U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-336/88-07

Docket No. 50-336 License No: DPR-65

Licensee: Northeast Nuclear Energy Company
P.O. Box 270
Hartford, CT 06101-0270

Facility: Millstone Nuclear Power Station, Waterford, Connecticut

Inspection at: Millstone Unit 2

Dates: March 22, 1988 through May 2, 1988

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Reporting
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E. C. McCabe, Chief, Reactor Projects Section 1B — 5/26/88 Date

Inspection Summary: March 22 - May 2, 1988 (Report 50-336/88-07)

Areas Inspected: Routine NRC resident and region-based inspection of: plant operations; surveillance; maintenance; radiation protection; physical security; outage activities; Temporary Instructions (TIs) 2500/19 and 2515/94; motor-operated valve testing; plant incident reports (PIRs) and Licensee Event Reports (LERs); allegations; and periodic and special reports.

Results: No unsafe operational conditions were identified. One violation and one deviation were identified involving the calibration of the control room ventilation radiation monitors (Section 8.2). Additional followup is warranted on containment boric acid buildup, PORV response time (TI 2500/19), and on-going motor-operated valve testing.

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DETAILS

1.0 Persons Contacted

Mr. S. Scace, Millstone Station Superintendent
Mr. J. Keenan, Unit 2 Superintendent
Mr. J. Riley, Unit 2 Maintenance Supervisor
Mr. F. Dacimo, Unit 2 Engineering Supervisor
Mr. D. Kross, Unit 2 Instrument and Controls
Mr. J. Smith, Unit 2 Operations Supervisor
Mr. M. Wilson, Unit 2 Operator Training Supervisor

The inspector also contacted other members of the Operations, Radiation Protection, Chemistry, Instrument and Control, Maintenance, Reactor Engineering, and Security Departments.

2.0 Summary of Facility Activities

Millstone Unit 2 operated at full power until April 8, when control element assembly (CEA) #22 dropped into the core during preparations to shut down for investigation of leakage inside containment. The licensee declared control element drive mechanism (CEDM) #22 inoperable. Plant shutdown began early on April 9, and CEA #4 dropped into the core during control rod insertion. The plant was in cold shutdown on April 11, repairing leaks in the reactor coolant system (RCS), specifically 2-SI-237 (safety injection check valve), 2-MS-11A (No. 1 S/G surface blowdown valve), 2-RC-402 (Power operated Relief valve) and 2-SI-707 (No. 2 safety injection tank sample valve). The licensee repaired CEDM #4, #22 and #5 (gripper coils were replaced) and cleaned boric acid buildup from the CEDM coolers, containment air recirculation coolers (CARs), and the CEDM ventilation shroud. The reactor was taken critical on April 14. Between April 15-17, power ascension testing was completed, and on April 18 the unit was at full power. The unit remained at full power throughout the rest of the inspection period.

3.0 Plant Tours (71707)

The inspector observed plant operations during regular and backshift tours of the following areas:

Control Room	Auxiliary Building
Vital Switchgear Room	Enclosure Building
Turbine Building	Intake Structure

Control Room instruments were observed for correlation between channels, proper functioning, and conformance with Technical Specifications. Alarm conditions in effect and alarms received in the control room were reviewed and discussed with operators. Posting and control of radiation and contamination areas, and control of high radiation areas were inspected. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication and equipment status. Records included various operating logs, turnover sheets, and tagout logs. Backshift inspections were performed on April 9 (7:00 a.m.), April 10 (4:30 a.m.), April 14 (3:45 p.m.), and April 25 (6:00 p.m). Routine power operations and outage activities were observed. No unacceptable operational conditions were observed.

3.1 Safety System Operability Review (71710)

The Low Pressure Safety Injection (LPSI) and Shutdown Cooling systems were reviewed for operability by examining documentation, plant conditions, maintenance (preventive and corrective), control room instrumentation and controls, compliance with technical specifications, training, and operating practices.

3.1.1 Documentation

The LPSI system is described in Final Safety Analysis Report (FSAR) Section 6.3, "Safety Injection System." The Shutdown Cooling System is described in FSAR Section 9.3, "Shutdown Cooling System." FSAR material was reviewed against plant procedures and compared with plant configurations and found to be accurate. Mechanical components and instrumentation for LPSI and Shutdown Cooling Systems are shown in Plant and Instrument Drawing (P&ID) 25203-26015, Rev. 20. The system configuration was reviewed during plant tours, utilizing the "valve-lineup" described in SP-2604L and M, Rev. 3, "LPSI Alignment Check and Valve Op. Test." No inadequacies were noted.

3.1.2 Plant Cleanliness

Plant areas for LPSI and shutdown cooling were inspected for housekeeping, radiological controls, and general equipment cleanliness. The following areas were inspected:

- Safeguards Rooms
- 5-Foot West Penetration Room
- Charging Pump Area
- Refueling Water Storage Tank (RWST)

All major low pressure safety injection (LPSI) and Shutdown Cooling components were found to be operable in the standby mode, and in good condition. Housekeeping was found to be good except in the enclosed RWST areas. The two RWST enclosures, located adjacent to the RWST, were poorly cleaned and showed excessive debris and tools. Inspector observations were discussed with licensee personnel. Subsequent inspections found the areas to be satisfactory.

The RWST pipe chase, which is accessed over the roof of an RWST enclosure, contained standing water and miscellaneous debris. Following inspection, the licensee was requested to perform a health physics survey of two valves in the pipe chase because of boric acid crystals on the valve stems. The results showed a count rate of 12,000 dpm/100 sq-cm. Since no posting of the area as "contaminated" was evident, this item was referred to the licensee. The inspector reviewed SHP 4906 Rev. 4, "Posting of Radiological Controlled Areas." The procedure requires plant areas to be posted as contaminated when general areas are in excess of 1000 dpm/100 sq-cm. The inspector identified no inadequacies with respect to procedural or 10 CFR Part 20 requirements. To this end the inspector expressed concern about localized contaminated components in non-radiologically controlled areas. The licensee committed to continue efforts to minimize undetected contaminated components. The inspector will follow these activities in future inspections and had no further questions at this time.

3.1.3 Maintenance

Review of instrumentation and controls (I&C) preventive maintenance (PM) records showed that LPSI and Shutdown Cooling instrumentation undergo routine PMs. Licensee records for LPSI flow instrumentation (FT/FE 312, 322, 332 and 342) and Shutdown Cooling flow instrumentation (FT/FE 3023 and 3024) were found to be complete. Identified I&C Maintenance activities for LPSI were reviewed. Only one item, a leaking Swagelock fitting on PI/PT-302X ("A" LPSI pump discharge pressure), was outstanding. This item was identified on February 25, 1988.

Pumps, valves, and valve motor-operators in the LPSI and Shutdown Cooling systems undergo routine PMs. One mechanical component had an outstanding identified maintenance item: LPSI Pump "B" was identified on October 20, 1987 as having a casing leak along several studs. Overall, the inspector concluded that the licensee had good maintenance on the LPSI and Shutdown Cooling systems.

During LPSI and Shutdown Cooling system inspection, it was noted that a number of valves exhibited varying amounts of crystalline boric acid buildup. The licensee plans to correct the buildup, and the inspector will re-examine the results during routine inspection.

3.1.4 Control Room Interface

The principal control location for LPSI and Shutdown Cooling is Control Room Panel C01. During the most recent refueling outage, panel C01 was repainted, and a new flow mimic diagram was installed as part of the control room design review. In addition, analog LPSI discharge pressure and flow instrumentation were replaced with digital instrumentation.

Two control room alarms are associated with LPSI/Shutdown Cooling: Alarm Windows A2 and B2 (Panel C01) indicate overload/trip conditions for LPSI Pumps A and B, respectively. The Control Room Annunciator Book (CRAB) describes the initiating devices for the LPSI overload/trip alarm as electrical relays 51X and 3. Procedure OP 2307, "Low Pressure Safety Injection System," Rev. 8, Section 8, describes the initiating devices for LPSI pump trip/overload as relays 51X, 3 or 74. The inconsistency between CRAB and OP 2307 was referred to the licensee for corrective action. The inspector had no further questions.

3.1.5 Compliance with Technical Specifications (TSs)

Licensee procedures SP-2604D, SP-2604C, SP-2604L, and 2604M address compliance with TS 4.5.2 for LPSI Surveillance and with TS 4.9.8 for Shutdown Cooling Surveillance. A sample of completed surveillance forms was reviewed with regard to completion dates and test results. The inspector concluded that the licensee is in compliance with TS 4.5.2 and 4.9.8.

TS 4.5.2a.10 requires certain valves to be verified in a designated position with the power to the valve operator removed. Valves 2-SI-659 and 660 are addressed in TS 4.5.2a.10 and are required by procedure SP2604L (M) to be locked open. Valves 2-SI-659 and 660 are air-operated valves controlled by separate solenoid valves. These valves provide a minimum flow path for LPSI and Containment Spray pumps following pump start and prior to injection. On a recirculation signal, the valves must be closed to prevent water from the containment sump from being pumped to the Refueling Water Storage Tank.

The valves are controlled from panel C01 where keylock switches (HS3659A and HS3660A) lock open and remove power from valves 2-SI-659 and 2-SI-660. The use of these keylock switches is described in FSAR Section 6.3.3.1. Inspection found that the requirements of TS 4.5.2a.10 and SP 2604 L (M) are satisfied for valves 2-SI-659 and 660. The inspector had no further questions.

3.1.6 Training

The C01 panel on the Unit 2 simulator was found to be almost identical to the control room panel C01 with regard to recent modifications. Differences were that the simulator panel did not have the new flow mimic and the digital discharge pressure instrumentation for the LPSI pumps. Otherwise, the improvements to control room panel C01 performed during the most recent refueling outage were reflected on the Unit 2 simulator panel.

The following documents were reviewed:

- Systems Description - LPSI-M2-OP-PRI-2307
- Lesson Plan 2306-9 - LPSI

The above documents were found to be thorough, well prepared and consistent with plant procedures and observations made at Millstone Unit 2.

Discussions were held with the licensee concerning understanding and preventing loss of shutdown cooling because of air binding in the LPSI pumps (Generic Letter 87-12, "Loss of RHR While RCS is Partially Filled"). The following documents were reviewed:

- Loss of Shutdown Cooling (RQ2-310-1(S))
- Operating Events (M2-OP-REQ-SOVR-87-2)

The "Loss of Shutdown Cooling" Lesson Plan involved using the Unit 2 simulator to demonstrate loss of shutdown cooling during a partial RCS drain down as a result of air binding of a LPSI pump. The "Operating Events" document contained a description and critique of an April 1987 loss of shutdown cooling at Diablo Canyon which resulted in boiling in the reactor.

The inspector concluded that the licensee had a heightened awareness of potential causes and consequences of loss of shutdown cooling. The inspector had no further questions in regard to this matter.

3.1.7. Operations

A representative of the licensee's operations staff was questioned about measures to prevent shutdown cooling loss during partial RCS drain down. The following were highlighted:

Use of Closed Circuit Television (CCTV) - During refueling and partial RCS drain-down, reactor water level is monitored via a length of Tygon tubing. The Tygon tube is monitored in the control room via CCTV, allowing operators to take prompt action on unexpected level changes.

Vacuum Pump - During partial drain-down of the RCS for nozzle dam placement or RCS Pump Seal maintenance, the water level in the RCS is drained down below the high point of the Shutdown Cooling System piping. In that case, a vacuum pump, taking suction from the Shutdown Cooling system piping highpoint, is operated twice per shift to assure that the shutdown cooling system remains full.

Revised Abnormal Operating Procedure (AOP) - A revised AOP is under preparation to incorporate all lessons learned to date regarding the loss of shutdown cooling during RCS drain-down. This AOP is expected to be issued prior to the next refueling outage.

In addition to the discussion, conduct of the following surveillances was observed:

SP 2604D-1, "Low Pressure Safety Injection Pump Operability Test - Facility II"

SP 2604M-1, "Facility II Low Pressure Safety Injection (LPSI) System Electrical Alignment Check"

No inadequacies were noted.

Based upon the inspection of the LPSI and Shutdown Cooling Systems, the following was concluded:

The components of LPSI and Shutdown Cooling Systems were operable and in overall good condition.

Inspected plant areas showed good housekeeping practices with the exception of the RWST pipe chase.

For the case of the RWST pipe chase, the licensee addressed the potential for corrosion due to standing water and properly implemented plant procedures concerning posting of radiologically contaminated areas.

Programs for completing identified and preventive maintenance were being effectively implemented.

The LPSI and Shutdown Cooling Systems were capable of performing their safety functions, and the licensee was in compliance with applicable Technical Specifications.

The Millstone Unit 2 training and operations staff were aware of the causes and symptoms of loss of shutdown cooling incidents. The operations staff has taken measures to prevent loss of shutdown cooling during partial drain-down of the RCS.

3.2 Outage Activities (92700/61726)

The inspector observed the following activities during the April 9-14 plant shutdown: CEDM cooler boric acid clean-up, containment air recirculation (CAR) cooler boric acid cleanup, repair of 2-MS-11A, repair of 2-SI-237, CEDM testing, and plant heatup.

3.2.1 Boric Acid Buildup

Cleanup of the boric acid on the CEDM coolers, control element drive assembly (CEDA) ventilation shroud, and CAR coolers began on 4/11. The CEDM coil cooling system consists of axial vane fans F-13A, F-13B, and F-13C, of which two normally are in service. Containment air is cooled by Reactor Building Component Cooling Water (RBCCW) at a flow of 165 gpm through each cooler. The CEDA ventilation shroud provides an air annulus above the reactor vessel head to cool the CEDM coil stacks. CAR system coolers remove heat from the containment atmosphere during normal operation. For a Loss of Coolant Incident (LOCI), the CAR coolers are a means of cooling the containment atmosphere to reduce containment pressure.

The boric acid on the above components was very fine and powdery. It was first discovered by the licensee during a containment entry at power on 4/6. On 4/10, the licensee performed an isotopic analysis on the boric acid powder and compared it to an evaporated RCS sample; total activities were 6.16E-3 microcuries/unit and 41.95 microcuries/unit, respectively. The licensee concluded that the boric acid results indicated very low activity.

On 3/30, the licensee had commenced addition of boric acid to the steam generators to inhibit stress corrosion cracking of U-tubes. At that time, as viewed from an in-containment close circuit television (CCTV) camera, steam leakage from 2-MS-11A was apparent. For the estimated 0.25 GPM leak rate from 2-MS-11A at 5 PPM boric acid, the boric acid deposition on the cooled surfaces in containment would be 1.2 lbs. Boric acid removed from the CEDM coolers and shroud including impurities (i.e. dust and dirt) was estimated to be 36 lbs. The source(s) of boric acid buildup were not identifiable by these estimates.

The licensee installed two additional CCTVs inside containment during the shutdown. One camera was located at the 38'6" refueling bridge handrail and directed toward the CEDM coolers and CEDA shroud on the south side of the refueling cavity. The other camera was directed to view RCS loop 2. The licensee implemented a surveillance to check CEDM fan currents, CAR fan currents, and CEDM coil stack temperatures by converting lower gripper coil resistance readings into temperature. Near the end of the inspection (4/27), the licensee cleaned the "A" and "C" CEDM coolers with a hot water lance. Boric acid buildup continued. The inspector will continue to follow licensee location and repair of the source of boric acid.

3.2.2 Failure of CEDMs During Plant Shutdown

At 10:37 p.m. on April 8, with a planned 56-hour downpower maneuver scheduled to begin at midnight, control rod CEA-22 dropped into the core. CEA-22 was declared inoperable at 11:55 p.m. per TS 3.1.3.1.e. A shutdown margin calculation was performed as required by TS 3.1.1.1, using surveillance procedure OP-2208-13. Plant shutdown was begun. At 4:55 a.m. on April 8, during CEA insertion, CEA-4 dropped into the core. At 5:55 a.m., all CEAs were inserted and the plant was in hot standby. In response to the dropped CEAs, the licensee commenced I/C 2421C "Reactor Vessel Head Cable Removal, Installation Testing Data." This procedure provides guidance for removal, installation, and testing of reactor head cables. The inspector reviewed licensee data taken on April 9 for completeness, accuracy, and comparison with allowable values. Upper gripper coils on CEA-22 and CEA-4 had resistance values less than the average of all other CEAs checked. CEA-4 resistance across the upper gripper coil was 1.13 ohms, CEA-22 was 0.98 ohms. The average resistance of all other CEAs was between 7-8 ohms. CEA-5 lower gripper coil resistence was 4.97 ohms, larger than the average of all CEAs by 1.5 ohms. Cable resistance checks for all CEAs met the acceptance criteria.

On April 12, the licensee replaced the CEA coil stacks for CEA-22, CEA-4, and CEA-5. On April 13, the licensee performed drop time testing on the affected rods using surveillance procedure EN 21010, "CEA drop times" to verify CEA operability. The affected CEAs satisfied the acceptance criterion of 2.75 seconds for full length drop time.

The inspector reviewed the safety evaluation prepared for change 4 to revision 1 of EN 21010. The purpose of the change was to allow for individual testing of CEA drop times at a RCS boron concentration less than the value typically present at the beginning of low power physics testing specified in TS

3.1.3.7. The licensee entered the action statement for TS 3.1.3.7 during the testing of CEA-4, -5, and -22, deeming that necessary for required testing of the three CEDMs which had their coil stacks removed. The licensee assumed the most reactive rod (either #5, #4 or #22 at beginning of core life) and cold RCS conditions (refueling) to maintain a 2.9% shutdown margin. The minimum boron concentration calculated by the licensee was 1078 ppm, with 100 ppm added for conservatism giving a final boron concentration of 1178 ppm. The inspector reviewed FSAR section 14.2 safety evaluations for the control element assembly withdrawal incident and the ejected CEA accident initial conditions. The inspector concluded no decrease to the margin of safety existed, and thus, the change did not involve an unreviewed safety question as defined by 10 CFR 50.59.

The inspector had no further questions in this area.

4.0 Plant Operational Status Reviews (71707)

4.1 Review of Plant Incident Reports (PIRs)

The plant incident reports listed below were reviewed to: (i) determine the significance of the events; (ii) review the licensee's evaluation of the events; (iii) verify the licensee's response and corrective actions were proper; and (iv) verify that the licensee reported the events in accordance with applicable requirements. The PIRs reviewed were: 88-17, 88-22, 88-23, 88-29, 88-30, 88-31, 88-32, 88-33, 88-40, and 88-41. The following items warranted inspector followup:

PIR 88-23 (Section 10.0) and PIR 88-32 (Section 4.2).

4.2 MOVATS Failures on Engineering Safety Feature (ESF) Injection Valves (61726/92703/62703)

On March 31, the licensee prepared Plant Incident Report (PIR) 88-32, identifying recently completed Motor-Operated Valve Automated Test System (MOVATS) test failures on six safety injection valves. The valves were 2-SI-653, 2-SI-655 (discharge cross-tie valves), and 2-SI-636, 2-SI-637, 2-SI-646, and 2-SI-647 (discharge injection valve. All of these valves are located on the high pressure safety injection system (HPSI). The licensee concluded the failure mechanism was excessive grease buildup in the spring pack. The spring pack consists of belleville springs used to actuate a cam to set the valve cut-off torque switch. Grease buildup rendered the cut-off torque switch settings inoperable due to hydraulic lockup. The licensee concluded the valves were operable based on satisfactory completion of in-service test T-87-38 "Safety Injection Valve Stroke Test Under Accident Condition Pressure," on February 18. The test required the HPSI injection valves, discharge isolation valves and cross-tie valves to open under full system pressure (1220 PSIG). This in-service test was part of the licensee's response to IE Bulletin 85-03 (Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings). All valves tested met the acceptance criteria. The inspector also concluded that the six affected HPSI valves were operable and that the intended safety function was satisfied from review of recent technical specification operability surveillances and IST-87-38. The licensee, as of May 1, had replaced the spring packs on the affected six valves. Each valve was retested satisfactorily. The inspector had no further questions regarding HPSI valve operability.

The inspector asked about methods of detecting grease buildup. The licensee developed a two-point program to detect and replace spring packs in motor-operated valves. The first objective was to calculate a "K" factor for each valve. The "K" factor is defined as belleville spring deflection divided by test equipment stem thrust. If large fluctuations over rated stem thrust occur, the licensee removes grease from the spring pack. The second objective was to check the pre-load force on the belleville springs. The assigned pre-load sets up a linear characteristic for the springs over a full thrust load range. If the belleville springs have inadequate pre-load, the spring pack is replaced. No inadequacies were noted.

The inspector reviewed Information Notice (IN) 86-29, "Effects of Changing Valve Motor-Operator Switch Settings." IN 86-29 alerts licensees about potential significant safety effects of changing valve motor-operated switch settings in response to IEB 85-03. Changes to switch settings can affect valve position indication and signals such as "permissives" to other equipment. The inspector questioned the licensee on IN 86-29 to determine applicability. The licensee adjusts the bypass torque switch setting as recommended by the vendor (LIMITORQUE). The setting is maintained between 5-7% from the valve unseat and re-seat position as determined from a thrust load recorder. The purpose of the torque bypass switch is to deactivate the torque valve motor cut-out as determined from valve position. The inspector had no further questions on this matter.

The licensee has not developed a long-range program to address hydraulic lock-up in spring packs for motor-operated valves. The licensee has committed to provide this information by July 1988 in response to NRC Bulletin 85-03. The NRC's Office of Analysis and Evaluation of Operational Data (AEOD) requested that the licensee provide information to the Electric Power Research Institute (EPRI) concerning the hydraulic lock-up. The licensee responded to the questionnaire provided by EPRI on April 14. The inspector reviewed the data and had no questions. The inspector will continue to follow licensee actions in response to IEB 85-03 and will review test program implementation for valves subject to hydraulic lock-up.

Motor-operated valve planned maintenance requirements are defined on Maintenance Form 2701J-7. Maintenance includes inspecting the physical condition of the electrical and mechanical (including lubrication) portions of the motor-operator. The limit switch housing cover is removed to facilitate the inspections, but the operator is not removed from the valve or disassembled unless corrective maintenance is required.

Motor-operator inspections are performed annually. If, due to operating constraints, a valve cannot be removed from service during plant operation, the planned maintenance is done during refueling. The schedule for all the motor-operators is included in the planned maintenance system.

If the planned maintenance program, a test, or operation of the valve reveals a problem with the operator, a trouble report is forwarded to the Maintenance Department. A work order is generated to investigate and determine the cause of the problem. If work is required on the motor-operator, a licensee supervisor determines the extent of the work and where it will be performed.

Many motor-operated valves are included in programs that require special consideration with respect to maintenance. Two such programs are MOVATS (motor-operated valve automated test system) and Environmental Qualification. In these cases special precautions have been taken to preclude maintenance from adversely affecting the qualification of the motor-operator.

MOVATS testing was performed to satisfy concerns raised in NRC Bulletin 85-03. The program includes 25 motor-operated valves. For each valve the PMMS (production maintenance management system) has been updated to include the acronym MOVATS under Procedures on the work order, with a caution that the work may change the operating characteristics of the valve. In addition, tags hung on MOVATS valves read:

MOVATS TESTED

Contact Maintenance Prior to
Working Valve or Operator

For motor-operators that require environmental qualification, PMMS identifies the status of the valve operator as EEQ. For EEQ valves the System/Component Equipment Worksheet (SCEW) number is referenced on the AWO (automated work order). The SCEW number can be used to determine environmental qualification requirements from the Electrical Equipment Qualification Maintenance Books. In addition, green EEQ tags have been placed on equipment requiring environmental qualification. No inadequacies were noted.

4.3 Boric Acid Use for Steam Generator Corrosion Inhibitor (79501/92700)

Beginning on March 31, 1988, boric acid was injected into the Millstone 2 secondary system to reduce corrosion, pitting and tube denting in the sludge pile area (central area) of the Steam Generators (SGs). Station Test Procedure 88-30, Boric Acid Treatment of Steam Generators, controls boric acid use during low power soaks and normal power operation.

The boron concentration throughout the secondary system on April 19, after injection had been secured for two days, was:

Sample Location	Boron, ppm
SG-1 Blowdown	8.91
SG-2 Blowdown	8.13
Moisture Separator/Reheater Drain Tank	1.04
Condensate Pump	0.23
Condensate Polisher Facility	0.30
Heater Drain Tank Pump	2.84
Feedwater #1 Heater	1.00

When this data was taken, the condensate polishers were releasing boron back into the system. This was evident because the boron injection was secured that day and the SG boron concentration was increasing with time.

The inspector reviewed the licensee's Balance of Plant (BOP) Safety Evaluation (SE) for Boric Acid Secondary Water Chemistry at Millstone Unit No. 2 and STP 88-30. Millstone 2 is the twelfth U.S. reactor and the nineteenth world-wide Pressurized Water Reactor (PWR) to use boric acid for secondary water chemistry. The SE addresses corrosion/erosion and concludes that there will be no change in the performance of any balance of plant components. The inspector had no further questions on this matter.

5.0 Observations of Physical Security (81064)

Selected aspects of site security were verified to be proper during inspection tours, including site access controls, personnel and vehicle searches, personnel monitoring, placement of physical barriers, compensatory measures, guard force staffing, and response to alarms and degraded conditions. No inadequacies were noted.

6.0 Safety Issues Management System

6.1 Inspection of Licensee's Actions Taken to Implement Unresolved Safety Issue A-26: Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors: Temporary Instruction 2500/19

A technical issue was identified concerning the safety margin to reactor vessel failure for pressurized water reactors (PWRs) subject to severe pressure transients while at a relatively low temperature. The majority of the transients that occurred during startup or shutdown when the reactor coolant system (RCS) was in a water-solid condition. During such conditions, the RCS is susceptible to a rapid increase in system pressure through thermal expansion of the RCS water or through injection of water into systems without adequate relief capacity or a discharge flow path to control the pressure increase.

Plants receiving an operating license before March 14, 1978 committed to design reviews, procedure changes, equipment modifications, operator training, and surveillance using a combination of operator personnel and automatic equipment.

The inspection items to be verified have been divided into several areas: design, administrative controls and procedures, training, and surveillance. Findings are listed below.

6.1.1 Design

The inspector reviewed the December 12, 1977 NRC safety evaluation (SE) supporting Amendment No. 50 to the technical specifications (TS). The TS change added low temperature overpressure protection system (LTOP) requirements. Paragraph 3.3.1.2 of the SE notes that the overpressure protection system is designed to protect the reactor vessel given a single failure in addition to the failure that initiated the overpressure. Redundant pressure protection channels and relief valves are used to satisfy this criterion. Redundant pressure and temperature sensors, bistables, two full capacity power-operated relief valves (PORVs), and independent power sources are provided for long-term overpressure mitigation.

The inspector verified that there are independent power supplies for each PORV solenoid (RC-404, RC-402). RC-402 receives electric power from the 201B-2 (D-22) 125 VDC distribution panel. Power to 201B-2 is supplied from the 201B 125 VDC battery. Valve RC-404 is supplied power from the 201A-2 (D-12) 125 VDC distribution panel. Power to 201A-2 (D-12) is from the 201A 125 VDC battery. The inspector also verified bistable alarms and redundant pressure and temperature sensor power supplies exist using the following licensee drawings:

-- 25203-30024 "Single line diagram, 125 VDC Emergency and 120 VAC vital systems"

-- 25203-32007 Sheet 23 "Pressurizer Relief Solenoid Operated Valve RC-402"

-- 25203-32007 Sheet 24 "Pressurizer Relief Solenoid Operated Valve RC-404"

-- 25203-28114 Sheet 15 "Logic Diagram Pressurizer PORV's and Alarms"

-- 25203-31166 Sheet 4A "Connection Diagram across Line Starters"

-- 25203-31166 Sheet 4 "Connection Diagram across Line Starters"

-- 25203-26014 "P&ID Reactor Coolant System"

The inspector reviewed licensee documentation showing that PORV setpoints are supported by plant specific analysis. Combustion Engineering "Generic Study Evaluating Over-pressurization Transients at Low Operating Temperatures and Pressures (12/3/76)," and Millstone 2 "Report on Low Temperature RCS Overpressure Protection" (11/19/77) evaluations indicate that 10 CFR 50 Appendix G limits are not violated assuming that a HPSI (high pressure safety injection) pump start would produce the worst case pressure transient. The licensee demonstrated that a single HPSI pump plus one charging pump mass input would produce a peak pressure of 465 PSIA (PORV setpoint). The assumed initial conditions for the mass addition transient model were: i) RCS assumed water-solid; ii) letdown flow isolated; iii) no RCS expansion during the transient; iv) a single PORV assumed to fail; v) mass addition using the highest density fluid; and subsequent PORV relief at its lowest density.

The inspector had no further ques[illegible] s matter.

6.1.2 Administrative Controls

The inspector reviewed: licensee controls to minimize plant time in a water-solid condition; controls to minimize temperature difference between the steam generators and reactor vessel; restrictions on the number of HPSI and charging pumps operable when in LTOP conditions; operator alarms to automatically annunciate operation in low temperature conditions; and procedure verifications to have operators manually align and remove low temperature overpressure protection.

The licensee addresses LTOP concerns in two sections of procedure OP-2201, Rev 16, "Plant Heat-Up." Step 4.22 (Precautions) identifies the temperature difference between the SGs and reactor coolant loops in relation to formation of a pressurizer bubble and starting of a reactor coolant pump (RCP). If SG temperature is greater than loop temperature, the operator is directed to form a bubble in the pressurizer to preclude a pressure transient when starting a RCP. TS 3.7.2.1 requires both the primary and secondary coolant temperature be greater than 70 F when the pressure of either coolant is greater than 200 PSIG. This ensures that the SG's do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70 F and 200 PSIG are based on a SG nil-ductility transition temperature of 50 F.

The inspector reviewed administrative controls over the HPSI and charging pumps when in the LTOP condition (RCS temperature below 280 F). In OP-2201, step 4.22 (Precautions), the first train is returned to operation when RCS temperature exceeds 175 F, and the second train is returned to service when RCS temperature is greater than 275 F. TS 3.5.3 requires one HPSI pump to be operable when RCS average temperature is less than 300 F. The licensee is required to ensure, at least once per twelve hours when RCS cold leg temperature is less than or equal to 275 F, that the remaining two HPSI pumps have no electrical power. TS 3.1.2.3 requires, as a minimum, one charging pump and one HPSI pump to be operable in cold shutdown and refueling plant conditions. No inadequacies in the associated licensee procedures were noted.

Control room alarms associated with LTOP are "Hi Reset," "Low Reset," and "Low Temperature Overpressure," all located on control panel C03. The "Low Reset" alarm annunciates when RCS pressure is less than 400 psig and RCS temperature is less than 275 F. This alerts the operator to place the PORVs in their low setpoint mode (450 psig). The "Hi Reset" alarm actuates when RCS temperature exceeds 275 F. This alarm is disabled when the operator places the LTOP control switch to "Hi," resetting the PORV set pressure to 2385 psig. The "Low Temperature Overpressure" alarm is energized when RCS pressure is greater than 400 psig and RCS temperature is less than 275 F. The alarm alerts the operator that the plant conditions are approaching the overpressure setpoint. To prevent inadvertent RCS blowdown at temperatures above 275 F, manual operation of two separate hand switches, one for each PORV, is required. The hand switch located on control panel C03 provides the following functions: i) low setpoint permissive signal, ii) high setpoint permissive, iii) isolation block valve open/closed signal. The inspector verified the alarm circuitry on licensee drawings (25203-32007 sheet 24, 25203-32007 sheet 23, "Pressurizer Relief SOV RC-402, RC-404") and in the "Plant Specific Report on LTOP, Millstone Unit 2." The inspector has no further questions on this matter.

In OP-2201, "Plant Heat-Up," Step 5.8.1, the operator is directed to manually remove LTOP when RCS temperature is greater than 275 F by selecting the separate PORV handswitch to "Hi." OP-2207, "Plant Cooldown," Step 5.17.5, directs the operator to position reset the PORV selector switch (HS-1402) to "Low" when the "Reset to Low" alarm (HS-1402) annunciates on panel C03. The inspector had no further questions on this matter.

6.1.3 Operator Training

The inspection objective in the training area was to determine whether all operators received training concerning RCS low temperature over-pressurization (LTOP) causes, on the operation and maintenance of the system that mitigates the event, and on the consequences of inadvertent actuation.

The licensee currently has 67 licensed Senior Reactor Operators (SROs) and Reactor Operators (ROs), including 8 recently licensed ones.

LTOP training was addressed in the following:

Module	Title
2021	Plant Materials/Brittle Fracture
2321	Reactor Cooling System
R02-1(B)	Plant Heatup Briefing
R02-7(B)	Plant Cooldown Briefing
R02-7(S)	Plant Cooldown Simulator

The inspector confirmed that, for the 8 license candidates in 1987, the above training material was given. Module 2321, Reactor Cooling System, was used in the 1987 Requalification Training for Presently licensed SROs and ROs. It was covered in Cycle 5 Module 5.3 between October 2 and December 3, 1987.

The inspector also reviewed the Lesson Plans listed above. Theory, design criteria, operation, and alarm and actuation setpoints were acceptably addressed. No inadequacies were noted.

6.1.4 Surveillance (62701)

The inspection objective was to determine whether: i) PORV electronics and setpoints are verified periodically; ii) PORV stroke times are compared with the design basis; iii) tests ensure operability of the system electronics before each cold shutdown; and iv) tests are performed before declaring the system operational after system, valve or electronics maintenance.

In accordance with TS 4.4.3.1 and 4.4.9.3.1, the licensee performs a monthly channel functional test using SP 2402I (Low Temperature/Overpressure Circuitry Functional Test), and a refueling outage channel calibration using SP 2402J (LTOP Protection Circuitry Channel Calibration). The inspector reviewed SP 2402I and SP 2402J and the latest data sheets (4/12/88 and 2/14/88, respectively). Twelve of the as-found pressure and temperature indicator readings were outside the acceptance criteria. Instrument Calibration Report (ICR) 88-24 documented the licensee evaluation and conclusion that the indications are acceptable as recalibrated. Pressure Indicator (PI) 103-1B was replaced and design change PDCR MP2-88-036 was prepared to protect against damage to the instrument.

TS 4.4.9.3.1.d requires PORV testing per ASME Section XI. The licensee sends the PORVs to Wyle Laboratories for required testing and furbishing. The inspector reviewed Certification Test Reports 49134-10 and -11, dated January 28, 1988, for each PORV. The test showed the PORVs operated at 2385 + 10 psig in less than 100 milliseconds. The reports document slight leakage through the main ports at test conditions. This leakage was reduced to zero by lapping the seating surfaces. No test data was available for LTOP conditions (450 psig). The inspector questioned the operability of the PORV when set to the low pressure value. A maintenance engineer contacted Dresser Consolidated (the manufacturer) and Wyle Laboratories (the testing facility) and found out that low pressure testing is not requested by licensees. This issue is left unresolved pending further NRC review (UNR 88-07-03).

The inspector was unable to determine the valve operation time assumed in the licensee's LTOP evaluations. In the NRC staff Safety Evaluation supporting Amendment 50, issued in 1979, mention is made of the quick opening time of the valves as about 10 milliseconds. The inspector learned of some EPRI tests showing valve operation in 0.17 seconds with water at 600 PSIA. The inspector noted that the PORVs are used during filling of the pressurizer and operate at very low pressure (about 50 PSIA). This indicates that the PORVs can mitigate an LTOP event. This issue is unresolved pending evaluation of opening time data (UNR 50-336/88-07-04).

6.2 Temporary Instruction (TI) 2515/94: Boron Dilution Incidents

By letter dated September 19, 1977, the NRC informed the licensee of an unanticipated dilution of the reactor coolant system (RCS) boron concentration at an operating PWR. The September 19, 1977 letter requested the licensee to review boron dilution analyses to assure that these are bounding for potential boron dilution events. The letter further requested the licensee to inform the NRC of any procedural or design changes resulting from their review of potential boron dilution events. The review of responses to the boron dilution inquiry was identified as Multiplant Action (MPA) B-3.

The licensee responded by letter dated December 28, 1977. The licensee stated that, while the facility was operating at rated conditions, no dilution incidents other than those addressed in the FSAR (Section 14.3, "Uncontrolled Boron Dilution") would occur; therefore, the FSAR analyses remain valid. For potential boron dilution incidents during Cold Shutdown (Mode 5) and Refueling (Mode 6), the existence of "abnormal valve lineups" could produce unexpected sources of boron dilution. The licensee identified the following paths where primary makeup water (PMW) could be introduced into the RCS during Mode 5:

-- RWST recirculation header (valve 2-CS-10A) to the safety injection header.

-- Valve 2-CH-080 to the safety injection header.

Both paths could potentially result in a dilution rate greater than the 132 gpm assumed in FSAR Section 14.3. Acceptability of the safety analysis dilution rate of 132 gpm is based upon the need for manipulation of more than one valve to produce the above "abnormal" valve lineups.

The licensee's December 28, 1977 letter also identifies "abnormal valve lineups" which could occur during Mode 6 and result in an unplanned boron dilution:

-- PMW to spent fuel pool, fuel transfer canal, and refueling pool via 2-RW-77.

-- PMW to spent fuel pool via 2-PMW-167.

The licensee concluded that the shutdown margin is not lost as a result of potential boron dilution via 2-RW-77 or 2-PMW-167.

The licensee's letter commits to requiring valves 2-CS-10A and 2-CH-80 to be closed and tagged in Mode 5 (per OP-2207, "Plant Cooldown") and to closing and tagging valves 2-CS-10A, 2-CH-080, 2-RW-77 and 2-PMW-167 in Mode 6 (per OP-2209A, "Refueling Operations"). By letter dated March 1, 1979, the NRC staff accepted the licensee's analysis and corrective actions.

On April 26, 1988, procedures OP-2207, Rev. 14, Change 4 and OP-2209A, Rev. 11, Change 2 were reviewed by the inspector. Step 5.31 of OP-2207 requires valves 2-CS-10A and 2-CH-080 to be closed and tagged. Valve 2-CH-195, PMW to Charging Pump Suction, identified as a potential dilution path, is also required to be closed and tagged by Step 5.31. Step 3.4 of OP-2209A requires valves 2-CH-10A, 2-CH-195, 2-CH-080 and 2-RW-77 to be closed and tagged. The following additional valves, identified as potential dilution paths, are also required to be closed and tagged by Step 3.4.

-- 2-CH-422, PMW to Volume Control Tanks.

-- 2-CH-091, PMW to Charging Pump Suction.

Based upon the review of OP-2207 and OP-2209A, the inspector concluded that the licensee has met their commitment, contained in their letter of December 28, 1977, to close and tag certain valves to prevent unanticipated boron dilution during Modes 5 and 6. Moreover, as additional boron dilution paths were created (via plant modifications) or identified as a result of subsequent review, the associated valves were also required to be closed and tagged in Modes 5 and 6, per OP-2207 and OP-2209A. No additional boron dilution paths were identified by NRC review. No inadequacies were noted.

7.0 Surveillance (61726)

Surveillance Procedure (SP) 2401E is used to insure the NI safety/control instruments are representative of the core neutron flux as indicated by incore detector measurements. The inspector observed calibrations completed per SP 2401E on April 19, 1988. This satisfied the TS 4.3.1.1.1 Table 4.3-1 requirement for monthly calibration of the excore nuclear instruments (NIs) following equilibrium at 100% power. The licensee normally performs this calibration every two weeks. The inspector reviewed SP 2401E and the completed SP 2401E forms for bistable trip data. Calorimetric data is entered into a computer with the as-found settings and, later, the as-left settings. A computer program calculates each detector voltage with allowable limits and the as-left Axial Shaping Index (ASI). After the calibration is completed, a computer report is prepared for each channel giving the input data, allowable limits and the final data. The print-out was attached to the surveillance cover sheets as permanent records. The Channel C as-found 15% Trip Permissive was outside the licensee's 14.5 +/- 0.2% acceptance criteria at 14.74%. Since it met the TS limit of less than 15%, an Instrument Calibration Report (ICR) was issued. A Plant Incident Report (PIR) would be required if TS limits were exceeded. The inspector had no further questions.

8.0 Allegations (92720/92702)

8.1 RI-88-A-29, Procedural Compliance in Meterology Lab

On March 18, the inspector received an allegation from a licensee employee. The alleger had two specific concerns dealing with meterology laboratory work. The meterology lab calibrates quality assurance (QA) standard instrumentation used for in-plant instrument calibrations. The alleger's concerns were:

-- Inadequate procedural compliance for safety-related instrument calibration.

-- Job discrimination from licensee management. [The alleger has taken this allegation to the Department of Labor (DOL).]

The inspector interviewed the alleger on March 22. For QA standards (QA-107, QA-110) calibrated per I/C 1101C, "Simpson multimeter calibration," the alleger stated that data error was in the opposite direction between one procedure revision and the next, and felt this was improper. To follow-up, the inspector reviewed data on the following multimeter calibrations.

Standard	Procedure	Instrument	Date Calibrated
QA-110	I/C 1101C	QA-2867	3/16/88, 1/27/88
		QA-2814	8/25/87, 7/8/86
		QA-2770	10/29/87, 10/23/86
		QA-2863	2/24/88, 1/19/87
QA-107	I/C 1101C	QA-2824	2/24/88, 10/13/86

The inspector reviewed the latest two revisions to procedure I/C 1101C and, on April 7, witnessed a calibration with QA standard 110, instrument QA-2824. No discrepancy was found in procedural compliance or adequacy. The revised procedure changed whether the calibrated instrument reading is subtracted from the reference reading (or vice versa). That changes the algebraic sign of the error but not calibration accuracy or validity. This part of the allegation was unsubstantiated.

Another technical concern from the alleger concerned procedural compliance during deadweight calibrations of Heise pressure gauges. The concern was that identical data readings recorded for increasing as well as decreasing pressure points indicated improper data logging and procedure performance. The inspector reviewed the data sheets from the following Heise gauges:

Standard	Procedure	Instrument	Calibration Dates
QA-177	I/C 1104A	QA-379	12/23/87
QA-176	I/C 1104A	QA-307	1/4/88, 10/9/87, 7/8/87, 3/30/87, 12/29/87, 9/23/87
QA-280	I/C 1104A	QA-280	1/3/88, 5/6/87, 2/9/87
QA-176	I/C 1104A	QA-282	1/3/88, 10/28/88, 7/10/87, 5/6/87, 2/19/87, 11/19/86, 8/20/86
QA-172	I/C 1104A	QA-284	3/10/88, 12/11/87, 8/31/87, 5/28/87, 2/27/87, 12/5/86

The data for the Heise gauge calibrations between 8/20/86 and 3/23/88 was reviewed. There were instances of data points on pressure increase being the same as in pressure decrease. The inspector independently took data in parallel with the licensee on a 0-15 psig Heise gauge (QA-303) on 4/7. Calibration results met the acceptance criteria, and no significant difference existed between the inspector's and the licensee's data. The inspector noted it was possible for a pressure gauge to indicate the same results in pressure rise as in pressure drop. This part of the allegation is unsubstantiated.

In regard to the alleged job discrimination, the inspector reviewed licensee administrative control procedures (ACP) 1.14 "Employee Complaints and Grievances," and ACP-1.14A "Nuclear Complaints and Concerns." The inspector also interviewed licensee management and first line supervisors concerning the alleged discrimination. No conclusion was reached. This aspect remains open pending receipt of the DOL disposition.

8.2 RI-88-A-40, Procedure Adequacy and Controls on Control Room Radiation Monitors (92720/83726)

On April 11, 1988, the inspector received an allegation from a licensee employee relating to the Unit 2 Control Room Ventilation Area Radiation Monitors (RIT 9799A and 9799B). The alleger raised two specific concerns:

a. The above monitors were not being adequately calibrated by the Unit 2 Instrument and Control (I&C) group.

b. Employee concerns regarding the adequacy of monitor calibration were not responded to appropriately by I&C supervision.

During the week the allegation was received, a routine Health Physics inspection was being performed onsite by NRC Region 1 based Radiation Specialists. The alleger's concerns relating to the technical adequacy of monitor calibration were reviewed by the NRC radiation specialist.

8.2.1 System Description

The Unit 2 Control Room Ventilation Area Radiation Monitors (RIT 9799A, 9799B) view the Heating Ventilation and Air Conditioning (HVAC) inlet ducting upstream of the Unit 2 Control Room. Upon detection of area radiation levels equal to or in excess of the alarm setpoint (1 mR/hr), the monitors switch the Control Room Ventilation to the Recirculation mode. The two monitors feature RD-1 GM detectors supplied by GA Technologies. The monitor range is from 0.1 to 10,000 mR/hr.

Calibration frequency for these monitors is once every 18 months. Initial system calibration was performed in June 1985. Both monitors were recalibrated in December 1986; the 9799A monitor was calibrated again in February 1988 after the readout module was replaced.

The inspector evaluated the technical adequacy of the licensee's calibration of these monitors by discussion with cognizant personnel and review of the following:

-- Procedure SP2404BA, "Control Room Ventilation Radiation Monitor Calibration."

-- Procedure SO2404AZ, "Unit 2 Control Room Ventilation Area Radiation Monitor RIT-9799A & 9799B Functional Test."

-- Results of monitor calibrations performed in 6/85, 12/86 and 2/88.

-- GA Technologies Operation and Maintenance Manual E-115-185.

-- ANSI N323-1978, "Radiation Protection Instrumentation Test and Calibration."

-- Unit 2 FSAR.

8.2.2 Procedural Adequacy

NRC revie· of calibration procedure SP2404BA identified the following steps performed as the radiological calibration of the Control Room Ventilation monitors:

-- Test sources are obtained with gamma flux values equivalent to the lowest decade (0.1 to 1.0 mR/hr) and as close as possible to the highest decade (1.0 to 10.0 R/hr) of the monitor.

-- The test source strengths are measured using HP Calibration Lab equipment. These measurements then become the "desired" values for calibration.

-- The test sources are held adjacent to the monitors and the monitor readout is compared with the "desired" value.

The inspector noted that no direction was given in the procedure to ensure that i) dose rates generated by the test sources were measured at a specified distance from the sources or that ii) test sources were held at the same distance from the detectc· during the calibration. No notations were made on completed data sheets indicating measurements of source to detector distance were made or that any specific geometry was maintained. Consequently, the dose rate used as the "desired" value during the calibration is not accurately known.

Acceptance criteria for monitor readout listed on the data sheet were also noted to be inappropriate. Acceptable tolerances for both calibration points equaled the desire value +/- 10 mR/hr. This was noted to be too large an error range for the lowest decade (0.1 to 1.0 mR/hr) and too small a range for the highest decade (1.0 to 10.0 R/hr).

ANSI N323 states in Section 3 that radiation fields used for calibration must be "thoroughly understood in terms of quality, quantity, and reproducibility." Section 6.1 of the same standard states that the exposure rate or the flux density of the radiation field "shall be known with an estimated uncertainty no greater than +/- 10 percent." Section 7.5.6.1.3 of the Unit 2 FSAR states that area radiation monitor calibration will be accomplished by placing the detector in a reproducible, fixed geometry configuration and exposing the detector to a calibrated radioactive source placed at a measured distance from the detector.

The inspector stated that the lack of specificity in procedure SP2404BA relative to source to detector distance and the subsequent large uncertainty in "desired" dose rates renders the procedure inadequate to assure calibration at a level commensurate with industry standards in accordance with the FSAR (NED 88-07-05).

In response, the licensee stated that an upgrade of procedure SP2404BA had been initiated. The inspector reviewed purchase orders, dated October 1987, which demonstrated that the licensee had ordered a calibration source assembly from the monitor vendor. This assembly mounts on the detector during calibration and provides a known dose rate in a reproducible geometry. The licensee stated that, although steps had been taken to improve the calibration procedure, the existing SP2404BA procedure was thought to be adequate and was consequently used in February 1988 on the 9799A detector.

8.2.3 Procedural Compliance

As noted above, calibration procedure SP2404BA requires that response of the Control Room Ventilation monitors be checked on the bottom decade (0.1 to 1.0 mR/hr) and at as high as practicable to the top decade (1.0 to 10.0 R/hr). During review of calibration data for the Control Room Ventilation monitors, the inspector noted the following discrepancies with the December 1986 calibration:

Source strength measurements were listed as 4, 8, and 7000 mR/hr in steps 7.4.1, 7.4.3, and 7.4.6, respectively. The "desired" monitor values reported in steps 7.4.2.1 and 7.4.4.1, however, were listed as 1.0 and 0.5 mR/hr, respectively. This shows that both monitors were checked at two points on the same decade (0.5 and 1.0 mR/hr) rather than of one point on the bottom decade and one point on the top decade (or as high as practicable) as required by SP 2404BA step 7.4.3. The 7000 mR/hr source strength reading listed in step 7.4.6 of the data sheets indicates a higher dose rate was available and thus "practicable" for the calibration.

Failure of the licensee to check monitor response on a low and high decade as required by procedure SP 2404BA violates TS 6.8 (NOV 88-07-05). The completed data sheet also indicates a basic lack of understanding on the part of the I&C technician, as evidenced by the difference between the source strength measurements and the values used as "desired" readings. Despite these deficiencies, no problems were noted during licensee supervisory review of the completed data sheets.

The inspector interviewed two I&C technicians responsible for completing the December 1986 calibration of the Control Room Ventilation monitors. Both technicians recognized the discrepancies on the data sheet when it was brought to their attention and neither could explain why measurements for only one decade were taken during the calibration. Both indicated they thought both the low and high decade had been checked.

8.2.4 Monitor Operability

TS Table 3.3-6 requires the Control Room Ventilation Monitors (listed in Table as Control Room Isolation Area Monitors) to be operable in all modes. Based on the above problems noted with calibration adequacy, the inspector questioned the operability status of the monitors. On April 13, Unit 2 Plant Management was asked to demonstrate operability of the monitors or otherwise comply with the TS 3.3.6 action statement. In response, the licensee performed an operability test on monitor 9799B. A test procedure (T88-32) was approved which required the licensee to expose the monitor to a known dose field of 0.6, 1.0, and 100 mR/hr. The inspector reviewed the test procedure and noted it required the use of a highly accurate HP survey instrument to initially measure source dose rates and also required placing the test source at measured distances from the monitor.

The inspector observed the completion of the test and noted the monitor responded adequately at all three test points. The initiation of the recirculation mode of the Control Room Ventilation also occurred at the 1.0 mR/hr setpoint as required. In light of the results of this test, coupled with the monthly functional surveillances of the monitor, the inspector concluded the monitor was operable despite the problems noted above.

8.2.5 Summary

Based on the above review, the inspector concluded the alleger's concerns regarding the adequacy of monitor calibration were substantiated. Calibration procedure SP 2404BA was found inadequate to ensure monitor calibration was performed in a repeatable fashion consistent with industry standards. Also, deficient compliance with the calibration procedure caused a violation.

Repeated deficiencies in radiation monitor calibrations have been noted in previous NRC inspections (see NRC Report No. 50-423/86-27 and NRC combined report No. 50-245/87-24, 50-336/87-21, and 50-423/87-19). Specific corrective actions by the licensee, prompted by NRC concerns, were noted to be ineffective. An NRC Follow-up item (245/87-24-01, 336/87-21-01, and 423/87-19-03) is currently open to track licensee actions to improve overall performance of radiation monitor calibrations. Subsequent licensee actions in this area will be reviewed as part of the above open items. Licensee responsiveness to concerns raised by employees will also be reviewed further on subsequent inspections.

9.0 Committee Activities (40700)

The inspector attended meeting 2-88-96 of the Plant Operations Review Committee (PORC) meetings on April 12. The inspector noted by observation and from the written record that committee administrative requirements were met for the meetings, and that the committees discharged their functions in accordance with regulatory requirements. The inspector observed a thorough discussion of matters before the PORC and a good regard for safety in the issues under consideration by the committee. No inadequacies were identified.

10.0 Review of Licensee Event Reports (LERs) (92700)

Licensee Event Reports submitted during the period were reviewed to assess LER accuracy, the adequacy of corrective actions, compliance with 10 CFR 50.73 reporting requirements, and to determine if there were any generic implications or if further information was required. The LERs reviewed were:

-- LER 88-005, Loss of Normal Power (LNP) Resulting from Operator Error and Subsequent discovery of Failure of EDG Sequencer.

-- LER 88-006, Combined Bypass Leakage Rate Exceeded.

-- LER 88-007, Changed Modes with D/G Inoperable.

NRC review of the licensee's followup and corrective actions for LER-88-005 was documented in Inspection Report 50-336/88-02.

LER 88-007 (PIR 88-23) concerned the entry into Mode 3 on February 16, 1988, while operating in accordance with Technical Specification Action Statement (TSAS) 3.8.1.1.(b), thereby violating TS 3.04. At the time, the "B" EDG was inoperable for routine preventive maintenance. An RCS heat-up from 190 to 532 degrees F was in progress. The licensee scheduled plant preventive maintenance during the time when RCS temperature exceeded the 300 degrees F point (Mode 3). The plant was at 295 degrees F when TSAS 3.8.1.1.(b) was entered.

The licensee reported this item, under 10 CFR 50.73(a)(2)(i), as an operation prohibited by the TS. The licensee's evaluation concluded that one operable EDG was capable of supplying all power required to return the reactor to a cold shutdown condition, and all safety-related plant equipment was operable when the plant entered Mode 3. The inspector reviewed the technical specifications, planned preventive maintenance on the "B" EDG, and corrective actions by the licensee, and concluded that the event described by LER 88-007 had minor safety significance.

Failure to meet TS 3.04 was considered for enforcement action. No notice of violation will be issued since this item meets criteria in 10 CFR 2 Appendix C. It was of minor safety significance, was identified by the licensee, and was reported as required. Corrective actions were timely and appropriate. This item would not have been prevented by corrective action on a past violation. The inspector had no further questions (NC4 88-07-02).

11.0 Review of Periodic (90713) and Special Reports (92700)

Upon receipt, periodic and special reports submitted pursuant to Technical Specifications were reviewed. This review verified that the reported information was valid and included the NRC required data; that test results and supporting information were consistent with design predictions and performance specifications. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Operating Report for Millstone Unit 2 for the month of March 1988.

-- Start-Up Test Report for Cycle 9

No deficiencies were noted.

12.0 Management Meetings (30703)

Periodic meetings were held with station management to discuss inspection findings during the inspection period. A summary of findings was also discussed at the conclusion of the inspection. No proprietary information was covered within the scope of the inspection. No written material was given to the licensee during the inspection period.