U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Report No.: 50-423/98-211

Docket No.: 50-423

License No.: NPF-49

Licensee: Northeast Nuclear Energy Company

Facility: Millstone Unit 3

Location: Millstone Nuclear Power Station
156 Rope Ferry Road
Waterford, Connecticut 06385

Dates: April 13 - 24, May 11 - 15, May 27 - 29, June 9 - 11, June 23 - 25, 1998

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EXECUTIVE SUMMARY

On April 13 through 24, May 11 through 15, May 27 through 29, June 9 through 11, and June 23 through 25, 1998, a team from the U.S. Nuclear Regulatory Commission (NRC), Special Projects Office, Office of Nuclear Reactor Regulation, performed the final phase of a corrective action implementation inspection of the Millstone Nuclear Power Station Unit 3 facility (Unit 3), licensed to Northeast Nuclear Energy Company (NNECO). This inspection was one part of an ongoing, multifaceted NRC evaluation of the Independent Corrective Action Verification Program (ICAVP) being conducted at Unit 3 by Sargent & Lundy (S&L). The corrective action inspection was conducted in several phases as corrective actions were completed by NNECO. The first phase of the corrective action implementation inspection results are documented in Inspection Report (IR) No. 50-423/98-205.

In accordance with SECY-97-003, "Millstone Restart Review Process," the scope of the second phase of the ICAVP corrective action inspection included an evaluation of the licensee's corrective actions and other ICAVP process activities, as follows:

- Corrective actions implemented in response to findings identified during NRC inspections associated with oversight of the ICAVP including inspections documented in NRC IR Nos. 50-423/97-206, Tier 1 Out-of-scope system inspection; 50-423/97-209, Tier 2/3 inspection; and 50-423/97-210, Tier 1 Inscope system inspection.

- Corrective actions for Licensee Event Reports (LERs) that were identified by the licensee while conducting its Configuration Management Plan (CMP) or S&L, the ICAVP contractor, within the ICAVP review scope..

- Corrective actions implemented for Confirmed Level 3 Discrepancy Reports (DRs) identified by the S&L.

- A sample of DRs that were Previously Identified during NNECO's CMP or those that were Non-discrepant to assess whether the process had properly categorized these DRs.

- A sample of Level 4 DRs to validate that they were appropriately classified as Level 4.

- DRs in which multiple interactions between S&L and the licensee occurred to determine whether the multiple interactions indicated an inadequate corrective action process.

- Unit 2 preliminary DRs that identified possible programmatic weaknesses to determine if similar issues existed in Unit 3.

- Self-assessments performed by NNECO in response to the NRC identified potential for vapor binding of the charging and safety injection pumps, and the self-revealing failure of the Recirculation Spray System (RSS) expansion bellows.

- Additional corrective actions taken by NNECO in response the RSS expansion bellows failure.

- Examining DRs collectively to determine if any trends in the findings could be identified.

During the current inspection, the NRC closed 13 LERs (LERs 50-423/96-050-00, 97-013-00, 97-020-00, 97-031-00, 97-032-00, 97-033-00, 97-036-01, 97-038-00, 97-039-00, 97-043-00, 97-045-00, 98-007-00, and 98-029-00). Closure of LER 50-423/98-007-00, that described the failure to include all containment bypass leakage pathways in the calculation for containment leakage resulted in a cited violation of NRC requirements. However, the licensee was not required to respond to this violation because the NRC had reviewed the corrective actions during this inspection. Also, in closing LER 50-423/98-029-00 the NRC considered escalated enforcement and a civil penalty. LER 50-423/98-029-00 documented a condition in which nonsafety-related exhaust fans in the Engineered Safety Features and Auxiliary Building Ventilation Systems could continue to operate, thereby increasing the potential for leakage of contaminated air to the atmosphere during a Loss-of-Coolant-Accident, with offsite power available). However, the NRC decided to exercise enforcement discretion pursuant to Section VII.B.2 of the NRC's Enforcement Policy, "Violations Identified During Extended Shutdowns or Work Stoppages," rather than issuing a formal Notice of Violation.

In addition, the report documents two examples of a violation of 10 CFR 50.73 for failure to submit LERs until questioned by the NRC for two conditions that were outside the design basis of the plant. These issues involved the maximum design steam flow through the turbine bypass valves (LER 50-423/98-026-00) and the maximum design temperature in service water pump cubicles (LER 50-423/98-035-00). However, the NRC decided to exercise discretion for these issues in accordance with Section VII.B.2 of the Enforcement Policy, rather than issuing a formal Notice of Violation.

The NRC team reviewed the two-part, NRC identified, violation for failure to comply with the requirements of the Boiler and Pressure Vessel Code promulgated by the American Society of Mechanical Engineers (ASME) (VIO 50-423/97-206-03). The first part of the violation involved failure to consider stress indices during the installation of the charging system restrictive orifices. The team concluded that NNECO's corrective actions for this part of the violation were appropriate. The second part of the violation involved sizing the discharge piping for charging system relief valves 3CHS*RV8119 and 3CHS*RV8123. After reviewing additional information, the team concluded, based on discussions with the staff, that the discharge piping for relief valves 3CHS*RV8119 and 3CHS*RV8123 is sized in accordance with Code requirements. Therefore, the NRC has withdrawn the second part of violation 50-423/97-206-03.

The team also reviewed the three-part, NRC-identified, violation for failure to ensure that the Final Safety Analysis Report (FSAR) contained the latest information (VIO 50-423/97-209-03). The team concluded that NNECO's response and corrective actions were appropriate for the parts of the violation that involved inadequate FSAR descriptions of the RSS pump seal cooling system and the Steam Generator Tube Rupture event analysis. The third part of the violation involved radiation dose assessments in FSAR Table 15.0-8. In this instance, based on additional information provided by the licensee, the team determined that the values contained in Table 15.0-8 were consistent with calculation results. Therefore, the third part of violation 50-423/97-209-03 was withdrawn.

The NRC team closed 2 Escalated Enforcement Items (EEIs). EEI 50-423/97-209-02 describes the failure to monitor Service Water (SW) leakage from the RSS heat exchangers in accordance with Technical Specification 6.8.4. EEI 50-423/97-209-06 describes the failure to implement adequate corrective actions for the single-failure vulnerability of control room inlet dampers. In a letter dated May 6, 1998, NNECO responded to the EEIs and identified

corrective actions. The NRC determined that these EEIs constituted violations of NRC requirements; however, they did not warrant escalated enforcement action because are of low safety significance. Recent testing identified that there was no SW leakage from RSS heat exchangers, and revised calculations indicated that the time required to shut a control room inlet damper following an accident was not critical. These EEIs are being cited as violations of NRC requirements.

The NRC staff determined that the DRs reviewed were properly categorized, and acceptable corrective actions were assigned to the LERs reviewed and the RSS bellows failure. The outage modification self-assessments appropriately identified areas that required additional actions, and the corrective actions for these self-assessment findings were acceptable. The corrective actions assigned to NRC inspection report findings and Level 3 DRs were also acceptable. The issues identified in the Unit 2 preliminary DRs reviewed were either previously identified and corrected during the Unit 3 CMP, or were not applicable to Unit 3. DRs with multiple interactions between S&L and the licensee did not indicate an inadequate corrective action process; rather, these repeated interactions were a consequence of the communications protocol between the licensee and S&L. DR trends in the areas of calculation control/accuracy and drawings/component information were identified, but were considered not to be significant because the types of errors identified in these trends, even when viewed collectively, did not suggest that an expansion of the ICAVP scope would have identified issues that would call into question conformance with the design and licensing bases.

Overall, the team found that NNECO's implementation of corrective actions during the CMP was acceptable in that conditions adverse to quality were identified and corrected in accordance with Criterion XVI, "Corrective Action," of Appendix B to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50). The violations identified during this inspection are categorized as being equivalent to ICAVP significance Level 3 findings. In a letter to NNECO dated January 30, 1998, the NRC stated that if the reviews conducted by either the ICAVP contractor or the NRC confirmed an ICAVP Significance Level 3 finding, the NRC staff would consider expanding the scope of the ICAVP taking into consideration your corrective actions. During this inspection the NRC staff determined that NNECO has taken effective corrective actions for ICAVP significance Level 3 findings identified by the NRC and the ICAVP contractor and that these corrective actions represented an appropriate expansion of the scope of NNECO's CMP to provide confidence that similar issues, if present, would likely have been found. Therefore, expansion of the ICAVP scope is not warranted.

1.0 Introduction

A team from the U.S. Nuclear Regulatory Commission (NRC), Special Projects Office, Office of Nuclear Reactor Regulation, performed the final phase of a corrective action implementation inspection of the Millstone Nuclear Power Station Unit 3 facility (Unit 3), licensed to Northeast Nuclear Energy Company (NNECO). The Independent Corrective Action Verification Program (ICAVP) corrective action inspection was conducted in phases as NNECO completed corrective actions. The results of the first phase of the corrective action implementation inspection are documented in Inspection Report (IR) No. 50-423/98-205. The scope of this phase of the ICAVP corrective action inspection included evaluating:

- Corrective actions implemented in response to findings identified during NRC inspections associated with oversight of the ICAVP including inspections documented in NRC IR Nos. 50-423/97-206, Tier 1 Out-of-scope system inspection; 50-423/97-209, Tier 2/3 inspection; and 50-423/97-210, Tier 1 Inscope system inspection.

- Corrective actions for Licensee Event Reports (LERs) that were identified by the licensee while conducting its Configuration Management Plan (CMP) or S&L, the ICAVP contractor, within the ICAVP review scope..

- Corrective actions implemented for Confirmed Level 3 Discrepancy Reports (DRs) identified by the S&L.

- A sample of DRs that were Previously Identified during NNECO's CMP or those that were Non-discrepant to assess whether the process had properly categorized these DRs.

- A sample of Level 4 DRs to validate that they were appropriately classified as Level 4.

- DRs in which multiple interactions between S&L and the licensee occurred to determine whether the multiple interactions indicated an inadequate corrective action process.

- Unit 2 preliminary DRs that identified possible programmatic weaknesses to determine if similar issues existed in Unit 3.

- Self-assessments performed by NNECO in response to the NRC identified potential for vapor binding of the charging and safety injection pumps, and the self-revealing failure of the Recirculation Spray System (RSS) expansion bellows.

- Additional corrective actions taken by NNECO in response the RSS expansion bellows failure.

- Examining DRs collectively to determine if any trends in the findings could be identified.

In conducting this inspection, the NRC team relied upon NRC Inspection Procedures 92701, "Followup," and 92702, "Followup on Corrective Action for Violations and Deviations."

2.0 Corrective Action

In a letter to the licensee dated January 30, 1998, the NRC stated that possible expansion of the ICAVP scope in response to individual findings would depend on the specific findings and the effectiveness of the licensee's corrective actions. For individual Level 3 ICAVP findings,

effective licensee corrective action must address the specific "defect," as well as any broader implications for other systems.

2.1 Violations and Escalated Enforcement Items

2.1.1 Scope of Review

During this inspection, the NRC team reviewed violations (VIOs) and Escalated Enforcement Items (EEIs) identified during the ICAVP. Through this review, the team verified the following:

- Licensee management assigned responsibility for implementing corrective actions, including any necessary changes in procedures and practices.
- Corrective actions were appropriate and implemented.
- The licensee performed a root-cause analysis when appropriate.
- The licensee identified and corrected repetitive deficiencies.
- The licensee's operability and reportability determinations related to violations and EEIs were appropriate.
- The licensee expanded the scope of its corrective actions to include applicable related systems, equipment, procedures, and personnel actions when appropriate.
- Deferred items and interim resolutions were acceptable.
- Corrective actions were scheduled to be implemented before entering the appropriate mode of operation.

2.1.2 Findings

In reviewing the Violations and EEIs identified during the ICAVP, the NRC team reviewed 24 open items as discussed in the following paragraphs.

(a) Accuracy of Information in Design-Basis Documentation Packages

(Closed) VIO 50-423/96-201-09 (EEI 50-423/96-201-09) and Significant Items List (SIL) NO. 79.1, "Failure to Ensure that Adequate Design Control Measures Were Established for Verifying and Checking the Accuracy of the Information in the Design Basis Documentation Packages (DBDPs)." This issue involved the licensee's failure to maintain current DBDPs. In the response to this violation, the licensee stated the following corrective actions:

- Recall the present DBDPs because they are not quality-controlled documents. Maintain the "Safety Functions Manual," MP3-DBDP-SFR, as a quality controlled document.

- Identify and resolve DBDP discrepancies before startup as necessary.

- Procedure U3-PI-29, "Development of Millstone Unit 3 Design Bases Summary Documents," issued for the restart program, addresses Millstone Unit 3 Maintenance Rule Group 1 and 2 systems that are either required to function to mitigate design-basis accidents and/or are considered to be risk significant. These Design Basis Summary Documents are intended to summarize the safety- or risk-significant functions and performance requirements, and to provide a "road map" to appropriate information (i.e., drawings, calculations, specifications, etc.) that support the design bases.

The team verified that all Design Basis Summary Documents were issued. The team reviewed the 26 DBDP discrepancies identified by the licensee; the resolution of 15 of the discrepancies were deferred until after startup, and 11 discrepancies were scheduled to be resolved before startup. The team concurred with the licensee's disposition and tracking of these discrepancies and determined that the corrective actions taken were acceptable.

(b) Sections of Final Safety Analysis Report (FSAR) Not Updated

(Closed) VIO 50-423/97-206-1, "Violation for Failure to Properly Update FSAR in Accordance with 10 CFR 50.72." This issue involved two examples where the FSAR was not updated. The first example involved FSAR Section 6.3.3.2 that stated that the makeup flow rate for one charging pump is adequate to sustain a pressure of 2235 psig for a break through a 0.375-inch diameter hole. However, the licensee had not conducted any testing to confirm this statement. Test data obtained during post-modification testing indicated that the charging pumps cannot perform as described in the FSAR statement. In its response to this violation, the licensee stated the following corrective actions:

- Westinghouse Calculation SAE/NSE-C-NEU-0101, "Millstone 3 Charging Pump Makeup Required for Spray Bypass Line Break," dated September 10, 1997, confirmed the required flow for 0.375-inch break/leak to be 127 gpm. The postulated 0.375-inch leak occurs at normal operating conditions and therefore, the normal operating pump curve is applicable. Post-modification tests conducted in accordance with Design Change Record (DCR) M3-96077, Rev. 0, "ECCS Orifices and Throttle Valves," confirmed that the charging pumps will provide a 240-gpm flow at 2235 psig.

- FSAR Change Request (FSARCR)-96-MP3-4, dated February 28, 1998, was issued to revise FSAR Section 6.3.3.2 by removing the statement regarding the 0.375-inch leak and placed this requirement in FSAR Section 9.3.4.1 (Chemical and Volume Control System).

The second example of this violation involved FSAR Section 9.4.3.1, item 12, which stated that air flow to the auxiliary building shall be maintained from the least contaminated to the more contaminated spaces. However, in winter alignment, air is recirculated from the potentially more contaminated charging pump rooms to less contaminated areas. As a corrective action, the licensee issued FSARCR-MP3-590, dated December 2, 1997, to revise Section 9.4.3.1 by adding the word "typically" to item 12 and including the phrase "except during winter mode alignment."

The team reviewed these corrective actions and determined that the licensee's response to the violation and corrective actions taken were acceptable.

(c) Failure to Follow Procedures

(Closed) VIO 50-423/97-206-2, "Violation for Failure to Follow Procedures in Accordance With TS 6.8.1." This violation identified three examples of failure to follow procedure. The first example involved a failure to update the Production Maintenance Management System (PMMS) and Bill of Materials (BOM) following the replacement of the charging system (CHS) relief valve 3CHS*RV8119 seating material with a material that required a new replacement frequency. In its response to this violation, the licensee stated that the PMMS and BOM for 3CHS*RV8119 would be updated to recognize the new seating material replacement frequency. The team reviewed procedure NGP 6.10, Figures 7.8A, "PMMS Update Request," dated March 17, 1998, and 7.8B, "Notification of PMMS Database Updates in the ID System or BOM," dated March 17, 1998, and verified that the PMMS and BOM for 3CHS*RV8119 were updated to recognize the new seating material replacement frequency.

The second example of the violation involved flange nuts on the Reactor Coolant Pumps (RCP) A and D seal water injection piping that were not fully engaged to their respective studs. In its response to this violation, the licensee stated that the flange studs were replaced with longer studs to allow full nut engagement in accordance with Design Change Notice (DCN) DM3-04-0095-97, "Reactor Coolant Pump Flange Studs," dated September 5, 1997, and Procedure NOQP 4.08, "Determination of Quality Control For Quality Activities," would be revised to include checking adequate thread engagement of high-strength bolting as an inspection plan attribute.

The third example of this violation involved the improper application of yellow duct tape on the stainless steel piping. In their response to this violation, the licensee stated that the tape was removed per Automated Work Order (AWO) M3-97-16361, and personnel would be retrained on the use of tapes.

The team reviewed these corrective actions and determined that the licensee's response to the violation and corrective actions were acceptable.

(d) Relief Valve Discharge Piping

(Closed) VIO 50-423/97-206-03, "Two Examples of Noncompliance with the ASME Code." This violation cited two examples in which NNECO failed to comply with the Boiler and Pressure Vessel Code promulgated by the American Society of Mechanical Engineers (ASME), with regard to relief valve discharge piping. The first example involved not calculating the stress indices for orifices 3SIH*R038, 039, and 041. The licensee stated in its violation response that this calculation would be revised to provide the appropriate documentation. The team reviewed Calculation 12179,NP(B)-X10700, CCN 7, "Containment Structure Annulus Piping - ASME Class 1 & 2," and verified that the calculation was revised to address the stress indices for the orifices.

The second example of this violation involved sizing the discharge piping for CHS relief valves 3CHS*RV8119 and 3CHS*RV8123 in accordance with ASME NC-3677. The team reviewed Calculation NSP-193-CHS, Rev. 0, Change 1, "CHS System Relief Valve Setpoints," and

Technical Evaluation M3-EV-98-0107, Rev. 0, "Review of QA Category 1 Relief Valves with Common Discharges." After further review, the team concluded, based on discussions with the staff, that the discharge piping for relief valves 3CHS*RV8119 and 3CHS*RV8123 was sized in accordance with Code requirements. The team, therefore, withdrew the second example of VIO 50-423/97-206-03.

(e) Valve Actuators Not in Environmental Qualification (EQ) Program

(Closed) VIO 50-423/97-206-06, "Violation of 10 CFR 50.59 for Failure to Have Certain Main Steam Valves in the EQ Program." This violation addressed the fact that the actuators for Motor-Operated Valves (MOVs) 3MSS* MOV18A, B, C, and D were inappropriately removed from the licensee's EQ program. The licensee stated in its response that these MOVs would be placed back into the EQ program before entering Mode 4, and that the Design Control Manual (DCM) would be revised to ensure that specification changes are considered a change to design bases. The team also reviewed DCN DM3-00-0061-98, "Reinstate 3MSS* MOV18A, B, C, D in EQ Program," dated February 4, 1998, and FSARCR 98-MP3-11, dated February 6, 1998, and verified that the actuators for 3MSS*MOV18A, B, C, and D were reinstated into the EQ program. The team also reviewed Chapter 6 of the DCM, Rev. 6, Change 4, and verified that it was revised to require the preparer to review the specification to see if it affects the licensing and/or design bases. The team reviewed these corrective actions and determined that the licensee's response to the violation and corrective actions was acceptable.

(f) Three Examples of Inadequate Corrective Action

(Closed) VIO 50-423/97-206-12, "Inadequate Corrective Action to Correct Identified Discrepancies in Accordance with 10 CFR Part 50, Appendix B, Criterion XVI." This violation identified three examples of inadequate corrective actions for deficiencies identified by the licensee. Specifically, the examples involved the failure to restrict the use of Teflon tape, and failure to fully revise FSAR Table 6.3.3 and FSAR Section 7.3.1.5. In its response, the licensee discussed the following corrective actions:

- Remove the Teflon tape from the charging pump gearbox thermowell and visually examine the remaining charging pumps for any inappropriate application of Teflon tape.

- Review maintenance and operations procedures for reference to the use of Teflon tape, issue instructions prohibiting the use of Teflon tape on components exposed to high radiation, and communicate management's expectations regarding the use of Teflon tape.

- Identify and evaluate the acceptability of any Teflon material to remain installed in the plant.

- Revise FSAR Table 6.3.3 and FSAR Section 7.3.1.5.

The team reviewed Evaluation M3-EV-98-0080, "Evaluation of Teflon Tape on Threaded Connections," dated April 16, 1998, building zone walkdown results (M3-98-7420, M3-98-7421, M3-98-7475, M3-98-7476, and M3-98-7477) and the completed AWOs for removal of the Teflon tape (M3-98-7690, M3-98-7700, M3-98-7701, M3-98-7841, M3-98-7842, M3-98-7843, and M3-98-7844) and concluded that the Teflon tape was removed or was acceptable to remain

installed. The team also reviewed FSAR Table 6.3.3, dated October 1997, that shows that 3CHS*8804B cannot be opened unless 3SIH*8702A, 8702B, or 8702C is fully closed. In addition, the team reviewed FSAR Section 7.3, dated November 1997, and verified that an inoperable Safety Injection (SI) Pump Cooling Pump was added to the list of items that would cause the SI Pump to be inoperable. The team found these corrective actions to be acceptable.

(g) Inadequate Procedures

(Closed) VIO 50-423/97-206-13, "Failure to Have Adequate Procedures in Accordance with Technical Specification 6.8.1." This violation identified five examples of inadequate procedures.

- Surveillance Procedure (SP) 3608.4, "High Pressure Safety Injection System Vent and Valve Lineup Verification," failed to verify the position of 13 unlocked/unsecured charging injection flow path valves as required by Technical Specification (TS) 4.5.2.b.2.

- Procedure SP 3604A.5, "Chemical and Volume Control System Valve Operability Test," failed to include a prerequisite that the isolation valves for the alternate charging path should only be tested during cold shutdown when normal charging is isolated.

- Procedure 3604C.1, "Borated Water Source and Flow Path Availability Verification," Rev.7, Change 2, stipulated the use of inaccurate wide range Reactor Water Storage Tank (RWST) level instruments to demonstrate TS compliance for minimum RWST level.

- Annunciator response procedure Operations Procedure (OP) 3353.MB1C, "Main Board 1C Annunciator Response," incorrectly directed the operator to secure the Train A SI pump for a Service Water (SW) system rupture with SI actuated.

- OP 33353.MBIC provided inadequate guidance for verifying that SW flow would be adequate to reestablish safety-grade cold shutdown.

In its response to this violation, the licensee stated the following corrective actions:

- Revise Procedure SP 3608.4 to include the missing valves and thereby restore compliance with the TS.

- Revise Procedure SP 3604A.5 to include a specific step noting the mode restraint.

- Issue an engineering evaluation to resolve outstanding questions related to the bases for TS-required RWST volumes.

- Revise OP 3353.MB1C and OP 3353.MB1B, "Main Board 1B Annunciator Response," to include a consistent procedure response for operation of the SI system following initiation.

- Revise Procedures OP 3353.MB1C and OP 3353.MB1A, "Main Board 1A Annunciator Response," to include a requirement to verify adequate SW flow.

The team verified that the licensee revised Form 3608.4-2, Rev. 4, Change 2, to add the missing valves, and SP 3604A.5, Rev. 10, to include a step noting the mode restraint in the

"Applicability" section of the procedure. The team also reviewed engineering evaluation M3-EV-970256, "Determination of the Need to Apply Instrument Uncertainty to TS LCO Volume Surveillance Procedure Acceptance Criteria," Rev. 0, dated November 25, 1997. The analyses concluded that the bases for the RWST volume with respect to TS minimum volume had adequate margin when uncertainties introduced by the wide range instruments were included. This evaluation demonstrated that procedure changes were not required to achieve TS compliance. This was acceptable to the team. The team also verified that the licensee revised OP 3353.MB1C, Rev. 1, Change 6, and OP 3353.MB1B, Rev. 1, to include a consistent procedure response for operation following SI initiation. In addition, the team verified that the licensee revised OP 3353.MB1C, Rev. 1, and OP 3353.MB1A, Rev. 1A, to include a requirement to verify adequate flow following system restoration. The revised procedures included a requirement to perform a surveillance for heat exchanger fouling per SP 3626.13, "Service Water Heat Exchanger Fouling Determination," Rev. 6, Change 1, if the alarm is received. The licensee planned to review additional annunciator response procedures in the future to identify if similar types of discrepancies existed. The team concluded that these corrective actions were adequate.

(h) Air in RSS Piping

(Closed) VIO 50-423/206-20, "Apparent Violation of 10 CFR Part 50 Criterion XVI for Failure to Identify and Take Corrective Actions for Air in the RSS Piping." This violation involved air in sections of the RSS piping being injected into the suction of the charging and SI pumps. The licensee stated in its response that the following corrective actions would be implemented:

- Evaluate if the air in the system rendered the charging or SI pumps inoperable.
- Perform an Integrated System Function Review (ISFR) to evaluate the dynamic interactions that take place between various systems during an accident scenario. This review would evaluate if other sections of RSS piping could contain air and if vents or revision of system fill procedures were needed to ensure that the air is removed.
- Discuss this issue with Unit 3 Engineering Design and Technical Support personnel.
- Perform a supplemental review of FSAR Chapters 3 and 6 focusing on critical Architect Engineer/Nuclear Steam System Supplier (NSSS) interfaces to ensure compatibility.
- Screen NRC Information Notices (INs) to ensure that the INs that impact Millstone Unit 3 system functional requirements have been evaluated.

The team reviewed Engineering Calculation 12179-NP(B)-163-FA, "Water Hammer Analysis of Recirculation Spray System," Rev. 3, CCN 3, and concluded that water hammer loading on the RSS piping that contained air was acceptable. The team also reviewed TM-1876A, "Transport of a Small Air Pocket," dated February 1998. This report evaluated if the air in the RSS would adversely affect the charging and SI pumps and concluded that the air pockets would not affect operation of the pumps. The team agreed with the calculation and report conclusions.

Although the system was operable with air in the RSS piping, the licensee planned to maintain the RSS piping full of water. The team reviewed the procedures for filling the RSS thermal expansion loops and maintaining them adequately filled. Procedure SP 3606.5(6),

"Containment Recirculation Spray System Train A (B) Valve Lineup and Loop Seal Verification," Rev. 3, now requires that the water levels in the four thermal expansion loops be checked monthly using ultrasonic testing and filled if necessary.

The licensee's corrective actions in response to the NRC identified potential for vapor binding of the CHS and SI pumps included an expansion of the CMP scope to evaluate, in an integrated manner, the potential for unanticipated adverse system interactions (the ISFR discussed above). The team reviewed report, 3 EAR-97-043, "Integrated System Functional Review for Millstone Nuclear Power Station, Unit 3," dated January 5, 1998. This report included a review of about 25 systems that would be actuated to mitigate the consequences of a small-break Loss-of-Coolant-Accident (LOCA). (This event was chosen because it results in the actuation of a majority of the accident mitigation systems.) This review resulted in 14 Condition Reports (CRs). Examples of conditions identified involved RWST level indication freeze protection, small pockets of air in RSS piping, and volume control tank boundary valve leakage. The team reviewed Surveillance Procedure 3608.4,"Safety Injection System Vent and Valve Lineup Verification," Rev. 2, Change 2, and verified that the small pockets of air in the RSS system to the charging and SI pumps were filled. The team reviewed the corrective actions assigned to the remaining CRs and concluded that the corrective actions were acceptable. The team concluded that the ISFR conducted by the licensee acceptably addressed the identified vapor binding issue and provided confidence that accident mitigation systems would perform their specified functions.

The team evaluated a supplemental review (performed by the licensee) of FSAR Chapters 3 and 6 to identify any discrepancies or inconsistencies in the two chapters. The licensee's review did not identify any major discontinuities in the interfaces between Chapters 3 and 6; however, the review did identify numerous minor discrepancies or inconsistencies within these chapters. The team agreed with the licensee that the discrepancies identified were minor in nature and concluded that the discrepancies did not call into question the operability of the associated equipment.

In response to NRC concerns about the thoroughness with which industry operating experience contained in INs had been reviewed, the licensee conducted an IN Screening Project. In the initial phase of the project, an expert panel made up of individuals of different technical disciplines briefly reviewed all INs from 1979 to the present to identify those that were applicable to one or more of the Millstone units and assigned each applicable IN to the appropriate department for an in-depth review. This in-depth review to be conducted by the individual departments was scheduled to be completed after startup of Unit 3. The expert panel also screened INs that could potentially affect interactions between safety systems and gave the ISFR team a list of those INs that had any potential for affecting system interactions. In total, the expert panel provided the ISFR team with some 420 INs that had the potential to impact system interactions. The ISFR team reviewed all 420 INs for immediate safety concerns. Of the 420 reviewed by the ISFR team, 120 were determined to require an in-depth review by the ISFR team.

On the basis of its review, the ISFR team generated three CRs related to INs. The team verified the completeness of the screening done by the expert panel, verified that ISFR team reviewed all INs referred to it, and evaluated selected findings of the ISFR team's review (including CRs MP3-98-1702, 1723, and 1724). The team concluded that the rescreening of INs was satisfactory to address the NRC's concerns associated with system interactions.

(i) Failure to Vent Emergency Core Cooling System (ECCS) Piping

(Closed) VIO 50-423/97-206-21,"Failure to Vent Piping in Accordance with TSs." TS 4.5.2.b.1 requires that in Modes 1 through 4, ECCS piping must be verified to be full of water by venting ECCS pump casings and accessible discharge high points every 31 days. Contrary to this requirement, Valve 3SIL*V992 was not vented every 31 days in Modes 1 through 4 to verify that the associated piping was full of water. In response to this violation, the licensee stated that the valve 3SIL*V992 was added to the monthly ECCS venting surveillance procedure SP 3610A.3, "RHR System Vent and Valve Lineup Verification." The team reviewed Rev. 2, Change 2, to SP 3610A.3 and verified that valve 3SIL*V992 was required to be opened every 31 days. The team concluded that this corrective action was acceptable.

(j) Failure to Leak Check RWST Boundary Valves

(Closed) EEI 50-423/97-206-22, "Apparent Violation of TS 6.8.4 and 10 CFR 50.55a for Failure to Leak Check RWST Boundary Valves." This issue involved failure of the TS-required program to account for potential valve leakage from systems which could carry radioactive water to the RWST following a postulated accident. In a letter dated March 9, 1998, the NRC exercised discretion for EEI 50-423/97-206-22 and did not issue a violation because this issue had previously been identified by the licensee.

The team reviewed AWOs M3-98-2208, M3-98-4495, and M3-98-4497, and verified that valves 3CHS*MV8511A/B, 3CHS*MV8512A/B, and 3CHS*V261 were seat checked. The team also reviewed calculation NUC-177, "MP3-Post-LOCA Radiological Impact of Sump Coolant Backflow into the Refueling Water Storage Tank," Rev. 0. The results of this calculation indicated that the additional release from the RWST caused by boundary valve leakage would not result in an increase of the consequences in excess of the values specified in the 10 CFR Part 100. The team found these calculations acceptable.

The licensee concluded that the leakage path into the RWST was an Unreviewed Safety Question (USQ) and, on May 7, 1998, submitted a Proposed License Amendment Request to the NRC to address back leakage into the RWST.

(k) Inadequate Procedures for Response Time Testing

(Closed) VIO 50-423/97-209-01," Inadequate Procedures for RTD Response Time Testing." This violation identified two examples of inadequate procedures. Resistance temperature detector (RTD) response time testing Procedure SP 3443E12," Protection Set 1 RCS Narrow Range RTD Time Response," failed to determine the acceptability of the data before using it, and did not increase the results by 10 percent to account for the tolerance of the analysis method as specified in the test equipment vendor manual. In their response to this violation, the licensee committed to revise the procedures. The team reviewed Procedure SP 3443E12, Rev. 3, and verified that the revision incorporates steps for evaluating data acceptability and to include a 10 percent margin for the measured values, as specified by the test equipment vendor manual. The team concluded that this corrective action was acceptable.

(I) RSS Heat Exchanger Testing

(Closed) EEI 50-423/97-209-02/(Closed) VIO 50-423/98-211-01, "Technical Specification Leakage Monitoring Program." This issue involved the licensee's failure to monitoring SW leakage from the RSS heat exchangers, and was identified as an apparent violation being considered for escalated enforcement. After receiving the licensee's written response on this issue, dated May 6, 1998, and after holding an enforcement panel to reconsider the issue, the NRC determined that this violation is classified as Severity Level IV. The violation is cited in the Notice of Violation accompanying this inspection report. As stated in the Notice of Violation, no additional written response to this violation is required. The corrective action in response to the violation was reviewed by the team. TS 6.8.4.a(2), in part, requires that leakage from the RSS components located outside of the containment be monitored by performing an integrated leak test at refueling cycles intervals or less. The failure to leak test the RSS heat exchangers was identified as VIO 50-423/98-211-01.

In its response to this EEI, the licensee stated that all RSS heat exchangers were tested for leakage after this issue was identified by the NRC and that the testing verified that all heat exchangers had zero leakage. The response stated that the licensee had performed an implementation review of TS Section 6.0, "Administrative Controls," including a review of program implementation effectiveness of TS 6.8.4.

The team evaluated the licensee's review of TS 6.0 that independently verified the ability of the applicable procedures to verify compliance with the TS. The results of the review are documented in Engineering Self-Assessment Report (EAR) 3CMT-98-001, dated April 3, 1998. The team concluded that the unit is currently in compliance with the TS.

TS 6.8.4 states that programs in the following shall be established, implemented and maintained: (1) primary coolant sources outside containment; (2) in-plant radiation monitoring; (3) post-accident sampling; and (4) accident monitoring instrumentation. The licensee determined that these four programs had not been fully developed, in that implementation of the programs was divided among many departments. In order to fully develop these programs and verify compliance with the TS requirements, the licensee developed four Engineering Record Correspondences (ERCs 25212-ER-98-0108, -0110, -0112, and -0117). Each of the ERCs is associated with one of the four programs mentioned above. The ERCs identify program owners and supporting departments and their respective responsibilities. The ERCs also discuss the program requirements listed in TS 6.8.4 and how these requirements are satisfied. The team reviewed the ERCs for thoroughness and for appropriate levels of management review and concluded that the licensee now meets the program requirements of TS 6.8.4.

As a long-term corrective action, the response stated that the licensee planned to develop a periodic leak testing procedure to test the RSS heat exchangers for leakage every refueling outage. The licensee also planned to revise calculation 1279-746P(R), "ECCS System Leakage Outside Containment," to reduce the allowable heat exchanger leakage from 60 cc/hr to 0 cc/hr for each heat exchanger. In addition, the licensee planned to implement administrative controls to assure organizational ownership of the programmatic requirements of TS 6.0 and to create a database to monitor ongoing compliance with TS 6.0 requirements (including those contained in Section 6.8.4).

(m) FSAR Updates

(Closed) VIO 50-423/97-209-03, "Failure to Ensure FSAR Contained the Latest Information." This violation identified three examples of FSAR discrepancies. For the first example, inadequate FSAR description of the RSS pump seal cooling, the licensee corrected the issue in FSARCR 98-MP3-72. The second example was that the description of the Steam Generator Tube Rupture (SGTR) event in the FSAR was not consistent with the Westinghouse analyses performed in accordance with WCAP-11002, "Evaluation of Steam Generator Overfill Due to Steam Generator Tube Rupture Accident," dated February 1986, and WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Turbine Rupture Accident," dated March 1986. The licensee corrected this issue in FSARCR 98-MP3-20. The team reviewed FSARCRs 98-MP3-72 and 98-MP3-20 and concluded that the corrective actions were acceptable.

The third example, LOCA doses in FSAR Table 15.0-8 were inconsistent with the results of the supporting calculations. Based on additional information provided by the licensee, the team determined that the values contained in the Table were in fact consistent with the results of the calculation. This part of the violation is withdrawn.

(n) Control Room Operator Dose

(Closed) EEI 50-423-97/209-06, (Closed) VIO 50-423/98-211-02, "Failure to Implement Adequate Corrective Actions In Accordance with 10 CFR Part 50, Appendix B, Criterion XVI." This item involved dose calculations for the control room that assumed an unfiltered in-leakage to the control room for 1 hour following a LOCA, but failed to address a single-failure vulnerability in the control room inlet damper arrangement that had the potential to result in unfiltered in-leakage to continue for more than 1 hour. The single-failure vulnerability in the damper arrangement was identified by the licensee, but was not adequately resolved when originally identified. After receiving the licensee's May 6, 1998, written response and after holding an enforcement panel to reconsider the issue, the NRC has determined that this violation is classified as Severity Level IV. The violation is cited in the Notice of Violation accompanying this inspection report. As stated in the Notice of Violation, no additional written response to this violation is required.

The team reviewed the corrective action in response to the violation. Appendix B, Criterion XVI, of 10 CFR Part 50, in part, requires that conditions adverse to quality be promptly identified and corrected. The failure to identify and correct the adverse conditions associated with a single-failure of the control room inlet damper arrangement was identified as VIO 50-423/98-211-02.

Calculation UR(B)-450, "MP-3 LOCA Doses at the Site Boundary, Control Room, and TSC Assuming Duct Leakage and Damper Bypass," Rev. 0, was performed, in part, to address this issue. This new calculation assumes that the dampers will not be realigned for up to 1 hour and 40 minutes following a LOCA. The licensee has concluded that such a delay is acceptable from a radiological standpoint and operator walkthroughs of the required actions. The team verified that procedure OP 3314F, "Control Building Heating, Ventilation, Air Conditioning, and Chilled Water," Rev. 16, Change 9, specifically requires the dampers to be repositioned. While the calculated doses for the Technical Support Center (TSC) contained in UR(B)-450 are below regulatory limits, they have increased slightly above what was previously calculated. Given that

fact, the licensee performed an operability determination that concluded systems were still able to perform their functions. However, the licensee concluded that this is a USQ and submitted the USQ to the NRC for review.

(o) Emergency Diesel Generator (EDG) Building Ventilation

(Closed) VIO 50-423/209-09, "Failure to Meet the Limiting Condition for Operation for TS 3.1.2.1 for the Train A EDG." This issue involved inadequate development and review of the safety and engineering evaluations associated with an EDG building ventilation system damper failure that occurred with a temporary modification installed in the EDG building ventilation system. In its response to this violation the licensee stated that the failed damper was replaced and procedures would be revised to improve the quality of safety evaluations. The team verified that Procedure RAC-12, "Safety Evaluation Screens and Safety Evaluations," Rev. 1, was revised to improve guidance in support of plant modifications, including specifying the training and qualification necessary for safety evaluation preparers. The team concluded that these corrective actions were adequate.

(p) Improper Installation of Temporary Modification

(Closed) VIO 50-423/97-209-10, "Inadequate Design and Installation of Temporary Modifications Contrary to 10 CFR Part 50, Appendix B, Criterion V." This issue involved the improper installation of temporary heaters in the Train A EDG building. In its response to this violation, the licensee stated that the temporary heaters were removed for the Train A EDG building and WC-10, "Temporary Modification," was revised to prompt the technical support engineer to determine if a post-installation walkdown should be performed. The team reviewed Rev. 1, Change 5, to WC-10, and verified that the procedure was updated to include guidance on the need for post-installation walkdowns for temporary modifications. The team concluded that these corrective actions were acceptable.

(q) Inadequate Temporary Modification Procedure

(Closed) VIO 50-423/97-209-11, "Inadequate Control of Calculations of Temporary Modifications Contrary to 10 CFR 50, Appendix B, Criterion V." This issue involved calculations supporting temporary modifications that were not controlled in accordance with the DCM. In its response to this violation, the licensee stated the temporary modification was removed and that design engineering personnel were trained on the use of the DCM as the applicable guidance document for preparation of calculations. The team reviewed Rev. 6, Change 4, to the DCM and concluded that it provides adequate guidance on the control and use of calculations in such situations. Therefore, the licensee's retraining of Design Control personnel which re-emphasized the importance of following the DCM when preparing or using calculations was an acceptable corrective action.

(r) Inadequate Temporary Modification

(Closed) VIO 50-423/97-209-12, "Failure to Adequately Translate Design Bases into a Temporary Modification in Accordance with 10 CFR 50, Appendix B, Criterion III." This issue involved the installation of temporary covers over the pilot valve for the C Main Steam Isolation Valves (MSIV). The temporary modification incorrectly stated that 65 psig conservatively bounded any pressure that could result from the loss of the Residual Heat Removal System

(RHR). In its response to this violation, the licensee stated that the approval of the inadequately designed temporary modification was attributed to an inadequate technical review of the modification when first proposed. The modification was approved, in part, on the basis of an internal memorandum concerning expected system pressures that was prepared for other purposes and was not an appropriate basis for the modification in question. The licensee concluded that if all of the proper disciplines had been involved in reviewing the modification, the improper use of the memorandum would likely have been recognized. The licensee's response to this violation stated that procedure WC-10, "Temporary Modifications," was revised to require that all disciplines potentially involved be included in the review process. The team reviewed Rev. 1, Change 5, to WC-10, and verified that the procedure was revised. The team concluded that this corrective action was acceptable.

(s) Final Safety Analysis Report Change Request (FSARCR) Safety Evaluations

(Closed) VIO 50-423/97-209-13, "Failure to Include Safety Evaluations with Records of FSAR Changes Where Safety Evaluations are Required by 10 CFR 50, Appendix B Criterion III." FSARCRs approved between the start of the CMP (mid-1996) and the retraining of plant personnel (mid-1997) did not include safety evaluations. In its response to this violation, the licensee stated that FSARCRs processed between mid-1996 and mid-1997 were reviewed to determine if the change created a USQ. The licensee's review concluded that no USQ existed. The response further stated that weaknesses in the area of preparing safety evaluation were previously identified during CMP and Procedure RAC 03, "Changes and Revisions to Final Safety Analysis Reports," dated March 31, 1997, was revised to include instructions to improve safety evaluations. The team reviewed these corrective actions and concluded that they were acceptable.

(t) Ventilation System Charcoal Filters

(Closed) VIO 50-423/97-210-03, "Violation for Inadequate Corrective Action." This item involved leakage of the fire protection water isolation valves that could have decreased the efficiency of the Supplementary Leak Collection and Release System (SLCRS) filter banks. In its response to the violation, the licensee stated that procedure SP 3670.4, "Routing PMs," was revised to include a weekly check for leakage past the fire protection water isolation valves. The team reviewed Rev. 16, Change 5, to SP 3670.4, and verified that the revised procedure provided instructions for checking for leakage. The team found this action acceptable.

(u) RSS Miniflow Valve Modification and Auxiliary Building Fan Logic Deficiencies

(Closed) VIO 50-423/97-210-04, "Violation for Inadequate Design Control." This issue involved two examples of design deficiencies. The first example involved a modification to the RSS that introduced the potential for cycling the RSS miniflow valves on a false low-flow signal. In its response to this violation, the licensee stated that miniflow valve control circuits were modified such that a false low-flow signal during an accident would not result in opening or cycling the miniflow valves and consequently diverting the flow. The team verified that the circuitry was modified to eliminate the potential for cycling based on its review of DCNs DM3-00-0079-98, "3RSS*MOV38A Logic Change," dated February 2, 1998, and DM3-00-0080-98, "3RSS*MOV38B Logic Change," dated January 28, 1998.

The second example of this violation involved the control circuitry for Auxiliary Building ventilation fans 3HVR*FN14A&B and 3HVR*FN13A&B. After a loss of power, the potential existed that the fans could fail to restart on restoration of power because of a relay race that could occur depending on fan coast down time. The team reviewed testing accomplished on January 22, 1998, that demonstrated the acceptability of the existing control circuit. The licensee's response to this violation stated that quarterly trending of fan coast down would be performed to periodically monitor the associated flow switch performance. The team reviewed the AWOs (M3-98-04309 through 04312) that accomplished the monitoring program, and concluded that the corrective actions were adequate.

(v) Inadequate Ventilation Procedures

(Closed) VIO 50-423/97-210-09, " Inadequate Procedures." This issue involved inappropriate substitution of manual operator action for automatic action, and failure to maintain the RSS as a closed system outside of containment. The licensee stated in its response to the violation that OP 3314A, "Auxiliary Building Heating Ventilation and Air Conditioning," and OP 3260, "Conduct of Operations," were revised to clarify the use of manual operator action. The response stated that operating and surveillance procedures were revised to identify the RSS seal head tank vent valves as containment boundary valves. The team reviewed OP 3314A, Rev. 21, Change 6, and OP 3260, Rev. 10, Change 4, and verified that the procedures were appropriately revised to require an approved safety evaluation before substituting manual operator action for automatic action. The team also reviewed SP 3606.1, "Containment Recirculation Pump 3RSS*P1A Operational Readiness Test," Rev. 11, Change 5, and OP 3306, "Containment Recirculation Spray System," Rev. 7, Change 2, and verified that these valves were maintained locked shut. The team concluded that the corrective actions were acceptable.

(w) RSS Sump Deficiencies

(Closed) VIO 50-423/98-205-02, "Failure to Implement Adequate Corrective Action, Two Examples." This issue involved the licensee's failure to annotate Maintenance Form 3704A-727 work instructions in AWOs for the removal/installation of containment sump cover plates, and failure to adequately perform periodic containment sump inspections. In its response to the violation, the licensee stated that the containment sump was assigned an identification number to track preventive maintenance items assigned to the sump, and that procedure SP 3612A, "Containment Inspections," was revised to specifically address containment sump internal inspection. The team reviewed AWO M3-95-12527 that was issued to remove/reinstall the containment sump cover plates, and verified that the AWO referenced Maintenance Form 3704A-727 work instructions. The team also reviewed Rev. 13, Change 4, to SP 3612A, and verified that containment sump inspection requirements were adequately addressed. The team concluded that the corrective actions were acceptable.

(x) Relay Setting Documents Not Revised

(Closed) VIO 50-423/98-205-03, "Failure to Implement Adequate Corrective Action." This issue involved the licensee's failure to properly update drawings and implement a procedure change associated with Quench Spray System (QSS) pump relay settings. In its response to this violation, the licensee stated that the applicable drawings were revised, and Revision 2 to the Control Document Library copy of Specification SP-EE-321 (the official record copy) was

properly entered. The team reviewed DCN DM-00-0210-98, "Drawing Discrepancies MFR DWGS 4.16KV SWGR 3ENS*SWG-A/B," dated February 28, 1998, and drawings 25212-31525, sheet 6; 2512-39010, sheet 418; and 25212-39010, sheets 447 and 460, and verified that the drawings were revised to indicated relay settings. The team also reviewed the Control Document Library copy of SP-EE-321 and verified that Revision 2 was properly entered. The team concluded that these corrective actions were adequate.

2.2 Unresolved Items and Inspector Followup Items

2.2.1 Scope of Review

The team reviewed Unresolved Items (URIs) and Inspector Followup Items (IFIs) identified in NRC ICAVP IRs 50-423/97-206, 50-423/97-209, 50-423/97-210, and 50-423/98-205. The team evaluated the additional information provided by the licensee and, for URIs, determined if a violation of NRC requirements occurred.

2.2.2 Findings

In reviewing the URIs and IFIs identified in the inspection reports, the NRC team reviewed 22 open items, as discussed in the following paragraphs.

(a) Charging System Check Valves

(Closed) URI 50-423/97-206-04, "URI to Resolve Check Valve Single Failure Requirements." This item involved a single active failure of a check valve that has the potential to incapacitate CHS. The licensee issued CR M3-97-2140 that led to its review of the licensing and design bases for the CHS. This review concluded that Unit 3 was licensed with single check valves for CHS pumps and, hence, the single active failure of these valves was outside of the licensing bases. The licensee's conclusion was based, in part, on the Westinghouse position that this is a standard NSSS design arrangement and the failure modes and effects analysis, documented in Chapter 6 of the FSAR, did not address failure of these check valves. The team concurred with the licensee's evaluation regarding the single active failure of the CHS pumps. The licensee planned to revise the FSAR to clarify that the check valve active failure exemption is only applicable to those valves within the NSSS vendor's original scope of design, and does not obviate the need to verify the operational readiness of check valves in accordance with plant TSs and the Inservice Test (IST) Program as applicable. The team concluded that this issue was not a violation of NRC requirements.

(b) RSS Sump Isolation Valves

(Closed) URI 50-423/97-206-05, "URI Concerning Qualification of RSS Isolation Valve Seats." This item involved qualification of material used for the containment isolation valve seats. The licensee issued CR M3-97-4621 that resulted in evaluations M3-EV-97-0319, "Technical Evaluation for Mechanical Equipment Environmental Qualification of Valve Seat Material for 3RSS*MOV23A, B, C and D," Rev. 0, and M3-EV-98-0004, "Technical Evaluation for Mechanical Equipment Environmental Qualification of BUNA-N as the Soft Material for Containment Isolation Valves 3RSS*MOV20A, B, C, and D and 3QSS*MOV34A and B," Rev. 0. These evaluations documented the acceptability of the qualification for these elastomer materials for its intended use. The team reviewed these reports and concurred that the

material used for soft seating is adequate to withstand the projected environmental conditions of service. The team also verified that the licensee is tracking the periodic replacement of these seats. The team concluded that this issue was not a violation of NRC requirements.

(c) Voltage Surge Protection

(Closed) URI 50-423/97-206-07, "URI Concerning the Adequacy of the Unit's Voltage Surge Protection." This issue involved the surge arresters on the 345kv side of the Reserve Station - Service Transformer (RSST). The team reviewed report MP3-15G-235AA, "Surge Transfer from the HV Winding to the LV Winding of RSST-A," that was developed by Stone & Webster and provided to the licensee on September 17, 1997. This report stated that the electromagnetically transferred surge to the switchgear is less than its withstand rating, that transformer withstand voltage and motor dielectric test voltages are greater than the surge voltage, and that no common failure modes exists. The team agreed with the report's conclusions and determined that this issue was not a violation of NRC requirements.

(d) 1-Hour Battery Connection

(Closed) URI 50-423/97-206-09, "URI Concerning 1-Hour Battery Connection." This issue involved a battery calculation for conditions during a SI actuation. The team reviewed BAT-SYST-1240E3, "Battery Capacity Calculation," Rev. 1, Change Notice 2, that added Table D1 for use in the calculation. Table D1 shows the order and magnitude of the breaker loads for the first minute of a containment depressurization actuation combined with a loss of offsite power (LOOP). It also demonstrates that such an event is the most limiting event and that the operation of the additional breakers does not affect the peak loading that is credited in the first minute of the event. The team agreed with the revised calculation conclusions and determined that this issue was not a violation of NRC requirements.

(e) Thermal Expansion of Containment Steel

(Closed) URI 50-423/97-206-17, "URI Concerning the Effects of Thermal Expansion of Rigidly Restrained Steel." This issue involved thermal expansion of steel structures in Containment that were not evaluated for Containment post-accident temperature that could be as high as 280 °F. The team reviewed Calculations S53.26, "MP3 - Misc. Platform Containment Structure," Rev. 1; 12179-NM(B)-127-JAK, "RHR Line Pipe Rupture Restraints and Cubicle Space Frame," Rev. 2, Change No.11; and S50.1044, "MP3 - Incore Instrumentation Platform," Rev. 1. In all cases, the team found that thermal growth effects did not cause loss of function and determined that this issue was not a violation of NRC requirements.

(f) Steam Generator Blowdown Isolation Valves

(Closed) URI 423-97209-07, "Verification That Appropriate Operator Actions are Defined to Isolate Steam Generator Blowdown in the Event the Automatic Isolation Valves Fails." This issue involved how the plant could assume the single failure of a blowdown isolation valve following a SGTR and still isolate the affected steam generator using safety-related valves. In response to this issue, the licensee questioned Westinghouse about the assumptions made for an SGTR event. In part, Westinghouse replied that taking credit for manual or automatic isolation of nonsafety-grade valves on small lines in non-critical areas is consistent with the analysis methodology. The team verified that Emergency Operating Procedure (EOP) 35 E-2,

"Faulted Steam Generator Isolation," Rev. 8, is consistent with this methodology in that Attachment A calls for the isolation of specific nonsafety-related manual blowdown isolation valves if the automatic isolation valves fail to shut. After reviewing the procedure and walking down the lines in question to ensure the accessibility of the valves, the team was satisfied that the valves could be used to isolate blowdown. The NRR Reactor Systems Branch agreed that the use of these nonsafety valves following an SGTR was acceptable. The team concluded that use of the valves as a backup means of isolating blowdown was acceptable and that this issue is not a violation of NRC requirements.

(g) Safety Evaluations for FSAR Changes

(Closed) URI 50-423/97-209-16, "Disagreements Between the FSAR and Calculations and the Need for Safety Evaluations to be Performed." This issue involved three examples where it may be necessary for the licensee to perform safety evaluations for differences found between the FSAR and calculations. In the example associated with the offsite dose calculation results contained in Section 15.0-8 of the FSAR, the team determined that the FSAR values were consistent with those of the calculations. In the other two cases (iodine spike dose calculations and SGTR operator response times), the team determined that the appropriate safety evaluations were done in support of changes made to the FSAR. The team therefore concluded that these issues did not violate NRC requirements.

(h) Seismic and Fluid Piping Loads

(Closed) URI 50-423/97-210-06, "Updating of FSAR On Use of Square Root Sum of Squares Modeling." This issue involved the licensee's design practice of combining seismic loads and fluid transient loads by the square root of the sum of the squares (SRSS) method for piping analysis and pipe support design that is not consistent with FSAR commitments to combine these loads by absolute sum. The team concluded that the licensee's use of the SRSS methodology was acceptable, but needed to be consistent with FSAR commitments. The licensee issued CR-M3-98-1370 on March 11, 1998, to address this issue and the corrective action for the CR required that the FSAR be revised to address the load combinations for pipe supports in addition to the load combinations for piping analysis. The team concluded that this was acceptable and determined that this issue was not a violation of NRC requirements.

(i) Seismic Qualification of the RSS Heat Exchangers

(Closed) URI 50-423/97-210-07, "Review of Revised Calculation for RSS Heat Exchangers." This issue involves the licensee's analysis of the RSS heat exchanger primary supports. In particular, the analysis did not document the vertical rigidity of the heat exchanger supports or the loads on those supports induced by the differential lateral displacements of the containment structure as a result of LOCA pressures and temperatures or the safe shutdown earthquake (SSE). The team reviewed Calculation 12179-NM(S)-748-CZC-004, "Support and Restrain Design for Containment Recirculation Coolers 3RSS*E1A, B, C, D," Rev. 3, CCN 4, and Calculation 12179-NM(B)-570-HBG, "Heat Exchangers 3RHS*E1A, B Supports and Restraints," Rev. 1, CCN 10. The CCNs to these calculations demonstrate that the SSE displacements of the containment structure induce negligible differential lateral displacements at the support locations of the RSS and RHR heat exchangers. The team agreed with the results of these calculations and determined that this issue is not a violation of NRC requirements.

(j) SLCRS Fan Qualification

(Closed) URI 50-423/97-210-08, "Review of Justification for Design Value and Embed Calculation." This issue involves the use of ZPA g-values to qualify the support for skid-mounted SLCRS fan 3HVR*FN12A. Specifically, these g-values were lower than the corresponding g-values specified in Design Specification 2176.30-141 to seismically qualify the fan. The team reviewed ERC 25212-ER-98-0163, "NETM-59 Structural Final Load Verification Program," Rev. 0, which demonstrated that the NETM-59 sample of embedded plates with duct support attachments had been appropriately qualified and bounded the existing population of duct support embedded plates. The licensee planned to locate and reproduce the complete listing of the remaining embedded plates included in the NETM-59 population after the unit is started up. The team concluded that this is a documentation issue and has no impact on the sample load reconciliation or qualification status of the embedded plates. The team concluded that the completed and scheduled action was adequate and determined that this issue is not a violation of NRC requirements.

(k) Hydrogen Analysis Indicators

(Closed) IFI 50-423/97-206-10, "Verify Labeling of Hydrogen Concentration Indicators." This issue involved the lack of labeling on the battery room hydrogen concentration indicators. The team walked down the hydrogen concentration indicators and verified that plastic tags were installed to identify each analyzer and that the units of measurement for hydrogen analyzer instrumentation was in percent hydrogen.

(l) Correction of Calculation Errors

(Closed) IFI 50-423/97-206-14, "Correct Calculation Errors in Accordance with CR M3-97-3169." This issue involved several minor errors in the RWST setpoint calculations. The team reviewed Calculation 34351B03-01232E3, "RWST Level Interlock Channel Calibration," Rev. 0, CCN 1, and verified that the minor errors were corrected.

(m) Pressurizer Level Discrepancies

(Closed) IFI 50-423/97-206-18, "Adequacy of the Licensee's Proposed Change to the Bases of the Pressurizer Level TS." This issue involved an inconsistency between a TS and an FSAR transient accident analysis. To resolve this issue, the licensee revised OP 3206, "Plant Shutdown," Rev. 7, Change 1, to prohibit increasing the pressurizer level above 40 percent, below 10 percent power. On March 10, 1998, the licensee also submitted for NRC review a proposed TS revision that replaces the pressurizer maximum water level inventory requirement with a pressurizer level requirement. The team concluded that once this TS amendment is approved, the TS and the FSAR transient/accident analysis will be consistent.

(n) TS Area Temperature Monitoring Requirements

(Closed) IFI 50-423/97-206-19, "Resolution of Charging Pump Area Temperature Inconsistencies." This issue involved charging pump cubicle temperature monitoring requirements specified by TS 3.7.14. The team reviewed OPS Form 3670.2-3, Rev. 8, and verified that the procedure was revised to ensure that area temperature monitoring adequately addresses possible equipment inoperability and EQ concerns, and that appropriate actions

would be taken if established limits were exceeded. The licensee plans to modify the Bases for TS 3.7.14 to clarify that the TS basis is equipment operability, not EQ as presently discussed in the Bases.

(o) Minimum Short-Circuit Currents not Evaluated

(Closed) IFI 50-423/97-209-05, "Unanalyzed Low-Level Fault Current Review." This issue involved a potential unanalyzed condition in that Calculatic BAT5-96-1247E3, "Potential for Cable Ignition for Battery 5 and Charger," did not evaluate short-circuit currents. The licensee developed a preliminary calculation that demonstrated satisfactory coordination for minimum fault current on the circuit supplied by Panel 3BYS-PNL-5AF. The licensee planned to revise Calculation BAT5-96-1247E3 to include the analysis of minimum short-circuit current and assessment of protective device actuation. The team agreed with the licensee's conclusion.

(p) Upgrade Safety Evaluation Procedure

(Closed) IFI 50-423/97-209-14, "Licensee to Upgrade Procedure NGP 3.12, Safety Evaluations" This issue involved a significant revision to the safety evaluation and screening procedure to correct identified weaknesses. Procedure NGP 3.12 was canceled and a new procedure, RAC 12, "Safety Evaluation Screens and Safety Evaluations," was issued March 1, 1998. The team reviewed the new procedure and concluded that NRC concerns had been addressed.

(q) Operator Actions Following an Auxiliary Feedwater (AFW) Pipe Break

(Closed) IFI 50-423/97-209-15, "Verification of Appropriate Operator Actions Following an Auxiliary Feedwater System Pipe Break Upstream of the Cavitating Venturi." The question was whether additional operator actions would be required to prevent the AFW pumps from running out following a pipe break in the steam generator AFW piping between the cavitating venturis and the check valves upstream of the cavitating venturis. The licensee performed a calculation that demonstrated that the AFW pumps will not runout following a pipe break between the cavitating venturis and check valves. The team reviewed Change Notice 4 to Calculation 96-067, "MP3 AFW Comprehensive Flow Analyses," Rev. 0, and concluded that operator action was not required to prevent the AFW pumps from running out following a pipe break.

(r) Pump Test Acceptance Criteria

(Closed) IFI 50-423/97-210-01, "Maximum Acceptance Value for Pump IST." This item involved a discrepancy in the maximum allowable design pump curves, as developed in design-basis calculations for QSS and RSS pumps, that were lower than the maximum allowable surveillance curve provided in the IST procedures. The team reviewed Technical Evaluation M3-EV-98-0110, "Assessment of Current Performance of Pumps in the IST Program Compared Against Their Pump Test Curves - Millstone Unit 3," Rev. 0. This evaluation identified that both the QSS and RSS pumps maximum flows are enveloped by the 5 percent allowance for maximum pump flow in Calculation US(B)-362, "Containment Recirculation System (RSS) Suction Hydraulic Analysis, Including Debris Transport, Water Holdup, and Available NPSH," Rev. 0. The team agreed with the conclusions in the Technical Evaluation.

(s) Operator Response Time

(Closed) IFI 50-423/97-210-02, "Change of Switchover Time from 10 to 25 Minutes." The operator response time assumed in Calculation US(B)-295, "RWST Draw-down Rates and Switch Over Levels," Rev. 6, was changed in Rev. 7. The team reviewed Revision 7 and found it to be acceptable.

(t) Radiation Monitor Flood Vulnerability

(Closed) IFI 50-423/97-210-05, "Vulnerability of Radiation Monitors to Flooding." This issue involved a concern that the concrete pit containing the radiation detectors used to monitor RSS leakage into the SW system were susceptible to in-leakage of rainwater, and that this might indicate that the drain line shown on original construction drawings was either not installed as shown, or was ineffective. The licensee inspected the pit and found that the drain inlet had been covered with concrete during original construction. The licensee subsequently cleared the drain opening per AWO M3-98-01981. The AWO recorded that the pit now drains freely

(u) Operator Training on RSS Modifications

(Closed) IFI 50-423/97-210-10, "Verification of Operator Training on RSS Modifications." This issue involved modification of the simulator to reflect RSS modifications before completion of operator training on the RSS modifications and a draft version of EOP ES-1.3, "Transfer to Cold-Leg Recirculation." The team discussed the simulator modification with a training instructor, who explained that the simulator was modified to reflect recent RSS modifications before operators were trained on the modifications. The simulator computer programs were initially revised to accurately model RSS modifications, but were later enhanced to accomplish the computer functions more efficiently. The team reviewed EOP ES-1.3, Rev. 8, and verified that it was the same as the draft copy used to train operators. The team concluded that the operators were properly trained on the RSS modifications.

(v) RSS Functions

(Closed) IFI 50-423-97-210-11, "Risk Significance of RSS Functions." This issue involved classification of the risk significance of the RSS fission product and containment heat removal functions. On February 3, 1998, the Maintenance Rule Expert Panel reclassified these as risk significant functions and performance criteria were subsequently developed and implemented. The team reviewed CR M3-98-0462 that documented the results of the Maintenance Rule Expert Panel, and concluded that the RSS risk significance classification and performance criteria were acceptable.

2.3 Licensee Event Reports

2.3.1 Scope of Review

The team reviewed LERs, and verified the following:

- Licensee management assigned responsibility for implementing corrective action, including any necessary changes in procedures and practices.

- Corrective actions were appropriate and implemented.
- The licensee performed a root-cause analysis when appropriate.
- The licensee identified and corrected repetitive deficiencies.
- The licensee properly handled all operability and reportability issues related to LERs.
- When appropriate, the licensee expanded the scope of its corrective actions to include applicable related systems, equipment, procedures, and personnel actions.
- Deferred items and interim resolutions were acceptable.
- Corrective actions were scheduled to be implemented before entering the appropriate mode of operation.

2.3.2 Findings

In reviewing the LERs addressed during the ICAVP, the NRC team reviewed 19 open items, as discussed in the following paragraphs.

(a) Fuel Drop Radiation Monitor Response Time

(Closed) LER 50-423/96-046-00, "Time Response of Containment Fuel Drop Radiation Monitor Non-Conservative." This issue involved the response time for the containment radiation monitors and containment purge exhaust and supply valves to support the fuel handling accident in containment analysis. The team reviewed DCR M3-97-032, "Fuel Drop Radiation Monitors 3RMS*RIY41 & RIY42 Hardware Change," dated August 6, 1997, and verified that this hardware modification was accomplished. The team also reviewed FSARCR 97-MP3-362 and verified that the response times were changed in Section 15.7.4.2.2 and Table 6.2-65. The team also reviewed Technical Requirements Manual (TRM) Table 3.3.2-1 and verified that the table was changed to reflect the new criteria of 16 seconds or less. In addition, the team reviewed SP 3443E10, "Four Channel RPS/ESFAS Time Response," Rev. 3, Change 6, and verified that response times were tested for monitors 3RMS*RIY41 & RIY42. The team concluded that the corrective actions were acceptable.

(b) Chilled Water Pump Gauges

(Closed) LER 50-423/96-050-00, "Range of Control Building Chilled Water Pump Suction Pressure Gauges Used for Surveillance Not in Accordance With ASME Chapter XI Requirements." The licensee identified that the chilled water pump suction pressure gauges (3HVK*P126A/B) did not meet the instrument range requirements imposed by Section XI of the ASME Code. The licensee revised SP Forms 3614F.5-1, "Control Building Chilled Water Pump A Operational Readiness Test," and 3614F.6-1, "Control Building Chilled Water Pump B Operational Readiness Test," both Rev. 6, Change 1, to delete the allowed use of gauges HVK*P126A/B, and substitute the use of digital plant process computer points 3HVK*151A/B.

The team reviewed the licensee's corrective action identified in the LER and considered it adequate. The use of chilled water pump suction pressure gauges (3HVK*P126A/B) that did

not meet the instrument range requirements imposed by Section XI of the ASME Code was identified as NCV 50-423/98-211-03. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(c) Turbine Driven Auxiliary Feedwater Pump Steam Supply Valves

(Closed) LER 50-423/97-013-00, "Failure to Meet General Design Criteria 57 for the Main Steam Supply Containment Isolation Valves to the Auxiliary Feedwater Pump Turbine." The licensee identified that the instrument air operated turbine driven AFW pump (TDAFWP) steam supply containment isolation valves (3MSS*AOV31A/B/D) did not satisfy the remote manual operation requirement of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 57, "Closed System Isolation Valves."

The licensee revised the TRM, Table 3.6.3-1, Rev. 6, to reflect the shift of the containment isolation boundary to the motor-operated non-return valves, (3MSS*MOV17A/B/D) and the bypass valves (3MSS*V885, V886, and V887). Drain valves 3MSS*V900, V902, and V904 are also containment isolation valves that were addressed in the TRM. Procedure OP 3260B, "Locked Component Checklist," Rev. 3, required that the bypass and manual valves be maintained in the locked closed position. FSAR Table 6.2-65, "Containment Penetration," Amendment 403, dated February 1998, stated that the motor-operated non-return, bypass, and drain valves and six manual valves were containment isolation valves. The IST Program, ISI-3.0, Rev. 5, Change 2, added the three motor-operated non-return valves as "active" valves and required that the valves be stroke tested. Calculation NM-027-ALL, "MP3 Active Valve Response Times," Rev. 2, Change 3, stated that 3MSS*MOV17A/B/D were containment isolation valves and specified a 60-second stroke time.

The team reviewed the licensee's corrective actions identified in the LER, and considered the actions adequate. The failure to ensure that the containment isolation valve design basis had been adequately maintained was identified as NCV 50-423/98-211-04. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(d) Operation of Containment Equipment During Fire

(Closed) LER 50-423/97-020-00, "Elevated Containment Temperature Due to Reduced Cooling Capability Restricts Ability to Enter Containment and Operate Equipment Required for Cold Shutdown During Certain Fire Events." The licensee identified that an analysis had not been performed to establish the containment temperature during a Loss-of-Power (LOP) event, where a fire-safe shutdown assumed that one Containment Air Recirculation (CAR) fan was operating and the fan cooler was supplied by the Component Cooling Water (CCW) system at 95 °F. The temperature in the containment is important because operators would be required to enter the containment to open the RHR system suction valves in order to meet the requirements of Branch Technical Position 9.5-1, "(Appendix R) Compliance Report," Rev. 2.

The licensee modified the CCW system to the CAR system to provide the design-basis flow rate (622 gpm) for operation of a single CAR fan. DCN DM3-00-00620-97, "Throttling of Valves 3CDS-V76 and 3CDS-V131," dated July 28, 1997, provided mechanical limiters for throttling, and OPS Form 3330C-2, "Reactor Plant Chilled Water System Lineup (Inside CTMT)," Rev. 5, Change 1, placed valves 3CDS-V076 and 3CDS-V131 in the full open position.

ERC 25212-ER-98-008, "Transmittal of Millstone 3 Containment Heatup Results for Appendix R and Loss of Power Scenarios," Rev. 0, demonstrated a steady-state containment temperature of 144.8 °F, approximately 12 hours after initiation of plant shutdown with one CAR fan running. DCR M3-96064, "Safety Grade Cold Shutdown Analysis -DWST Inventory, CCP Temperature Limitations, and Spent Fuel Pool Cooling," Rev. 0, stated that with an LOP, 11 hours (6 hours boration plus 5 hours cooldown) would be required to reach hot shutdown (350 °F).

The team reviewed the licensee's corrective actions identified in the LER and considered the actions adequate. The failure to ensure that the containment temperature design basis had been adequately maintained was identified as NCV 50-423/98-211-05. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(e) Steam Generator Tube Rupture Analysis

(Closed) LER 50-423/97-029-00, "Design Basis Concern on Steam Generator Tube Rupture Analysis for Main Steam Pressure Relief Bypass Valves." This LER described a condition in which the minimum number of main steam pressure relief bypass valves flow paths assumed by the plant design may not have been available. In response to that issue, the licensee committed to control these flow paths in the TSs. The proposed TS change was approved by the NRC on October 2, 1997.

(f) Technical Specification Low-Temperature/Pressure Limits

(Closed) LER 50-423/97-30-00, "Non-Conservatism in the Low Temperature Overpressure/ Temperature Limit Curves in Technical Specifications." NRC IN 93-58, "Non-Conservatism in Low Temperature Overpressure Protection For Pressurized-Water Reactors," notified the licensee of this generic issue. In the LER, the licensee stated that the TS low-temperature overpressure/temperature curves would be revised, and a review would be conducted to determine if cold overpressure setpoints were exceeded. The team verified that the licensee incorporated TS Amendment 157 that revised low-temperature overpressure protection limits. The team also reviewed Calculation ENG-01476-M3, "Review if COP Cold Overpressure Setpoints Were Exceeded," Rev. 0. This calculation concluded that low temperature overpressure setpoints were not exceeded. The team concluded that the licensee's corrective actions were acceptable.

(g) RHR Valve Low-Pressure Permissive Bistable

(Closed) LER 50-423/97-031-00, "RHR Valve Low Pressure Open Permissive Bistable Setting Set Non-Conservatively." The RHR low-pressure interlock setpoint did not comply with TS 4.5.2.d.1 surveillance requirements. Specifically, the setting would allow the RHR isolation valves to be opened at pressures higher than the pressure stipulated in the TS. However, the settings were below the RHR pressure limits and relief valve settings, including instrument uncertainties. The condition was identified by the licensee and had existed since initial startup.

The licensee subsequently amended the TS (Amendment 156) to revise the setpoint from 390 psia to 412.5 psia and recalibrated the bistables to comply with the TS. The team reviewed Calculation 3442J01-01525E3, "RCS Wide Range Pressure Channel Calibration Data," Rev. 0, and verified that the calculation was appropriately revised. The team also reviewed DCN

DM3-00-1912-97, "Recalibration of 3RCS*PB403A and 3RCS*PB405A per PTSCR 3-36-97," dated March 5, 1998, as well as the implementing AWOs (M3-98-01459 and M3-98-01457) and verified that the setpoint revision was implemented. The team determined that the licensee's corrective actions, identified in the LER, were adequate. The failure to maintain the RHR low-pressure interlock consistent with the TS was identified as NCV 50-423/98-211-06. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(h) Containment Drain Sump Pump Timer Relay Calibration

(Closed) LER 50-423/97-032-00, "Calibration of Pumped Capacity Monitoring System Timers not in Accordance with 18 Month Surveillance Requirement of the Technical Specifications." TS 4.4.6.1.b requires that the licensee calibrate the Containment Drain Sump Level and Pumped Capacity Monitoring System every 18 months. This LER identified that a portion of the system, specifically, the sump pump timer relays, was calibrated on a 3-year frequency instead of the required 18-month frequency. The licensee revised the procedure to require calibration at the specified frequency. The team reviewed GTS PT 31459A, "MP3 Timing Device Calibration Program," Rev. 6, and verified that the timers were required to be calibrated at the specified 18-month frequency.

The team also reviewed the licensee's corrective actions identified in the LER and considered the actions adequate. The failure to calibrate the containment drain sump pump timer relays in accordance with the 18-month frequency required by the TS was identified as NCV 50-423/98-211-07. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(i) Intermediate Range Nuclear Instrument Reset Value

(Closed) LER 50-423/97-033-00, "P-6 Bistable Reset Function Set Non-Conservatively Resulting in TS Violation." The licensee identified a discrepancy involving the reset value for the P-6 interlock (intermediate range neutron flux permissive to allow manual block of source range reactor trip during power escalation). This reset value controls when the interlock function is automatically reinstated on decreasing power. Specifically, the reset value was set lower (less conservative) than the Allowable Value specified in TS Table 2.2-1. This would result in instances between 5.0 and 6.0 X 10^{-11} amps, where the P-6 interlock would be bypassed and the source range flux monitoring instrumentation would not be re-energized and operable as required by the TS. The condition has existed since initial startup.

The licensee amended the TS Bases to reflect the existing reset value and to allow the reset value to be below the "set" value and, thereby, accommodate a deadband to prevent chatter attributable to drift (the amendment was approved by the NRC on September 29, 1997). The team confirmed that the amended TS Bases had been implemented and found this acceptable.

The team also reviewed the licensee's corrective actions identified in the LER and considered the actions adequate. The failure to set the P-6 interlock reset value consistent with the TS was identified as NCV 50-423/98-211-08. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(j) Pipe Breaks Exclusion Zones

(Closed) LER 50-423/97-036-00 and (Closed) LER 50-423/97-036-01, "Pipe Breaks in Exclusion Zones Differ From Those Provide in The Final Safety Analysis Report." The licensee identified that incorrect stress analysis limits may have been used in the design of moderate-energy fluid system piping and that a design change that revised the containment design from subatmospheric to atmospheric did not evaluate containment leakage. These two conditions were reported by the licensee as conditions outside the design basis. The licensee determined that the containment penetration areas needed to be considered as break exclusion zones in order to preclude the postulation of out-leakage and as such, required that some weld locations be included in an augmented Inservice Inspection (ISI) program.

The team reviewed updates to applicable tables and figures in FSAR Sections 3.6.1 and 3.6.2, per FSAR Change Request 97-MP3-585. The team also reviewed Calculation 97-ENG-01329-M3, "Review of Break Exclusion Zones for MP3," Rev. 1, Change 1, and verified that the licensee's piping stress calculations were acceptable.

In addition, the team reviewed the licensee's corrective actions, as identified in the LERs, and considered the actions adequate. The failure to ensure that the design basis had been adequately maintained was identified as NCV 50-423/98-211-09. This licensee-identified and corrected the violation that is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(k) Fuel Building Exhaust Fans

(Closed) LER 50-423/97-038-00, "Single Failure Potential for Fuel Building Exhaust Fans." The Fuel Building standby fan would not automatically start in the event of a single failure of the operating Fuel Building filter fan inlet or outlet damper. The team reviewed the elementary diagrams revised by DCNs DM3-00-1207-97, "Fuel Building Filter Exhaust Fan Start Logic Modification (orange train)," dated January 21, 1998, and DM3-00-1208-97, "Fuel Building Filtered Exhaust Fan Start Logic Modification (purple train)," dated January 21, 1998, and verified that a single failure of a Fuel Building filter fan inlet or outlet damper would not prevent automatic startup of the standby fan, and the alarm circuit would respond properly to a single failure.

The team also reviewed the licensee's corrective actions, as identified in the LER, and considered the actions adequate. The failure to ensure that the design basis had been adequately maintained was identified as NCV 50-423/98-211-10. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(l) Turbine Building Sump Effluent Monitor Failure

(Closed) LER 50-423/97-039-00, "Inoperability of Turbine Building Sump Effluent Monitor DAS-RE50 Due to a Failure to Recognize a Design Deficiency that Resulted in a TS Violation." The licensee did not detect the inoperability of the Turbine Building Sump Effluent Monitor when the sample line had become partially blocked because of a rusted and partially shut inlet sample solenoid valve. The design and use of the installed instrumentation was not adequate for timely detection of its failure. TS 3.3.3.9(b) requires the Turbine Building Sump Effluent

Monitor to be continuously operable to monitor Turbine Building floor drain activity levels. This condition was identified by the licensee and existed from May 13 through June 14, 1997.

The team reviewed DCN DM3-00-1200-97, "3DAS-RE50 Low Flow Alarm and Interlock," dated January 19, 1998; P&ID 12179-EM-106C-29, Rev. 29; SP 3450G01, "Turbine Building Floor Drains Monitor Channel Calibration," Rev. 6; and elementary diagrams 12179-ESK-6AFV DCN Rev. B, and 12179-ESK-7SE DCN Rev. A. The team also verified that the modification replaced the existing pressure instrument with a more reliable flow measurement and changed the interlock/alarm scheme such that an inoperable monitor would divert flow to liquid radwaste processing and provide an alarm. The team found this revised design acceptable.

The team also reviewed the licensee's corrective actions identified in the LER, and considered the actions adequate. The failure to monitor the Turbine Building sump effluent in accordance with the TS was identified as NCV 50-423/98-211-11. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(m) Engineered Safety Features Activation System (ESFAS) Testing

(Closed) LER 50-423/97-043, "Inadequate TS Surveillance for the ESFAS and Reactor Trip System Instrumentation Interlocks." Surveillance test procedures did not adequately test the ESFAS permissive P-11 (pressurizer pressure signal that allows manual block of SI signal and other ESFAS signals), the steam dump permissive P-12 (low T_{avg} signal that blocks steam dump), and the P-13 permissive (turbine impulse chamber pressure signal that permits feedwater isolation). Channel test overlap between the analog protection sets and the solid-state protection system was not complete because the procedures did not provide for observing a change in the channel bistable status lights during the testing. This condition was identified by the licensee and had existed since initial startup. The requirements for operability of the P-11 and P-12 permissives are established in TS Table 3.3-3, and for the P-13 permissive in TS Table 3.3-1.

The team reviewed channel calibration procedures SP 3443A21, "Protection Set Cabinet I Operational Test," Rev. 10, Change 7; SP3443B21, "Protection Set Cabinet II Operational Test," Rev. 10, Change 9; SP 3443C21, "Protection Set Cabinet III Operational Test," Rev. 10, Change 8; SP 3443D21, "Protection Set Cabinet IV Operational Test," Rev. 10 Change 7; SP 3442A10, "Delta T/T_{avg} Channel I Calibration," Rev. 0, Change 14; SP 3442A20, "Delta T/T_{avg} Channel II Calibration," Rev. 0, Change 13; SP 3442A30, "Delta T/T_{avg} Channel III Calibration," Rev. 2; SP 3442A40, "Delta T/T_{avg} Channel IV Calibration," Rev. 0, Change 13; SP 3442C10, "Pressurizer Pressure Narrow Range Channel 1 Calibration," Rev. 3; SP 3442C20, "Pressurizer Pressure Narrow Range Channel 2 Calibration," Rev. 3; SP 3442C30, "Pressurizer Pressure Narrow Range Channel 3 Calibration," Rev. 2; and SP 3445A01, "Turbine Impulse Chamber Pressure Calibration," Rev. 2, Change 8. As a result, the team confirmed that the procedures were appropriately revised. The team discussed with the licensee the scope of its investigation of the testing of other permissives and found the extent of its investigation acceptable.

The team also reviewed the licensee's corrective actions identified in the LER, and considered the actions adequate. The failure to adequately test the ESFAS permissives consistent with TS

was identified as NCV 50-423/98-211-12. This licensee-identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(n) Reactor Trip Response Time Testing

(Closed) LER 50-423/97-045-00, "Reactor Trip System Response Time Testing not in Verbatim Compliance with the TSs." This issue involved a conflict between the definitions of Reactor Trip System response time testing in the FSAR and TS. Testing was accomplished per the FSAR definition in lieu of the TS definition. The team reviewed Procedure SP 31024, "Calculation of Reactor Trip and ESF Response Times," Rev. 3, and verified that 150 milliseconds was added to the response time acceptance criteria for the gripper coil voltage collapse time. The team reviewed FSARCR 97-MP3-221 and verified that the definition of "Reactor Trip System Response Time" was revised to match the definition contained in Section 1.28 of the TS.

The team reviewed the licensee's corrective actions identified in the LER, and considered the actions adequate. The failure to test gripper coil voltage collapse response time in accordance with the TS was identified as NCV 50-423/98-211-13. This licensee identified and corrected violation is being treated as a Noncited Violation consistent with Section VII.B.1 of the NRC Enforcement Policy.

(o) A EDG Inoperable

(Closed) LER 50-423/97-57-00, "A Emergency Diesel Generator Inoperable Due to Ventilation Alignment Prohibiting Tornado Recovery." This issue was identified as VIO 50-423/97-209-09 and is discussed in Section 2.1.2(o) of this inspection report.

(p) ECCS Piping Not Verified Full of Water

(Closed) LER 50-423/98-002-00, "Verification of ECCS Piping Full of Water Per Technical Specification 4.5.2.b.1 Not Being Met." This issue was identified as VIO 50-423/97-206-21, and corrective actions are discussed in Section 2.1.2(i) of this report.

(q) Containment Bypass Leakage

(Closed) LER 50-423/98-007-00, "Containment Bypass Leakage in Excess of Technical Specification Limits." This LER documented that containment bypass leakage specified in TS 3.6.1.2 had historically been exceeded from February 3, 1988, through May 11, 1989 (Cycle 2). This condition was identified by S&L in DR-MP3-1016 and occurred because not all bypass leakage pathways were included in the calculation for containment leakage. As a result of omission of the bypass leakage pathways, the calculation of as-left containment bypass leakage during Cycle 2 was 130.1 percent of the limit established in TS 3.6.1.2.c. However, the licensee's evaluation verified that current leakage is well within this limit, and the licensee had implemented all corrective actions associated with this LER. These corrective actions included evaluating all containment boundary bypass leakage paths to ensure that all such penetrations are included in FSAR Table 6.2-65 and OPS Form 3273-3/4.3.6.1.2, Table 3.6.1.2-1, "Containment Leakage," Rev. 0. The team reviewed the corrective actions and determined that they were acceptable. The failure to calculate the containment leakage rate in accordance with TS 3.6.1.2.c is identified as VIO 50-423/98-211-14.

(r) Manual Control of an Automatic Safety Function

(Closed) LER 50-423/98-015-00, "Historical Failure to Enter Technical Specifications for Manual control of an Automatic Safety Function." This issue was previously identified as VIO 50-423/97-210-09, and is discussed in Section 2.1.2(v) of this report.

(s) Unfiltered Ventilation System Leakage

(Closed) LER 50-423/98-029-00, "Supplementary Leak Collection and Release System Bypass Leakage May Result in Unaccounted Control Room And Technical Support Center Doses." This LER reported a condition that could exceed the 10 CFR Part 50, GDC 19, "Control Room," control room operator dose limits. During a LOCA with offsite power available, five exhaust fans in the Engineered Safety Features (ESF) and Auxiliary Building Ventilation System (ABVS) could fail to trip and continue to operate. This would increase the differential pressure across the closed dampers downstream of the exhaust fans and increase leakage of contaminated air to the atmosphere. The control room ventilation system takes a suction from the atmosphere and the dose in the control room would increase due to the introduction of airborne radioactivity. The licensee established a maximum allowable control room dose of 30 rem to the thyroid over a 30-day period, in accordance with GDC 19. Under the worst-case LOCA conditions (offsite power available, all five fans run for 30 days, and a single failure causes one train of dampers to remain open), the approximate control room dose for 30 days would be 70 rem.

In 1992, the licensee implemented Plant Design Change Record MP3-89-013, "MP3 Containment Design Pressure Change." This changed containment operating pressure from subatmospheric to near atmospheric. Before this change, containment pressure would be subatmospheric within 1 hour after the LOCA. After the design change, leakage from the containment would increase in that it occurred for 30 days, rather than 1 hour. When implementing the design change, the licensee did not evaluate the effect of containment leakage over an extended period on operation of the ESF/Auxiliary Building Ventilation and Control Room Ventilation systems.

The licensee revised procedure OP 3314F to require that operators manually secure the fans following a LOCA. The licensee identified that this is a USQ and, in a letter dated June 6, 1998, submitted the USQ for NRC review. The team reviewed OP 3314 F, "Control Building Heating, Ventilation, Air Conditioning and Chilled Water," Rev. 16, Change 9, and verified that operators are directed to trip the fan breakers before realigning the Control Room dampers.

The team reviewed the deficiencies identified by the licensee during the CMP that were reported in LER 50-423/98-029-00 and determined that a violation of 10 CFR 50.59 requirements occurred. However, after consultation with the Director, Office of Enforcement, it was determined that enforcement discretion can be exercised pursuant to Section VII.B.2 of the NRC Enforcement Policy and to not issue a formal Notice of Violation because the violation was (1) based on the licensee activities prior to the events leading the shutdown; (2) not classified higher than a Severity Level II; (3) not willful; and (4) plant restart required NRC concurrence. Discretion is appropriate because the NRC has in place a formal plan that is providing a broad-based evaluation of Millstone readiness for restart that confirmed that NNECO has taken corrective action for this issue and further enforcement action is not necessary to achieve remedial action. (NCV 50-423/98-211-15)

2.4 Closed Confirmed Level 3 DRs

2.4.1 Scope of Review

The team reviewed closed Confirmed Level 3 DRs and verified that corrective actions were adequate and implemented. A closed Confirmed Level 3 DR is a discrepancy that the licensee agreed was a new discrepant condition (e.g., a design- or licensing-basis discrepancy that the licensee had not previously identified), the DR resolution was finalized, and S&L accepted the licensee's resolution. The NRC team reviewed 21 of the 22 Confirmed ICAVP Significance Level 3 DRs (see Appendix C for a list of Level 3 DRs reviewed). The licensee had requested the deferral of the implementation of corrective actions for DR-MP3-0328 until after restart. Following the review of the licensee's Operability Determination that verified that the existing installation met the requirements of ANSI/ASME B31.1 for seismic loading, the staff approved the licensee's request. The corrective actions associated with DR-MP3-0328 have been completed by the licensee.

2.4.2 Findings

(a) DR-MP3-0001

This DR identified two discrepancies between the TS (6.8.1 and 6.8.3.c) and Station Procedure DC-1, "Administration of Procedures and Forms," Rev. 5. The first discrepancy identified a difference in approval authority for temporary procedure changes (paragraph 1.6.3.a.4 of Procedure DC-1). In the second discrepancy, classified as an editorial discrepancy, (Section 1.1 of procedure DC-1) had an obsolete statement about the TS 6.8.1 requirements for Security Plan and Emergency Plan procedures.

The licensee's response and corrective actions included the following:

- Station Procedure DC-1, Rev. 6, Change 3, corrected references to TS 6.8.1 and added approval and definition of the Station Qualified Reviewer Program personnel.

- Station Procedure DC-15, Rev. 2, Change 1, clarified and added definition to Attachment 1 that uses different titles applicable to the Station Qualified Reviewer Program Department Manager.

The team reviewed the corrective actions and verified that they were acceptable and implemented.

(b) DR-MP3-0006

The DR identified that the licensee's design change program was not compliant with respect to requirements for reviews of safety-related Minor Modifications (MMODs) by the Plant Operating Review Committee/Station Operating Review Committee. Specifically, the 10 CFR 50.59 screening requirements in the DCM for safety-related or nonsafety-related modifications may not always show that a safety evaluation is required when a MMOD affects nuclear safety.

The licensee's response and corrective actions included revisions to the DCM, (Rev. 5, Change 3, and Rev. 6, Change 2, that ensures modifications affecting nuclear safety are

subjected to onsite committee action and any modifications resulting in such a change are processed via a DCR rather than a MMOD. The team reviewed the licensee's corrective actions and verified that they were acceptable and implemented.

(c) DR-MP3-0029

FSAR Section 9.2.1 requires two SW pumps in the same train to operate following a LOCA; however, the licensee's design-basis calculations for the SW house cubicle temperature were based on operation of a single SW pump per train.

To resolve this discrepancy, the licensee implemented the following corrective actions:

- Review and supersede initial SW pump house cubicle calculations that were based on one SW pump operating per train during post-LOCA conditions.

- Issue Calculations SWP-01515, "Service Water Pump Cubicle Temperature for Operability Determination," Rev. 1, and T-01613-S3, "Gothic Analysis of Service Water Pump Cubicle Temperature in Winter and Summer Mode," Rev. 0.

- Issue Technical Evaluation M3-EV-98-0038, "Service Water Pump Cubicle Temperature with Two Pumps Operating," Rev. 2.

- Issue FSARCR 98-MP3-101 to revise FSAR Section 9.4.8.1.1 "Design Bases Service Water System," and Appendix 3B, "Environmental Design Conditions," to reflect the revised post-LOCA design-basis SW pump house cubicle temperature of 119 °F with two pumps operating in one train (was 104 °F with one pump operating per train).

The team concluded that these corrective actions were acceptable and implemented. However, a violation of NRC requirements was identified. This violation involves reportability in accordance with 10 CFR 50.73. 10 CFR 50.73(a)(2)(ii)(b) requires that the licensee report conditions that are outside of the design basis of the plant. The failure to report to the NRC that the design-basis SW cubicle temperature limit of 104 °F did not bound two-pump post-LOCA operation was identified as the first example of a violation of 10 CFR 50.73(a)(2)(ii)(b). However, after consultation with the Director, Office of Enforcement, it was determined that enforcement discretion can be exercised pursuant to Section VII.B.2 of the NRC Enforcement Policy and to not issue a formal Notice of Violation because the violation was (1) based on the licensee activities prior to the events leading the shutdown; (2) not classified higher than a Severity Level II; (3) not willful; and (4) plant restart required NRC concurrence. Discretion is appropriate because the NRC had in place a formal plan that provided a broad-based evaluation of Millstone readiness for restart that confirmed that NNECO had taken corrective action for this issue (issued LER 50-423/98-035-00) and further enforcement action was not necessary to achieve remedial action. (NCV 50-423/98-211-16)

(d) DR-MP3-0035

This DR identified that the internals of the four containment recirculation heat exchangers were not visually inspected on a routine interval to verify that fouling was not occurring. The team reviewed SP 3626.13, "Service Water Heat Exchangers Fouling Determination," Rev. 15, Change 1, and verified that the procedure was revised to require that the containment

recirculation heat exchangers be periodically visually inspected. The team concluded that this corrective action was acceptable and implemented.

(e) DR-MP3-0051

This DR identified that Calculation 12179-CFSK-732E-E66, "Qualification of Embedded Plate/ ESF Building CFSK," did not properly qualify embedded plate E66 that supports loads from two pipe supports. The embedded plate was over stressed by 11 percent as a result of loads imposed by the first support, and the calculation documented that the loads on the embedded plate from the second support as minor without a formal analysis of the effects of those loads. The team reviewed Rev. 1, Change 1, to Calculation 12179-CFSK-732E-E66, and verified that a less conservative but more accurate method was used to qualify embedded plate E66. The new method was based on ME-323, "Preparation and Revision of Pipe Support Analysis," Version 2, Level 3. The team found that the licensee's corrective action was acceptable and implemented.

(f) DR-MP3-0331

This DR identified that drawings EM-148E-10 and EM-148A-24, and FSAR Table 1.8-1 disagreed on the position of filter drain valves 3HVR-V964, V970, V993, and V999. Specifically, drawings EM-148E-10 and EM-148A-24 showed these valves as normally open, while FSAR Table 1.8-1 stated that the valves are normally closed. The licensee determined that the drawings specified the correct position of the valves. The team reviewed FSARCR 98-MP3-38, dated March 13, 1998, and verified that the licensee revised FSAR Table 1.8-1 to indicate that the valves are normally open. The team concluded that this corrective action was acceptable and implemented.

(g) DR-MP3-0355

This DR identified that one of three 480-volt power cables running between Motor Control Center 3EHS*MCC1B4 and the tray system, enters a 120-volt control cable tray (C service level) rather than a 480-volt cable tray (K service level) as required by cable routing procedures. Corrective action consisted of rerouting the errant cable into the tray per AWO M3-97-23210, which was completed January 22, 1998. The team reviewed AWO M3-97-23210 and verified that the corrective actions were acceptable and implemented.

(h) DR-MP3-0434

During the performance of a routine surveillance, the licensee identified that the level in each of the containment trisodium phosphate (TSP) baskets was below the required level. This issue was documented in Adverse Condition Report (ACR) 12327. The licensee attributed the low levels to settling, and refilled the TSP baskets to a level of one-half inch above the fill line in anticipation of future settling. DR-MP3-0434 identified that this action was accomplished without considering the impact of the additional TSP on the maximum transient pH. As a corrective action, the licensee revised Calculation 03703-US(B)-350, "Trisodium Phosphate System for Sump pH Control." The team reviewed Rev. 2 to 07303-US(B)-350, and found that the revised calculation supported the conclusion that the pH in the local areas surrounding the TSP baskets would not exceed 11, which was acceptable. The team also reviewed FSARCR 98-MP3-39, dated March 6, 1998, that revised the maximum pH range following a design-basis

LOCA from approximately 10.5 to approximately 11. The team concluded that these corrective actions were acceptable and implemented.

(i) DR-MP3-0588

This DR identified that the design and licensing bases for the backup cooling of the SLCRS and the ABVS exhaust filter units took credit for the fire protection sprinkler system that is not safety-related. As a corrective action, the licensee changed the design and licensing bases for SLCRS and ABVS backup cooling. The team reviewed Calculation 12179-UR(B)-452, "Estimate of the Iodine Loading and Temperature Rise in the SLCRS/Auxiliary Building Exhaust and Fuel Building Charcoal Filters," Rev. 1, Change 1. This calculation determined that there was adequate cooling through heat transfer in the sheet metal skin of the filtration units, and the heat generated from post-LOCA conditions is insufficient to raise the carbon bed temperature above 250 °F, with no system airflow and with no credit for the nonsafety-related sprinkler system in cooling the charcoal. The team agreed with the results of this calculation and verified that the corrective actions were acceptable and implemented.

The licensee issued ERC-25212-ER-98-103, "MP3 Charcoal Filter Heatup Analysis," Rev. 0, to approve an FSARCR to change the design and licensing bases. The team reviewed ERC-25212-ER-98-103 and concluded that an ERC was not the appropriate document to use for processing a DCR. Procedure NGP 5.31, "Engineering Record Correspondence (ERC) and Technical Evaluation," Rev. 3, Section 1.1, states that, "The ERC will not be used to provide authorization for licensing-basis changes, design-basis changes, or physical work."

After the team questioned the use of an ERC for processing a design change, the licensee issued DCR M3-98031, "MP3 SLCRS, ABVS, and Fuel Building Charcoal Adsorbent Cooling," Rev. 0, to make the design- and licensing-basis change. The team concluded that with the issuance of DCR M3-98031 the licensee had adequately corrected this issue.

(j) DR-MP3-0624

This DR identified deficiencies in the storage of vendor-supplied information. As a result, the licensee revised Station Procedure DC2, "Developing and Revising Millstone Procedures and Forms," to require copies of all vendor-supplied information used as a basis for preparing a procedure to be processed by Station Procedure DC16, "Vendor Equipment Technical Information Program." The team reviewed DC2, Rev. 1, Change 4, and verified that DC2 refers a procedure writer to DC16, which addresses control of vendor information. The team verified that the corrective actions were acceptable and implemented.

(k) DR-MP3-0639

Temporary Alteration 3-97-027, dated March 27,1997, provided instructions for installing a temporary non-Code patch over a pinhole leak in the "A" train SW piping. This DR identified that Code relief was not obtained from the NRC for this repair. The licensee concluded that the specific issue reported in the DR does not represent a discrepant condition, but the programmatic issue represents a discrepant condition with respect to the licensee's policy toward performing repairs during plant outages. The team concluded that NRC relief is not required when a Code repair of a flaw is completed within 30 days of discovery.

The licensee revised Specification SP-ST-ME-947, "Standard Specification for Non-Code Repairs in Safety Class 3 Piping," Rev. 1, to clarify the requirements for obtaining NRC relief during both operational and shutdown modes within a 30-day period. The team reviewed this specification and concluded that this corrective action was acceptable and implemented.

(l) DR-MP3-0667

This DR identified that the calculation for sizing the component cooling pumps' temporary fans did not include all necessary heat loads, did not provide an adequate basis for inlet supply temperature, and did not provide an adequate basis for fan pressure rating. In response, the licensee reperformed the calculation for the fan heat load, provided additional documentation for supply inlet temperature, and provided a sufficient basis for the pressure rating of the fans. The team reviewed Calculation P(B)-1130, "Temporary Ventilation of CCP Pump Area," Rev. 2, Change 5, and verified that the relatively minor additional heat sources were included in the calculation. The team therefore agreed with the licensee's conclusion that the present temporary fans are adequately sized for their intended use.

This DR also identified that there was procedural guidance to open the doors that would provide the discharge path for the temporary fans, but that there was no similar guidance that would provide an air supply path from outside the Auxiliary Building. The team reviewed OP 3314 J, "Auxiliary Building Emergency Ventilation and Exhaust," Rev. 4, and verified that the procedure was revised to provide guidance for establishing an air supply path. The team concluded that these corrective actions were acceptable and implemented.

(m) DR-MP3-0669

This DR identified a discrepancy with fan brake horsepower and motor rating between the calculation and vendor data for the CHS pumps and CCW pump area ventilation supply fans, 3HVR*FN14A&B. With high air density during winter operation, Calculation 3-92-103-191-M3, "Fan 3HVR*FN14A&B Motor Requirements," Rev. 1, Change 5, calculates a fan brake horsepower greater than the motor nameplate rating. The licensee prepared Technical Evaluation M3-EV-98-0046, "Technical Evaluation of Charging and Component Cooling Water Pump Area Supply Fan Motor Loading," Rev. 0. The Technical Evaluation concluded that review of initial startup data and test data taken on January 16, 1998, demonstrated that the fan curve showed higher brake horsepower requirements than test data indicated the fan was actually drawing. During the course of the review, the fan vendor was contacted and agreed with the licensee's methodology. The team reviewed the Technical Evaluation and concluded that the 40-horsepower motors are satisfactory and that the corrective actions were acceptable and implemented.

(n) DR MP3-0686

This DR identified a potential unfiltered leakage path from the ABVS to the atmosphere that could occur during LOCA conditions. The team reviewed Calculation UR(B)-450, "MP-3 LOCA Doses at the Site Boundary, Control Room, and TSC Assuming Duct Leakage and Damper Bypass", Rev. 0, and verified that the calculation was revised to account for the unfiltered leakage path. The team noted that the increased dose from the unfiltered leakage path was not significant and concluded that the licensee's corrective action was acceptable.

(o) DR-MP3-0670

This DR raised a number of questions concerning the air temperatures in the Auxiliary Building during cold weather conditions. The most significant questions was whether one train of heaters installed as part of DCR MP3-93-067, "Auxiliary Building CHS/CCP Area Ventilation Heaters," was sufficient to ensure that the Auxiliary Building temperature could be kept above 32 °F with the lowest assumed outside air temperature (0 °F) and a degraded voltage condition. The team reviewed Calculation 3-92-103-191M3, "CCP and CHS Pump Area Ventilation," Rev. 1, CCN 9, and concluded the calculation demonstrates that temperatures in the building will remain above 32 °F under the postulated conditions and that the corrective actions were acceptable and implemented.

(p) DR-MP3-0687

This DR identified that the fan blade missile evaluation included the fan casing, but did not include the flexible connection for fans 3HVR*6A/B, 10A/B, 13A/B, and 14A/B. The licensee inspected the fans and documented the results in Walkdown MP3-WALKDOWN-98-042, dated April 14, 1998. The licensee concluded that fan blade missiles were precluded as hazards from the flexible connections for fans 3HVR*6A/B and 3HVR*10A/B. However, the inspection results did not eliminate the possibility of fan blade missiles from fans 3HVR*13A/B and 3HVR*14A/B. The licensee issued Change 4, to Calculation 12179-NM(S)-DKB, "Fan Blade Missiles," Rev. 0, and DCR M3-98025, "Installation of Flexible Connection Missile Barrier For Fan No. 3HVR*13A/B and 3HVR*14A/B," Rev. 0, to provide missile shields to the flexible connections to fans 3HVR*13A/B and 3HVR*14A/B. The team reviewed these corrective actions and verified that they were acceptable and implemented.

(q) DR-MP3-0762

This DR identified that vibration test results for SLCRS exhaust fans 3HVR*12A/B exceeded the vibration levels specified in ANSI N509-1976, "Testing of Nuclear Air-Cleaning Systems." As a corrective action, the licensee evaluated the current vibration levels for fans 3HVR*12A/B and modified FSAR Table 1.8-1 to recognize that ANSI N509-1976 vibration levels were not utilized. The team reviewed Technical Evaluation M3-EV-0033, "Fan Performance Acceptance Criteria for Quality Related Fans," dated March 6, 1998, and FSARCR 98-MP3-36, dated March 16, 1998, and verified that the corrective actions were acceptable and implemented.

(r) DR-MP3-0795

This DR identified that the procurement specifications for ABVS exhaust fans 3HVR*FN6A/B, specified acceptable fan vibration test levels that exceeded the vibration levels specified in ANSI N509-1976. As a corrective action, the licensee evaluated the current vibration levels for fans 3HVR*FN6A/B and modified FSAR Table 1.8-1 to recognize that ANSI N509-1976 vibration levels were not utilized. The team reviewed Technical Evaluation M3-EV-0033, and FSARCR 98-MP3-36 and verified that the corrective actions were acceptable and implemented.

(s) DR-MP3-1011

This DR identified that a previous modification, DCR MP3-91-96, "Emergency Diesel Fuel Oil Transfer Pump Discharge Strainer Element Removal," Rev. 0, did not evaluate the effects of

unstrained fuel oil on the operation of the EDGs. The licensee issued FSARCR 98-MP3-8 to clarified that Table 1.8-1 and Section 9.5.4, to recognize the control of fuel oil quality through both delivery and periodic sampling (every 31 days) in accordance with TS 3/4.8 and that the TS requirements were more stringent than those specified by the diesel manufacturer. The team found that the licensee's response and corrective actions were acceptable and implemented.

(t) DR-MP3-1016

This DR is discussed in Section 2.3.2(q) of this report.

(u) DR-MP3-1026

This DR identified that Calculation US(B)-353, "Recirculation Spray System Piping Analysis," did not address scenarios involving a Main Steam Line Break (MSLB). The licensee evaluated RSS and QSS piping and support analyses for MSLB scenarios and determined that for QSS, the MSLB was a more limiting case. The revised calculations listed in Memo MP3-DE-98-0226, "Owner's Review of Calculations Associated with DR-MP3-1026," dated May 14, 1998, demonstrated that the QSS and RSS piping meet design-basis criteria for both LOCA and MSLB events, and no modifications were required in response to this reanalysis. The team reviewed the scope of the calculations included in the reanalysis and found that the licensee's corrective actions were acceptable and implemented.

2.5 Corrective Actions for RSS Pump Expansion Joint Failure

2.5.1 Scope of Review

During the outage, the licensee installed flow-restricting orifices between the discharge nozzle of each RSS pump and the expansion joint downstream of the pump discharge nozzle. While performing the post-modification testing, the licensee identified that the expansion joint vibrations exceeded allowable values. The proposed corrective action was to change the location of the flow-restricting orifice from upstream of the expansion joint to downstream of the expansion joint. During the implementation of this change, the licensee discovered that the liner inside one of the expansion joints was fractured with a portion of the liner missing. The team reviewed the licensee's corrective actions.

2.5.2 Findings

The licensee implemented a number of corrective actions. The immediate corrective actions were to retrieve the missing pieces, and determine if all of the pieces were accounted for. The licensee determined that there was a potential that nine small pieces were left in the system. The licensee could not make a conclusive determination because of the irregularity of the retrieved pieces and the shape of the remainder of the sleeve (i.e., it is possible that all pieces were retrieved). The licensee performed an extensive evaluation of the effects of the remaining pieces on plant safety. Additional corrective actions included formal root cause evaluation, revision of the modification to replace the expansion joints with a hard pipe, and development of the long-term programmatic changes to prevent a future recurrence of similar problems.

The team found the proposed corrective actions and the evaluation of the effects of the remaining pieces to be adequate with one exception. Specifically, the licensee's initial evaluation did not address the potential for the remaining pieces to interact with the ECCS pump seals following a LOCA for a scenario where the pieces are located in the crossover pipe and the LOCA takes place before the next refueling outage (the licensee's proposed corrective actions include flushing the crossover pipe during the next refueling outage). The licensee concurred with the team's observations and revised the evaluation to address the effect on the ECCS pump seals. The results of this evaluation indicated that there were no adverse effects associated with this scenario. The team reviewed the revised evaluation and concurred with the licensee's conclusions.

2.6 Corrective Action Conclusions

The team found that the licensee's corrective actions for NRC-identified violations were acceptable, and implementation of these corrective actions was adequate. Two EEIs from a previous inspection were identified as violations. The first involved the failure to monitor SW leakage from the RSS heat exchangers in accordance with TS 6.8.4 (VIO 50-423/98-211-01). The second involved the failure to implement adequate corrective actions for the single-failure vulnerability of control room inlet dampers (VIO 50-423/98-211-02). The team found that the URIs and IFIs were adequately resolved.

The team found that the corrective actions for the LERs met the requirements of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." In all, 13 LERs were determined to be violations of NRC regulations. Since these violations were licensee-identified and corrected Severity Level IV violations, enforcement discretion was exercised pursuant to Section VII.B.1 of the NRC Enforcement Policy for the LERs listed below:

LER	Finding No.	LER	Finding No.
50-423/96-050-00	NCV 50-423/98-211-03	50-423/97-036-01	NCV 50-423/98-211-09
50-423/97-013-00	NCV 50-423/98-211-04	50-423/97-038-00	NCV 50-423/98-211-10
50-423/97-020-00	NCV 50-423/98-211-05	50-423/97-039-00	NCV 50-423/98-211-11
50-423/97-031-00	NCV 50-423/98-211-06	50-423/97-043-00	NCV 50-423/98-211-12
50-423/97-032-00	NCV 50-423/98-211-07	50-423/97-045-00	NCV 50-423/98-211-13
50-423/97-033-00	NCV 50-423/98-211-08		

In addition, a violation was issued for the item identified by S&L that was reported in LER 50-423/98-007-00 (VIO 50-423/98-211-14), and enforcement discretion was exercised pursuant to VII.B.2 of NRC Enforcement Policy for LER 50-423/98-029-00 (NCV 50-423/98-211-15).

The team also identified an example in which the licensee failed to report a condition outside of the design basis (NCV 50-423/98-211-16). The team found that for the Closed Confirmed Level 3 DRs reviewed, the licensee's corrective actions were adequate in that they corrected all conditions adverse to quality. The team also found that the corrective actions for the failed RSS expansion joint liner were acceptable. Based on this inspection, the team does not recommend expanding the scope of ICAVP.

3.0 Discrepancy Reports

3.1 Closed Previously Identified, Closed Non-Discrepant and Confirmed Level 4 DRs

A closed Previously Identified DR is a discrepancy identified by S&L that the licensee had previously identified during the CMP. During the DR review process, S&L reviewed the licensee's responses to DRs and agreed that the licensee previously identified the discrepancies on the basis of information provided by the licensee. A closed Non-discrepant DR is a condition that S&L originally considered discrepant but after the licensee responded to the DR, S&L concluded that the condition was not discrepant. A closed Confirmed Level 4 DR is a discrepancy that the licensee agreed was a new discrepant condition. In such cases, the DR resolution was finalized and S&L accepted the licensee's resolution.

3.1.2 Scope of Review

The team reviewed a sample of DRs that were determined to be Previously Identified and Non-discrepant to assess whether the process had properly categorized these DRs. The team also reviewed a sample of Level 4 DRs to validate that they were appropriately classified as Level 4. The Non-discrepant and Previously Identified DRs reviewed are listed in Appendix D and the closed Confirmed Level 4 DRs reviewed are listed in Appendix E.

3.1.3 Findings

On April 24, 1997, the licensee identified a concern that the maximum flow capacity of the turbine bypass control valves may exceed the limit of 970,000 lb/hr stated in the FSAR. FSAR Section 10.4.4.1 states that the capacity of any single turbine bypass valve does not exceed 970,000 lb/hr of steam at the main steam supply system design pressure, 1185 psig, as supplied by the NSSS. FSAR Section 10.4.4.1 further states that the failure of a turbine bypass valve to close will not cause an uncontrolled plant cooldown and excessive reactivity excursion. The licensee's investigation determined that the valve vendor had not satisfied the requirement of the Stone and Webster Engineering Corporation Specification 2472.120-183, Technical Data Sheet 3-3, for a maximum permissible flow through these valves of 970,000 lb/hr at 1200 psia. The corrective action for this error included a design modification to shorten the valve stroke to limit the turbine bypass control valve maximum capacity to 970,000 lb/hr.

The team reviewed the reportability evaluation performed by NNECO that determined the event was not reportable. 10 CFR 50.73(a)(2)(ii)(B) requires that the licensee shall submit an LER within 30 days of the discovery of any event or condition that resulted in the nuclear power plant being in a condition that was outside the design basis of the plant. The licensee's reportability evaluation determined that the consequences of the maximum flow capacity of the unmodified turbine bypass control valves was bounded by the accident analysis for an MSLB and was within the design basis of the plant. Since the specific values chosen to control the maximum flow capacity of the turbine bypass control valves exceeded the FSAR limits, and Unit 3 was considered to be in a condition outside its design basis. The failure to report this issue, as required by of 10 CFR 50.73, was identified as the second example of a violation of 10 CFR 50.73(a)(2)(ii)(b). However, after consultation with the Director, Office of Enforcement, it was determined that enforcement discretion can be exercised pursuant to Section VII.B.2 of the NRC Enforcement Policy and to not issue a formal Notice of Violation because the violation was (1) based on the licensee activities prior to the events leading the shutdown; (2) not classified

higher than a Severity Level II; (3) not willful; and (4) plant restart required NRC concurrence. Discretion is appropriate because the NRC had in place a formal plan that provided a broad-based evaluation of Millstone readiness for restart that confirmed that NNECO has taken corrective action for this issue (issued LER 50-423/98-026-00) and further enforcement action was not necessary to achieve remedial action. (NCV 50-423/98-211-16).

3.1.4 Conclusion

The team concluded the licensee had performed and documented adequate technical evaluations to support the DR classification, and all of the licensee's responses contained adequate technical justification that a design- or licensing-basis issue did not exist. The licensee's resolution process had provisions to ensure appropriate review of DR responses, and the licensee performed appropriate evaluations to support the DR classification.

3.2 Trending of DRs

3.2.1 Scope of Review

The team collectively examined all Confirmed DRs to identify adverse trends.

3.2.2 Findings

A trend in the area of calculations (control and accuracy) was identified by both S&L and the NRC. This trend was determined not to be safety significant because the errors did not materially alter the results or conclusions of the calculations. This trend does indicate a need for improved accuracy and thoroughness in the licensee's independent calculational review. The team determined that these improvements could be made on an ongoing basis.

The team also reviewed DRs in the area of drawings/component information and related to RSS. A significant number of DRs were identified in both that could be indicative of a trend. In the area of drawings/component information, the team concluded that the this trend was not safety significant because the as-built configuration in the plant was correct and other drawings existed that correctly represented the existing configuration. In reviewing the RSS-related DRs, the team concluded that the DRs were of minor safety significance, since most were in the areas of calculations and drawings/component information that were already determined collectively to be not safety significant.

In addition, the team reviewed the time sequence for identifying RSS-related DRs. Early in the ICAVP, CMP closure dates were chosen such that S&L could classify a DR as Previously Identified by the licensee if the licensee had identified the concern before the closure date. The NRC imposed a strict interpretation for "Previously Identified" in that the identification by the licensee had to be specific to the issue of concern. As modifications and calculation revisions for the RSS system were being implemented after the closure date, there were several issues that the licensee identified after the CMP closure date but before S&L. While credit was not given to the licensee for self-identification of these issues, the team concluded that the licensee's identification of these issues was a strength, indicative of the licensee having a low threshold for problem identification after the completion of the CMP.

3.2.3 Conclusion

DR trends in the areas of calculation control and accuracy, drawings/component information, and the RSS were identified but were not considered safety significant because the types of errors identified, both individually and collectively, would not likely result in the identification of a nonconformance with the unit's licensing and design bases. Consequently, the scope of the ICAVP was not expanded. A strength was identified in the area of having a low threshold for problem identification after the completion of CMP.

3.3 DRs for which S&L Rejected the Licensee's Initial Response

3.3.1 Scope of Review

The team reviewed DRs for which there were multiple interactions between S&L and the licensee to determine whether the multiple interactions could be indicative of an inadequate corrective action process.

3.3.2 Findings

Within the ICAVP process, the licensee is required to respond to all valid DRs. The DR response could be rejected by S&L for a number of reasons including S&L's need for additional information; new questions raised by S&L based on the licensee's initial response; differences between S&L and the licensee on significance level; or differences on deferability or the extent of corrective actions. If S&L and the licensee can not agree on the appropriate closure of the DR, the DR is classified as "Unresolved" and forwarded to the NRC for final adjudication.

There were 204 DRs (out of 974 valid DRs) for which S&L rejected the licensee's initial response (rejected DRs). Of these 204 rejected DRs, 141 were rejected because of the need for additional information by S&L to complete its review, 21 were rejected because S&L raised new questions on the licensee's initial response, 37 were rejected because S&L determined that additional corrective actions were necessary, and 5 were classified as Unresolved at the time of this inspection.

The team performed a detailed review of all 37 DRs that were rejected because S&L determined that additional corrective actions were necessary. These DRs are listed in Appendix F. The team concluded that the additional corrective actions were generally confirmatory in nature or documentation-related issues. For example, in a number of the DRs, the original DR raised a strictly technical issue (the majority of which were Significance Level 3). The licensee's responses to these DRs typically focused on and adequately addressed the technical issues. S&L typically concurred with the licensee's technical evaluation, and downgraded the DR from a Level 3 to a Level 4, but indicated the documents that should be revised to document the licensee's technical resolution. The team concluded that this was an acceptable consequence of the communication protocol in that the licensee's corrective action program adequately addressed the technical issues raised in the DRs. In a number of other DRs, the licensee felt that undocumented engineering judgment was adequate to conclude that a design- or licensing-basis issue did not exist, while S&L indicated that a formal calculation or engineering evaluation would be required for it to reach the same conclusion. The team reviewed the calculations and engineering evaluations performed by the licensee in response to S&L's concerns. The team concluded that the licensee's engineering judgement was generally

validated by the calculations and engineering evaluation, providing additional confidence in the ability of the licensee's corrective action program to address technical issues.

3.3.3 Conclusion

S&L's rejection by the licensee's original responses to 204 DRs did not call into question the adequacy of the licensee's corrective action program.

3.4 Review of Unit 2 Preliminary DRs

3.4.1 Scope of Review

The team reviewed preliminary DRs identified by the Unit 2 ICAVP contractor (Parsons Power Group Inc.) that raised possible programmatic concerns to determine if similar programmatic discrepancies existed at Unit 3.

3.4.2 Findings

In reviewing the preliminary programmatic DRs identified during the ICAVP at Unit 2, the NRC team determined that four of these DRs might affect Unit 3, as discussed in the following paragraphs.

(a) Unit 2 Preliminary DR-0034

This DR identified the potential that design change processes not related to modification may not be adequate in that they did not provide documented evidence of design control measures commensurate with those applied to the original design. Specifically, design change control procedures not related to modifications did not require a 10 CFR 50.59 screening or safety evaluation, nor did they require an independent reviewer. The team reviewed the Unit 3 Engineering Self-Assessment Report, 3ESAR-97-001, "Administrative DCN (Category 8) Review," dated February 8, 1997, and ERC 25212-ER-98-0125, "Administrative/Category 8 DCN Civil/Mechanical Review," Rev. 0, and concluded that the same type of discrepancies were identified and corrected during the Unit 3 CMP.

(b) Unit 2 Preliminary DR-0076

This DR identified potential discrepancies in the Commercial Grade Dedication (CGD) Program. In an internal memorandum dated May 19, 1998, the licensee explained its basis for concluding that the potential issues identified in Unit 2 DR-0076 have little or no impact on Unit 3. Primarily, the licensee concluded that 87 of the 88 examples of potential problems raised in the DR could likely be satisfactorily resolved by providing the ICAVP contractor with additional documentation to demonstrate that there are no actual problems. The team concluded that the issues raised with regard to the CGD Program for Unit 2 should not significantly impact Unit 3. The basis for that conclusion is that the issues raised in the DR are from the same time period and are similar in content to those issues raised in the NRC June 1991, assessment of the Millstone Procurement and CGD Programs. None of the issues raised in that assessment resulted in any operability concerns. Further, the licensee's July 1992, implementation of the Nuclear Management and Resources Council (NUMARC) procurement initiative provides assurance that such documentation problems should not recur.

(c) Unit 2 Preliminary DR-0364

This DR identified possible programmatic issues associated with the Millstone Unit 2 Instrumentation Setpoint and Loop Uncertainty Analysis. This discrepancy was not considered applicable to Unit 3 because of significant differences implemented by the programs between the two units. Westinghouse developed the Unit 3 setpoint and channel uncertainty calculations using the standard accepted Westinghouse methodology. The Unit 2 setpoint and channel uncertainty calculations were developed by the licensee.

(d) Unit 2 Preliminary DR-0538

The DR identified potential weaknesses with the licensee's Measuring and Test Equipment (M&TE) Program, including improper use of calibration equipment and failure to follow procedural guidance. The team reviewed the licensee's program for controlling, calibrating, procuring, and specifying M&TE, and did not identify discrepant conditions with the Unit 3 M&TE Program. The program is implemented in accordance with Station Procedure WC 8, "Control and Calibration of Measuring and Test Equipment, " Rev. 2. During the ICAVP assessments, the NRC inspection team did not identify issues related to calibration.

3.4.3 Conclusion

The potential Unit 2 preliminary programmatic discrepancies identified by the Unit 2 ICAVP contractor did not significantly impact or apply to Unit 3.

4.0 Self-Assessments of Plant Modifications Made During the Current Outage

4.1 Scope of Review

As a result of the failure of the RSS expansion bellows during testing and other modifications made to plant systems during the current outage that resulted in some adverse impacts on system equipment and/or performance, NNECO conducted three self-assessments as part of its corrective actions. These self-assessments were conducted in order to determine whether causal factors in those instances resulted in problems in any of the other modifications made to the plant during the outage. The team reviewed these self-assessments.

4.2 Findings

In reviewing the licensee's self-assessment of plant modifications implemented during the current outage, the NRC team reviewed three self-assessments, as discussed in the following paragraphs.

(a) Outage Modification Review

The team reviewed Self-Assessment 3DE-SA-98-2, "MP3 Current Outage Modification Review," dated April 17, 1998. This self-assessment reviewed the 194 modifications made to the plant during the current outage. This was not an in-depth technical review of each modification; instead, the modifications were reviewed technically to the extent necessary to understand them and evaluate what was done and how it was done against the causal factors. The causal factors that resulted in the Train A RSS pump vent line vibration failure, the air

binding of the boric acid transfer pumps, the damage to the RSS expansion joints, and the replacement of the RHR valves with the incorrect solenoid operators were utilized. The assessment did not identify any single causal factor or group of factors that would point to significant weaknesses in any of the modifications reviewed.

(b) Recirculation Spray System Modification Review

The team reviewed Self-Assessment 3DE-SA-98-12, "Review of Recirculation Spray System - Design Modifications for MP3," dated May 1, 1998. Given the large number of modifications made to the RSS during the current outage, the licensee conducted a review of all 12 of the design changes made to the system. This review verified the adequacy of the subject modifications.

(c) RSS Cubicle Sump Modification

The team reviewed self-assessment, "RSS Cubical Sump Modification Implementation Weakness," Rev. 0. The licensee determined that the RSS cubicle sump pump modification was inadequate because interaction with the RSS pumps was not properly considered during the development of this modification; therefore, the difficulty of developing the modification was considered to be low. The licensee developed a screening process to identify other modifications containing vulnerabilities similar to the RSS cubicle sump pump modification. The licensee reviewed the modifications, focusing on interactions between systems. The licensee also reviewed all of the RSS and AFW system modifications. The licensee's review of approximately 19 modifications did not identify any significant deficiencies.

The team reviewed the following DCRs and verified that the changes were made in accordance with applicable procedures, the plant design and licensing bases were updated as appropriate, and no USQs were introduced as a result of the changes:

- DCR M3-97005, "Revise EOPs (and Design) to Remove Credit for Shutdown Compressors," Rev. 0.

- DCR M3-97007, "Replacement of 3RCS*MV8000A/B Valves," Rev. 0.

- DCR M3-96076, "Replacement of the Emergency Diesel Generator Jacket Water Temperature Control Valve," Rev. 0.

- DCR-M3-97058, "Modify the Logic for Ventilation Supply Fans in the Component Cooling Water Pumps and Charging Pumps Areas," Rev. 0.

The team concluded that the changes were done in accordance with the appropriate procedures, and the documentation contained in the change packages adequately supported the conclusion that the modifications did not result in a USQ.

4.3 Conclusion

The licensee's corrective actions were appropriate and satisfactorily assessed the potential for programmatic weakness in the change processes. These self-assessments provided an additional level of confidence that modifications performed during the outage were acceptable.

Based on the acceptability of the licensee's corrective actions, the team concluded that an expansion of the ICAVP was not warranted.

5.0 Entrance and Exit Meetings

After completing the onsite inspection, the team leader conducted an exit meeting with the licensee on July 7, 1998, that was open for public observation. Appendix B presents a partial list of persons who attended the exit meeting.

Appendix A

Summary of Inspection Results

Violations Identified During the Inspection

Item Number	Type	Section	Status	Title
50-423/98-211-01	VIO	2.1.2(l)	closed	TS Leakage Monitoring Program
50-423/98-211-02	VIO	2.1.2(n)	closed	Failure to Implement Adequate Corrective Actions in Accordance with 10 CFR 50, Appendix B, Criterion XVI
50-423/98-211-03	NCV	2.3.2(b)	closed	Chilled Water Pump Pressure Gauges Did Not Meet ASME Requirements
50-423/98-211-04	NCV	2.3.2(c)	closed	Failure to Ensure that the Containment Isolation Valves Design Basis had been Adequately Maintained
50-423/98-211-05	NCV	2.3.2(d)	closed	Failure to Ensure that the Containment Temperature Design Basis had been Adequately Maintained
50-423/98-211-06	NCV	2.3.2(g)	closed	Failure to Maintain the RHR Low-Pressure Interlock Consistent with TSs
50-423/98-211-07	NCV	2.3.2(h)	closed	Failure to Calibrate the Containment Drain Sump Pump Timer Relays per TS Interval
50-423/98-211-08	NCV	2.3.2(i)	closed	Failure to Set the P-6 Interlock Reset Value Consistent with TSs
50-423/98-211-09	NCV	2.3.2(j)	closed	Failure to Maintain Design-Basis Pipe Break Exclusion Areas
50-423/98-211-10	NCV	2.3.2(k)	closed	Failure to Maintain Design Basis-Fuel Building Fans
50-423/98-211-11	NCV	2.3.2(l)	closed	Failure to Monitor Turbine Building Sump Effluent in Accordance with TSs
50-423/98-211-12	NCV	2.3.2(m)	closed	Failure to Test the ESFAS Permissives in Accordance with TSs
50-423/98-211-13	NCV	2.3.2(n)	closed	Failure to Test Gripper Coil voltage Collapse Response Time In Accordance with TSs
50-423/98-211-14	VIO	2.3.2(q)	closed	Failure to Calculate the Containment Bypass Leakage Rate in Accordance with TSs

Item Number	Type	Section	Status	Title
50-423/98-211-15	NCV	2.3.2(s)	closed	Inadequate Safety Evaluation for Containment Pressure Design Change
50-423/98-211-16	NCV	2.4.2(c) 3.1.3	closed	Failure to Report a Condition Outside of Plant Design Basis per 10 CFR 50.73 (Two Examples)

Unresolved Items, Inspector Followup Items, Licensee Event Reports, and Violations

Item Number	Type	Section	Status	Title
50-423/96-201-09 SIL 79	VIO/SIL	2.1.2(a)	closed	Failure to Ensure That Adequate Design Control Measures Were Established for Verifying and Checking the Accuracy of the Information in the Design Basis Documentation Packages
50-423/97-206-01	VIO	2.1.2(b)	closed	Violation for Failure to Properly Update FSAR in Accordance with 10 CFR 50.72
50-423/97-206-02	VIO	2.1.2(c)	closed	Violation for Failure to Follow Procedures in Accordance With TS 6.8.1
50-423/97-206-03	VIO	2.1.2(d)	closed	Stress Indices for Orifices-Part A
50-423/97-206-03	VIO	2.1.2(d)	withdrawn	Relief Valve Discharge Piping-Part B
50-423/97-206-06	VIO	2.1.2(e)	closed	Violation of 10 CFR 50.59 for Failure to Have Certain Main Steam Valves in EQ Program
50-423/97-206-12	VIO	2.1.2(f)	closed	Inadequate Corrective Action to Correct Identified Discrepancies in Accordance with 10 CFR 50, Appendix B, Criterion XVI
50-423/97-206-13	VIO	2.1.2(g)	closed	Failure to Have Adequate Procedures in Accordance with TS 6.8.1
50-423/97-206-20	VIO	2.1.2(h)	closed	Violation of 10 CFR 50, Appendix B, Criterion XVI for Failure to Identify and Take Corrective Actions for Air in the RSS Piping
50-423/97-206-21	VIO	2.1.2(i)	closed	Failure to Vent Piping in Accordance with TSs
50-423/97-206-22	EEI	2.1.2(j)	closed	Apparent Violation of TS 6.8.4 and 10 CFR 50.55A for Failure to Leak Check RWST Boundary Valves
50-423/97-209-01	VIO	2.1.2(k)	closed	Inadequate Procedures for RTD Response Time Testing
50-423/97-209-02	EEI	2.1.2(l)	closed	TS Leakage Monitoring Program

Item Number	Type	Section	Status	Title
50-423/97-209-03	VIO	2.1.2(m)	closed	Failure to Ensure that the FSAR Contained the Latest Information-Parts A and B (RSS pump seal cooling and FSAR description of SGTR)
50-423/97-209-03	VIO	2.1.2(m)	withdrawn	Failure to Ensure that the FSAR Contained the Latest Information-Part C (FSAR Table 15.0-8 dose rates)
50-423/97-209-06	EEI	2.1.2(n)	closed	Failure to Implement Adequate Corrective Actions in Accordance with 10 CFR 50, Appendix B, Criterion XVI
50-423/97-209-09	VIO	2.1.2(o)	closed	Failure to Meet the Limiting Condition for Operation for TS 3.1.2 for the Train A EDG
50-423/97-209-10	VIO	2.1.2(p)	closed	Inadequate Design and Installation of Temporary Modification Contrary to 10 CFR 50, Appendix B, Criterion V
50-423/97-209-11	VIO	2.1.2(q)	closed	Inadequate Control of Calculations of Temporary Modifications Contrary to 10 CFR 50, Appendix B, Criterion V
50-423/97-209-12	VIO	2.1.2(r)	closed	Failure to Adequately Translate Design Bases into a Temporary Modification in Accordance with 10 CFR 50, Appendix B, Criterion III
50-423/97-209-13	VIO	2.1.2(s)	closed	Failure to Include Safety Evaluations with Records of FSAR Changes Where Safety Evaluations are Required by 10 CFR 50, Appendix B, Criterion III
50-423/97-210-03	VIO	2.1.2(t)	closed	Violation for Inadequate Corrective Action
50-423/97-210-04	VIO	2.1.2(u)	closed	Violation for Inadequate Design Control
50-423/97-210-09	VIO	2.1.2(v)	closed	Inadequate Procedures
50-423/98-205-02	VIO	2.1.2(w)	closed	Failure to Implement Adequate Corrective Action, Two Examples
50-423/98-205-03	VIO	2.1.2(x)	closed	Failure to Implement Adequate Corrective Action
50-423/97-206-04	URI	2.2.2(a)	closed	URI to Resolve Check Valve Single Failure Requirements
50-423/97-206-05	URI	2.2.2(b)	closed	URI Concerning Qualification of RSS Isolation Valve Seats
50-423/97-206-07	URI	2.2.2(c)	closed	URI Concerning the Adequacy of the Unit's Voltage Surge Protection
50-423/97-206-09	URI	2.2.2(d)	closed	URI Concerning One-Hour Batter Connection

Item Number	Type	Section	Status	Title
50-423/97-206-17	URI	2.2.2(e)	closed	URI Concerning the Effects of Thermal Expansion of Rigidly Restrained Steel
50-423/97-209-07	URI	2.2.2(f)	closed	Verification That Appropriate Operator Actions are Defined to Isolate Steam Generator Blowdown in the Event the Automatic Isolation Valves Fail
50-423/97-209-16	URI	2.2.2(g)	closed	FSAR Disagreements and the Need for Safety Evaluations
50-423/97-210-06	URI	2.2.2(h)	closed	Updating of FSAR on the Use of Square Root Sum of Squares Modeling
50-423/97-210-07	URI	2.2.2(i)	closed	Review of Revised Calculation for RSS Heat Exchangers
50-423/97-210-08	URI	2.2.2(j)	closed	Review of Justification for Design Value and Embed Calculation
50-423/97-206-10	IFI	2.2.2(k)	closed	IFI to Verify Labeling of Hydrogen Concentration Indicators
50-423/97-206-14	IFI	2.2.2(l)	closed	IFI to Correct Calculation Errors in Accordance with CR M3-97-3169
50-423/97-206-18	IFI	2.2.2(m)	closed	IFI Concerning the Adequacy of the Licensee's Proposed Change to the Basis of Pressurizer Level
50-423/97-206-19	IFI	2.2.2(n)	closed	IFI Concerning the Resolution of Charging Pump Are Temperature Inconsistencies
50-423/97-209-05	IFI	2.2.2(o)	closed	Unanalyzed Low-Level Fault Current Review
50-423/97-209-14	IFI	2.2.2(p)	closed	Licensee to Upgrade Procedure NGP 3.12, Safety Evaluations
50-423/97-209-15	IFI	2.2.2(q)	closed	Verification of Appropriate Operator Actions Following an Auxiliary Feedwater System Pipe Break Upstream of the Cavitating Ven
50-423/97-210-01	IFI	2.2.2(r)	closed	Maximum Acceptance Value for Pump IST
50-423/97-210-02	IFI	2.2.2(s)	closed	Change of Switchover Time From 10 to 25 Minutes
50-423/97-210-05	IFI	2.2.2(t)	closed	Vulnerability of Radiation Monitors to Flooding
50-423/97-210-10	IFI	2.2.2(u)	closed	Verification of Operator Training on RSS Modifications
50-423/97-210-11	IFI	2.2.2(v)	closed	Risk Significance of RSS Functions

Item Number	Type	Section	Status	Title
50-423/96-046-00	LER	2.3.2(a)	closed	Time Response of Containment Fuel Drop Radiation Monitor Non Conservative
50-423/96-050-00	LER	2.3.2(b)	closed	Range of Control Building Chilled Water Pump Suction Pressure Gages Used for Surveillance Not in Accordance with ASME Chapter XI Requirements
50-423/97-13-00	LER	2.3.2(c)	closed	Failure to Meet General Design Criteria 57 for the Main Steam Supply containment Isolation Valves to the Auxiliary Feedwater Pump Turbine
50-423/97-020-00	LER	2.3.2(d)	closed	Elevated Containment Temperature Due to Reduced Cooling Restrict Ability to Enter Containment and Operate Equipment Required for Cold Shutdown During Certain Fire Events
50-423/97-029-00	LER	2.3.2(e)	closed	Design-Basis Concern on Steam Generator Tube Rupture Analysis of Main Steam Pressure Relief Bypass Valves
50-423/97-030-00	LER	2.3.2(f)	closed	Non-Conservatism in the Low-Temperature Overpressure/Temperature Limit Curves in TSs
50-423/97-031-00	LER	2.3.2(g)	closed	RHR Valve Low-Pressure Open Permissive Setting Set Non-Conservatively
50-423/97-032-00	LER	2.3.2(h)	closed	Calibration of Pumped Capacity Monitoring System Timers not in Accordance with 18-Month Surveillance Requirement of TSs
50-423/97-033-00	LER	2.3.2(i)	closed	P-6 Bistable Reset Function Set Non-Conservatively
50-423/97-036-00	LER	2.3.2(j)	closed	Piper Breaks in Exclusion Zoned Differ From Those Provided in the FSAR
50-423/97-036-01	LER	2.3.2(j)	closed	Piper Breaks in Exclusion Zoned Differ From Those Provided in the FSAR
50-423/97-038-00	LER	2.3.2(k)	closed	Single Failure Potential for Fuel Building Fans
50-423/97-039-00	LER	2.3.2(l)	closed	Inoperability of Turbine Building Sump Effluent Monitor Due to a Failure to Recognize a Design Deficiency the Resulted in a TS Violation
50-423/97-043-00	LER	2.3.2(m)	closed	Inadequate TS Surveillance for the ESFAS and Reactor Trip System Instrumentation Interlocks
50-423/97-045-00	LER	2.3.2(n)	closed	Reactor Trip System Response Time Testing not in Verbatim Compliance with TSs

Item Number	Type	Section	Status	Title
50-423/97-057-00	LER	2.3.2(o)	closed	Emergency Diesel Generator Inoperable Due to Ventilation Alignment Prohibiting Tornado Recovery
50-423/98-002-00	LER	2.3.2(p)	closed	Verification of ECCS Piping Full of Water per TSs not being met
50-423/98-007-00	LER	2.3.2(q)	closed	Containment Leakage in Excess of TS Limits
50-423/98-015-00	LER	2.3.2(r)	closed	Historical Failure to Enter TS for Manual Control of Automatic Safety Function
50-423/98-029-00	LER	2.3.2(s)	closed	Supplementary Leak collection and Release System Bypass Leakage May Result in Unaccounted Control Room and Technical Support Center Doses

Appendix B

Exit Meeting Attendees

NAME	ORGANIZATION
Michael Brothers	Vice President, Operations
Martin Bowling	MP2 Recovery Officer
Raymond Necci	Director, Configuration Management Plan
Harry Miller	Director, Unit Services
Joe Fougere	Manager, ICAVP
Carol Pietriyk	ICAVP Team oordinator
Brain Krauth	Nuclear Special Project
William Travers	NRC/Director/SPO
Gene Imbro	NRC/Deputy Director/ICAVP/SPO
Peter Koltay	NRC/Acting Branch Chief/ICAVP/SPO
Stephen Tingen	NRC/ICAVP/SPO
John Nakoski	NRC/ICAVP/SPO
Tony Cerne	NRC/Senior Resident Inspector
Beth Korona	NRC/Resident Inspector

Appendix C

Confirmed Level 3 DRs Reviewed

DR No.	Discrepancy Titles
0001	Consistency with Technical Specifications
0006	PORC/SORC review of Minor Modifications
0029	Service Water Pumphouse Ventilation Calculation Discrepancy
0035	Not all Service Water heat exchangers are included in the surveillance procedure per LER 90-020
0051	Calculation 12179-CFSK-732E-E66 did not properly qualify embedded plate E66
0331	Filter Unit Drain Valve Normal Position
0355	Cable Routing is Not Consistent with TSC2
0434	Conclusions Documented in Technical and Reportability Evaluations for ACR No. 12327
0588	SLCRS and ABVS Filter Units Adsorbent Cooling
0624	Storage of Reference Material as QA Records
0639	Not Obtaining NRC Relief for Temporary Non-Code Repair
0667	Calculation P(B)-1130 Temporary Ventilation for CCP Pump Area
0669	Fan 3HVR*FN14A/B Motor Requirements
0670	CCP and CHS Area Ventilation System Winter Operation
0686	ABVS Filter Unit Bypass Leakage
0687	Fan Blade Missiles
0762	PDCR MP3-92-024 Fan 3HVR*FN12B Vibration Test
0795	Specification 2176.430-141 and 2170.430-140 Vibration Test Requirements
1011	Unreviewed Safety Questions Concerning the MP-3 Emergency Diesel Generators
1016	Secondary Containment Bypass Leakage Penetrations
1026	Revision of Calculation US(B)-353 for DCR M3-97045

Appendix D

Non-Discrepant and Previously Identified Discrepancy Reports Reviewed

DR No.	Discrepancy Title	Category
0011	Failure to Implement Corrective Action in ACR M3-96-035	Non-Discrepant
0022	Westinghouse Comments in FSAR 15.1	Non-Discrepant
0098	Qualification of End Loads on RSS Expansion Joint Not preferred	Non-Discrepant
0123	ESK-6RR and LSK-22-12B Discrepancy	Non-Discrepant
0132	Inadequate Corrective Action for ACR M3-96-032	Non-Discrepant
0157	High-Frequency Modes Not Accounted For in Response Spectra	Non-Discrepant
0175	FSAR Description MCC & RCC Area Booster Pump	Non-Discrepant
0206	Drawing Discrepancy LSK-27-12C	Non-Discrepant
0211	Drawing Discrepancy QSS 027-1-2; 028-1-2	Non-Discrepant
0213	Drawing Discrepancy-Schematic QSS 054A, B	Non-Discrepant
0234	Drawing Discrepancy - LSK-9-10G, L	Non-Discrepant
0336	Pipe Support Discrepancies	Non-Discrepant
0442	Equipment Qualification Documents Discrepancy	Non-Discrepant
0459	SLCRS & ABVS Filter Unit DP Switches	Non-Discrepant
0499	Inconsistent Logic Between Drawings LSK-27-11G and LSK-11F With Respect to Bypass Alarms	Non-Discrepant
0601	Setting of Bus Tie Overcurrent Relays for Buses 34C & 34D	Non-Discrepant
0628	Simulated Test Pressure Selected For Performance of TS Surveillance	Non-Discrepant
0644	SLCRS and ABVS Filter Unit Drain Lines	Non-Discrepant
0663	Operation of EDG Enclosure Supply Fan B	Non-Discrepant
0712	NPSH for ECCS Pump During Recirculation Mode	Non-Discrepant
0728	Inconsistencies Between SER and Calculation with Respect to Sump Water Level	Non-Discrepant
0761	Setpoint Change Without Modification	Non-Discrepant
0779	Class 1E Cable Qualified to DOR Guidelines	Non-Discrepant
0780	Class IE Cable Test Discrepancy	Non-Discrepant

DR No.	Discrepancy Title	Category
0788	Value Used for Motor Contribution Fault is not Recorded	Non-Discrepant
0797	Cable Lengths Used do not Agree with Calculation	Non-Discrepant
0836	Calculation Methodology vs RG 1.105 Commitment	Non-Discrepant
0935	Class 1E BIW Cable Qualified to DOR Guidelines	Non-Discrepant
0973	Corrective Action Incorrectly Scheduled	Non-Discrepant
0976	Insufficient Data Provided for Corrective Action Implementation on ACR M3-96-0181	Non-Discrepant
0981	SGTR EOP Instruments Need to BE RG 1.97	Non-Discrepant
1067	DCR M3-97-102 & FSARCR 97-MP3-569 are not Consistent	Non-Discrepant
0004	Design Control Procedure Control and Use of Vendor Manuals	Previously Identified
0008	Design Control Procedure Control & Use of Vendor Manual	Previously Identified
0010	Licensing Document Fuel Cycle Analysis	Previously Identified
0109	Pipe Crack Postulation Discrepancy	Previously Identified
0200	Mass Flow Capacity of Turbine Bypass Control Valves Is Not Verified	Previously Identified
0208	Fast Closure Time for Turbine Control Valves is Not Verified	Previously Identified
0261	Discrepancy between FSAR Section 15.4.6.1 and Operating Procedure for Boron Dilution	Previously Identified
0266	Minimum RWST Level During ECCS Suction Switchover	Previously Identified
0292	Inconsistencies Between FSAR 6.1-1 and 6.3-4	Previously Identified
0316	Temperature Used in Stress Calculation Not Consistent	Previously Identified
0324	Inconsistencies Between FSAR and Vendor Specifications and Drawings	Previously Identified
0384	Verification of Time Critical Operator Actions	Previously Identified
0437	Lack of Documentation to Demonstrate Qualification	Previously Identified
0592	Design Requirement for RSS after Main Steam Line Break	Previously Identified
0704	RWST Level Calculation Discrepancies	Previously Identified
0711	Minimum C Initiation Time in FSAR Table 6.2-60	Previously Identified
0794	Inconsistencies Between Specification 2362.200-164 and Calculation	Previously Identified

DR No.	Discrepancy Title	Category
0852	Service Water Strainer Tube Sheet Replacement	Previously Identified
0865	Inadequate Safety Evaluation	Previously Identified
0866	Incorrect ACR Closure Relating to RSS Spray Piping and Supports	Previously Identified

Appendix E

Closed Confirmed Level 4 Discrepancy Reports Reviewed

DR No.	Discrepancy Title	Level
0005	Internal Procedure Inconsistencies	4
0012	Inconsistencies in MP-3 FSAR, Technical Specifications, and Westinghouse Safety Analysis	4
0016	Incorrect Values in FSAR Table 15.7-9	4
0018	Misidentified Pump Designator in Step of Procedure	4
0024	Westinghouse Comments on FSAR Section 15.4	4
0025	FSAR Discrepancy Regarding ATWS Turbine Impulse Pressure Signal	4*
0027	Incorrect Nuclide Identified in FSAR Section 15.0	4
0028	Inconsistency in FSAR Section 15.7	4
0032	Water hammer Analysis - Error in Nozzle Thrust & Centrifugal Force Calculation	4
0034	Piping System Data Package - Reference Error	4
0040	Walkdown Discrepancies of the SWP in the Diesel Generator Building	4*
0041	Upper Tier to Lower Tier Drawings Review for SWP in Pump House	4
0042	Walkdown Discrepancies of the SWP in the Pump House - Trains B & D	4
0046	Pipe Support Calculation Discrepancy	4
0050	Liner Plate Calculation Discrepancy	4*
0053	Pipe Support Calculation NP(F)-Z019B-405 Discrepancy	4
0055	Incomplete Adverse Condition Report	4
0056	Pipe Support Calculation NP(F)-Z19B-035 Discrepancy	4
0060	Use of an Unapproved Code Case	4
0061	Conduit Support ES-1219 not in Agreement with Design Drawing	4
0062	Inadequate Support of Conduit 3CX763NA	4
0064	Duct Conduit Numbering	4
0065	Tray Support Location Drawing Inconsistent with Field Installation	4
0066	Bonding Conductor Between Trays not Installed	4

DR No.	Discrepancy Title	Level
0070	Identification of Tray on Tray Location Drawing is Incorrect	4
0074	Upper Tier to Lower Tier Drawing Review for RSS-QSS in ESF Building	4
0080	Incorrect Loads Used to Evaluate Containment Penetration 110	4
0087	Discrepancy Between Vendor Information and Design Calculations	4
0093	Pipe Support Calculation 12179-NP(F)- Z79B- 161 Discrepancy	4
0094	Pipe Support Calculation Discrepancy	4
0095	Structural Steel Calculation Discrepancy	4
0101	QSS/RSS Pump Room Ventilation Calculation Discrepancy	4
0114	Upper Tier to Lower Tier Drawing Review for SWP in ESF Building	4
0117	Notification of Failed Surveillance Test for 3MSS*RV26C	4*
0129	Adverse Condition Report ACR M3-96-0272 Discrepancy	4
0138	Tornado Damper Safety Classification	4*
0140	Emergency Diesel Generator Enclosure Tornado Dampers	4
0142	Unsupported Design Assumption	4
0172	Design Basis Summary Requirement for Valves 3SWP*MOV130A & B is Not in Agreement with Drawings	4
0181	Configuration Discrepancies Between FSAR Sections 15.6.2 , FSAR Table 6.2-65, and P&IDs	4
0183	The Reference for the Dispersion Parameter Used in the Fuel Handling Accident is Not Current	4
0196	Missing Guidance in the Containment Inspection Procedure	4
0201	Inappropriate Commercial Grade Procurement	4*
0205	Drawing Discrepancy - LSK-27-12B	4
0207	Drawing Discrepancy - LSK-27-12D	4
0212	Drawing Discrepancy - Schematics 3QSS-040A, B	4
0214	Drawing Discrepancy - Schematic 3QSS-060	4
0229	Missing Equipment Tags	4
0240	Drawing Discrepancy - Schematics 3SWP-152A, B	4

DR No.	Discrepancy Title	Level
0255	Reactor Coolant Pump Underspeed - Calibration Data Conversion (Discrepancy for Loop RCS*SY495)	4
0269	Chemistry Action Limits for RWST Not Specified.	4
0485	Installation Not in Accordance with Design Documents	4
0508	Discrepancy with respect to PDCR MP3-90-106, 3QSS*AOV27/28 Engaging Key Removal	4*
0546	Inadequate Support of Conduit	4
0549	Installed Supports Not in Agreement with Drawings	4
0553	Design Documents Not in Agreement	4
0556	Inadequate Spacing	4
0557	Wall Penetration Sealing Not in Accordance with Commitment	4
0558	Inadequate Raceway Protection	4
0577	Installed Conduit Type Not in Agreement with Design Document	4
0772	Closure of Design Deficiency Report (DDR-1027)	4*

*Indicates DRs originally identified as level 3

Appendix F

Rejected Response Discrepancy Reports Reviewed

DR No.	Discrepancy Title	Level
0081	Incorrect Operating Temperature Used in Stress Analysis.	3
0328	SLCRS and ABVS Filter Unit Drain Valves	3
0588	SLCRS and ABVS Filter Units Adsorbent Cooling	3
0639	Not Obtaining NRC Relief for Temporary Non-Code Repair	3
0667	Calculation P(B)-1130 Temporary Ventilation for CCP Pump Area	3
0670	CCP & CHS Area Ventilation System Winter Operation	3
0738	Supply and Return Air Registers	3
0835	Incorrect Calculation Methodology for: NSP-107,108,109,124-HVR, & SP-3HVR-29	3
0039	Upper Tier to Lower Tier Drawings Review for SWP Support in Diesel Generator Building	4
0052	Pipe Support Calculation Discrepancy	4
0057	Liner Plate Pad Calculation Discrepancy	4
0122	Schematic and Wiring Drawing Discrepancies at Motor Control Centers	4
0139	Effect of Fluid Transient Induced Header Movements on Branch Piping not Considered	4
0166	Calculation SP-3SWP-1 Methodology vs. RG 1.105 Commitment	4
0180	Technical Justification for Changes to FSAR	4
0246	Undocumented Structural Connections to Tray Supports	4
0247	Difference in Material Type Between TS02 and Conduit Support Log	4
0249	Missing Support Details	4
0269	Chemistry Action Limits for RWST not Specified	4
0287	Calculation 3-ENG-106 Data Discrepancy	4
0291	SLCRS Filter Housing Design Pressure	4
0294	Lack of Documentation for Qualification of Tie Rods	4
0297	Design Pressure in Calculation P(R)-1171	4
0315	SLCRS HEPA Filter Airflow Rating and Pressure Drop	4

DR No.	Discrepancy Title	Level
0418	The Minimum Wall Calculations were Reviewed and Several Discrepancies were Found	4
0428	The Setpoint for (7) Valves Could not be Verified in Calculation SP-3SWP-29, Rev. 0, CCN #1	4
0480	ESF Filter Unit Compliance with RG 1.52, Rev. 2, Position C.3.g	4
0509	Areas Maintained at a Negative Pressure by SLCRS and ABVS	4
0575	Auxiliary Building Filter Unit Charcoal Absorber Face Velocity	4
0616	RSS Pump Actuation Time in Design Basis Summary Document	4
0623	Criteria Determining Which Vendor Technical Manuals Need Upgrading by Startup	4
0658	ABVS Filter Unit Electric Heater Capacity	4
0659	SLCRS Filter Unit Electric Heating Coil Capacity	4
0724	SLCRS and ABVS Filter Unit Backup Adsorbent Cooling Mechanism	4
0823	P&ID EM-117A Rev. 10 , Emergency Generator Fuel Oil System, Review of Piping Line Sizes	4
0971	Calculation SWS-MOV-1380M3	4
1087	CR M3-96-1222 Corrective Action Implementation	4

Appendix G

List of Acronyms

ABVS	Auxiliary Building Ventilation System
ACR	Adverse Condition Report
AFW	Auxiliary Feedwater System
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AWO	Automated Work Order
BOM	Bill of Materials
CAR	Containment Air Recirculation
CCN	Calculation Change Notice
CCW	Component Cooling Water
CGD	Commercial Grade Dedication
CFR	*Code of Federal Regulations*
CHS	Charging System
CMP	Configuration Management Plan
CR	Condition Report
DBDP	Design Basis Documentation Package
DCM	Design Control Manual
DCN	Design Change Notice
DCR	Design Change Record
DR	Discrepancy Report
EAR	Engineering Assessment Report
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EEI	Escalated Enforcement Item
EOP	Emergency Operating Procedure
EQ	Environmental Qualification
ERC	Engineering Record Correspondence
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FSAR	Final Safety Analysis Report
FSARCR	Final Safety Analysis Report Change Request
GDC	General Design Criterion
ICAVP	Independent Corrective Action Verification Program
IFI	Inspector Followup Item
IN	Information Notice
IR	Inspection Report
ISFR	Integrated System Functional Review
IST	Inservice Test

LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss-of-Coolant-Accident
LOP	Loss-of-Power
MMOD	Minor Modification
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
M&TE	Measuring and Test Equipment
NCV	Noncited Violation
NNECO	Northeast Nuclear Energy Company
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam System Supplier
PDR	Public Document Room
PMMS	Production Maintenance Management System
PM	Preventive Maintenance
OP	Operations Procedure
QSS	Quench Spray System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RSS	Recirculation Spray System
RSST	Reserve Station Service Transformer
RWST	Reactor Water Storage Tank
S&L	Sargent & Lundy
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIL	Significant Items List
SLCRS	Supplementary Leak Collection and Release System
SP	Surveillance Procedure
SRSS	Square root of the sum of the squares
SW	Service Water
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TRM	Technical Requirements Manual
TS	Technical Specification
TSC	Technical Support Center
URI	Unresolved Item
USQ	Unreviewed Safety Question
VIO	Violation