ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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License Nos.:	DPR-80 DPR-82
Report No.:	50-275/99-04 50-323/99-04
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Nuclear Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	March 7 through April 17, 1999
Inspectors:	D. L. Proulx, Senior Resident Inspector D. G. Acker, Resident Inspector D. E. Corporandy, Resident Inspector G. A. Pick, Senior Project Engineer J. F. Melfi, Project Engineer
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ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Diablo Canyon Nuclear Power Plant, Units 1 and 2
NRC Inspection Report No. 50-275/99-04; 50-323/99-04

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report documents the inspection performed during a 6-week period by the resident inspectors.

Operations

- Operators ramped Unit 1 to 50 percent power on indication of elevated differential pressure across the main condenser because of mussel shells. The root cause analysis was thorough, determined that the influx of mussel shells resulted from incomplete cleaning of the Unit 1 traveling screens during Refueling Outage 1R9, and categorized this as a maintenance preventable functional failure (Section O1.2).

- A noncited violation was identified for failure to adequately maintain procedures, as required by Technical Specification 6.8.1.a. Specifically, procedures did not direct operators to energize the pressurizer backup heaters to ensure mixing when the pressurizer steam space sample valves were opened. Knowledge of the need to perform this action was not within the skill of the newer operators. The licensee performed a good investigation, and this noncited violation was documented in the corrective action program as Action Request A0482289 (Section O1.3).

- A noncited violation of Technical Specification 6.8.1.a was identified because operators placed two reactor protection system channels for Overtemperature ΔT and Overpower ΔT in bypass simultaneously, which was prohibited by Technical Specification 3.3.1. The shift foreman and licensing personnel demonstrated excellent attention to detail in identification of this issue. This noncited violation was documented in the corrective action program as Action Request A0481553 (Section O1.4).

- An NRC-identified noncited violation of Technical Specification 6.8.1.h was identified. Operators failed to implement a fire protection procedure for establishing a fire watch, because of weak knowledge of fire protection program requirements. Although this condition existed for approximately 15 hours, the issue was mitigated because the area was patrolled hourly for an unrelated fire impairment. This noncited violation was documented in the corrective action program as Action Request A0481645 (Section O4.1).

- The failure to properly follow an administrative procedure for controlling maintenance activities was identified as a noncited violation of Technical Specification 6.8.1.a. Operators performed the risk assessment, used to determine if on-line maintenance is acceptable, <u>after</u> declaring the auxiliary saltwater supply to Component Cooling Water Heat Exchanger 2-1 inoperable for replacement of the inlet expansion joint. The operators also failed to consult with the onsite Probabilistic Risk Assessment group, as required. In addition, the licensee identified several performance weaknesses, such as failure to properly document actions taken and poor communication among operators and between operators and management related to the risk involved with the emergent maintenance. This noncited violation was documented in the corrective action program as Action Request A0476004 (Section O4.2).

Maintenance

- A questioning attitude by a system engineer revealed a design deficiency in the construction of piping leaving the condensate storage tank, in that the carbon steel piping sat on concrete. Because ground water leakage corroded the piping for Level Transmitter LT-40 and resulted in a failure, the licensee concluded that the structure failed to perform its Maintenance Rule function and classified this as a maintenance preventable functional failure. The inspectors concluded that the licensee effectively implemented the Maintenance Rule and corrective action programs (Section E8.2).

Engineering

- Because of previous problems with the appropriateness of Technical Specifications interpretations, the inspectors assessed a sample of five Technical Specifications interpretations. The inspectors concluded that the licensee properly implemented NRC guidance for license amendment submittals and use of interim administrative controls when performing Technical Specifications interpretations (Section E8.4).

Plant Support

- A high level of attention to plant housekeeping in Unit 1, following Refueling Outage 1R9, resulted in an excellent level of housekeeping being maintained throughout safety-related areas in both units (Section O1.1).

- Licensee investigation of the circumstances that led to an individual receiving a hot particle exposure was thorough and appropriate to the circumstances. The licensee determined that personnel had input an incorrect geometry configuration into the multichannel analyzer. This resulted in the estimated dose (160 rem) exceeding regulatory limits. However, the estimated dose significantly improved (14 rem) using the correct geometry. The individual's exposure to the skin did not exceed regulatory limits, and no violations of NRC requirements were identified (Section R1.1):

Report Details

Summary of Plant Status

Unit 1 began this inspection period in Mode 5 (Cold Shutdown) with Refueling Outage 1R9 in progress. On March 10, 1999, operators commenced reactor heatup to normal operating temperature and pressure (Mode 3, Hot Standby). On March 13, operators commenced reactor startup (Mode 2, Startup) and took Unit 1 critical. The licensee paralleled the main generator to the grid on March 15, ending Refueling Outage 1R9, and reached 100 percent power on March 20. On March 23, Unit 1 operators reduced power to 50 percent because of high condenser differential pressure as a result of mussel shells. Following condenser cleaning, operators returned Unit 1 to 100 percent power on March 24 and remained at essentially 100 percent power until the end of this inspection period.

Unit 2 began this inspection period at 100 percent power. Unit 2 continued to operate at essentially 100 percent power until the end of this inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors visited the control room and toured the plant frequently when on site, including periodic backshift inspections. Overall, the performance of plant operators was professional and reflected a focus on safety. The use of three-way communications continued to improve, and operator responses to alarms were observed to be prompt and appropriate to the circumstances. The Unit 1 reactor startup and power ascension were performed carefully in accordance with procedures.

During plant tours to verify the operational readiness of safety-related systems, the inspectors noted that housekeeping was excellent in Unit 2. Also, the inspectors determined that a high level of management attention, following the Unit 1 refueling outage, resulted in restoration of housekeeping in Unit 1 to an excellent level.

O1.2 Unit 1 Power Reduction

a. Inspection Scope

The inspectors evaluated the operator response to indication of high differential pressure across the Unit 1 main condenser and subsequent power reduction.

b. Observations and Findings

On March 20, 1999, operators noted increasing differential pressure across the northeast quadrant of the Unit 1 main condenser. Operators expected the differential pressure to read approximately 5 psid, since Unit 1 had just completed a refueling outage and traveling screen differential pressures were normal. However, by March 22, the condenser differential pressure reached 8.5 psid and was increasing slowly. On March 23, the maximum condenser differential pressure read 9.9 psid and was rising

slowly. Procedure OP AP-7, "Degraded Condenser," Revision 19, Section B.1, required operators to immediately ramp the unit to 50 percent power if condenser differential pressure reached 10.0 psid and was rapidly increasing. Although the conditions requiring a power reduction did not exist, operators conservatively ramped Unit 1 to 50 percent power to allow condenser cleaning. Operators carefully performed the downpower in accordance with procedures.

Craftsmen performing the Unit 1 main condenser cleaning identified that the source of the elevated condenser differential pressure was the buildup of mussel shells. The root cause analysis determined that mussel shells adhered to the Unit 1 traveling screens and were then released into the circulating water system after the Unit 1 startup. Craftsmen failed to completely clean the Unit 1 traveling screens during Refueling Outage 1R9. The licensee initiated Action Request (AR) A0481449 and Quality Evaluation Q0012119 to enter this item into the corrective action program. The licensee considered this event to be a maintenance preventable functional failure. The inspectors reviewed the root cause and corrective actions and determined that they were appropriate.

Following condenser cleaning, operators returned Unit 1 to 100 percent power without incident. Unit 1 condenser differential pressures returned to normal.

c. Conclusions

Operators ramped Unit 1 to 50 percent power on indication of elevated differential pressure across the main condenser because of mussel shells. The root cause analysis was thorough, determined that the influx of mussel shells resulted from incomplete cleaning of the Unit 1 traveling screens during Refueling Outage 1R9, and categorized this as a maintenance preventable functional failure.

O1.3 Soluble Poison Management

a. Inspection Scope (71707)

The inspectors evaluated the response to AR A0482289, which discussed an event in which the boron concentration in the pressurizer significantly exceeded that of the reactor coolant system.

b. Observations and Findings

On April 1, 1999, chemistry personnel requested that operators open the pressurizer steam space sample valves to degas the reactor coolant system because of an elevated concentration of hydrogen. Operators energized the pressurizer backup heaters to initiate pressurizer spray flow and to promote good mixing within the reactor coolant system. This mixing would prevent boric acid from concentrating in the pressurizer while the steam space sample valves were open.

However, the need for maintaining this configuration while the pressurizer steam space sample valves remained open was not adequately turned over to the oncoming shift. The oncoming crew assumed that the pressurizer heaters were energized for one-time mixing of the pressurizer and the reactor coolant system. This crew did not recognize the concentrating effect that open pressurizer steam space sample valves would have on pressurizer boric acid concentration. Thus, on April 2, operators secured the pressurizer backup heaters, when chemistry personnel reported that the pressurizer and reactor coolant system boric acid concentrations were equalized. The pressurizer steam space sample valves remained open for the next 4 days without adequate mixing of the fluid in the reactor coolant system and the pressurizer.

On April 6, during routine sampling, chemistry personnel determined that the boron concentration of the pressurizer was 1502 ppm, while the boron concentration of the reactor coolant system indicated 1294 ppm, a difference of 208 ppm.
Procedure OP L-4, "Normal Operation at Power," Revision 44, Section 5.3, limited the differential boron concentration between the pressurizer and reactor coolant system to 50 ppm. Procedure OP F-5:I, "Chemical Control Limits and Action Guidelines," Revision 17, required operators to take corrective action, if the reactor coolant system and pressurizer liquid space boron concentrations diverged by +10/-2 percent. In this instance, the limit was 129.4 ppm. Operators initiated an AR to enter this item into the corrective action program.

The shift supervisor, knowledgeable on integrated system response, recognized that, with the pressurizer steam space sample valves open but without continuous pressurizer spray flow, boric acid tends to concentrate in the pressurizer liquid space. The shift supervisor directed that the backup pressurizer heaters be started manually, which induced continuous spray flow, ensuring good mixing between the reactor coolant system and pressurizer. To compensate for the increase in reactor coolant system boric acid concentration resulting from the mixing, operators diluted the reactor coolant system to balance reactivity. On April 7, the pressurizer and reactor coolant system boric acid concentrations were equalized.

Licensee investigation revealed that operators did not energize the backup pressurizer heaters to induce additional spray flow to ensure good mixing of the coolant in the pressurizer and reactor coolant system when pressurizer steam space sample valves were opened. Procedures OP L-4 and OP B-9:I "Primary Sampling System - Make Available and Place in Service," Revision 3, did not require energization of the backup heaters when opening the pressurizer steam space sample valves. More experienced licensed operators considered this evolution to be within the skill of the craft. However, the inspectors noted that licensee training manuals did not cover operation with the pressurizer steam space sample valves open. Consequently, without adequate procedure guidance, less experienced operators failed to properly ensure that the pressurizer heaters remained energized, which ensured mixing when the steam space sample valves were opened.

The inspectors concluded that Procedures OP L-4 and OP B 9:I were not appropriate to the circumstances. The failure to adequately maintain procedures for power operations is a violation of Technical Specification 6.8.1.a. However, this Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the Enforcement Policy. This violation is in the corrective action program as AR A0482289 (50-275/99004-01).

The licensee evaluated the effect of the high boron concentration in the pressurizer. The licensee noted that, if a downpower was required with elevated boric acid in the pressurizer, an unexpected increase in reactor coolant system boric acid concentration would occur. However, the licensee determined that this increase would be slow and have little effect on the plant response during a transient.

c. Conclusions

A noncited violation was identified for failure to adequately maintain procedures, as required by Technical Specification 6.8.1.a. Specifically, procedures did not direct operators to energize the pressurizer backup heaters to ensure mixing when the pressurizer steam space sample valves were opened. Knowledge of the need to perform this action was not within the skill of the newer operators. As a result over a 4-day period, boric acid concentrated in the pressurizer and the differential boron concentration between the pressurizer and the reactor coolant system exceeded procedure limits. Incomplete operator turnovers contributed to this issue. The licensee performed a good investigation, and this noncited violation was documented in the corrective action program as AR A0482289.

O1.4 Bypassing Overtemperature ΔT and Overpower ΔT Protective Functions

a. Inspection Scope (61726, 92902)

The inspectors evaluated the response to AR A0481553 and Nonconformance Report (NCR) N0002093, which discussed an event that involved two concurrent bypassed channels in the reactor protection system and required entry into Technical Specification 3.0.3.

b. Observations and Findings

On March 20, 1999, technicians performed Procedure STP R-21, "Reactor Coolant System Temperature Instrumentation Data," Revision 6B, on Unit 1. This procedure obtained the data necessary to provide inputs to the solid state protection system for the Overtemperature ΔT and Overpower ΔT protection functions. Portions of Procedure STP R-21 required use of Procedure MP I-36-M.1 "Operating Instructions for Eagle 21 and Man-Machine Interface," Revision 5, for guidance. These procedures required technicians to place one channel of the protective function in "bypass" to obtain the required data.

The shift foreman authorized technicians to place the Loop 1 temperature bistables to "bypass." Approximately 5 minutes later, the shift foreman authorized placement of the Loop 2 bistables in "bypass" by authorizing the parameter updates to be processed. The technicians did not communicate to the shift foreman that the Loop 1 temperature bistables were still bypassed. Therefore, two channels of the Overtemperature ΔT and Overpower ΔT were bypassed simultaneously. The Loop 2 bistable was returned to service. The same process of having the shift foreman authorize bypassing the bistables and returning them to service occurred with the Loops 3 and 4 bistables, as well, with the Loop 1 bistables bypassed. This resulted in two channels of the Overtemperature ΔT and Overpower ΔT being bypassed for 38 minutes total, without operator knowledge.

Technical Specification 3.3.1, Action 6, requires four channels of reactor protection each for the Overtemperature ΔT and Overpower ΔT protection functions. Technical Specification 3.3.1 allows one channel to be inoperable provided it is placed in the tripped condition within 6 hours. The affected channel or a different channel may be placed in bypass for up to 4 hours. Technical Specification 3.3.1 does not have an action statement for two or more channels placed in bypass; therefore, if two channels are placed in bypass simultaneously, Technical Specification 3.0.3 applies. Technical Specification 3.0.3 requires that, when a limiting condition for operation is not met and there are no specified action requirements, the licensee must take action within 1 hour to place the plant in at least hot shutdown within the next 6 hours. Therefore, on March 20, during performance of Procedure STP R-21, the licensee inadvertently entered Technical Specification 3.0.3 for approximately 38 minutes.

During the subsequent shift foreman's review of operator logs and annunciator printouts, the licensee identified the inappropriate entries into Technical Specification 3.0.3. The oncoming shift foreman initiated an AR, and licensee management subsequently elevated the issue to an NCR to determine the root cause and develop corrective actions. In addition, licensing personnel identified an additional instance of entry into Technical Specification 3.0.3. The inspectors concluded that the operations and licensing personnel demonstrated excellent attention to detail.

Licensee investigation revealed that neither Procedures STP R-21 nor MP I-36-M.1 contained precautions that prevented bypassing of two protection channels simultaneously. Such a precaution would be prudent because the design of the Eagle 21 protection system permitted two channels to be bypassed simultaneously. The failure of Procedures STP R-21 and MP I-36-M.1 to contain precautions or prohibitions against bypassing two channels of reactor protection was a violation of Technical Specification 6.8.1.a, which resulted in a condition prohibited by Technical Specification 3.3.1. However, this Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the Enforcement Policy. This violation is in the corrective action program as AR A0481553 (50-323/99004-02).

For short-term corrective actions, the Operations Director initiated a shift order reiterating the requirements of Technical Specification 3.3.1. The licensee was also evaluating revisions to the applicable procedures or an Eagle 21 design change.

c. Conclusions

A noncited violation of Technical Specification 6.8.1.a was identified because operators had placed two reactor protection system channels for Overtemperature ΔT and Overpower ΔT in bypass simultaneously, which was prohibited by Technical Specification 3.3.1. The shift foreman and licensing personnel demonstrated excellent attention to detail in identification of this issue. This noncited violation was documented in the corrective action program as AR A0481553.

O4 Operator Knowledge and Performance

O4.1 Implementation of Equipment Control Guidelines

a. Inspection Scope (71707, 71750)

The inspectors reviewed licensee implementation of Technical Specifications and Equipment Control Guidelines on a daily basis to ensure the applicable requirements were properly followed.

b. Observations and Findings

On March 26, 1999, the inspectors noted that operators cleared the Unit 2 positive displacement pump (PDP) for routine preventive maintenance. The inspectors reviewed this maintenance item to determine if the actions credited in the Updated Final Safety Analysis Report had been implemented. Although the PDP was considered nonsafety-related, the fire protection analysis took credit for the PDP to mitigate the effects of a design basis fire. The safety-related centrifugal charging pumps, located in the same room with no fire rated train separation, could not be credited in the fire hazards analysis.

Chapter 9.5 of the Updated Final Safety Analysis Report and Equipment Control Guideline 8.1 provided the compensatory actions credited in the fire hazards analysis to be taken in the event the PDP was inoperable. These references required that an hourly fire patrol watch be initiated at the centrifugal charging pumps within one hour if the PDP was inoperable. If the fire detection or suppression systems in the area were coincidentally inoperable, a continuous fire watch was required.

However, at 10:30 a.m. on March 27, upon review of the Unit 2 Technical Specifications tracking sheets, the inspectors determined that an hourly fire watch had not been established when operators cleared the PDP. This condition had existed since 7:30 p.m. on March 26 for approximately 15 hours. The inspectors informed the shift supervisor and shift foreman, who in turn informed the individual performing the hourly roving fire watches. In addition, the shift supervisor initiated AR A0481645 to enter this item into the corrective action program.

Licensee investigation revealed that the shift foreman who authorized clearing the PDP normally worked in the maintenance department and was not normally assigned to an operating crew. Thus the shift foreman had poor knowledge of the compensatory actions taken when the PDP was cleared.

Procedure OM8.ID2, "Fire System Impairments," Revision 6A, Section 5.1.2, required that each fire impairment be tracked on the "Fire Watch Status Log" to ensure proper implementation of compensatory measures. On March 26, because of unfamiliarity with fire protection requirements, the shift foreman did not notify the roving fire patrol watch so that a Fire Watch Status Log could be initiated for the charging pumps. The failure to implement Section 5.1.2 of Procedure OM8.ID2 is a violation of Technical Specification 6.8.1.h. However, this Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the Enforcement Policy. This violation is in the corrective action program as AR A0481645 (50-323/99004-03).

The licensee noted that this issue was mitigated since the entire auxiliary building, including the location of the centrifugal charging pumps, was patrolled with an hourly roving fire watch because of inoperable fire rated seals. Therefore, the hourly roving fire watch was in the area of the charging pumps for an unrelated reason. However, the inspectors noted that this issue indicated weaknesses among operators, with respect to fire protection.

c. Conclusions

An NRC-identified noncited violation of Technical Specification 6.8.1.h was identified. Operators failed to implement a fire protection procedure for establishing a fire watch, because of weak knowledge of fire protection program requirements. Although this condition existed for approximately 15 hours, the issue was mitigated because the area was patrolled hourly for an unrelated fire impairment. This noncited violation was documented in the corrective action program as AR A0481645.

O4.2 Determination of Risk Significance for Replacement of Component Cooling Water (CCW) Heat Exchanger (HX) Inlet Expansion Joint

a. Inspection Scope (71707, 92901)

The inspectors evaluated Unit 2 operations activities related to an inappropriate risk determination associated with emergent maintenance, involving replacement of an inlet expansion joint for CCWHX 2-1. The inspectors evaluated plant procedures for determining on-line risk, interviewed licensee personnel who evaluated the event, and evaluated the corrective actions implemented. On February 3, 1999, operations personnel recognized that maintenance performed on January 27 had extremely high risk and initiated AR A0476004 to document the deficiency and implement corrective actions.

b. Observations and Findings

On January 26, operators had removed Battery Charger 2-2 from service for scheduled maintenance. Operators declared CCWHX 2-1 inoperable on January 27 at 12:50 p.m. after engineering personnel had notified them that the inlet expansion joint could not be assured to remain operable during an accident because of its age (26 years). In addition, operators deenergized and opened the 4160V breaker for Auxiliary Saltwater Pump 2-1 on January 27 at 6:36 p.m. in order to perform maintenance on the auxiliary saltwater inlet expansion joint for CCWHX 2-1.

During a postmaintenance review of the completed risk assessment form for the combined maintenance activities of Battery Charger 2-2 and CCWHX 2-1, as specified in Procedure AD7.DC6, "On-Line Maintenance Risk Assessment," Revision 2, Attachment 9.13, an individual noted several deficiencies. Specifically, the probabilistic risk assessment (PRA) allowed outage time (AOT), the number of hours that a safety system could be removed from service, was marked "NA." Operators erroneously believed that "NA" in the matrix for risk assessment meant "not applicable" and nonrisk significant, when it meant "not analyzed." As a result, the operators incorrectly concluded that the scheduled work was less than the PRA AOT. The operators also assumed that PRA AOT was greater than the threshold PRA AOT (36), based on having the single component with the highest risk out of service. The individual who performed the postmaintenance review of this initial risk assessment indicated that the operators had no basis to make these conclusions. The evaluator determined that the form did not indicate that the Operations Director had been notified, as required for the level of risk. However, the Operations Director had been notified. Also, the evaluator determined that the key safety function value, which was a relative rating of risk, exceeded the acceptance criteria.

The licensee determined that the offgoing shift supervisor requested the oncoming shift supervisor to perform a second, after-the-fact, risk assessment for the combination of both Auxiliary Saltwater Pump 2-1 and Battery Charger 2-2 being out of service, since the auxiliary saltwater pump had been recently removed from service. This combination of components had a PRA AOT of 8, which represented an increase of 4128 percent above the base core damage frequency. The operators then correctly identified that this PRA AOT was less than the work duration and the threshold PRA AOT and that the safety function value of 10 exceeded the acceptance criteria of 8. However, the oncoming shift supervisor thought that the Operations Director had been notified of this high risk combination of the auxiliary saltwater pump and the battery charger and did not contact him.

The licensee noted that operators did not recognize that they should have contacted the onsite PRA group for any combination of components identified by Procedure AD7.DC6, Attachment 9.4, as "NA," which indicates that the combination had not been specifically analyzed in the PRA. In addition, the licensee determined that the operators made errors in filling out the initial risk assessment, failed to perform the second risk assessment prior to the evolution, and failed to notify the Operations Director of the extremely high risk situation identified by the second risk assessment.

Procedure AD7.DC6, Attachment 9.4, notes that "NA" meant that the PRA AOT for this combination was "not analyzed." The inspectors noted that Procedure AD7.DC6 was a quality-related administrative procedure; as such, the procedure established program requirements for controlling activities related to items covered by the quality assurance program. The inspectors concluded that a violation of Technical Specification 6.8.1.a occurred since the operators failed to obtain guidance from the onsite PRA group when they obtained an "NA" on Attachment 9.4 of Procedure AD7.DC6 and failed to properly complete a risk assessment as specified in Procedure AD7.DC6 prior to the second evolution. The licensee performed an investigation for this deficiency, entered this deficiency into their corrective action program, and implemented appropriate corrective actions. The licensee provided training to operators that described the appropriate usage of this procedure, emphasizing the breakdowns that occurred. This licensee-identified Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the Enforcement Policy. This violation is in the corrective action program as AR A0476004 (50-275; 323/99004-04).

Because of the relatively high risk value associated with the combined maintenance on CCWHX 2-1 and Battery Charger 2-2 with Auxiliary Saltwater Pump 2-1 deenergized, the licensee reperformed the risk assessment using realistic conditions as part of the corrective actions for AR A0476004. Specifically, although Auxiliary Saltwater Pump 2-1 was deenergized, no work was performed on the pump, and the licensee estimated that the pump could have been restored and aligned to inject into CCWHX 2-2 within 30 minutes. Also, power from the station batteries would have been available for some period of time and would not be lost at the same time as offsite power. These more realistic assumptions resulted in a risk increase above the base core damage frequency of 680 percent and a PRA AOT of 51 hours, which the licensee considered as a low risk activity.

c. Conclusions

The failure to properly follow an administrative procedure for controlling maintenance activities was identified as a noncited violation of Technical Specification 6.8.1.a. Operators performed the risk assessment, used to determine if on-line maintenance is acceptable, after declaring the auxiliary saltwater supply to CCWHX 2-1 inoperable for replacement of the inlet expansion joint. The operators also failed to consult with the onsite PRA group as required. In addition, the licensee identified several performance weaknesses such as failure to properly document actions taken and poor communication among operators and between operators and management related to the risk involved with the emergent maintenance. This noncited violation was documented in the corrective action program as AR A0476004.

O8 Miscellaneous Operations Issues (92700, 92901, 90712)

O8.1 Violation Closures

The inspectors performed an in-office review of outstanding violations in the operations area. The Severity Level IV violations listed below were issued in Notices of Violation prior to March 11, 1999. On this date, the NRC changed the policy for treatment of Severity Level IV violations (Appendix C of the Enforcement Policy). Because these violations would have been treated as noncited violations in accordance with Appendix C, they are being closed out in this report, consistent with the new Enforcement Policy for Severity Level IV violations. The inspectors verified that the licensee had included these violations in their corrective action program. The corrective action program reference (AR or NCR) for each of the violations is listed below. In addition, these violations already have docketed responses or were generated with no response required.

Violation Number	Description	CA Program Reference
50-323/98021-01	Failure to secure circulating water pump and failure to properly set atmospheric dump valve	N0002078
50-275/98010-01	Mechanics drained oil from the wrong auxiliary feedwater pump	A0461604
50-275; 323/98007-05	Failure to implement design of level indicating system into abnormal procedure	A0451605
50-275; 323/98007-04	Multiple failures to implement clearance procedure	A0461891 N0002056 N0002054
50-323/98007-03	Failure to restore high flux alarm during core reload	A0455551
50-323/98007-02	Failure to provide appropriate procedure for nonseismic hoist storage	A0458146
50-275; 323/98002-01	Failure to implement sealed valve program	A0450497
50-275/97-369-03 (03014)	Auxiliary saltwater trains made inoperable	N0002034
50-275/97-369-02 (02014)	Valves unacceptable and negatively impacted the auxiliary saltwater system	A0444138 A0444100 N0002034
50-275/97010-01	Failure to follow procedure for alignment of 480V power supply Panel PY-16	A0440311

Violation Number	Description	CA Program Reference
50-323/97006-01	Failure to maintain an operator at-the-controls as required by procedure	A0435466
50-275; 323/96021-01 96-469-02 (02014)	Failure to update the Final Safety Analysis Report	A0427852 A0417757 A0415113 N0002003
50-275; 323/96021-01 96-469-01 (01014)	Failure to perform 10 CFR 50.59 evaluation when revising Procedure E-1.3	A0416238

Review of the effectiveness of the corrective actions for selected violations will be performed in the future as a routine part of the review of the corrective action program.

O8.2 (Closed) Licensee Event Report (LER) 50-323/98-005-00: manual reactor trip because of heavy debris loading of the circulating water system during Pacific ocean storm.

The issues identified in this LER were appropriately addressed, as documented in NRC Inspection Report 50-275; 323/98-21. No further review is required.

O8.3 (Closed) LER 50-275/97-010-00 and 01: unplanned start of Diesel Engine Generator 1-1 because of a 4160V Bus H startup feeder phase potential transformer fuse that opened.

After eliminating other possible causes for the unplanned diesel engine generator start, the licensee concluded that the unplanned start of Diesel Engine Generator 1-1 resulted from a failed fuse. The licensee lost the failed fuse and had not performed diagnostic examinations on the failed fuse. The licensee did perform tests on eight similar fuses installed at approximately the same time as the failed fuse (circa 1970). These tests revealed that one similar fuse had a higher internal resistance than the normal acceptable resistance range for the fuses. Current surges through a fuse with a high internal resistance will eventually cause fuse failure because of accumulated fuse element degradation.

The licensee replaced all similar fuses and now performs periodic continuity checks on these fuses to ensure that the internal resistance is within specifications. The licensee excluded the potential for a circuit problem through these fuses since there had not been any other diesel engine generator starts because of this failure. Based on the corrective actions, this LER is closed.

O8.4 (Closed) LER 50-323/97-003-01 and -00: manual reactor trip on loss of normal feedwater because of unknown condensate/feedwater transient.

The issues described in this LER were discussed in NRC Inspection Report 50-275; 323/97-10, Section O1.2. The licensee did not identify a root cause for the condensate/feedwater transient. The inspectors identified no new issues during the inoffice review of this licensee event report. The inspectors found the licensee evaluation into the event and corrective actions taken satisfactory. No further review is required.

O8.5 (Closed) LER 50-275;323/97-002-00: refueling water storage tank outside its design basis because of insufficient water margin at completion of switchover to cold-leg recirculation.

Background

Following a loss-of-coolant accident, the centrifugal charging, safety injection, residual heat removal (RHR), and containment spray pumps inject into the containment building, taking suction from the refueling water storage tank (RWST). The times allowed to accomplish switchover to cold-leg recirculation are described in Table 6.3-5 of the Updated Final Safety Analysis Report. At 33 percent RWST level, the RHR pumps are automatically shut off, while the other pumps remain running. Operators are required to manually realign the suction source for the RHR pumps to the containment recirculation sump and manually realign the other pump suctions to the discharge of the RHR pumps. This switchover must occur before the RWST level reaches 4 percent where vortexing can occur, causing the pumps to become air bound and therefore inoperable. The original analysis used by the licensee included a failure of one RHR pump to trip and used the times stated in the Updated Final Safety Analysis Report.

Calculations performed before initial plant startup showed that, at the completion of the switchover to cold-leg recirculation using emergency operating procedures, the remaining usable volume in the RWST was approximately 32,500 gallons. Using the more conservative assumptions with Emergency Operating Procedure E-1.3, "Transfer to Cold Leg Recirculation," the licensee would have less than this licensing basis margin.

Following questions by NRC inspectors, the licensee initiated this LER to document that Procedure E-1.3 had been revised without evaluating the impact of the revisions on the calculation that estimated the volume remaining in the RWST following switchover. Operators using these unevaluated revisions could have placed the plant outside its licensing basis. Various calculations that provided the licensing basis for the emergency operating procedure were nonconservative.

Findings and Conclusions

Following NRC identification of this issue, a predecisional enforcement conference was held on December 18, 1996, to address apparent violations involving failures to perform safety evaluations for the changes to Procedure E-1.3. The NRC subsequently issued

violations concerning this issue for not performing an adequate 10 CFR 50.59 safety evaluation (50-275:323/96021-01) and for their design control process for this calculation (50-275;323/96024-02). The remaining open issues were to determine the safety significance of the reduction in RWST volume and to assess the licensee's need to automate the process for switchover to cold-leg recirculation (refer to NRC Inspection Report 50-275; 323/97-02, Section E1.2).

The licensee submitted a letter dated April 7, 1997, addressing the need to automate the switchover process. The licensee reanalyzed this situation and provided the reanalysis to the NRC in a letter dated March 11, 1999. The staff evaluated the submittals and on April 8, 1999, concluded that the licensee was not required to automate the switchover process.

To assess the safety significance of this issue, the licensee wrote NCR N0002015. The licensee ensured that the design basis was satisfactorily translated into the current procedures. The licensee performed a reanalysis of the transfer to cold-leg recirculation, including instrument accuracy of the RWST level transmitter, and verified operators could perform the required actions within the assumed times. The analysis showed that the level transmitters had the required accuracy, that the licensee timed the activities in the emergency operating procedure to assure that they could be performed within the times shown in Updated Final Safety Analysis Report, Table 6.3-5, and that sufficient inventory existed. The inspectors concluded that the analysis identified a reduced volume margin in the RWST; however, sufficient volume remained to perform the switchover to the recirculation phase.

The NRC previously issued a violation of 10 CFR 50.59 and a violation involving the design control process. The RWST level uncertainty was evaluated in Section E8.3 of NRC Inspection Report 50-275; 323/98-13 and in a March 11, 1999, letter that found the level uncertainty of the RWST level instruments acceptable. The licensee verified that previous versions of Procedure E-1.3 would still have left sufficient inventory in the RWST. The new analysis confirms that there would be water inventory left after the switchover of the suction of the RHR pump to the recirculation sump. Based on the licensee's actions and previous NRC enforcement action, this LER is closed.

O8.6 (Closed) LER 50-275; 323/97-001-00: CCW system has operated with procedural guidance that permitted operation in a condition outside the design basis of the plant.

This LER documents a condition where, using the emergency operating procedures, operators could have placed the plant in a configuration in which the ability of the RHR system to cool the core was lost. The inspectors reviewed the licensee corrective actions and previous NRC inspection reports related to this issue.

The original design of the auxiliary saltwater and CCW systems included separation into two redundant trains for long-term cooling during a loss-of-coolant accident as described in the Updated Final Safety Analysis Report, Section 9.2.7.2, thus assuring their ability to withstand a single passive failure. This was reflected in Emergency Operating Procedure E-1.4, "Transfer to Hot Leg Injection," Revision 12, which required train separation at approximately 10½ hours after the start of a loss-of-coolant accident.

However, the licensee discovered that, when the trains were separated, the accident heat removal capabilities of these systems, in conjunction with the RHR system, were vulnerable to a single active failure in two scenarios. First, for the postulated loss of electrical Bus F, CCW cooling flow in Loop B and auxiliary saltwater cooling flow in Loop A would be lost. Second, for the postulated loss of electrical Bus G, auxiliary saltwater cooling flow in Loop B and RHR flow in Loop A would be lost.

The inspectors evaluated this issue in NRC Inspection Reports 50-275; 323/97-202, -98-05, and -98-09. The NRC concluded that this deficiency involved a nonsubstantial potential unreviewed safety question, and discretion was exercised as provided by Section VII.B.6 of the Enforcement Policy. The licensee concluded that, for long-term cooling, the CCW and auxiliary saltwater system plant configurations were appropriate.

To protect against an active single failure, which could render long-term heat decay removal inoperable, the licensee modified Procedure E-1.4 to no longer require immediate train separation after the transfer to hot-leg recirculation. The decision to separate the trains would be made by the Technical Support Center, after an evaluation of actual plant conditions. The licensee also upgraded two cross-tie valves to allow credit for their operation after a loss-of-coolant accident. The inspectors concluded that, based on these actions, this LER is closed and that no new issues were identified.

O8.7 (Closed) Unresolved Item 50-275; 323/96021-07: nonconservative assumptions in timing of switchover to cold-leg recirculation.

This item is addressed by the closeout of LER 50-275; 323/97-002-00 in Section O8.5.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments on Maintenance Activities

a. Inspection Scope (62707)

The inspectors observed portions of work activities covered by the following work orders:

Work Order	Description
R0190730	Clean/Inspect Seawater Side of CCWHX
C0159543	Repair Seat Leak for Valve PCV-21

b. Observations and Findings

The inspectors concluded that the applicable work orders were performed properly. Operators initiated the clearances properly, and probabilistic safety assessments were satisfactorily performed prior to the work.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

On April 6, 1999, the inspectors observed performance of Procedure STP I-2D, "Nuclear Power Range Incore/Excore Calibration," Revision 42.

b. Observations and Findings

The inspectors found that this surveillance test was conducted properly and the data was properly documented. The probabilistic safety assessment was properly performed prior to removing the equipment from service for the test.

M8 Miscellaneous Maintenance Issues (92700, 92902)

M8.1 (Closed) Violation 50-275/97-369-01 (01014): failure to monitor performance of auxiliary saltwater check valves.

The inspectors performed an in-office review of this violation. This Severity Level IV violation was issued in a Notice of Violation prior to March 11, 1999. On this date, the NRC changed the policy for treatment of Severity Level IV violations (Appendix C of the Enforcement Policy). Because this violation would have been treated as a noncited violation in accordance with Appendix C, it is being closed out in this report, consistent with the new Enforcement Policy for Severity Level IV violations. The inspectors verified that the licensee had generated NCR N0002034 for this violation. In addition, this violation already has a docketed response. Review of the effectiveness of the corrective actions for selected violations will be performed in the future as a routine part of the review of the corrective action program.

M8.2 (Closed) LER 50-275/97-016-01: Technical Specifications 3.3.1 and 3.3.2 not met following inadequate testing of reactor trip/engineered safety feature functions because of inadequate vendor design information discovered as a result of Generic Letter 96-01.

The issues identified in this licensee event report were appropriately addressed, and NRC had previously issued a noncited violation, as documented in NRC Inspection Report 50-275; 323/97-19, Section M1.4. No further review is required.

M8.3 (Closed) LER 50-275/97-011-01 and -00: auxiliary saltwater systems outside design basis for flooding because of inadequate corrective actions resulting from personnel error.

The issues identified in these licensee event reports were appropriately addressed, and NRC had previously issued a Notice of Violation, as documented in NRC Inspection Report 50-275; 323/97-14 and Enforcement Action 97-369. The licensee documented this deficiency and their corrective actions in ARs A0444100 and A0444138 and NCR N0002034. No further review is required.

M8.4 (Closed) LER 50-275/95-016-02: Technical Specification 3.4.2.2 not met during pressurizer safety valve surveillance testing because of random setpoint spread.

The issues identified in this licensee event report were appropriately addressed, and NRC had previously issued a noncited violation, as documented in NRC Inspection Report 50-275; 323/97-07, Section E8.2. No further review is required.

M8.5 Closed LER 50-275/94-010-01: Unit 1 shut down in accordance with Technical Specification 3.8.1.1, Action b, because of a degraded condition on the Phase C 500 kV transformer.

This LER discussed an event in which the licensee discovered a degraded condition in the 500 kV transformer, which resulted in a Unit 1 shutdown. On May 10, 1994, plant instrumentation alerted operators to arcing in Phase C of the Unit 1 main transformer. Chemical analysis of the transformer oil revealed the presence of acetylene in the oil. The licensee declared the 500 kV transformer inoperable. Because Technical Specification 3.8.1 credits the 500 kV transformer as a required source of offsite power, the licensee commenced a reactor shutdown to affect repairs.

Revision 0 of this LER did not include the root cause and corrective actions. Revision 1 contained this information. The licensee attributed the root cause to manufacturer design deficiencies. For corrective actions, the licensee initially replaced the inoperable transformer with a spare. Subsequently, during Refueling Outage 1R7, the licensee replaced the entire Unit 1 main transformer with a model from a different vendor. The inspectors concluded that the corrective actions were satisfactory.

III. Engineering

E8 Miscellaneous Engineering Issues (92903, 92700)

E8.1 Violation Closures

The inspectors performed an in-office review of outstanding violations in the engineering area. The Severity Level IV violations listed below were issued in Notices of Violation prior to March 11, 1999. On this date, the NRC changed the policy for treatment of Severity Level IV violations (Appendix C of the Enforcement Policy). Because these violations would have been treated as noncited violations in accordance with Appendix C, they are being closed out in this report, consistent with the new Enforcement Policy for Severity Level IV violations. The inspectors verified that the licensee had generated a corrective action program reference (AR or NCR) for each of the violations listed. In addition, these violations already have docketed responses or were generated with no response required.

Violation Number	Description	CA Program Reference
50-275; 323/98009-01	10 CFR 50.59 violation because changes to CCW system allowed boiling in containment fan cooler units and changes to emergency operating procedures affected a Technical Specification	A0468466 N0002008
50-275/98009-02	10 CFR 50.59 violation because auxiliary saltwater system modification located piping on ground other than bedrock	N0002008
50-275; 323/98008-03	failure to perform 10 CFR 50.59 evaluation for the change to the containment fan coil unit motor leads	A0461257
50-275/97006-06	failure to ensure spray shields enclose reactor coolant pump lift	N0002025
50-275; 323/97003-05	failure to adequately incorporate design basis into emergency operating procedures	N0002011

Review of the effectiveness of the corrective actions for selected violations will be performed in the future as a routine part of the review of the corrective action program.

E8.2 (Closed) Inspection Followup Item 50-275; 323/98020-02: evaluate adequacy of Maintenance Rule program for assessing structures.

A system engineer determined that a 1-inch diameter instrument line for Level Transmitter LT-40, which penetrated the side wall of the condensate storage tank, leaked. The licensee performed a prompt operability assessment that demonstrated, with condensate storage tank level above 83 percent, that sufficient inventory remained to cool down the plant using auxiliary feedwater and accommodates the added leakage from the 1-inch instrument line. The inspectors initiated this item to evaluate whether the Maintenance Rule program should have previously identified this deficiency.

During this inspection, the inspectors reviewed the circumstances surrounding identification of the leak. The inspectors determined that a questioning attitude by a system engineer revealed a design deficiency in the construction of piping leaving the condensate storage tank. The system engineer noted that the vault area below the condensate storage tank (located below ground elevation at the base of water storage tanks) indicated leakage, while other similar vault areas had no indication of leakage.

The licensee initiated NCR N0002082 to document the corrosion and effect corrective actions. The design deficiency consisted of carbon steel pipes setting on concrete and becoming wet because of ground water. As immediate corrective action to prevent further corrosion, the licensee removed the concrete from around the carbon steel pipes for the condensate storage and fire water storage tanks. Subsequently, the licensee performed ultrasonic testing of the pipes and determined that the wall thickness met specifications. For long-term corrective action, the licensee relocated the instrument line taps for the condensate storage tank level transmitters. In addition, the licensee added notations to the procedure for inspection of structures to ensure that engineers will assess any leakage for damage to components.

The inspectors reviewed the application of the Maintenance Rule to onsite structures. The inspectors concluded that this deficiency did not result from an inadequate Maintenance Rule program or program implementation. The licensee determined that Level Transmitter LT-40 was within the scope of the Maintenance Rule. Because ground water leakage corroded the piping for Level Transmitter LT-40 and resulted in a failure, the licensee concluded that the structure failed to perform its Maintenance Rule function and classified the structural failure as a maintenance preventable functional failure. The licensee placed the structure in 10 CFR 50.65a(1), established corrective actions, and was establishing goals. The inspectors concluded that the licensee effectively implemented the Maintenance Rule and their corrective action program.

E8.3 (Closed) LER 50-275; 323/98-013-00: actuations of engineered safety features - diesel engine generators started when startup power was lost because of an inappropriate relay setpoint.

This LER was described in NRC Inspection Report 50-275; 323/98-18, Section E2.1. The licensee implemented appropriate actions and documented this nonsafety-related deficiency in NCR N0002077.

E8.4 (Closed) Unresolved Item 50-275; 323/97023-03: Questionable Technical Specifications interpretations.

a. Inspection Scope

The inspectors evaluated the licensee actions with respect to five Technical Specifications interpretations.

b. Observations and Findings

Technical Specifications Interpretation 88-01, applicability of 25 percent grace period for conditional surveillances. In answer to the question/concern about "When does T.S. 4.0.2 apply?" The Plant Staff Review Committee (PSRC) interpretation stated: "T.S. 4.0.2 applies to the surveillance requirements given in Section 4 of the Technical Specifications unless an exception is stated. T.S. 4.0.2 also applies when a surveillance requirement is referenced in an action statement. T.S. 4.0.2 does not apply to remedial actions required by Action statements (i.e., grab samples every 24 hours, hourly fire watches)."

The inspectors considered this interpretation to be confusing because it did not appear to be consistent with the improved Technical Specifications submitted by the licensee for NRC review and approval. For example, the Improved Technical Specifications submittal specifies that the 25 percent extension does not apply to the initial performance of a required action or surveillance, since the initial action or surveillance usually verifies that no loss of function has occurred by checking the status of redundant or diverse components. The bases of the Improved Technical Specifications submittal does allow the 25 percent extension for subsequent performance of required actions or surveillances. The inspectors questioned the licensing department about Technical Specifications Interpretation 88-01. Licensing's explanation of Technical Specifications Interpretation 88-01 was consistent with the Improved Technical Specifications submittal. However, the inspectors also interviewed four senior reactor operators and a shift technical advisor about Technical Specifications Interpretation 88-01. Each of these individuals stated that they would not apply the 25 percent extension to any surveillance requirements referenced in an action statement.

The last sentence of the interpretation contained the abbreviation "i.e." The inspectors asked if ". . . grab samples every 24 hours, hourly fire watches" was to be interpreted to be inclusive of all the remedial actions to which Technical Specification 4.0.2 did not apply or if there were other examples, in which case "e.g." should have been used. The licensing engineers acknowledged that "e.g." should have been used.

Although the inspectors considered Technical Specifications Interpretation 88-01 to be unclear, they acknowledged that the licensee was aware of the issue, that actions were underway to clarify Technical Specifications Interpretation 88-01 in the bases of the Improved Technical Specifications, and that licensed operators appeared reluctant to extend any surveillance requirements referenced in an action statement without consulting licensing.

Technical Specifications Interpretation 89-07, compliance with off-site power specifications. In answer to the question/concern about "What constitutes compliance with Technical Specification 3.8.1.1, Action a if an independent circuit from the offsite transmission network is inoperable with respect to only one 4kV vital bus (e.g. the Bus G startup feeder breaker is racked out)?" the PSRC interpretation stated: "If one circuit from the offsite transmission network to only one of the vital 4kV busses is inoperable, it is necessary to perform Technical Specification 3.8.1.1, Action a only for the affected 4kV bus and affected diesel generator. It is not necessary to verify power supplies or start the diesel generators on the unaffected 4kV busses."

The inspectors considered this interpretation to be clear and noted that it was consistent with the Improved Technical Specifications submittal.

Technical Specifications Interpretation 94-08, allowance of charging pump usage during low temperature/over pressure conditions. In answer to the question/concern about an exception note in Technical Specifications 3.1.2.3 and 3.1.2.4, which specifies that an inoperable charging pump may be OPERABLE for testing in accordance with Technical Specification 4.0.5, the question asks if the inoperable charging pump may "be made available for other testing that may require the charging pump available (i.e. diesel

testing, pump acceleration time testing, safeguards testing, etc.)?" The PSRC interpretation explains that the provisions provided by the exception in the Technical Specifications may be applied to other types of testing besides Technical Specification 4.0.5 provided that the pump being tested is isolated from the reactor coolant system. The bases section in the Improved Technical Specifications states "An inoperable pump may be energized for test . . . provided the discharge of the pump is isolated from the reactor coolant system by closed isolation valve(s) with power removed from the valve operator(s) or by manual isolation valve(s) sealed in the closed position." The inspectors considered the Technical Specifications interpretation to be consistent with the Improved Technical Specifications submittal which does not refer to any specific type of testing.

Technical Specifications Interpretation 96-05, actions to be taken when both undervoltage relays are inoperable. The licensee rescinded this PSRC interpretation, because the clarification provided by Technical Specifications Interpretation 96-05 was made as a change to Technical Specification 3.3.2 by License Amendments 127 and 125 issued June 5, 1998.

Technical Specifications Interpretation 94-07, use of positive displacement pump during low temperature/overpressure conditions. Technical Specifications 3.2.1.3, 3.2.1.4, and 3.5.3 specifies the limitations on the number of operating centrifugal charging pumps for specific reactor coolant system conditions to ensure that the system is not overpressurized when cold. Results from analyses were summarized in Technical Specifications Interpretation 94-07 to clarify limitations considering centrifugal and positive displacement pump operation to ensure that the system is not cold overpressurized. The interpretation attachment provides clear direction on the use of the two centrifugal charging pumps and the positive displacement pump for specified reactor coolant system conditions. This is consistent with the Improved Technical Specifications submittal.

c. Conclusions

Because of previous problems with the appropriateness of Technical Specifications interpretations, the inspectors assessed a sample of five Technical Specifications interpretations. The inspectors concluded that the licensee properly implemented NRC guidance for license amendment submittals and use of interim administrative controls when performing Technical Specifications interpretations.

E8.5 (Closed) Unresolved Item 50-275; 323/96024-04: performance of primary flow determination at the end of the cycle.

Background

Historically, the licensee calculated the reactor coolant system primary flow at the beginning of the cycle. Based on information in the calculation, the licensee calibrated the reactor coolant system flow transmitters, which is an input into the reactor trip setpoints. To calculate the reactor coolant system flow, the licensee used the secondary calorimetric and the actual reactor coolant system cold-leg and hot-leg

temperatures. The reactor coolant system flow rate is calculated every cycle. From this flow determination, the reactor coolant system flow was verified to be above design minimum values, and a reactor coolant system low flow trip was established.

In 1994, the licensee changed the reactor coolant system flow calibration procedure to calculate the primary flow at the end of the cycle to account for hot-leg temperature streaming effects. Hot-leg streaming occurs when the measured hot-leg temperature is higher than the actual bulk temperature. This effect resulted from a combination of core design (i.e., low leakage cores), physical location of the hot-leg thermowells, and the changes in power generation that occur over core life. The hot-leg streaming effect resulted in total reactor coolant system flow being under predicted.

Observations and Findings

Technical Specification Surveillance Requirement 4.2.3.5 requires a reactor coolant system total flow measurement once every 18 months while in Mode 1. The Technical Specifications did not specify whether the calculation should be performed at the end or the beginning of the cycle. However, standard industry practice was to perform this test at the beginning of the cycle to account for any changes in the primary or secondary flow characteristics introduced during refueling.

The licensee had changed the test procedure without a 10 CFR 50.59 review based upon the fact that: (1) Technical Specifications did not specify when the flow measurement was required and (2) there were no specific Updated Final Safety Analysis Report statements that specified when this measurement should be made. Engineers changed the procedure and included a penalty factor to account for larger measurement uncertainty at the end of the cycle versus the beginning of the cycle.

NRC had previously cited this issue as a violation of 10 CFR 50.59 in Inspection Report 50-275; 323/96-24. The procedure change did not consider the assumed measurement error noted in a Technical Specifications table for the reactor coolant system flow. The licensee admitted the violation, modified their 10 CFR 50.59 processes, modified the calibration procedure, returned the reactor coolant system flow calibration to the beginning of the cycle values, and performed training on this issue to preclude this in the future.

The inspectors reviewed Updated Final Safety Analysis Report, Sections 4, 5, 7, and 15, and the Technical Specifications. The inspectors did not find any reference that identified a time frame for performing the flow calibration, except every refueling cycle. However, major changes to the reactor coolant system flow can occur during refueling outages (e.g., steam generator tube plugging, new fuel assemblies, and reactor coolant pump maintenance). The inspectors determined that no significant changes to reactor coolant system flow occurred while the incorrect procedure change was in effect. Because of the previous violation of 10 CFR 50.59 and the lack of significant changes in reactor coolant system flow, the inspectors concluded that no additional violations occurred.

E8.6 (Closed) LER 50-275; 323/96-16-00: lack of 10 CFR 50.59 safety evaluation for changes to reactor coolant system precision flow calorimetric.

This LER is addressed by the closeout of Unresolved Item 50-275; 323/96024-04 discussed in Section E8.5.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Hot Particle Exposure

a. Inspection Scope (71750)

The inspectors evaluated the licensee response to AR A0478718, which described an event in which an individual was exposed to a hot particle.

b. Observations and Findings

On February 27, 1999, during Refueling Outage 1R9, radiation protection personnel discovered a discrete particle on an individual who had alarmed a portal monitor. Radiation Protection personnel immediately removed the particle from his chest using a moist wipe. Initial surveys of the particle with an RO-2 survey meter revealed 1.2 mr/hr using a closed window meter and 19 mr/hr with an open window survey meter on contact. The Radiation Protection Foreman performed a rough calculation and determined that the particle contained approximately 1 µCi of activity. The particle was packaged properly and was transported to the chemistry laboratory for formal counting.

The initial analysis of the discrete particle using a multichannel analyzer indicated a 10 µCi Cobalt-60 particle. The radiological engineer performed a calculation of the individual's skin dose based on 3 hours work in the radiologically controlled area. The engineer initially assigned a skin dose of 160 rem, significantly exceeding the NRC skin dose limit of 50 rem.

Because of the potential for an overexposure, the licensee performed a confirmatory analysis of the discrete particle. This time the analysis determined that a 1.174 µCi particle was present. Using this revised activity level, engineers calculated the skin dose of the exposed individual, resulting in an assigned dose of approximately 14 rem, less than regulatory limits. Repeated analyses continued to indicate the presence of a 1 µCi particle. The licensee determined empirically that the original dose assessment was in error because of improper geometry of the detector setup during analysis. Therefore the licensee assigned a skin dose of 13.648 rem and a whole body dose of 0.039 rem. The inspectors reviewed the dose assessment and investigation and determined that it was appropriate.

The inspectors reviewed the exposed individual's radiation work permits to determine if licensee and NRC requirements were properly followed. The individual entered the radiologically controlled area and supervised work associated with reactor split-pin replacement. He did not enter the posted "Hot Particle Zone." Approximately 1 hour later he installed shielding on the letdown line for 2 hours. The individual stated that he placed his right arm around the piping to facilitate shielding installation. The licensee believed that the hot particle rubbed off inside the exposed individual's protective clothing during this evolution. The inspectors reviewed the evaluation and considered it to be reasonable. In addition, the inspectors reviewed the applicable radiation work permits and determined that the individual was authorized for the work and took the requisite radiological precautions.

c. Conclusions

Licensee investigation of the circumstances that led to an individual receiving a hot particle exposure was thorough and appropriate to the circumstances. The licensee determined that personnel had input an incorrect geometry configuration into the multichannel analyzer. This resulted in the estimated dose (160 rem) exceeding regulatory limits. However, the estimated dose significantly improved (14 rem) using the correct geometry. The individual's exposure to the skin did not exceed regulatory limits, and no violations of NRC requirements were identified.

R8 Miscellaneous Radiation Protection and Chemistry Control Issues (92904)

R8.1 (Closed) Violation 50-275; 323/98001-01: failure to follow radiation work permit instructions.

The inspectors performed an in-office review of this violation. This Severity Level IV violation was issued in a Notice of Violation prior to March 11, 1999. On this date, the NRC changed the policy for treatment of Severity Level IV violations (Appendix C of the Enforcement Policy). Because this violation would have been treated as a noncited violation in accordance with Appendix C, it is being closed out in this report, consistent with the new Enforcement Policy for Severity Level IV violations. The inspectors verified that the licensee had generated AR A0431905 for this violation. In addition, this violation already has a docketed response. Review of the effectiveness of the corrective actions for selected violations will be performed in the future as a routine part of the review of the corrective action program.

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.

F8 Miscellaneous Fire Protection Issues (92904)

F8.1 (Closed) Violation 50-275; 323/98014-02: performance of 15-minute roving patrols in lieu of continuous fire watches.

The inspectors performed an in-office review of this violation. This Severity Level IV violation was issued in a Notice of Violation prior to March 11, 1999. On this date, the NRC changed the policy for treatment of Severity Level IV violations (Appendix C of the Enforcement Policy). Because this violation would have been treated as a noncited violation in accordance with Appendix C, it is being closed out in this report, consistent with the new Enforcement Policy for Severity Level IV violations. The inspectors verified that the licensee had generated AR A0465904 for this violation. In addition, this violation already has a docketed response. Review of the effectiveness of the corrective actions for selected violations will be performed in the future as a routine part of the review of the corrective action program.

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 23, 1999. 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

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

INSPECTION PROCEDURES (IP) USED

IP 37551	Onsite Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observation
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 90712	Inoffice Review of Written Reports of Nonroutine Events at Power Reactors
IP 92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901	Followup - Operations
IP 92902	Followup - Maintenance
IP 92903	Followup - Engineering
IP 92904	Followup - Plant Support
IP 93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED AND CLOSED

Opened

None.

Closed

50-323/98021-01	VIO	Failure to secure circulating water pump and failure to properly set atmospheric dump valve (Section O8.1)
50-275/98010-01	VIO	Mechanics drained oil from the wrong auxiliary feedwater pump (Section O8.1)
50-275; 323/98007-05	VIO	Failure to implement design of level indicating system into abnormal procedure (Section O8.1)
50-275; 323/98007-04	VIO	Multiple failures to implement clearance procedure . (Section O8.1)
50-323/98007-03	VIO	Failure to restore high flux alarm during core reload (Section O8.1)
50-323/98007-02	VIO	Failure to provide appropriate procedure for nonseismic hoist storage (Section O8.1)
50-275; 323/98002-01	VIO	Failure to implement sealed valve program (Section O8.1)
50-275/97-369-03 (03014)	VIO	Auxiliary saltwater trains made inoperable (Section O8.1)
50-275/97-369-02 (02014)	VIO	Valves unacceptable and negatively impacted the auxiliary saltwater system (Section O8.1)
50-275/97010-01	VIO	Failure to follow procedure for alignment of 480V power supply Panel PY-16 (Section O8.1)
50-323/97006-01	VIO	Failure to maintain an operator at-the-controls as required by procedure (Section O8.1)
50-275; 323/96021-01 96-469-02 (02014)	VIO	Failure to update the Final Safety Analysis Report (Section O8.1)
50-275; 323/96021-01 96-469-01 (01014)	VIO	Failure to perform 10 CFR 50.59 evaluation when revising Procedure E-1.3 (Section O8.1)

50-323/98-005-00	LER	Manual reactor trip because of heavy debris loading of the circulating water system during Pacific ocean storm (Section O8.2)
50-275/97-010-01 and -00	LER	Unplanned start of Diesel Engine Generator 1-1 because of a 4160V Bus H startup feeder phase potential transformer fuse that opened (Section O8.3)
50-323/97-003-01 and -00	LER	Manual reactor trip on loss of normal feedwater because of unknown condensate/feedwater transient (Section O8.4)
50-275;323/ 97-002-00	LER	Refueling water storage tank outside its design basis because of insufficient water margin at completion of switch over to cold-leg recirculation (Section O8.5)
50-275; 323/ 97-001-00	LER	CCW system has operated with procedural guidance that permitted operation in a condition outside the design basis of the plant (Section O8.6)
50-275;323/ 96021-07	URI	Nonconservative assumptions in timing of switchover to cold-leg recirculation (Section O8.7)
50-275/97-369-01 (01014)	VIO	Failure to monitor performance of auxiliary saltwater check valves (Section M8.1)
50-275/97-016-01	LER	Technical Specifications 3.3.1 and 3.3.2 not met following inadequate testing of reactor trip/engineered safety feature functions because of inadequate vendor design information discovered as a result of Generic Letter 96-01 (Section M8.2)
50-275/97-011-01 and -00	LER	Auxiliary saltwater systems outside design basis for flooding because of inadequate corrective actions resulting from personnel error (Section M8.3)
50-275/95-016-02	LER	Technical Specification 3.4.2.2 not met during pressurizer safety valve surveillance testing because of random setpoint spread (Section M8.4)
50-275/94-010-01	LER	Unit 1 shutdown in accordance with Technical Specification 3.8.1.1, Action b, because of a degraded condition on the "C" phase 500 kV transformer (Section M8.5)
50-275; 323/98009-01	VIO	10 CFR 50.59 violation because changes to CCW system allowed boiling in containment fan cooler units and change to emergency operating procedures affected a Technical Specification (Section E8.1)
50-275/98009-02	VIO	10 CFR 50.59 violation because auxiliary saltwater system modification located piping on ground other than bedrock (Section E8.1)

50-275; 323/98008-03	VIO	Failure to perform 10 CFR 50.59 evaluation for the change to the containment fan coil unit motor leads (Section E8.1)
50-275/97006-06	VIO	Failure to ensure spray shields enclose reactor coolant pump lift (Section E8.1)
50-275; 323/97003-05	VIO	Failure to adequately incorporate design basis into emergency operating procedures (Section E8.1)
50-275;323/ 98020-02	IFI	Evaluate adequacy of Maintenance Rule program for assessing structures (Section E8.2)
50-275; 323/98-013-00	LER	Actuations of engineered safety features - diesel engine generators started when startup power was lost because of an inappropriate relay setpoint (Section E8.3)
50-275; 323/ 97023-03	URI	Questionable Technical Specifications interpretations (Section E8.4)
50-275; 323/96024-04	URI	Performance of primary flow determination at the end of the cycle (Section E8.5)
50-275; 323/96-16-00	LER	Lack of 10 CFR 50.59 safety evaluation for changes to reactor coolant system precision flow calorimetric (Section E8.6)
50-275; 393/98001-01	VIO	Failure to follow radiation work permit instructions (Section R8.1)
50-275; 323/98014-02	VIO	Performance of 15-minute roving patrols in lieu of continuous fire watches (Section F8.1)

Opened and Closed

50-275/99004-01	NCV	Inadequate procedures for pressurizer sampling (Section O1.3)
50-275;323/ 99004-02	NCV	Inadequate procedures for controlling reactor protection channel bypassing (Section O1.4)
50-323/99004-03	NCV	Failure to implement the procedure for establishing a fire watch (Section O4.1)
50-275; 323/ 99004-04	NCV	Failure to follow administrative controls for on-line maintenance (Section O4.2)

LIST OF ACRONYMS USED

AOT	allowed outage time
AR	action request
CCW	component cooling water
CFR	Code of Federal Regulations
HX	heat exchanger
IFI	inspection followup item
IP	inspection procedure
LER	licensee event report
NCR	nonconformance report
NCV	noncited violation
PDP	positive displacement pump
PDR	Public Document Room
PRA	probabilistic risk assessment
PSRC	Plant Staff Review Committee
RHR	residual heat removal
RWST	refueling water storage tank
STP	surveillance test procedure
VIO	violation