U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee: Northeast Nuclear Energy Company
P. O. Box 128
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Facility: Millstone Nuclear Power Station, Units 1, 2, and 3

Inspection at: Waterford, CT

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EXECUTIVE SUMMARY

Millstone Nuclear Power Station

Combined Inspection 50-245/99-09; 50-336/99-09; 50-423/99-09

Operations

- At Unit 2, on September 17, 1999, operators appropriately implemented the required actions for a dropped control rod and performed a reactor shutdown in a controlled and deliberate manner. Operator performance was particularly good in that the dropped rod occurred at a time they were responding to numerous main condenser alarms that resulted when the "D" circulating water pump was secured. (Section U2.O1.2)

- At Unit 2, the circulating water system operating procedure was weak in that it did not specify the reactor power level to allow securing a circulating water pump. As a result when a circulating water pump was secured at 85% power, numerous main condenser alarms and reduced main condenser vacuum occurred, which unnecessarily challenged plant operators. (Section U2.O1.2)

- At Unit 2, the licensee failed to revise the plant cooldown procedure to reflect a change in reactor protection system (RPS) setpoints in April 1999. As a result, an unplanned RPS actuation on low steam generator level occurred while the plant was shut down. The failure to adequately maintain the plant cooldown procedure is being treated as a Non-Cited Violation (NCV 50-336/99-09-02). (Section U2.O1.3)

- The licensee incorrectly retracted their initial notification of an unplanned RPS actuation. This retraction constitutes a failure to report a condition as required by 10 CFR 50.72 and is being treated as a Non-Cited Violation (NCV 50-336/99-09-03). (Section U2.O1.3)

- At Unit 2, during the forced outage that occurred due to a dropped control rod, the licensee's corrective actions were found acceptable in addressing an electrical ground in the wire to the lower gripper coil which caused the rod to drop. The pre-evolution briefing of operators prior to reactor startup was thorough with a good discussion of industry operating experience and extra operator staffing was provided during the startup. (Section U2.O1.4)

- At Unit 2, the NRC identified that the licensee failed to initiate a condition report to document that, during the reactor startup, reactor criticality occurred at a higher power level than expected which resulted in the recording of critical data at power level that was higher than power level specified in the procedure. A condition report is necessary to ensure that the cause of the higher power level is evaluated and that corrective actions are taken to prevent recurrence during future reactor startups. The failure to initiate a condition report as required by the corrective action procedure is being treated as a Non-Cited Violation (NCV 50-336/99-09-04). (Section U2.O1.4)

- At Unit 2, when a control circuit card failed for the main feedwater regulating bypass valve, reactor operator performance was good in immediately recognizing, before any alarms were received, that the bypass valve had closed. Due to prompt action by the reactor operator to restore main feedwater flow, a plant transient was averted. (Section U2.M1.1)

- The licensee responded well to a leak in the Unit 3 condensate demineralizer system. Operators isolated the leak and stabilized power at 80 percent to control the transient which followed. Effective operations command and control and appropriate licensee cleanup efforts were observed during and after the transient. The licensee demonstrated thorough followup to determine why a damaged demineralizer drain valve did not fail closed, as expected. (Section U3.O1.1)

- On September 10, 1998, with Unit 3 in Mode 1, both trains of the service water (SW) system were declared inoperable when check valves associated with sodium hypochlorite injection failed to reseat following a surveillance test. The valve failures resulted from internal valve corrosion, caused by the placement of hypochlorite-susceptible materials in portions of the system subjected to relatively high concentrations of hypochlorite. The failure to adequately implement design controls to ensure that the SW design basis was correctly translated into specifications, drawings and procedures was a violation and was appropriately identified, documented, and corrected. Associated LER 50-423/98-37 is closed. This violation is being treated as non-cited. (NCV 50-423/99-09-08) (Section U3.O8.7)

- At Unit 3, the licensee reported on January 6, 1998, that the motor pinion gear keys in three out of four SI accumulator isolation valves were sheared. The condition was adequately responded to and resolved by the licensee. The root cause was a failure to adequately translate the valve design criteria into appropriate design specifications for the pinion keys. This failure is a violation and is being treated as non-cited (NCV 50-423/99-09-09). LER 50-423/98-01 is closed. (Section U3.O8.11)

- The licensee reported on September 9, 1997, that the Unit 3 Emergency Diesel Generator (EDG) coolers would not have met their design criteria for thermal performance due to fouling that resulted from historical fuel oil leaks into the EDG fresh water cooling water system from the fuel oil injectors. The fouling was not identified because of inadequate thermal performance testing. The condition was adequately responded to and resolved by the licensee. The failure to adequately implement design controls to ensure that the EDG design basis was correctly translated into specifications, drawings and procedures is a violation and is being treated as non-cited. (NCV 50-423/99-09-10) LER 50-423/98-01 is closed. (Section U3.O8.12)

- On February 6, 1998, the licensee reported that the Unit 3 Containment Air Monitor (CAM) alarm and alert setpoints were set above those indicated in the Millstone Unit 3 Final Safety Analysis Report (FSAR). The condition was adequately responded to and resolved by the licensee. Failing to adequately implement design controls to ensure that the CAM design basis was correctly translated into specifications, drawings and procedures is a violation and is being treated at non-cited. (NCV 50-423/99-09-11) LER 50-423/98-09 is closed.

Maintenance

- At Unit 2, the replacement of a control circuit card for the main feedwater regulating bypass valve was well coordinated with operations and appropriate contingency measures were in place in the event an unexpected feedwater transient occurred during the card replacement. (Section U2.M1.1)

- Observed Unit 3 surveillance activities were performed in a controlled manner, in accordance with approved procedures. Where testing problems arose or failures occurred, the licensee developed action plans to evaluate and correct the identified concerns. To address any generic questions of component or system operability, the licensee prudently scheduled additional testing on an expedited basis, as permitted by the overall plant conditions. (Section U3.M1.1)

- Inspection-tours of Unit 3, including observation of ongoing maintenance and modification activities, identified some issues that required follow-up, for which the licensee appropriately issued condition reports. While most of these items were minor, one finding involving the ineffective implementation of safety-related design control measures for the relocation of an emergency lighting box resulted in the identification of a Non-Cited Violation (NCV 50-423/99-09-12). (Section U3.M2.1)

Engineering

- At Unit 2, the NRC found that design control measures were inadequate to assure that the 4160 volt switchgear room coolers were capable of maintaining a suitable environment for the vital switchgear under post-accident conditions. This Severity Level IV violation of design control requirements is being treated as a Non-Cited Violation (NCV 50-336/99-09-05). The licensee's determination that the "A" train 4160 volt switchgear remained operable when the switchgear room cooler was removed from service for corrective maintenance was found acceptable. (Section U2.E1.1)

- At Unit 2, the licensee identified in 1997 that a valve that isolates the shutdown cooling system from the reactor coolant system was vulnerable to a fire-induced hot short. The licensee's corrective actions were found acceptable. This violation of 10 CFR 50, Appendix R, Fire Protection, is being treated as a Non-Cited Violation (NCV 50-336/99-09-06). Licensee Event Report 50-336/97-35-00 is closed. (Section U2.E8.2)

- At Unit 2, the licensee identified in 1998 that the design analysis of the interaction between the reactor vessel internals and the reactor vessel did not properly address dynamic loading associated with a loss of coolant accident or a design bases earthquake. This violation of 10 CFR 50 Appendix B, Criterion III, Design Control, is being treated as a Non-Cited Violation. (NCV 50-336/99-09-07) The licensee's corrective actions were found acceptable. Licensee Event Report 50-336/98-03-00 is closed. (Section U2.E8.4)

- At Unit 3, the repetitive failure of the Recirculation Spray System cubicle sump pumps appears to stem from ineffective root cause evaluations by engineering, in that, the evaluations did not identify that swelling and elongation of the air motor vanes could be a potential failure mechanism for the recirculation system cubicle sump pumps. An unresolved item (URI 50-423/99-09-13) was identified pending further review of the analysis of failed recirculation system cubicle sump pump 3DAS*P15A, review of Technical Evaluation M3-EV-99-0098, review of the 10 CFR Part 21 reportability determination, and review of the results of the monthly measurements of vane elongation. (Section U3.E2.1)

Plant Support

- Continuing investigation into a July 8, 1999, 7.04 rem personnel TLD exposure is focused on resolving the potential for deliberate tampering and irradiating of the TLD. While the overall investigation is not yet completed, the licensee has initiated several corrective measures to address contributing or potential causes, and effect improved positive control of TLDs used to monitor personnel exposure. (Section R1.1)

- Radiological controls were effectively implemented during a Unit 2 forced outage in late September 1999. (Section R1.2)

- The internal exposure measurement and dose assessment program at Millstone is effective. (Section R1.3)

- The RP program has an active oversight and self-assessment program that engages problems in an effective manner. (Section R7.1)

Report Details

Summary of Unit 1 Status

At Unit 1, the licensee continued to progress toward decommissioning of the unit. Overall, licensee activities were conducted in a safe and deliberate manner, with no significant safety issues identified to date. Transfer of 184 new fuel assemblies from the spent fuel pool to the new fuel storage vault began, with shipment to a fuel vendor to follow. In addition, the licensee continued in the preparation for the programmatic and physical separation of Unit 1 from the other two operating units.

U1.I Operations

U1 O1 Conduct of Operations

O1.1 Spent Fuel Pool Heatup Test

a. Inspection Scope (71707)

The inspector observed the licensee's preparation for and performance of a spent fuel pool (SFP) heatup test.

b. Observations and Findings

The licensee conducted a SFP heatup test on September 8-9, 1999. The heatup test was conducted to validate a vendor heat load calculation in support of the cooling system design for the decommissioning spent fuel pool island project. The inspector reviewed the safety evaluation initiated in support of the heatup test implementing procedure, SPROC OPS-99-1-05, "Spent Fuel Pool Decay Heat Measurement Test," and determined that the evaluation adequately supported the procedure.

The inspector also observed the pre-job brief conducted on September 8. The inspector determined that the brief covered all applicable topics as required by unit procedures, was attended by the appropriate personnel, and included a good example of lessons learned from a similar test that had been performed at another site. In addition, termination criteria were discussed, as well as plant conditions that could have impacted the performance of the test.

The test was initiated on September 8, when the fuel pool cooling (FPC) pumps were secured in accordance with the applicable operating procedure, and the spent fuel pool at approximately 89°F. On September 9, 1999, the 105°F termination criteria was reached and the FPC pumps were subsequently started. No adverse conditions were observed during performance of the test.

c. Conclusions

The inspector concluded that the licensee's performance of the spent fuel pool heatup test was satisfactory.

O1.2 Preparation For New Fuel Recovery

a. Inspection Scope (60705)

The inspector reviewed the licensee's activities in preparation for the transfer of 184 new fuel assemblies from the spent fuel pool (SFP) to the new fuel storage vault (NFSV).

b. Observations and Findings

In support of the unit decommissioning, the licensee commenced preparations for the removal of 184 new fuel assemblies that have been stored in the SFP, and the subsequent transfer of these assemblies to the NFSV. The inspector verified that both the fuel movement supervisors and the refuel bridge operators had been appropriately trained and qualified prior to performance of actual fuel movement. In addition, the licensee performed the appropriate tests and inspections to support the operation of the refuel bridge crane for in-pool fuel movement, as well as the reactor building overhead crane for both channel removal and new fuel assembly transfer to the NFSV. The licensee also completed modifications to the fuel preparation machine to support the new fuel transfer, and installed the underwater filter skid for pool clarity. Minor delays occurred due to emergent issues, such as difficulties in filter replacement and in-pool alignment for the underwater filter, as well as administrative engineering qualification concerns. The licensee maintained an appropriate focus on the satisfactory completion of the applicable activities, as indicated by the number of Plant Operations Review Committee (PORC) meetings that occurred before the licensee approved the main procedure for the fuel move, SPROC ENG 99-1-03, "New Fuel Recovery."

The inspector observed a number of licensee PORC meetings where the various design modifications, procedures, and safety evaluations in support of the new fuel recovery were reviewed and assessed for adequacy and approval. The inspector observed an appropriate level of questioning from the various PORC members regarding the submittals, and overall, the PORC was effective in fulfillment of the technical specification requirements. In addition, the inspector reviewed a number of safety evaluations performed in support of the new fuel recovery, such as SPROC ENG 99-1-03, and identified no significant issues.

In preparation for the new fuel recovery, as well as to increase the available space on the refueling floor, the licensee successfully disassembled the reactor vessel head insulation package. This process was conducted in a safe and deliberate manner, had appropriate health physics coverage, and was completed without incident. Due to the presence of asbestos in the insulation package, an approved vendor successfully removed the affected portion of the package which was stored for future abatement. No abnormal radiological or safety issues were identified throughout the procedure, with the exception of some minor detectable contamination that was successfully decontaminated such that further disassembly could continue. No airborne activity was detected and health physics personnel conducted the appropriate surveys throughout the entire process.

The inspector observed the pre-job brief for the new fuel recovery, and observed that the majority of personnel involved with the evolution were present. Personnel who were not present were provided with the appropriate pre-briefs prior to the performance of any work. Overall, the brief provided an excellent overview of the various parallel activities that would be performed for the new fuel recovery, and included management's expectations regarding safe and deliberate performance of each activity. The brief also included good use of visual aids, and an appropriate use of operating experience and lessons learned that included events that had occurred previously at Unit 1. In addition, the importance of documentation required for the movement of special nuclear material, the possible radiation exposure hazards that could occur during the evolutions, and the foreign material exclusion issues that are inherent in SFP activities were also discussed.

c. Conclusions

The inspector concluded that the licensee's preparation for the new fuel recovery was good, and an appropriate focus on personnel and radiological safety was maintained.

O1.3 Hurricane Floyd Preparations

a. Inspection Scope (71707)

The inspector observed the licensee's preparation and actions both prior to, and during the arrival of Hurricane Floyd.

b. Observations and findings

The inspector observed the licensee's activities as they prepared for the approach of Hurricane Floyd on September 16, 1999. The licensee performed the applicable actions from the appropriate unit and site procedures to address the high winds, tropical storm warnings, and the hurricane approach.

In addition, the licensee exhibited appropriate prioritization regarding the expeditious return to service of the "D" service water pump (SWP) during the hours preceding the hurricane's arrival at the site. Specifically, the SWP had been electrically disconnected following receipt of test results that indicated a potential wet or dirty motor. As such, the Unit 1 emergency diesel generator (EDG), which is supplied cooling water by the "D" SWP, was declared unavailable to support Unit 2 operations, specifically, for station blackout and Appendix R events. The licensee subsequently performed the necessary work and returned both the "D" SWP and the Unit 1 EDG to service prior to the arrival of Hurricane Floyd.

While the overall actions by the licensee in preparation for the hurricane were good, the information exchange among the three units appeared to be inconsistent. Specifically, the classification as a hurricane or a tropical storm was different among the three control rooms, however, the overall impact was minimal based on the substantive actions already completed by the licensee in preparation for the Hurricane Floyd's arrival. This issue was appropriately captured by the licensee and detailed in a condition report.

c. Conclusions

The inspector concluded that the licensee's actions prior to, and during the approach of Hurricane Floyd were good. In addition, the licensee expeditiously returned vital Unit 1 equipment to service prior to the hurricane's arrival in support of the continued operation of Unit 2.

U1 O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Closed) Unresolved Item (URI) 50-245,336,423/96-09-03: Organizational Changes: The inspector reviewed the licensee's actions following their identification and evaluation of potential non-compliances regarding organizational changes across the Millstone site. These original organizational change issues were detailed in NRC Inspection Report (IR) 50-245,336,423/96-09, and culminated in the issuance of Violation (VIO) 50-245,336,423/96-09-04. Subsequently, the NRC determined that the licensee had implemented appropriate corrective actions for VIO 96-09-04, and the violation was closed in NRC IR 97-203, in November 1997. Specific to URI 96-09-03, the inspector identified that the licensee implemented numerous corrective actions. For example, procedure RAC-13, "Organizational Changes," was implemented to establish a process for future organizational changes to ensure compliance with the licensing basis and applicable regulatory requirements. While this specific corrective action was appropriate, a number of procedural non-compliances with RAC-13 were identified by Nuclear Oversight in February 1999, which indicated that corrective actions implemented to prevent recurrence were not effective. As a result, the licensee performed specific actions to resolve the issue, which included: (1) a root cause investigation was performed; (2) the Licensing Basis Ownership Group was formed to restore compliance to the current licensing basis; (3) a stop work order was issued against further organizational changes; and (4) transition plans were developed and provided guidance for the licensee during the implementation of the organizational realignment.

NRC IR 50-423/99-07 (40500 team inspection June 1999) documents the NRC disposition of these issues as a Non-Cited Violation (NCV 99-07-03). The NCV addressed the adequacy of corrective actions for the organizational changes, which were implemented both before and during the transition to the current site management structure, and were initiated in a manner that did not preserve the licensing basis. As a result, the inspector concluded that both the licensee's numerous corrective actions and the NRC's disposition from an enforcement perspective adequately address the relevant issues identified in the URI, therefore, **URI 50-245,336,423/96-09-03 is closed.**

O8.2 (Closed) Unresolved Item (URI) 50-245/97-02-02: RP-4 Interface with Lower Tier Reporting Processes: The inspector reviewed the licensee's actions regarding URI 97-02-02, which were discussed in NRC Inspection Report (IR) 97-02. The URI was initiated due to the potential that the multiple site deficiency identification programs that existed at the Millstone site could have circumvented the corrective action process as detailed in procedure RP-4, "Corrective Action Program." The inspector reviewed the licensee corrective actions as detailed in condition report (CR) M1-97-1119, which was generated to address the RP-4 interface issue, and found the corrective actions to be appropriate and adequate. However, the inspector identified one corrective action from the CR regarding security department instructions that had been listed as "complete," but no evidence of implementation could be identified. The licensee subsequently initiated a revision to the department instruction that adequately addressed the relatively minor administrative corrective action. In addition, the inspector identified inconsistencies between guidance contained in the security instruction and the emergency classification for security events contained in the emergency plan implementing procedures. However, the revision to the security instruction previously discussed adequately resolved the inconsistency issue. The inspector concluded that the corrective actions to address the RP-4 interface issue were adequate, and included numerous improvements to the corrective action program. In addition, while a corrective action was identified to have not been completed, the inspector concluded that the issue was minor and no other violations of NRC requirements were identified. Therefore, **URI 50-245/97-02-02** is **closed.**

O8.3 (Closed) Violation (VIO) 50-245/95-07-01: Control Room Habitability/Use of Self-Contained Breathing Apparatus (SCBAs): The inspector reviewed the licensee's actions to address VIO 95-07-01, specifically, the actions to address a previous NRC inspection of this issue as discussed in NRC Inspection Report 50-245/96-01. The inspector concluded that the licensee established appropriate corrective actions that adequately resolved the issue, therefore, **VIO 50-245/95-07-01 is closed.**

O8.4 (Closed) Licensee Event Report (LER) 50-245/97-013-00 & 01: Evaluation of Impact of Refueling Platform Fuel Grapple: LERs 97-013-00 and 01, generated in March 1997, and December 1998, respectively, documented conditions outside of the licensee's design basis. Specifically, the licensee-identified condition regarded additional kinetic energy from the weight of lower sections of the refueling platform fuel mast that were not included in both the impact load for the design of the spent fuel pool (S.P.) storage racks, and the applicable portions of the fuel handling accident contained in Chapter 15 of the Final Safety Analysis Report (FSAR), which directly related to fuel stored in the reactor vessel.

The inspector reviewed the design basis issue, as well as the licensee's corrective actions. As a result, the inspector concluded that the failure to include the weight of the lower mast in support of both the S.P. storage rack design and the fuel handling accident of the FSAR is a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." However, the licensee's certification to the NRC in accordance with 10 CFR 50.82, mitigates the current consequences of the fuel handling accident issue in that fuel has been permanently removed from the reactor vessel. In addition, the licensee has implemented adequate corrective actions for the impact analysis on the S.P. storage racks with revised analyses that have been incorporated into the appropriate design basis documents. Therefore, this Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into the licensee's corrective action program. Therefore, **LERs 50-245/97-013-00 and 01** are **closed. (NCV 50-245/99-09-01)**

O8.5 <u>(Closed) Licensee Event Reports (LERs) 50-245/96-023-00 & 01: Movement of New Fuel Assemblies Over the Spent Fuel Pool</u>: The licensee generated LERs 96-023-00 & 01 in April 1996, and May 1998, respectively, after they identified that new fuel assemblies had been transferred over the spent fuel pool (S.P.) that introduced a new potential radiological consequence that was not addressed by design and licensing bases. However, the licensee subsequently performed calculations that demonstrated the radiological consequences that resulted from a drop of a new fuel assembly onto irradiated fuel in the S.P. from the maximum height possible by the reactor building overhead crane, were enveloped by the existing fuel handling accident in the Final Safety Analysis Report. While the licensee conservatively implemented a number of administrative corrective actions to prevent the movement of new fuel over irradiated fuel to mitigate the consequences of a potential fuel drop, the inspector concluded that no violations of NRC requirements had occurred. Therefore, **LERs 50-245/96-023-00 & 01** are **closed**.

Report Details

Summary of Unit 2 Status

Unit 2 entered the inspection period in Operational Mode 1, power operation, with the plant at 100 percent power. Operators reduced power to levels between 80 and 90 percent on the following four occasions to clear fouling of the "A" and "C" main condenser waterboxes and circulating water pump bays by saltwater mussels: August 27 through 28, 1999, August 29, 1999, August 31 through September 5, 1999, and September 10 through 12, 1999.

When debris generated by Tropical Storm Floyd fouled the main condenser on September 17, 1999, a power reduction was initiated to allow cleaning the "D" main condenser waterbox. While inserting the regulating bank of control rods during the power reduction, a single control rod in the bank became misaligned when a drive mechanism latch failed to engage the control rod due to a ground in its power supply. When operators were unable to realign the control rod within the required time, the licensee entered Technical Specification 3.0.3 and executed a reactor shutdown to Operational Mode 3, hot standby. In addition to other maintenance activities, the licensee repaired the failed control rod drive mechanism, checked other control rods for similar problems, and completed cleaning the main condenser waterboxes and circulating water bays before starting the reactor on September 23, 1999. On September 25, 1999, the plant reached 100 percent power. At the conclusion of the inspection period, the plant remained in operation at 100 percent power.

U2.I Operations

U2 O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspector conducted frequent reviews of ongoing plant operations, including observations of operator evolutions in the control room; walkdowns of the main control boards; tours of the Unit 2 radiologically controlled area and other buildings housing safety-related equipment; observations of onsite safety review committee meetings; and observations of several management planning meetings.

In general the inspectors noted good turnovers and good communication practices among operators in the control room. In addition, the inspectors found the licensee's preparations for Hurricane Floyd were generally good and the actions prescribed in abnormal operating procedure (AOP) 2560, "Storms, High Winds and High Tides," were properly implemented. The hurricane had been downgraded to a tropical storm prior to its arrival. During the storm, there were no significant equipment problems or operational challenges.

On September 9, 1999, the licensee notified the NRC in accordance with 10 CFR 50.72(b)(1)(i)(A), that they were initiating a plant shutdown because service water temperature was 77°F, which was 2°F above the ultimate heat sink temperature limit specified in technical specifications. With the ultimate heat sink inoperable, Technical Specification 3.7.11 requires the unit to be placed in hot standby within 6 hours, and cold shutdown within the following 30 hours. Approximately 10 minutes after commencing plant shutdown, the service water temperatures returned to normal and the technical specification that required a shutdown was exited. Reactor power was initially at 85% to facilitate cleaning of mussels from the circulating water bays and was reduced to 84% during this 10 minute period. The licensee found the temperature increase at Unit 2 was caused by thermal backwashing at Unit 3, which is the first time this activity has significantly affected Unit 2. Licensee corrective actions included evaluating the tidal effects and the timing of the Unit 3 backwash and providing procedural guidance to prevent recurrence. The licensee's actions were found acceptable.

O1.2 Technical Specification Required Shutdown Due to Failed Control Rod Drive Mechanism

a. Inspection Scope (71707)

The inspector observed a reactor power reduction for condenser waterbox cleaning and control room operator response to a subsequent failure of one control rod drive mechanism. The inspector evaluated procedural adherence and conformance with technical specification requirements during the evolution.

b. Observations and Findings

While conducting a planned power reduction for main condenser waterbox cleaning on September 17, 1999, control rod No. 65 became misaligned from the remaining control rods in the regulating group. When the regulating group was being inserted to 162 steps withdrawn to control axial power distribution during the power reduction, control rod No. 65 slipped to a position of 151 steps withdrawn. In response, operators appropriately entered the Technical Specification (TS) 3.1.3.1 action statement for a misaligned rod and performed the actions of abnormal operating procedure (AOP) 2556, "Control Element Assembly Malfunction."

During the attempt to recover control rod No. 65, the control rod slipped further to 122 steps withdrawn. Operators appropriately entered the TS 3.1.3.1 action statement for a dropped rod, which applies when one or more control rods are misaligned from other control rods in their group by 20 steps or more. As required by the action statement operators reduced reactor power to less than or equal 70 percent within one hour. This action statement also requires that, within one hour after reducing reactor power, the licensee must restore the control rod to within the specified alignment requirements. During this one hour period, the licensee initiated troubleshooting but was unsuccessful in determining the cause of the control rod failure. While troubleshooting, control rod No. 65 slipped to 77 steps withdrawn when they attempted to move the rod. When they determined that the control rod could not be recovered within the one hour, operators entered TS 3.0.3 and completed the required shutdown to Operational Mode 3, hot standby, without incident. The licensee properly reported this to the NRC in accordance with CFR 50.72(b)(1)(i)(A), as a plant shutdown required by technical specifications.

The licensee identified a procedural enhancement in that the existing abnormal operating procedure for a dropped control rod assumed that the failed control rod was fully inserted, and, consequently, the procedure did not provide specific instructions to complete insertion of the affected control rod group. The licensee elected to trip the failed control rod and proceed with the reactor shutdown in accordance with the written procedures. The inspector found this acceptable.

A procedural weakness in the circulating water system operating procedure resulted in a minor plant transient that unnecessarily challenged plant operators. Because no specific procedural guidance was provided, operators used personnel experience to estimate the reactor power level that would allow securing one circulating water pump. Operators reduced reactor power to 85 percent, which was sufficient in the past, but was not sufficient in this instance due to the degree of fouling that existed in the other three condenser waterbox es. As a result, when the "D" circulating water pump was secured, numerous alarms occurred related to high condenser differential temperatures and reduced main condenser vacuum. Response to these alarms distracted the operators during their response to the misaligned control rod, which occurred 10 minutes after the "D" circulating water pump was secured. The licensee plans to implement a procedural enhancement to better define the necessary power reduction to support securing a circulating pump under various conditions.

While shut down, the licensee determined that the cause of the dropped control rod was an electrical ground in the wire to the lower gripper coil. This ground was corrected, other control rods were tested for grounds and none were identified.

c. Conclusions

Operators appropriately implemented the required actions for a dropped control rod and performed a reactor shutdown in a controlled and deliberate manner. Operator performance was particularly good in that the dropped rod occurred at a time they were responding to numerous main condenser alarms that resulted when the "D" circulating water pump was secured.

The circulating water system operating procedure was weak in that it did not specify the reactor power level to allow securing a circulating water pump. As a result, when a circulating water pump was secured at 85% power, numerous main condenser alarms and reduced main condenser vacuum occurred, which unnecessarily challenged plant operators.

O1.3 Inadvertent Actuation of the Reactor Protection System

a. Inspection Scope (71707)

The inspector reviewed an inadvertent actuation of the reactor protection system (RPS) that occurred on September 19, 1999, due to low steam generator water level, while the unit was shut down.

b. Observations and Findings

While cooling down the plant, operators were maintaining steam generator levels in the lower end of the control band of 45 percent to 75 percent as specified in procedure OP2207, "Plant Cooldown." However, because procedure had not been updated to reflect a change in RPS setpoint, a steam generator low level trip was processed on all RPS channels when steam generator levels reached the new trip setpoint of 49.5 percent. At the time, the plant was in Operational Mode 3, hot standby, and the reactor trip breakers were already open.

Technical Specification 2.2.1 requires that while in Operational Modes 1 and 2, power operation and startup, the RPS trip for low steam generator level must be in service with a setpoint of greater than or equal to 48.5 percent. On April 8, 1999, the NRC approved Technical Specification Amendment No. 232 which changed the RPS setpoint for low steam generator water level from greater than or equal to 36 percent to greater than or equal to 48.5 percent. However, procedure OP 2207 was not updated to reflect this change. Technical Specification 6.8.1 requires that written procedures be established and maintained covering activities listed in Appendix "A" of Regulatory Guide 1.33, which includes the plant cooldown procedure. The failure to adequately establish and maintain procedure OP 2207 to reflect the change in RPS setpoint is a violation. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program. This issue was documented in Condition Report M2-99-2482.

On September 19, 1999, the licensee reported this RPS actuation in accordance with 10 CFR 50.72(b)(2)(ii), as an event or condition that results in an actuation of an engineered safety feature (ESF), including the RPS. Upon further review, the licensee determined that the RPS trip was not reportable because: (1) an ESF was not needed to mitigate the consequences of an event because steam generator levels were being maintained with the range specified in procedure OP 2207 and, (2) because the reactor trip breakers were already open, an ESF signal did not process to an ESF component and therefore, an actuation of an ESF did not occur. Accordingly, on September 29, 1999, the licensee notified the NRC they were retracting their previous notification of this event.

The inspector evaluated the licensee's reportability evaluation associated with this inadvertent RPS actuation, determined that this actuation was reportable and that their decision to retract the notification was incorrect. Valid ESF actuations, which are actuations that result from valid signals that are initiated from actual plant conditions, are required to be reported unless the actuation is part of preplanned test or evolution. In this case, the low steam generator water level was an actual condition that created a valid signal resulting in a valid ESF actuation when the steam generator low level trip was processed on all RPS channels at the prescribed setpoint of 49.5 percent. The NRC considered that a valid RPS actuation occurred even though the reactor trip breakers were already open. In addition, although the reduction in steam generator level was specified in procedure OP 2207, the RPS actuation was not part of the planned procedure.

The inspector discussed the NRC position regarding the reportability of the RPS actuation with licensee management who agreed with the NRC position. On October 19, 1999, the licensee notified the NRC that their original determination on October 15, 1999, that the RPS actuation was reportable was correct and should not have been retracted. This retraction constitutes a failure to report a condition as required and is a violation of 10 CFR 50.72(b)(2)(ii). This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into the corrective action program. This issued was entered as Condition Report M2-99-2686.

c. Conclusions

The licensee failed to revise the plant cooldown procedure to reflect a change in RPS setpoint in April 1999. As a result, an unplanned RPS actuation on low steam generator level occurred while the plant was shut down. The failure to adequately maintain the plant cooldown procedure is being treated as a **Non-Cited Violation (NCV 50-336/99-09-02)**. The licensee incorrectly retracted their initial notification of this unplanned RPS actuation. This retraction constitutes a failure to report a condition as required by 10 CFR 50.72(b)(2)(ii) and is being treated as a **Non-Cited Violation (NCV 50-336/99-09-03)**.

O1.4 Reactor Startup Following Forced Outage

a. Inspection Scope (71707)

The inspector reviewed the licensee's actions to correct the cause of an inoperable control rod drive mechanism that resulted in a technical specification required shutdown on September 17, 1999. The inspector also observed the subsequent reactor startup on September 23, 1999, and power ascension activities.

b. Observations and Findings

While shut down, the licensee determined that the cause of the dropped control rod was an electrical ground in the wire to the lower gripper coil. The licensee corrected this ground, tested other control rod mechanisms for grounds, and identified no similar deficiencies. The inspector found these corrective actions acceptable.

The inspector found that the pre-evolution briefing of operators prior to startup was thorough with a good discussion of industry operating experience related to reactor startups. Extra operator staffing was provided including a reactor operator for control rod manipulations and a dedicated unit supervisor to oversee the startup of the reactor.

During the startup, the reactor was brought to a critical condition at a higher power level than expected. Procedure 2202, "Reactor Startup," step 4.4.5, provides the instructions for withdrawing the regulating group of control rods to establish criticality. Step 4.4.10 instructs operators to raise and stabilize reactor power to approximately 1×10^{-4} percent using control rods. Step 4.4.11 states that when power has stabilized at approximately 1

X 10^{-4} percent, refer to OPS Form 2208-1, "Estimated Critical Position Data and Analysis Sheet," and record critical data. However, operators were unable to follow Steps 4.4.10 and 4.4.11 because, when the reactor was declared critical, reactor power was already greater than 1 X 10^{-4} percent. Reactor power was stabilized and critical data was taken at approximately 2 X 10^{-3} percent. Operators discussed with reactor engineering whether they should insert control rods to reduce power to approximately 1 X 10^{-4} percent. Because inserting rods at that point would drive the reactor subcritical, they decided to record critical data at approximately 2 X 10^{-3} percent.

The inspector was concerned that although operators appropriately followed the instructions in Step 4.4.5 of procedure OP 2202 to establish criticality, they were not able to perform Steps 4.4.10 and 4.4.11 as written because reactor criticality occurred at a higher power level than expected which resulted in the recording of critical data at approximately 2 X 10^{-3} percent rather than approximately 1 X 10^{-4} percent that was specified. Because the reactor had just been taken critical, this was not a point in the procedure that would allow stopping to process a procedure change. Therefore, the inspector was primarily concerned with the licensee's failure to document this procedural adherence concern in a condition report to ensure that the cause of the higher power level is evaluated and that corrective actions are taken to prevent recurrence during future reactor startups. The inspector discussed this concern with licensee management who agreed and created Condition Report M2-99-2613. Administrative procedure RP4, "Corrective Action Program," requires plant personnel to initiate condition reports for conditions that have a potential or actual adverse effect on plant safety. The failure of the licensee to initiate a condition report as required by administrative procedure RP4, to document that critical data was not taken at the specified power level was a violation of 10 CFR Part 50, Appendix B, Criterion V, "Instruction, Procedure, and Drawing." This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into the corrective action program.

c. Conclusions

During the forced outage that occurred due to a dropped control rod, the licensee's corrective actions were found acceptable in addressing an electrical ground in the wire to the lower gripper coil which caused the rod to drop. The pre-evolution briefing of operators prior to reactor startup was thorough with a good discussion of industry operating experience and extra operator staffing was provided during the startup.

The NRC identified that the licensee failed to initiate a condition report to document that during the reactor startup, reactor criticality occurred at a higher power level than expected which resulted in the recording of critical data at power level that was higher than power level specified in the procedure. A condition report is necessary to ensure that the cause of the higher power level is evaluated and that corrective actions are taken to prevent recurrence during future reactor startups. The failure of the licensee to initiate a condition report as required by their corrective action procedure is being treated as a **Non-Cited Violation (NCV 50-336/99-09-04)**.

U2 O8 Miscellaneous Operations Issues (92700)

O8.1 (Closed) LER 50-336/97-22-00, -01, -02, & -03: Technical Specification Violations

The inspector conducted in-office and onsite reviews of Licensee Event Report (LER) 50-336/97-22-00, -01, -02 & -03, which involves examples of various historical technical specification (TS) violations that the licensee identified during the extended shutdown (February 1996 to May 1999). Three supplements to the original LER provided additional examples of historical TS violations for which various corrective actions were taken. The inspector reviewed the original LER and its supplements, the Unit 2 TS, and the associated corrective actions documented in the licensee's corrective action process.

A sample of the licensee's completed corrective actions showed that the issues were properly tracked and resolved, including those that required procedure changes, TS change requests and updates to the Unit 2 Technical Requirements Manual. The licensee identified the TS violations during their extensive reviews of the design and licensing basis that occurred during the extended shutdown. These reviews were conducted as a result of a number of NRC violations involving design and licensing basis discrepancies. Therefore, the TS violations discussed in the LER have already been dispositioned from an enforcement perspective and no additional violations of NRC Enforcement Policy were identified. Licensee Event Reports 50-336/97-22-00, -01, -02, & -03 are **closed**.

U2.II Maintenance

U2 M1 Conduct of Maintenance

M1.1 General Maintenance Observations

During routine plant inspection tours, the inspectors observed, on a random sampling basis, maintenance and surveillance activities to evaluate the propriety of the activities and the functionality of systems and components with respect to technical specifications and other requirements.

The inspectors reviewed maintenance work orders and interviewed licensee field personnel to verify the adequacy of work controls. The inspector observed a portion of activities performed under the following automated work orders (AWOs) or surveillance procedures:

- AWO M2-99-02877 — "B" Auxiliary Feedwater Pump Motor and Cable Testing
- AWO M2-99-10945 — Main Feedwater Regulating Bypass Valve Card Replacement
- Procedure SP 2606C-2 — Containment Spray System Alignment, Operability and Operational Readiness Tests
- Procedure SP 2604L-2 — Low Pressure Safety Injection System Alignment Check and Valve Operability Check

The inspector found that maintenance work was being performed in accordance with approved work orders present at the work site. A review of the work packages found that they were complete with respect to work authorizations, procedures, and inspection requirements. The replacement of a control circuit card for the main feedwater regulating bypass valve was well coordinated with operations and appropriate contingency measures were in place in the event an unexpected feedwater transient occurred during the card replacement. When the card initially failed, reactor operator performance was good in immediately recognizing, before any alarms were received, that the main feedwater water regulating bypass valve had closed. Due to prompt action by the reactor operator to restore main feedwater flow, a plant transient was averted. The plant equipment operator who conducted the two surveillances for verifying the alignment of containment spray and low pressure safety injection systems was found to be knowledgeable and was diligent in checking for proper operation of other equipment in the areas.

U2 M8 Miscellaneous Maintenance Issues

M8.1 <u>(Closed) Inspector Follow-up Item 50-336/96-08-21: Unit 2 Material Condition Program</u>

a. <u>Inspection Scope (71750, 90712, 92903)</u>

The inspector conducted in-office, in-field and document reviews of Inspector Follow-up Item 50-336/96-08-21, which related to a Material Condition Program (MCP) that was discontinued at Unit 2. The IFI was opened to review the resolution of Adverse Condition Report (ACR) 10172, which addressed the conversion from a site-wide MCP to unit specific programs.

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

The licensee initially established a site-wide MCP to: provide an organized method to identify material problems and deficiencies; establish a reporting process to initiate corrective actions; and implement a management oversight process to assure accountability and commitment of personnel and other resources. It was subsequently replaced by the condition report (CR) process and other unit specific procedures. The inspector reviewed: the MCP Manual, revision 0; ACR 10172; CR M2-97-0732; Millstone Unit 2 Material Condition Upgrade Program, dated March 1999; Procedure M2-UI-1.01, "Unit 2 Housekeeping"; and a sample of MCP punch list items that were outstanding at the time the MCP was discontinued. The inspector found that the CR process adequately tracked and resolved newly identified material condition deficiencies. The inspector also reviewed a sample of MCP punch list items that were outstanding when the MCP was discontinued and found the items were adequately resolved. The Unit 2 Material Condition Upgrade Program includes extensive painting efforts that are currently underway.

c. Conclusion

The Unit 2 Material Condition Upgrade Program and the site-wide CR process adequately replaced the site-wide Material Condition Program. Items on the MCP punch list that existed when the MCP program was discontinued were found to be adequately resolved. No violations of NRC requirements were identified. Inspector Follow-up Item 50-336/96-08-21 is **closed**.

U2.III Engineering

U2 E1 Conduct of Engineering

E1.1 Vital Switchgear Cooling Design Control Issues

a. Inspection Scope (37551/71707)

The inspector evaluated the licensee's contingency actions for loss of vital switchgear cooling for the lower 4160 volt switchgear room. The inspector also reviewed the calculations supporting the contingency actions and interviewed engineering and operations department personnel involved with the issue.

b. Observations and Findings

On September 1, 1999, the licensee identified a service water leak from a tube in the lower 4160 volt switchgear room cooler. At the time, the reactor was in Operational Mode 1, power operation, and the vital 4160 volt switchgear was required to be operable by Technical Specification 3.8.2. Operators isolated the cooler using Section 4.11 of procedure OP 2315D, "Vital Electrical Switchgear Room Cooling Systems," and began implementing the specified compensatory measures. The required compensatory measures included periodically measuring the room temperature and, at specified temperatures, turning off room lights and breaker cubicle heaters to reduce the rate of heat generation in the room.

On September 2, 1999, the inspector discussed the switchgear room cooler leak with the control room operators and reviewed procedure OP 2315D. The inspector had concerns whether the compensatory measures were sufficient to ensure continued operability of the 4160 volt switchgear. In addition to the heat generated inside the room by electrical equipment, the affected switchgear room is located in a warm area of the turbine building adjacent to where the main steam lines enter the turbine building. The inspector reviewed licensee calculation 92-FFP-932ES, Revision 2, "MP2 Lower Switchgear Room, 4.16 & 6.9 KV, Heat Gains and Maximum Room Temperature," and found that it failed to consider effect of a HELB. Instead, the calculation evaluated the peak steady-state temperature within the switchgear room under design normal operating conditions for a summer day with no room cooling. Under these conditions, the calculated room temperature with the room lights and breaker cubicle heaters off was 121.9°F, which was just below the switchgear room design temperature of 122°F.

Section 9.9.15, "Vital Switchgear Ventilation System," of the Unit 2 Final Safety Analysis Report (FSAR) states that the function of the vital switchgear ventilation system is to maintain a suitable environment for the safety-related electrical equipment during normal operation, loss-of-offsite power, and post-accident conditions. A high-energy line break (HELB) in the turbine building with off-site power remaining available could increase both the heat generation within the room by energizing additional equipment and the heat addition from the surrounding area affected by the HELB.

Based on the fact that the "A" train switchgear room cooler was out of service and the design calculation did not consider the affects of a HELB, the inspector discussed his operability concern with the licensee who issued Condition Report (CR) M2-99-2382 and developed an operability determination for the "A" train 4160 volt switchgear. The licensee determined that the switchgear was operable based on the conservative nature of the steady-state switchgear room temperature calculation and the short duration of the heat addition associated with a HELB in the turbine building. Based on the short duration of the HELB effects and the measured temperatures within the switchgear room, the inspector agreed that the switchgear room temperature was likely to remain below the 122°F temperature limit after a postulated high-energy line break. The vital switchgear room cooler was returned to operable status on September 5, 1999.

During further review of the Unit 2 FSAR, procedure OP 2315D, and the calculation the inspector noted the following additional discrepancies:

(1) Section 9.9.15 of the FSAR states that the normal cooling requirements exceed the cooling requirements for emergency operation, but the calculation shows that the heat gain following a loss-of-coolant accident with off-site power available exceeds the heat gain during normal operation by about 8 percent.

(2) The lower 4160 volt switchgear room contains both the "A" train 4160 volt vital bus and the "swing" vital 4160 volt bus, which often powers pump motors aligned to the "B" train, but the calculation does not include heat gain from the components powered from the "swing" vital bus.

(3) The heat gain from additional equipment operated to mitigate a loss-of-coolant accident may not bound the heat gain from turbine building high-energy line break effects, but neither the FSAR nor the calculation provide a basis for evaluating only the loss-of-coolant accident heat gains.

(4) Procedure OP 2315D recommends the use of portable electric blowers as a compensatory measure when the lower 4160 volt switchgear room cooler is not functioning, but the calculation does not consider the additional heat gain from the blowers.

The inspector discussed these issues with the responsible engineering supervisor. This discussion also included the potential future use of compensatory measures for loss of room cooling and the extent similar concerns are applicable to other rooms containing safety-related equipment. The engineering supervisor agreed that the issues were valid.

Subsequently, the licensee documented the concerns with the future application of compensatory measures for loss of room cooling in CR M2-99-2586 and developed their basis for an expectation of continued operability when those compensatory cooling measures would be employed.

Criterion III, "Design Control," of Appendix B to 10 CFR Part 50 requires that the design basis of safety-related structures, systems, and components be correctly translated into specifications and procedures. The inspector determined that design control measures were inadequate to assure that the 4160 volt switchgear room coolers were capable of maintaining a suitable environment for the vital switchgear under post-accident conditions. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. This violation is in the licensee's corrective action program as Condition Reports M2-99-2382 and M2-99-2586.

c. Conclusions

The NRC found that design control measures were inadequate to assure that the 4160 volt switchgear room coolers were capable of maintaining a suitable environment for the vital switchgear under post-accident conditions. This Severity Level IV violation of design control requirements is being treated as a **Non-Cited Violation (NCV 50-336/99-09-05)**. The licensee's determination that the "A" train 4160 volt switchgear remained operable when the switchgear room cooler was removed from service for corrective maintenance was found acceptable.

U2 E8 Miscellaneous Engineering Issues

E8.1 (Closed) LER 50-336/96-30-01 & -02: In-service Test Program Deficiencies

Licensee Event Report (LER) 50-336/96-30-01 & -02 are administratively **closed**. The licensee's corrective actions to address in-service test program deficiencies, which were described in these LER supplements, were inspected and closed in NRC Inspection Report 50-336/99-02 as part of Unit 2 Significant Items List Item No. 49.

E8.2 (Closed) LER 50-336/97-35-00: Shutdown Cooling System Isolation Valve Does Not Comply with Appendix R Requirements

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in Licensee Event Report (LER) 50-336/97-35-00. The reviews included inspection of the licensee's corrective actions and supporting references, and limited discussions with licensee engineering and regulatory affairs personnel.

b. Findings and Observations

On November 11, 1997, a licensee reanalysis of 10 CFR 50 Appendix R requirements identified that the interface between the reactor coolant system (RCS) and the shutdown cooling suction line was not in conformance with NRC Generic Letter 86-10, "Implementation of Fire Protection Requirements. They found that a fire-induced three phase hot short on the power cable for the shutdown cooling suction header isolation valve could cause this valve to spuriously open and result in the inability to maintain RCS boundary isolation. The licensee responded to this discovery by rerouting the power supply for this valve.

The licensee's corrective actions were found to be adequate. Failing to adequately implement design controls to ensure that safety-related valves are not subject to fire induced hot shorts is a violation of 10 CFR 50 Appendix R, Fire Protection. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program. This issue was entered as Condition Report M2-97-2604.

c. Conclusions

The licensee identified in 1997 that a valve that isolates the shutdown cooling system from the RCS was vulnerable to a fire-induced hot short. The licensee's corrective actions were found acceptable. This violation of 10 CFR 50, Appendix R, Fire Protection, is being treated as a **Non-Cited Violation (NCV 50-336/99-09-06)**. LER 50-336/97-35-00 is **closed**.

E8.3 (Closed) LER 50-336/98-02-00 & -01: Emergency Core Cooling System Single Failure Vulnerability

Licensee Event Report (LER) 50-336/98-02-00 & -01 are administratively **closed**. The licensee's corrective actions to address the emergency core cooling water system single failure vulnerability discussed in the LER were inspected as part of Unit 2 Significant Items List Item No. 53, which was closed in NRC Inspection Report 50-336/99-04.

E8.4 (Closed) LER 50-336/98-03-00: Inadequate Evaluation of the Interaction Between the Reactor Internals and the Reactor Vessel

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in Licensee Event Report (LER) 50-336/98-03-00. The reviews included inspection of the licensee's corrective actions and supporting references, and limited discussions with licensee operations and regulatory affairs personnel.

b. Findings and Observations

On November 13, 1998, while the unit was shut down, the licensee identified that their design analysis of the interaction between the reactor vessel internals and the reactor vessel did not properly address dynamic loading associated with a loss of coolant accident or a design bases earthquake. The licensee reperformed the analysis and submitted it to the NRC for approval prior to startup from the extended shutdown.

The inspector reviewed the LER, the Unit 2 Final Safety Analysis Report, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. Failing to adequately implement design controls to ensure that the design basis of the reactor vessel internals was correctly translated into specifications, drawings and procedures is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program. This issue was entered as Condition Report M2-98-0593.

c. Conclusions

The licensee identified in 1998 that their design analysis of the interaction between the reactor vessel internals and the reactor vessel did not properly address dynamic loading associated with a loss of coolant accident or a design bases earthquake. This violation of 10 CFR 50 Appendix B, Criterion III, Design Control, is being treated as a **Non-Cited Violation. (NCV 50-336/99-09-07)** The licensee's corrective actions were found acceptable. Licensee Event Report 50-336/98-03-00 is **closed**.

Report Details

Summary of Unit 3 Status

Unit 3 began the inspection period operating at 100 percent power. On September 4, 1999, operators reduced power to approximately 89 percent to perform a thermal backwash on the main condenser. Power was subsequently returned to 100 percent on September 5.

On September 16, operators reduced power to approximately 80 percent power in response to a broken condensate demineralizer drain valve and subsequent effects on the feedwater system. (See Section U3.O1.1.) Following plant stabilization and isolation of the leak, operators restored the plant to 100 percent power later that evening.

Also on September 16, a tropical storm, later upgraded to a hurricane, was predicted to affect the site from late evening into early morning. (See Section U3.01.2.) The site was minimally affected by the storm, which made landfall west of the site. However, at approximately midnight, operators began reducing power due to degrading condenser vacuum conditions. Power was stabilized at approximately 80 percent. After all condenser bays were backwashed, operators restored reactor power to 100 percent on September 17, where it remained through the end of the report period, October 4th.

U3.I Operations

U3 O1 Conduct of Operations

O1.1 Failed Condensate Demineralizer Drain Valve

a. Inspection Scope (71707)

At the September 16, 1999, morning managers' meeting, the inspector was told that operators were in the middle of a plant transient. The inspector responded to the control room and observed operator actions to stabilize the plant and discussed the transient with control room personnel and management.

b. Observations and Findings

Operators responded well to a demineralizer trouble alarm and field reports of leakage in the turbine building. The demineralizers were bypassed and the leak was isolated. Upon restoring the demineralizers to service, normal automatic level control valves for the fourth point heater did not respond quickly enough to control the level in the heaters and two of the three fourth point heater drain pumps tripped on low level. Engineering review subsequently determined that the system operated as designed.

Operators responded well to the pump trips and reduced power to 80 percent to stabilize the plant and control the resultant feedwater transient. Proper command and control was observed in the control room. Briefs were conducted at appropriate intervals to ensure all operators understood plant conditions.

Contractor personnel were painting portions of the condensate demineralizer area. As one of the painters was descending the area, he accidentally damaged the "F" condensate demineralizer (CND) drain valve (3CND-AOV31F) solenoid. The valved opened, draining water from the condensate system into the turbine building. The inspector questioned why the valve would have been designed to fail open and noted good follow-up and timely response by the licensee. Engineering personnel determined that although the valve was designed to fail closed, the physical damage to the solenoid prevented this operation and instead the valve failed in an intermediate position, thereby allowing water to drain from the in-service demineralizer.

The inspector examined the area in the turbine building where the CND valve failure occurred and observed proper coordination to control access to the building and clean up the approximately twenty to thirty thousand gallons of uncontaminated water and spent resin that spilled from the condensate system onto the turbine building floor. The inspector interviewed available licensee personnel to determine the extent of health physics and environmental sampling prior to allowing work in the area. Plant personnel worked to limit releases not approved by their environmental permit. The inspector also examined the damaged CND valve solenoid, noting how the failure mechanism prevented full valve closure. Equipment tagging was consistent with the system isolation status and cleanup activities.

c. Conclusions

The licensee responded well to a leak in the Unit 3 condensate demineralizer system. Operators isolated the leak and stabilized power at 80 percent to control the transient which followed. Effective operations command and control and appropriate licensee cleanup efforts were observed during and after the transient. The licensee demonstrated thorough follow-up to determine why a damaged demineralizer drain valve did not fail closed, as expected.

O1.2 Hurricane Floyd Response

a. Inspection Scope (71707)

The inspector observed the licensee's actions taken in response to Hurricane Floyd, predicted to reach Connecticut on September 16 and 17, 1999.

b. Observations and Findings

In response to the harsh weather predictions for the northeastern United States, the NRC Region I office activated the regional emergency response center the afternoon of September 16. NRC inspectors provided onsite coverage of licensee actions from 7 am through midnight, after the storm made landfall in the western portion of Connecticut.

The licensee took appropriate actions in preparation for the storm; including tying down or moving outdoor equipment, verifying any in progress work on screen house equipment was terminated and the equipment returned to service, and staffing extra onsite and on call personnel to cope with potential weather-related equipment degradation. In addition, four of the six condenser bays were backwashed early on September 16th. (The final two bays were not completed due to rising bay temperature effects on Unit 2.) The inspector noted that the "A" quench spray system (QSS) pump was taken out of service for planned surveillances early the morning of September 17th and returned to service later that afternoon, as planned. Since the storm did not reach Connecticut until later that evening, this period of pump inoperability had no adverse impact on plant safety. The inspector examined the inoperable status of the QSS pump in the engineered safety features (ESF) building, discussed its return to service with the operations shift manager, and witnessed a portion of the shift briefing for the operational test.

The inspector conducted a walkdown of all eleven Unit 3 watertight doors, specified in AOP 3569 to be closed during hurricane preparations, and confirmed all doors were properly closed and latched. The inspector also examined conditions at the service water intake structure and observed mechanics on station to respond to potential circulating water fouling concerns. The licensee detailed an appropriate number of personnel for storm coverage during all shifts at Unit 3.

Unit 3 operators appropriately entered and followed common operating procedure C OP 200.6, Storms and Other Hazardous Phenomena, and AOP 3569, Severe Weather Conditions, as conditions warranted. As discussed in Section U1.O1.3, the classification of the storm as a hurricane or tropical storm differed among the three units. No adverse impact resulted from this difference and the licensee issued CR M3-99-3221 to investigate the discrepancy.

Following the plant transient discussed in the previous section, operators restored reactor power to 100 percent at approximately 11 pm on September 16th. To further prepare for the storm's expected after effects on the intake, operators decided to backwash the last two bays. However, at approximately midnight, they noted that condenser vacuum was degraded such that their procedures would not have allowed them to continue the backwash once started. This is because while one bay is out of service during the backwash, condenser vacuum is expected to degrade. As a result, the operators reduced power, eventually to 80 percent, to improve condenser vacuum to an acceptable level.

The inspector observed appropriate control of the downpower and discussion of where condenser vacuum needed to be for a safe backwash. The operators performed the backwash on the final two bays early that morning in accordance with approved procedures. Before power was restored to 100 percent later that day, the other four bays were again backwashed.

Throughout the evening, the inspector noted proper staffing at the intake and good communication between the operators and the system engineer, who was onsite to support operations through the storm. The storm was downgraded to a tropical storm before reaching Millstone and brought minimal rain and non-severe wind to the Millstone site. The licensee determined that the degraded condenser vacuum was caused by a combination of condenser fouling and higher water temperatures due to the storm.

c. <u>Conclusions</u>

Unit 3 personnel appropriately prepared for and responded to Hurricane Floyd. Operators deliberately reduced reactor power and stabilized the plant before backwashing two condenser bays during the storm. Before restoring power to 100 percent the following day, operators properly backwashed the remaining four bays.

U3 O8 Miscellaneous Operations Issues (92700)

O8.1 <u>(Closed) LER 50-423/98-46: Manual Reactor Trip Due to a Difference in Indicated and Demand Rod Positions</u>

This LER documented that on December 26, 1998, with the Unit in Mode 3, a manual reactor trip was initiated when operators identified that there was a disagreement between the digital rod position indicator (DRPI) and the rod control system demand position. Operators initiated the action statements associated with Technical Specification (TS) 3.1.3.3. The licensee determined that the cause of the disagreement between the two indications was the failure of a main board rod bank selector switch.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited industry material history data review was performed. Based on the material review, the failure appeared to be wear related with no salient common mode failure or generic implications. The operators properly responded to plant conditions and the licensee appropriately documented, tracked and corrected the condition in accordance with the licensee's corrective action process. No violation of NRC requirements was identified. This LER is **closed.**

O8.2 <u>(Closed) LER 50-423/98-45: Reactor Trip Due to a Main Steam Isolation Valve Closure during Partial Stroke Testing Due to Solenoid Valve Failure</u>

This LER documented that on December 11, 1998, with the Unit in Mode 1, an automatic reactor trip resulted from a low steam generator level reactor protection system function. A steam generator level transient was caused by the inadvertent closing of a main steam isolation valve (MSIV) during a surveillance test. The MSIV fully closed as a result of a failure of an associated solenoid operator.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited industry material history data review was performed. Based on the material review, the failure appeared to be related to corrosion on a solenoid not related to the safety function of the valve, with limited generic implications.

The operators properly responded to plant conditions and the licensee appropriately documented, tracked and corrected the condition in accordance with the licensee's corrective action process. No violation of NRC requirements was identified. This LER is **closed.**

O8.3 (Closed) LER 50-423/98-44: Reactor Trip Due to High Differential Pressure Between "A" and "B" Condensers

This LER documented that on November 11, 1998, with the Unit in Mode 1, a manual reactor trip was initiated when operators noted high differential pressure between the "A" and "B" circulating water system (CWS) Condensers. The differential pressure was determined to have been caused by high levels of debris impinging on the traveling screens due to severe weather.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited Millstone station operating data history review was performed. The licensee has experienced an ongoing problem with the impact of debris on traveling screen operation. The operators properly responded to plant conditions and the licensee appropriately documented, tracked and corrected the condition in accordance with the licensee's corrective action process. No violation of NRC requirements was identified. This LER is **closed.**

O8.4 (Closed) LER 50-423/98-43: Manual Reactor Trip Due to High Conductivity in the Condensate System

This LER documented that on October 28, 1998, with the Unit in Mode 1, a manual reactor trip was initiated when operators noted a salt water intrusion into the "C" condenser waterbox. The salt water intrusion resulted from a leak into the condenser waterbox and resulted in high conductivity water samples. A manual reactor trip was initiated by control room operators after entering abnormal operating procedure 3558, Condenser Tube Leak.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited Millstone station operating data history review was performed. The licensee has experienced an ongoing problem with condenser water quality, including chlorine and oxygen. The operators properly responded to plant conditions and the licensee appropriately documented, tracked and corrected the condition in accordance with their corrective action process. No violation of NRC requirements was identified. This LER is **closed.**

O8.5 <u>(Closed) LER 50-423/98-40: Technical Specification 3.0.3 Entry Due to Both Quench Spray System Pumps Placed in "Pull-to-Lock"</u>

This LER documented that on October 17, 1998, with the Unit in Mode 1, both trains of the Quench Spray System (QSS) were made inoperable by placing the "A" and "B" QSS pumps in the "pull to lock" position. The placement of the pumps in "pull to lock" resulted in an annunciated condition in the control room, which was responded to and resolved by the operator that had previously manipulated the pumps. No plant transient or perturbation resulted from the alignment or annunciator.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited Unit 3, human performance data history review was performed. No specific negative human performance trends were noted. The operators properly responded to plant conditions, limiting QSS inoperability to less than one minute. The licensee appropriately documented, tracked and corrected the condition in accordance with the corrective action process. The placement of both QSS pumps in the "pull to lock" position is a violation of procedural adherence as required by TS 6.8.1. However, the inspector concluded that this event constituted a minor violation based on the short duration that both QSS pumps were in the "pull to lock" position. This LER is **closed.**

O8.6 <u>(Closed) LER 50-423/98-39: Technical Specification 3.0.3 Entry Due to Vital inverter Failure Resulting in the "A" and "C" Recirculation Spray System (RSS) Pumps Being Inoperable While the "B" RSS Pump Was Out of Service for Maintenance</u>

This LER documented that on October 1, 1998, with the Unit in Mode 1, both trains of the RSS were made inoperable as a result of an equipment failure combined with a pre-existing plant equipment alignment. A vital inverter failure resulted in the "A" and "C" RSS pumps being inoperable at the same time that the "B" RSS pump was out of service for maintenance. Although the failure did not result in an annunciated condition in the control room, it was identified and resolved by plant operators within two hours. The licensee replaced the blown fuse associated Silicon Control Rectifier (SCR) circuit boards and other inverter components. The removed components were sent to suppliers for analysis.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited Unit 3, inverter maintenance history review was performed and found the licensee's corrective actions to be adequate. The operators properly responded to plant conditions, limiting RSS inoperability to less than two hours from the time of discovery. The licensee appropriately documented, tracked and corrected the condition in accordance with the licensee's corrective action process. The October 1, 1998, inverter failure resulted from a blown fuse which was preceded by two previous similar failures (August 23 and September 23, 1998). No violation of NRC requirements was identified. This LER is **closed.**

O8.7 (Closed) LER 50-423/98-37: Technical Specification 3.0.3 Entry Due to Both Service Water Trains Declared Inoperable Following Failure of Check Valves Associated with the Injection of Sodium Hypochlorite

This LER documented that on September 10, 1998, with the Unit in Mode 1, both trains of service water (SW) were declared inoperable when check valves associated with the injection of sodium hypochlorite failed to reseat following a surveillance test. The condition was responded to and resolved by the unit operators within three hours.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. In addition, a limited review of the maintenance activities that resulted in the plant condition was performed. The licensee's corrective actions were found to be adequate. The inspector determined that the check valve failures resulted from internal valve corrosion, caused by the placement of hypochlorite susceptible check valves in portions of the SW system that were subjected to relatively high concentrations of hypochlorite. Failing to adequately implement design controls to ensure that the SW design basis was correctly translated into specifications, drawings and procedures is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy, which permit closure of most Severity Level IV violations based on the issue being entered into the corrective action program **(NCV 50-423/99-09-08)**. This LER is **closed.**

O8.8 (Closed) LER 50-423/99-01: Technical Specification 3.0.3 Entry Due to a High Energy Line Break (HELB) Boundary Door Latch Failure

This LER documented that on January 16, 1998, with the Unit in Mode 1, the east main control room door became inoperable as the result of the door latch sticking in the withdrawn position. The east main control room door is a HELB boundary. The cause of the failure was determined by the licensee to be a latch failure. The condition was identified and initially corrected within two hours.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited review of maintenance history data was also performed. The inspector noted that the January 16, 1999 failure was followed by a similar failure but that a general trend of door failures did not exist. No violation of NRC requirements was identified. This LER is **closed.**

O8.9 (Closed) Inspection Follow-up Item (IFI) 50-423/97-202-03: Loss of Spent Fuel Pool Cooling

IFI 97-202-03 was generated to address licensee actions in response to a loss of spent fuel pool cooling that occurred on June 25-26, 1997, and is detailed in NRC Inspection Report 50-423/97-202. The inspector reviewed (1) the licensee's overall assessment of the event, (2) the licensee's assessment of generic implications of the event relative to other licensee programs, and (3) the adequacy and implementation of corrective actions. The inspector also reviewed the licensee's actions to address the programmatic configuration control and generic issues that were identified following the loss of spent fuel pool cooling event. The inspector determined that while operations department configuration control events continued to occur following the June 1997 event, the licensee self-identified these events, had initiated appropriate evaluations, and implemented numerous corrective actions that, in general, have been successful in preventing recurrence of the specific root cause of each event. The inspector concluded that the licensee's actions to address both the broad configuration and control issues for the loss of spent fuel pool cooling event, including subsequent events, and their implementation of appropriate corrective actions have been adequate, therefore, **IFI 50-423/97-202-03** is **closed.**

O8.10 (Closed) Violation (VIO) 50-423/98-208-04: Failure to Implement Plant Heatup Procedure

NRC Inspection Report 50-423/98-208, detailed the licensee's failure to implement a plant heatup procedure, in that they failed to direct test personnel to close and lock applicable valves during an operational mode transition in June 1998. The inspector reviewed the licensee's activities following the NRC's identification of the violation, and concluded that the licensee appropriately entered the violation in their corrective action program and has implemented appropriate corrective actions, therefore, **VIO 50-423/98-208-04** is **closed.**

O8.11 (Closed) LER 50-423/98-01: Safety Injection (SI) Accumulator Outlet Isolation Valve Motor Pinion Gear Key Failure

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of licensee corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On December 12, 1997, with Unit 3 in Mode 5, the licensee identified that the motor pinion gear keys in three out of four SI accumulator isolation valves were sheared. The licensee reported the condition to the NRC on January 6, 1998. The condition was responded to and resolved by performing maintenance on the pinion gears, replacing the keys with keys having different material properties and testing the valves consistent with the requirements of Generic Letter 89-10, Motor Operated Valves.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and surveillance procedures, the design change documentation associated with the key replacement, and the associated corrective actions and work requests documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. The inspector determined that the root cause of the pinion key failure was a failure to adequately translate the valve design criteria into appropriate design specifications for the pinion keys. Failing to ensure that the SI accumulator valve design basis was correctly translated into specifications, drawings and procedures is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program (NCV 50-423/99-09-09).

c. Conclusions

The licensee reported on January 6, 1998, that the motor pinion gear keys in three out of four SI accumulator isolation valves were sheared. The condition was adequately responded to and resolved by the licensee. The root cause was a failure to adequately translate the valve design criteria into appropriate design specifications for the pinion keys. This failure is a violation and is being treated as non-cited (**NCV 50-423/99-09-09**). LER 50-423/98-01 is **closed**.

O8.12 (Closed) LER 50-423/98-04: Emergency Diesel Generator (EDG) Service Water (SW) Fouling

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of licensee corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On September 9, 1997, with Unit 3 in Mode 5, the licensee identified that the "A" and "B" EDG intercoolers would not have met their design criteria for thermal performance. The intercoolers would not have met their design criteria for thermal performance because of fouling that resulted from historical fuel oil leaks into the EDG fresh water cooling water system from the fuel oil injectors. The root cause of the leaks was improper fuel injector maintenance and the inadequate implementation of EDG testing. The condition was responded to and resolved by performing an adequate engineering analysis, conducting preventive maintenance on the EDG fuel injectors, performing testing consistent with that described in NRC Generic Letter 89-13, heat exchanger testing, and by the routine successful performance of TS required surveillance testing.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and surveillance procedures, EDG design change documentation, and the associated corrective actions and work requests documented in the licensee's corrective action process. In addition, a limited review of Generic Letter responses from 1988 and 1999 was performed. This review was limited to inspecting summary responses and the documentation of assigned corrective actions. The inspector determined that the adequacy of Generic Letter responses was included in the licensee's restart readiness process and had been the subject of considerable licensee effort. The licensee's corrective actions were found to be adequate. The inspector determined that the root cause of the failure to ensure that EDG performance parameters were maintained consistent with the design described in the FSAR was inadequate thermal performance testing procedures. Failing to adequately implement design controls to ensure that the EDG design basis was correctly translated into specifications, drawings and procedures is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program (NCV 50-423/99-09-10).

c. Conclusions

The licensee reported on September 9, 1997, that the Unit 3 Emergency Diesel Generator (EDG) coolers would not have met their design criteria for thermal performance due to fouling that resulted from historical fuel oil leaks into the EDG fresh water cooling water system from the fuel oil injectors. The fouling was not identified because of inadequate thermal performance testing. The condition was adequately responded to and resolved by the licensee. The failure to adequately implement design controls to ensure that the EDG design basis was correctly translated into specifications, drawings and procedures is a violation and is being treated as non-cited **(NCV 50-423/99-09-10)**. LER 50-423/98-01 is **closed**.

O8.13 (Closed) LER 50-423/98-09: Containment Atmosphere Monitoring System Radiation Monitor Setpoints

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of licensee corrective actions and supporting references and discussions with licensee personnel.

b. Findings and Observations

On February 6, 1998, with Unit 3 in Mode 5, the licensee identified that the containment atmosphere monitoring (CAM) alarm and alert setpoints (gaseous and particulate) were set above those indicated in the Millstone Unit 3 Final Safety Analysis Report (FSAR). The condition was responded to and resolved by resetting the setpoints, documenting the condition in a CR, and conducting other generic setpoint reviews..

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. In addition, a limited review of the setpoint calibration activities that resulted in the plant condition was performed. The licensee's corrective actions were found to be adequate. The inspector determined that the root cause of the failure to adequately set the CAM setpoints at those levels required by the plant design was a failure to establish a formal control over radiation monitor setpoints in general. A sample of additional radiation monitor setpoints was taken and found to be adequate. Failing to adequately implement design controls to ensure that the CAM design basis was correctly translated into specifications, drawings and procedures is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Appendix C of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program (NCV 50-423/99-09-11).

c. Conclusions

On February 6, 1998, the licensee reported that the Unit 3 Containment Air Monitor alarm and alert setpoints were set above those indicated in the Millstone Unit 3 Final Safety Analysis Report (FSAR). The condition was adequately responded to and resolved by the licensee. Failing to adequately implement design controls to ensure that the CAM design basis was correctly translated into specifications, drawings and procedures is a violation and is being treated as non-cited **(NCV 50-423/99-09-11)**. LER 50-423/98-09 is **closed**.

O8.14 (Closed) LER 50-423/98-38: Manual Reactor Trip Due to a High Condensate Pump Discharge Conductivity

This LER documented that on September 15, 1998, with the Unit in Mode 1, a manual reactor trip was initiated when operators identified that condensate pump discharge conductivity exceeded the limits allowed by an abnormal operating procedure. The licensee determined that the cause of the high conductivity condition was an inadequate secondary system (steam generator blowdown) operating procedure.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and abnormal procedures, and the associated corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were adequate. In addition, a limited industry material history data review was performed. The operators properly responded to plant conditions and the licensee appropriately documented, tracked and corrected the condition in accordance with their corrective action process. No violation of NRC requirements was identified. This LER is **closed**.

U3.II Maintenance

U3 M1 Conduct of Maintenance

M1.1 Surveillance Observations

a. Inspection Scope (61726)

The inspector observed portions of selected surveillance activities, including operational tests, in-service testing (IST), and equipment tests performed for cause, to establish component and system operability. Systems were selected for surveillance observation based upon their risk significance, the opportunity for test witness, and the timing and priority for returning equipment to service to comply with the allowed outage times delineated in the Unit 3 technical specifications.

b. Observations and Findings

The inspector witnessed portions of the following surveillance tests, discussed the conduct of work and ongoing controls with operations and engineering personnel, and reviewed selected test results:

- SP 3622.3 Turbine Driven Auxiliary Feedwater (TDAFW) Pump 3FWA*P2A Operational Readiness Test
- SP 3635B.2 Containment Recirculation Cubicle Air Driven Sump Pump Test
- SP 3646A.2 Emergency Diesel Generator B Operability Test (performed in conjunction with special procedure, SPROC 95-3-26, 3EGS*E1B/2B Thermal Performance Test)
- SP 3646A.8 Slave Relay Testing - Train A (containment spray actuation, quench spray system pump 3QSS*P3A)
- EN 31121 IST Pump Operational Readiness Evaluation (recirculation spray system pump 3RSS*P1B)

During the conduct of the IST portion of the operational test for pump 3FWA*P2A on September 13, 1999, two separate test measurements of recirculation flow resulted in flow rates less than the minimum established as part of the surveillance criteria of SP 3622.3. A condition report, CR M3-99-3178, was issued to document the apparent TDAFW pump low recirculation flow rates. On September 14, 1999, an operability determination (OD MP3-033-99) was written, and later approved by the plant operations review committee (PORC) on September 15, 1999, establishing the operability of pump 3FWA*P2A, while documenting that this TDAFW pump was not fully qualified. The inspector reviewed the OD, noting that technical specification operability surveillance criteria had been satisfied by the developed pump discharge head pressure calculated for the obtained recirculation flow. The pump was determined to be capable of delivering forward (versus recirculation) flow to the steam generators and therefore operable with respect to the TDAFW system safety function. An action plan was developed by the licensee for resolving the discrepant recirculation flow data. This is discussed further in Section U3.E1.1 of this inspection report.

On September 22, 1999, during the conduct of SP 3635B.2, the "A" RSS cubicle air driven sump pump (3DAS*P15A) failed its test by not demonstrating the ability to pump water from the "A" RSS cubicle sump. Based upon this failure, operators declared the "A" RSS train inoperable because of the failure of the 3DAS*P15A support equipment, appropriately entering technical specification (TS) limiting conditions for operation for both the RSS and the emergency core cooling (ECCS) systems. The inspector witnessed the "B" train surveillance test of pump 3DAS*P15B, confirming satisfactory flow results. This ensured that one operable train was available, and that continued plant operation was being properly controlled by the 72-hour allowed outage time of TS 3.5.2 and 3.6.2.2. Subsequently, an action plan was developed for the failure analysis, repair, and restoration of 3DAS*P15A to an operable status. This is discussed further in Section U3.E2.1 of this inspection report.

c. Conclusions

Observed Unit 3 surveillance activities were performed in a controlled manner, in accordance with approved procedures. Where testing problems arose or failures occurred, the licensee developed action plans to evaluate and correct the identified concerns. To address any generic questions of component or system operability, the licensee prudently scheduled additional testing on an expedited basis, as permitted by the overall plant conditions.

U3 M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Field Modifications, Maintenance, and Material Conditions

a. Inspection Scope (62707, 37551)

During periodic inspection-tours of Unit 3, the inspector observed maintenance and modification activities in progress and checked the material condition of various pieces of equipment, subjected to the field environment. As appropriate, the inspector reviewed preventive maintenance documents, Maintenance Rule action plans, design drawings, and licensee procedures delineating the criteria and controls for routine maintenance and design change implementation. The applicable maintenance and test records were evaluated with respect to the standards documented in the Unit 3 FSAR and other procedurally referenced guidelines.

b. Observations and Findings

The inspector conducted inspections of equipment in the engineered safety features (ESF) building, including the turbine driven auxiliary feedwater (TDAFW) pump room, the recirculation spray system (RSS) cubicles, and separate supplementary leak collection and release system (SLCRS) areas, as well as various auxiliary building levels. While general housekeeping observations were noted and provided to the cognizant licensee maintenance personnel, the following types of equipment and specific components were selected for a more detailed review:

(1) Various containment penetration assemblies (e..g, Z24), and the piping, pipe supports, and containment isolation valves (CIVs) associated with the selected penetrations;

The inspector noted some rigid hardware connections crossing the seismic "shake" space between the containment structure and the auxiliary building, and confirmed with engineering personnel that these installations had been properly reconciled for stresses associated with building design and assumed differential movement. Some safety-related CIVs were found to be labeled with non-safety identification tags. These discrepancies were subsequently handled in accordance with the provisions of station procedure OA 9, "System and Component Labeling". The inspector also identified a discrepancy between the power supply design and the FSAR containment penetration listing (Table 6.2-65) for three TDAFW pump steam supply drain trap CIVs. The inspector verified the proper containment isolation valve design considerations in accordance with 10 CFR 50 general design criterion (GDC) 57, checked that other safety-related (non-CIV) valves provided redundant isolation capability against a main steam line break accident, and noted that the licensee initiated a condition report (CR M3-99-3302) to document the FSAR error. Such discrepancies and findings constitute issues of minor safety significance for which the licensee initiated appropriate corrective action.

(2) Safety-related instrument level switches (3DAS*LS66A & B) on the floor of the redundant RSS cubicles and non-safety temperature switches (3HVQ-TS45C & D) in their respective train related SLCRS areas;

The inspector noted setpoint discrepancies between the two HVQ-TS instruments for the ESF building vent controls. Subsequent discussion with a Unit 3 instrumentation and control (I&C) supervisor led to re-calibration of both temperature switches.

With respect to the reactor plant aerated drain system (DAS) level switches, the inspector reviewed a technical evaluation (M3-EV-970281, Rev. 2) documenting the safety-related function of these instruments relative to the emergency core cooling system (ECCS) passive failure design criteria. Upon checking the preventive maintenance requirements for both level switches, along with the post-maintenance testing criteria for 3DAS*LS66B, which had been replaced in 1998, the inspector noted that only loop calibrations are specified on a periodicity of 48 months. Since such loop calibrations involve manually lifting the limit switch and not floating the device, the inspector questioned both the functional testing accuracy and the test interval for these DAS level switches.

The inspector reviewed a Unit 3 Maintenance Rule (a)(1) system action plan for the DAS aerated drains system (3335B-1), based upon functional failures of 3DAS*LS66B and other DAS components. When the replacement of the switch was completed in February 1998, consideration of increasing the test frequency of these level switches was documented, but determined to not be necessary. The inspector also reviewed the production maintenance management system (PMMS) history file for the 3DAS*LS66B replacement work, the referenced I&C retest procedure (IC 3408A12), and the Unit 3 Post-Maintenance Testing procedure (U3 WPC 3). The guidance documented in a referenced Institute of Electrical and Electronics Engineers standard (IEEE 338) indicates that equipment performance history, as well as the need for the test to check the monitored variable (in this case, water level) are important considerations in the development of both the specific test procedure and the appropriate test interval.

The inspector further discussed these criteria and the questions of testing accuracy and periodicity with a responsible I&C supervisor. The licensee documented the subject of this NRC inquiry in CR M3-99-3112, which the inspector believes will adequately track the need for any future corrective actions, relative to the 3DAS*LS66A & B instruments. With regard to the identified setpoint discrepancies on the HVQ temperature switches, the licensee initiated immediate and effective corrective action for an item of minor safety significance.

(3) Construction of a high energy line break (HELB) vestibule in the service building outside control room door C-49-1;

As documented in Inspection Report 50-423/99-08, construction of the HELB vestibule outside the east control room door commenced during the last inspection period, in accordance with the design details of minor modification (MMOD M3-99022). During this inspection period, the inspector examined ongoing construction activities, evaluating completed work with respect to the design drawings and discussing the work plans and construction standards with the craftsmen in the area. The inspector noted that the job had been generally designated as non-safety-related, but with certain aspects of the job affecting safety-related structures and components (e.g., drilling concrete anchors into the seismic category I control building; relocating a fire-related, safe-shutdown emergency lighting unit electrical box). For these activities the minor modification details specified some quality assurance (QA) program applicability. Based upon the HELB basis for this vestibule construction and an analysis of risk implications associated with the historical control room door C-49-1 usage, the inspector concurred with the overall quality categorization of the MMOD controls.

The inspector discussed the need for QA inspection coverage with a Nuclear Oversight (NOS) manager, learning that while NOS inspectors were engaged in coverage of some of the construction work, such inspection was aimed at ensuring general quality for the special processes (e.g., concrete anchor installation) in accordance with approved procedures. In addition to the NRC expansion anchor comments on the in-process work, the inspector raised questions regarding structural bolting criteria and the use of slotted joints to provide differential movement between the control and service buildings. Also, the inspector noted that the relocated emergency lighting box (SB 30) had been re-attached to the wall with only two of the four concrete anchor bolts holding it in place. Since the operability of this lighting is controlled by a Technical Requirements Manual (TRM 7.4.I) limiting condition for operation (LCO), the inspector apprised the NOS manager of the concern for the structural configuration of this lighting box and notified the Unit 3 operations manager of the TRM/LCO implications. The need for the application of fire protection QA controls was delineated in the MMOD details.

Subsequently, NOS initiated CR M3-99-3417 to determine whether SB 30 was adequately, temporarily supported for continued service. Shortly thereafter, an operations shift manager's examination of the questioned SB 30 condition identified that the emergency lighting was unplugged and not functional. CR M3-99-3426 was then issued by the operations department to document both the inoperability of the emergency lighting and the fact that the TRM 7.4.I LCO Action 3 should have been entered on September 30, 1999, consistent with when the lighting box was first moved. The operations department then initiated action to verify battery power to the subject emergency lighting area on a daily basis, as required by the TRM action statement.

The inspector determined that the licensee had failed to assure that the design controls, specifying QA requirements for the relocation of emergency lighting box SB 30 and its impact upon the TRM safe-shutdown requirements, were adequately implemented. This finding is considered a violation of 10 CFR 50, Appendix B. However, based upon the licensee's corrective measures, promptly initiated and documented in the two CRs noted above, and consistent with Appendix C of the NRC Enforcement Policy, this Severity Level IV violation is being treated as a Non-Cited Violation, which permits closure of most Severity Level IV violations based on the issue being entered into the corrective action program (NCV 50-423/99-09-12).

Additionally, during this period, the inspector evaluated the licensee's maintenance action plan for the repair to service water pump 3SWP*P2B, a booster pump providing cooling water to one train of the control building chilled water (HVK) system. During planned HVK system preventive maintenance, delamination of the epoxy coating on the pump impeller backing plate was discovered. The problem was first identified and documented in CR M3-99-3083 on August 31, 1999. Subsequently, the action plan was issued with various options for repair. Good coordination amongst the operations, maintenance, and engineering departments was observed, along with the necessary liaison with the procurement group to obtain spare parts. Because of the 30-day allowed outage time requirements for the HVK system, as specified in the Unit 3 TRM, maintenance personnel worked through the weekend to effect coating repairs. The pump, 3SWP*P2B, was made available for service on September 7, 1999, and the HVK system was declared operable on September 8, 1999.

c. Conclusions

Inspection-tours of Unit 3, including observation of ongoing maintenance and modification activities, identified some issues that required follow-up, for which the licensee appropriately issued condition reports. While most of these items were minor, one finding involving the ineffective implementation of safety-related design control measures for the relocation of an emergency lighting box resulted in the identification of a Non-Cited Violation **(NCV 50-423/99-09-12)**.

U3 M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Inspection Follow-up Item (IFI) 50-423/97-202-07: Letdown Heat Exchanger ASME Code Compliance

IFI 97-202-07 was generated following the replacement of the letdown heat exchanger due to flange leakage problems, as detailed in NRC Inspection Report 50-423/97-202. The inspector verified that the licensee had appropriately performed a heat exchanger performance test with satisfactory results. In addition, the inspector reviewed the required ASME code documentation for heat exchanger testing provided by the licensee, and verified that the test was performed and documented satisfactorily. As a result, **IFI 50-423/97-202-07** is **closed**.

M8.2 (Closed) URI 50-423/97-203-06: Rosemount Transmitters

During Inspection 50-423/97-203 the NRC identified incomplete broad corrective actions with respect to incomplete identification and correction of inadequate plugging of spare ports in Unit 3 instruments. The licensee issued a CR to document the incomplete review and broadened the scope of the corrective actions. Unresolved Item 97-203-06 was opened pending NRC review of the subsequent corrective actions.

The NRC documented the acceptability of the licensee's completed broader instrument review and corrective actions in IR 50-423/97-208. No violations of NRC requirements were identified. Therefore, **URI 50-423/97-203-06 is closed.**

M8.3 (Closed) IFI 50-423/98-208-05: Material, Equipment, and Parts Lists (MEPL) Review

a. Inspection Scope (92902)

The Unit 3 MEPL program was reviewed to ensure that the issues identified in this IFI were entered into the licensee's corrective action process, that there were no apparent operability issues, and that adequate corrective action was being implemented to ensure that safety related components were adequately qualified and maintained.

b. Observations and Findings

Background

The Unit 3 material, equipment, and parts lists (MEPL) program was reviewed in inspection report 50-423/98-208 and several preceding inspections. The licensee was determined to have invested substantial effort in improving the program and had significantly upgraded both the program and the evaluations for many components and parts in Unit 3. ACR M3-96-0912 was inspected and EEI 96-201-43 of significant item list (SIL) Item No. 25 was closed in inspection report 423/98-207. The MEPL program was deemed to meet regulatory requirements and was determined to be adequate to support the restart of the unit.

Inspector Follow-up Item (IFI) 50-423/98-208-05 identified six aspects of the MEPL program that were of interest to the NRC.

Item 1 - The numbering scheme for the production maintenance management system (PMMS) resulted in differences between the identification for components in the field/on drawings and in the PMMS database, when the number of characters exceeded 15.

Item 2 - The designation of safety related components with an asterisk (*) as indicated in FSAR, Figure 3.2.2 stated that an asterisk indicates the equipment is quality assurance Category 1 (i.e., safety related). However, the inspector noted that not all safety related components use the asterisk as noted in the FSAR, e.g., SR snubbers. There is some ambiguity in the use of the asterisk for relays. Some identification tags and signs in the plant did not use the asterisk

Item 3 - The licensee's efforts to re-evaluate all components (in the safety related and augmented quality categories) for the proper quality consistent with plant design and licensing bases.

Item 4 - Licensee efforts to close CR M3-98-2667 and Operability Determination (OD) MP3-070-98, which were initiated to resolve, on an interim basis, the qualification of a number of components.

Item 5 - Licensee efforts to consolidate the corrective actions for TE-0022, Nonconformance Report (NCR) 397-010 and CR M3-98-2667.

Item 6 - Licensee efforts to improve the control of parts and consumables as described in CR M3-98-0407.

Inspection Activity

The inspector conducted an on-site review of the IFI, including a sample of Unit 3 operability determinations (ODs), selected design change documentation, associated corrective action documentation (including M3-98-2343, 98-2667, 98-1855, and 98-0407), selected Technical Evaluations (TE) and selected action request (AR) tracking assignments. In addition, a limited sample of dedication activities was performed. The licensee's actions in response to the IFI items were found to be adequate. Therefore, **IFI 50-423/98-208-05 is closed.** No violations of NRC requirements were identified.

c. Conclusions

The Unit 3 material, equipment, and parts lists (MEPL) program was reviewed in inspection report 50-423/98-208 and several preceding inspections. The licensee invested substantial effort to improve the program and had significantly upgraded both the program and the evaluations for many components and parts. The MEPL program was deemed to meet regulatory requirements and was determined to be adequate to support the restart of the unit. Six issues of interest were identified by the NRC for further review. Following a review of a sample of Unit 3 operability determinations, selected design change documentation, associated corrective action documentation, selected Technical Evaluations and selected AR tracking assignments, the licensee's actions in response to the IFI items were found to be adequate.

U3.III Engineering

U3 E1 Conduct of Engineering

E1.1 Site Engineering Activities

a. Inspection Scope (37551)

The inspector reviewed engineering activities related to two emergent equipment concerns during this report period. One involved the turbine driven auxiliary feedwater (TDAFW) pump recirculation flow concerns identified during the conduct of a surveillance test, as previously noted in Section U3.M1.1 of this report. The other issue involved the development of an action plan and engineering guidance with respect to the discovery of increased dissolved oxygen levels at the discharge point of the condensate pumps.

b. Observations and Findings

(1) In response to the low TDAFW pump recirculation flow rate (i.e., < 90 gpm) identified during surveillance testing, the licensee developed an operability determination (OD MP3-033-99) that established the operability of the TDAFW pump to perform its safety function. This OD documented additional analysis of the trend of decreasing measurements in recirculation flow rate since the conclusion of the Unit 3 refueling outage (RFO6), when the pump impellers were replaced, in June 1999. The inspector reviewed the engineering data and logic inherent in the OD and found consistency between the

measured discharge pump head, determined to meet technical specification requirements during recirculation, and the flow calculations associated with this differential head pressure.

The OD indicated that it might be appropriate to establish a new in-service testing (IST) flow reference value, based upon the lower flow measurement. However, the inspector questioned whether such action was consistent with the provisions of the relevant pump IST guidance (ASME/ANSI OM-6), which specifies that reference values can only be established when a pump is known to be operating acceptably. A licensee condition report (CR M3-99-3196), noting that the TDAFW pump was operable, documented the need to resolve the conflict between the minimum 90 gpm flow rate specified in the Unit 3 Final Safety Analysis Report (FSAR) and the surveillance test results. The licensee then developed an action plan to investigate the flow differences, establish contingencies, and plan for additional pump testing.

The licensee's action plan included a re-assessment of the pump vendor (i.e., Sulzer-Bingham) recommendation in 1992 to increase the intermittent flow requirements (e.g., during periodic surveillance testing) to better protect the pump against cavitation-erosion problems that may arise during low flow conditions. The licensee also conducted recirculation line orifice and valve radiography to confirm that the flow path was free of restrictions that might be causing an adverse impact upon the flow measurements. Additionally, since external, "strap-on" acoustic flow measurement devices ("controlotrons") are used during the conduct of the surveillance tests, the controlotron vendor was asked by the licensee to review the test setup and diagnostic data. This evaluation determined a difference between the controlotron drive frequency used in the calibration process and that used in the conduct of the surveillance test. This discrepancy was found to be caused by some metallurgical differences in the piping itself, even though the same material was used.

Subsequently, the surveillance test (SP 3622.3) was repeated using two different controlotrons, an earlier 960 model which had been used for surveillance testing prior to the recent RFO6 modification work on the TDAFW pump, and the newer 990 model which had yielded the lower flow results that were being questioned. The subsequent surveillance test resulted not only in an acceptable flow measurement >90 gpm, but also in correlatable test data which explained the offset results of the two controlotron models. The inspector reviewed the technical evaluation (M3-EV-99-0099) authorizing the use of two different controlotrons during the last acceptable test, and discussed with test and engineering personnel the test setup, the work controls, and interpretation of the surveillance results. As of the end of this inspection report period, the licensee was continuing to evaluate the minimum TDAFW pump flow criterion, the need for FSAR and surveillance procedure revisions, and the guidance of NRC Bulletin 88-04, relative to the overall value and need for IST using the recirculation flow path.

(2) As documented in Inspection Report 50-423/99-08, dissolved oxygen concentration levels at the condensate pump discharge (CPD) reached Action Level 1 limits (>10 ppb) following the Unit 3 restart from RFO6. These condenser oxygen levels had no immediate impact upon safety-related component operability. However, the Unit 3

secondary water chemistry program endorses the Electric Power Research Institute (EPRI) guidelines, which specify plant downpowers under certain action level conditions and time frames, in order to provide long-term protection to the steam generator materials. While the licensee was able to take corrective measures (e.g., backwashing to improve the condenser efficiency for oxygen removal) to exit Action Level 1 before plant downpowers were required, the elevated oxygen concentration adversely affected the overall unit chemistry performance index (CPI) and the licensee's ability to meet stated CPI goals. The licensee initiated an action plan to address this overall problem.

The licensee's action plan, which was implemented and updated periodically throughout this inspection period, included provisions, among a number of more detailed activities, for helium leak testing of the condenser and components, sealing the condenser shell and boot assemblies, repairing valve and various air ejector component leaks, increasing hydrazine injection to the condenser, searching for underwater leakage with the help of a contracted condenser efficiency expert, continued thermal backwashing, as necessary, and planning for additional component repairs and condenser cleaning activities during future cold shutdown plant conditions. During the period of this review, the dissolved oxygen concentration varied from approximately 8 - 12 ppb.

On September 23, 1999, the licensee's material performance group issued a memorandum (ME-MP-99-274) to the Unit 3 operations manager, temporarily raising for one month the CPD Action Level 1 dissolved oxygen limit from 10 ppb to 12 ppb. The inspector reviewed this memorandum, including the attachments discussing the options, standards, and impacts (e.g., increased potential for iron and copper transport to the steam generators), associated with operating at elevated CPD oxygen levels. The inspector noted that licensee investigation had determined that the dissolved oxygen levels in the feedwater system were normal and that there was no evidence of increased iron or copper transport or changes in the main steam conductivity levels. The licensee concluded that operation with the slightly higher dissolved oxygen concentration was acceptable. The inspector noted that the one time, time-specific change was made to CP 3802B to increase the limit to 12 ppb, as recommended in the memorandum.

In evaluating this licensee position, the inspector also reviewed Chemistry Procedure CP 3802B and Station Procedure NGP 2.17 for the Secondary Water Chemistry Program. The inspector noted that the procedural requirements for continued CPD oxygen level >10 ppb involved a power reduction below 30% and performance of a technical evaluation for continued operation. Since the licensee had already performed such an evaluation, with no evidence of the material transport that could adversely affect the steam generators, and since a power reduction would not significantly decrease the potential for adverse impact, if such existed, the inspector determined that the licensee's logic on extending the Action Level 1 limit for CPD dissolved oxygen was sound. In addition, inspector discussion with NRC materials engineers in NRR confirmed that the minimal increase in CPD dissolved oxygen Action Level 1 limits for the short term was not a safety significant concern. Near the end of this inspection period, with the seasonal falling service water temperatures and resulting increase in condenser efficiency, the CPD oxygen level dropped to approximately 7 ppb, with the likelihood of trending lower through the winter months. Despite this trend, the licensee continued to update its action plan for the CPD oxygen issue, recognizing that the problem will recur with rising service water temperatures next Summer.

c. Conclusions

The licensee developed detailed action plans to address two Unit 3 technical concerns that emerged or continued to persist throughout this inspection period. Engineering and operations coordination was effective in assessing the short-term component operability, as well as the longer-term impact on safety-related equipment. The licensee action taken to correct the immediate concerns had a sound engineering basis, while continued measures are planned to preclude problem recurrence.

U3 E2 Engineering Support of Facilities and Equipment

E2.1 (Open) Unresolved Item (URI) 50-423/99-09-13: Recirculation Spray System Cubicle Sump Pump Vane Growth

a. Inspection Scope (92903)

The inspector observed the following concerning the safety related recirculation spray system (RSS) cubicle sump pumps: 1) the shop disassembly of sump pump "B" (3DAS*P15B); 2) shop measurements of the length of the air motor vanes for both pumps (3DAS*P15A and B); and 3) briefings on pump status. Also, the inspector reviewed the pump manufacturer's operations and maintenance manual and the root cause evaluation for the May 1999 pump failures, including the test report and failure evaluation issued by National Technical Systems (NTS), the pump's qualification vendor.

b. Observations and Findings

The RSS cubicle sump pumps were installed to remove ground water intrusion past the degraded containment basemat rubber membrane. On September 22, 1999, sump pump "A" (3DAS*P15A) failed a monthly surveillance test. The train "B" pump passed its surveillance test. Subsequent examination of 3DAS*P15A found the air motor to be seized, apparently due to elongation of the air motor vanes from absorption of oil and/or moisture. Since the swelling of the vanes could represent a potential common mode failure, Northeast Nuclear Energy Company (NNECo) removed and disassembled 3DAS*P15B on September 23, 1999, after installing a spare pump in the "A" train.

Measurements of the vane length in the 3DAS*P15B air motor found them elongated beyond the original specification, but shorter than the inside height of the air motor housing. NNECo issued condition report M3-99-3273 to evaluate the condition for reportability, operability, and actions necessary to prevent recurrence.

The licensee initially wrote a reasonable expectation of continued operability (RECO) which stated the pumps were operable. After the licensee trimmed the pump blades, they considered not requiring a follow-up operability determination (OD) for the pumps. The inspector questioned this decision as compensatory measures were to be taken to ensure operability. Nuclear Oversight (NOS) had raised the same question in parallel and the inspector observed good discussion between NOS, engineering, and operations personnel on this issue. The associated operability determination, MP3-034-99, classified the pumps as operable. Both pumps had their air motor vanes restored to the original specification lengths and were reinstalled and tested satisfactorily. NNECo stated the pumps will be tested monthly, with one pump being disassembled to measure and evaluate blade elongation.

On September 23, 1999, NTS forwarded information to NNECo regarding a pump which had been utilized for the initial qualification. The pump had been in the warehouse for approximately two years. When removed, the air motor vanes measured 3.0000, 3.0015, 3.0070, and 3.0035 inches in length. This indicates that the vanes in both sump pumps had elongated to a substantial portion of the expected growth in the four months they had been in service. It can be reasonably expected that any additional blade elongation will not adversely impact pump operation because NNECo will conduct vane measurements and will evaluate blade elongation on a monthly basis.

"Test Report for the Failure Evaluation of Chicago Pneumatic Pumps Used at Northeast Utilities MP-3," issued by National Technical Systems on May 27, 1999, included an evaluation of the failure mechanism of both sump pumps during their surveillance test failures on May 16, 1999. The evaluation found that both air motors contained grit and corrosion, and that the vanes had elongated beyond the original installed lengths, but were all less than or equal to 3.000 inches. The vane lengths were evaluated as acceptable by NTS on the basis of "... the motor liner has an overall length of greater than 3 inches." Although the licensee appropriately addressed and corrected the problems with grit and corrosion found in the air motors as part of the earlier failure evaluation, the inspector questioned why the vane growth mechanism had not been further analyzed as a potential, future failure mechanism.

The inspector also questioned the applicability of 10 CFR Part 21 reporting requirements to the identified vane growth problem. The licensee's associated reportability determination was in progress at the end of the report period. Pending completion of the analysis for the September 16, 1999, failure of 3DAS*P15A, issuance of Technical Evaluation M3-EV-99-0098, review of the 10 CFR Part 21 reportability determination, and review of the results of the monthly measurements of vane elongation, this issue remains unresolved **(URI 50-423/99-09-13)**.

c. <u>Conclusion</u>

The repetitive failure of the Recirculation Spray System cubicle sump pumps appears to stem from ineffective root cause evaluations by engineering, in that, the evaluations did not identify that swelling and elongation of the air motor vanes could be a potential failure mechanism for the recirculation system cubicle sump pumps. An unresolved item was identified pending further review of the analysis of the failed recirculation system pump 3DAS*P15A cubicle sump pump, review of Technical Evaluation M3-EV-99-0098, review of the 10 CFR Part 21 reportability determination, and review of the results of the monthly measurements of vane elongation.

IV Plant Support

(Common to Unit 1, Unit 2, and Unit 3)

R1 Radiological Protection and Chemistry Controls

R1.1 (Update) URI 50-423/99-08-12: 7.04 REM Exposure Event

a. Inspection Scope (83750)

On July 8, 1999, during routine processing of second quarter 1999 personnel TLD, a personnel TLD indicated 7.04 rem, which is above the annual personnel exposure limit of 5 rem. The initial investigation conducted by the radiation protection (RP) technical staff was thorough and comprehensive as previously reported in Inspection Report 50-423/99-08. The licensee determined that the individual did not receive the radiation exposure, but that the TLD exposure was real and may have been caused by tampering. A subsequent security investigation was conducted.

A preliminary security investigation report and the status of corrective actions was reviewed with respect to this condition.

b. Observations and Findings

The preliminary security investigation determined that there were likely circumstances where TLD tampering may have occurred. However, the preliminary investigation did not establish any conclusive facts to substantiate tampering. In view of the fact that the investigation was not extensive or comprehensive, the Radiation Protection Manager and the Station Director for Operations decided to initiate a more thorough investigation of this matter.

Notwithstanding the continuing investigation, the licensee did initiate corrective measures to address possible contributing and potential causes. For example, the licensee is evaluating methods to improve (1) the control and accountability of personnel TLDs when not in use; (2) the distinction between TLDs that may be used for exposure experiments and measurements versus personnel monitoring applications; and (3) the positive identification of TLDs assigned to personnel upon entry to the Radiologically Controlled Area.

c. Conclusion

Continuing investigation into a July 8, 1999, 7.04 rem personnel TLD exposure is focused on resolving the potential for deliberate tampering and irradiating of the TLD. While the overall investigation is not yet completed, the licensee has initiated several corrective measures to address contributing or potential causes, and effect improved positive control of TLDs used to monitor personnel exposure.

R1.2 External Exposure Controls During Unit 2 Emergent Outage

a. Inspection Scope (83750)

The inspection included accompaniment of a work crew assigned to replace three snubbers inside containment during a 48 hour emergent shutdown condition. The work crew entry included a confined space entry and high radiation area (greater than 1 R/hr) entry. Observations included all pre-job briefings and containment entry controls.

b. Observations and Findings

The pre-job meeting covered the pertinent confined space entry controls and the containment entry utilized two dedicated attendants outside containment. Containment atmosphere was appropriately sampled, regular communications were established and maintained with the attendants during performance of the work. RP controls included an appropriate radiological briefing and constant RP technician accompaniment during work performance. No discrepancies were noted.

c. Conclusions

Radiological controls were effectively implemented during a Unit 2 forced outage in late September 1999.

R1.3 Internal Exposure Program Review

a. Inspection Scope (83750)

Internal exposure procedures and gamma spectroscopy instrument calibration documents were reviewed and interviews were conducted with applicable personnel.

b. Observations and Findings

Based on a previous inspection, (NRC Inspection No. 50-336/99-08; 50-423/99-08 conducted on July 26-30, 1999), the procedure RPM 2.10.2, Rev. 6, "Air Sample Counting and Analysis," did not specify sufficiently low sensitivity to ensure 0.1 DAC of alpha emitting radionuclides were detectable. During this inspection, the licensee had revised the subject procedure to a 0.1 DAC alpha instrument counting sensitivity.

A review of bioassay and internal exposure assessment procedures identified two areas where exposure assessment with respect to alpha emitting radionuclides may be improved. In procedure RPM 1.3.12, Rev. 5, "Internal Monitoring Program," the decision to obtain fecal samples to perform alpha measurements depends on positive air sample results analysis. In cases where workers' internal exposure is not adequately represented by an air sample, the procedure does not specify a method of monitoring for the presence of alpha emitting transuranics. In procedure RPM 1.3.14, Rev. 4, "Personnel Dose Calculations and Assessments", there was a reference suggesting the consideration of non-gamma emitting radionuclides but no specific requirement to apply the information in internal dose assessments. Notwithstanding, current waste stream analysis indicates the presence of several non-gamma emitting radionuclides having potential dose significance at each Millstone Unit. In response, the Health Physics Support Group initiated action to re-evaluate the applicable procedures and revise as necessary.

During 1999, there were no internal exposures assigned to Millstone personnel. Two internal exposure assessments performed in 1998 were reviewed indicating the appropriate use of whole body counting measurements and also accounting for the undetectable alpha emitters through the use of calculations.

Calibrations of whole body counters and air sample counting gamma spectroscopy instruments were current and daily quality performance tests were reviewed indicating continuing good performance of the gamma spectroscopy instruments.

c. Conclusions

The internal exposure measurement and dose assessment program at Millstone is effective.

R2 Status of RP&C Facilities and Equipment

R2.1 (Update) Violation 01172/EEI 50-245/96-003-01: Liquid Radwaste Management System

During the inspection period, inspectors toured and reviewed licensee activities regarding the liquid radwaste management system at Unit 1. Specifically, the inspectors reviewed the extent of the licensee's implementation of their Radwaste Remediation Program that followed the NRC's identification and subsequent issuance of a violation as detailed in NRC Inspection Report 50-245/96-03.

The inspectors determined that the historical remediation program did not appear to have been effectively implemented due to the numerous resources the licensee had directed at the recovery of the remaining two operating units. While the inspectors did observe that the physical condition and operation of the equipment had improved, questions regarding the overall functional capabilities of various systems to adequately perform their required functions have been raised. However, the licensee has recently planned to re-define the radwaste remediation program such that relevant issues are resolved, consistent with both the current plant status, and the future needs of the system during the decommissioning of the unit. Therefore, Violation 01172 (EEI 96-003-01) will remain open pending the licensee's implementation of a revised Radwaste Remediation Program.

7 Quality Assurance in Radiological Protection and Chemistry Activities

R7.1 Problem Identification and Resolution of RP Activities

a. Inspection Scope (83750)

A sample of 20 condition reports issued in 1999, monthly summary RP surveillance reports for 1998 and 1999, and a quality assurance RP program audit conducted in November 1998 were reviewed.

b. Observations and Findings

The problems identified in the various licensee corrective action processes were minor with the exception of the 7.04 rem TLD exposure mentioned earlier in this report and the identification by the licensee of an increasing trend of radiation worker deficiencies. The licensee has been addressing the radiation worker deficiencies through establishment of a common cause investigation team to determine any common underlying causes and recommendations. Some of the underlying causes include separate Unit RP departments with different RP practices, and different Unit RP facility layouts with variable availability to radiation workers. These causes and others are being addressed through the condition report process.

c. Conclusions

The RP program has an active oversight and self-assessment program that engages problems in an effective manner.

R8 Miscellaneous RP&C Issues

R8.1 Post Shutdown Decommissioning Activities Report Environmental Review (Unit 1)

a. Inspection Scope

On August 24 - 25, 1999, NRC staff from the Office of Nuclear Reactor Regulation performed an audit of the environmental records associated with the Millstone Unit 1 Post Shutdown Decommissioning Activities Report (PSDAR), submitted by letter dated June 14, 1999. The audit addressed environmental records associated with the PSDAR, the licensee's decommissioning Environmental Report (ER), and a report prepared by TLG Services, Inc. that formed its basis. The purpose of the staff audit was to determine whether the licensee had an adequate basis for concluding that the environmental impacts associated with the anticipated decommissioning activities at Unit 1 were bounded by appropriate, previously issued impact statements, as required by 10 CFR 50.82(a)(4)(i). Because the Final Environmental Statement for Millstone Unit 1 (issued at the operating license stage) did not address decommissioning impacts, the Generic Environmental Statement on Decommissioning of Nuclear Facilities (GEIS) and its related support documents provide the envelope for all previously issued impact statements.

b. Observations and Findings

Dose Estimates

Using methodology from the TLG report and dose surveys of general radiation areas conducted in March 1999, the licensee presented the results of a cubical-by-cubical assessment of decommissioning and decontamination (D&D) at various times after shutdown, which also included the SAFSTOR decommissioning option.

The licensee's estimated occupational exposure from decommissioning activities was combined with the estimated dose to the public (including considerations of dose to transportation onlookers and the general public) to obtain a cumulative total dose for each decommissioning scenario. The licensee's estimates for both the occupational exposure and the dose to the public were less than the values in Table 5.3-2 of the GEIS. The cumulative dose estimated to result from D&D and from modified SAFSTOR at Millstone Unit 1 is within the bounds assessed for the Decon alternative presented in the GEIS.

Waste Volume

The licensee's ER summarized the breakdown of waste classifications, locations, and anticipated disposal site, which are all based on the TLG report. The licensee's estimates of waste volume, however, do not take credit for waste compaction/volume reduction technology currently in use on-site.

The estimated volume of contaminated soil requiring off-site disposal was based on: (1) historical survey information on file at the site; (2) interviews with operations department personnel; and (3) a review of operational logs to determine where activities had occurred. Mixed waste was also included in the total estimated 165,000 ft^3 of contaminated soil that will be shipped off-site by truck for disposal. This volume is less than the estimated burial volume of low level radioactive waste and rubble for the reference BWR in Table 5.4-1 of the Decommissioning GEIS.

Accidents

The only accident considered by the licensee in its environmental review for decommissioning is the fuel handling accident and applicable consequences that are already addressed as part of the Chapter 15 analysis in the FSAR.

Transportation Shipments

The licensee expects that all shipments, except for shipments of greater than Class C waste (to be transported by rail), will be accomplished by truck. The 1382 shipments needed to dispose of radioactive waste resulting from decommissioning activities at Unit 1 is lower than the 1495 shipments assumed in Table I.3-2 of NUREG-0672. This includes shipments for disposal of the remaining wastes from the operational period (also known as legacy waste).

c. Conclusions

The NRC determined that the licensee has developed and documented an adequate basis for concluding that the environmental impacts associated with the anticipated decommissioning activities at Millstone Unit 1 are bounded by appropriate, previously issued impact statements, as required by 10 CFR 50.82(a)(4)(i).

R8.2 (Closed) Violation (VIO) 50-245/97-01-08: Failure To Monitor Gaseous Effluents from the Radwaste Storage Building: The inspector reviewed the licensee's actions in response to the NRCs issuance of VIO 97-01-08, as discussed in NRC Inspection Report 50-245/97-01. Specifically, the violation involved the licensee's failure to monitor effluents released to the environment from the Radwaste Storage Building through exhaust fan HVE-14. The licensee appropriately initiated condition reports to address the adverse condition through the corrective action program. The inspector verified that the licensee has completed all corrective actions to resolve the unmonitored release violation, with the exception of a remaining action to establish the adequacy of the installed filtration system in accordance with applicable regulatory requirements and standards. The remaining corrective action is currently being planned by the licensee, therefore, the inspector had no further questions regarding the completion of the corrective actions. As a result, the inspector concluded that the licensee's actions are adequate and support the closure of the violation, and **VIO 50-245/97-01-08** is **closed.**

P1 Conduct of Emergency Preparedness Activities

P1.1 Off-Year Emergency Preparedness Exercise (71750)

On September 15, 1999, the inspector observed portions of the licensee's off-year emergency preparedness exercise in the operations support center (OSC) and technical support center (TSC). The inspector subsequently attended the player/controller and evaluator debrief meetings and the drill critique. Appropriate control of OSC teams was observed with proper tracking, prebrieifing and debriefing. Effective communication was observed between the OSC and TSC personnel.

The licensee's evaluation of the exercise was thorough and identified both positive and negative findings and concluded that the health and safety of the public would have been protected. The licensee's critique communicated the failure of five exercise objectives in the areas of accident projection, emergency classification, notification of onsite/offsite responders, communications between facilities, and magnitude of release components. The inspector noted senior licensee management support of the EP program during the critique, evidenced by their statements to take any actions necessary to improve the organization's performance in this area. The inspector noted that a CR was written to document the evaluation findings and effect corrective actions.

S1 Conduct of Security and Safeguards Activities

S1.1 Inspection of Vital Area Doors, Hatches and Alarms (71750)

During inspection-tours of the plant, the inspector observed station security officers performing routine patrols, security checks, and compensatory functions. While examining the integrity of some roof hatches on the Unit 3 engineered safety features (ESF) building roof, the inspector noted one member of the guard force checking a door with a locked and disabled key reader. Discussion with this security officer revealed that access to certain vital areas (e.g., supplementary leak collection and release system [SLCRS] doors) had been disabled because of the personnel safety hazard associated with entry into these areas without the proper equipment (e.g., flashlights). After radio communication with the central alarm station, the guard coordinated the unlocking of the key reader, allowing the inspector access for inspection. The inspector examined the SLCRS area for material condition, "shake" space integrity, and the presence of equipment that might require immediate operator response to the area. No inspector concerns were identified.

As discussed in Section U3.O1.2 of this report, the inspector checked the watertight door status in Unit 3 in accordance with the hurricane preparations delineated in an abnormal operating procedure, AOP 3569. Since some of these doors also represented vital barrier boundaries, the inspector assessed the security condition of each door at the same time, along with certain doors specified in AOP 3569 as those provided for tornado protection. After the tropical storm passed the site, the inspector visited the secondary alarm station (SAS) at the plant, observing the compensatory measures enacted in response to any storm damage to the protected area security detection devices. The inspector interviewed security officers in the SAS and determined that the appropriate actions were in effect.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection period. The licensee acknowledged the findings presented.

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 37551	Onsite Engineering
IP 60705	Preparation for Refueling
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 83750	Occupational Radiation Exposure
IP 90712	Power Reactor Facilities
IP 92700	Onsite Follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901	Follow-up - Operations
IP 92902	Follow-up - Maintenance
IP 92903	Follow-up - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-245/99-09-01	NCV	Failure to include the weight of the lower mast in support of both the SFP storage rack design and the fuel handling accident of the FSAR
50-336/99-09-02	NCV	Failure to adequately establish and maintain plant cooldown procedure to reflect a change in RPS setpoint
50-336/99-09-03	NCV	Failure to report an unplanned actuation of RPS
50-336/99-09-04	NCV	Failure to initiate a condition report to document that critical data was not taken at the specified power level during a reactor startup
50-336/99-09-05	NCV	Inadequate design control measures to assure that the 4160 volt switchgear room coolers were capable of maintaining a suitable environment for the vital switchgear under post-accident conditions
50-336/99-09-06	NCV	(Closure of LER 97-35) Shutdown cooling isolation valve does not comply with Appendix R
50-336/99-09-07	NCV	(Closure of LER 98-03) Inadequate design controls to evaluate interaction between the reactor internals and reactor vessel
50-423/99-09-08	NCV	Failure to adequately implement design controls to ensure that the SW design basis was correctly translated into specifications, drawings and procedures
50-423/99-09-09	NCV	Failure to ensure that the SI accumulator valve design basis was correctly translated into specifications, drawings and procedures
50-423/99-09-10	NCV	Failure to adequately implement design controls to ensure that the EDG design basis was correctly translated into specifications, drawings and procedures
50-423/99-09-11	NCV	Failure to adequately implement design controls to ensure that the CAM design basis was correctly translated into specifications, drawings and procedures
50-423/99-09-12	NCV	Ineffective implementation of safety-related design control measures for the relocation of an emergency lighting box
50-423/99-09-13	URI	Failure of 3DAS*P15A, issuance of Technical Evaluation M3-EV-99-0098, review of the 10 CFR Part 21 reportability determination, and review of the results of the monthly measurements of vane elongation

Closed

The NCVs opened above are closed.

50-245/95-07-01	VIO	Control Room Habitability/Use of Self-Contained Breathing Apparatus
50-245,336, 423/96-09-03	URI	Organizational Changes
50-245/97-02-02	URI	RP-4 Interface with Lower Tier Reporting Processes
50-336/96-08-21	IFI	Unit 2 Materiel Condition Program
50-423/97-202-03	IFI	Loss of Spent Fuel Pool Cooling
50-423/97-202-07	IFI	Letdown Heat Exchanger ASME Code Compliance

50-423/97-203-06	URI	Rosemount Transmitters
50-423/98-208-04	VIO	Failure to Implement Plant Heatup Procedure
50-423/98-208-05	IFI	Material, Equipment, and Parts Lists (MEPL) Review

Discussed

50-245/96-03-01	EEI	Liquid RadWaste Management System
50-423/99-08-12	URI	7.04 REM Exposure Event

The following LERs were also closed during this inspection:

LER 50-245/96-023-00 & 01	Movement of New Fuel Assemblies Over the Spent Fuel Pool
LER 50-245/97-013-00 & 01	Evaluation of Impact of Refueling Platform Fuel Grapple
LER 50-336/96-30-01& 02	In-service Test Program Deficiencies
LER 50-336/97-22-00,01, 02,&03	Technical Specification Violations
LER 50-336/97-35-00	Shutdown Cooling System Isolation Valve Does Not Comply with Appendix R Requirements
LER 50-336/98-02-00&01	Emergency Core Cooling System Single Failure Vulnerability
LER 50-336/98-03-00	Inadequate Evaluation of the Interaction Between the Reactor Internals and the Reactor Vessel
LER 50-423/98-01	Safety Injection (SI) Accumulator Outlet Isolation Valve Motor Pinion Gear Key Failure
LER 50-423/98-04	Emergency Diesel Generator (EDG) Service Water (SW) Fouling
LER 50-423/98-09	Containment Atmosphere Monitoring (CAM) System Radiation Monitor Setpoints
LER 50-423/98-37	Technical Specification 3.0.3 Entry Due to Both Service Water (SW) Trains Declared Inoperable Following Failure of Check Valves Associated with the Injection of Sodium Hypochlorite
LER 50-423/98-38	Manual Reactor Trip Due to a High Condensate Pump Discharge Conductivity
LER 50-423/98-39	Technical Specification 3.0.3 Entry Due to Vital inverter Failure Resulting in the "A" and "C" Recirculation Spray System (RSS) Pumps Being Inoperable While the "B" RSS Pump Was Out of Service for Maintenance
LER 50-423/98-40	Technical Specification 3.0.3 Entry Due to Both Quench Spray System Pumps Placed in "Pull-to-Lock"
LER 50-423/98-43	Manual Reactor Trip Due to High Conductivity in the Condensate System
LER 50-423/98-44	Reactor Trip Due to High Differential Pressure Between"A" and "B" Condensers
LER 50-423/98-45	Reactor Trip Due to a Main Steam Isolation Valve (MSIV) Closure during Partial Stroke Testing Due to Solenoid valve Failure
LER 50-423/98-46	Manual Reactor Trip Due to a Difference in Indicated and Demand Rod Positions
LER 50-423/99-01	Technical Specification 3.0.3 Entry Due to a High Energy Line Break (HELB) Boundary Door Latch Failure

LIST OF ACRONYMS USED

ACR(s)	adverse condition report(s)
AOP(s)	abnormal operating procedure(s)
ASME	American Society of Mechanical Engineers
AWO(s)	automated work order(s)
CAM	containment atmosphere monitoring
CIV(s)	containment isolation valve(s)
CND	condensate demineralizer
CPD	condensate pump discharge
CPI	chemistry performance index
CR(s)	condition report(s)
CWS	circulating water system
DAC	derived air concentration
DAS	plant aerated drain system
DRPI	digital rod position indicator
ECCS	emergency core cooling system
EDG(s)	emergency diesel generator(s)
EPRI	Electric Power Research Institute
ESF	engineered safety feature
FPC	fuel pool cooling
FSAR	Final Safety Analysis Report
GDC	general design criterion/criteria
HELB	high energy line break
HVK	control building chilled water
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
IFI	inspector follow-up item
IST	in-service testing
LCO	limiting condition for operation
LER(s)	licensee event report(s)
MCP	material condition program
MEPL(s)	material, equipment, and parts list(s)
MMOD	minor modification
MSIV	main steam isolation valve
NCR(s)	nonconformance report(s)
NCV(s)	non-cited violation
NFSV	new fuel storage vault
NNECO	Northeast Nuclear Energy Company
NOS	Nuclear Oversight
NRC	Nuclear Regulatory Commission
NTS	National Technical Systems
OD	operability determination
OP	operating procedure
OSC	Operations Support Center
PDR	Public Document Room
PMMS	production maintenance management system

PORC	plant operation review committee
QA	quality assurance
QSS	quench spray system
RCS	reactor coolant system
RFO	refueling outage
RP	Radiation protection
RPS	reaction protection system
RSS	recirculation spray system
SAS	secondary alarm station
SCR	silicon controlled rectifier
SFP	spent fuel pool
SI	safety injection
SIL	significant item list
SLCRS	supplementary leak collection and release system
SP(s)	surveillance procedure(s)
SPROC	special procedure
SW	service water
SWP	service water pump
TDAFW	turbine driven auxiliary feedwater
TE(s)	technical evaluation(s)
TLD	Thermoluminescent dosimeter
TRM	Technical Requirements Manual
TS(s)	technical specification(s)
TSC	technical support center
URI	unresolved item
VIO	violation