U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report/ Docket No.:	50-336/91-15
License No.:	DPR-65
Licensee:	Northeast Nuclear Energy Company P. O. Box 270 Hartford, CT 06141-0270
Facility Name:	Millstone Nuclear Power Station, Unit 2
Inspection At:	Waterford, CT
Dates:	May 14 - June 22, 1991
Inspectors:	P. J. Habighorst, Resident Inspector, Unit 2 W. J. Raymond, Senior Resident Inspector, Millstone R. A. McBrearty, Reactor Engineer, MPS, DRS

Approved by: Gene Kelly July 8, 1991
Eugene M. Kelly, Chief Date
Reactor Projects Section 4A

Areas Inspected: Routine NRC inspection of plant operations, radiological controls, maintenance, surveillance, outage activities, licensee self-assessment, and periodic reports.

Results: See Executive Summary

EXECUTIVE SUMMARY

MILLSTONE UNIT 2 INSPECTION 91-15

Plant Operations

Generally good performance was noted by plant personnel to complete a shutdown following a steam generator tube leak. An exception to good performance was a deficiency in crew coordination and communication during the shutdown of the main turbine, resulting in a subcritical automatic reactor trip. NNECO actions were appropriately focused to correct the performance deficiency.

Controls during mid-loop operations were satisfactorily implemented. Licensee maintenance of both emergency power sources availability and implementation of containment integrity controls exemplified good awareness of and management of shutdown risk.

Radiological Protection

Radiation protection controls for outage activities inside the containment were well implemented, and a significant personnel exposure reduction was noted in the removal of steam generator manways.

Maintenance/Surveillance

NNECO corrective actions to address the cracked charging pump block were satisfactory. The preventive program to trend the modified block performance was a good initiative in response to a design problem. Other maintenance and surveillance activities observed were acceptably implemented. The NNECO program to use limiting conditions for operations to perform preventative maintenance is adequate to assure plant safety while improving equipment reliability.

Engineering and Technical Support

A questioning attitude regarding quality indicators for the emergency diesel air vent valves exemplified a good safety ethic; however, engineering interface with operations was untimely regarding the operability status of the valves.

An unresolved item (91-15-01) was opened to determine whether the Millstone 2 off-site electrical distribution system complies with 10 CFR 50 Appendix A, Criterion 17.

Scope and selection basis for the steam generator tube inspections were both extensive and methodical.

Safety Assessment/Quality Verification

LER 91-003-00 documented an inadequate design modification as a contributing root cause for a postulated loss of DC switchgear cooling, yet no corrective action existed at the time of this inspection. Furthermore, NRC Generic Letter 88-14 requested verification of system design based on instrument air failure modes. NNECO actions to meet their commitments and to correct identified deficiencies in this instance were an exception to generally good performance. This item is unresolved pending future inspection of NNECO action address root causes of deficiencies (UNR 91-15-02).

TABLE OF CONTENTS

The inspection procedure (IP) or temporary instructions (TI) from NRC Manual Chapter 2515 that was used as guidance is parenthetically listed for each report section.

DETAILS

1.0 SCOPE

Within this report period, interviews and discussions were conducted with members of Northeast Nuclear Energy Company (NNECO) management and staff as necessary to support inspection activity.

Resident activities during this report period included 167 hours of inspection. In addition to normal working hours, review of plant operations was conducted during periods of backshifts (evening shifts) and deep backshifts (weekends, holidays, and midnight shifts). Inspection coverage included 31 hours of backshift and 16 hours of deep backshift.

2.0 SUMMARY OF FACILITY ACTIVITIES

Millstone 2 began the inspection period at full rated power. On May 25 at 1:25 a.m., the licensee identified an increase in primary-to-secondary leakage on the main steam line (Nitrogen-16 monitors). The leakage in the No. 2 steam generator increased to approximately 50 gallons per day (gpd) at 6:00 a.m. At 1:55 p.m., NNECO management ordered a plant shutdown. The shutdown was originally scheduled for 18 days to identify and repair the steam generator tubes. On May 30, NUSCO management decided to extend the outage to perform a comprehensive and expanded steam generator tube inspection.

At the end of the inspection period, the plant was in cold shutdown, and steam generator eddy current inspections were in progress.

3.0 PLANT OPERATIONS (IP 71707)

3.1 Operational Safety Verification

This inspection consisted of selective examinations of control room activities, operability reviews of engineered safety feature systems, plant tours, review of the problem-identification systems, and attendance at periodic planning meetings.

Control room reviews consisted of verification of staffing, operator procedural adherence, operator cognizance of control room alarms, conformance with technical specification limiting conditions of operation, and electrical distribution status verifications.

Administrative control procedure (ACP) - 6.01, "Control Room," identifies the minimum staffing requirements and the required protocol within the control room. During this inspection period, the requirements were observed by the inspectors to met for the number of licensed operators in the control room during power and shutdown conditions. Operator attentiveness and cognizance of plant conditions was acceptable.

Inspection was performed of operator procedural adherence during the plant shutdown and reactor trip (report detail 3.2), and routinely during periodic equipment outages, in-service tests, daily technical specification surveillances, steam generator leak testing, and operability equipment surveillance. Operator actions were coordinated and well controlled. Control room operators were cognizant of control room alarms both during power operation, and shutdown operations. Actions taken in response to alarms were timely and appropriate.

Primary to Secondary Leakage

The inspector reviewed the status of primary-to-secondary leakage following the shutdown to repair the tube leak in the #1 steam generator. The review was completed on May 13 with the plant operating at 100% full power. NNECO monitored steam generator leakages using the nitrogen-16 (N-16), blowdown and offgas radiation monitors, and by direct chemistry sampling and analysis of secondary water.

The leakage rates in both generators were below the technical specification limit of 144 gallons per day. Residual activity was present in the SG No. 1 secondary side as part of hide-out return from the activity and tube leak present before the shutdown. All leak indicators showed low, decreasing trends for primary-to-secondary leak rates.

NRC review noted an abnormal value on the daily Chemistry Status Sheet on May 13. The 4:30 a.m. entry for N-16 showed a 115 gallon per day (GPD) leak rate on the #1 steam generator. The inspector noted that the N-16 readouts at 5:00 p.m. were 0.16 and 0.09 gpd for SG No. 1 and SG No. 2, respectively. The abnormal value was discussed with control room personnel, who determined the entry was in error. The inspector noted that this abnormal entry had not been highlighted by operators during routine review of chemistry data, which was contrary to otherwise good performance in maintaining operating status records. NNECO performance in operating status reviews will be reviewed in the future as part of routine resident inspections.

Radiation Monitoring LCOs

The inspector reviewed adherence to technical specification (TS) requirements. The TS limiting conditions for operations were entered during the outage for various radiation monitors and fire protection seals. The radiation monitors that were out of service at various times during the outage were the steam generator blowdown monitor, stack radiation monitor, aerated liquid radiation monitor, and containment radiation monitors. NNECO took appropriate actions for all instances reviewed, including chemistry grab samples and control of the ventilation systems. The inspector verified through fire watch log entries, that the required hourly watches were being performed on inoperable fire barriers.

System Alignments

Inspection of the onsite electrical distribution system determined that operability of the emergency core cooling pumps and valves, emergency diesel generators, radiation monitors, and various engineered safety feature equipment was acceptable. The distribution system was also reviewed during outages on the non-emergency 4.16KV 24A and 24B buses to verify independence, when required, and proper alignments. During emergency diesel outages, the facility emergency core cooling pumps on the operable emergency source was preserved.

Operability reviews of the engineered safety feature systems included periodic verification of valve lineups, power supplies, and flow paths for the high pressure safety injection system, the low pressure safety injection system (during shutdown cooling operations), the charging system, the containment spray system, the containment purge isolation system, control room air conditioning system, emergency diesel generators and auxiliaries, and the auxiliary feedwater system. Satisfactory conditions were observed.

Jumpers

Various bypass jumpers were reviewed for conformance with ACP-QA-2.06B with emphasis on installation, and, if applicable, the content of the safety evaluations. Specific bypass jumpers reviewed were 2-91-36, "Tie in temporary outside compressor station air supply during station air compressor maintenance," 2 91-37, "Installation of a temporary gauge for the "A" boric acid pump discharge," and 2-91-24, "Removal of automatic closure of shutdown cooling suction valves." Additionally, the inspector periodically reviewed all open jumpers for age, and periodic Plant Operations Review Committee (PORC) evaluations to disposition longstanding evaluations. The jumpers reviewed were found to be in accordance with administrative requirements.

Tagouts

Inspection was performed of equipment tagouts according to applicable sections of ACP-QA-2.06A. Tagouts reviewed were: 2-1042-91, "Prevent Inadvertent Loss of Shutdown Cooling on loss of the normal station service transformer"; 2-1361-91, "Caution tagging of potential injection sources without nozzle dams"; 2-1097-91, "Isolation of the No. 2 atmospheric dump valve"; 2-1146-91, "A service water pump isolation"; 2-1044-91, "Low Temperature Overpressurization"; 2-1061-91, "Removal and installation of primary manways"; 2-1098-91, "Overhaul on non-return main steam line check valve (2-MS-1A)"; 2-1240-01 and 2-1239-91, "Electrical insulation tests on non-emergency bus 24A." Aside from specific tagouts reviewed, partial reviews were completed of the tags installed in the plant by comparison with the tagout sheet maintained in the control room. A review of the tagouts verified that the proper equipment was tagged, equipment identified in technical specifications was appropriately controlled, and equipment isolation was proper based on work observations, and consideration of controlled diagrams and procedural guidance.

Log Keeping and Turnovers

Control room logs, night order logs, radwaste logs, plant incident report log, and crew turnover sheets were reviewed. Satisfactory conditions were noted. On June 5, NNECO altered the format of the crew turnover sheet. The alterations included the addition of sections on evolutions in progress, long term evolutions, changes in equipment in the last 24 hours, security key accountability, and beeper information. On May 31, the operations supervisor documented expectations on the additional turnover sheet format. The inspector considered the turnover sheet additions to be a generally positive step towards providing clarifications and accountability of plant activities.

NRC inspection verified that shift turnovers were satisfactory, with the shift supervisor controlling the turnover and additional information presented by the work control supervisor. Accurate information on plant conditions and evolutions in progress were discussed with all members of the crew.

Control room trouble reports were reviewed for age, NNECO planned action, and operator awareness of the reason for the trouble report. The trouble reports were generally recent with very few longstanding items.

Attendance at daily planning meetings identified upcoming quality service department audits, maintenance and surveillance activities in progress, exposure estimates, and actual exposure incurred for the previous day, and discussions of work control and authorizations. Upcoming quality services and audits planned for late 1991 include: ASME Section XI program, Fitness for Duty program, EEQ Maintenance Program Review, Authorized Work Order and Work Control, and Long term effects of Non-Conformance Reports.

Periodic plant tours were conducted of the auxiliary building, turbine building, containment building, enclosure building, and intake structures. Plant housekeeping was satisfactory, no fire hazards were observed, and good material condition of safety equipment was noted.

3.2 Onsite Followup of Operational Events

3.2.1 Subcritical Reactor Trip During Plant Shutdown

Event Description

During plant operation at 100% full power on May 25, a primary to secondary leak developed on the No. 2 steam generator. The leakage was detected by radiation monitors on the main steam line (N-16) and on the steam jet air ejector discharge from the main condenser. The leak rate increased from zero at midnight to about 50 gallons per day (gpd) at 6:00 a.m. Plant operators entered abnormal operating procedures to monitor leakage status and notified the NNECO duty officer and management. The leakage stabilized between 45 to 50 gpd for the next 8 hours, with periodic spikes to 61 and 73 gpd. After plant engineering review of leak rate trends and with leak rate at 55 gpd and trending up, plant management at 1:55 p.m. ordered the plant shutdown to conduct repairs. The resident inspector went to the site to monitor the shutdown. The plant status and licensee actions were reviewed in a telephone conference call with NRR and Region I management at 4:00 p.m.

While inserting control rods to shutdown the reactor, an automatic scram on low steam generator pressure (nominal setpoint 600 psia) occurred at 5:32 p.m. The reactor was subcritical at the time of the scram with insertion of regulating group four in progress. Plant operators noted the decreasing Tavg and steam pressure, but actions to remove excess steam loads were unsuccessful to avert the scram. The plant responded as expected following the scram.

The steam generator leakage remained at about 55 gpd following the scram and during the shutdown. The reactor was placed on shutdown cooling using the "A" LPSI pump at 1:25 p.m. on May 26, and the plant entered Mode 5 (Tavg less than 200F) at 4:40 p.m. on May 26.

NNECO reported the automatic reactor trip at 6:12 p.m. to the NRC Operations Center, pursuant to 10 CFR 50.72 (b)(2)(ii) as any event that results in automatic actuation of the reactor protection system. NNECO initially reported that the low steam generator pressure was caused by the plant cooldown attributed to low reactor decay heat following 13 days of operation after the last outage. Subsequent NNECO review of the post-trip data on May 26 concluded that the plant cooldown occurred because the main turbine was not removed from service before the reactor was shut down. NNECO corrected its report to the NRC duty officer.

Sequence of Events

The following lists the chronology of critical plant parameters or actions during the event:

Time	Event
17:01:00	Operators take the main generator off-line by opening off-site breaker 15G-9T-2
17:08:32	Reactor Coolant System Average - Reference Temperature Low Alarm
17:16:26	Reactor Protection System Pre-Trip Alarm
17:31:24	Steam Generator Low Pressure, Reactor Protection System Channel "A"
17:32:01	Steam Generator Low Pressure, Reactor Protection System Channel "B"
17:32:01	Reactor Trip
17:32	Operators Commenced Emergency Operating Procedure (EOP) 2525, "Standard Post-Trip Actions"
17:40	Operators Completed Actions Within EOP 2525 and initiated actions within EOP 2526, "Reactor Trip Recovery"
18:15	Completed Actions within EOP 2526

The inspector reviewed the licensee's pre-trip, post-trip, sequence of event log, post-trip review summary, and the duty officer trip report. NRC reviews focused on plant response and operator actions.

The inspection confirmed an appropriate reactor protection system response to a low steam generator pressure condition. The review included a consideration of setpoint values as detailed in Technical Specification Table 2.2-1, the "as-left" functional test data per procedure SP-2402P, "Spec 200 Safety Parameters Functional Test," and review of the post-trip steam header pressure. The inspector concluded the reactor tripped within design specifications and requirements.

NRC review evaluated plant cooldown values and compared them with the requirements of Technical Specification 3.4.9.1. The cooldown rate is limited to 80 degrees Fahrenheit (F) per hour. The calculated average cooldown rate for a one hour period prior to the reactor trip was 46.8 degrees F/hr.

The inspector reviewed reactor power and shutdown margin prior to and after the trip to determine the effects of the inadvertent plant cooldown from 532 degrees F to 507 F. Reactor conditions were evaluated to determine whether the criticality requirements of Technical Specification 3.1.1.5 were met with the reactor below 515 F. At the time the reactor coolant system (RCS) was at 515 degrees F and, prior to the trip, the reactor was subcritical with a 2% delta k/k shutdown margin (keff 0.98). This shutdown margin was attributed to the inserted control element assembly (CEA) reactivity worth (rod group 5 inserted to 45 steps) and the RCS boron concentration. Additionally, just prior to the plant trip, the reactor shutdown margin was 2.6% delta k/k based on a CEA pattern with group 4 fully inserted. The shutdown margin was offset somewhat by the contributions from the moderator temperature coefficient of reactivity during the cooldown to 507 degrees F. The positive reactivity added by the cooldown as calculated to be 0.225% delta k/k. At the time of the trip, the shutdown margin was further increased by the insertion of the worth from the remaining control and shutdown rod groups. The reactor was subcritical at 515 degrees F just prior to the automatic reactor trip. The technical specification requirements were met.

Operating Crew Performance

The assessment of operating the crew performance was based on interviews, a review of procedural actions, and the overall control of plant activities. Crew performance was assessed on inter-crew communication, command-and-control, and procedural adherence.

Overall inter-crew communications were acceptable; however, deficiencies were noted. Generally good communications were noted in the awareness of plant conditions prior to the trip by the senior reactor operators, the reactor operator to senior operator communications, and the communications between both senior reactor operators on shift. However, the communication between the senior reactor operators prior to the trip was deficient in that the shift supervisor was generally unaware of the plant cooldown and the supervisory control operator actions to reduce excess steam loads. Reactor operator to senior operator communications were generally satisfactory, with a noted exception involving the status of the main turbine. Crew awareness of plant conditions and the initial actions to address the ongoing cooldown were acceptable; however, general understanding of the primary cause of the cooldown was lacking.

Command-and-control was acceptable with a noted deficiency on the supervisory control operator's decision to leave the control room to perform switchyard breaker alignments in the Unit 1 control room at a time when reactor protection system pre-trip alarms existed on low steam generator pressure. Additionally, the shift supervisor was distracted from his primary operational oversight functions by unrelated communication from utility management and outside organizations.

Procedure adherence by control room operators was assessed. Procedures in-use prior to and after the reactor trip were: OP-2204, "Load Changes"; OP-2205, "Plant Shutdown"; OP-2206, "Reactor Shutdown"; OP-2323A, "Turbine";, OP-2324A, "Main Generator"; EOP-2525, "Standard Post-Trip Actions"; EOP-2526, "Post-Trip Recovery"; and AOP-2569, "Steam Generator Tube Leak." The assessment of procedural adherence was based on control room log entries, sequence of events logs, and operator interviews. Procedural adherence was generally good.

One procedure adherence deficiency was noted in completing the action step 3.17 EOP-2525 which states that if steam generator pressure is not controlled between 880-920 psia, then as a contingency step, at 800 psi shut the main steam isolation valves (MSIVs). The senior reactor operator on-shift made a decision not to shut the MSIVs based on indications that reactor coolant temperatures were rising and steam generator pressure was returning to its normal hot shutdown value. Additionally, NNECO management felt that the operator's actions were reasonable and expected based on the restoration of critical plant parameters to the expected values and on the preservation of a monitored and controlled pathway for the ongoing primary-to-secondary leakage.

Inspector assessment of procedural adherence included consideration of NNECO expectations as documented in ACP-QA-3.02E, "Procedural Compliance," and OP-2260, "Emergency Operating Procedure (EOP) Users Guide." ACP-QA-3.02E states full and total compliance is expected for controlled procedures. OP-2260 states that if an EOP action results in an expected plant response and that response is obtained, then the user of the EOP passes to the next step or substep. Inspector evaluation of plant conditions at the time EOP-2525 was being implemented concluded that the expected response occurred, and thus the intentions of OP-2260 were preserved, and the operators were in control of the plant.

Root Cause and Corrective Actions

NNECO determined that the root cause of the event was the initiation of a reactor shutdown without fully securing the main turbine and the lack of a specific step in the plant shutdown procedure OP-2205 containing criteria for closing the MSIVs. NNECO identified that contributing causes for the trip were inter-crew coordination, crew communications, and pressures felt from external NNECO and NRC management interest in the status of the plant shutdown.

NNECO corrective actions included: addressal of performance deficiencies within the operating crew, additional procedural improvements to the plant shutdown procedure to address specific plant conditions to close the MSIVs, and reinforcement of management expectations for senior licensed individuals to maintain overall plant perspective and oversight.

NNECO reviewed specific crew performance issues for the event for programmatic deficiencies and found none. The conclusion was based on management expectations presented to the crew and ongoing observation of the balance of the crew's performance. The inspector noted that the crew associated with this particular event successfully performed a plant shutdown in April, 1991, with a larger primary-to-secondary leakage value and an extraction steam line rupture in progress (refer to Inspection Report 50-336/91-09).

Assessment and Conclusions

Inspector assessment of the reactor trip concluded that the plant responded as designed and requirements were preserved. Appropriate reporting criteria were met. The operating crew performance was deficient in inter-crew communications and in selected coordination efforts. NNECO planned corrective actions were appropriately focused on crew performance deficiencies. Procedural adherence prior to and after the trip was acceptable. NNECO also identified an area of improvement in the plant shutdown procedure to develop a specific criterion to close the MSIVs.

3.3 Outage Activities

On May 25, based on an increase in primary-to-secondary leakage in the No. 2 steam generator, NNECO decided to shut the unit down to identify and repair the cause of the leakage. The initial shutdown duration was 18 days to accomplish a limited eddy current inspection program. On May 30, NNECO management decided to extend the shutdown to increase the scope of the steam generator examinations.

The inspection of outage activities included plant shutdown activities, licensee controls during mid-loop operations, observations and assessments of the work control center, and outage maintenance activities.

3.3.1 Plant Shutdown Activities and Steam Generator (SG) Releases

The inspector reviewed plant shutdown and cooldown activities in progress on May 25 and 26 to independently assess reactor safety and plant conditions. The inspector also reviewed actions by control room operators and chemistry personnel relative to the following plant procedures: AOP 2569, "Steam Generator Tube Leak"; OP 2204, "Load Changes"; OP 2323A, "Turbine"; OP 2205, "Plant Shutdown"; OP 2206, "Reactor Shutdown"; OP 2207, "Plant Cooldown"; EOP 2525, "Standard Post-Trip Actions"; and, EOP 2526, "Reactor Trip Recovery." The inspector verified that the activities to shut down the plant were generally in accordance with NNECO procedures and regulatory requirements. An exception to generally good operations crew performance is discussed in Section 3.2 of this report relative to the automatic reactor scram on low steam generator pressure.

Plant operators completed the initial phase of the cooldown on May 25 - 26 using the main condenser as the plant heat sink to minimize the release of radioactivity. This action was taken as a conservative measure since the tube leakage was small and the concentrations of radioactivity were low in the steam generator blowdown and the off gas system effluent. The cooldown was completed using the atmospheric dump valves per the normal operating procedure OP 2207.

The releases via the blowdown and offgas effluent pathways were monitored by radiation monitors RM 5099 (off gas) and RM 4262 (blowdown). Both channels are used to monitor the blowdown liquid effluent pathway. The offgas monitor is not the final gaseous effluent monitor, since the offgas discharges are routed through the Millstone 1 stack, a monitored release point. The alarm and trip setpoints for both radiation channels are established in accordance with the Radiation Monitoring Manual and the Offsite Dose Calculation Manual using methodologies that will assure that releases will not exceed 10 CFR 20 limits.

The inspector reviewed the blowdown and off gas radiation monitoring channels during the initial part of the plant cooldown on May 25 to verify that the channels were functioning properly and that releases were below the established limits. Based on the leakage rate, measured isotopic concentrations, and a review of the setpoint calculations for RM channels 5099 and 4262, the inspector verified that the releases were less than the 10 CFR 20 limits. The inspector also reviewed the stack effluent monitor during the plant shutdown and noted that the releases via that pathway remained below the normal background levels for the channel of 10 counts per second.

The steady state primary-to-secondary leakage rate was 55 gallons per day (0.038 gpm), or less than half the technical specification limit of 0.1 gpm. The inspector reviewed operator actions in response to increases in SG No. 2 leak rate, such as at 5:09 p.m., when the blowdown radiation monitor spiked to 13,000 counts per minute, causing an automatic isolation of the blowdown system. Inspector review determined that actions were taken as required by procedure AOP 2569, "Steam Generator Tube Leak," including actions to respond to the alarm, verify blowdown isolation, monitor the nitrogen-16 channels and to obtain chemistry samples.

The inspector verified that NNECO increased radiological and chemistry sampling of the secondary system to verify conditions throughout the plant shutdown. Chemistry analysis was used to supplement radiation monitors to assess the radiological release rates. The inspector noted that operators removed the steam jet air ejector monitor (RM 5099) from service at 5:25 p.m. on May 25 due to a design problem that resulted in sample line flooding at reduced loads. The inspector verified that the requirements of TS 3.3.3.9 and Table 3.3-12 were met with RM 5099 unavailable, since the blowdown monitor remained in service. Additionally, chemistry samples were taken in accordance with the required action section of the specification.

NNECO evaluated the releases from the No. 2 SG resulting from the tube leak and provided the estimate to the NRC for comparison with regulatory limits. The release estimates were based on radiation monitor readings and steam generator samples, and covered the period from midnight on May 25 until the plant entered cold shutdown on May 26. The total body and organ doses for the liquid effluent pathway were estimated to be 3.8X10-4 mRem and 3.88X10-3 mRem, which were less than the quarterly limits of 1.5 mRem and 5.0 mRem, respectively. The gamma and beta doses from noble gas effluent were estimated to be 1.5X10-6 mrad and 4.18X10-6 mrad, which were less than the quarterly limits of 5 mrad and 10 mrad, respectively. Based on the above, the TS 3.11 release limits were not exceeded as a result of the tube leak.

In summary, generally good performance was noted by plant personnel to complete the plant shutdown following the start of a leak in the No. 2 SG on May 25. The operators demonstrated a good regard for safety by choosing shutdown methods that minimized offsite releases. An exception to generally good performance was the operation crew actions just prior to the automatic reactor scram while shutting down the reactor. These actions had minimal impact on plant safety.

3.3.2 Control of Mid-loop Operations

The inspector reviewed licensee implementation of his commitments to NRC Generic Letter 88-17, "Loss of Shutdown Cooling." The program consisted of procedure OP-2301E, "Draining the Reactor Coolant System," and operations night orders. The inspection included a review of available reactor coolant system vent path, vessel level monitoring, availability of injection sources, control of work activity affecting shutdown cooling performance, incore vessel temperature monitoring, operability of shutdown cooling monitoring instrumentation, containment integrity controls, and operability of control room alarms associated with vessel level indications.

At the initiation of mid-loop controls, the initial vent path was the power-operated relief valves, and during nozzle dam installation, the pressurizer manway. Licensee actions were in accordance with generic letter commitments.

Level Monitoring Instrumentation

Reactor vessel level instrumentation available to operators were the two trains of reactor vessel level monitoring system (RVLMS), the No. 1 reactor coolant system hot leg level transmitter LT-112, and a local level indicator LI-112 consisting of a float in the standpipe with high visibility magnetic flags attached to the standpipe. The remaining narrow range indication in the No. 2 hot leg, LT-122, was not operational. Control room operators were cross-checking level outputs during the reactor coolant draindown sequence and logging hourly all the available level readings. The inspector observed satisfactory level outputs. Operable reactor vessel level channels met the generic letter commitments in that two independent level indicators were available during reduced inventory operations.

Inspection review of procedure OP-2301E determined that the No. 2 hot leg level instrument is required to be operable pursuant to prerequisite step 3.11.6. Prior NNECO investigation determined that inoperability of the No. 2 hot leg instrument was an ongoing problem. In April 1991, the licensee processed procedural change 3 to Revision 15 of OP 2301E to remove the requirement that indicator LT-122 be operable. OP-2301E, Revision 16, issued on May 1, 1991, did not incorporate the intended change. This was an oversight. NNECO attributed this procedure problem to administrative error between the procedural upgrade program and the timing of the procedural change, in that both occurred within a few days of each other. The change was corrected to address the inoperable condition of LT-122.

The inspector reviewed the performance of the reactor vessel level instrumentation from April 30 through May 2, 1991. This period was during the previous outage to inspect and repair damaged steam generator tubes. The reactor was in reduced water inventory with level being maintained at approximately mid-level of the hot leg nozzles.

The inspector observed that diverse methods for monitoring reactor vessel level were being used. In discussions with licensee personnel, a high degree of confidence was placed on the post accident Reactor Vessel Level Monitoring System (RVLMS). The system of heated junction thermocouples provided two channels of discrete level monitoring at the top, center line, and bottom of the reactor vessel hot legs by monitoring level within the reactor vessel. The system was supplemented with (unheated) core exit thermocouples. The system, which is in service during reactor operation, was available because the reactor vessel closure head was in place during the outage. The inspector observed the licensee taking action to add inventory based on a change in the center line level indicator output.

The licensee discovered various problems with the three other channels of level instrumentation during this period. A narrow range level instrument, L-122, which uses an ultrasonic probe attached to the Reactor Coolant System (RCS) Hot Leg No. 2 piping, was inoperable due to software start up problems. Repairs were being supported by the vendor. Additional problems were discovered with RCS Hot Leg No. 1 wide range level indicator and level sensor. These devices operate through a standpipe which is vented to the RCS. The licensee discovered that the vent line was improperly designed and may have allowed a

differential pressure equal to about five inches of water to develop between the standpipe and the RCS. This was corrected by installing larger vent tubing and taking care not to allow low points which would create water traps.
An additional problem was discovered with level gauge LG-112, which is monitored by closed circuit television (CCTV) in the control room. The gauge glass uses indicating flags to display water level to the CCTV camera. Licensee personnel discovered that the flags may stick and not change position as water level changes.

Licensee representatives expressed their concern over performance of these systems and continued to take actions to resolve these individual deficiencies.

System Alignment

The inspector verified the flow path valve alignments and the emergency power source availability, and concluded that two charging pumps and one high pressure safety injection pump were available during mid-loop operations. At the time of the mid-loop inspection, no corrective or preventive maintenance on the reactor coolant system, chemical and control volume system, and the shutdown cooling system was performed. One equipment tagout (2-1043-91, a bypass of the sump recirculation actuation signal) and a bypass jumper (2-91-34) were installed to prevent inadvertent loss of the low pressure safety injection pump on shutdown cooling.

The inspector verified all available instrumentation to monitor shutdown cooling performance was operable and within the calibration frequency. Control of containment integrity was in accordance with commitments to NRC Generic Letter 88-17. At one point during mid-loop controls, the containment equipment hatch was opened for approximately ten hours. NNECO performed a safety evaluation for this condition and provided training to assigned mechanics and electricians. Further, NNECO performed a timing run to preserve full closure within two hours in the event of a postulated loss of decay heat removal. The timed evolution to bring the equipment hatch from full open to closure was nine minutes. Discussions with the mechanics and electricians determined that they were familiar with their responsibilities and actions. The licensee demonstrated a strong commitment to mitigation and prevention of a loss of decay heat removal during mid-loop controls.

All available core exit thermocouples were in service pursuant to requirements procedure OP-2301E. The licensee complied with the commitments of Generic Letter 88-17. Control room alarms for low vessel level were operational for the RVLMS and LT-112 transmitter.

During mid-loop operations, NNECO management preserved operability of both emergency sources of power and the cross-connect power from Unit 1. In addition, administrative controls and reviews were implemented on the onsite transformer work to reinforce the availability of electrical power necessary to maintain shutdown cooling operations. The licensee assured the availability of the necessary controls to assess and to mitigate the consequences of a loss of shutdown cooling.

3.3.3 Activities of the Work Control Center

The work control center is located next to the shift supervisor's office in the control room. The outage coordinator is an experienced senior reactor operator. The coordinator is also qualified as a shift technical advisor. The coordinator was knowledgeable of work activities in progress and details of the components taken out of service. The outage coordinator controls the major work activities during the outage, and supervises daily planning sessions with the maintenance and instrument and control department planners. The coordinator also attends unit daily planning meetings in which he is a major contributor. In addition, the coordinator approves all work orders related to the outage and directs the implementation of the scheduled activities. At the time of the inspection, the coordinator utilized two assistants to confirm work activities at the work location and to review the necessary tagouts. NNECO maintained two outage support shifts (10 hours/shift) during the outage.

3.3.4 Major Maintenance Activities

Major maintenance activities during the forced outage included: overhaul of the main steam non-return check valves, overhaul of the "A" service water pump, replacement of relays for both control element drive mechanism motor-generator sets, non-emergency 4.16 KV bus inspections, installation of charging pump relief valves, and scheduled preventive maintenance and surveillance. Steam generator inspections and repairs were the major contributors to personnel exposure and dictated the outage duration.

3.3.5 Conclusion and Assessments

Activities to shutdown the plant were in accordance with NNECO procedures and regulatory requirements. Procedures and controls in place during reduced inventory operations were adequate to mitigate the consequences of a postulated loss of shutdown cooling.

3.4 Operator Requalification Exams

During the week of June 3, the NRC administered requalification examinations to eight senior reactor operators and six reactor operators. In addition to the requalification examinations, inspection of open item 50-336/88-10-04 and the medical program for licensed operators were conducted. One weakness was observed during the examinations in that the senior reactor operators had some difficulty accessing information on the plant process computer. This item was discussed with NNECO management for consideration in future operator training. The results of the examination will be documented NRC in inspection report 50-336/91-13.

3.5 NRC Information Notice 90-54: Summary of Requalification Program Deficiencies

On June 18, the inspector attended a video presentation on emergency operating procedure (EOP) implementation guidelines. The licensee developed the video primarily as an operator training tool to present management expectations on command-and-control and the conduct of operations. The tape, in addition, presented issues discussed in NRC Information Notice 90-54.

Included in the video were NNECO identified "good practices" from various crew requalification simulator scenarios, training department insights, and procedure references (OP-2260, "EOP Users Guide," and ACP-6.01A, "Structured Communication"). The video setting was the Millstone 2 simulator and control room. The tape defines command-and-control, the preservation of the senior reactor operator roles, and "key" activity control by senior licensed operators.

The licensee presented the video to two of the six operating crews at Millstone 2 and to the unit operations managers from Millstone 1 and 3.

The inspector concluded the video was well-structured and clearly defined NNECO management expectations regarding professionalism and operator performance.

4.0 RADIOLOGICAL AND CHEMISTRY CONTROLS (IP 71707)

4.1 Posting and Control of Radiological Areas

During plant tours, the posting of contaminated, high airborne radiation and high radiation areas was reviewed with respect to boundary identification, locking requirements, and appropriate hold points.

The inspector had no significant observations.

4.2 Radiological Controls During the Forced Outage

Radiological controls during the outage focused on work activities in containment, improvements in monitoring airborne radioactivity from the primary steam generator plenums, and outage exposure estimates.

During containment tours, the inspector evaluated various control point activities, radiation protection personnel actions, briefings with workers, and survey results. The inspector concluded that implementation of radiation health practices was acceptable. The relocation of the contamination clothing dress-out area outside of containment and the moving of the access point for steam generator work to as-close-as-possible to control points were noted as positive initiatives. Technicians were cognizant of survey results, monitoring requirements, and necessary pre-job training requirements for steam generator plenum work. The online monitoring and control of airborne radioactivity from the primary steam generator plenums were noted enhancements. The continuous online monitoring provided representative samples, minimized overall exposure to technicians performing periodic sampling, and provided tools necessary to maintain personnel exposure as-low-as-reasonably achievable (ALARA).

A significant exposure reduction was noted in steam generator manway removal. Historically, this job task required 2.0-3.0 man-rem exposure. Exposure during this outage for manway removal was 0.690 man-rem. Reduction in exposure was attributed to good job coordination of three separate work force teams, preserving the efficiency of work by minimizing stay time, and the use of experienced personnel for the job. The original exposure estimate for the outage was 27.5 man-rem. On May 31, the exposure estimate was increased to 135.0 man-rem based primarily on the installation/removal of steam generator nozzle dams, the expanded steam generator inspections and repairs, and the additional staging and shielding necessary to accomplish the inspections. The steam generator work accounted for 77% of the total outage exposure estimate. At the end of the inspection period, actual man-rem exposure was at 46.7 man-rem, approximately 30% below projected values.

In conclusion, radiation protection controls in containment were well implemented, and a notable reduction was achieved in the steam generator manway removal. Exposure estimates were effectively changed to take into account the additional steam generator inspection scope.

5.0 MAINTENANCE/SURVEILLANCE (IP 61726/62703/37701/92701)

5.1 Observation of Maintenance Activities

The inspector observed and reviewed selected portions of preventive and corrective maintenance to verify compliance with regulations, use of administrative and maintenance procedures, compliance with codes and standards, proper QA/QC involvement, use of bypass jumpers and safety tags, personnel protection, and equipment alignment and retest. The following activities were included:

-- AWO M2 90 06235 Preventive Maintenance on the "A" EDG, 6/10/91
-- AWO M2 91 05928 "C" Wide Range Nuclear Instrument Repairs, 6/11/91
-- AWO M2 91 06321 Insulation Tests on Electrical Bus 24B, 6/20/91
-- AWO M2 91 06319 Insulation Test on Potential Transformer, 6/20/91
-- AWO M2 91 04530 Overhaul of 2-MS-1A, 6/3/91
-- AWO M2 91 03765 Blowdown Radiation Monitor Troubleshooting

Activities observed were satisfactory.

5.2 Charging Pump Block Crack

During corrective maintenance on May 14 on the "A" charging pump, a mechanic identified that cylinder no. 1 was cracked from the outside face to the internal intake valve. The crack was in the pump block. The corrective maintenance in progress at the time was to replace the cylinder cap 0-rings due to identified leakage from the 0-rings. Upon identification of the crack, the charging pump was placed in an inoperable status.

History and Component Description

Historically, the unit has experienced charging pump block cracks. On May 25, 1984, NNECO submitted a 10 CFR Part 21 report to document the cracked blocks. In the time interval between 1982 and 1987, the licensee identified six cracked blocks. Prior to the May 1991 failure, the most recently documented cracked block was on the "A" charging pump which occurred in 1987. In addition to the submittal of the 10 CFR 21 report, the licensee has documented the previous events as informational licensee event reports (LERs): 82-36, 85-13, 86-23, and 87-05.

The charging pumps are manufactured by APV Gaulin, Inc., model number NP-18-3.1 TFS. The pumps are horizontally mounted, constant speed, three stage positive displacement, and have a rated flow rate of 44 gallons per minute (gpm). The pumps are classified as a QA category I component. The charging system is equipped with three pumps of which a minimum of two are required to be operable per Technical Specification 3.1.2.4.

During postulated emergency operations, the charging pumps inject concentrated boric acid into the reactor coolant system for reactivity control. A safety injection actuation signal will automatically start the charging pumps. In normal operation, the charging pumps control programmed pressurizer level and are used for dilution, or boration within the reactor coolant system.

Corrective Actions

Licensee corrective actions included: initiation of plant incident report 91-53, replacement of the pump block pursuant to the ASME Section XI replacement program, and decontamination of the old block. The new block was installed and the pump was returned to operable status on May 22, 1991.

Actions to prevent recurrence include past modifications to the block design to minimize cracking, tracking, and trending of block performance and initiation of a preventive maintenance program geared to periodically non-destructively examine the cylinder walls for cracks. Modification of the block design included shot peening the cylinder/bore transition area, using a larger bore transition radii, and changing the block material from SA182 F304 stainless steel to 17-4 PH martensitic stainless steel. The "A" charging pump block did not have the recent block modifications and had approximately 8,431 hours of operation. The modified block design was intended to minimize the initiation of fatigue failure (cause of block cracking). NNECO has opted to preferentially operate the "C" charging pump to acquire longevity data on the new designed block. The lead charging pump has approximately three years (25,796 hours) of continuous operation without failure.

The current preventive maintenance program non-destructively examines the cylinder walls for cracks each time the pump is required to be repacked or before exceeding 3500 hours of operation. The inspector verified implementation of the program in the last year of operation with no noted deficiencies. The NNECO maintenance department tracks the operation time of each charging pump, the times of pump repacking, and rework activities to evaluate reliability of the new block design.

Conclusion and Assessment

NNECO actions to address the charging pump block crack were satisfactory. Improved performance was noted on the two charging pumps with the modified blocks.

5.3 LCO Maintenance

Objective

The objective of this inspection was to randomly select safety-related equipment subjected to the preventative maintenance (PM) program and to review implementation of this program during power operations. Also, the inspector determined whether the licensee has reasonable expectation that an on-line preventative maintenance program improves safety by making equipment more reliable and that the overall risk to safe operation of the facility decreases taking into account equipment inoperability (i.e., the performance of the preventive maintenance).

Scope

The components selected for this review included the emergency core cooling pumps/motors, charging pumps/motors, auxiliary feedwater pump/motors, and the emergency diesel generators. The preventative maintenance program for each of these components is documented under procedure MP-20701J, "Preventative Maintenance," and MP-2701F, "Lubrication."

Results

The interval and type of preventative maintenance varies based on the component. Table 1 identifies the component, the frequency of the preventative maintenance, and the governing procedure. The inspector noted in review of the preventative maintenance program and in discussion with cognizant maintenance personnel that all semi-annual, annual, and refueling frequency PMs require the equipment to be inoperable, and for the emergency diesel generators all PMs performed require inoperability.

Inspection of completed preventative maintenance activities performed in the year 1990 concluded that the average outage time for annual PMs were as follows: core spray pump/motor - 14 hours; high pressure safety injection pumps/motor - 10 hours; and auxiliary feedwater pump/motor - 4 hours. No refueling PMs (overhauls) were noted during power operations.

The maintenance department has a matrix established to coordinate the preventative maintenance program with required operational technical specification surveillances and in-service inspection surveillances. The matrix is planned for the entire year independent of scheduled outages. The specific implementation date for the equipment outage is chosen by the operations department.

The coordination of inter-system PMs is reviewed within the daily planning meetings as well as during operations review prior to work approval. The coordination of corrective maintenance activities is also input in the scheduled outage date for the equipment. The inspector determined that the preventative maintenance activity did not impact the allowable surveillance interval.

The licensee does not have a formal mechanism to track the outage duration versus preventative maintenance nor a formal process to include the risk assessment on the outage duration to perform preventative maintenance activities. Component reliability is the primary focus to assess the effectiveness of the preventative maintenance program. Indicators on corrective maintenance backlog duration, ratios of preventative maintenance/corrective maintenance, and rework items are tracked and reported on a quarterly basis to assess, in part, component reliability.

Qualitatively, the licensee believes that the overall safety benefit from the performance of preventative maintenance exceeds the loss of equipment availability to perform the maintenance activity. This conclusion is supported in part by the lack of repetitive, corrective maintenance on the components reviewed and, in addition, the plant design has increased redundancy for the high pressure safety injection pump/motor and charging pump/motor.

Two previous examples of refuel preventative maintenance accomplished prior to the commencement of the outage relate to intake bay work and the hydrolasing of the reactor building component cooling water heat exchangers. In both activities it was not NNECO's intention to shorten the outage, but rather to improve work efficiency. In both examples, the safety-related equipment is designed with increased redundancy, and the balance-of-plant equipment affected is generally considered of high reliability. This type of work activity required approval by a NUSCO management.

Conclusion

The inspector concluded that a reasonable expectation of improved equipment reliability and plant safety exists with respect to management's decision to perform preventative maintenance activities on-line. Continuing evaluation will ensure careful planning and safety benefits are presented prior to future preventative maintenance during operations or shutdown conditions.

5.4 Reactor Building Component Cooling Water Seal Return

On April 23 at approximately 9:55 p.m., an operator identified leakage on the reactor building component cooling water (RBCCW) return from the "A" reactor coolant pump (RCP) seal package. The leak was from a 4-inch stainless steel braided hose at a rate of approximately 10 milliliters per minute. At the time of identification, the RCP was not in operation and the plant was entering a cold shutdown condition.

The primary purpose of the RBCCW cooling to the RCP seals is to cool the coolant bleed-off to protect the elastomer material within the seal package. FSAR Section 4.3.3 states that no damage is expected to the reactor coolant pumps seal for up to five minutes without cooling water at normal operating temperature within the reactor coolant system. The cooling water to the RCP seals is considered non-essential according to the analyzed loss-of-coolant incident as documented in Final Safety Analysis Report (FSAR) 9.4.3.2.

The flexible RBCCW hose is attached on the inlet and outlet of the RCP seals in close proximity to the RCP motor. Each hose is approximately 4 feet 6 inches in length. The hose is procured under a QA requisition and a post-installation hydrostatic test is performed prior to operation. The flexible hose is incorporated into NNECO's preventive maintenance program for replacement every second refuel outage. The particular hose that was leaking was replaced in June 1990. The flexible hose material is rated for 300 degrees F, and pressure tested to 350 psi. The hard pipe connected to the hose is designed to ANSI B31.1, to material specification ASTM A-53B carbon steel with a designed pressure rating of 150 psi, and a design temperature rating of 200 degrees F. Normal system operating conditions are 65 psi at 95 degrees F. The inspector verified that the flexible hose is not considered a part of the ASME Section XI boundary and, therefore, not subject to actions pursuant to NRC Generic Letter 90-05.

NNECO corrective actions at the time of discovery were to initiate non-conformance report (NCR) 291-054 and, under a maintenance work order, to install a splash shield to protect the RCP motor in the event of a catastrophic failure of the flexible tube. The NCR disposition of the hose was "use-as-is." The engineering basis for the interim use of the hose was the limited leak size, the hose design ratings, the minimal effects on system performance, and the preventive measures for potential spray down of components in the region.

On May 30, NNECO replaced the hose pursuant to work order M2-91-04661. The failed hose was hydrostatically tested, and the failed section was identified. The cause of the failure was not determined at the end of the inspection period. Future inspections will evaluate the licensee efforts to determine the cause of the hose failure. NNECO actions were acceptable.

5.5 Previously Identified Items

5.5.1 (Closed) Unresolved Item 91-02-02: Improper Maintenance Work Control

This item was opened to evaluate NNECO actions to prevent recurrence of inadequate corrective maintenance activities that resulted in the reactor/turbine trip on January 10, 1991. Licensee corrective actions focused on the technical aspects associated with the problems in the electro-hydraulic control (EHC) system (cause of turbine trip) and the personnel performance aspects associated with the corrective maintenance.

The corrective actions associated with the EHC system were: pump discharge filter changeout, replacement of the 0-rings at the suction strainer, and replacement of the "B" EHC pump. The root cause for inadequate EHC system performance was air-inleakage past the 0-ring seating surfaces at the suction strainer and excessive oxidation of the holddown spring retainer sleeve assembly for the "B" EHC pump. The oxidation of the spring retainer sleeve assembly prevented stroke adjustments of the pump. Both anomalies resulted in failure of the supply pump to maintain constant system hydraulic pressure. Action to prevent recurrence was storage of the "spare" EHC pump in hydraulic oil to prevent oxidation.

Corrective actions for personnel performance were: letters of reprimand to all assigned workers; discussion of the event and lessons learned at maintenance department meetings; and management reinforcement of expectations for mechanics and job supervisors.

The inspector reviewed the licensees corrective actions and actions to prevent recurrence and considers this item closed.

5.5.2 (Closed) Unresolved Item 90-28-02: Steam Generator Primary Manway Leakage

This item concerned licensee actions to address inadequate maintenance procedure requirements and insufficient engineering oversight of repeating requisitions for gasket material used for the steam generator primary manways. The above items caused an unplanned outage on December 29, 1990, resulting from failures in the steam generator primary manway gaskets.

NNECO corrective actions changes to procedure MP-2705E, "Installation/Removal of Steam Generator Primary Manway Covers," and MP 2706A, "Pressurizer Manway Removal/ Reinstallation." The changes require that the quality control inspector verify correct part number and dimension of the gaskets prior to installation on the steam generator and the pressurizer manways. In addition, NNECO's investigation revealed that miscommunication between the stores and purchasing departments contributed to the reordering of unapproved gasket materials. Corrective action was to initiate a draft Purchasing and Materials Management Guideline on May 14, 1991. The guideline documents actions if the purchasing department initiates changes to purchase records.

In addition to procedure changes and draft guidelines, NNECO management discussed the need for procedural compliance with emphasis on personnel making necessary procedural changes prior to commencement of the work activity. The licensee concluded that engineering review is not required in non-QA procured stock, as documented in administrative control procedure ACP-QA-4.02C, and no change to that control is necessary since the procurement error occurred in the communications between the purchasing and stores departments. NNECO corrective actions were satisfactory.

This item is closed.

5.6 Observation of Surveillance Activities

The inspector observed and reviewed portions of completed surveillance tests to assess performance in accordance with approved procedures and limiting conditions of operation, removal and restoration of equipment, and deficiency review and resolution. The following tests and procedures were reviewed:

- OP-2316C, Steam Generator Leak Test, 5/30/91
- OP-2619A, Daily Technical Specification Surveillances
- OP-2301E, Drain Down of the Reactor Coolant System, 5/29/91
- OP-2397-1, Nozzle Dam Surveillances
- SP-2403B, 8 Engineered Safety Actuation System Undervoltage Bistable Test, 6/25/91
- SP-404AJ, Steam Generator Blowdown Monitor Calibration, 5/13/91

No significant observations were noted.

5.6.1 SP404AJ, Steam Generator Blowdown Monitor Calibration

On May 13, the inspector reviewed the results of troubleshooting activities performed under AWO M2-91-03765 on the steam generator blowdown (SGBD) radiation monitor, RM 4262. The AWO was issued to investigate the calibration of the monitor because of a high reading at 12,000 cpm. NNECO suspected the sample canister might be contaminated.

Work completed on May 12 under AWO 91-03765 determined that the sample canister was contaminated. A calibration check of the channel showed good correlation with the response obtained during the last formal calibration per SP 404AJ on April 16, 1991. The following count rates were recorded for the monitor during troubleshooting activities: (i) 1000 cpm - detector inside the lead shield with no sample canister; (ii) 4000 cpm - detector inside the lead shield with a dry (but contaminated) canister installed; and (iii) 12,000 cpm with the process stream valved into the monitor. The higher detector counts recorded with the process stream valved in was attributable to residual activity in the #1 steam generator secondary side caused by the hide-out return from the activity and tube leakage present before the plant shutdown.

The monitor was returned to service and considered operable upon completion of troubleshooting activities. The inspector noted that NNECO was taking credit on May 13 for the blowdown monitor to meet the requirements of Technical Specification 3.3.3.9b, since the steam jet air ejector monitor (RM 5099) was out of service. The technical specifications require that either monitor be in service during plant operations to automatically isolate the blowdown liquid effluent pathway in the event of a steam generator (SG) tube leak. NNECO was sampling the SGs once per 12 hours as an initiative to follow SG performance during the plant startup. This action would satisfy the TS LCO if it is assumed that both the SJAE and SGBD monitors were inoperable.

SP 404AJ Test Method

The inspector reviewed the calibration test method performed in accordance with SP 404AJ. The calibration used in the procedure is to expose the detector to three concentrations of solutions containing radioisotopes in a liquid sample container with an equivalent geometry to the process chamber. The detector response to each source is plotted to obtain an average calibration factor. The procedure does not use a blank cartridge filled with deionized water to obtain the "background" count rate for the detector. Instead, the normal sample canister is removed from the lead-shielded housing, the detector is reinstalled in the housing, and the background readings are recorded for use in correcting the detector readout when exposed to the calibration sources.

The inspector questioned whether the calibration method was adequate if no "reproducible geometry" was established when measuring background radiation for determining the background correction factor. A similar question regarding background corrections for the reactor building closed cooling water (RBCCW) process radiation monitor was addressed in Inspection Report 336/90-06. The question addressed in that report was whether the RBCCW calibration method was adequate if no detector-source geometry was established for the background corrections using a radioactively "clean" canister.

Prior NRC reviews showed that for a "clean" sample canister, it is meaningless to establish a detector-source geometry for background radiation that is entering the detector from all directions in a 4-pi geometry. But, in the present instance, the SGBD radiation monitor sample canister was found to be radioactively contaminated. A fixed detector-source geometry is a meaningful concept when the radiation is present in the process canister as the source of "background" radiation.

The inspector assessed the error introduced by the test method in the existing procedure. NNECO calibrates the monitor using SP 404AJ by introducing three concentration of radioactive solutions and recording the channel output. Typical calibration values were as follows: Test Point 1 - Source A (2.38x10-5 uCi/cc) - produced a channel output of 6000 cpm; Test Point 2 - Source B (5.99x10-4 uCi/cc) - produced a channel output of 2x10+5 cpm; and Test Point 3 - Source C (2.92x10-3 uCi/cc) - produced a channel output of 5.5x10+5 cpm. The background correction used to adjust the channel output was the 100 cpm value obtained with the detector installed in the shielded housing without the canister. A more accurate background correction would be the 4000 cpm value recorded with the detector and the canister installed in the housing. Both background numbers results in an insignificant correction to the data recorded at the upper two test points.

The failure to account for the contamination activity as part of the background correction factor will have the most significance on the lowest test point. The lower background correction will have the effect of introducing a systematic error in the channel calibration that will in all cases cause the channel to overestimate the radioactive source term actually present in the process stream. Using a regression analysis of actual test data, the inspector determined that the systematic error would have an insignificant effect on the slope of the detector response curve, and thus, the channel calibration factor and the setpoint determinations.

A channel output that is erroneously biased high will conservatively isolate SG blowdown in response to tube leakage. Since NNECO relies on chemistry analysis of the process effluent to establish absolute radioactive concentrations and source term, the erroneously high channel output has minimal operational or safety significance. The inspector concluded that NNECO's decision to consider the monitor operable was acceptable.

Notwithstanding the above, the inspector concluded it is important to make primary calibration test methods as accurate as possible. NNECO has also recognized the need to improve monitor calibration procedures, as described in the Radiation Monitor Manual prepared by the Radiological Assessment Branch in NUSCO engineering. The RM manual states that when primary calibrations are performed in the field and significant background corrections are necessary, the calibrations should be done using the exact geometry. NUSCO recommended that procedures be reviewed and improved as necessary to assure more exact geometries are employed, including the use of blank sample canisters filled with clean water.

NNECO initiatives to improve radiation monitor calibration techniques by adding the noted enhancements are desirable. NNECO is revising procedures in phases, with the radiation monitor and maintenance procedures scheduled to be upgraded in 1991 and 1992, respectively. The I&C supervisor stated the present procedures are scheduled to be revised as necessary per the RAB recommendations by December 31, 1991. The inspector had no further questions in this area.

5.7 Previously Identified Items

5.7.1 (Closed) Unresolved Item 89-18-01: Implementation of NRC Guidance for Anticipated Transient Without Scram (ATWS) Equipment Operability and Surveillance Requirements

This item concerned the licensee implementation of NRC guidance for the ATWS equipment operability and surveillance requirements as specified in the unit's technical specifications.

The NRC staff continues to evaluate the need for technical specification operability and surveillance requirements, including actions considered appropriate when operability requirements cannot be met to ensure that equipment installed per the ATWS rule will be maintained in an operable condition.

The evaluation of NRC staff action is controlled under TAC number 59114 and, therefore, future decisions based on the need for surveillance requirements for ATWS will be controlled under this number, independent of regional inspections. This item is closed.

5.7.2 (Closed) Unresolved Item 89-11-02: Root Cause Determination on Failure of a Containment Isolation Valve

This item concerned NNECO actions to identify the root cause for the failure of containment isolation valve 2-EB-91 (inboard containment isolation valve for the hydrogen purge system). The valve failed in that it did not stroke within the required acceptance criteria. The event was documented in licensee event report (LER) 89-006-00 and NRC routine inspection report 50-336/89-11.

Inspection of licensee corrective actions concluded that the root cause was not identified. NNECO corrective actions to identify the failure included corrective maintenance overhaul on the valve and successful post-maintenance restoration of the valve within its surveillance acceptance criteria.

The inspector reviewed surveillance results since failure of the valve on May 4, 1989, to assess current performance and operability of valve 2-EB-91. Three surveillances: SP-2605G, "Containment Isolation Valve Operability Test"; SP-21129, "Quarterly In-Service Inspection Testing of Containment Isolation Valves"; and SP-2605D, "Containment Leak Test - Type C"; periodically surveil operation of valve 2-EB-91. The inspection identified no other documented failures of the valve and, in addition, no identified corrective maintenance activities were accomplished on the valve.

The inspector concluded that NNECO corrective actions were reasonable since the additional valve failures occurred. The enforcement disposition of this event were documented in Inspection Report 50-336/89-11. This item is considered closed.

6.0 ENGINEERING/TECHNICAL SUPPORT (IP 92701/93702)

6.1 Change in Engineering Organization

NUSCO implemented changes in the offsite engineering organization that became effective on June 1, 1991. Engineering support of projects was utilized, and selected functions and field services were consolidated. The change completed a phased restructuring of the engineering organization designed to complete the transition from a construction to an operations focus; to improve decision making, communication and teamwork; and to clarify responsibilities.

Mr. E. Mroczka is the Senior Vice President of Nuclear Engineering & Operations, with the following groups reporting to him:

-- Mr. E. DeBarba, Vice President of Engineering Services, with Project Services (H. Risley), Engineering Support (R. Harris) and Field Services (G. Johnson)

-- Mr. F. Sears, Vice President of the Nuclear Services Division, with Nuclear Engineering (M. Bonaca), Stores & Record Services (G. Baston), Nuclear Training (J. Black), Quality Services (D. Nordquist), and Environmental Programs (W. Renfro)

There was no change to the Nuclear Operation Division, with Mr. W. Romberg as Vice President and with the Millstone Station Director (S. Scace) and the Haddam Neck Station Director (J. Stetz) reporting to him.

The inspector noted that the changes had no effect on communications and responsibilities in the operational line of authority established for all three Millstone plants in the technical specifications. No unsatisfactory conditions were noted.

6.2 Previously Identified Items

6.2.1 (Closed) Unresolved Item 89-06-01: Administration of SNT-TC-1A Tests as "Closed Book"

This item was opened based on licensee action to address the identified weakness in the administration of examinations in accordance with the American Society of Nondestructive Testing (ASNT) document SNT-TC-1A.

The licensee has revised its certification program to comply with the SNT-TC-1A requirement for "closed book" administration. The test questions, which are greatly improved over the questions in the original program, are computer-data-based and can be selected at random by the computer. The inspector reviewed the licensee's revised program, including specific tests and the accompanying reference material, and concluded that the tests are administered as "closed book" tests, and the program does comply with the SNT-TC-1A requirement.

Based on the above, this item is considered closed.

6.3 Emergency Diesel Generator Air Start Vent Solenoid Classification

Description

On April 11, the NNECO maintenance personnel questioned the qualification of the emergency diesel generator (EDG) air vent (AV) solenoid valve quality (Q) indicators. The basis for the qualification evaluation (i.e., Master Equipment Parts List (MEPL)) were questioned because (i) the evaluation did not address the credible failure mode of the valve failing open and (2) the solenoid electrical power is from the same circuitry as the air start solenoids without any isolation devices.

The function of the AV solenoid is to vent air from the air start distributor when the air start valves are closed. The venting of air prevents wear on the EDG air start distributor cam and the pilot valves. During intervals when the EDG is in standby, the AV valve is energized and opened. During the startup of the EDG, the valves de-energize and close to port starting air to the distributor.

In 1986, pursuant to administrative control procedure (ACP) QA-4-03B, a determination of the Q applicability for the AV valves was completed. The evaluation concluded that the AVs were non-Q components based on the failure mode to de-energize thus allowing starting air to the distributor as designed and creating no impact on EDG performance.

NNECO engineering reevaluated the quality assurance controls for the solenoid based on the maintenance department request. The evaluation concluded that the AVs for each diesel are category I Q components. The revised failure mode addressed a "hot short" failure as a result of a seismic event. The solenoid failure would result in the EDG starting normally; however, the output breaker would not automatically close in on the emergency bus on a loss of normal power event. The EDG output breaker would not function automatically because the "hot short" in the AV solenoid would blow fuses in the breaker control power circuit. The reevaluation resulted in the initiation of non-conformance report 291-60.

An assessment of continued operability was developed by NNECO engineering. The EDG was determined to be operable with known non-Q components because (i) the qualified solenoid characteristics were identical to the installed solenoid, (ii) the installed solenoids have been subjected to EDG running vibration with no known failures, (iii) operators are trained to respond to a failure of the diesel output breaker to automatically close in on the emergency bus. Prior to 1986, no specific component quality evaluation was developed.

The AV solenoids were replaced with qualified solenoids during the forced outage which began on May 25.

Conclusion and Assessment

This item represents a lack of thorough engineering evaluation to appropriately classify components. The engineering evaluation was deficient in that the AV failure mechanism was not fully evaluated with respect to the integrated system interaction.

Further, once the results of the reevaluation were known, NNECO engineering did not inform the operations department on a timely basis regarding the potential operability impact on the EDGs. Specifically, it was known by NNECO engineering on approximately May 2, that the solenoid failure mechanism could affect operation of the EDGs yet the operations department was not formally made aware of this issue until May 28. The judgement and conclusion of the operability assessment were adequate.

NNECO questioning of past quality assurance controls for the AVs exemplified a good safety ethic. The untimely communications had no significance in this case since there was no impact on diesel operability. The NRC is concerned that the communication of potentially reportable issues could be improved.

6.4 Electrical System Compliance with GDC 17

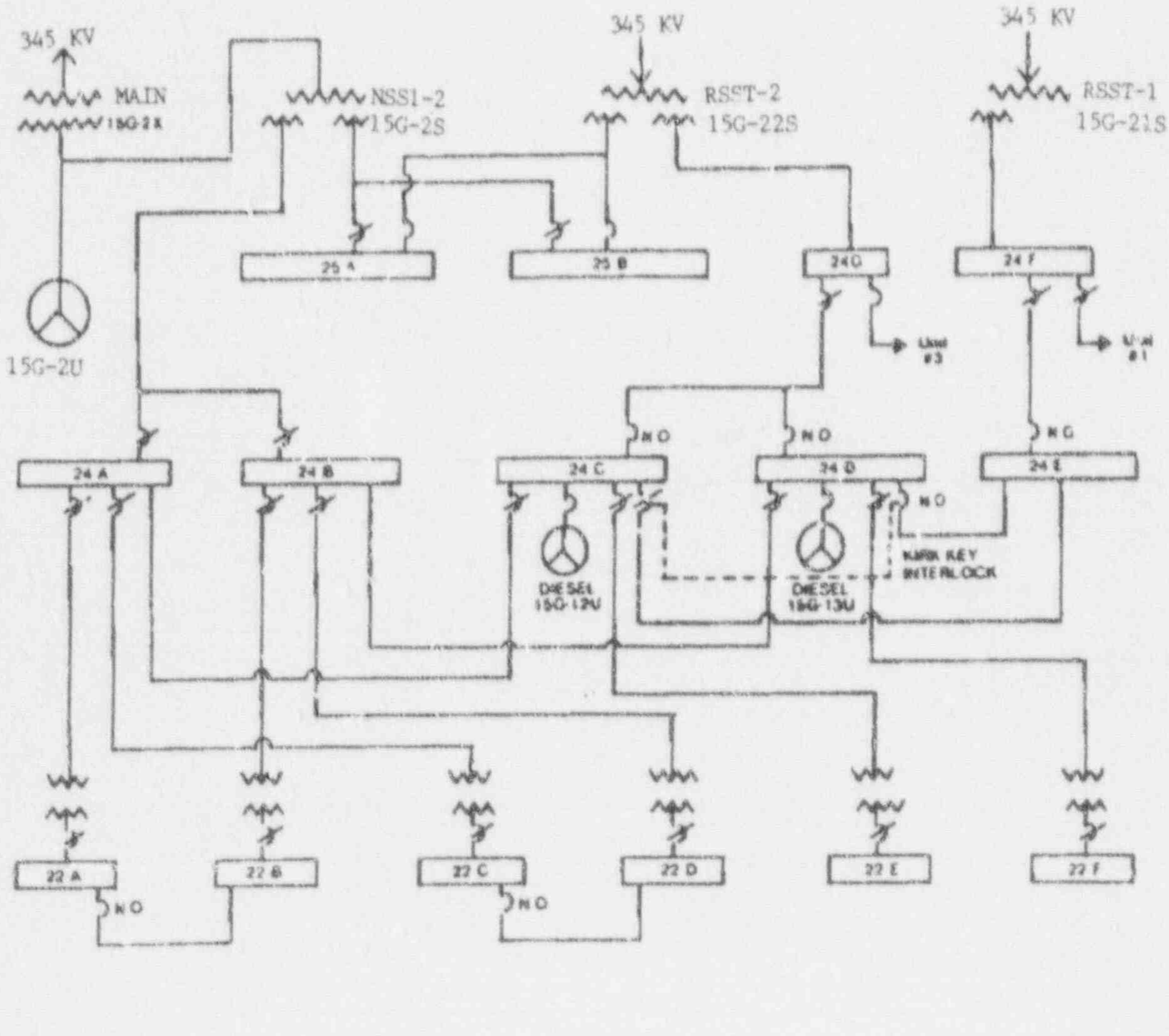
NNECO performed a safety system functional inspection (SSFI) on the reactor building closed cooling water (RBCCW) system in August 1988. The inspection reviewed the ability of the RBCCW system to perform its intended design and safety functions and included an evaluation of the adequacy of emergency power for RBCCW components. At Millstone 2, two redundant diesel generators provide the Class 1E emergency AC power for the RBCCW and other safety systems. Several deficiencies identified during the SSFI were dispositioned at the time of the inspection and resolved.

The electrical system issue discussed below was SSFI Observation No. 61 and is the last item to be resolved by NNECO. The issue involves the postulated loss of the emergency diesel generator (EDG) supply under certain scenarios involving a faulted condition combined with the occurrence of a limiting single active failure. NUSCO engineering reviews of the electrical issue raised the question as to whether the Millstone 2 electrical system design was in compliance with certain 10 CFR 50 Appendix A general design criteria (GDC), and specifically with GDC 17, Electric Power System Design. Compliance with GDC 17 is a requirement for Millstone 2 as a condition of the plant operating license.

The issue was discussed within NUSCO, and after different engineering groups could not arrive at a consensus, NNECO hired an outside consultant to study the matter. The results were reported to the Manager of Generation Facilities Licensing by a memorandum dated April 30, 1991, with the conclusion that Millstone 2 was in compliance with GDC 17. At the end of this inspection period, the matter was still under NNECO review with the study recommendations presented to the NUSCO Vice President for Engineering Services for dispositioning.

Millstone 2 AC Electrical System Design

At Millstone 2, AC electric power is provided by the offsite power system comprising a normal station service transformer (NSST) and a reserve station service transformer (RSST), both powered from the 345 KV switchyard. The NSST is powered by the Millstone 2 main generator and provides normal power to 4160 volt non-Class 1E buses 24A and 24B. Class 1E buses 24C and 24D are powered from the non-safety buses via cross tie breakers. Buses 24A and 24C comprise onsite electric division A; buses 24B and 24D comprise electric division B. Refer to the unit two electrical distribution system figure below.



On loss of normal power, Class 1E buses 24C and 24D, can each be powered by an emergency diesel generator (EDG) in about 10 seconds. Additionally, the RSST provides automatic backup power to buses 24C and 24D in less than one second. One type of protection relays senses an undervoltage condition on the safety buses, which operates to isolate the safety buses 24C and 24D and to start and connect the EDGs to the buses. Another type of protective relay senses overcurrent conditions indicative of an electrical fault. The 4160 volt overcurrent protection scheme is coordinated to isolate the unaffected portions of the AC distribution system from a faulted section. The issue essentially involves the lack of coordination between the overcurrent and bus undervoltage protective relay schemes used in the 4160 volt system.

Postulated Fault Scenario

The GDC 17 issue involves the response of the 4160 volt AC distribution system to the following postulated scenario. The plant must be designed to satisfactorily respond to a postulated large break loss of coolant accident (LOCA) followed by a single failure. The LOCA will result in an automatic reactor and turbine trip and the automatic transfer of both onsite electric divisions to the offsite RSST supply. If the single failure is a fault on one of the onsite electric divisions (e.g., Division A), excessive current will flow on the faulted bus. Low voltage will occur on the Division A and B buses, since both are fed from the same transformer winding. The undervoltage relays will trip before the Division A supply breaker on overcurrent to isolate the fault. Thus, instead of the fault being isolated by a trip of the Division A overcurrent relays, the fault affects both divisions and appears to the operators to be a loss of offsite power. Since no overcurrent relays activate, no indication is available to the operators to signal that an overcurrent initiated the undervoltage trip condition.

Once divisions A and B have been isolated from the RSST by the operation of the undervoltage protection scheme, both Millstone 2 EDGs will start and connect to their respective buses. The division A EDG should then trip on overcurrent as a result of the fault remaining on the division A bus. The division B bus would be powered from its EDG, unaffected by the division A fault. The systems and equipment powered by division B alone has the necessary capacity and capability to cool the core and safety shutdown the plant should this scenario occur.

Event though the breaker supplying power to division B from the RSST tripped as a result of the undervoltage relay operation, the breaker could be shut by the operator from the control room to re-energize division B from the offsite supply. This action should not be necessary immediately because of operation of the division B EDG. However, if such a fault occurs during a LOCA, it is doubtful the offsite circuit would be restored by the operator within a few seconds because of the lack of overcurrent indication and due to the activity in the control room. The question raised by NNECO is whether Millstone 2 meets the requirements of GDC 17 if the AC distribution system design (inclusive of the protection schemes) would not preclude the division A fault from causing the loss of immediate supply of power to the unaffected onsite electric division from the offsite RSST supply.

GDC 17 Requirements

NNECO review determined that the following GDC 17 requirements were the most relevant to the question regarding compliance with the offsite circuit criteria. Two electric power systems are required, onsite and offsite. Each power system shall provide its safety function assuming the other is not functioning. The safety functions are to maintain reactor design limits during anticipated transients and to cool the core during postulated accidents.

The onsite power system shall meet the single failure criterion. Offsite power shall be supplied by two physically independent circuits designed to minimize the likelihood of their simultaneous failure. Each of the two offsite circuits shall be available in sufficient time to prevent exceeding reactor design limits, given a failure of the onsite power supplies and the other offsite circuit. One of the offsite circuits shall be designed to be available within a few seconds following a LOCA to assure adequate core cooling. Provisions shall be included to minimize the probability of losing electric power form any of the remaining supplies as a result of loss of the other supplies.

The conclusion that the Millstone 2 design is in compliance with the GDC 17 relies on the NNECO interpretation that the offsite circuit design need not consider single failure of the onsite power system during accident conditions.

Conclusions

NNECO used a probablistic risk assessment to estimate the safety significance of this issue. NNECO concluded that the contribution of the current protective relay configuration to the risk of core melt was very low at less than 1X10-7. Based on the above, the inspector concluded the issue had low safety significance and that no further immediate NNECO action was warranted pending the completion of NRC management review of the issue.

The question of whether Millstone 2 meets the requirements of GDC 17 is a licensing matter. This issue was referred to NRC:NRR on May 20 for review to determine whether further action by NNECO for Millstone 2 is required. This matter is unresolved pending further review by the NRC staff (50-336/91-15-01).

6.5 **Steam Generator Tube Inspections**

Background

During plant operation at 100% full power on May 25, a primary to secondary leak developed on No. 2 steam generator (SG). The leak rate increased from approximately zero at midnight to about 50 gallons per day (GPD) at 6:00 a.m. At 1:55 p.m., with the leak rate at approximately 60-70 gpd, management ordered the plant shutdown. Millstone 2 was placed in cold shutdown on May 26 for inspection and repairs of the steam generators.

Leak Identification

NNECO activities to locate the cause of the primary-to-secondary leakage included a secondary pressure test and eddy current testing to confirm the leaking tube. On May 30, following plant cooldown and steam generator primary manway removal, the steam generators were pressurized to 506 psi for tube leakage investigations. The investigations by video camera inside the primary plenum identified two leakage locations in the No. 2 steam generator hot leg plenum. The locations were row 64 line 150 (in-service tube) and row 74 line 80 (plugged tube).

On May 31, an eddy current trace was performed on the tube at row 64 line 150. The primary-to-secondary leakage was confirmed by eddy current test data. The defect was located just below the U-bend region, in proximity to the diagonal support strap. The support strap is not physically attached to the tube but is between rows of tubes. The defect was characterized as a circumferential crack approximately 112 degrees in circumference thru-wall with an axial offset of 1/4 inch. The last recorded inspection of this tube was in 1986.

Inspection Scope, Expansion, and Results

On May 31, NNECO developed the eddy current examination scope. The inspection scope used three different probe types: a 3-coil rotating pancake coil; a standard bobbin coil; and a flexible rotating pancake coil. The scope included examination of both steam generators and included inspections in three principal areas within the steam generator.

The first area was full length tube exams on 50.1% of all available tubes in the No. 1 steam generator, and 53.2% of the tubes in the No. 2 steam generator. The full length exams covered all tubes not inspected since 1986 and a random selection of 20% in each steam generator. The random population focused on tubes in contact with partial eggcrate supports No. 8 and No. 9 and supports No. 10 and No. 11. The full length examinations were performed with a standard bobbin coil.

The second area included a partial tube examination at the U-bend to the first horizontal support. The exams originated at the hot leg plenum. The number of tubes examined with this technique were 426 in the No. 1 steam generator, and 570 in the No. 2 steam generator. The tube rows examined focused on No. 8 and No. 9 partial eggcrate support. The examination was performed with a flexible rotating pancake coil to evaluate potential cracked indications.

The final area included partial tube examinations from approximately one inch below the tube sheet to three inches above the tube sheet. The inspections focused on previously identified circumferential cracks in the tube to tube sheet transition area. The scope initially was twenty percent of all available tubes in the susceptible area, which on June 17 was expanded to 100% of the susceptible area. The examination purpose was to confirm the crack mechanism had decreased and was within predicted values for the cycle of operation. Confirmation and depth characterization of identified tube cracks were subjected to ultrasonic evaluation.

Conclusion and Assessment

The inspection scope for the steam generators exceeded the required and normal refuel frequency scope. NNECO actions to shutdown the facility prior to exceeding any primary-to-secondary limits precluded the requirement to perform a "first sample" inspection pursuant to Technical Specification 4.4.5.1.3. The developed scope exceeded the requirements of inspection scope pursuant to Technical Specification 4.4.5.1.2 and Table 4.4-5. At the end of the inspection period, examinations were ongoing. Evaluation of the results will be subject to future inspections. Engineering support of the examinations and repair was thorough.

6.6 Seismic Qualification of Diesel Gages

The inspector noted that NNECO personnel raised a question regarding the seismic installation of pressure gauges on the service water supply strainers to the emergency diesel generators (EDG) skid. The inspector toured the EDGs on May 7 with the Millstone 2 I&C engineer to review the installation and to assess the impact of a potentially nonconforming condition on EDG operability. The inspector requested NNECO to provide its assessment of the non-seismic gauges on diesel operability. The inspector requested NNECO address the issue on Millstone 1 as well.

Background

Each EDG at Millstone 2 is supplied cooling water from separate service water headers. The service water system provides cooling for the diesel engine jacket and lube oil heat exchangers. An in-line strainer is provided in each header upstream of the engine skid; either side of the duplex strainer can provide for 100% of the required EDG cooling supply. Two pressure gauges are mounted on the strainer, one on the inlet side and one on the outlet side. The gauges provide differential pressure (delta-P) readings across the strainers and must be read locally. This installation is identical on both Millstone 2 EDGs, as well as on the single Millstone 1 diesel. The associated pressure gages are labeled PI 6340A&B and 6351A&B for Millstone 2, and PI-4-66 and 4-67 for Millstone 1.

As-Built Configuration

The installation consisted of a 2.5 lb Ashcroft pressure gage and 1/4 inch isolation valve mounted on the top side of the strainer. The strainer was about 12 inches tall, and the valve and gage were 4 and 18 inches above the strainer, respectively. Even though the assembly was unsupported, the installation uses 1/4 inch stainless steel pipe, instead of tubing. The construction of the strainer mounting and base plate made the strainer rigidly attached to the floor. The inspector reviewed the gauge mounting and noted that the installation, based on inspector judgment and experience, had inherent seismic characteristics.

Inspector review also noted a similar configuration for the strainer assembly on the Millstone 1 EDG. However, the installation at Unit 1 differed in the smaller size and weight of the isolation valve and in the length of the tubing, such that both the gauge and the isolation valve were mounted closer to the top of the strainer, effectively reducing the equivalent moment arm associated with the installation. The inspector concluded the seismic evaluation for the Millstone 1 installation would be bounded by that for the Millstone 2 installation.

Safety Significance

Inspector review of diesel operating procedures and assessment of the gauge installation noted several design characteristics and functions. The MEPL evaluation states that the gauges are not safety related, as non-Q, and are not relied upon for operation of the diesel. A fouled strainer would reduce cooling flow to the diesel skid, cause jacket cooling temperatures to increase, and cause a high temperature alarm to be annunciated at the local diesel panel and in the control room. The plant equipment operator (PEO) responding to the alarm would note the high delta-P and would manually switch to the alternate strainer to provide continued flow of cooling water to the EDG.

The inspector also considered the potential consequences of a rupture of the instrument tubing and the affects on diesel operation. The inspector calculated the amount of flow that would be lost of the 3/8-inch strainer instrument opening at normal service water system conditions. The flow lost out of the postulated broken tube(s) would not divert enough flow to jeopardize diesel operation. The inspector also noted that spray from the break would not affect essential equipment in the area. Potential flooding in the engine room would be mitigated by the elevated mounting of essential equipment, the floor drain system, and an estimated 30-square-inch opening at the bottom of the east door in the engine rooms. A PEO is dispatched to the diesel room for routine scheduled starts of the EDG and in response to any automatic starts to verify proper operation of the engine within established operating limits. The operator would identify abnormal strainer leakage caused by degradation in the instrument piping, and other actions could be taken to mitigate the affects of flooding. Based on the above, the inspector concluded that a rupture in the instrument lines would not jeopardize EDG operation.

Seismic Analysis

NNECO reported its preliminary evaluation that the strainers were not provided as original equipment for the EDGs, but were added as a plant modification in 1977. NNECO review of the PDCR determined that the strainers were installed to meet seismic requirements. NNECO I&C engineering completed a walkdown of the EDG skids determined that (i) the strainer gauge installation conformed to the isometric diagram used for construction and (ii) the strainer and gage installation conformed with the Bechtel construction specification for gauges. However, NNECO could identify no reference to seismicity of the gauges in the PDCR package and no calculation demonstrating seismic qualification was on file. The matter was referred to NUSCO mechanical engineering for review. Based on a description of the installation configuration and material of construction, NUSCO completed a preliminary engineering assessment on May 9 concluding that the installation was seismic and that there was a high probability that an operability determination would result from a detailed seismic evaluation.

NNECO provided the results of a detailed seismic analysis in calculation MP2-LOE-1403 dated May 22, 1991, for Millstone 2, and a seismic qualification evaluation in a memorandum dated May 29, 1991, for Millstone 1. NNECO determined that the existing gauge installations were acceptable for pressure, deadweight, and seismic loading conditions. The inspector reviewed the Millstone 2 calculation and the Millstone 1 evaluation to verify they reflected actual plant conditions. For the Millstone 2 case, the maximum stress for worst case SSE loading was 16460 psi, which was less than the allowable stress of 28260 psi for the Type 316L stainless steel tubing. No inadequacies were identified.

Conclusions

In summary, NNECO review of existing plant conditions during routine operating activities identified a potential nonconforming condition on a safety related EDG. NNECO reviews identified that the 1977 PDCR documentation was discrepant in that the seismicity of the instrument gages on the seismically installed strainers were not specifically addressed. Present day calculations confirmed the adequacy of the installation such that EDG operability was not impacted. The I&C supervisor stated that the May 1991 calculation to qualify the gauges will be added to the I&C files, the MEPL file, and/or the NUSCO engineering calculation file for future reference in plant design work. NNECO actions were acceptable. The inspector had no further comments on this item.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (IP 90713/92702/90712)

7.1 Licensee Event Reports

Licensee event reports (LERs) were reviewed to assess accuracy, adequacy of licensee corrective actions, compliance with 10 CFR 50.73 reporting requirements, and to determine whether there were generic implications or if further information were required. The following LER was reviewed:

LER 91-003: Potential Loss of DC Switchgear Cooling During Loss of Instrument Air

NNECO documented, pursuant to 10 CFR 50.73 (a)(2)(V)(a), a design inadequacy in that a postulated loss of instrument air coincident with a safety injection actuation signal would result in loss of DC switchgear cooling. The licensee concluded the root causes for the event were a design inadequacy and inadequate guidance in the controlling abnormal operating procedure (AOP). NNECO corrective action was to incorporate a procedural change to AOP-2563, "Loss of Instrument Air," to provide instructions to maintain cooling to the DC switchgear rooms.

The inspector verified that the revised AOP-2563 adequately addressed the procedural deficiency. With respect to the design, inspector review of plant modification safety evaluation 2-294-76 found that the modification did not adequately address the system configuration on loss of instrument air and the impact on system operability. NNECO corrective actions focused on procedural guidance, yet actions under NRC Generic Letter 88-14, "Loss of Instrument Air," requested verification that design of the air system is in accordance with its intended function. As part of the design review, an analysis was required to verify that air-operated component failure positions are correct for assuring their safety function.

NNECO review of the affected chilled water valves evaluated the loss-of-air failure position, yet the evaluation did not consider the impact on system operation adequately. NNECO identified the inadequate design modification as a root cause for the event, yet no corrective actions exist to address this aspect. Further, NRC Generic Letter 88-14 requested a verification of design function in the instrument air system considering component failure modes. This item is unresolved pending further NRC review of NNECO actions in response to Generic Letter 88-14 and NNECO corrective actions to address root causes of deficiencies (50-336/91-15-02).

7.2 Periodic Reports

Upon receipt, periodic reports submitted pursuant to technical specifications were reviewed. This review verified that the reported information was valid and included the required NRC data. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following report was reviewed:

-- Monthly Operating Report, May 1991

No significant observations were noted.

7.3 Previously Identified Items

7.3.1 (Closed) Notice of Violation Item 90-18-02: No Reactor Vessel Monitoring System Operable During Reduced Inventory Operations

This item was open based on the licensee not having available an independent reactor vessel level monitoring system to assess reactor coolant system conditions and shutdown cooling performance during reduced inventory operations on September 19, 1990. This condition was prohibited by controlling operations procedure OP-2301E. On December 12, 1990, NUSCO responded to the Notice of Violation documenting the root cause of the event, corrective actions, and actions to prevent recurrence. The inspector reviewed licensee actions and their implementation and considers them appropriate.

Procedure IC 2421C was revised to provide for proper operation of the reactor vessel level monitoring system prior to system draindown. This is in agreement with prerequisites of the controlling operations procedure (OP-2301E). Operation of the reactor vessel level monitoring system in reduced inventory operations on several occasions since the issuance of the violation was found acceptable. This item is closed.

8.0 MANAGEMENT MEETINGS

Periodic meetings were held with station management to discuss inspection findings during the inspection period. A summary of preliminary findings was discussed on July 1 with Mr. J. Keenan 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.

8.1 Radiation Monitor Meetings

A meeting was held with NNECO representatives in the NRC's Region I office on June 11 to discuss programmatic features of the Millstone Station radiation monitoring systems. The meeting was chaired by Mr. E. Wenzinger of NRC's Division of Reactor Projects and was attended by members of all three Region I technical divisions. The licensee's presentation was introduced by Mr. J. Keenan, Unit 2 Director, and principals included R. Crandall of NU's RAB, the Unit 2 and 3 I&C Managers, and a GFL representative.

Attached is a handout outlining the major facts presented by NNECO. The meeting afforded for a good mutual understanding of the progress in upgrading procedures across the site, and the priority and status given radiation monitor calibration procedures. Calibration techniques were discussed, including the greater-than-required frequencies employed on Unit 2. Design problems and expected modifications were also highlighted. A major initiative was the Manual assembled over a year ago and in use for reference of technicians and operators regarding various technical aspects of all types of monitors used at the Station. Also, the ad hoc committee and assignment of a full-time RAB representative who spends approximately half of his time on site was described.

Conclusions reached from the meeting were:

-- Clear "ownership" of monitors by I&C group.
-- Several lapses of independent oversight and continuing technical assessment to monitor performance.
-- Recognized design problems (e.g., U2 blowdown monitor flooring) and modifications proposed.
-- High priority given to calibration procedure upgrades -- by the technicians who use them -- which are approximately 50% completed and targeted to be finished in December 1991.
-- Reliable performance, which is closely tracked and promptly addressed when necessary.

Table 1

Component	Frequency	Procedure
Auxiliary Feedwater Pump/Motor	Weekly, Quarterly Semi-annual, Annual	MP-2701J49 MP-2701F-P9 MP-2701J-3
Low Pressure Safety Injection Pump/Motor	Annual, Refueling	MP-2701J3 MP-2703E3
High Pressure Safety Injection Pump/Motor	Semi-annual, Annual Refueling	MP-2701J3 MP-2703E3 MP-2701F
Containment Spray Pump/Motor	Annual Refueling	MP-2701J3 MP-2703E4 MP-2701F
Emergency Diesel Generators	Bi-weekly, Quarterly Semi-annual, Annual Refueling	MP-2701J12 MP-2701J15 MP-2701J19 MP-2721A
Charging Pumps/Motors	Semi-Annual, Annual Special Frequency	MP-2701J-18 MP-2701F

ENCLOSURE

MILLSTONE NUCLEAR POWER STATION
RAD MONITORS/INFORMATION EXCHANGE

JUNE 12, 1991

INTRODUCTION AND MEETING PURPOSE	W. E. HUTCHINS SENIOR ENGINEER NUCLEAR LICENSING
CALIBRATION/FUNCTIONAL TESTING	R. A. CRANDALL SUPERVISOR RADIOLOGICAL ASSESSMENT BRANCH
RAD MONITOR REVIEW o WHY NECESSARY o SCOPE OF REVIEW o GENERAL RESULTS o ROOT CAUSES o CORRECTIVE ACTIONS	R. A. CRANDALL
RAD MONITOR PROCEDURE IMPROVEMENTS	J. D. BECKER MANAGER MP2 I&C
RAD MONITOR HARDWARE UPGRADES	J. D. BECKER
CONTINGENCY ACTIONS FOR INOPERABLE EQUIPMENT (I.E., REDUNDANT INSTRUMENTATION)	R. C. ENOCH MANAGER MP3 I&C
SUMMARY/OVERVIEW	J. S. KEENAN DIRECTOR MP2
QUESTIONS/ANSWERS	AS APPROPRIATE

CALIBRATION/FUNCTIONAL TESTING

R. A. CRANDALL

FUNCTIONAL TEST/CALIBRATION TECHNIQUE DEPENDENCIES

- TYPE OF DETECTOR
  - -- GM, IONCHAMBER, BETA SCINTILLATION, GAMMA SCINTILLATION
- PURPOSE OF DETECTOR
  - -- EFFLUENT, TREND, AREA RAD
  - -- AUTO INITIATION, SAFETY SIGNIFICANCE
- ANALYSIS MODE
  - -- GROSS, SINGLE CHANNEL
- READOUT TYPE
  - -- ANALOG, DIGITAL, LINEAR, LOG
- READOUT UNITS
  - -- CPM, μCi/CC, mR/hr
- NUCLIDES OF INTEREST
- VENDOR DESIGN/ELECTRONICS
- VENDOR RECOMMENDATIONS
- DETECTOR RANGE
- DETECTOR LOCATION
- NORMAL BACKGROUND
- PERSONAL PREFERENCES

FUNCTIONAL TEST/CALIBRATIONS

CONCLUSIONS

- o THERE IS NO STANDARD FUNCTIONAL OR CALIBRATION PROCEDURE
  - -- ALL OF THE ABOVE FACTORS NEED TO BE CONSIDERED
  - -- RESULTS IN MANY DIFFERENT PROCEDURES FOR 3 MP UNITS
- o BASICS OF FUNCTIONAL TESTS
  - -- SIGNAL IN ONE END--READOUT ON OTHER END
  - -- ALARMS OCCUR AT SETPOINT
  - -- AUTO ISOLATION FUNCTIONS
- o BASICS OF CALIBRATIONS
  - -- KNOWN SOURCE IN REPRODUCIBLE GEOMETRY--KNOWN RESPONSE
  - -- VARIOUS STRENGTHS TO TEST LINEARITY

FREQUENCY OF FUNCTIONALS/CALIBRATIONS

- BASICALL, SIMILAR TO STANDARD TSs
- TYPICAL FOR SAFETY RELATED RAD MONITORS
  - MONTHLY FUNCTIONAL
  - REFUEL OR QUARTERLY CALIBRATION
- TYPICAL FOR EFFLUENT RAD MONITORS
  - QUARTERLY FUNCTIONAL
  - REFUEL CALIBRATION
- MP1 AND MP3 FOLLOW TS FREQUENCY
- MP2 HAS BEEN PERFORMING AT FREQUENCY SHORTER THAN TS

RAD MONITOR REVIEW

R. A. CRANDALL

RAD MONITOR REVIEW

- WHY
  - -- RECURRING WEAKNESSES NOTED IN NRC AND NRB AUDITS
  - -- RECOGNIZED NEED FOR RADIOLOGICAL TECHNICAL REVIEW OF I&C PROCEDURES
- SCOPE
  - -- ALL PROCESS, EFFLUENT AND ARMs
  - -- REVIEW OF PROCEDURES FOR REGULATORY COMPLIANCE AND TECHNICAL CONTENT
  - -- RECOMMENDATIONS OF GOOD PRACTICES AND IMPROVEMENTS
  - -- DEVELOPMENT OF RAD MONITOR MANUALS
- RESULTS
  - -- NO SAFETY CONCERNS
  - -- MET INTENT OF TSs
  - -- ALTHOUGH SOME INACCURACIES WERE POSSIBLE, MONITORS WOULD FULFILL THEIR INTENT OF PROTECTING WORKERS AND PUBLIC
  - -- NUMEROUS RECOMMENDATIONS FOR IMPROVEMENTS

RAD MONITOR REVIEW (CONT)

- ROOT CAUSES
  - -- LACK OF RADIOLOGICAL TECHNICAL REVIEW OF PROCEDURES
  - -- MULTIDISCIPLINE RESPONSIBILITY
  - -- COMPLEXITY OF ISSUE
  - -- LACK OF A CLEAR BASIS FOR MONITOR REQUIREMENTS
- CORRECTIVE ACTIONS
  - -- APPRAISAL--ONE-TIME TECHNICAL REVIEW
  - -- IMPLEMENTATION OF RECOMMENDED IMPROVEMENTS
  - -- ADMINISTRATIVE CONTROL PROCEDURE REQUIRING RADIOLOGICAL ASSESSMENT BRANCH TECHNICAL REVIEW OF ALL NEW/REVISED PROCEDURES
  - -- ISSUANCE OF RAD MONITOR MANUAL
  - -- ASSIGNMENT OF "RESPONSIBLE" INDIVIDUAL
  - -- FORMATION OF RAD MONITOR SUBCOMMITTEES

RAD MONITOR REVIEW (CONT)

- **CORRECTIVE ACTIONS** (CONT)

  RESPONSIBILITIES OF RAD MONITOR INDIVIDUAL

  -- CHAIR A RAD MONITOR COMMITTEE FOR EACH UNIT; COMMITTEE RESPONSIBILITIES WILL INCLUDE:

    - ACTIONS ON INOPERABLE MONITORS--REVIEW THE ADEQUACY AND THE PRIORITY THE LINE ORGANIZATION IS GIVING TO REPAIR OF MONITOR, ENSURE ADEQUATE AUGMENTED SAMPLING IS BEING ACCOMPLISHED, REVIEW PIRs/LERs

    - INVESTIGATE MONITOR PROBLEMS--EXAMPLES FROM THE PAST INCLUDE MP1 OFF-GAS MONITOR RESPONSE CHANGES, MP2 SJAE SAMPLE FLOW,MP3 CONTROL ROOM INTAKE MONITOR SPIKES, LIQUID WASTE MONITOR HIGH BACKGROUND AT ALL THREE UNITS; THE COMMITTEE WILL ENSURE THE RIGHT GROUPS ARE INVESTIGATING THE PROBLEM WITH THE APPROPRIATE PRIORITY

    - EVALUATE MONITOR OR PROCEDURE IMPROVEMENTS--EXAMPLES INCLUDE ELIMINATING MP3 MOVING FILTERS OR CHANGING MP2 CALIBRATION TECHNIQUE FROM GASEOUS SOURCE TO BUTTON SOURCES; THE COMMITTEE WILL DISCUSS THE FEASIBILITY FOR SUCH IMPROVEMENTS AND ASSIGN APPROPRIATE FOLLOW-UP ACTION

    - SUPPORT NEW MONITOR PROJECTS--FOR EXAMPLE, THE NEW MP2 SJAE MONITOR; THE COMMITTEE WILL ENSURE THE RIGHT GROUPS ARE INVOLVED IN MONITOR DESIGN, SELECTION, INSTALLATION, AND TESTING

RAD MONITOR REVIEW (CONT)

- CORRECTIVE ACTIONS (CONT)

  RESPONSIBILITIES OF RAD MONITOR INDIVIDUAL (CONT)

  -- REVIEW AND SIGN OFF ON RAD MONITOR PROCEDURE CHANGES

  -- PERFORM MONTHLY CHECKS OF "NORMAL" READING vs. SET POINTS AND MONITOR READINGS vs. GRAB SAMPLE RESULTS

  -- INITIATE RAD MONITOR MANUAL CHANGES

  -- ASSIST I&C/CHEMISTRY DURING SELECT MONITOR CALIBRATIONS

  -- INITIATE OR RECOMMEND SET POINT CHANGES

  -- PROVIDE TECHNICAL CONTACT FOR AUDITORS FROM NRC, INPO, ANI, AND NRB AND ASSIST IN PREPARATION OF AUDIT RESPONSE

RAD MONITOR PROCEDURE IMPROVEMENTS

J. D. BECKER

RAD MONITOR PROCEDURE IMPROVEMENTS

- o PROCEDURES ARE BEING UPGRADED TO ADDRESS SEVERAL ISSUES:
  - -- FORMAT
  - -- LEVEL OF DETAIL
  - -- TECHNICAL METHOD AND CONTENT
- o RADIOLOGICAL ASSESSMENT BRANCH ASSESSMENT OF RAD MONITOR PROCEDURES PROVIDED THE MAJORITY OF TECHNICAL IMPROVEMENT COMMENTS:
  - -- PHASE I (SAMPLED 80% OF THE TYPES, 20% OF THE TOTAL NUMBER)
  - -- PHASE II (REMAINING 100%)
  - -- REPORTS PROVIDED DETAILED COMMENTS; COMMENTS RANKED BY THE FOLLOWING PRIORITIES:
    1. POSSIBLE REGULATORY COMPLIANCE ISSUE
    2. SIGNIFICANT TECHNICAL IMPROVEMENT SUGGESTED
    3. GENERAL GOOD PRACTICE IMPROVEMENT
  - -- RAD MONITOR MANUAL COMPLETED EARLY 1990--GENERAL INFORMATION CONCERNING FUNCTION, SPECIFICATIONS, SET POINTS, AND CALIBRATION METHODS
- o INCREASED RADIOLOGICAL ASSESSMENT BRANCH INVOLVEMENT IN RAD MONITOR PROCEDURE PROCESS. ADMINISTRATIVE CONTROL PROCEDURES NOW REQUIRE RADIOLOGICAL ASSESSMENT BRANCH SIGN-OFF ON TECHNICAL RAD MONITOR PROCEDURE REVISIONS

RAD MONITOR PROCEDURE IMPROVEMENTS (CONT)

- MP1
  - -- ALL PRIORITY 1 COMMENTS ADDRESSED

    80% OF PRIORITY 2 AND 3

- MP2
  - -- INCORPORATION OF RADIOLOGICAL ASSESSMENT BRANCH COMMENTS INTO PROCEDURES
    - PRIORITY 1 COMMENTS IMMEDIATELY RESOLVED BY PROCEDURE CHANGES
    - PRIORITY 2 AND 3 COMMENTS BEING RESOLVED DURING THE PROCEDURE UPGRADE PROGRAM
    - 38 RAD MONITOR PROCEDURES TO UPGRADE OR WRITE:

      13 UPGRADE COMPLETE
      11 WORKING
      14 NOT STARTED
      38 TOTAL

    - A COMMENT REMAINS OPEN UNTIL RADIOLOGICAL ASSESSMENT BRANCH INDICATES CONCURRENCE WITH THE RESOLUTION; E.G., MP2 I&C MAINTAINS A PUNCHLIST THAT TRACKS ALL MP2 OPEN ITEMS

- MP3

  SPECIFIC RADIOLOGICAL ASSESSMENT BRANCH COMMENTS HAVE BEEN INCORPORATED

RAD MONITOR PROCEDURE IMPROVEMENTS (CONT)

FUTURE ACTIONS

- o CONTINUED ENHANCEMENTS OF THE PROCEDURE PROCESS
  - -- STANDARDIZED OUTLINES BEING DEVELOPED FOR SIMILAR MP2 RAD MONITOR CALIBRATIONS AND FUNCTIONAL TESTS
  - -- ADDITIONAL PROCEDURE WRITING TRAINING BEING CONDUCTED FOR ALL MP2 I&C PERSONNEL
  - -- ENHANCEMENT OF THE DEPARTMENT INSTRUCTION GOVERNING PROCEDURES IS IN PROGRESS
  - -- ADDING ADDITIONAL RESOURCES TO I&C DEPARTMENT TO INCREASE THE UPGRADE PROGRAM'S PRODUCTIVITY
- o CONTINUED TRAINING ON RAD MONITOR TOPICS SUCH AS DETECTOR GEOMETRY

RAD MONITOR HARDWARE UPGRADES

J. D. BECKER

RAD MONITOR HARDWARE UPGRADES

- MP1

  -- UPGRADING PIPING DESIGN TO SJAE RAD MONITOR TO IMPROVE SAMPLING

  -- RELOCATION OF LIQUID WASTE RAD MONITOR TO A LOWER BACKGROUND AREA

- MP2

  -- REPLACING THE FLOW INDICATING SWITCHES TO IMPROVE RELIABILITY (MAGNAHELIC TO PHOTOHELIC)

  -- SJAE MONITOR BEING REPLACED WITH NEW DESIGN USING ADJACENT-TO-LINE METHOD

  -- DUE TO DIFFICULTY IN OBTAINING SPARE PARTS FOR THE EXISTING ORIGINAL EQUIPMENT AND A CONTINUED DESIRE TO IMPROVE RELIABILITY, THE PLAN IS TO CONTINUE INDIVIDUAL RAD MONITOR UPGRADES:

    - SJAE IN PROCESS

    - CONTAINMENT GAS AND PARTICULATE

    - WASTE GAS

    - SPENT FUEL POOL AREA

RAD MONITOR HARDWARE UPGRADES (CONT)

- MP3
  - -- CONTINUED EFFORTS TO RESOLVE INADVERTENT ACTUATIONS CAUSED BY THE CONTROL ROOM INLET VENTILATION MONITOR
    - NOISE SPIKES
    - NOISE FILTER SOFTWARE IMPROVEMENTS
    - INCREASED DETECTOR REPLACEMENT FREQUENCY
    - CONSIDERING ADDITIONAL DESIGN ENHANCEMENTS
  - -- EVALUATING CHANGING "MOVING FILTER" PARTICULATE RAD MONITOR DESIGN TO A FIXED FILTER DESIGN
- MP2 HISTORY
  - -- REPLACED CLEAN AND AERATED LIQUID EFFLUENT MONITORS WITH MORE RELIABLE DIGITAL PIOPS UNITS AND RELOCATED TO SOLVE HIGH BACKGROUND AND SHIELDING PROBLEMS
  - -- RELOCATED LOCAL ALARM HORNS TO SOLVE EMI PROBLEMS
  - -- INSTALLED LOCAL BYPASS KEYS TO ALLOW THE ALARM HORN TO BE SILENCED WHEN THE MONITOR IS NOT IN USE AND THE SIGNAL FAIL IS IN
  - -- INSTALLED NEW SHIELDING AND DETECTORS ON RBCCW AND SGBD MONITORS TO INCREASE SENSITIVITY

RAD MONITOR HARDWARE UPGRADES (CONT)

- o MP2 HISTORY (CONT)
  - -- INSTALLED SJAE/SGBD BYPASS SWITCH TO ALLOW MAINTENANCE ON EITHER MONITOR WHILE MAINTAINING THE PROCESS IN SERVICE WITH ISOLATION CAPABILITY FROM THE OPERABLE RAD MONITOR
  - -- INSTALLED SPRING RETURN TEST SWITCHES ON QA MONITORS TO PREVENT INADVERTENTLY LEAVING THE MONITOR IN TEST MODE; THIS DESIGN IMPROVEMENT WAS AN LER ACTION TO PREVENT RECURRENCE
  - -- INSTALLED HIGHER CAPACITY PUMPS ON CONTAINMENT PARTICULATE AND GAS MONITORS TO INCREASE SAMPLING FLOW RATE AND IMPROVE SAMPLING EFFICIENCY
  - -- CONNECTED ALL MONITORS TO THE PLANT COMPUTER FOR BETTER DISPLAY AND TREND ANALYSIS CAPABILITIES
  - -- INSTALLED NEW CONTAINMENT HIGH RANGE, CONTROL ROOM VENTILATION, MAIN STEAM LINE, STACK HIGH RANGE
  - -- INSTALLED N16 MONITORS TO MONITOR LOW STEAM GENERATOR LEAK RATES

CONTINGENCY ACTIONS/
INOPERABLE EQUIPMENT

R. C. ENOCH

CONTINGENCY ACTIONS/INOPERABLE EQUIPMENT

- o ALARMS
  - -- OPERATING PROCEDURES (HIGH RAD ALARM RESPONSE)
- o TROUBLE/EQUIPMENT FAILURE
  - -- IF TS:
    - DECLARE INOPERABLE
    - LOG INTO LCO
    - TAKE REQUIRED ACTION
  - -- CONTACT:
    - I&C
    - HEALTH PHYSICS (AREA MONITORING AND SAMPLING)
    - CHEMISTRY (SAMPLING MEDIUM)
  - -- REVIEW
    - PROCEDURE FOR ALARM RESPONSE

CONTINGENCY ACTIONS/INOPERABLE EQUIPMENT (CONT)

DEPARTMENT ACTIONS

- o I&C
  - -- TROUBLESHOOT AND REPAIR
  - -- RETEST
- o HEALTH PHYSICS
  - -- AREA SURVEYS
- o CHEMISTRY
  - -- SET UP CONTINUOUS SAMPLE RIGS
  - -- PERFORM GRAB SAMPLES
  - -- INDEPENDENTLY VERIFY RELEASE RATE CALCULATION
- o OPERATIONS
  - -- INDEPENDENT VERIFICATION OF VALVE LINEUP

CONTINGENCY ACTIONS/INOPERABLE EQUIPMENT (CONT)

- o SAFETY GRADE MONITORS
  - -- HAVE BACKUPS EXAMPLES INCLUDE:
    - CONTAINMENT HIGH-RANGE AREA
    - FUEL DROP MONITOR
    - SERVICE WATER EFFLUENT
    - CONTROL BUILDING
    - GASEOUS EFFLUENT HIGH RANGE HAVE A NORMAL RANGE
- o LIQUID MONITORS
  - -- TYPICAL ACTIONS:
    - TWO INDEPENDENT SAMPLES
    - INDEPENDENT VERIFICATION OF:
      - + RELEASE RATE CALCULATION
      - + VALVE LINEUP
    - GRAB SAMPLES EVERY 12 TO 24 HOURS DEPENDING ON SECONDARY COOLANT ACTIVITY

CONTINGENCY ACTIONS/INOPERABLE EQUIPMENT (CONT)

- GASEOUS/PARTICULATE MONITORS
  - -- IODINE/PARTICULATE
    - CONTINUOUS SAMPLES WITH AUXILIARY SAMPLING EQUIPMENT
  - -- GASEOUS
    - GRAB SAMPLES TAKEN EVERY 12 HOURS
- SPENT FUEL POOL AREA MONITORS
  - -- SURVEY EVERY 24 HOURS (MP2)
  - -- PORTABLE CONTINUOUS (MP3)

SUMMARY/OVERVIEW

J. S. KEENAN

SUMMARY/OVERVIEW

- o EXPENDITURE OF SIGNIFICANT RESOURCES TO DATE AT MP SITE TO ENSURE RAD MONITORS ARE OPERABLE
  - -- RAD MONITOR REVIEW
  - -- IMPLEMENTATION OF IMMEDIATE/CORRECTIVE ACTIONS
  - -- TECHNICAL PROCEDURE REVIEW AND UPGRADE PLANS
- o HEIGHTEN SENSITIVITY OF MANAGEMENT TO FUNCTIONAL AND OPERATIONAL RAD MONITORS
- o SOME GOOD PRACTICES HAVE BEEN IMPLEMENTED WHICH EXCEED CURRENT INDUSTRY STANDARDS
- o CURRENT PROGRAM ESTABLISHED IS BOTH CONTINUOUS AND EVOLVING