U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report Nos.: 50-245/92-25; 50-336/92-27; 50-423/92-24

License Nos.: DPR-21; DPR-65; NPF-49

Licensee: Northeast Nuclear Energy Company
P. O. Box 270
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Facility: Millstone Nuclear Power Station, Units 1, 2, and 3

Inspection at: Waterford, CT

Dates: September 8, 1992 - October 27, 1992

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Reactor Projects Section 4A

Scope: NRC resident inspection of core activities in the areas of plant operations, radiological controls, maintenance, surveillance, security, outage activities, licensee self-assessment, and periodic reports. The inspectors performed special reviews in the following areas: operation of the Unit 1 reactor vessel level reference leg backfill system, maintenance errors on the Unit 1 emergency service water strainer, retest of the Unit 2 enclosure building filtration system, operability of motor-operated valves at Unit 2, operability determinations for the Unit 3 auxiliary building filter system, and effectiveness of the site material receipt inspection program.

The inspectors reviewed plant operations during periods of backshifts (evening shifts) and deep backshifts (weekends, holidays, and midnight shifts). Coverage was provided for 77 hours during evening backshifts and 22 hours during deep backshifts.

Results: See Executive Summary

EXECUTIVE SUMMARY
Millstone Nuclear Power Station
Combined Inspection 245/92-25; 336/92-27; 423/92-24

Plant Operations

Unit 1 operated safely throughout this inspection period. However, the Unit 1 operator requalification program was declared unsatisfactory for the second consecutive year when 5 of 15 operators failed an NRC-administered requalification examination. Unit 2 was shutdown and defueled as the steam generator replacement outage continued. The transfer of new reactor fuel into the spent fuel pool was conducted safely. Unit 3 operated at full power until September 29 when the licensee identified that both trains of the supplementary leak collection and release system (SLCRS) were inoperable due to system degradation, and design and operating deficiencies involving the auxiliary building filter system (ABFS). Unit 3 was shut down and remained in the cold shutdown condition through the end of this inspection.

The inspectors were concerned that operation of the needle valves which regulate flow to certain Unit 1 reactor vessel level instrumentation reference legs to compensate for flow reductions could adversely affect the instrumentation if the material which caused the blockage was dislodged. In response, the licensee developed operating limits for the needle valves which are bounded by the design safety analysis.

Prior to and during the Unit 3 maintenance outage, several tests were performed to define more accurately the operating parameters and design basis of the ABFS and SLCRS. The licensee clarified the dependency between SLCRS and ABFS; ABFS operation is necessary to support SLCRS in meeting the design requirement for a vacuum in the enclosure and auxiliary buildings. The licensee also identified that adequate surveillance tests of SLCRS have not been performed since initial plant startup. The issues of SLCRS and ABFS operability and the adequacy of surveillance testing of these systems are the subject of NRC inspection report 50-423/92-23.

Maintenance/Surveillance

While performing preventive maintenance during this inspection period, the licensee identified that lubricating oil had not been added to the Unit 1 emergency service water strainer gearbox during its installation in August 1992. This omission rendered the strainer inoperable and raised concerns with the adequacy of the licensee's control over spare parts maintained by the maintenance department and proper system verification prior to its return to service. The issue was unresolved pending licensee implementation of corrective actions.

Following discovery of a pinhole leak in the service water header in the engineered safeguards building, Unit 3 personnel performed a comprehensive inspection of susceptible areas of piping. Pipe replacements were made as necessary with a high regard for preserving service water system integrity. To maintain system integrity, the licensee is developing a service water system monitoring program; the inspectors concluded that this was a good initiative.

Engineering and Technical Support

The licensee completed a review of the auxiliary building filter system design, equipment, and testing problems when the plant was shut down for this issue. Several design and equipment issues which necessitated system modifications were found. Once the full scope of the problems was identified, the licensee initiated a comprehensive effort to resolve the problems.

Inspector concerns were identified with the implementation of motor-operated valve upgrades in Unit 2. These concerns included the quality of the technical basis for operability of certain valves and the timeliness of completion of operability/reportability evaluations. These issues were unresolved pending further review by NRC.

Safety Assessment/Quality Verification

The inspectors reviewed the licensee's report of the discovery that the Unit 1 service water (SW) and emergency service water (ESW) systems would not be able to meet design loads occurring during a safe shutdown earthquake. The degraded areas of SW and ESW had experienced corrosion when the piping exterior protective coating wore off. The pipe walls were further thinned during restoration of the rusty areas without evaluation of the effect of the thinning on system stress analyses. This condition was identified during the July/August 1992 maintenance outage and repaired by the installation of reinforcing saddles around the tie-in points of the SW and ESW piping. The identification and repair of these weak areas resulted from an individual inspector's initiative rather than a defined erosion/corrosion program. Unit 1 has committed to the development of a comprehensive SW and ESW maintenance and inspection plan.

A failure to report degraded Unit 1 fire barriers was of minor significance and the licensee committed to submit the required reports. Accordingly, enforcement discretion was exercised.

Following licensee identification of dimensional nonconformances on Unit 3 reactor head stud replacements, the licensee adequately addressed the nonconforming conditions. In addition, the licensee caused the vendor to make changes to its program for controlling subcontractors and revised their program for auditing vendor controls on subcontractors. However, the inspectors concluded that the licensee may have resolved the matter sooner had their review of three prior NRC information notices not been so narrowly focused.

In following corrective actions for previous violations at Units 1 and 2, the inspector noted discrepancies in the licensee's implementation of committed corrective actions. In one case, the licensee committed to review a matter with operators at the Millstone and Haddam Neck sites. These actions were completed satisfactorily at Millstone, but Haddam Neck had not been informed of this commitment. In the second case, efforts to improve the timeliness of processing plant information reports have not been implemented effectively by plant management. In particular, NRC noted that licensee performance in this area was inconsistent with recent management emphasis on procedural adherence.

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The inspection procedures (IP) from NRC Manual Chapter 2515, Light Water Reactor Inspection Program, that were used as guidance are listed parenthetically for each report section.

DETAILS

1.0 PRINCIPALS CONTACTED

Within this report period, interviews and discussions were conducted with members of the licensee's management and staff as necessary to support the inspection.

2.0 SUMMARY OF FACILITY ACTIVITIES

During this report period, with the exception of small power reductions to conduct turbine stop valve, and main steam isolation valve testing, Millstone Unit 1 remained at 100% of rated thermal power. On October 2, 1992, the licensee reported to the NRC that, because of excessive wall thinning caused by exterior corrosion, the service water and emergency service water systems had been inoperable prior to their repair during the July-August 1992 maintenance outage. NRC review and followup to this issue is discussed further in this report. During the week of September 21-25 1992, the NRC administered requalification examinations to fifteen operators. Five operators failed the examination and the Unit 1 requalification program was declared unsatisfactory. On October 7, 1992, the licensee met with NRC management to discuss the operator examination failures and the basis for continued plant operation; and to outline a corrective action plan to restore the operator requalification program to a satisfactory status. On October 9, 1992, the NRC issued Confirmatory Action Letter 1-92-014 to the licensee which outlined the licensee's commitments for corrective actions, and indicated that additional inspection of the training program would be conducted.

Millstone Unit 2 remained shutdown and defueled throughout the inspection period. Major outage activities included installation, welding, and nondestructive testing of the new steam generators; replacement of vital 120 volt ac inverters; diagnostic testing of motor-operated valves; overhaul and series conversion of the 'A' emergency diesel generator; and modification of the engineered safety features actuation system.

Millstone Unit 3 entered the report period at 100% power. On September 29 the licensee declared an unusual event and commenced a plant shut down after identification that both trains of the supplementary leak collection and release system (SLCRS) were inoperable. The unusual event was terminated on October 1 after the plant reached cold shutdown, a condition in which the SLCRS is not required to be operable. The plant remained in cold shutdown through the end of this inspection to modify the auxiliary building filter system, which is required to support SLCRS operation.

3.0 PLANT OPERATIONS (IP 71707, 71710, 93702)

3.1 Operational Safety Verification (All Units)

The inspectors performed selective examinations of control room activities, operability of engineered safety features systems, plant equipment conditions, and problem identification systems. These reviews included attendance at periodic plant meetings and plant tours.

The inspectors made frequent tours of the control room to verify sufficient staffing, operator procedural adherence, operator cognizance of control room alarms and equipment status, conformance with technical specifications, and maintenance of control room logs. The inspectors observed control room operators response to alarms and off-normal conditions. No noteworthy deficiencies or safety concerns in this area were identified during this inspection.

The inspectors verified safety system operability through independent reviews of the following: system configuration, outstanding trouble reports and event reports, and surveillance test results. The selection of safety systems for review was made using risk-based inspection guidance developed by NRC. During system walkdowns, the inspectors made note of equipment condition and tagging. The inspectors performed detailed walkdowns of the Unit 1 core spray system 'B' train, the Unit 2 spent fuel pool purification system, and the Unit 3 safeguards building ventilation system. The system configuration for the Unit 1 core spray system was correct. The results of the Unit 2 and Unit 3 walkdowns are discussed in sections 3.4 and 3.5 of this report, respectively.

During plant tours and system walkdowns, the inspectors verified the use of installed jumpers, bypasses, and lifted leads in accordance with ACP-QA-2.06B, "Jumper Lifted Lead and Bypass Control (NEO 8.05)," Revision 12. The inspector reviewed jumpers at Millstone 2 involving the installation of a temporary spent fuel pool filter, recirculation of the 'A' boric acid storage tank, temporary service water system seismic restraints. Safety evaluations for these temporary modifications were well-written and promptly reviewed by the plant operations review committee. The modifications were reflected properly in the applicable control room drawings.

The accessible portions of plant areas were toured on a regular basis. The inspectors observed plant housekeeping conditions, general equipment conditions, and fire prevention practices. The inspectors also verified proper posting of contaminated, airborne, and radiation areas with respect to boundary identification and locking requirements. Selected aspects of security plan implementation were observed including site access controls, implementation of compensatory measures, and guard force response to alarms and degraded conditions.

3.2 Operation of the Reactor Vessel Level Reference Leg Backfill Modification - Unit 1

In August of 1992, the licensee installed a system in Unit 1 which provided a source of water from the control rod drive system to the bottom of the reference legs of the cold condensate pot reactor vessel level indicating system. The licensee believes that the addition of water into the reference legs of the cold condensate pot system will prevent the buildup of noncondensable gases which could cause large level fluctuations during rapid reactor coolant system depressurizations. Some level fluxuations were noted during the July 4, 1992, plant shut down. The flow of water into the reference legs is controlled by needle valves which maintain the flow at .008 gallons per minute (gpm). Adjustment of the valves is conducted and controlled in the presence of a reactor engineer through use of Instrumentation and Controls Procedure 428C, "Reference Leg Flow Metering Valve Adjustment." NRC inspection of the modification installation and startup testing is documented in NRC inspection report 50-245/92-22.

During the months of August and September 1992, the setting of the 'A' side reference leg needle valve was adjusted twice while the setting of the 'B' needle valve had to be adjusted four times. All six adjustments were in the open direction to compensate for a reduction in flow across the needle valve. On September 10, 1992, the needle valve on the 'B' reference leg backfill system was replaced after three consecutive adjustments. Examination of the 'B' needle valve revealed that a brown film on the needle valve stem caused the gradual reduction in flow. At the close of the report period, the licensee had not determined the material composition of the film or why the 'B' needle valve required more adjustment than the 'A' valve. However, foreign material in the tubing to the 'B' needle valve that was not removed by flushing during installation, is suspected as the cause. The 'B' valve would be more susceptible to this type of failure mechanism since its has a longer run of tubing.

While reviewing the operation of the modification, the inspector noted that the safety assessment that was performed for installation of the modification was based upon a maximum flow rate through the needle valve of .0085 gpm. The inspector was concerned that if the licensee continued to adjust the needle valve as the flow rate decreased due to film buildup, and the blockage suddenly dislodged, the flow rate into the reference leg would be greater than the .0085 gpm analyzed amount. To address the inspector's concern, the Unit 1 engineering manager issued a memorandum to the supervisor of the Unit 1 reactor engineers which instructed him to limit operation of the valve to values which would be bounded by the safety analyses. This memorandum limiting operation of the valve provides interim guidance until additional analysis can be completed to allow for a potentially higher flow rate. The inspector considered this action to be an acceptable resolution of this concern.

3.3 New Fuel Storage in the Spent Fuel Pool - Unit 2

The inspector witnessed transfer of several new fuel assemblies from the new fuel storage racks to the spent fuel pool. Through review of documents the inspector verified that the

new fuel elevator, cables, slings, and cranes had been tested in accordance with applicable requirements. Movement of the fuel assemblies into Region B of the spent fuel pool was controlled properly using station material transfer form SF-1422. Appropriate cleanliness controls were implemented during the evolution.

The inspector verified through system walkdowns and review of completed surveillances that applicable technical specification (TS) requirements were satisfied. These included:

- Spent fuel pool bulk temperature was less than 140°F (TS 3.9.3.2).
- A minimum of 23 feet of water was maintained over the top of fuel assemblies seated in pool storage racks (TS 3.9.12).
- At least two spent fuel pool area radiation monitors were operable and one train of enclosure building filtration system was in operation in the auxiliary exhaust mode (TS 3.9.13 and 3.9.15).
- Pool boron concentration was greater than 800 parts per million (TS 3.9.17).
- The storage pattern of Region B of the spent fuel pool was controlled by properly positioned blocking devices to maintain Keff less than or equal to 0.95 (TS 3.9.12.2).

The inspector noted that since the new fuel inspection machine was not operable, certain inspection steps of procedure EN-21028, "New Fuel Assembly and Control Element Assembly Receipt and Inspection," had not been performed for some of the fuel assemblies. The licensee dispositioned these items as acceptable in nonconformance report 292-660. Since detailed fuel inspections were conducted prior to shipment of the new fuel and there was no evidence of shipment damage, the licensee considered the particular waived inspection steps to be redundant and the fuel assemblies acceptable for use as is. The inspector reviewed the report and had no concerns regarding the disposition of the items. The inspector concluded that the evolution had been conducted safely and in accordance with applicable technical specifications and procedures.

3.4 Spent Fuel Pool Cooling and Purification System Walkdown - Unit 2

The inspector performed a detailed walkdown of the Millstone 2 spent fuel pool cooling and purification system using as guides the system operating procedure OP-2305, "Spent Fuel Pool Cooling and Purification System," and associated valve lineup check sheets, the final safety analysis report, and the control room system drawings. Material conditions were acceptable. A caution tag dated 1991 on the local power switch for the 'A' cooling pump indicated a long-standing oil leak, and some oil was present in the drip tray. The inspector did not consider the amount of oil present sufficient to constitute a fire hazard. An automated work order to overhaul the pump is scheduled for completion during the present refueling outage. The inspector also noted a small body-to-bonnet leak on discharge header check valve 2-RW-8. This discrepancy was corrected promptly. The inspector verified that system flow, pressure, temperature, and differential pressure instruments were calibrated properly and that all operating parameters were within the limits specified by the operating procedure. The inspector found that while the valve lineup check sheets were consistent with

the procedure for placing the system into operation, they were not consistent with the control room "operations critical" drawing. For example, system makeup valve 2-PMW-167 is closed by the procedure, but is shown open on the drawing. The licensee stated that system operation is controlled by the procedure rather than the drawing. Finally, the inspector found that the system drawing in the updated final safety analysis report (FSAR) had been superseded by three subsequent revisions. The licensee committed to correct the discrepancy in the next scheduled (FSAR) update.

3.5 Safeguards Building Ventilation System Walkdown - Unit 3

The inspector performed a review of the Unit 3 engineered safeguards features (ESF) building emergency ventilation system on September 18 to verify system operability. The inspection included a review of system alignment, equipment condition, associated surveillances, and a comparison of the plant system drawings to the as-built configuration.

The ESF building ventilation system is composed of five safety-related subsystems designed to maintain the residual heat removal, safety injection, quench spray, containment recirculation, and the auxiliary feedwater pump area temperatures suitable for equipment operation and personnel access. Each subsystem is designed to automatically start and limit the temperature in the applicable area to the maximum allowed by design whenever any of the safety-related pumps within their respective area starts.

The inspector noted that the systems were properly aligned and appeared to be in good physical condition. However, four minor deficient items were identified: the condenser discharge piping appeared to be rusty and sitting in standing water; there were a number of valves with label plates missing or inconsistencies between piping and instrument diagram (P&ID) labels versus plant labels; the P&ID did not show the hot gas bypass valve or the correct arrangement of the condenser vent line; and the P&ID indicated that the compressor inlet and outlet valves were to be locked open while the system lineup specified in operating procedure OP 3314D, "Engineered Safeguards Feature Building Ventilation and Air Conditioning," indicated the required position as open. The inspector informed the operations supervisor of these issues; he indicated that they would be addressed.

At the end of this inspection, the licensee informed the inspector that the P&ID would be changed to incorporate the inspector's findings and that the engineering department will address the maintenance issue of the rusty pipe. The licensee also indicated that the missing valve label plates had been replaced, and that the inconsistent valve labeling issue would be reviewed to determine the correct valve identification and ensure accurate plant equipment labeling. The operations department has turned these issues over to the engineering department for resolution. The engineering department has issued an assignment tracking sheet for these issues, however no action to resolve them had been taken by the end of this inspection.

The inspector concluded that the inconsistency between the P&ID and the plant labels was of minor safety significance since the valve lineup and label plate identifications were identical and proper positioning of the system was achieved. Although some valves were not identified on the P&ID, the valves were listed in the system valve lineup and were in the correct position for proper system operation. The inspector considered the dispositioning of these items to be appropriate. The inspector also noted that system surveillance and operating procedures for the safety-related equipment in the ESF building did not require verification of its respective ventilation unit operation. Without verification of satisfactory ventilation system operation, the basis for operability of this necessary support system to various ESF components could not be determined. The inspector discussed this concern with the operations supervisor and indicated that this issue remained **unresolved** pending clarification of the basis for operability of support systems (**UNR 50-423/92-24-01**).

3.6 Supplementary Leak Collection and Release System Inoperability - Unit 3

On September 29, 1992, the licensee determined that both trains of the supplementary leak collection and release system (SLCRS) were inoperable and initiated a technical specification (TS) required shutdown. The licensee declared an unusual event and made all the required offsite notifications.

The SLCRS is designed to ensure that during a loss of coolant accident any containment leakage into the enclosure building and contiguous buildings will be filtered prior to discharge to the atmosphere. The auxiliary building filter system (ABFS) is designed to control the release of radioactive material from the area of the charging pump and reactor plant component cooling water pumps and heat exchangers in the auxiliary building by directing releases through a filtered path and assisting SLCRS in maintaining a negative pressure within the secondary enclosure around containment. The two systems work together to draw and maintain a 0.25 inch water gage vacuum within one minute inside the enclosure and auxiliary building.

As part of ongoing investigations regarding the ABFS design basis and its interaction with the SLCRS (see NRC inspection report 50-423/92-23) and in response to inspector concerns regarding failure to perform surveillance tests in a configuration which demonstrates the ability to perform as designed following an accident, the licensee performed numerous inservice tests. Inservice Test (IST) 3-92-019, "Auxiliary Building Filter Test," was performed to determine ABFS instrument behavior and to identify parameters which contribute to unreliable fan performance with the fan variable inlet vanes in automatic. Test data identified that the 'B' train ABF filter fan takes approximately 75 seconds to come on line due to time delay settings in the system logic. Due to the SLCRS dependency on the ABFS to assist in drawing the 0.25 inch vacuum, the licensee concluded that the 'B' train of SLCRS would be unable to draw the required vacuum within one minute and that it was questionable whether the 'A' train of SLCRS/ABFS would be able to meet this design requirement. Previous testing had demonstrated that the SLCRS was unable to draw the required 0.25 inch vacuum without the assistance of the ABFS fans. Based on this

information, the licensee concluded that the 'B' train of SLCRS would not perform its design function and was therefore inoperable. Additionally, the licensee determined that an adequate surveillance of the 'A' train SLCRS capability to draw a vacuum had not been performed as required by TS 4.6.6.1.d(3). Accordingly, on September 29 at 12:45 p.m., the licensee entered two TS action statements; TS 3.6.6.1 for one train (B) of SLCRS inoperable and TS 3.0.3/4.0.3 for failure to perform a surveillance on the 'A' train within the required interval.

On September 29 the licensee performed IST 3-92-020, "'A' Train Drawdown From Sequencer Panel," to verify satisfactory drawdown capability of the 'A' train of the SLCRS and ABFS when sequenced onto the emergency diesel generator as they would be in the event of a safety injection signal with a concurrent loss of power. Test results indicated that the 'A' train of SLCRS and ABFS could not meet the TS requirement of 0.25 in. wg vacuum within the required time. The system was able to draw the required vacuum after approximately 80 seconds of operation. The 'A' train of SLCRS was also declared inoperable at 11:47 p.m. following the unacceptable drawdown test. The licensee initiated a plant shutdown in accordance with TS 3.0.3. The licensee declared an unusual event and made all the required offsite notifications. The unusual event was terminated on October 1 at 8:14 a.m. after the plant reached cold shutdown, a condition in which SLCRS operability is not required.

This issue regarding ABFS and SLCRS operability and the adequacy of surveillance testing performed on these systems is the subject of NRC inspection report 50-423/92-23.

4.0 MAINTENANCE (IP 62703)

The inspectors observed and reviewed selected portions of preventive and corrective maintenance to verify compliance with regulations, use of administrative control procedures and appropriate 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 inspectors reviewed portions of the following work activities:

- AWO M3-92-18903, Inspect and verify damper position of 3HVR*AOD 40B (SLCRS Boundary)
- AWO M3-92-16073, MEQ, 3 year required maintenance - neoprene seal inspection and verify damper position of 3HVR*AOD 33B (SLCRS Boundary)
- AWO M1-92-10152, Restore oil in 'A' ESW strainer
- AWO M2-90-15619, Inspect and calibrate instruments (spent fuel pool temperature)

- AWO M2-92-10849, Retest engineered safeguards actuation system modules

- AWO M2-92-07260, Overhaul 'B' enclosure building filtration system fan motor

Maintenance activities were observed to be good; the following observations were made.

4.1 Emergency Service Water Strainer Declared Inoperable - Unit 1

Millstone Unit 1 has two redundant Emergency Service Water (ESW) system trains which supply cooling water to the low pressure coolant injection (LPCI) system heat exchangers. Each ESW system has a rotating self cleaning strainer which keeps debris out of the individual LPCI heat exchangers. On September 16, 1992, while performing preventative maintenance checks on the 'A' train ESW system strainer at Millstone Unit 1, maintenance personnel noted that the strainer gearbox assembly did not contain lubricating oil. Maintenance restored the level in the gear box by adding six quarts of oil and tested the strainer for galling by verification of proper motor amperage. The abnormally low oil level was documented by plant incident report (PIR) 1-92-137.

On September 18, 1992, the licensee inspected the gearbox assembly of the 'A' ESW strainer. During the inspection, small metal fines were noticed in the gearbox oil; however, no significant wear was identified on the worm or drive gear assemblies. The gearbox assembly was subsequently removed and replaced with a spare assembly.

The gearbox assembly of concern was installed on August 26, 1992, when the previously installed ESW system strainer tripped on thermal overload during a surveillance test. The strainer motor trip and other issues concerning strainer procurement/installation were discussed in NRC inspection report 50-245/92-22. During installation of the strainer assembly, the lube oil level in the gearbox was not checked prior to declaring the strainer operable. According to the strainer manufacturer, without adequate lubricating oil in the gearbox assembly, sustained operation of the strainer during an accident event could not be assured. Therefore, the 'A' train of the ESW system was potentially inoperable from the time of the gearbox installation on August 26, 1992, until replenishment of the oil on September 16. Since this time period exceeded the seven day out-of-service time allowed in Unit 1 Technical Specification (TS) 3.5, "Containment Core Cooling Subsystems," the licensee reported the TS violation in Licensee Event Report (LER) 92-24, dated October 16, 1992.

The strainer assembly of concern was ordered from the manufacturer during the Summer 1991 refuel outage to replace an old strainer body which failed a hydrostatic test. According to the licensee engineers, the strainer gearbox assembly was shipped from the manufacturer with oil. However, prior to installing the strainer body, the licensee elected to utilize the old strainer gearbox assembly. Accordingly, maintenance personnel drained the oil from the new gearbox assembly, removed the gearbox from the strainer element and installed the old drive assembly. This modified strainer body was then installed in the ESW system in August

1991. The unused gearbox assembly was left in the maintenance shop until August 1992 when it was placed on a spare strainer element and installed in the ESW system.

To prevent recurrence of the event, the licensee revised the procedures governing the repair and installation of the ESW strainers, to require verification of proper lubricating oil prior to declaring the component operational. All procedures governing the repair of rotating equipment will be reviewed to ensure lubrication is verified prior to work completion. Additionally, because of the numerous component operability issues which arose during the strainer replacement activity, the licensee will conduct an overall review of the plant design change process and how it was followed during the strainer replacement activities. This review will be accomplished to determine if any generic deficiencies exist in the design change process.

NRC Review

Several program weaknesses and personnel performance issues, which were not documented in LER 92-24, became evident as a result of this event. The first error occurred when Unit 1 maintenance personnel who drained the oil out of the assembly in 1991, failed to document this action under the description of work section of automated work order AWO M1-91-09985 which installed the original strainer drive assembly on the new strainer element. Therefore, maintenance personnel who may have referred to that AWO in August 1992, when installing the new gearbox assembly on the strainer body would not be alerted to the fact that oil had been drained from the gearbox.

When the strainer gearbox was removed from the strainer element, the Unit 1 maintenance department left the gearbox assembly in the maintenance shop for an extended period of time. Based upon this event the inspector was concerned that licensee personnel were not adequately controlling QA category 1 material once it was issued from the store room and that material issued for previous work was being used for new work orders without proper re-certification of the applicability and material condition of the locally stored component.

Additionally, the Millstone work control processes do not require specifically that personnel verify that the applicable preventative maintenance actions have been performed on components prior to installation. If preventative maintenance checks were performed on the strainer gearbox prior to declaring the system operable, the low lube oil level would have been detected.

At the close of the report period, the licensee had not completed its investigation of the event, closed the PIR, or developed a final corrective action plan. Therefore, the inspector could not determine if adequate corrective action will be taken to warrant the use of enforcement discretion for this issue. Accordingly, **unresolved item 50-245/92-25-02** will be opened until the inspector has reviewed the licensee's final corrective actions and determined if enforcement discretion is warranted for the violation of TS 3.5.

4.2 Enclosure Building Filtration System Retest - Unit 2

During a routine walkdown of the auxiliary building on October 14, 1992, the inspector identified air leakage at the discharge of the 'B' train enclosure building filtration system (EBFS) fan. Technical specifications require the EBFS to be operable during fuel movement or crane operation over the spent fuel pool. Also, the system must be able to draw a negative pressure in the area when the irradiated fuel in the pool has decayed for sixty days or less. Whenever irradiated fuel is in the fuel pool and the EBFS automatic initiation function is inoperable, the system must be operating in the auxiliary exhaust mode. At the time of this walkdown, new fuel was being moved into the spent fuel pool, all of the irradiated fuel had decayed for more than 60 days, and the EBFS was lined up to ventilate that area, operating continuously in the auxiliary exhaust mode because an out-of-service radiation detector rendered the automatic initiation function inoperable.

Immediately upon being informed of the inspector's finding, the licensee investigated, found more leakage around the fan housing, and initiated a plant incident report to track resolution of the problem. A test was performed which verified that the technical specification flow requirement of 9000 +/- 10% cfm was satisfied. Also, temporary repairs were made to the ductwork until the fan could be taken out of service for permanent repairs. The inspector found these short term corrective actions to be acceptable.

The inspector reviewed the maintenance retest requirements of administrative procedures ACP-QA-2.02B, "Retest," and ACP-QA-2.02C, "Work Orders," and the automated work order packages (AWO M2-92-07259 and M2-92-07260) associated with the overhaul of the 'B' EBFS fan and motor in late July 1992. Regarding post-maintenance retests, the ACPs require that tests be proposed by the appropriate department head, documented on the AWO, and reviewed and concurred with by the operations shift supervisor or supervisory control operator. If the retest is performed as part of an approved surveillance procedure, the appropriate data sheets are to be appended to the AWO package and forwarded to the document control facility. This provides an auditable trail of the successful completion of the maintenance activity. At Millstone 2, the procedure MP-2701X, "Maintenance Retest Guidelines," implements the requirements of the ACPs. Form 2701X-3, "Miscellaneous Equipment Retest Matrix," for fans requires the performance of operational checks to verify component operability, vibration monitoring, and a check of motor running current. These items were documented on the AWO. The inspector noted that item two of the matrix, which requires a check of mechanical joints for leakage, is not applicable to fans. The licensee indicated to the inspector that the adequacy of the retest matrix for fans would be reviewed.

The inspector found that surveillance procedure SP-2609D-1, "Enclosure Building Filtration System," had been performed satisfactorily on August 6 prior to placing the system into service. The surveillance tests system flow and differential pressure across the combined high efficiency filters and charcoal absorber. In this case, previous testing of the EBFS system in the auxiliary exhaust mode had demonstrated the importance of eliminating leaks

from system ductwork and across dampers, and that the existence of required system flow alone did not demonstrate operability of the system. No verification of fan boundary integrity had been specified or performed. Additionally, the completed surveillance forms were not appended to the work package, nor was the work order annotated to indicate that the surveillance had been performed.

Through review of other work packages during the outage and discussions with Unit 2 operations and maintenance personnel, the inspector considered this occurrence to have been an isolated case. Nonetheless, failure to perform an adequate post-maintenance retest of the fan and establish an auditable trail tying the retest to the maintenance activity is a violation of Millstone procedures and 10 CFR 50, Appendix B, Criteria XI, "Test Control," and XVII, "Quality Assurance Records." Similar problems have been noted recently on the Unit 3 hydrogen recombiner system (NRC Inspection 50-423/92-23). As this issue had minor safety significance, enforcement discretion was exercised in accordance with section VII.B of the NRC Enforcement Policy. However, licensee review and upgrade of the work control process including retest and record keeping remains **unresolved** under the formerly issued identification number **(50-423/92-23-002)**.

4.3 Service Water Piping Degradation - Unit 3

On October 5, 1992, while performing a leak test of the 'A' train safety injection pump cooling water (CCI) system the licensee identified a pinhole leak on the service water (SW) header in the engineered safeguards feature (ESF) building. The SW system supplies cooling water for heat removal for various components/systems. The SW to the ESF building supplies cooli water to the air conditioning (AC) units in the ESF building and the CCI system heat ex angers. The ESF building AC units start whenever any safety related pump within their respective area starts to provide a suitable environment for equipment operation. The CCI system provides cooling for the safety injection pump lube oil coolers. The licensee determined that the leak was caused by erosion/corrosion.

As part of the investigation, the licensee removed the lagging from all the SW header fittings in the ESF building and visually inspected them for indication of system degradation. Of the fittings inspected the licensee identified one leak and four areas which had indication of salt deposits. The licensee repaired the leaking joint and performed ultrasonic testing (UT) of the other potential problem areas as well as ten additional areas to determine if there was any indication of wall thinning and to evaluate overall system condition. Of the fourteen areas inspected the licensee found four indications of system degradation. All but one of the piping areas measured were significantly above the minimum required wall thickness. The licensee repaired one of the degraded areas and processed non-conformance reports (NCRs) to document the deficiencies for the other three areas. One problem was determined to be the result of piping outside diameter defects. The other two NCRs documented piping degradation. In addition to inspecting the ESF building SW header, the licensee visually inspected the lagged and exposed areas of the small bore piping in the ESF building, the SW

header to the control building AC water chillers, and the auxiliary building SW supply header to the charging pump coolers, and the motor control center and rod control area ventilation units.

The licensee determined that the mechanism causing the wall thinning was erosion/corrosion impingement attack. This is a highly localized form of corrosion caused by high local turbulence downstream of elbows, tees, and valves. This corrosion typically affects only a few inches of pipe length. Since 1987, the SW system has experienced chronic minor leaks. The licensee has determined that many of the leaks have occurred in the small/medium bore copper/nickel pipe. The leaks appear near pipe tees, elbows, or reducers. Where possible, the licensee has replaced elbows with wide radius sweeps to decrease erosion. The licensee also replaced some copper/nickel piping with monel which has a higher corrosion resistance. The licensee informed the inspector that they are in the process of developing a monitoring program for inspecting the SW system.

The inspector visually inspected portions of the SW header in the ESF building and reviewed the discrepancies identified by the licensee. The inspector concluded that the scope of inspection performed by the licensee was comprehensive and demonstrated a high regard for preserving service water system integrity.

5.0 SURVEILLANCE (IP 61726)

The inspectors observed and reviewed selected portions of surveillance tests, and reviewed test data, to verify compliance with: procedures; technical specification limiting conditions for operation; removal and restoration of equipment; and, review and resolution of test deficiencies. The inspector reviewed portions of the following tests:

- SP 3626.5, Service Water Pump 3SWP*P1B Operational Readiness Test
- SP 3440A01, Plant Startup Surveillance, Operational Test of Source Range Neutron Flux Reactor Trip
- IC 412G, Break Detection Valve Permissive Functional and Calibration Test
- SP 611.1, Main Steam Isolation Valve Closure Functional Test
- SP 611.3, Main Steam Isolation Valve Closure Scram Functional Test
- SP 668.2, Gas Turbine Emergency Fast Start Test
- IC 2435A, Balance of Plant Instrument Calibration - Operating (Spent Fuel Pool Temperature)

- SP 2402AO, Spent Fuel Pool Area Radiation Monitors Calibraticn

- SP 2404AN, Spent Fuel Pool Area Radiation Monitors Functional Test

Surveillance activities were observed to be good; the following observation was made.

5.1 Area Radiation Monitor Calibration - Unit 2

The inspector witnessed a spent fuel pooi area radiation monitor logic function test conducted in accordance with portions of the system calibration procedures SP 2404A0 and SP 2404AN. The calibration constituted a retest of work performed on associated engineered safety features actuation system (ESFAS) modules. The monitors provide indication of radiaion levels around the spent fuel pool; any two of the four monitors will initiate emergency ventilation of the area in the event of a fuel handling accident. Since the ESFAS actuation cabinets were deenergized for maintenance, the ventilation system was operated continuously in the emergency mode. The inspector verified that the calibration had been authorized properly by the operations shift supervisor; that the procedures in use contained the most recent revisions and changes; and that the test equipment was in calibration. While walking down the system, the inspector found that the detector for the spent fuel pool southwest area radiation monitor (RM 8139) was partially blocked by a portable steel cabinet. The licensee promptly moved the cabinet. None of the remaining three monitors were obstructed. Though the instrument indicated approximately the same as the other area monitors, the inspector was concerned that the cabinet could attenuate area radiation levels at the monitor and result in nonconservative readings. Upon further review by the NRC Region I staff, it was concluded that the presence of a cabinet in front of the radiation monitor would be unlikely to cause a problem with the monitor readings. However, it is a good practice to keep radiation monitors unobstructed. The inspector had no further questions regarding this surveillance activity.

6.0 ENGINEERING/TECHNICAL SUPPORT (IP 37700, 37828)

6.1 Motor-Operated Valve Operability - Unit 2

In July 1992 a partial loss of normal power (LNP) event occurred at Millstone 2. The event is documented in inspection report 50-336/92-22. In response to the event, the licensee reviewed the design of the engineered safety features actuation system and found certain failure modes which could result in inadvertent opening of the pressurizer power-operated relief valves (PORVs). This would require isolation of the PORVs by closing the motor-operated PORV block valves.

The PORVs provide overpressure protection for the reactor coolant system by opening on receipt of a pressurizer high pressure reactor trip signal. The valves are sized to pass the maximum steam surge resulting from a control element assembly withdrawal accident initiated at low power, and to prevent lifting of pressurizer code safety valves on a loss of

load accident from full power operation. Each PORV is provided with a motor-operated block valve to permit isolation in the event of valve leakage. In operating modes 1, 2, and 3, technical specifications require both PORVs and block valves to be operable. Operation with one PORV inoperable is permitted provided that its associated block valve is shut with its power removed within eight hours. Otherwise the plant must be placed in the cold shutdown condition within the following 36 hours.

No credit is taken for PORV operation in the overpressurization accident analysis contained in the Millstone 2 Final Safety Analysis Report. Failure of both PORVs in the open position is an analyzed loss of reactor coolant inventory event with acceptable consequences provided that the charging and/or safety injection systems are operable. However, failure of a PORV to close was a significant contributor to the Three Mile Island (TMI) Unit 2 accident. Item II.D.1 of the NRC TMI Task Action Plan (NUREG-0660) required, in part, that licensees determine expected PORV and block valve operating conditions through use of accident analyses and apply the single failures of those analyses such that the dynamic forces on the valves are maximized. The licensee participated in a PWR owners group program which tested typical block valves and correlated the results to the valves actually installed in Unit 2. The NRC reviewed and approved the licensee's approach to assuring block valve operability in a safety evaluation report and subsequent correspondence issued in 1988. Due to the recent licensee discovery, in response to the LNP event, of risk-significant failure modes involving inadvertent PORV actuation with safety injection systems unavailable, the inspector initiated a detailed review of PORV block valve operability.

In May 1992, the licensee initiated a reportability/operability determination (REF) based on new motor-operated valve (MOV) target thrust calculations which suggested that the block valves would not have been capable of isolating a stuck open PORV under worst case design basis conditions. A preliminary finding in July 1992, supplemented by updates in August and September, concluded that the valves were operable. The inspector reviewed the information relied upon by the licensee to support this conclusion.

NRC Generic Letter (GL) 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance," requires licensees to develop and implement a program to assure that valve motor-operator switch settings for valves in specified systems are selected, set, and maintained so that MOVs will operate under design basis conditions. Millstone 2 has included the PORV block valves in this program.

The licensee contracted Babcock & Wilcox Nuclear Services (BWNS) to perform MOV evaluations using Babcock & Wilcox's (B&W) PC-based motor-operated valve evaluation (MOVE) program. The program determines the thrust required to operate an MOV under design conditions and evaluates the thrust producing capability of the valves. The MOVE program utilizes a standard valve operator sizing formula to determine design thrust requirements and compares it to operator thrust and torque limits, spring pack capability,

motor undervoltage capability, and valve load limits. Design inputs, including valve factor, stem coefficient of friction, and system line and differential pressure, are provided to BWNS by the licensee.

The inspector reviewed the input assumptions for the PORV block valves and the MOVE program results for calculations performed on April 24, July 2, and July 23, 1992, and had the following observations:

- Valve seat diameter was changed from port size to actual seat diameter. This results in a higher thrust requirement and is conservative.
- Use of a stem coefficient of 0.2 results in a higher thrust requirement and is probably conservative.
- Valve diagnostic data (VOTES) indicates that a stem packing load assumption of 1500 pounds is conservative.
- For worst case scenarios, an assumed differential and line pressure of 2485 psig may be conservative.
- A valve factor of 0.3 is assumed. This is the value specified by Limitorque, and was standard industry practice, for solid wedge gate valves.

The MOVE calculations indicated that the PORV block valves would not have developed the required closing thrust without exceeding the thrust and torque limits of the actuator, and that in undervoltage conditions at the motor, the valve would have stalled before torque switch trip. In order to relate the theoretical conclusions to conditions in the field, the inspector compared the licensee's cold, static VOTES data taken in June 1990 for block valve 2-RC-403 and in May 1992 for block valve 2-RC-405 and compared them with design thrust requirements assuming a valve factor of 0.3, a stem coefficient of friction of 0.15, and normal operating pressure of 2250 psig. The inspector found that the switch settings for valve 2-RC-403 provided a margin less than that accounted for by VOTES instrument accuracy and torque switch repeatability. Sufficient thrust was available for valve 2-RC-405, but the valve was over thrusting by approximately 200 pounds. The inspector also noted that, since the calculations use nominal bus voltage, the valves may have stalled under design basis undervoltage conditions. Finally, the inspector noted that the licensee was modifying the valve actuator gear ratio and changing spring packs to provide more thrust capability. The plant design change records state that the modifications are necessary since initial calculation results could not provide reasonable assurance that the valves were capable of performing their design functions. This conclusion was consistent with the inspector's findings and appeared to conflict with the preliminary operability determination in the REF.

Due to the uncertainties associated with operability of the PORV block valves, the inspector reviewed BWNS calculations for the remaining valves evaluated by the licensee during the

current refueling outage. Fifteen other valves in the main steam, auxiliary feedwater, safety injection, and component cooling systems were identified by the MOVE program as being inoperable, and all are being modified to increase thrust capability. Noting that REFs were initiated for only some of these valves, the inspector questioned the basis for this decision. The licensee responded that the other valves were considered operable based on the engineering judgement that input differential and line pressures were overly conservative. The valves selected for evaluation under the REF process represented those where more serious problems were felt to exist. The licensee also stated that for NRC reporting purposes, GL 89-10 analysis results were not applicable to past operating cycles before which time the generic letter was not part of the plant licensing basis. Where operability problems were not found in the past, the licensee found no requirement to evaluate historical operability using the new methodology. The inspector stated that this position appeared to conflict with the NRC position expressed in Supplement 1 to the generic letter, which states that licensees must meet the reporting requirements of 10 CFR 50.72 and 50.73 with regard to identification of deficiencies.

The inspector observed that the BWNS design thrust calculations upon which the licensee's valve modifications were based assumed a valve factor of 0.3. Previous NRC information notices, Generic Letter 89-10, industry experience, and the results of licensee valve testing pursuant to NRC Bulletin 85-03 conflicted with this assumption. In a memorandum dated December 1989, the licensee compared its test results with those of NRC and industry sponsored test programs and concluded that for solid wedge gate valves, a valve factor of 0.6 probably was reasonable. While agreeing that the assumed valve factor is irrelevant for valves which are demonstrated to be operable when tested under design basis conditions, the inspector was concerned that the licensee's current program did not take adequate account of industry experience. For those valves which will not be tested under design basis conditions, the inspector considered that the licensee had not justified adequately the use of a potentially nonconservative valve factor.

The inspector concluded that the PORV block valves may have been inoperable under certain design basis conditions, and questioned the licensee's failure to identify and report this condition to the NRC. The inspector also questioned why the design basis of the block valves remained undetermined in light of a long-standing TMI Action Plan requirement. With regard to the other valves in the generic letter program, the inspector questioned the validity of assuming potentially nonconservative valve factors despite industry experience to the contrary, and was concerned that design basis inputs to the MOVE program appeared not to have been well thought out by the licensee. Finally, the inspector was not satisfied that the licensee had understood and applied NRC reporting requirements to the generic letter program findings.

This item is **unresolved (50-336/92-27-03)** pending the following licensee actions:

- Justification of using a valve factor of 0.3.

- Reconciliation of the apparent conflict between the preliminary operability determination and the implementation of valve modifications.

- Explanation of why the valve design basis, supposedly settled under the TMI Action Item between 1982 and 1988, was still undetermined in 1992.

- Resolution of the reportability of valves found inoperable by GL 89-10 standards.

6.2 Auxiliary Building Filter Operability Determination - Unit 3

Since initial startup the licensee has not fully understood the auxiliary building filter system (ABFS) design basis and operating parameters. In addition, the licensee did not test the ABFS and the supplementary leak collection and release system (SLCRS) in a manner so as to verify the system capability to perform its design function following an accident. This issue is the subject of NRC inspection report 50-423/92-23. At the conclusion of that inspection in mid-September 1992, the licensee's evaluation of the design, operation, and testing of the SLCRS and ABFS was continuing.

The SLCRS is designed to ensure that any containment leakage into the enclosure building and contiguous buildings during a loss of coolant accident will be filtered prior to discharge to the atmosphere. The ABFS is designed to control the release of radioactive material from the area of the charging pump and reactor plant component cooling water pumps and heat exchangers in the auxiliary building by directing these releases through a filtered path and to maintain these areas above 65°F. The ABFS also assists SLCRS in maintaining a negative pressure within the secondary enclosure around the containment building. The two systems work together to draw and maintain a vacuum (0.25 inch water gage) inside the secondary enclosure buildings within 50 seconds of a design basis accident.

As part of the ongoing investigation regarding the ABFS design basis and its interaction with the SLCRS, the licensee conducted system inspections and performed extensive system testing to obtain information regarding SLCRS and ABFS performance. As a result of their investigation, the licensee determined that the ABFS would not perform its designed accident function and that the SLCRS performance has degraded since initial plant startup such that the ABFS must function with SLCRS to meet the accident analysis assumptions for SLCRS performance. Through tests and further evaluation, the licensee determined that the SLCRS/ABFS function was inoperable due to the system degradation and design problems. Both trains of SLCRS were declared inoperable and the plant was shut down on September 29 (see section 3.6 of this report).

As a result of timing delays in the ABF fan circuitry, the required vacuum in the secondary enclosure buildings could not be achieved within the 50 second time requirement. During testing the SLCRS was able to draw approximately 0.20 in vacuum without the assistance of the ABFS. About 80 seconds passed before the 0.25 inch vacuum was achieved with SLCRS and ABFS running. The licensee also duplicated the test conditions during initial startup testing in which filter fan cycling and instability with the ABFS in the automatic mode occurred. The licensee determined that the filter fan instability was caused by low flow to the filter fans and that the filter fans cycled due to problems associated with the automatic starting feature of the differential pressure switch. The licensee concluded that the filter fans cannot operate reliably at flows less than approximately 15,000 cfm. Other problems identified included the need for a fan interlock between the supply and exhaust fan to ensure that two supply fans will not operate with only one exhaust fan running. This condition would prevent SLCRS/ABFS from providing a negative pressure in the auxiliary building in the event of an accident. The licensee also identified that the ABFS is vulnerable to single failure. A single failure of various components in the ABFS could render both trains inoperable by effecting the capability of the system to maintain greater than 0.25 inch vacuum or exceeding the design basis high/low temperature limits in the affected areas.

At the end of this inspection the licensee was in the process of developing a plant design change record (PDCR) to modify the ABF control circuitry, sequence timing, and control dampers such that the system will be operated in conjunction with SLCRS to draw the required vacuum following an accident signal and maintain the charging and reactor plant component cooling water areas above 65°F. The plant will remain shut down until the modifications are complete and testing is performed to verify system operability.

Safety Significance

As a result of the SLCRS performance degradation and the misoperation of the ABFS, the licensee concluded that in the event of an LOCA, the exposure to the low population zone and the exclusion area boundary would be affected but doses would still remain below the part 100 limit. Assuming a delay in drawdown to a time of up to two minutes, the licensee calculated a total exclusion area boundary dose increase from 150 Rem to approximately 280 Rem (thyroid dose) using ICRP 2 methodology (current licensing basis). If the same conditions are assumed but ICRP 30 were used to determine the increased dose the result would be an increase to about 180 Rem. The calculations using both methodologies assume that an unfiltered radiological release occurs until the 0.25 inch vacuum is achieved. The failure of the ABFS to automatically actuate would have been annunciated on the main control board. Plant operating and emergency procedures specify operator actions to restore system operation, but not within the time frame assumed in the accident analysis. In addition, the licensee has now determined that operator action to start the filter fans may not have been successful due to system instability and flow balancing problems.

Summary and Conclusions

The licensee did not have a full understanding of the system design, equipment, and testing problems until the plant was shut down in September 1992. Once the problems were identified and the plant shut down, the licensee initiated a comprehensive investigation to resolve the concerns. The licensee did identify several necessary design and equipment modifications which are being reviewed as part of the PDCR. The inspectors will continue to follow the licensee's closure of these issues. Resolution of this issue will be followed under items opened in NRC inspection report 50-423/92-23.

7.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (IP 40500, 90712, 92700)

7.1 Review of Written Reports

Periodic Reports and Licensee Event Reports (LERs) were reviewed for root cause and safety significance determinations and the adequacy of corrective action. The inspectors determined whether further information was required and verified that the reporting requirements of 10 CFR 50.73, station administrative and operating procedures, and technical specifications 6.6 and 6.9 had been met. The following reports and LERs were reviewed:

LER 245/92-24	Emergency Service Water Train Rendered Inoperable
LER 423/92-16	Both Trains of the Auxiliary Building Filter System Inoperable
LER 423/92-19	Both Trains of the Hydrogen Recombiner System Inoperable
LER 423/92-20	Both Trains of the Auxiliary Building Filter System Inoperable

Unit 1 Monthly Operating Report covering operation for September 1992, dated October 14, 1992

Unit 2 Monthly Operating Report covering operation for August 1992, dated September 8, 1992

Unit 2 Monthly Operating Report covering operation for September 1992, dated October 8, 1992

Unit 3 Monthly Operating Report covering operation for August 1992, dated September 8, 1992

Unit 3 Monthly Operating Report covering operation for September 1992, dated October 14, 1992

The reports reviewed were accurate; the following observations were made.

7.1.1 Reporting of Fire Barrier Deficiencies

While reviewing Plant Incident Reports (PIRS) which were written during the 1989-1991 time period, the inspector noted that PIR 1-91-25 which reported an unsealed fire barrier penetration in the turbine building was not reported to the NRC via a Licensee Event Report (LER). The unsealed penetration of concern was a longstanding condition that was not sealed during the implementation of the Appendix R program at Unit 1. A review of PIRS written during the 1990 time period identified five other instances where fire barriers required by technical specifications (TS) were found unsealed but not reported to the NRC.

Millstone Unit 1 TS 3.12, "Penetration Fire Barriers," requires all fire barriers which protect safety related components to be operable. If a penetration is not operable, the TS requires a temporary fire barrier to be established of equal effectiveness or a continuous fire watch be established on at least one side of the fire barrier. When the degraded fire barriers were discovered, a continuous fire watch was posted until the discrepant condition was corrected. However, the violations of TS 3.12 were not reported to the NRC as required by 10 CFR 50.73(a)(2)(i)(B) as any operation or condition prohibited by the plant TS. The NRC position that the discovery of a longstanding degraded fire seal was reportable as a violation of TS, was discussed with the Unit 1 engineering manager. The inspector noted that Unit 3 routinely reports longstanding degraded fire barriers when discovered, the most recent example being the issuance of LER 50-423/91-29 dated December 23, 1991.

To prevent recurrence of the event, Unit 1 engineering agreed to disseminate guidance to the engineering and operations departments that longstanding degradations of TS-required fire barriers should be reported to the NRC via an LER. The licensee has committed to review previously unreported fire barrier degradations which were documented in PIRs and to make the necessary reports by November 30, 1992.

The aforementioned reporting inconsistency between Units 1 and 3 is another example of a failure of personnel to share information and communicate issues between individual units. This weakness was identified in the most recent Systematic Assessment of Licensee Performance Report. The inspector noted that the failure to report the degraded fire barriers to the NRC via an LER is of minor significance since a fire watch was posted when the degraded barriers were identified and the barriers were eventually resealed. The inspector determined that the corrective actions taken by the licensee to prevent recurrence of the failure to report were adequate. Therefore, enforcement discretion was exercised in accordance with section VII.B of the NRC Enforcement Policy.

7.1.2 Report of Degraded Service Water and Emergency Service Water Systems - Unit 1

On October 2, 1992, the licensee reported to the NRC that an engineering evaluation concluded that areas of thinned Service Water (SW) and Emergency Service Water (ESW) piping discovered during the July-August outage would not have been able to meet design

loads imposed by a Safe Shutdown Earthquake (SSE). The licensee reported this finding in accordance with 10 CFR 50.72(b)(l)(ii)(B) as a condition that was outside the design basis of the plant.

The SW system at Millstone Unit 1 consists of one common 34 inch header that can be supplied by four pumps. Major cooling loads for the SW system consist of the diesel generator, reactor building component cooling heat exchangers, shutdown cooling heat exchangers, turbine building secondary component cooling heat exchangers, and the turbine building component cooling heat exchanger. The ESW system at Unit 1 consists of two 100% capacity systems which are each supplied by two ESW pumps. Each train supplies cooling water to its respective low pressure coolant injection heat exchanger.

The areas of degraded ESW and SW piping are located underneath the floor of the intake structure. In this structure, one 34 inch SW header and two 20 inch ESW headers cross the individual intake bays. The degraded SW piping concerns the area where the 'A', 'B', 'C', and 'D' SW pump discharge piping ties into the main 34 inch SW header. Each 'T' joint section of piping had thinned from a nominal wall thickness of .375 inches to .260, .254, .194, and .209 inches, respectively. The degraded sections of ESW piping, consisted of the area where the 'A', 'C', and 'D' ESW pumps discharge into the their respective 20 inch headers. Those areas of piping had degraded from .365 inches to .236, 250, and .300 inches, respectively.

As documented in NRC inspection report 50-245/92-16, the pipe thinning was the result of corrosion which occurred when the protective coating wore off the exterior of the main SW and ESW headers and their associated piping tie-in from the SW and ESW pumps. The coating degradation exposed the carbon steel piping to the saltwater environment in the intake bays allowing general corrosion to occur. In the 1985-1991 time period, the licensee commenced a program of restoring the piping in the intake bays by removing the corrosion from the piping and painting the exposed areas. However, the effect of the wall thinning caused by corrosion and grinding and how that thinning affected pipe integrity was not fully assessed on the limiting SW and ESW tie-in points. Specifically, following the restoration effort, licensee engineers measured and evaluated the piping thickness of some discrete areas of the affected piping, but not the T-joints where the minimum wall thickness requirements were most critical. A representative sample of the entire restored area would have been more appropriate. Further, site engineering evaluated the measured wall thinning against thickness criteria derived from the pipe pressure criterion and did not consider the more limiting stresses including joint configuration and seismic stresses. Therefore, the significance of the thinning at the T-joints was not evaluated at that time. In response to another SW pipe integrity concern which caused Unit 1 to shutdown for repairs in July 1992, the licensee conducted augmented inspections designed to assess the current status of previously identified degraded areas. While setting up to monitor a degraded pipe area above the T- joint, the Unit 1 Inservice Test (IST) technician decided to measure the wall thickness of one T-joint

for data gathering purposes. Engineering evaluation of the T-joint data determined that repairs were necessary because of the small margin designed into this high stress joint. The licensee subsequently measured and repaired all of the T-joint connections.

To restore the degraded areas to an operable condition, the licensee installed reinforcing saddles around the tie-in points of the SW and ESW piping; thus redistributing the system stresses and reducing the minimum pipe wall thickness needed to withstand design transients. As a long-term corrective action item, the licensee is investigating the possibility of moving the SW and ESW piping away from the salt environment by installing it at the ground level in the intake structure. The licensee also has committed to develop and implement by February 1993, a comprehensive SW and ESW maintenance and inspection strategy. This strategy would be used to target areas in the SW and ESW systems for additional inspection, maintenance and monitoring based upon past performance, system configuration and industry experience.

To evaluate the significance of the thinning, the licensee contracted with Stone and Webster (S&W) engineering company to analyze the ultrasonic test (UT) data taken on only the SW pipe 'T' joints. Based upon the data supplied, S&W concluded that with the exception of the 'D' SW 'T' joint, all tie-in points would remain operable during normal operating pressure stresses and the stresses of an Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE). However, the ('D') joint was found to have insufficient wall strength (0.209 vs 0.335 inches) to remain operable during a SSE event. Loss of this joint affects the operability of the entire SW system which would prevent normal safe shutdown of the unit. Based upon the conclusions reached by S&W, the licensee conservatively postulated that the ESW systems would also have been rendered inoperable during an SSE event.

NRC Review

During inspection 50-245/92-16, inspector review of this event concluded that there was weak engineering oversight of past maintenance activities which could affect the pipe strength of the SW and ESW systems. Specifically, when maintenance removed the external corrosion from the ESW and SW systems during previous refueling outages, the licensee did not ensure that all areas of the SW and ESW piping still contained adequate wall strength to meet all design requirements. During that inspection, the inspector further concluded that the lack of a systematic approach/program for maintaining the condition of these systems contributed to the degradation of the SW and ESW piping. Specifically, the licensee did not maintain and restore the protective paint on the exterior of the ESW and SW systems before significant wall loss occurred. During this inspection, the inspector determined that no previous instances of significant external corrosion have occurred at the site. The inspector also determined that there are no specific code requirements to establish programs to monitor or evaluate exterior corrosion on piping. Also, periodic volumetric examination of Class 3 systems is not a code requirement. In addition, the identification of the degraded T joint piping during the July 1992 outage was due to good initiative by one technician. The

initiative resulted from the increased licensee inspection and engineering review of service water systems initiated following the plant shutdown to repair another SW pipe degradation.

Nonetheless, chapter 3.7, "Seismic Design," of the Millstone Unit 1 Final Safety Analyses Report states that the SW and ESW systems are designed to remain operable in the event of an OBE of .07g and an SSE of .17g. This is to ensure the systems are available to shutdown of the plant and maintain it in a shutdown condition. Millstone Unit 1 Technical Specifications (TS) 3.5.B, "Core and Containment Cooling Subsystems," and TS 3.5.F., "Minimum Core and Containment Cooling System Availability," require the SW and ESW systems to be operable whenever irradiated fuel is in the reactor vessel. If these TS cannot be met, the reactor is to be placed in cold shutdown within 24 hours. The licensee failed to maintain the SW and ESW systems in a systematic manner. Those systems eventually degraded due to exterior corrosion and were rendered inoperable. The inspector noted that the licensee identified and corrected the deficient condition. Further, the long term SW and ESW maintenance plans should prevent recurrence of the event.

The loss of the ESW and SW systems during a seismic event would significantly reduce the options available to an operator to place and maintain the plant in a safe shutdown condition. Continued reactor cooling using seismically-qualified equipment would be lost. Specifically, the isolation condenser would be available to remove reactor heat. However, the isolation condenser requires make-up water within 30 minutes of actuation and no qualified source of makeup exists. The licensee stated that a qualified flow path from the reactor vessel to the torus via the safety relief valves provides some additional cooling which would allow time for operators to find a makeup source for the isolation condenser.

In accordance with Section VII.B of the Enforcement Policy, the NRC exercised enforcement discretion for the service water system degradation and inability to withstand the SSE because of the licensee's identification and correction of the problem; and the fact that no prior instances of significant external corrosion of this nature nor any required programmatic inspections were expected to identify and resolve this issue sooner.

7.2 Plant Operations Review Committee/Nuclear Review Board - Unit 2

The inspector attended one plant operations review committee (PORC) meeting and a combined PORC/Nuclear Review Board meeting during the inspection period. The PORC meeting agenda included discussion of a plant incident report concerning the failure of a local leak rate test of a containment penetration; approval of a control room ventilation system in-service test, and approval of a plant design change for the 'A' emergency diesel generator. The combined meeting was held to discuss plant design changes proposed to mitigate the consequences of a main steam line break inside the containment and issues involving the July 1992 loss of normal power event. Both committees discharged their functions in accordance with relevant technical specification requirements and, through frank and thorough discussion, demonstrated an appropriate regard for nuclear safety.

7.3 Combustion Engineering Reactor Head Studs

The inspector reviewed the licensee's disposition of discrepant conditions involving the reactor head studs purchased for Unit 3 from Combustion Engineering (CE). Nonconforming conditions were identified at Millstone during receipt inspection in September 1990, but were not identified by CE or their subcontractor, PCI Energy Services (PCI), prior to delivery to Millstone. The Millstone staff subsequently generated five NCRs (NCRs 290-566, 290-575, 290-589, 290-235, and 290-239) that documented these nonconforming conditions discovered during the receipt inspection and subsequent reinspection of the critical dimensions by the vendor.

The inspector reviewed the five NCRs, including the actions taken to resolve the nonconforming conditions. The nonconforming conditions consisted of discrepancies in documentation and dimensions. All 60 of the reactor head studs were dispositioned use-as-is in October 1990. The documentation problems were resolved by receiving corrected documentation from the suppliers. The dimensional problems were dispositioned as use-as-is based on either chasing out internal threads to remove excessive coatings, load testing, running the reactor head nuts over the studs to assure proper fit, or discussions with the supplier about the safety significance of the particular dimensional nonconformance.

Based on this review, the inspector concluded that the licensee adequately addressed the reactor head stud nonconformances.

The inspector also reviewed the actions the licensee took with regard to the performance of CE. The initial action was to change CE's status on the approved suppliers list from "approved" to "conditionally approved." This required anyone who wanted to place an order with CE to contact Procurement Quality Services to determine what the conditions for placing an order with CE were. This included source inspection for all orders for items purchased to the requirements of the ASME Boiler and Pressure Vessel Code (ASME Code). The status of CE has been returned to "approved"; however, the condition of source inspection for ASME Code orders is still in place. The inspector questioned why the source inspection was only applied to ASME Code orders. The licensee responded that ASME Code orders are the only ones for which problems were detected during receipt inspection. The licensee attributed this to the fact that CE was placing suppliers of ASME Code items on the their approved supplier list solely on the basis of the suppliers' ASME certificate. Combustion Engineering was not verifying that the suppliers had a quality assurance (QA) program to control characteristics not covered by the ASME Code, (e.g., dimensions, non-pressure boundary parts) or performing inspections to verify the items supplied met all requirements. The company has since modified their QA program to include tests and inspections to cover characteristics not controlled by the ASME Code.

The inspector noted that the NRC has issued three information notices (IN) that addressed issues similar to those surfaced with CE as above: IN 90-03, "Malfunction of Borg-Warner Bolted Bonnet Check Valves Caused By Failure Of The Swing Arm," IN 88-95, "Inadequate

Procurement Requirements Imposed By Licensees On Vendors," and IN 88-35, "Inadequate Licensee Performed Vendor Audits." The inspector reviewed the licensee's actions taken following receipt of the INs. Each IN was assigned to an individual for review and the need for a response was placed in a tracking system.

The licensee concluded that no action was necessary in response to IN 90-03 because none of the Millstone units had Borg-Warner valves. The licensee did not recognize or address the generic issue raised by the IN concerning the effectiveness of the qualification and oversight of vendors by licensees.

In response to IN 88-95 the licensee concluded that revisions being prepared for two procedures would address the issues raised by the IN. The two procedures being revised addressed the preparation and review of purchase requisitions and the use of commercial grade items in safety related applications. The issue of the qualification and oversight of vendors, which was also raised by the IN, was not addressed by the licensee.

Actions taken in response to IN 88-35 were documented in a memorandum issued in September 1988 that gave a very positive assessment of the licensee's vendor audit process. There was no indication the licensee's review process determined if the problems identified by the IN had or could occur under the licensee's program.

In August 1992, the licensee modified the checklist used to perform audits of vendors to include verification that vendors assure that their sub-suppliers control important attributes for the items supplied. Combustion Engineering had been audited in 1987 and was not due to be audited again until 1990. The inspector observed that if the licensee had fully used the information supplied by the INs to assess their vendor audit programs at the time the INs were issued, that would not have affected this event because of the delay between the notifications and the next triennial audit. However, if the licensee factored the information from the INs into the annual assessment of vendors, the fact that CE's program for controlling sub-suppliers was not adequate may have been discovered prior to the receipt inspection of the reactor head studs.

Based on the changes the licensee recently made to their program for auditing vendors, the inspector had no further questions concerning the program for procuring material or monitoring vendor performance.

7.4 Followup of Previous Inspection Items

7.4.1 Effectiveness of Licensee Receipt Inspection Program

Unresolved item (245/92-12-01) involves three issues: commercial grade electrical connectors installed in nuclear instrumentation, nonconforming commercial grade fasteners accepted by Stone & Webster (SWEC) and transferred to Millstone stores, and a programmatic issue concerning the effectiveness of SWEC's and the licensee's receipt

inspection programs. The inspector reviewed and updated the status of all three issues during this inspection.

Stone & Webster Fastener Materials - Unit 3

An update of the licensee's efforts to resolve the SWEC fastener issue was previously provided in NRC inspection report 50-423/92-16. As noted in that report, the licensee had selected a total of 90 fasteners for tensile testing; 30 from each of the three manufacturers that supplied fasteners. The sample included fasteners with porosity, linear indications, oversized heads, and undercut. During this inspection, the inspector reviewed the test results. With the exception of one cap screw, all of the fasteners tested had acceptable test results. One cap screw manufactured by Bethlehem Steel, identified by the manufacturer's mark BS, only had a tensile strength of 121.3 ksi, compared to a specified minimum of 125 ksi. However, the cap screw did have an acceptable yield strength and when pulled to failure; it failed in the threaded portion, the portion of the fastener expected to fail. Based on these results, the licensee concluded that the nonconforming conditions have not adversely affected the ability of the fasteners to perform their function. The licensee initially stated that the remaining fasteners that were transferred from SWEC to Millstone stores would be scrapped. However, due to ongoing maintenance, the licensee did install some of the fasteners in the units after performing additional inspections and dedicating the fasteners prior to installation. This portion of the unresolved item is considered closed.

Nuclear Instrumentation Electrical Connectors

As noted in NRC inspection report 50-245/92-12, a procurement engineer preparing to order electrical connectors for nuclear instrumentation from General Electric (GE) noted that on the previous order GE had supplied commercial grade connectors when safety-related connectors had been specified. Based on this discovery, the licensee initiated plant incident report (PIR) 1-92-052 to document the problem and initiate a review to determine the cause of this error. To characterize the extent of the problem, the licensee reviewed documentation for purchase orders issued for safety-related items during the 1988 - 1989 time frame, including purchase orders issued to GE. For the GE purchase orders, the licensee limited the scope of the review based on changes in the licensee's program during 1989 and changes GE made in 1989 to their program for handling discrepancies between the purchase order and the items being supplied. In addition, as reported in the inspection report referenced above, the inspector reviewed eight purchase orders from the 1988 - 1989 time frame and found an example of GE supplying commercial grade electrical connectors when safety-related connectors were ordered; this was not identified previously by the licensee.

The licensee's review of the documentation on purchase order 116668 showed that the electrical connectors had been ordered from GE as safety-related for use in nuclear instrumentation. General Electric ordered the connectors from Reuter-Stokes, a subsidiary of GE. The parts were received by the licensee with certificates of conformance that stated that the supplied products were supplied in conformance with the purchase order quality

requirements. At receipt inspection, the licensee identified that the part numbers on the connectors did not match those ordered and issued a nonconformance report (NCR) on December 6, 1988. In response to the NCR, on December 14, 1988, GE sent a facsimile of a memorandum dated November 18, 1988, (previously sent to the purchasing department supervisor at Millstone) that stated that the part numbers had been changed and that the parts provided had the same fit, form, and function as the original parts. Based on the information contained in the facsimile from GE, the licensee accepted the parts.

However, during the review of the facsimile the licensee failed to notice that the November 18, 1988, memo included a statement that Reuter-Stokes needed to know the system where the parts were to be used to assure that the parts were suitable for the application. In addition, the memo also stated that the quotation for the parts was for an unspecified nonsafety-related application.

When the problem was brought to GE's attention in March 1992, GE performed a 10 CFR Part 21 reportability evaluation for the connectors. This evaluation concluded that the problem was not reportable. Their conclusion was based on reviewing the failure modes and concluding that either the failure was in the safe direction or that the failure was very unlikely given the actual system conditions where the parts were installed. Based on the information provided by GE, the licensee concluded that the installed connectors were acceptable. General Electric did conclude that it was an error on their part to complete the order without resolving the discrepancy between the stated request for safety-related connectors and the listing of a nonsafety-related part number in the purchase order. As corrective action, GE instructed their employees to resolve all customer purchase order discrepancies prior to filling the order. In addition, Reuter-Stokes has since revised their certificates of conformance to more clearly note whether the material being supplied is safety-related.

To date the licensee has not determined why engineering personnel were not initially aware of the November 18, 1988, memo informing Millstone that the parts ordered were nonsafety-related prior to the parts being receipt inspected on December 6, 1988. The licensee has also not determined why engineering did not recognize that the parts supplied were nonsafety-related when engineering used the memo as the basis to disposition the parts as use-as-is on December 14, 1988. The licensee's review of this issue is ongoing and has been referred to the Project Services Division for an independent assessment of the root cause of the problem.

This portion of the unresolved item remains open pending NRC evaluation of the licensee's review and corrective action.

Programmatic Receipt Inspection Issues

As a result of the questions raised concerning the effectiveness of the licensee's and SWEC's receipt inspection programs discussed above, the licensee initiated a corrective action request (CAR) 92-04. The purpose of the CAR was to provide reasonable assurance that SWEC's

quality assurance program for Category I, non-engineered items identified nonconforming items and prevented their installation in Unit 3.

To accomplish this, the licensee reviewed SWEC's program for establishing purchase order and receipt inspection requirements. The licensee concluded that appropriate procedures existed to assure the quality of Category I, non-engineered items. To review the implementation of the procedures, the licensee reviewed approximately 4,500 receipt inspection reports (RIR) and selected about 1,000 of these that identified nonconforming conditions for detailed review. From this review, the licensee concluded that SWEC's program was effective in assuring the quality of Category I, non-engineered items and closed the CAR on August 24, 1992.

The inspector reviewed the licensee's analysis and a portion of the background information on which the analysis was based. The background information was segregated by the basic specification under which the item was procured. The inspector reviewed the background information associated with the specifications for installation of electrical items (E350), erection and installation of piping (M968), and instrumentation installation (I943). In addition to the information reviewed by the licensee, the inspector also reviewed the requirements for Category I, non-engineered items in each of the basic specifications mentioned above.

The inspector reviewed 19 RIRs from the background information for items purchased under the electrical installation specification. Thirteen of the RIRs were for fasteners, one was for terminal blocks, and five were for support parts and materials. Several inconsistencies between RIRs for similar items and between the RIRs and the specification requirements were noted. The electrical installation specification required that fasteners purchased as Category I, non-engineered were inspected upon receipt for manufacturing imperfections. The inspector believed that the attribute from Quality Assurance Directive (QAD) 7.7, "Receiving Inspection - General," that covers manufacturing imperfections is fabrication. Of the thirteen RIRs for fasteners, three included the fabrication inspection attribute and ten did not.

Specification E350 requires that the receipt inspection for terminal blocks include a random check to verify proper sizes, lengths, and compatibility. In addition, for terminal blocks subject to radiation, the block insulation material is required to be polysulfone produced by a specific company. The RIR included the attributes of manufacturer's documentation, damage, marking, and cleanliness. The RIR did not include dimensions, fabrication, or material properties.

The inspector reviewed eight RIRs from the background information for items purchased under the piping installation specification. Two of the RIRs were for Category I, Engineered items rather than Category I, non-engineered items. The remaining six RIRs were for items purchased to the quality requirements of Appendix B or Section III of the ASME Boiler and

Pressure Vessel Code. No discrepancies were noted in these RIRs, although it should be noted that the two RIRs for Category I, Engineered should not have been included in the review.

The inspector reviewed five RIRs from the background information for items purchased under the instrumentation installation specification. Three of the RIRs were for fasteners, one was for tubing, and the other was for tube clamps. One of the RIRs for zinc plated fasteners did not include the attribute for coating/preservatives, which should have been inspected according to QAD 7.7. The other RIRs correctly reflected the specification requirements.

With the exception of the discrepant bolts addressed above, their have been no other accepted non-engineered items which have subsequently been found to be non-conforming. Therefore, it appeared that the SWEC receipt inspections had been effective. However, the inspector was concerned that the receipt inspections may not have inspected all the required attributes. Additionally, the inspector concluded that the licensee closed the CAR without adequate justification that the SWEC receipt inspections had been conducted in accordance with QA program requirements.

Licensee review of these concerns identified that SWEC receipt inspections for non-engineered items relied heavily on the experience of the inspector and did not strictly follow QAD 7.7 Specifically, the receipt inspector would decide what needed to be inspected by review of procurement documents. He conducted the inspections and documented the results on a generic checklist. Therefore, any required attribute could have been inspected and documented in another attribute of the inspector's choice.

The SWEC program was not consistent in documenting inspection attributes reviewed and may not always have assured that all inspection requirements delineated in the basic specifications were included in the receipt inspections. However, the licensee's current receipt inspection program is deliberate and controlled, and consistent in the choice of required attributes and the documentation of the result. Since there have been no unacceptable SWEC parts identified in use in the field, the inspector had no further questions in this area. This portion of the unresolved item is closed.

7.4.2 Technical Specification Violations Involving Containment Integrity and Safety System Operability - Unit 2

The first **violation (336/90-22-01)** involved two failures to maintain containment integrity during fuel movement. The first incident occurred when a supervisory control operator opened a steam generator atmospheric dump valve while a steam drum manway was open, thus creating a direct path from the containment to the atmosphere. The valve was opened in spite of a procedure cautionary note and valve tagging instructions. The second incident occurred when containment purge system valves were opened without automatic isolation

capability. Another combined **violation (336/92-04-01 and 336/92-04-02)** involved entry into and operation in mode 4 (Hot Standby) with the required high pressure safety injection system train inoperable.

The licensee attributed both violations to a combination of inadequate administrative controls, personnel error, and lack of attention to detail. Corrective actions included procedure revisions, operator shift turnover process enhancements, control room status board improvements, and operator training. The inspector reviewed these actions and noted in particular that operating procedure OP-2201, "Plant Heatup," and operations department instruction 2-OPS-1.14, "Conduct of Operations," had been strengthened by the addition of independent reviews of system status prior to changing operating modes and following shift turnover. During the present refueling outage, an assessment of plant status in terms of shutdown risk is performed daily. Plant operators have received formal training on teamwork, and human performance, and have attended briefings on lessons learned from previous operational events.

Through attendance at shift turnovers and discussions with plant staff, the inspector observed that operators at Millstone 2 typically perform thorough and professional shift changes and are knowledgeable of plant conditions and system configurations. Since the inspector considered that the licensee's corrective actions addressed the causes of the violations acceptably, these items are **closed**. However, a recent loss of normal power/loss of spent fuel pool level event, attributed to personnel error in the performance of an electrical system tagout, indicated a continuing need for aggressive licensee response to human performance issues. The effectiveness of the licensee's corrective actions will be assessed further by the Millstone resident inspector staff during fuel reload and startup from the present refueling outage.

7.4.3 **Flooding of Feedwater Heater - Unit 2**

This **violation (336/92-14-02)** involved inadvertent flooding of a feedwater heater while personnel were working inside the heater shell. The inspector attributed the incident to the licensee's failure to follow administrative work controls properly prior to releasing the heater for maintenance. The licensee's corrective actions included changes to the operations department instruction regarding system venting and draining; operator briefings on the violation, its causes, and corrective actions; and a commitment to review the incident with operators at the other Millstone units and at the Haddam Neck plant. The inspector considered the licensee's corrective actions to be acceptable and verified that they had been implemented at the Millstone units. However, the inspector found that Haddam Neck was not aware of the issue. The inspector discussed with the licensee the NRC expectation that corrective actions to which the licensee commits will be implemented in a timely fashion. While this item is **closed**, implementation of corrective actions for enforcement issues at the licensee's other nuclear units is **unresolved (336/92-27-04)** pending NRC review of future licensee performance.

7.4.4 Verification of Plant Records Accuracy (TI 2515/115)

This temporary instruction (TI) provided guidance to the inspectors for evaluating the licensee's ability to obtain accurate and complete log readings from both licensed and non-licensed operators. The need for this type of inspection became apparent following the identification by several licensee's that plant records (operator logs) may have been falsified.

In April 1992, in response to industry findings that plant equipment operators (PEOs) were falsifying their rounds log sheets information, the licensee conducted an internal review of PEO performance at the three Millstone units. A management team from the line organization was assembled for this task. The team compared rounds log sheets and security information for every individual performing PEO rounds over a six month period (October 1991 to March 1992) at a preferred frequency of about one shift per month per PEO. Several PEOs were identified with discrepant logs. Details regarding the management team investigation, results, and licensee response are contained in NRC inspection reports 50-245/92-12, 50-336/92-12, 50-423/92-11; and 50-245/92-14, 50-336/92-15, and 50-423/92-14.

An independent review team (IRT) was chartered to determine the root cause and generic implications of this problem and recommend corrective actions. The team was comprised of personnel from each unit and the corporate office with experience in the areas of training, licensing, engineering, operations, and human factors disciplines. The IRT final report was submitted to NRC by a letter dated May 21, 1992. In addition to personnel accountability, the team identified a number of causal factors which influenced PEO performance including: rounds procedures, management expectations, communications, PEO oversight, and training. The IRT made a number of recommendations in the areas of procedures, management expectations, communications, and training to correct this situation. Closure of this matter is pending completion of NRC review of the results of the IRT investigation (see NRC inspection report 50-245/92-14, 50-336/92-15, 50-423/92-14).

At the close of this inspection, the licensee was preparing its response to NRC describing the long-term corrective actions planned to prevent future problems with the accuracy of plant records.

7.4.5 Control Rod Drive Hydraulic Control Unit Valve Out of Position - Unit 1

Violation 245/91-16-01 concerned the failure of operators to restore a charging header isolation valve for a control rod drive hydraulic control unit (HCU) accumulator to the condition required by the plant operating procedure following charging of the accumulator. This incident was similar to an event which occurred six months earlier involving the same valve. Following the first event, a plant incident report (PIR) was generated and the corrective action to prevent reoccurrence was developed to provide procedural enhancement and independent valve verification. However, this corrective action was not implemented prior to the occurrence of the second event. The inspector reviewed the licensee's response

to the Notice of Violation, and commitments to change procedures as described in the licensee's letter dated December 24, 1991.

In response to the initial finding, the licensee took immediate corrective actions to restore the system to its normal configuration. In addition, the Operations Manager issued a memorandum concerning the charging of accumulators which stressed the importance of verifying that the charging isolation header valves are fully opened after charging operations, and instituted requirements for dual verification of the hydraulic control unit valves upon accumulator charging restoration.

Long term corrective action to prevent recurrence of the violation included a change in both the plant equipment operators (PEO) logs and the accumulator charging procedure. The change to the charging procedure will ensure that the charging header isolation valves are fully open upon restoration of a charging evolution by requiring dual verification. To address the failure of corrective action for the first event to prevent the second, the licensee also revised Administrative Control Procedure (ACP) 10.01, "Plant Incident Report," to emphasize the importance of evaluating and closing PIRs in a timely fashion. Specifically, the revised ACP requires a PIR to be investigated in two phases. The first phase, which would be used to detect adverse trends in performance, is to be completed typically within a few days of the event. The second phase, which would develop the corrective action to prevent recurrence of the event, should be completed within 90 days unless an extension was granted by the unit director. Administrative Control Procedure 10.1 also requires a quarterly trend report providing statistics of the number of PIRs initiated and closed during the last quarter and the current backlog. No guidance or restrictions regarding the timeliness of implementation of corrective actions was included in ACP 10.1.

The inspector reviewed the PEO logs, accumulator charging procedure, and ACP 10.1 and verified that the changes had been made. However, while reviewing the PIR backlog, the inspector noted that the intent of ACP 10.01 was not followed. Specifically Units 1 and 2 each had 28 PIRs which had not received a Phase 1 investigation within a few days of the event. In Unit 3, 47 PIRs had not received Phase 1 investigations and 35 PIRs which were greater than 90-days old had not received the required extensions from the unit director.

The inspector noted that the licensee revised the PIR investigation process in late 1991 to quicken the investigative phase of the PIR process, but personnel do not appear to be able to meet the new requirements. Personnel who are performing the Phase 1 and 2 investigations indicated that they are not able to meet these requirements due to the pressure of higher priority tasks. Additionally, department supervision did not emphasize the importance of PIR investigation and closeout in their respective departments.

The timeliness of investigation and closeout of Millstone issues has been an NRC concern particularly at Unit 3. Specific issues include the Millstone Unit 3 investigation of the fouling of the 'B' train of the service water system (NRC inspection report 50-423/91-15), and the failure of fire dampers on the supplementary leak collection and release system

(reference NRC inspection report 50-423/91-26). The licensee recognized this weakness and subsequently developed a corrective action plan for both deficiencies as outlined in letters to the NRC dated October 30, 1991 and March 17, 1992, respectively. These letters also informed the NRC that the PIR investigative process would be revised to require a timely Phase 1 evaluation of an event.

Based upon the inspectors' review, it appeared that the corrective action for the two problems originally cited was not implemented appropriately. Also, the untimely response to PIRs contributed to the second HCU event. In addition, the backlog in the PIR investigative process at all three units as described in ACP 10.01 may be contributing to the procedural adherence issues at Millstone. Specifically, licensee management has emphasized that employees follow procedures as written or change the procedure, yet licensee management allows the PIR backlog to occur without addressing the conflict with ACP 10.01. This process could send conflicting signals to licensee personnel regarding procedural adherence. The corrective actions to **violation 245/91-16-01** have not yet been implemented effectively, this violation remains **open**.

The backlog of PIRs was also identified by the Nuclear Safety Engineering Group (NSE) while conducting an internal evaluation of the PIR process. This was documented in an NSE report issued on September 9, 1992. At the close of the inspection report period, the licensee had not responded to these findings. Therefore, the effectiveness of the corrective actions taken for NSE findings cannot be assessed. In response to the inspectors observations, the Unit 1 engineering manager issued guidelines which stated that Phase 1 investigations should be completed within five working days of the event. Phase 2 investigations should be completed within 60 days of the event. At the end of this inspection, the corrective action plans for Units 2 and 3 had not been developed.

8.0 MANAGEMENT MEETINGS

Periodic meetings were held with station management to discuss inspection findings during the inspection period. Following the inspection, an exit meeting was held on November 24, 1992, to discuss the inspection findings and observations. No proprietary information was covered within the scope of the inspection. No written material regarding the findings of this inspection was given to the licensee during the inspection period.