U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee:	Northeast Nuclear Energy Company P. O. Box 270 Hartford, CT 06141-0270
Facility Name:	Millstone Nuclear Power Station, Units 1, 2, & 3
Inspection At:	Waterford, CT
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Inspection Summary: Combined Report Nos. 50-245/90-25, 50-336/90-28, and 50-423/90-27

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

Results: See Executive Summary

EXECUTIVE SUMMARY

MILLSTONE NUCLEAR POWER STATION
UNITS NO. 1, 2, AND 3
NRC REGION I INSPECTION REPORT NOS.
50-245/90-25, 50-336/90-28, AND 50-423/90-27

Plant Operations

Unit 1

On December 4, 1990, the "A" LPCI pump failed while in service for torus cooling. Operators responded well to assess and mitigate the event, and to comply with the technical specification requirements.

Unit 2

Overall plant control during the steam generator manway repair outage was implemented in an acceptable manner.

General plant housekeeping tours of the facility identified numerous nonessential equipment adrift, inadequate lighting in some areas, and examples of poor equipment preservation. NNECO actions were noted to improve the condition of the unit at the end of the inspection period.

Unit 3

On December 31, 1990, two moisture separator reheater discharge drain pipes ruptured because of erosion/corrosion. Operators responded well to the initial event and the subsequent transient caused by the loss of containment instrument air.

Two non-cited licensee-identified violations occurred involving (1) the failure of operations personnel to record ventilation sample flow data associated with a temporary sample log every four hours, for a period of eleven hours, when turbine building vent radiation monitor 3HVR*RE10B was declared inoperable (50-423/90-27-01) and (2) the inadequate restoration of monitor 3HVR*RE10B when the monitor was declared operable without transferring the filter element from the temporary monitoring skid (50-423/90-27-02).

Untimely corrective action (development of a radiation monitor restoration procedure), which was specified in Licensee Event Report (LER) 89-27, resulted in a similar event, LER 90-28, when a monitor was not restored to service. The restoration procedure, OP 3265.2, "Radiation Monitor Restoration," has since been developed.

NNECO troubleshooting and corrective actions taken in response to an overspeed trip of the steam driven auxiliary feedwater pump were determined to be good.

Radiological Protection

Unit 1

One non-cited station violation (50-245/90-25-01, 50-336/90-28-01, 50-423/90-27-03) was noted concerning the shipment of by-product material to the Duane Arnold nuclear power station that exceeded a licensed limit.

Unit 2

No significant findings were noted during this inspection period.

Unit 3

Good health physics response to a leak from a reactor coolant pump seal injection filter was noted.

Emergency Preparedness

Unit 2

During the inspection period, a partial participation emergency preparedness exercise demonstrated licensee capabilities to provide adequate protective measures for public health and safety.

Maintenance/Surveillance

Unit 1, 2, and 3

NNECO procedures to assure diesel generator fuel oil quality were found acceptable.

Unit 2

One unresolved item (50-336/90-28-02) was identified concerning maintenance procedure controls of steam generator manway gasket material, and engineering oversight of repeating purchase requisitions for gaskets used in the reactor coolant system.

Unit 3

Failure of a reactor coolant pump seal injection filter cover 0-ring resulted in a decrease of normal reactor coolant pump seal water injection, a spill of reactor coolant, and contamination of licensee personnel. Health physics and operations response to the event was good.

Engineering and Technical Support

Unit 1

NRC inspection noted good technical support to plant operations by site engineering completion of a quality PDCR to install a replacement LPCI pump.

Unit 2

Evaluation of the degraded "C" service water pump was aggressive and comprehensive. Final NRC approval of the relief request from ASME Section IV pursuant to 10 CFR 50.55 (g)(6)(i) was still pending at the end of the inspection period.

Unit 3

NNECO has elected to install positive indication of the cold overpressure protection system status. Temporary Instruction 2500/19, low temperature overpressure protection, is closed.

NNECO has established administrative controls for containment access and personnel control that provide adequate short term compensatory measures to address a containment voice page/alarm inadequacy.

Safety Assessment/Quality Verification

Unit 1

A safety-conscious orientation and uninhibited exchange of technical views were exhibited during a safety system functional inspection team meeting.

Unit 2

Ineffective communications and coordination in authorization and implementation of charging system corrective maintenance activities resulted in isolation of all charging flow (LER 90-21). Corrective actions to prevent recurrence were adequate.

A number of NNECO employee concerns presented to the NRC were referred to the licensee for review and disposition.

Unit 3

Corrective actions contained in LER 90-19-01 to address control rod cable corrosion concerns were determined to be adequate.

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DETAILS

1.0 PERSONS CONTACTED

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.

2.0 SUMMARY OF FACILITY ACTIVITIES

2.1 MILLSTONE 1 ACTIVITIES

At the start of the inspection period Millstone Nuclear Power Station Unit 1 (Millstone 1) was operating at 100% of rated thermal power. Except for power reductions for routine main steam valve testing and to support maintenance activities, Millstone 1 remained at full power throughout the inspection period.

On December 4, 1990, while operating in the torus cooling mode, the "A" low pressure coolant injection pump failed, requiring entry into a 30 day technical specification action statement. A replacement motor was installed and tested, and the action statement satisfied, prior to expiration of the time limit. Details of NNECO activities regarding this occurrence are in section 3.1.1 of this inspection report.

A detailed chronology of Millstone 1 events occurring during the inspection period is included in Attachment I.

2.2 MILLSTONE 2 ACTIVITIES

Millstone 2 began the inspection period at 75% of rated thermal power, and was in power ascension testing after startup from the recent cycle 11 refueling outage. On November 21, Millstone 2 achieved full rated thermal power.

On December 29, NNECO commenced a downpower to facilitate a containment entry to identify a calculated increase in reactor coolant system leakage. The containment entry identified leakage from all four steam generator primary manways. The facility reached cold shutdown on December 30 in preparation to evaluate and replace all primary gaskets for the steam generators.

At the end of the inspection period, the plant was in a hot shutdown condition making preparation for reactor criticality.

On December 5, 1990, NNECO conducted a partial participation emergency preparedness exercise at Millstone 2. The conduct of the exercise was evaluated by an NRC review team, which included two resident inspectors. NNECO performance provided adequate protective measures for public health and safety.

2.3 MILLSTONE 3 ACTIVITIES

Millstone 3 began the report period at 100% of rated thermal power. Plant power remained essentially constant until December 14 when, due to boron depletion and fuel burnup, plant power reduction commenced. On December 31, with the plant at approximately 87% of rated power, two six-inch lines on the discharge of the moisture separator reheater drain pumps ruptured. In response to the event, operators manually tripped the reactor and isolated steam to the turbine building by closing the main steam isolation valves. Plant recovery from the event was delayed pending recovery of turbine building loads centers 32-A and 32-P, which supply power to turbine building equipment, and inverter six in the turbine building, which powers the plant process computer and also powers the containment instrument air isolation valve. While the turbine building equipment was being restored to service, the plant was maintained in mode 3 with plant temperature being controlled through use of the atmospheric dump valves. Makeup water to the steam generators was supplied from the condensate storage tank via the auxiliary feedwater pumps.

On January 8, after completing a review of the event, and with turbine building equipment restored to service, a plant startup was commenced.

2.4 NRC ACTIVITIES

On December 3 and 4, 1990, the NRC Region I Regional Administrator visited the site to tour all three Millstone units and to meet with licensee management. The inspection findings during the tours are discussed elsewhere in this report. The meetings with unit and site management were beneficial for the review of site issues and licensee activities.

On January 3, an eight person augmented inspection team (AIT) arrived on site to investigate the cause and effect of the December 31, 1990, moisture separator reheater steam drain line ruptures in the turbine building. The team consisted of Region I Division of Reactor Safety materials and fluid system specialists, headquarters personnel from the human performance section of the Office for Analysis and Evaluation of Operational Data and the licensing and materials branches of the Office of Nuclear Reactor Regulation and the senior resident inspector from the Connecticut Yankee Haddam Neck Plant. On January 7, the team conducted an exit meeting with NNECO management. The section of pipe which failed was inadvertently excluded from NNECO's erosion/corrosion program. Operator response to the transient was good. Final inspection results will be contained in IR 50-423/91-80.

Routine review of plant operations was conducted during periods of backshifts (evening shifts) and deep backshifts (weekends, holidays, and midnight shifts). Inspection coverage was provided for 33, 8, and 36 hours during backshifts and 7, 2, and 18 hours during deep backshifts for Millstone 1, 2 and 3, respectively.

3.0 PLANT OPERATIONS

3.1 CONTROL ROOM OBSERVATIONS

Control room instruments were observed for correlation between channels, proper functioning, and conformance with technical specifications. Using indicators at the main control board, reactor, electrical, and safety system lineups were verified to be aligned properly. Alarm conditions in effect and alarms received in the control room were discussed with operators. The inspector periodically reviewed the night order log, tagout log, plant incident report log, key log, and bypass jumper log. Each of the respective logs was discussed with operation department staff.

NNECO activities in this area were satisfactory.

3.2 PLANT TOURS

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

Unit 1

Control Room	Reactor Building
Main Battery Rooms	Cable Vault
Diesel Generator Room	Intake Structure
Turbine Building	

Unit 2

Control Room	Auxiliary Building
Main Battery Rooms	Cable Vault
Diesel Generator Room	Intake Structure
Turbine Building	

Unit 3

Control Room
Engineered Safety Features Building
Spent Fuel Pool Building
Emergency Diesel Generator Building
Intake Structure
Auxiliary Building

During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication and equipment status.

NNECO activities in this area were satisfactory.

3.2.1 MILLSTONE 1 TOURS

During a facility tour with the NRC Region I Regional Administrator on December 3, a number of items were noted and referred to NNECO for review and followup. The items and NNECO's actions were as described below.

(1) Two fire extinguishers were observed in the reactor building that were sitting on the floor and were not mounted on their associated wall bracket. Additionally, the wall bracket for fire extinguisher #157 was damaged.

(2) The inspector questioned the status of the drywell hydrogen and oxygen analyzers because of apparent anomalous flow indications on the panel at the reactor building 82-foot 9-inch elevation.

(3) During a walkdown of the standby liquid control system, the inspector noted less than full thread engagement on the two packing retaining nuts for valve 1-SL-27. A similar condition was noted on valve 1-SL-28. The valves are normally closed isolation valves for a 1-inch drain line on the SLC header just upstream of the Squibb valves. While the condition did not affect valve function, it did demonstrate a lack of attention to detail in the completion of a routine activity.

In regard to item 1, NNECO stated that the normal fire extinguishers had been sent out for hydrostatic testing and were due back on December 6. Those observed by the inspector were temporary replacements. Trouble report #03M1121054 was submitted to repair the mounting bracket for fire extinguisher #157. In response to item 2, NNECO operators and technicians checked the hydrogen and oxygen analyzer and found the system to be functioning appropriately with normal flow indications. In response to item 3, trouble reports #03M1134918 and #03M1135332 were initiated to address the packing nuts on the SLC valves.

The inspector noted that, in general, NNECO identifies and corrects plant discrepancies via the trouble report system, and the above findings appear to be isolated problems. The effectiveness of the trouble reporting program will be reviewed during subsequent routine inspections.

During a walkdown of the standby liquid control system, the inspector noted that an operator aid affixed to valve 1-SL-32, which is one of two isolation valves on a 1-inch test line for the SLC header. The aid was a pipe fitting that would allow connection of a fire system hose to the 1-inch drain line and thereby provide an alternate injection path to the reactor as part of the contingency plan per emergency operating procedure (EOP) 590.7. The fitting appeared to the inspector to be too small to fit onto the 1-inch pipe nipple after the cap was removed. This concern was discussed with the duty shift supervisor on December 5. The supervisor went to the valve and demonstrated to the inspector's satisfaction that the coupling did fit onto the drain line. The inspector noted that the staged equipment was properly labeled and controlled to assure availability as part of EOP 590.7. No discrepancies were identified.

During tours of the Millstone 1 control room, the inspector noted shift activities were conducted in a quiet, professional and orderly manner. However, the inspector noted that the numbers of personnel in the control room for the conduct of routine operations and maintenance activities was somewhat high in comparison with other facilities. This observation was discussed with licensee management, who acknowledged the comment and stated that an initiative was in progress to relocate certain functions requiring operations to interface with other plant departments to adjoining offices within the control room proper. This action will require some modifications to the control room entrance points to facilitate access to the operations support office without having to traverse the main control room operating area. The inspector noted that this action should reduce the level of personnel in the control room for routine activities, and thereby reduce the potential for operator distractions.

3.2.2 HOUSEKEEPING TOUR OF MILLSTONE 2

On December 3, the resident inspector was accompanied by the NRC, Region I Regional Administrator in a tour of Millstone 2. The areas toured were the auxiliary building, emergency diesel generator rooms, enclosure building, and the control room. The tour identified numerous items of non-essential equipment adrift, inadequate lighting in selected areas, and poor equipment preservation.

The inspector presented the list of items to the operations engineer. Millstone 2 management response was to set up a team of maintenance personnel for a general clean-up of the identified areas. The inspector witnessed and reviewed the actions of the clean-up team. All items identified were properly dispositioned. The condition of the facility had improved at the end of the inspection period.

3.3 ONSITE FOLLOWUP OF OPERATIONAL EVENTS

3.3.1 LPCI PUMP FAILURE - MILLSTONE 1

During plant operation at 100% of rated full power on December 4, the operators completed a routine test of the low pressure core cooling system. The "A" LPCI pump was started as part of this test, and was left in service for torus cooling after the taking of test data per SP 622.7. At 2:13 pm, after about 45 minutes of operation on torus cooling, a motor overload alarm was received on control room panel (CRP) 9-3. The shift supervisor dispatched an operator to the 4KV switchgear, who reported that the "A" LPCI pump had high motor current. This report was followed immediately by a report from the reactor building that smoke was observed coming from the southwest corner room. The shift supervisor ordered that the "A" LPCI pump be shut down and that the fire brigade respond to the area. The pump breaker was racked down. The reactor building was evacuated of nonessential personnel at 2:16 pm. The NRC resident inspector responded to the control room and the southwest corner room to assess conditions.

NNECO responders found smoke but no fire in the corner room. A leak from the pump seal, estimated to be about 5 gallons per minute, was stopped by isolating the pump suction from the torus at 3:00 pm. The leakage was well within the 50 gpm capacity of the room sump and no flooding occurred. A fan was set up to evacuate smoke from the area. Initial assessments noted no apparent damage to the "A" LPCI pump or motor other than a potential seal failure, and no damage to other components in the southwest corner room. Since all other components in the low pressure core cooling system were operable, plant operators entered a 30-day action statement per Technical Specification 3.5.2, which permits plant operation at full power with one LPCI pump inoperable. Plant procedures were reviewed and a determination made that no offsite notifications were required.

The inspector interviewed shift operations personnel and reviewed alarm response and surveillance procedures. The inspector also reviewed motor protection features as shown on circuit wiring diagram 25202-3100, Sheet 761. The normal operating current for the LPCI pump is about 50 amps with flow throttled to 4000 gpm for torus cooling. The maximum normal running current for the motor at 5200 gpm full flow is about 60 amps. The pump motor is protected by instantaneous (device 50) and time delayed (device 51) overcurrent relays. Stator winding currents above 62 amps cause an alarm in the main control room (LPCI system overload or trip). An automatic trip of the motor breaker will occur for sustained currents in excess of 120 amps, or instantaneous currents in excess of 800 amps. The maximum current noted on the motor prior to manual tripping by plant operators was about 100 amps. Plant procedures require the operator to check for failed flow control valves in response to an overload alarm, and to switch to alternate pumps if the pump has tripped. The inspector noted plant operators followed the plant procedures. Based on the above, and subsequent licensee findings that the motor stator windings were damaged by melt-down of the rotor bars, the inspector concluded that the response to the failure by plant operators and motor protection circuits were appropriate.

NNECO's onsite investigations of the "A" LPCI pump included visual inspections of the pump and motor, and electrical check of the motor and cables. The pump was found to turn freely after it was disconnected from the motor. The motor windings were checked using procedure PT 1405 on December 5, 1990. The windings passed the Baker AC surge test in which 4000 AC volts was applied, which provided a gross check of insulation of the motor windings. The windings initially passed a "Hypot" test with up to 6000 volts DC applied, but test personnel noted evidence of insulation break down as the voltage was increased to 7000 vdc. Winding insulation is considered successful under "Hypot" testing if 9200 vdc can be applied. Based on these results, NNECO concluded an electrical fault had occurred in the pump motor. The "A" LPCI motor was shipped to an offsite vendor (General Electric - GE) facility for further inspection and evaluation.

The "A" LPCI pump uses a GE 500 HP motor, Model 5K6329XC3A, that operates at 3750 rpm. Upon disassembly at the vendor's facility, the motor was found to have extensive damage to the rotor and minimal damage to the stator. The squirrel cage type induction motor is constructed of cast-aluminum rotor bars mounted inside stacked laminated discs of silicone embedded steel that provides electrical insulation. Of the 40 bars in the rotor, 8 were found melted down over the length of the rotor. Some of the melted aluminum had splattered onto the stator windings, which caused the damage noted during the onsite testing. The vendor root cause assessment was that bar failure resulted from a manufacturing defect in which voids are left in the bars during the casting process. The vendor reportedly considers the existence of voids to be a random defect in the manufacturing process. There were reportedly two other failures noted by the vendor on similar motors in service.

NNECO's inservice test (IST) program for the LPCI pump includes evaluation of pump performance based on periodic measurement of mechanical vibration. The ISI test results for the "A" LPCI pump showed vibration levels that were gradually increasing with time, but were below the alert level and thus acceptable.

A new test method that was not formally included in the licensee's program involves the analysis of the frequency spectrum of the motor current. Frequency peaks in the spectrum can be correlated with motor operating characteristics, including winding damage and rotor degradation. A predominant peak occurs at 60 hertz, the operating frequency of the 4KV motors. Degraded rotor bars would be indicated by the relative height of frequency lines (two side band peaks) at two times the running speed of the motor. In a normal rotor, the side band peaks have a 60 db fall off from the 60 hertz peak. An increase in the side band peak height by about 20 db or greater relative to the main peak indicates that potential rotor damage exists.

This technique was used by NNECO to evaluate the other safety related 4KV pump motors, including those on the four emergency service water pumps, the four service water pumps, the two core spray pumps and the other three LPCI pumps. The inspector reviewed the test results for the LPCI and core spray pump motors, and observed the testing on the remaining 4KV motors. NNECO's acceptance criterion was that rotor damage was not present if at least a 60 db fall-off was observed on the dual side band peaks. The inspector noted this acceptance criterion was met for the safety related pumps, and thus no rotor damage was evident. The testing demonstrated that other safety related motors did not appear to have a pending failure of the type experienced by the "A" LPCI pump.

NNECO was able to obtain a replacement motor from the Pooled Inventory Maintenance System. The new, fully qualified motor, manufactured by Reliance Electric Company, has the same electrical characteristics as the failed motor, except that it is rated at 600 horsepower (HP). Design considerations regarding the suitability of this pump motor as a replacement were documented by the licensee in plant design change record (PDCR) 1-77-90, which is discussed in section 8.3 of this inspection report.

Prior to installation of the replacement motor, NNECO performed DC Hypot, Baker surge, 500 vdc insulation resistance, and phase resistance tests on the motor pursuant to procedure PT-1405, Testing of 4.16 KV and 6.9 KV Motors and Surge Capacitors, revision 4, dated May 1, 1990. The inspector reviewed the satisfactory test data and had no questions regarding this activity.

While the replacement motor was being prepared for installation, NNECO performed a complete overhaul of the "A" LPCI pump in accordance with maintenance procedure MP-726.7, Bingham, Vertical Single Stage Centrifugal Pumps. The pump was disassembled and cleaned, clearances were checked, and the shaft and impeller were balanced and aligned using a new coupling and coupling spacer.

Since the new motor was not directly compatible with the pump, NNECO manufactured and installed a certified steel transition plate in order to match the motor and pump mounting surfaces. The plate also compensated for the difference in shaft length between the pump and the original and new motors. The inspector witnessed the rigging, landing, and alignment of the motor into the pump/transition plate. The evolution was performed professionally and was adequately supervised. Housekeeping conditions and radiological controls were good. NNECO management was actively involved in the maintenance activity.

The inspector verified through review of test data that the appropriate post-maintenance tests were performed and that the test results supported the NNECO conclusion that the pump met technical specification requirements. The procedures reviewed were:

-- SP 622.7, LPCI System Operability Test, revision 16, dated July 3, 1990

-- SP 622.10, LPCI System Narrow Range Flow Verification , revision 0, dated February 16, 1989

-- SP 1060, ISI Program Pump Vibration and Hydraulic Test, revision 9, dated June 6, 1990

NNECO declared the "A" LPCI pump operable and exited the technical specification limiting condition for operation at 4:00 pm on December 13, 1990.

In summary, the inspector concluded that NNECO operators responded to the event appropriately and that technical specification and NRC reporting requirements were met. Appropriate management involvement in all phases of the event was evident. Communication between unit and corporate engineering staffs was excellent. Maintenance, testing and quality assurance activities were well-planned and coordinated.

3.4 REVIEW OF PLANT INCIDENT REPORTS

Millstone Units 1 and 3 plant incident reports (PIRs) were reviewed during the inspection period to (i) determine the significance of the events; (ii) review the NNECO evaluation of the events; (iii) verify that NNECO response and corrective actions were proper; and (iv) verify that the NNECO reported the events in accordance with the applicable requirements, if required.

PIRs 1-90-99, 1-90-100, 3-90-160, 3-90-176, 3-90-174, and 3-90-183 warranted inspector followup and are discussed in the inspection report sections that follow or in other sections of this report.

3.4.1 TURBINE DRIVEN AUXILIARY FEED PUMP TRIP - MILLSTONE 3

Plant incident report (PIR) 390-160 documented an October 1990 event in which the Turbine driven auxiliary feedwater (TDAFW) pump was declared inoperable when it reportedly tripped, due to overspeed during testing. NNECO investigation of the event revealed that prior to turbine start, the turbine shaft had been rotating due to steam leakage past the steam admission valves MSS*AOV-31A, B and D. It was concluded that the initial shaft rotation, prior to turbine start, was sufficient to admit oil to the Woodward Governor speed setting piston. The combination of the initial rotational velocity and the position of the speed setting piston was sufficient to bring the TDAFW turbine above the electrical overspeed setpoint of 4752 rpm following a start demand. Following consultation with the equipment vendors, the licensee blew down the steam that had collected in the turbine inlet bowl which brought the TDAFW shaft to rest. Subsequent testing of the TDAFW turbine, from rest, was successful, and the TDAFW turbine/pump was declared operable. To ensure the pump is not rotating during plant operation, operators closed valve 3MSS-17C which had exhibited the most significant seat leakage.

NNECO is currently pursuing two efforts regarding the TDAFW pump. The first effort is to assure that the TDAFW shaft remains at rest when the pump is not in service. To assure that TDAFW turbine shaft rotation will be detected, NNECO has implemented a temporary surveillance (OPS Form 3670.3-4) that requires visual confirmation, every four hours that the TDAFW turbine shaft is not rotating. The inspector confirmed that this temporary surveillance is being performed. The second effort involves corrective maintenance to eliminate leakage past the steam admission valves. The inspector confirmed that valves AOV 31A, B, and D have been scheduled for maintenance during the next refueling outage which is expected to begin on February 1, 1991.

The inspector discussed NRC report AEOD/C602, dated August 1986, with NNECO staff. The subject report describes overspeed trips of steam-driven turbines at other nuclear power facilities. The report suggests that a modification to the Woodward Governor, which provides an auto-bleed device for the speed setting cylinder, will prevent a turbine trip on overspeed if the shaft is rolling prior to turbine start. The modification is known to be applicable to the type PG-PL governor; however, the Millstone Unit 3 TDAFW turbine utilizes a type PGG governor. NNECO indicated that they would review the report and consider a modification as part of a long-term corrective action.

The inspector asked the Unit 3 engineering staff if they had contacted the Unit 2 engineering staff regarding PIR 390-160 in that Unit 2 also utilizes a Terry Turbine with a Woodward Governor for a TDAFW system. The Unit 3 engineering staff indicated that they did not pass along this information since it was known that the Unit 2 TDAFW turbine/governor was of a different design and was not susceptible to the problems experienced at Unit 3.

Based upon review of the licensee actions, the inspector considered the licensee approach to the turbine driven feedwater pump to be acceptable.

3.4.2 INCOMPLETE RESTORATION OF RADIATION MONITOR 3HVR*RE10B - MILLSTONE 3

During the report period, the turbine building vent radiation monitor 3HVR*RE10B was declared out of service. Technical Specification 3.3.3.10 "Radioactive Gaseous Effluent Monitoring Instrumentation" requires taking grab samples and estimating flow rates if the monitor is declared inoperable. Proper performance of these actions, however, did not occur which resulted in the generation of two plant incident reports and license event report 90-29. These events are discussed below:

First Event PIR 3-90-174

This event occurred on November 23, 1990, and included the failure of operations personnel to record flow data associated with a temporary sample log, which was initiated when monitor 3HVR*RE10B was declared inoperable. The inspector reviewed the subject temporary sample log, OPS Form 3670.2-4, and noted that the data were taken every four hours, as required, except for the period from 8:35 am on November 23, 1990, to 7:54 pm on November 23, 1990, a span of approximately 11 hours. NNECO attributed the failure to take the required data to personnel error in that the primary equipment operator (PEO) did not properly note the temporary logs on the shift turnover report and briefing as required. Additionally, the shift supervisor failed to note the omission during routine review of the shift turnover logs.

As corrective action, the personnel involved in the event were counseled. Additionally, OPS Form 3670.2-4 was modified to require issuance of a new temporary log daily. Prior to the modification of the form, when radiation monitor flow rates would be recorded every four hours, as many as 20 entries could exist on a single form. By limiting the number of entries to a single form, management overview of this surveillance would be increased reducing the possibility for error. Additionally, this modification would make the temporary sample log consistent with other temporary logs which are issued on a daily basis.

Inspector review of this event concluded that this incident was an isolated result of minor personnel error. NNECO had reported a similar incident in LER 88-17 in which data were not taken when radiation monitor 3HVR*RE10B was inoperable. However, that event was attributed to poor communication between the operations department (responsible for requesting the data) and chemistry department (responsible for recording the data). The inspector noted that the event had minor safety significance, since alternate radiation monitors, which monitor the effluent path were operable during the time period and showed no increase in activity. The inspector considered this event to be an isolated example of poor communication during turnover. The inspector verified that the criteria of 10 CFR 2 Appendix C section V.G.1 were met, and no violation will be issued **(50-423/90-27-01)**.

Second Event PIR 3-90-176

The subject PIR describes an incident involving the inadequate restoration of radiation monitor 3HVR*RE10B. NNECO was returning the subject radiation monitor to operable status on November 28, 1990. Prior to declaring the monitor operable, a filter element was to be transferred from the temporary monitoring skid to 3HVR*RE10B. The monitor, however, was declared operable without the filter having been transferred. LER 89-27 dated November 30, 1989, reported a similar case in which radiation monitor HVQ-99 was declared operable prior to transfer of the filter element from the temporary sample rig. The corrective action specified in LER 89-27 stated that a "... procedure will be developed covering restoration of radiation monitors." Operating Procedure 3250.62 "Restoring Radiation Monitors to Service" was implemented on December 3, 1990. The inspector reviewed the procedure and determined it to be adequate. The inspector concluded that although radiation monitors 3HVR*RE10B and HVQ-99 are different in design, the root cause of PIR 390-176 and LER 89-27 is the same in that the monitors were declared operable prior to adequate restoration. NNECO's failure to take timely corrective action in the case of LER 89-27 resulted in the incident reported in PIR 390-176. The inspector verified that this incident was of minor safety significance and that the criteria of 10 CFR 2 Appendix C, section V.G.1 were met. No violation will be issued **(50-423/90-27-02)**.

3.5 SYSTEM WALKDOWNS

3.5.1 INSTRUMENT AIR SYSTEM - MILLSTONE 3

The inspector performed a review of the Millstone 3 instrument air system. The review consisted of the following: a verification that the instrument air system lineup as specified in Operating Procedure 3332A "Instrument Air System" matched the actual lineup in the field, a walkdown of Abnormal Operating Procedure AOP 3562 "Loss of Instrument Air," a comparison of the Final Safety Analysis Report system description to the as-built and installed instrument air system, and a comparison of the plant system diagrams to as-built configuration.

Results

During the plant walkdown, the inspector noted that the instrument air system was in good condition, valves were properly tagged for maintenance activities, and system compressors appeared to be in good condition. The inspector noted that valves IAS-500, IAS-V34, IAS-V756 and IAS-V757 were not labeled. Additionally, valve IAS-V786 "cold shutdown instrument air filter 4A inlet" was found to be open vice the required closed position. This finding, however, was of little safety significance since the filter outlet isolation valve was shut and the opposite filter was on line as specified in the valve lineup. When the shift supervisor was informed of the out-of-position valve, it was promptly closed.

The inspector noted that while the system valve lineup established the position of the compressor cylinder unloader valves, isolation valves which are on the instrument air supply from the unloader valves to the compressor cylinders are not positioned. If these valves were shut for a maintenance activity and not reopened, the compressor unloader would be isolated. This condition could lead to premature compressor wear since the compressor would be forced to start under full cylinder compression load. The inspector discussed this issue with the operations manager who indicated that he would review the valve lineup and consider the inspector's comment.

During the system walkdown, the inspector noted that portions of bypass jumper 3-89-47 were installed. This jumper utilized compressed air hoses which supplied air to portions of the instrument air system that were taken out of service to facilitate performance of a maintenance activity during the 1989 outage. The jumper was not valved into the system; however, it was listed as cleared and the system restored in the jumper bypass log. This status is contrary to ACP-QA-2.06B "station bypass/jumper control" which states that jumpers are to be removed prior to being cleared from the jumper bypass log. The jumper was subsequently removed by the operations department when informed of the finding. The inspector discussed this finding with the operations manager who indicated that a decision was made subsequent to the 1989 outage to leave the hoses installed to support future work activities. The manager indicated that the hoses should have been identified to ensure control of system status is maintained. The inspector considered this discovery to be an isolated example of incomplete bypass jumper restoration as other examples have not been identified. The inspector will continue to follow jumper restoration in future routine inspections.

A comparison of plant diagrams to as-built configuration and the system description as described in the Final Safety Analyses Report revealed three minor discrepancies. The shutdown instrument air compressors were labeled 3IAS-SKID-1A and 1B vice 3IAS-C2A and 2B as specified on the plant diagrams. Additionally, a line from the shutdown compressor air receivers to the compressor unloader was not labeled on the plant system diagram. Section 9.3.1.2.4 of the FSAR incorrectly referred to the containment instrument air compressors which are no longer installed. The inspector informed the cognizant system engineer and operations manager of these minor items.

No significant weaknesses were noted with AOP 3562. The inspector did note, however, that two valves which the procedure operates, AOV-33 which isolates the service air system if the instrument air pressure decreases and AOV-95 which unisolates the shutdown instrument air compressors on a loss of offsite power, are not cycled by the licensee. Therefore, there is no assurance that the valves could operate as designed. The inspector discussed this issue with Millstone Unit 3 plant engineering personnel who indicated that this item would be reviewed.

During the plant walkdown, the inspector noted that the shutdown instrument air compressors which are powered off class 1E vital power supplies, are cooled by a section of the reactor plant component cooling water (RBCCW) system which isolates on a loss of offsite power. Section 9.3.1.1 of the Millstone Unit 3 FSAR which contains the design bases of the instrument air system states that the shutdown instrument air compressors are designed to start upon receipt of a loss of offsite power (LOP) and supply air to vital plant components to assure an orderly plant shutdown. Although the compressors will start on an LOP signal, it is clear that they would not operate for an extended period of time without RBCCW cooling water. NNECO has recognized this deficiency and intends to modify the compressor cooling system so that it would tie into a section of the RBCCW piping which does not isolate on a LOP signal. This is currently scheduled for the fourth refuel outage. The inspector noted that this effort would substantially improve recovery of the unit if a LOP occurred.

In summary, the instrument air system was found to be in good condition, valves were properly aligned and system configuration maintained. The abnormal operating procedure for loss of instrument air was determined to be acceptable. NNECO is actively trying to improve instrument air reliability by modifying the shutdown instrument air compressor cooling water supply.

3.6 OUTAGE ACTIVITIES - MILLSTONE 2

Millstone 2 was shut down between December 29, 1990, and January 7, 1991, primarily to replace steam generator primary manway and pressurizer manway gaskets. NNECO activities on this issue are documented in report section 7.3.

Other major outage maintenance activities include placement of the "C" reactor coolant pump seal, replacement of the "B" heater drain pump upper motor bearing, leak repairs of containment valves, and repair of the "A" atmospheric dump valve.

On December 31, 1990, at 9:58 pm Millstone 2 entered reduced inventory operation as defined in NRC Generic Letter 88-17, "Loss of Decay Heat Removal." The inspector reviewed controls on containment integrity, establishment of a reactor coolant system vent path, reactor vessel level and temperature indications, and availability of alternate injection sources into the reactor coolant system. All equipment and controls were being implemented as committed to by NNECO. Operators were knowledgeable of the controls and implementation of postulated contingency actions.

The inspector reviewed the adequacy of various equipment tagging orders. The review considered the adequacy of component isolation based on the maintenance activity, verification of the equipment tags hung, and administrative controls as documented in administrative control procedure (ACP)-QA-2.06A. Tagging orders reviewed and components worked on were:

Tagging Order	Component
2-9-91	"A" Emergency Diesel Generator DC Air Compressor
2-8-91	"B" Main Feedwater Pump Main Feedwater Check
2-5-91	Main Feedwater Check Valve (2-FW-5A)
2-3138-90	No. 2 Steam Generator Heater Drain Valve (2-MS-296)
2-3127-90	"A" Reactor Coolant Pump Oil Leak Inspection
2-3125-90	"C" Reactor Coolant Pump Seal Replacement

No deficiencies were identified in isolation of components or in controls as prescribed in ACP-QA-2.06A.

In conclusion, the inspector determined that plant control of the outage, based on review of equipment tag review, reduced inventory operations, and control of maintenance activities was implemented in an acceptable manner.

3.7 PREVIOUSLY IDENTIFIED ITEMS

3.7.1 (CLOSED) OPEN ITEM 50-336/88-10-04: EMERGENCY OPERATING PROCEDURE PROCEDURAL DEFICIENCIES

Inspection report 50-336/88-10 identified a number of deficiencies associated with emergency operating procedure implementation and labeling of related equipment. These items have been reviewed by NNECO and upgrades have been made in response to the open item. The inspector verified the upgrades which included relabeling of control panel mimic components, upgrade of steam generator pressure and level gauges to improve readability, and labeling of controls for the containment hydrogen monitors. No deficiencies were identified. This item is considered closed.

4.0 RADIOLOGICAL CONTROLS

4.1 POSTING AND CONTROL OF RADIOLOGICAL AREAS - ALL UNITS

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

The inspectors had no significant observations.

4.2 RADWASTE PACKAGING - CONTROL ROD BLADE CUTTER SHIPMENT - MILLSTONE STATION

NNECO management notified the inspector on November 19, 1990, of the receipt of additional information regarding a radwaste shipment in January 1990. NNECO was notified by the Duane Arnold licensee on November 15 of a new problem associated with the shipment of a control rod blade shearer-compactor to that facility.

The shipment was made on January 17, 1990, from the Millstone site under authorized work order MP 90-00241. Upon receipt at the Duane Arnold facility on January 19, 1990, it was determined that the radiation level on the package was 300 millirem per hr, which exceeded the 200 millirem per hour limit of 49 CFR 173.411(a) and 10 CFR 71.47. NRC followup of this matter was documented in Combined Inspection Report 50-245/90-03, 50-336/90-04 and 50-423/90-03.

The new information reported to NNECO on November 15 was that the concentration of Cobalt 60 in the package exceeded the limits allowed by the Duane Arnold facility license. Duane Arnold license condition 2.B(4) allows that licensee, pursuant to 10 CFR 30, to receive up to 100 millicuries of byproduct material. The control rod blade cutter had 119 millicuries of Co-60 when the tool was first shipped to Millstone from Duane Arnold, and had 114 millicuries of Co-60 when it was sent back to Duane Arnold on January 17, 1990. The Millstone operating license permits NNECO to receive byproduct material, but does not contain a similar limit on the allowable quantity.

NNECO stated that its review of the incident identified no applicable reporting requirements, but the matter constituted a violation of the regulations. 10 CFR 30.41(c) requires, in part, that "Before transporting byproduct material..., the licensee transferring the material shall verify that the transferee's license authorizes the receipt of the type, form, and quantity of byproduct material to be transferred."

The Radioactive Shipment Checklist Form, RW 6004/26004/36004, requires that NNECO verify that a copy of the transferee's license is on file at Millstone and that it has not expired. NNECO personnel indicated that the operating licenses are normally reviewed, but that the limitation in the Duane Arnold license was overlooked for the January 1990 shipment.

NNECO personnel revised the radwaste check off lists to include a specific instruction to review licenses for limitations in type, form and quantity of byproduct materials to be shipped. The inspector reviewed Revision 5 of the Radioactive Shipment Checklist to verify the change had been made and noted this action would be appropriate to preclude a recurrence of the error.

This matter was reviewed with Region I personnel, including members of the Division of Radiation Safety and Safeguards. The inspector noted that the shipment of the control rod shearer containing 114 millicuries of Co-60 to Duane Arnold on January 17, 1990 was a violation of the requirements of 10 CFR 30.41 and the Duane Arnold facility license. The matter has low safety significance. The implementation of NNECO's corrective actions will be reviewed further during a subsequent routine inspection of NNECO's radwaste program by NRC Region I personnel. The inspector verified that the criteria of 10 CFR 2, Appendix C, Section V.G.1 were met, and no violation will be issued **(245/90-25-01, 50-336/90-28-01, and 50-423/90-27-03)**.

5.0 EMERGENCY PREPAREDNESS

On December 5, 1990, NNECO conducted a partial participation exercise at Millstone 2. The exercise scenario simulated a leak from a steam generator nozzle dam during refueling operations, that resulted in a loss of water in the spent fuel pool and created the potential for core uncovery. The exercise began with an Alert declaration due to radiation levels in excess of 1000 times normal for 5 minutes and escalated to a Site Area Emergency due to a fire that affected safety systems. A General Emergency was declared based on a blackout of Millstone 2 with the potential for lasting more than 2 hours.

Nine state government agencies, ten towns and two private organizations participated in offsite activities. Since this was a partial participation exercise, the offsite activities were not evaluated by the NRC.

NNECO's conduct of the exercise was evaluated by an NRC review team, which included two resident inspectors. The full summary of NRC findings is provided in NRC combined inspection report 50-245/90-84, 50-336/90-83 and 50-423/90-84.

6.0 SECURITY

Selected aspects of site security were verified to be proper during inspection tours, including site access controls, personnel searches, personnel monitoring, placement of physical barriers, compensatory measures, guard staffing, and response to alarms and degraded conditions. The inspector also reviewed a personnel matter that had implications in NNECO's fitness for duty program, as described below. Site security controls were effectively implemented.

NNECO management notified the inspector on December 13 of a personnel matter on December 12 that had potential fitness for duty implications. A guard was relieved from duty by a guard supervisor because of behavior that was considered disruptive to shift activities. The guard was sent home at 5:00 p.m. The guard supervisor evaluated the incident as a personnel matter due to known personal and family problems being experienced by the guard. The guard supervisor specifically evaluated the guard's behavior and had no reason to believe the guard was unfit for duty.

When the incident was reported to NNECO station security supervision, NNECO concluded that the guard's behavior could have been considered as "aberrant", and, as a conservative action, the guard should have been subjected to a for-cause drug test in accordance with the NNECO fitness for duty program. The guard was contacted at home at about 6:00 p.m. on December 12 and initially agreed to come back to the site that evening for a test. When contacted again at about 6:30 p.m. to arrange for transport back to the site, the guard refused to come in and submit to the test. The guard was suspended and protected area access was restricted pending resolution of the matter.

NNECO efforts to arrange for a test continued until the guard submitted to a test on December 18. The results were negative. The guard had also tested negative for random drug screenings in October and November, 1990. The guard was referred to an employee assistance program. The suspension remained in effect pending completion of the EAP and re-evaluation by the medical review officer for approval to resume duty.

The inspector reviewed the issue for reportability and no reports were necessary in accordance with the requirements of 10 CFR Part 26.73. The inspector also reviewed NNECO's action in this matter in regard to "for-cause" testing requirements per 10 CFR 26.24(a)(3). NNECO actions were conservative and appropriate.

7.0 MAINTENANCE/SURVEILLANCE

7.1 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:

Millstone 1

M1-90-10636, "1A" LPCI Pump Motor, 12/4/90

M1-90-10638, "1A" LPCI Motor Overhaul, 12/5/90

M1-90-10991, Test LPCI Motor for Suitability, 12/19/90

M1-90-11053, Manufacture New LPCI Pump Mounting Plate, 12/18/90

M1-90-11059, Install New LPCI Motor per PDCR 1-77-90, 12/20/90

M1-90-11147, "A" LPCI Pump Replacement Tests, 12/21/90

Millstone 2:

M2-90-14902, "F-212" Flow Orifice Repair, 11/27/90

M2-90-13311, "C" Service Water Pump Repair, 11/6/90

Millstone 3:

M3-90-12248 ESF SCIF Contained Air Conditioning Unit Repairing and Fan Belt Changeout, 12/13/90

M3 90-20728 "A" Emergency Diesel Generator Maintenance, 12/12/90

M3-90-21838 Diesel Generator Service Water Heat Exchanger Outlet Valve Maintenance, 12/11/90

M3-90-20604 Meter Calibration, 12/14/90

Except as noted below, the inspector had no noteworthy observations.

7.1.1 SEAL INJECTION FILTER LEAK - MILLSTONE 3

On December 6, 1990, at 7:00 p.m., a leak developed in the "B" reactor coolant pump seal injection filter assembly. The loss of reactor coolant decreased seal injection flow from 9 to 8.5 gallons/minute; however, it did not reach the low flow alarm which is set at 6.5 gallons/minute. Operators who responded to the event unisolated the "A" filter assembly and isolated the "B" stopping the leak. Isolation of the assembly was hampered by the inability of the remote filter reach rods to operate. The failure necessitated an operator entering the cubicle to operate the isolation valves locally. Operation of the valves locally resulted in the contamination of the individual's clothing despite the wearing of a plastic suit and face shield. The individual's clothing was subsequently deconned by health physics personnel. NNECO investigation attributed the cause to a torn filter 0-ring. The 0-ring in which the filter sits was inspected prior to filter installation and no defects were found. However, the filter was apparently damaged during the installation process which is performed remotely using long-handled tools. Plant incident report 3-90-183 documented the event and licensee corrective actions. NNECO is investigating the use of a different filter assembly with an improved seating surface which may prevent similar failures.

The inspector noted that in response to the decrease in reactor coolant pump seal injection flow, operators appropriately utilized AOP 3555 "Reactor Coolant Leak." Additionally, health physics personnel, in response to an earlier spill of reactor coolant from a seal injection filter, staged a spill kit at the entrance to the filter cubicle. This was good preplanning which minimized personnel response time to the event. Review of prior work orders revealed that of 45 previous filter changeouts, only 3 were due to filter leaks. Therefore, this failure appears to be an isolated instance of improper filter installation. The reach rod problems identified during the event were attributed to the operators unfamiliarity with the operation of the reach rod. Specifically, the reach rods are designed with clutches which disengage when under severe torque. This feature is installed to prevent valve damage. The clutches however can be overridden by pushing the remote operator handwheel into a spline. This information is being disseminated to operating personnel to address this issue. Upon reviewing this event, the inspector found operator, health physics, and engineering followup to be adequate and had no further questions.

7.2 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:

<u>Millstone 1</u>

PT 1405, Testing of 4.116KV and 6.9 KV Motors and Surge Capacitors

SP 621.10, Core Spray System Operability Test

SP 622.7, LPCI System Operability Test

SP 622.10, LPCI System Narrow Range Flow Verification

SP 623.18, Emergency Systems Valve Position Check

<u>Millstone 2</u>

SP 2618G-2, Fire Damper Operability Test, 12/18/90

OP 2301E, Reactor Coolant System Draindown, 12/31/90

Millstone 3

SP 3443A21, Protection Set I Operational Test, 12/10/90

SP 3606.4, Containment Accumulation Pump 3RSS+PID Operation Readiness Test, 12/27/90

SP 3616A.1, Main Steam System Valve Operability Test, 12/14/90

Except as noted below, the inspector had no noteworthy observations.

7.2.1 EMERGENCY DIESEL GENERATOR FUEL OIL - ALL UNITS

NNECO's programs and procedures that are used to ensure the availability of the proper quantity and quality of emergency diesel generator (EDG) fuel oil to meet the technical specification requirements for both the fuel oil and EDG operability were previously addressed in NRC Inspection Reports 50-245/90-11; 50-336/89-16; and 50-423/89-02 for Units 1, 2 and 3, respectively. This inspection was made to address in more detail certain aspects of NNECO's fuel oil programs including improvements and changes made based upon NNECO self evaluation of the fuel oil program.

Millstone 1

In station procedure SP 667.11, revision 4, Unit 1 Diesel Fuel Sampling Analysis, NNECO implemented the use of the preferred all levels sampling method of ASTM D4057 for new incoming fuel oil tankers. This method provides a more representative sample than the top sample method previously used. In addition to analyses based on the requirements of ASTM D975-81, the new fuel is analyzed for total particulate contamination in accordance with ASTM D2276.

Monthly samples of stored fuel oil are taken from the day tanks and the fuel oil storage tank. Near bottom grab "spot" samples are taken from the storage tank. A day tank drain cock from near the bottom is used for collecting day tank "drain" samples. Analyses of these samples now include the "clear and bright" and "water content" acceptance criteria of ASTM D4176. The samples are also analyzed for total particulate contamination in accordance with ASTM D2276 (changed from ASTM D2274) with an upper limit of 20 mg/liter. Instructions are provided in NNECO's Fuel Sampling Analyses Procedure SP 668.11 which requires corrective actions if the particulate content is greater than 10 mg/liter or if the sample fails to pass the "clear and bright" tests.

The inspector noted the above improvements/changes in the licensee's fuel oil program/procedures as being positive steps to better ensure the availability of the proper quality fuel oil for operation of the EDG units.

Millstone 2

In station procedure SP 2613E, revision 4, Diesel Generator Fuel Oil Delivery and Sampling, NNECO has now implemented the use of the all levels sampling method of ASTM D270-65 for new incoming fuel oil tankers. This is expected to provide a more representative sample than the top level method previously used. The new fuel is analyzed to meet the requirements of ASTM D975-1975 and to meet specific gravity requirements and fuel impurity level restrictions (20mg/liter) when tested in accordance with ASTM D2274. Fuel oil tankers that are compartmented now require separate sampling and analysis for each compartment.

Fuel oil in the main storage tank (T47A) and in the two EDG unit storage tanks (T48A, T48B) is sampled/analyzed for particulates at least once each 92 days in accordance with ASTM D2276 (changed form ASTM D2274). In addition, the main storage tank T47A is sampled immediately after a new fuel delivery for analyses to confirm that the stored fuel oil meets the fuel oil and particulates requirements of ASTM D975 and ASTM D2276 (20mg/liter). Fuel oil samples from all three of the storage tanks could be considered as "lower tank" samples as shown in ASTM D270 due to the fact that the pumped sample (T47A) and the drain line samples (T48A and T48B) pick up locations are approximately six-inches above the tank bottoms. Fuel oil particulates concentration is trended for indications of particulates buildup. If there are particulate buildup indications, the analyses frequency is increased to once a month until the problem is resolved.

The inspector noted the above improvements/changes in NNECO's fuel oil program procedures as being positive steps to better ensure the availability of the proper quality fuel oil for operation of the EDG units.

Millstone 3

Since the NRC EDG fuel oil inspection reported in 50-423/89-02, NNECO changed the incoming new fuel oil sampling procedure from the "top sample" to the "all levels" procedure of specification ASTM D4057. The "all levels" method is considered superior for obtaining the samples because of potential stratification that may exist in the fuel delivery tanker truck. This stratification may include water near the bottom.

Inspection of the sampling/analysis of the stored EDG fuel oil for particulates contamination verified that NNECO is following specification ASTM D2276-78 method A3 (sample bottle method) in performing this evaluation. This sampling is performed at least once each 31 days. The samples taken are not taken directly from the fuel storage tanks (due to tank design and location) but from the pumped fuel lines that go from the storage tanks to the EDG day tanks by means of a sampling point in the common fuel crossover line. Prior to taking the samples, the fuel oil recirculation line valve is opened and the fuel transfer pump is operated for approximately 45 minutes to purge the lines. This provides some mixing of the fuel (10 gpm for a total of 450 gallons in these 35000 gallon tanks). The fuel oil pick-up in these storage tanks is a few inches above the bottom of the storage tanks, therefore the sample approximates a lower or clearance sampling point as described in ASTM D4047. The sample taken from this point in the tank would be representative of the fuel oil pumped to the EDG unit after 45 minutes of operation.

The inspector further observed that the fuel oil line has a course 70 micron 200 mesh strainer/filter near the discharge of the fuel oil transfer pump. This strainer is equipped with differential pressure indication/alarm. The purpose of this strainer is to prevent relatively large contaminants from being pumped to the day tanks and then to the 5 micron EDG duplex filters. The strainers do not prevent smaller particulate sizes, indicative of fuel oil stability or possible tank degradation from being detected. Further, the inspector observed that the fuel oil storage tanks are equipped with a bottom recessed sump well. In accordance with procedure SP 2646B.5, this sump well is pumped monthly to remove accumulated water and check the fuel to determine if it is "clear and bright." This monthly pumping of the sump is also expected to detect and remove heavy particulates and bottom sediment. NNECO considers that the monthly sampling method in use meets the intent of ASTM D4057 in obtaining a fuel oil sample which can provide reliable indications of fuel oil quality when analyzed to the appropriate ASTM specifications (ASTM D975 and ASTM D2276).

The inspector noted the above improvements/changes made in NNECO's fuel oil program, since the previous inspection reported in NRC report 50-423/89-02, as being positive steps to better ensure the availability of the proper quality fuel oil for operation of the EDG units. Although other sampling methods of ASTM D4057 might provide more representative or possibly worst case samples of the fuel oil from the storage tanks for monthly analyses, the inspector concurs with NNECO that the method currently used can provide reliable indications of diesel fuel oil quality.

7.2.2 REACTOR PROTECTION SYSTEM CONDENSER VACUUM SWITCHES OUT OF CALIBRATION - MILLSTONE 1

On December 12, 1990, with the plant at 100% of rated power, all four condenser vacuum scram switch setpoints were found to be non-conservative. The condition was discovered during performance of monthly surveillance procedure SP-408J, Condenser Low Vacuum Scram Functional Test/Calibration, revision 8, dated January 31, 1990. In accordance with its emergency plan implementing procedures, the licensee notified the NRC of the occurrence pursuant to 10 CFR 50.72 (b)(2)(iii), any event or condition that alone could have prevented the fulfillment of a safety function. NNECO also initiated plant incident report 1-90-100 to document the event. Since the switches were re-calibrated and satisfactorily tested immediately, no reactor shutdown was initiated as a result of the condition.

The four Barksdale vacuum switches provide scram signals to the reactor protection system at 23.0 inches of mercury (" Hg), nominal, when the mode switch is in the "run" position. A decreasing vacuum in the main condenser could lead to closure of the turbine stop valves and loss of reactor heat sink. The low vacuum scram function ameliorates a reactor pressure and power transient by providing an anticipatory scram prior to the turbine stop valve closure scram at 22.5" Hg. On December 12, two of the pressure switches actuated at 22.9" Hg and two at 22.5" Hg.

Technical specification table 3.1.1, Reactor Protection System (Scram) Instrumentation Requirements, requires that the vacuum switch setpoint be greater than or equal to 23 "Hg. A minimum of two operable instrument channels per trip system were required to be operable at the time of the event. With both trip systems inoperable, all control rods must be inserted within four hours, or the main steam isolation valves must be closed within eight hours. The inspector discussed corrective activities and reviewed switch test data with operations and instrumentation and controls personnel to assess the event. Since the switches were re-calibrated immediately upon discovering the discrepant condition, plant shutdown pursuant to technical specification table 3.1.1 was not initiated. NNECO verified the accuracy of the test standard used during the performance of the surveillance test. Atmospheric pressure and ambient temperature measurements were performed near the switches in order to identify the cause of the setpoint drift. A review of previous surveillance test results indicated that the switch setpoints tend to drift downward with decreasing ambient temperature. This tendency was confirmed by technical information provided by the vendor.

In order to monitor the switches for setpoint variations, NNECO has scheduled the surveillance test for performance weekly. Surveillance test results for other safety-related Barksdale switches were reviewed and no similar problems were identified. Finally, NNECO is evaluating the need for a plant design change to account for temperature drift effects on switch setpoints in order to provide greater margin to the technical specification limit.

The inspector reviewed the updated final safety analysis report accident analysis to assess the safety significance of the non-conservative setpoints. The analysis assumes an immediate loss of condenser vacuum. The consequences of the event are similar to a turbine trip coincident with failure of the main steam bypass system. Closure of the turbine stop valves would result in a rapid increase in reactor coolant system pressure and reactor power. The power transient is limited by a scram from closure of the turbine stop valves, while the heat energy in the reactor coolant system is dissipated by operation of the safety relief valves. The transient is bounded by a load reject without bypass system event. Thus, no reactor thermal or hydraulic limits are exceeded. Finally, the vacuum switch scram function is backed up by scram on turbine stop valve closure, high reactor pressure scram, and the high-high average power range monitor trip.

The inspector determined that the safety significance of this event was low. NNECO properly evaluated the event for reportability and took prompt and thorough corrective action to correct the discrepant condition. The inspector had no further questions.

7.3 STEAM GENERATOR MANWAY LEAKAGE - MILLSTONE 2

Event Description

On December 20, NNECO personnel entered containment to investigate an increase in containment sump pumping frequency. The licensee identified that containment sump fill rate increased by approximately 6 gallons per hour (0.1 gallon per minute) on December 18 to a total calculated unidentified leakage of 0.2 gpm. No corresponding increase was noted in containment bulk air temperatures, or containment particulate and gaseous radiation readings. The containment entry identified potential leakage from the No. 2 steam generator hot leg primary manway. The leakage point could not be specifically identified based on distance and radiation considerations. On December 29, at approximately 11:40 am NNECO reported pursuant to 10 CFR 50.72 (b)(2)(vi) an unplanned shutdown to address steam generator manway leakage. At the time, the facility was at 5% rated power with insulation removed from all four manways on both steam generators. NNECO identified all four manways indicated leakage by boric acid residue and steam wisps around the gasketed areas.

Background

On November 21, 1990, the facility attained full power operation following the cycle 11 refuel outage. During the outage, all four primary manways were removed to facilitate steam generator tube eddy current testing and tube repairs. At the end of the outage, all manway gaskets were replaced and were visually inspected at normal operating temperatures and pressures for leakage. No leakage was noted.

Two primary manways are located on each steam generator, one each for the hot and cold leg plenums. Each manway has an inside diameter of 16 inches. The manways are installed to facilitate entry into the primary plenums for each steam generator. The manway covers are fastened to the steam generator vessel by twenty, 14 1/4 inch long, 1 1/2-8UN-2A manway bolts. The manway cover holds the diaphragm plate into the gasket.

The replacement gasket was manufactured by Flexitallic, with a trade name of "Flexite Super." The gasket type is spiral wound with graphite and mineral paper constituted the filler material, and winding material made of type-304 stainless steel.

Historically, NNECO installed flexitallic asbestos gaskets (Combustion Engineering part number 120-06). This type of gasket is identified in the initial steam generator drawing (CE E233645). During the time interval between June 15, 1977, and July 16, 1986, asbestos gaskets were purchased by the licensee. In the 1988 timeframe, the warehouse initiated a program to remove asbestos containing material from storage. As a result of this program, and by control by a repeating purchase requisition, the warehouse ordered non-asbestos "Flexite Super" in late 1988. According to the supplier, this type of gasket material was a replacement for the asbestos containing gasket. No engineering oversight was performed that reviewed replacement gasket acceptability.

NNECO Actions

NNECO actions to address the cause of gasket leakage included review of past NRC correspondence on steam generator primary manway leakage, review of recent industry events and conclusions, gasket accountability and past purchase requisitions, visual inspections of the gasket surfaces, dimensional analysis of primary diaphragms, and inspections of fasteners.

Past NRC correspondence of primary manway gasket failures was identified in IE Notice 82-06, "Failure of Steam Generator Primary Side Manway Closure Studs." The notice alerted NNECO of failures of closure studs when subjected to sealing compounds such as Fermanite. The notice did not specifically address failures of gasket materials.

Two recent industry events at Diablo Canyon in August 1988 and Zion 1 in March 1989 identified steam generator primary manway leakage. The Diablo Canyon licensee used "flexite super" gaskets that experienced compressive creep and loss of resiliency due to changes in organic filler material when exposed to high temperatures (greater than 400 degrees fahrenheit). The Zion 1 failure was caused by a change in installation from asbestos to non-asbestos gasket material.

A dimensional review of the diagrams, gaskets, and acceptable gasket "crush" depths was performed by the licensee. The maximum "crush" depth for the nominal gasket thickness is 0.055 inch. The minimum recommended gasket "crush" by Combustion Engineering is .035 inch. Based on actual diaphragm measurements, and nominal gasket thickness, the actual crush depth was 0.031 inch. The unacceptable "crush" depth was a result of all four steam generator primary diaphragms being outside the allowable specification for thickness. The actual diaphragm thickness measurements were 0.233 inches and the allowable is 0.255 (+.003/-.000) inch. According to NNECO, the diaphragms have been installed in the steam generator manways since approximately 1985. Based on dimensional analysis of the diaphragms, NNECO replaced all four steam generator diagrams.

NNECO completed a visual examination of all 100 steam generator and pressurizer manway studs and nuts (20/manway). NNECO replaced 27 studs and 20 nuts due to boric acid surface corrosion, steam cutting, and mechanical damage during disassembly. The examination and replacements were controlled under American Society of Mechanical Engineers (ASME) Section XI Repair/Replacement Program with procedure ACP-QA-2.18.

Between 1977 through 1986, NNECO purchased and subsequently installed recommended asbestos manway gaskets manufactured by Flexitallic. The gaskets were recommended by Combustion Engineering initially during construction of the facility and afterward during plant operations. In late 1988, in an effort to remove asbestos-containing material from the warehouse locations, the licensee purchased non-asbestos replacement gaskets from Flexitallic with a trade name of "Flexite Super." The control of the alternate material gasket was controlled by a repeating purchase requisition. Based on NNECO review, the "Flexite Super" gaskets have been previously installed in the steam generator manways.

NNECO reviewed all other manway-gasketed areas inside containment. The review considered if the appropriate gasket material was installed in the manway areas. No deficiencies were noted in this review.

NNECO assessment of the root cause for the leakage from the primary manways did not identify a specific cause, but rather a collection of contributing factors. The assessment concluded that diaphragms out of specification resulted in a potential insufficient "crush" of the gaskets. In addition, the type of gasket material was a contributor. (i.e. non-asbestos v. asbestos)

In summary, NNECO replaced four steam generator primary gaskets and the pressurizer gasket with asbestos containing material, replaced 27 studs and 20 nuts and completed a review of all other manway gaskets inside containment to assure that original specification, asbestos-containing material was installed.

Inspector Assessment and Conclusions

The inspector review of the type of leakage from steam generator primary manways concluded that the classification of leakage was unidentified. This conclusion was based on review of technical specification definitions for pressure boundary leakage, unidentified and identified leakage, and review of NRC Regulatory Guide 1.45. The limitation on unidentified leakage is 1.0 gpm. NNECO actions to shut down the facility were taken to prevent exceeding limits on unidentified leakage.

Two items identified by the inspector concerned the maintenance procedure controls of gasket material, and engineering oversight of repeating purchase requisitions for gaskets used in the reactor coolant system. Maintenance procedure MP-2705E "Installation and Removal of Steam Generator Manway Covers" step 5.4.2 requires a quality control inspector to verify the correct gasket by part number. The part number provided in the procedure is 120-06. Part 120-06 is the asbestos-containing flexitallic gasket, not the part number for the "Flexite Super." During the most recent refuel outage, the quality control inspector could not verify the gasket part number and therefore dimensionally verified the gasket specifications provided within the procedure. In addition, no engineering involvement into the acceptability of the alternate gasket material was requested or implemented. Coordination between the warehouse and applicable department was deficient based on no procedure change to reflect the change in material part number for the maintenance procedure, and no oversight on the acceptability of the replacement gasket material. This item is unresolved **(50-336/90-28-02)** pending NNECO resolution of this issue.

Based on a review of NNECO actions and investigations into the cause of the leakage, the inspector concluded corrective actions were extensive.

8.0 ENGINEERING/TECHNICAL SUPPORT

8.1 PREVIOUSLY IDENTIFIED ITEMS

8.1.1 (CLOSED) UNRESOLVED ITEM 89-03-01: NNECO ACTIONS TO INSTALL A COPS ARMING DEVICE - MILLSTONE 3

This item tracked NNECO evaluation of the necessity to provide direct control room indication that the cold overpressure protection system (COPs) was armed. This issue was raised subsequent to a January 17, 1988, event which increased reactor coolant system pressure while the reactor was in cold shutdown. Followup of the event revealed that NNECO personnel were unaware that the COPs system was not operable in violation of Millstone 3 technical specification 3.1.9.3.

In an October 3 , 1990, letter, NNECO informed the NRC staff that the evaluation had been completed and that a positive indication of COPs status would be installed in the control room during the fifth refueling outage. However, exact scheduling would be determined by the ranking the modification achieved in the Integrated Safety Assessment Program. The NRC staff has determined that NNECO's approach to address this issue is acceptable and this item and TI 2500/19, low temperature overpressure protection, is closed.

8.1.2 (CLOSED) OPEN ITEM 85-62-08: VOICE PAGE/EVACUATION ALARM ADEQUACY - MILLSTONE 3

This item was opened to track the NNECO dispositioning of unsatisfactory test results obtained during construction testing while performing steps 7.2 and 7.3 of 3 INT 3031 "Voice/Page Evaluation Alarm Test in Containment." During the performance of this test, high background noise obscured the audible acuity of the evacuation alarm/voice page, rendering the test results unsatisfactory. The highest noise levels were located by the reactor coolant pump motors, the lowest existed by the regenerative heat exchanger room.

To ensure personnel are safely evacuated from the containment structure in an emergency, NNECO instituted the following controls to compensate for the inadequate evacuation alarm system. During Modes 1-4, containment access is controlled by OP 3212A "Containment Entry." This procedure requires personnel who enter the containment to be issued vibrating beepers and are instructed to exit the containment if beeped. During cold shutdown and refuel periods, Modes 5 and 6, area noise is reduced since reactor coolant pumps are not routinely operated. If an evacuation is required, health physics (HP) technicians were instructed via memorandum to ensure personnel who have signed in on radiation work permits (RWPs) have exited the area prior to themselves exiting the containment structure. Areas such as the 24' 6" level of the containment loop areas, which are open for general area access and as such do not require a specific RWP for entry, would also be checked by an HP technician for personnel prior to exiting. The inspector was informed by the Millstone 3 HP supervisor that the guidance to HP technicians would be addressed during subsequent refuel outages and would be tracked as an item via the Millstone Station outage planning group.

Based upon review of NNECO administrative controls that have been established to date for containment access and personnel control, the inspector determined that NNECO has adequately provided short term compensatory measures to address the containment alarm deficiency. Therefore, this item is closed.

8.1.3 (CLOSED) OPEN ITEM 90-08-03: CONTROL ROD CORROSION INVESTIGATION - MILLSTONE 3

This item followed NNECO actions to identify the cause of a June 6, 1990, reactor trip which occurred when control rod G-13 dropped into the reactor core. Investigation of the trip revealed that semi-fluid corrosive material in the stationary gripper coil power cable connector at the conductor/pin interface caused the rod to de-energize.

The corrosive material, when analyzed by an independent laboratory, could not be positively identified. However, it appears that a resin, which is used to seal out moisture, lacked the required catalyst which completely cures the resin. The laboratory hypothesized that the uncured resin formed a corrosive material which deteriorated the connector, de-energized the cable, and dropped the control rod.

NNECO will inspect 10 additional connectors during the upcoming refuel outage commencing February 2, 1991. If the semi-fluid material or damage to the connectors is not found, no action will be taken. If either semi-fluid material or damage to the connectors is found, all 61 connectors will be inspected and repaired as necessary. The inspector has reviewed the licensee response to the reactor trip and root cause investigation, and determined it to be satisfactory. The inspector has no further questions, and this item is closed.

8.2 "C" SERVICE WATER PUMP DISCHARGE HEAD CRACK - MILLSTONE 2

Description of Event

On November 2, 1990, during an inservice inspection, NNECO identified excessive vibration readings on the "C" service water pump. The vibration readings changed from 0.30 inch/sec to 1.30 inch/sec. Since the vibration readings exceeded the alert limit, NNECO secured the pump and performed corrective maintenance activity on the pump. The maintenance activity identified a motor imbalance. NNECO decided to conduct a complete overhaul of the pump. Overhaul of the pump identified cracks and defects at the intersection of the suction bowl and guide vanes and at the discharge elbow.

Background

The pump manufacturer is Hayward Tyler. The base material of the discharge elbow is American Society of Mechanical Engineers (ASME) SA 351 Grade CF3M (316 series stainless steel), and the suction bowl material is A296 CF-68M (316 series stainless steel casting). In 1983, NNECO changed from original cast iron (ASTM A048 class 40) to the existing cast 316 stainless steel material on the pump columns and discharge elbows for all three service water pumps. The modification was controlled by plant design change request (PDCR) 2-64-83, and was incorporated to provide a better corrosion resistant material.

Two independent service water headers are required by technical specification 3.7.4 to ensure that sufficient cooling capacity is available for operation of vital components and engineered safety feature equipment during normal and accident conditions. Supplying the two service water headers are three service water pumps, of which one is considered the "swing pump" and normally not in operation.

The quality indicators for the service water pump are QA-category 1, and seismic class I. The pumps are subject to the inservice inspection program under ASME Section XI. (1980 edition)

NNECO Actions

NNEC0 characterized, evaluated, and repaired the cracks at the suction bowl guide vane welds as documented under non-conformance reports (NCRs) 290-383, 290-385, and 290-386. Based on measurements of the excavated area, the cracks at the suction bowl guide vane welds varied up to approximately five inches in length, 3/4 inch in depth, and 7/8 inch in width. After the removal of the crack indications, the excavated area was inspected with liquid penetrate (LP), built-up by weld repair, and a final LP examination occurred. NNECO did not consider the above to be an ASME Section XI repair since the identified cracks at the suction bowl guide vane welds did not affect the pump pressure boundary. In accordance with ASME Section XI (1980 edition) IWD-4110 repairs to class 3 components and supports are developed for pressure retaining boundary. Pressure retaining components apply to vessel shells, heads and nozzles, pipes, tubes and fittings, valve bodies, pump casings and covers, and bolting which joins pressure retaining items. The suction bowl guide vanes are not part of the service water pump pressure boundary.

NNECO characterized, evaluated, and repaired the cracks at the discharge elbow with NCR 290-394. The excavation of the affected area resulted in an irregular cavity on the inside wall of 9.5 inches by 4 inches, with depths ranging from .15 to .32 inch. In addition, an area of degradation was identified at the same elevation on the external surface of the pump. The external excavated area was approximately 3 inches by 2 inches with a maximum depth of .375 inch. The nominal wall thickness of the pipe elbow is one inch. In addition to the excavated areas, a thru-wall crack was identified. The crack was evaluated by Northeast

Utilities Service Company (NUSCO) engineering to be a result of cast shrinkage. Specifically, an internal chill bar was not consumed in the original casting process, and thus the lack of fusion and cast shrinkage developed the thru-wall crack.

On November 30, NNECO had a conference call with the NRC staff, to discuss the extent of the service water pump corrosion, repair techniques, and additional information to support an ASME Section XI relief request pursuant to 10 CFR 50.55(g)(6)(i). On December 21, 1990, NNECO submitted to the NRC the ASME Section XI relief request for an alternative repair based on the characterization of defects identified. As of the end of the inspection period, NNECO had completed the alternate repairs to the discharge elbow on the "C" service water pump. The pump was installed, and NNECO is controlling the pump as inoperable until a decision is made by the NRC on the status of the relief request.

Inspector Review, Conclusions, and Assessment

Inspection of this item consisted of review of applicable NCRs, authorized work orders (AWOs), discussions with cognizant NNECO and NUSCo engineers, observations of ongoing repair activities, and review of applicable sections to ASME Section XI, ASME Section III, and past maintenance history for all three service water pumps.

The NNECO evaluation of the degraded service water pump was aggressive and comprehensive. Calculation of required minimum wall thickness, and the retrieval and incorporation of original seismic calculations were timely.

Evaluation of the maintenance history of all three service water pumps in the past four years revealed that all pumps had been subjected to at least one complete overhaul and pump alignment. Over this time interval, no material deterioration was identified in the "B" service water pump, and one area of pitting in the "A" service water pump column was identified in November 1989. The previous overhauls of the "C" service water pump in January 1987 and November 1988 revealed degraded areas at the pump columns and welds. All degraded areas were repaired successfully under the ASME Section XI repair program.

The inspector discussed with NNECO personnel the potential susceptibility of the remaining pumps (A and B) to a thru-wall crack based on the conclusion that the internal chill bar was not consumed during the original casting process. NNECO stated that the other pump discharge elbows were visually inspected during the refueling outage with no signs of thru-wall leakage. Further, NUSCO engineering concluded that the consumption of the chill bar was an isolated occurrence based on recent pump overhauls of the A and B service water pumps. The inspector had no further comments.

8.3 "A" LOW PRESSURE COOLANT INJECTION PUMP REPLACEMENT - MILLSTONE 1

The inspector reviewed NNECO plant design change record (PDCR) 1-77-93, Replace LPCI Pump Motor 1502-A, to assess whether NNECO engineering and safety evaluations adequately addressed the effects on plant design of installing a 600 horsepower (HP) motor as a replacement of the original 500 HP motor.

The new LPCI pump motor essentially has the same electrical and mechanical characteristics as the failed motor. The PDCR addressed the following electrical considerations:

-- Pump acceleration time

-- Motor performance under degraded voltage conditions

-- Adequacy of existing cables

-- Effects on protective trip coordination and existing setpoints

-- Effect on emergency diesel generator transient and steady-state loading

-- Effect of starting load on the reserve station services transformer and margin to loss of normal power (LNP) limits on the 345 KV electrical system

The inspector noted that the increased load on the emergency diesel generator is well within the design capacity of the machine, and that, while margin to the Millstone 1 345 KV system LNP limit is reduced, the effect of the modification on the RSST is acceptable. The inspector concluded that the PDCR adequately addressed the relevant electrical effects of this modification.

The new motor and pump/motor transition plate add approximately 1,000 pounds to the existing installation. The PDCR addressed the effect of the added weight on foundation loading and concluded that the modification was bounded by existing analyses. The structural integrity of the new drive coupling and pump-to-motor assembly was acceptable. The inspector verified that the seismic qualification of the new motor was consistent with the assumptions in the updated final safety analysis report.

Due to slip characteristics, use of the 600 HP motor results in a slight increase in pump speed. While pump discharge pressure and net positive suction head requirements increase slightly, the effects are well below system design limits.

Finally, the inspector reviewed the vendor environmental qualification data and the licensee system component evaluation work sheet and concluded that the new motor is suitable for operation in the southwest corner room environment.

In summary, PDCR 1-77-90 comprehensively addressed the mechanical and electrical considerations required to assure the operability of he LPCI pump. The package was complete and conformed to NNECO administrative procedures. The inspector had no further questions regarding this modification.

9.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

9.1 LICENSEE EVENT REPORT REVIEW

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

9.1.1 LER 90-018, LOW CONDENSER VACUUM SWITCHES OUT OF TOLERANCES DUE TO INSTRUMENT DRIFT - MILLSTONE 1

NRC review of this event is documented in section 7.2.2 of this inspection report. NNECO properly reported the event pursuant to 10 CFR 50.73(a)(2)(v), any event or condition that alone could have prevented the fulfillment of a safety function. NNECO corrective action adequately addressed the cause of the event. The inspector had no further questions regarding this LER.

9.1.2 LER 90-021-00, ENTRANCE INTO TECHNICAL SPECIFICATION ACTION 3.0.3 WITH ALL CHARGING FLOW ISOLATED - MILLSTONE 2

Event Description

On November 27, corrective maintenance work on several components in the charging and letdown system resulted in a tag-out of the charging system rendering the system inoperable.

The unit at the time of the event was at full-rated power.

NNECO Actions

Operations had prepared the system tag-out on the operating shift prior to commencement of work activities. The work activities included leak repairs on the 'A' charging pump discharge valve, charging header flow element adjustments and leak repairs, repair of a fire barrier around a letdown pipe penetration, and troubleshooting of the control channel for pressurizer level.

Unit management had authorized the commencement of the work activities at the daily planning meeting on November 27. The decision was based on as-low-as-reasonably achievable exposure, and efficiency in work control. The initial approval of the tag-out was based on the operators understanding of management expectations to accomplish the work activities.

At 9:59 a.m. on November 27 the operators secured charging and letdown and commenced the tag-out sequence. The operators reexamined the applicable technical specification requirements and recognized that the requirements of limiting condition for operation 3.1.2.4 and 3.5.2.d were not satisfied. At this time, operators entered TS 3.0.3 which requires that within one hour action shall be initiated to place the unit in an operational mode in which the specification does apply.

At 10:55 a.m. on November 27, NNECO revised the equipment tag-out, reestablished charging and letdown and exited TS 3.0.3.

NNECO reported the event on December 27 pursuant to 10 CFR 50.7[illegible](2)(i)(B). The report was filed based on guidance provided in NUREG-1022 Supplement 1 concerning the use of LCO 3.0.3.

Inspector Assessment and Conclusions

In discussion with operations management, the inspector noted that a management error occurred in authorization of work activities on the charging and letdown system. Unit management did not fully appreciate the extent of the equipment tag-out.

The inspector reviewed NRC technical guidance on entry into technical specification limiting condition for operation 3.0.3. The basis for TS 3.0.3 is not intended for operational convenience but to provide time limits for an "orderly" shutdown when other specifications cannot be complied with. Voluntary entry removes the last line of defense against potentially harmful events. Removal of a single safety train from service for test and preventive maintenance is acceptable. Removal of redundant safety trains for the same reason is not justified.

NNECO actions to remove the charging system from service was based on corrective maintenance activities, however, when the charging system was isolated, the maintenance activities were of insufficient importance to continue with isolation of the charging system.

Operations department input into the scheduled work activities was not aggressive in identifying the impact of isolation of the charging system, nor was unit management aggressive in understanding the impact of the maintenance activities prior to work approval.

Inspector assessment concluded that NNECO actions were not intended for operational convenience, but rather ineffective communication in authorization and implementation of charging system corrective maintenance activities occurred.

Inspector review determined that the corrective actions in LER 90-21 were adequate to prevent recurrence. No technical specification violation occurred since the limiting condition for operation was not exceeded.

9.2 FOLLOWUP OF PAST EVENTS

9.2.1 REACTOR SCRAM EVENT FOLLOWUP - MILLSTONE 1

On September 14, 1990, during the performance of an annual low reactor pressure alarm switch surveillance, the plant tripped on low reactor vessel water level. Details of the event and NNECO corrective actions are documented in Region I inspection report 50-245/90-17, section 3.3.2, dated October 5, 1990. The inspector identified three items for followup.

--Pursuant to 10 CFR 50.73, NNECO reported the event to the NRC in licensee event report (LER) 90-015, Reactor Scram on Low Water Level, dated October 12, 1990. The event was caused by leaking pressure switch isolation valves which apparently perturbed the reactor vessel level reference legs, causing false high level signals to be transmitted to the feedwater control system. As part of its corrective action plan, NNECO committed to review instruments associated with the reactor vessel reference legs, and which have the potential to cause a reactor scram, to assure that proper procedure guidance is provided. NNECO identified nine instrument calibrations needing specific procedure guidance. Detailed procedures were developed and approved by the plant operations review committee to address the finding. The inspector reviewed procedures IC-427A, Reactor Pressure Gage Calibration, revision 0, and IC-427, Reactor Fuel Zone Level Calibration, revision 0, both dated December 28, 1990. Both procedures provide precautions and specific guidance to minimize the potential for a similar event. The inspector concluded that the new procedures adequately fulfilled the LER commitment, and had no further questions.

--During the September 14 event, the reactor recirculation pumps failed to run back automatically to minimum speed on low feed flow. NNECO determined the cause to be malfunction of a non-Class 1E Agastat time delay relay (TDR). The inspector asked whether similar relays were used in safety-related applications and requested NNECO to provide information regarding plans to replace the relays, if any.

In a letter to NNECO dated March 28, 1989, Amerace Corporation - Agastat Division notified the licensee that Series E7000 TDRs, first sold to nuclear facilities in 1979 for Class 1E applications, had a projected qualified life of 10 years in the de-energized state, that the period was about to expire, and that the TDRs may not be capable of performing their intended function. NNECO initiated a replacement program on May 16, 1989. Forty of the TDRs in the gas turbine generator governor system already had been replaced during the

1989 refueling outage. Seven Series E7000 TDRs were identified in Class 1E circuits. In addition, 33 Series 2400 TDRs, the non-qualified precursors of the 7000 series, were identified in Class 1E circuits.

NNECO stated that one-third of all Agastat relays are tested every operating cycle under production test calibration program 1441. This provided reasonable assurance that the TDRs would function if challenged. Nonetheless, NNECO has scheduled the 40 relays to be replaced with fully qualified electronic time relays during the 1991 refueling outage. The inspector had no further questions regarding this item.

--During the recovery phase of the September 14 reactor trip, Millstone 1 management expressed a concern that too many non-essential personnel were in the control room. The inspector identified no confusion or unsafe conditions as a result of this condition. NNECO administrative control procedure ACP-6.01, Control Room Procedure, revision 21, dated August 24, 1990, assigns the unit shift supervisor/supervisory control operator the authority to limit access to the control room during emergency conditions. The Millstone 1 operations department manager has supplemented this guidance in a memorandum which limits access to the control room under emergency conditions to specified key individuals. The inspector had no further questions regarding this issue.

9.2.2 MANUAL REACTOR SCRAM ON LOSS OF COOLING WATER FLOW - MILLSTONE 1

On October 4, 1990, the plant was manually scrammed on degraded service and circulating water systems flow when three of five intake structure traveling screens were damaged during severe weather conditions. The event is documented in Region I inspection report 50-245/90-20, section 3.3.1, dated December 28, 1990. NNECO corrective action commitments are documented in LER 90-016, Manual Reactor Trip Due to Loss of Cooling, dated November 2, 1990. Additional guidance regarding operator actions to be taken in severe weather to preclude screen damage were incorporated into off-normal procedure ONP-514A, Natural Occurrences, revision 5, dated January 9, 1991. NNECO also reviewed past plant design changes to assure that no other protective functions similar to those which contributed to the event had been removed. None were identified. After reviewing these corrective actions, the inspector considered that NNECO had been responsive to its LER commitments, and had no further questions.

9.3 PERIODIC REPORTS

Upon receipt, periodic reports submitted pursuant to technical specifications were reviewed. The inspector ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

Millstone 1 Monthly Operating Reports for November - December 1990
Millstone 3 Monthly Operating Reports for October - December 1990

No significant observations were made.

9.4 SAFETY SYSTEM FUNCTIONAL INSPECTION - MILLSTONE 1

On November 8, 1990, the Millstone 1 safety system functional inspection (SSFI)_ team reconvened at NNECO management request to review the resolutions of team findings. The SSFI on the condensate/feedwater coolant injection system had been conducted by an in-house team in 1988 and a final report was issued in December 1988. The plant was then tasked with closeout of team findings and the resolutions were published in the team report. In September 1990, NNECO decided to reconvene the SSFI team to review plant closeout actions to ensure that the responses adequately addressed the original findings.

At a meeting on November 8, the team reviewed all 96 of the SSFI findings. A document was assembled containing team comments on the items where it was determined that further clarification by the unit would be required. A meeting between the team and plant representatives was conducted on December 13 to discuss resolution of these 23 items. At this meeting all but four items were closed. NNECO has scheduled a further meeting to resolve the items.

The inspector attended the December 13 meeting. An open and uninhibited exchange of technical considerations was observed. Closeout of open items required concurrence of all team members. Meeting minutes were maintained with the intent of providing a final report of the session to senior NNECO management. The inspector had no questions regarding this self assessment activity.

10.0 MANAGEMENT MEETINGS

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

ATTACHMENT I

MILLSTONE UNIT 1 STATUS

November 16	Millstone 1 at 100% of rated power.
November 22	At 1:15 a.m., power is reduced to 80% for testing of main steam system valves. Full power operation is restored at 1:55 a.m.
November 29	At 5:40 a.m., power is reduced to 30% to perform minor valve maintenance in the heater bay and to repack an isolation condenser system valve. Full power operation is restored on November 30 at 1:58 a.m.
December 6	At 12:06 a.m., power is reduced to 80% for testing of main steam system valves. Full power operation is restored at 1:35 a.m.
December 13	At 12:44 a.m., power is reduced to 65% for routine main steam isolation valve testing. Full power operation is restored at 2:00 a.m.
December 20	At 12:00 a.m., power is reduced for testing of main steam system valves. Full power operation is restored at 1:15 a.m.
December 23	At 5:30 a.m., at the request of the New England Power Exchange, power is reduced to 80%. Full power operation is restored at 6:45 a.m.
December 24	At 3:00 a.m., at the request of the Connecticut Valley Electric Exchange, power is reduced to 87%. Full power operation is restored at 5:30 a.m.
December 28	At 6:50 a.m., power is reduced to 65% to isolate leaks in "B" bay of the main condenser. Full power operation is restored at 11:36 a.m.
January 4 1991	At 12:25 a.m., power is reduced to 80% for testing of a main steam system valves. Full power operation is restored at 1:20 a.m. At 7:50 p.m., power is reduced to 61% to correct a high conductivity condition in "D" bay of the main condenser. Routine main steam isolation testing is performed. Full power operation is restored on January 5 at 2:45 a.m.