U.S. NUCLEAR REGULATORY
REGION I

Report No. 50-423/87-24

Docket No. 50-423

License No. NPF-49

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

Facility Name: Millstone Nuclear Power Station, Unit 3

Inspection At: Waterford, Connecticut

Inspection Conducted: November 3, 1987 - December 7, 1987

Inspectors: W. J. Raymond, Senior Resident Inspector
G. S. Barber, Resident Inspector
E. L. Conner, Project Engineer

Reporting Inspector: G. S. Barber, Resident Inspector

Approved by: E. C. McCabe, Jr. 1/4/88
E. C. McCabe, Chief, Reactor Projects Section 1B Date

Inspection Summary: Inspection 50-423/87-24 (11/3/87 - 12/7/87)

Areas Inspected: Routine, unannounced inspection on day and back shifts by the resident inspectors of: outage activities, including operational status reviews and facility tours; foreign object discovered in lower core plate flow hole; abnormal RWST and RCS sodium concentration; ESF and waste disposal building contamination during cavity drain down; mechanical and hydraulic snubber inspection; IE Bulletin 85-03; review of committee activities; physical security; maintenance; surveillance testing; loop stop valve interlock design change for three loop operation; and followup of licensee event reports (LERs). The inspection involved 179 hours.

Results: No violations were identified. Routine reviews of plant activities identified no conditions adverse to safe plant operations. Unresolved issues were identified in the following areas: amendment to allow testing of containment penetration overcurrent devices to NEMA criteria (UNR 87-24-01); licensee root cause identification and followup of RWST and RCS sodium contamination (UNR 87-24-02); inadequate licensee control of boundary valves during reactor cavity drain down led to ESF and Waste Disposal Building Contamination (UNR 87-24-03).

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DETAILS

1.0 Persons Contacted

Inspection findings were discussed periodically with the supervisory and management personnel identified below.

S. Scace, Station Superintendent
C. Clement, Unit Superintendent, Unit 3
J. Harris, Acting Operations Supervisor
R. Rothgeb, Maintenance Supervisor
K. Burton, Staff Assistant to Unit Superintendent
M. Gentry, Engineering Supervisor
L. Fusaro, Assistant Building Services Supervisor
D. McDaniel, Reactor Engineer
R. Satchatello, Health Physics Supervisor
M. Pearson, Operations Assistant

2.0 Summary of Facility Activities

At the beginning of the inspection, the plant was in Mode 5 (cold shutdown) for a refueling outage. Final head detensioning was completed and the reactor v ssel head lift was commenced at 7:25 p.m., 11/13. Cavity fill was commenced at 7:50 p.m., 11/13, and proceeded concurrently with head lift. Mode 6 (refueling) was entered on 11/13 and cavity fill was completed at 1:23 a.m., 11/14. A complete core offload was completed at 12:05 a.m. on 11/21 to allow inspection of the lower core plate and plenum for debris (see Section 7.0). Core reload resumed upon completion of the inspection and retrieval of the debris found (ocking cups from the reactor coolant pumps). Refueling activities were completed at 6:30 a.m., 12/1. Cavity drain down was commenced on 12/1 and finished on 12/2. Head bolt tensioning was in progress at the end of the inspection period as the plant was made ready for the transit to Mode 5. The licensee then announced their decision to extend the outage by 8 weeks for disassembly, inspection and repair of the reactor coolant pumps.

3.0 Review of Outage Activities

Performance of operators and equipment was reviewed to ensure safe operation. The following items required inspector followup.

3.1 Loss of an Emergency Bus 34C During Mode 5

During operations in Mode 5 with decay heat being removed by the "A" RHR train, Emergency Bus 34C de-energized at 10:35 a.m. on November 10, when the Bus 34A/34C tie breaker opened due to an apparent spurious actuation of the 86E lockout relay. Loss of Bus 34C resulted in a momentary loss of RHR cooling from the "A" RHR pump. The "A" diesel generator (DG) started as required and powered Bus

34C. The "A" RHR pump was restarted after the DG sequencer timed out (40 seconds). The "B" RHR pump was off but could have been powered from Bus 34D had it been needed. The normal power supply to bus 34D was available throughout the event. Decay heat removal was successfully reestablished and the inspector noted that RCS temperature increased from 92 degrees F to 94 degrees F during the event. The licensee found that the 86E lockout relay had activated, but there were no conditions evident to cause that lockout. There were no indications (flags) of instantaneous overcurrent, overcurrent, or ground fault on the breaker enclosure. Normal power was reestablished at 11:10 a.m. by closing the Bus 34A/C tie breaker and the "A" diesel generator was returned to a standby status.

The licensee reported the event per 10 CFR 50.72 at 11:05 a.m. Subsequent licensee interviews with personnel in the area at the time of the breaker trip identified that a janitor had banged the breaker relay cabinet with his broom. The shock from the broom handle on the cabinet door appears to have caused momentary connection in contacts (8&10) of the 62AR diesel generator timing relay. Even though the diesel was not running at the time and the protection circuit was not operating, the momentary make-up of contacts 8&10 was sufficient to actuate the 86E lockout relay. The inspector verified the licensee's root cause determination for the event by independent review of the timing and lockout relays and by review of drawings of ESK 5BF and ESK 7J. The subject relay contacts were very light, had a 10 to 15 mil gap, and were very sensitive.

The janitor was counseled on the need for caution when working in areas containing electrical equipment. The licensee reinstated restricted access control to the spaces in the front of the switchgear. These are normally in effect during power operations but were relaxed for the outage.

The inspector verified that barricades with warning signs were returned to use in the switchgear rooms. The licensee reported this item as an LER as required by 10 CFR 50.73(a)(2)(iv). No safety inadequacies were identified. During routine inspection, ability to meet Operating Basis Earthquake (OBE) criteria for the 86E relay will be assessed.

3.2 ESF Room Contamination

On November 11, at 9:50 a.m., during routine surveillance testing with the plant in Mode 6, the "B" Safety Injection/Quench Spray pump room became partially contaminated when reactor water sprayed out of an open vent valve in the hot leg recirculation header. The event occurred as the control room operator shifted the discharge from the operating "A" RHR pump from cold leg injection to hot leg recirculation to verify the operability of check valves in the recirculation path. The operator immediately noted an anomalous system response and returned the valve lineup to cold leg injection in less than a minute. Fast operator action significantly minimized the amount of water that sprayed into the ESF pump room.

The vent valve in the hot leg recirculation header was open because of an abnormal system lineup created by local leak rate testing (LLRT) that was in progress on the system but not active at the time, and which resulted in a portion of the hot leg recirculation piping being drained and vented. The control room operator was unaware of the LLRT test status for the system. The event caused no personnel contamination but equipment and floor areas affected by the spray were contaminated to levels of about 2000 to 4000 dpm/100 square centimeters. There were no adverse effects on plant equipment. Actions were taken to begin decontaminating affected areas.

The licensee concluded that the cause of the event was inadequate communication between shift operations personnel and the LLRT group. Licensee corrective actions included procedure changes and the use of caution stickers on the main control board to improve communications between the operations and testing groups. The resident inspector reviewed the corrective actions for adequacy and had no further questions on this licensee-identified, minor item.

3.3 Boric Acid in Reactor Vessel Stud Holes

The reactor vessel stud holes were found to contain reactor coolant borated to about 2300 parts per million (ppm) when one of the studs was removed at 11:00 p.m. on 11/11. At the time, the licensee was making the transition from Mode 5 to Mode 6. The first of two passes of reactor vessel head stud detensioning was completed at 8:00 a.m. on 11/11. Reactor vessel level was being maintained at 5 inches below the vessel flange. At 9:50 a.m., during surveillance on the hot leg recirculation flow path check valves, a sudden surge of water into the hot leg is postulated to have sloshed water up onto the flange area. Since the head was detensioned, water leaked into the flange area and entered the empty bolt holes when the studs were removed.

By isotopic analysis, the licensee identified the water source as being from the vessel. Water was identified in 46 of 54 stud holes. The licensee flushed the holes with demineralized water to reduce the boron concentration to less than 50 ppm. The licensee then placed the stud holes into wet layup for Mode 6 operations by filling them with demineralized water containing 200 ppm hydrazine and installing plugs to seal the holes from the cavity. The licensee evaluated the potential impact on the vessel flange stud cavity and concluded that, when the stud holes are in wet layup, the resulting pH is greater than 8.0. The licensee stated that accelerated general corrosion from boric acid occurs at pH values less than 5.8 if higher temperatures occur than are experienced during the refueling mode.

The inspector reviewed the event with the licensee. The licensee calculated that a pH of 8.57 is expected at 50 degrees Celsius (50C) when 200 ppm hydrazine is mixed with 100 ppm boron. The expected pH at 25C is 8.97. Actual reactor head temperature was between 25C and 50C since RCS temperature was about 33C (92 degrees Fahrenheit, or 92F).

The inspector reviewed Information Notice (IN) 86-108, Supplement 2 dated 11/19/87, regarding the effects of boric acid corrosion and noted that a corrosion rate of 400 mils/month was expected when an aerated 25% boric acid solution was exposed to a 200F carbon steel environment. Since actual boron concentration was less than 2%, the worst case of corrosion damage to the RV studs could be estimated at less than 1 mil since they were only exposed for 2 hours maximum. The licensee concluded that no significant safety hazard existed. The inspector has no further questions in this area.

3.4 Inoperable Containment Penetration Overcurrent Devices

The licensee declared the instantaneous overcurrent devices for containment equipment controlled by molded case circuit breakers inoperable at 12:45 p.m., 11/18. The licensee performed the required 18-month surveillances on containment penetration molded case circuit breakers and found that the instantaneous overcurrent trips did not trip within 20% of their trip setpoint, as required. The Technical Specifications (TS) tolerance of 20% was derived from revision 5 of Westinghouse Standard TS. Initial testing during the startup test phase met the less restrictive tolerances specified in NEMA AB-2 of +40% and -25% of the trip setpoint. Review of the current circuit breaker trip setpoints shows that the tested breakers meet the NEMA standards.

Based on their conclusion that the NEMA AB-2 criteria are sufficient, the licensee is pursuing a TS amendment to allow testing to NEMA AB-2 criteria. The licensee classified this event in accordance with 10 CFR 50.72 (b)(2)(i) at 12:45 p.m. and reported it via the ENS at 1:20 p.m. NRC Specialist Inspectors will review this item, which is unresolved (UNR 87-24-01) pending determination of resolution of the TS amendment request.

3.5 Minor Burns to Chemistry Contractor During SG Sample

A Nuclear Support Services chemistry technician (chem tech) received second degree burns to his left ear and neck while drawing a Steam Generator (SG) sample. The "B" SG was being sampled to comply with National Pollutant Discharge Elimination System (NPDES) while draining it to Long Island Sound over the course of many hours. A sample taken at 2:00 p.m., November 2 showed a temperature of 120 degrees F. As the draining progressed, there was a smaller volume of water to absorb heat from the tube sheet. When the sample was drawn 10 hours later, the water was hotter and the chem tech involved opened the sample valve too rapidly. A slip joint between two tygon tubes separated, spraying him with hot water. He was treated by the site nurse for second degree burns on 5% of his body and was subsequently transported offsite to L&M Hospital where he was treated and released. The sample that was released was non-radioactive secondary water. The licensee is installed a hard pipe connector with hose clamps to prevent recurrence.

The inspector reviewed the existing sample locations and noted that the licensee planned to install cold SG sampling capability during the current outage. Originally, the installation of SG cold wet layup (WLU) recirculation skids included SG draindown and sample capability. The SG WLU skids have been installed but no longer contain SG draindown and sample capability. The licensee stated that consideration is being given to a permanent sample connection for drain down or clarification of the procedure to ensure adequate protection of personnel. No nuclear safety inadequacies were noted.

3.6 Containment Health Physics Coverage

The inspector toured the Unit 3 containment building on 11/16/87 to review activities in progress and health physics controls. The inspector interviewed health physics technicians controlling activities on the containment 56 ft elevation, and at the health physics check point set up on the 24 ft 6 inch elevation. The inspector's review verified the technicians were cognizant of their duties and that workers complied with radiological controls. The review included consideration of: technician knowledge of work activities, including a job in progress to remove sludge from the steam generators; surveys taken and radiological controls for RWP 2216, Steam Generator Sludge Lancing; worker briefings for radiological hazards, technician briefings and interface with supervisors to obtain information from the daily planning meetings; and familiarities with duties, exposure limits, dose tracking and administrative procedures. The inspector also verified personnel exposures were within the established administrative limits for a sampling of workers signed in on RWP 2216. The inspector concluded that containment health physics activities were well controlled. No inadequacies were identified.

3.7 Reactor Cavity Seal Installation

The licensee installed the reactor cavity seal on November 9, 1987 in preparation for cavity floodup. However, the seal ring did not pass a leak test performed when the space between the dual O-ring gaskets was pressurized per MP3 790 AN. The test required that the space between the gaskets be pressurized to 5 psi with less than 1 psi pressure drop observed after 1 hour. The outer O-rings passed the leak test but the inner rings would not hold the test pressure. The licensee removed the seal ring from the cavity and replaced the O-rings. The old rings were inspected and found to have small irregularities at a few locations along their seating surface. However, retest of the seal after installation of the new O-rings proved unsuccessful.

The seal ring is attached to the cavity floor on the outer edge by 36 bolts. As the bolts are tightened, the redundant O-rings on the outer edge are compressed to form a positive seal. If the bolts are overtightened, a spacer bar on the seal ring will contact the cavity floor and prevent over-compression of the ring. The seal ring also has radial cantilever beams which act as stiffeners and are rigidly attached on the outer edge of the seal plate and free on the inner edge. The cantilever beams are equipped with jacking bolts, one per beam, along the inner edge of the ring. Once the ring is in place and the outer bolts are tightened down, the jacking bolts are tightened to force the inner edge of the seal onto the reactor vessel flange.

The licensee tried to better seat the inner O-rings by tightening the outer bolts, and by tightening down on the jacking bolts to the extent allowable by the design. These efforts failed to seat the seal. Subsequent measurements around the circumference of the seal ring along its inner and outer radii showed the ring was slightly warped (most likely from routine handling). The measurement showed as much as a 3/16th inch difference in the space between the seating surface and the inner radius of the ring, such that the O-ring seals were just making contact with the vessel flange. There was insufficient compression of the O-rings to provide an adequate seal with no water in the cavity.

The licensee developed special test procedure IST 2-87-0?1 to install a local leak rate test rig to pressurize the space between the inner O-rings as the cavity was flooded. The pressure between the rings would be sufficient to prevent water leakage, but small enough so that the rings would not roll prior to seating. The licensee calculated the depth of water needed to provide sufficient force to seat the O-rings and the cavity seal ring. The calculations showed that ring would seal with less than three feet of water in the cavity and that full compression would occur (metal-to-metal contact) with about 11 feet of water. The normal refueling level is 23 feet of water. The licensee concluded that the weight of the water would further act to flatten the ring so as to provide a permanent deflection of the ring to assure a positive seal stays in place for the drain down evolution. The special test procedure contained a safety evaluation that was reviewed and approved by the PURC. That evaluation included monitoring for leakage; the possible effects of leakage on equipment in the annulus around the reactor vessel; and arrangement and possible failure modes of the air pressurization system. The inspector reviewed the licensee's plans and identified no inadequacies.

The refueling cavity was subsequently flooded while implementing IST 3-87-031. No major problems occurred. The licensee estimated that about 250 gallons of water leaked to the vessel annulus during the floodup. The inner O-rings sealed with about 1 foot of water in the cavity. The refueling cavity seal plate provided a leak tight seal between the vessel and the cavity floor that was problem free throughout subsequent refueling operations. The licensee plans to inspect the annulus area during the shutdown to verify no adverse affects occurred from the leakage to the annulus. The licensee's planned actions are acceptable.

The inspector reviewed the licensee's followup actions to resolve the problem. The inspection included interviews with engineering and maintenance personnel; inspection of the seal ring and O-ring gaskets; review of the seal ring design details per Drawings 12179-EV-19A-2; and independent inspector calculations to verify the licensee's conclusion regarding the forces required to seal the ring; review of IST 387-031; and attendance at Plant Operations Review Committee (PORC) meeting 3-87-162 on 11/11/87. The licensee's approach to resolving the problem was found thorough and technically sound, and demonstrated adequate regard for safety. No inadequacies were identified.

4.0 Facility Tours

The resident inspectors observed plant operations, maintenance, surveillance and outage activities during regular and back shift hours to verify safe operating practices and that activities were conducted in accordance with approved procedures. Fuel loading and fuel offloading activities were observed and fuel handling procedures were reviewed to ensure licensee compliance during fuel movement. The back shift inspections included tours made at 7:30 p.m. on 11/16, 12:00 noon on 11/14, and 8:00 a.m. on 11/29. Posting and control of radiation, contamination and high radiation areas was reviewed. The use of personnel monitoring devices and compliance with the radiation work permits (RWPs) was verified. Plant housekeeping controls were observed, including the control of flammable and hazardous materials. No inadequacies were identified.

4.1 Unattended Radioactive Materials Cart

While on a routine tour, the inspector found an unattended Radioactive Materials push cart 20 feet east of the Unit 3 diesel generator building. These carts are used by Building Services personnel to transfer bagged Low Specific Activity (LSA) contaminated trash to the Unit 1 laundry and to other storage locations. This particular cart contained no bagged contaminated trash. However, there were loose paper towels inside the cart. Since the cart was painted yellow and posted as a Radioactive Materials area, the inspector notified Health Physics (HP) personnel. The HP supervisor and a technician responded and observed the situation and stated that the cart was not required to be attended since there was no bagged material inside the cart. The inspector pointed out that, although the dirty paper towels appeared to be uncontaminated trash, they should be treated as contaminated because they were inside the cart. The HP supervisor noted the comment and the HP technician moved the cart to a proper storage area.

The Assistant Building Services supervisor indicated that these carts are routinely used by other than Building Services personnel for various miscellaneous uses. The inspector questioned the licensee about actions to minimize or prevent unauthorized use of Radioactive Material Carts. The licensee stated that he would chain the carts together under lock and key in a posted area of the Waste Disposal Building. The carts are now numbered and are required to be signed out by Building Services and other plant personnel prior to use. While in use the carts will be continuously monitored and attended to minimize the likelihood of spreading contamination. LSA trash picked up by building services personnel not directly accessible at the step off pad will be double bagged and the carts will be wiped down using yellow Maslin cloth and surveyed by HP twice a shift. The inspector had no further questions.

4.2 Post Shutdown Trip of the "A" Emergency Diesel Generator (EDG) Overspeed Device

During log review, the inspector noted that the "A" EDG tripped approximately 10 minutes after shutdown at 2:40 p.m., November 4. The "A" EDG was being run to verify its operability in accordance with Technical Specification (TS) 4.8.1.1.2. It was shutdown after the required 1 hour full load run at 2:30 p.m. Approximately 10 minutes after shutdown, the control room received an "A EDG Not Ready for Auto Start" alarm.

At the time, the plant was in a Reserve Station Service Transformer (RSST) A/B and main switchyard south bus outage. The "B" RSST was out of service. The north switchyard bus and Normal Station Service Transformer (NSST) A/B remained operable and the Limiting Condition for Operation (LCO) action statement was met. (LCO 3.8.1.2 was exited at 5:55 p.m. after the diesel was returned to operable status, before the 8-hour time limit for depressurizing and venting the reactor coolant system was reached.)

A Production Test (PT) Technician (Tech) responded to a trouble call from Operations and noted that the following annunciators were lighted on the local gageboard:

-- Engine Overspeed
-- Diesel Generator Not Ready for Auto Start
-- Diesel Generator Not Reset

The PT Tech visually identified that the overspeed switch was not actuated from the centrifugal arm and that the overspeed switch did not appear to be closed. It appeared to be in the reset position and damage free. Inside the gageboard (3EGS*TBEG1A), the following relays were energized as a result of the overspeed trip:

-- Engine Shutdown Relays (5A & SDR); energized by the Engine Overspeed Relay (EOR) relay.
-- EOR Relay; Engine Overspeed energized (EOS) contact (operated by EOS microswitch).
-- Control Power Relays CF3 & CF4 (energized when 125VDC available).

Without lifting wires, the PT Tech could not check the EOS overspeed microswitch contact status because the EOR relay was sealed in through its own series contact. Since relays CF3 and CF4 were energized, control power was available There was no apparent electrical fault in the control power circuit. When an attempt was made to reset the "A" EDG, it successfully reset, all annunciators cleared, and engine relays returned to their normal position. Operations started the diesel and ran it unloaded for five minutes and then shut the diesel down without an overspeed actuation. After ten minutes, an operator fingered the overspeed switch and duplicated the previously described conditions. An hour later, Operations successfully completed a one-hour diesel surveillance.

The licensee concluded that either the overspeed switch (EOS) was physically disturbed by personnel or a falling object or that the prime mover was still coasting down through critical speeds and vibration actuated the overspeed switch. The latter scenario is not consistent with Operations' description of the event. During the current Millstone 3 refueling outage, PT31458, Millstone 3 Emergency Diesel Logic Test, will be performed on Emergency Diesel A to test proper operation of the diesel engine shutdown scheme. That may provide further indication as to the cause of the overspeed trip.

The licensee interviewed each of the individuals who were in the Emergency Diesel "A" Enclosure during the time the overspeed alarm was received. The interviews indicate no one made contact with the overspeed limit switch or the overspeed relay cabinet on the east end of the engine at the time of the event. In addition, the licensee conducted a walkdown on November 7 to review the activities and location of the individuals present during the event.

The licensee's conclusion from reviewing the troubleshooting activities, the interviews, and the walkdowns conducted is that there was no evidence of tampering or physical contact to cause the over-speed alarm and that the alarm must be spurious or due to reasons unknown. Since there was no evidence of tampering and since the licensee complied with all LCO 3.8.1.2 requirements, this event was non-reportable. The inspector will continue to monitor the licensee's root cause assessment for the event.

4.3 "B" Emergency Diesel Fuel Oil Leak after Satisfactory Completion of LOP Test

The licensee satisfactorily completed the "B" Emergency Diesel Generator (EDG) Loss-of-Power (LOP) Test at 9:24 a.m., 11/29. The test required the de-energization of emergency bus 34D and the verification that the diesel reenergized the bus and sequenced on the required loads without overloading. Diesel generator load was increased to 5.5 megawatts, after completion of the LOP test, for the diesel 24-hour surveillance run. Approximately 2 hours into the test, at 12:23 p.m., a flexible hose for the "B" EDG injector return to the day tank failed, spraying oil inside the diesel building. A security guard discovered the leak, promptly notified the control room, and the EDG was shut down immediately. The site fire brigade was dispatched to the diesel building to assess the damage. Offsite assistance was not requested. The inspector, who was on-site monitoring fuel loading activities, responded to the scene and noted that approximately 250 gallons of fuel oil was sprayed in the northwest corner of the "B" EDG Building. Reports by the fire brigade indicated that the fuel oil in the building was about 2 inches deep in the "B" EDG building's northwest corner. The fuel oil was cleaned up by the fire brigade and an oil absorbent agent was placed on the floor. In addition, the fire brigade cleaned up fuel oil discovered on cable runs located beneath the floor level.

The licensee has replaced the flexible hose with a hard pipe and the diesel has been returned to an operable status. The licensee plans to perform the same modification on the flexible hoses attached to the "A" EDG. The licensee has been investigating EDG vibration to determine the effect on skid mounted instrumentation and attached piping systems. The inspector will continue to review the licensee's monitoring and long term corrective actions regarding EDG vibration. The inspector reviewed the licensee's immediate corrective actions and had no further questions. Licensee identification and correction of the vibration induced problems will be reinspected prior to plant restart to assure conditions that could impact diesel operability are corrected.

5.0 Plant Operational Status Reviews

The inspector reviewed plant operations from the control room and reviewed the operational status of plant safety systems to verify safe operation of the plant in accordance with the Technical Specifications (TS) and plant operating procedures during Mode 5 and Mode 6 conditions. Actions taken to meet TS requirements when equipment was inoperable were reviewed to verify the limiting conditions for operations were met. Plant logs and control room indicators were reviewed to identify changes in plant operational status since the last review and to verify that changes in the status of plant equipment was properly communicated in the logs and records.

Control room instruments were observed for correlation between channels, proper functioning and conformance with technical specifications. Alarm conditions in effect were reviewed with control room operators to verify proper response to offnormal conditions and to verify operators were knowledgeable of plant status. Operators were found generally cognizant of control room indications and plant status (Sections 9 and 10 address this aspect further). Control room manning and shift staffing were reviewed and compared to TS requirements. No inadequacies were identified. The following specific activities were also addressed.

5.1 Decay Heat Removal Operability Review

The residual heat removal system was reviewed to verify that the system was operable and running in its decay heat removal mode. The review included consideration of: proper positioning of major flow path valves; adequate flows and proper temperatures in supporting cooling systems; operable normal and emergency power supplies; indicators and controls functioning properly; and visual inspection of major components for leakage, cooling water supply, lubrication and general condition. No inadequacies were identified.

5.2 Review of Plant Incident Reports

Selected plant incident reports (PIRs) were reviewed to (i) determine the significance of the events; (ii) review the licensee's evaluation of the events; (iii) verify the licensee's response and corrective actions were proper; and, (iv) verify that the licensee reported the events required. The PIRs reviewed were: number's 182-87 dated 8/11, 186-87 dated 10/7, 192-87 dated 10/18, 193-87 dated 10/22, 194-87 dated 10/23, 195-87 dated 10/24, 196-87 dated 10/31, 197-87 dated 11/1, 198-87 dated 11/3, 200-87 dated 11/4, 202-87 dated 11/8, 206-87 dated 11/10, 207-87 dated 11/10, 209-87 dated 11/4, 215-87 dated 11/17, 216-87 dated 10/30, 218-87 dated 11/17, 219-87 dated 11/17, 226-87 dated 11/21, 229-87 dated 11/23. The following items warranted inspector followup:

PIR 205-87, Main steam Safety Valve Setpoint Drift. This item is reviewed in Section 13.

PIR 210-87, Water in Reactor Vessel Stud Holes. This item is reviewed in Section 3.3.

PIR 214-87, Inadequate Testing of Containment Penetration Breakers. This item is reviewed in Section 3.4.

PIR 220-87, Sodium in RWST and RCS. This item is reviewed in Section 9.0.

PIR 224-87, Foreign Objects found in Reactor Vessel. This item is reviewed in Section 7.0.

6.0 Observations of Physical Security

Selected aspects of site security were verified to be proper during inspection tours, including site access controls, personnel and vehicle searches, personnel monitoring, placement of physical barriers, compensatory measures, guard force staffing, and response to alarms and degraded conditions. The following items warranted inspector followup.

6.1 Lost Badge in a Locked Office in the Protected Area

The licensee notified the inspector of the results of his investigation of the following event. An individual left his badge in the fabricating shop inside the protected area at 12:30 a.m., November 3. He discovered his badge was missing when he tried to exit through the North Access Point (NAP). He notified a guard inside the NAP that he didn't have his badge and needed to talk to him. The guard had another individual key him into the NAP. The individual stated that he left his badge in a locked office in the Unit 3 Welding Fabrication shop. The guard incorrectly sent the individual home without attempting to retrieve the lost badge. The guard failed to notify his supervisor of the event and left at the end of his shift at 7:00 a.m., November 4. The individual's coworkers found the badge in the locked office at approximately 12:45 p.m., November 3 and notified the Security department. Security deactivated the badge within 10 minutes and verified the badge had not been used for vital area access. Security assumed that the badge had been lost on dayshift and attempted to locate the individual on-site. When they were unable to locate the individual, they left a note at the NAP for the individual to report to security when he arrived on-site. The individual arrived at 3:20 p.m. and reported to security. The incident was investigated by security personnel and the resident inspector was informed at 4:30 p.m. The event was classified as reportable per 10CFR 73.71c and reported to the NRC via the ENS at 5:08 pm November 3. The security guard involved has been suspended pending further investigation. This event will be reviewed further after completion of licensee's actions.

6.2 Undetected Dangerous Instrument Potentially Brought On-Site

The licensee reported that a dangerous instrument was brought on site and removed without being detected. The device was an electrical stun gun classified by the State of Connecticut as a dangerous instrument. The state puts various devices into this category to control their use by requiring the owner to have a permit to carry them.

The device itself is black and rectangular, slightly larger and similar in shape to a pack of cigarettes. It could be confused with electronic test equipment that is used onsite.

The licensee discovered the incident when a worker reported that a Northeast Nuclear Energy Company (NNECO) employee who worked in computer services told of bringing a stun gun on-site. He reportedly brought the device on-site on either 11/15 or 11/16 when he stepped into the office to retrieve some items prior to beginning his vacation. Reportedly, he stated that he realized he had brought the stun gun inside the protected area (PA), was worried that he could not get it out, and decided to leave it locked in his desk. He then returned on 11/21 with a duffle bag full of various items and reportedly stored the stun gun in the bottom to remove it from the site, which he did successfully.

After the licensee discovered the incident on 12/2, he classified the event in accordance with 10CFR 73.71(c) at 10:30 am and reported via the ENS at 10:51 am. The licensee then investigated the incident and determined that the search guards at the entry and exit points either performed an inadequate search or did not recognize the stun gun as a dangerous instrument. In addition, the licensee reviewed the transaction history for 11/16 from 4:30 pm to 5:30 pm and noted the individual entered and exited on 3 separate occasions within a 10 minute period with a maximum stay time in the PA of 2 minutes. The licensee was unable to determine if the individual had an adequate amount of time to store the stun gun in his office. When the individual was questioned on 12/2 he stated that he did bring the stun gun into the PA and exited without it being detected. Later that day the individual recanted his original story and stated that he never did bring the stun gun on-site but wanted to identify a problem with security personnel not performing adequate searches of personnel entering and exiting the PA. The licensee questioned him further to determine if he owned a stun gun and he stated he did and it was at home. He stated he was willing to bring it in for examination. He brought it in for examination and the licensee photographed it for future reference.

The inspector reviewed the security investigation with the licensee and questioned the licensee regarding his corrective actions. The licensee stated that the individual has been suspended pending further evaluation of his actions. The individual's unescorted access to the PA was removed. The training program for entry and exit searches will be upgraded to include the list of dangerous instruments identified by the State of Connecticut. The inspector noted the corrective actions and asked to be informed of the final dispositions. Additionally, the inspector asked if the stun gun was used. The licensee reviewed the matter and stated that he had no indication the stun gun was used. Further, compensatory measures were discussed and resolved to the inspector's satisfaction.

The inspector concluded that no violation of security requirements occurred. Potential weaknesses in the ability to detect dangerous weapons were identified and corrective actions are in progress to address the issue. No inadequacies were identified in the licensee's followup investigations and corrective actions.

7.0 Foreign Objects on the Lower Core Plate

A foreign object was discovered on the lower core plate for core location C05 at 10:30 a.m., November 17, as a fuel assembly (FA) was being moved to the upender for transfer to the fuel building. The object blocked one of the four 2-inch diameter core flow holes in the lower core plate. The licensee used a submersible TV camera and offloaded the entire core to search for additional debris. A Westinghouse core internals expert was called on-site to help identify the part. The resident inspector reviewed licensee actions including: review of the object via videotapes and after retrieval from the vessel to identify it and its source; review of fuel shuffle activities; interviewing licensee personnel and consultants to assess the significance of the object, its potential impact on the reactor during past operations, and the significance of the debris relative to future operations; and the licensee's plans to address the above concerns.

The licensee discovered a total of 8 objects on the lower core plate. Seven of the objects were similar. The eighth object was flat and about the size of a dime. Radiation levels taken on contact with the object identified on November 17 measured 300 R/hr 2.5 feet underwater. The pieces showed a buildup of a black oxide layer. The 7 similar pieces are cylindrical and approximately 2.25 - 2.5 inches in diameter and 1/32 of an inch thick. The interior of the pieces have a turned finish. There are two holes 180 degrees apart and approximately 1/16 inch in diameter. There is also a tab an eighth of an inch square protruding from the upper circumference of each piece and there are two dimples, diametrically opposed 90 degrees from the tab, that indent each piece by 1/16 of an inch.

All 8 of the objects were successfully retrieved from the reactor vessel. Core off-loading was completed at 12:05 p.m. on 11/21 and there were no more objects visible on the lower core plate. The licensee performed a detailed inspection of the area below the lower core plate by lowering a submersible camera through every other flow hole in a given row of the plate. Additionally, the reactor vessel bottom head area was inspected and no further foreign objects were identified. The foreign objects are not part of the core internals based on a review of the object's size and geometry.

The dime-shaped, flat object has yet to be identified. The licensee identified the 7 cylindrical objects as locking cups for Westinghouse Model 93A1 reactor coolant pump (RCP) turning vane diffuser hold down bolts. The current vendor drawings show that the dimensions of the locking cup are similar to the objects. However, the current drawings show a set of tabs protruding from the cup at its midpoint and that the piece has a 359 degree circumference to allow for installation. Westinghouse personnel identified the licensee's locking cups as an earlier design used only on the Millstone 3 RCPs. The licensee's RCP locking cups have tabs at one end and are 360 degrees in circumference. There are 23 locking cups per pump (92 total cups for Millstone 3).

Each locking cup was designed to hold one bolt, which is similar to an allen screw, in position. The bolts were 8.5 inches long with a 1.5-inch thread diameter and a 2.5-inch head size. There were detents on the bolt heads to allow one end of the locking cups to be crimped into the detent holding the bolts in place. The bolts were to be torqued to 2000 ft-lb and then the cups crimped on to the bolt heads. Tabs on one end of the cups fit into detents in the turning vane diffuser assembly, thereby preventing rotational motion of the bolts.

The licensee reviewed core performance records for the first operating cycle and identified no apparent impact on the thermal hydraulics for the fuel assembly positioned at core location C05. The licensee contacted Westinghouse for detailed analysis of the potential risks of operating with loose RCP locking cups. The analysis was to include the effects of various configurations of all of the remaining 85 locking cups on core thermal hydraulics. Additionally, the potential for damaging the RCPs and loop instrumentation was included. Although the hold down bolts are torqued to 2000 ft-lb, their effects on the RCPs and loop instrumentation were also considered. The licensee's proposed analyses and action plans were reviewed during a conference call with NRC regional management and NRR technical personnel on November 25. The licensee noted NRC staff comments during the conference call and included them in the analysis.

The licensee received the analysis results from Westinghouse. The effects of loose locking cups on the lower core plate in various configurations were assessed as minimal. There was no additional DNBR penalty except in the most severe case, which assumed all 85 locking cups ended up in a 4X5 array with a locking cup in each flow hole. The analysis showed that the worst case transient DNBR would then be acceptable only at a reduced power level. The inspector questioned the licensee on the details of the analysis and noted no inadequacies.

The licensee also reviewed the potential effects of loose bolts on the RCP and on core instrumentation. The effect of bolt impact on loop instrumentation was unacceptable to the licensee. Direct impact could significantly degrade the RCS boundary. The licensee decided to extend the outage eight weeks to tack weld the remaining locking cups in place.

The area will be reviewed further on subsequent routine inspections to follow the licensee's corrective actions to repair the locking cups. Preliminary estimates by the licensee indicated the repair could cost 110 man-Rem of exposure per RCP. The licensee initiated actions to reduce this exposure. The inspection will include a review of the licensee's actions to complete the repair while keeping the exposures as low a reasonably achievable.

8.0 Loop Stop Valve Interlock Design Change for Three Loop (N1) Operation

The inspector reviewed the licensee's changes to the Millstone 3 loop stop valve interlock design made during construction. These changes were not reflected in Final Safety Analyses Report (FSAR) Chapters 7 and 15 as amended to date. The following chronology was identified.

DATE	DOCUMENTATION
10/29/82	Northeast Nuclear Energy Company (NNECO) application for Operating License including FSAR.
05/31/83	NRC requested additional information including thermal hydraulic analysis for N-1 operation.

04/09/84	NNECO provided the information requested on 5/31/83.
08/15/84	NRC issued Safety Evaluation Report (SER); Section 15.4.4/15.4.5 concluded that the results of a RCP startup at an incorrect temperature was acceptable.
11/20/84	NNECO requested authorization for three loop (N-1) operation.
12/07/84	NNECO submitted draft TS for 4-loop operation; TS for N-1 loop operation was to be submitted later.
03/04/85	NNECO submitted Amendment 12 to FSAR including revised Section 15.4.4, Startup of an Inactive Reactor Coolant Pump at Incorrect Temperature and Boron Concentration.
Mar-Nov,'85	NRC issued SER Supplements 1, 2, and 3; each stated that N-1 loop operation was under staff review.
04/30/85	NNECO provided response to N-1 questions on thermal-hydraulic parameters, inlet flow maldistribution, flow instability, and changes required in the TS.
Jun-Sep,'85	NRC/NNECO meetings (licensee states at least 3) on N-1 loop operation.
07/05/85	NNECO answers staff questions and provides draft Technical Specifications (TSs) for N-1 loop operation.
09/20/85	NNECO submitted Proof and Review TSs including Limiting Condition for Operation (LCO) 3.4.1.6, Isolated Loop Startup.
10/09/85	NNECO updated the Proof and Review TSs including LCO 3 4.1.6, RCS Isolated Loop Startup.
10/25/85	NRC requested additional information on Chapter 15, Anticipated Operational Occurrences (AOOs) and Postulated Accidents (PAs) for N-1 loop operation.
11/25/85	NRC issued Low Power (5 %) Operating License (OL) with TS LCOs for N-1 operation but prohibited such operation (License Condition 2.C.6) until outstanding issues are resolved to the satisfaction of the staff.
12/06/85	NRC issued SER Supplement 4, which addressed N-1 loop operation in Sections 4.4.7 and 15.1 through 15.5.6. A number of N-1 loop operation issues were left unresolved.

01/31/86	NRC issued Operating License (OL) with TS LCOs for N-1 operation but prohibited such operation (License Condition 2.C.4) until outstanding issues are resolved to the satisfaction of the staff.
02/20/86	NRC issues SER Supplement 5; no evaluation of N-1 loop operation issues was included.
07/01/86	NNECO provided the additional information requested by NRC letter dated 10/25/85.
07/28/86	NRC/NNECO meeting held to discuss the NRC staff's concerns regarding the aspects of the protection system which were unique to plant operation with a loop out of service.
08/25/86	NNECO documented above meeting and provided the requested information in the form of FSAR Chapter 7 marked-up pages.
11/14/86	NRC/NNECO meeting held at site to discuss the design details of the modifications made to the Reactor Coolant System (RCS) loop isolation valve interlocks.
01/21/87	NRC issued SER for N-1 loop operation. Instrumentation and control evaluation containing 6 open items and a list of drawing errors. Additional information for human factors review was requested.
02/12/87	NRC/NNECO meeting held to discuss the 6 SER open items, the drawing errors, and provide clarification for human factors review.
02/20/87	NNECO letter documented 11/14/86 and 02/12/87 meetings with the staff, provided requested information, and included draft FSAR Section 15.4 page changes.
07/31/87	NNECO letter supplemented the 02/20/87 information and provided a draft copy of FSAR Sections 7.2, 7.6, and 15.4 pages; final changes to be submitted later.
11/16/87	NRC issued SER for N-1 loop operation with some open issues.
01/31/88	Date when FSAR update submittal is required per 10 CFR 50.71(e); licensee has made several update submittals ahead of the due date to reduce the size of the update package.

Upon establishment of N-1 loop operation, NNECO had originally planned to allow restart of a Reactor Coolant Pump (RCP) during low power operation (below 25 %, in Mode 1 or 2). Reactor Protection System (RPS) interlocks to limit reactivity additions due to temperature and/or boron concentration differences were provided. In 1985, the RCPs were tested in the loop isolated mode (isolation valves closed). Extensive pump vibration occurred due to undersized bypass piping. Loop operation, necessary to equalize temperature and boron concentration prior to opening the loop isolation valves, was not recommended by Westinghouse. Restart of a RCP in Modes 1 or 2 was dropped in favor of RCP restart in Modes 5 and 6 (cold shutdown or refueling) only. This allowed simplification of the interlock design. A revision to the proposed (Proof and Review) TS was provided in the September 20 and October 9, 1985 submittals. These TSs, as proposed, were issued with the OL but negated by License Condition 2.C.4 which prohibited N-1 loop operation until outstanding issues are resolved to the satisfaction of the staff.

The licensee stated that the redesigned interlock system was discussed with the staff in at least three meetings held between June and September, 1985, during the July 28 and November 14, 1986 meetings, and at the February 2, 1987 meeting. The 1985/1986 meetings were documented in NNECO's January 21 and February 20, 1987 submittals. The current FSAR does not describe the present N-1 loop operation interlocks accurately. The licensee stated that FSAR changes are submitted only after system designs are finalized. Copies of marked up FSAR pages describing the current interlock design have been provided to the NRC staff. The N-1 loop interlock design modification has had outstanding NRC staff issues since 1985. The modified instrument and control design has not yet been approved by the NRC staff.

Prior to Operating License (OL) issuance, licensees are obligated to keep the FSAR up-dated to reflect modifications so the FSAR accurately describes the as-built plant at OL. In this case, the licensee modified the N-1 loop isolation interlocks described in the FSAR after issuance of the staff's SER and three Supplements but prior to OL issuance. However, the modifications had no safety significance because NRR review was ongoing and License Conditions 2.C.6 (5 % License) and 2.C.4 (Full Power License) prohibited N-1 loop operation.

After OL issuance, changes to the facility as described in the FSAR are allowed pursuant to 10 CFR 50.59. Without prior NRC approval, such changes cannot involve a TS change or an unreviewed safety question. Changes made under 10 CFR 50.59 are to be submitted, in accordance with 10 CFR 50.71(e), as replacement FSAR pages on an annual basis commencing two years after OL issuance. The first such FSAR update for Millstone 3 is due January 31, 1988.

Inasmuch as N-1 loop operation does not require a TS change but does require NRC review to resolve outstanding questions prior to removal of the license condition that prohibits such operation, no noncompliance with 10 CFR 50.59 or 10 CFR 50.71 was identified by the inspector. Since three loop operation remains prohibited by License Condition 2.C.4, no safety concern was generated during the inspector's review. The inspector had no further questions at this time.

9.0 Abnormal RWST and RCS Sodium Concentration

The licensee discovered an abnormal concentration of Sodium (Na) in the refueling water storage tank (RWST) and reactor coolant system (RCS) based on the 7:00 am, 11/18 chemistry sample. RWST and RCS Na concentration were 200 parts per million (ppm) and 10 parts per billion (ppb), respectively.

The oncoming Senior Control Operator (SCO) for the 3:30 pm shift on 11/17 noted a Chemical Addition Tank (CAT) low level alarm during board walkdown just prior to shift turnover. He questioned the on-shift SCO and assumed that action would be taken to correct the alarm condition before he departed the control room to relieve the Senior Reactor Operator (SRO) monitoring fuel handling activities in the containment. The SCO informed the shift supervisor (SS) prior to exiting the control room. The SS noted that the level was low in the Technical Specification (TS) band, at 19,200 gallons. TS 3.6.2.3 states that the CAT tank shall be operable in Modes 1 through 4 with a volume between 19,100 and 20,100 gallons of 1.35% to 2% weight percent solution of sodium hydroxide (NaOH). Additionally, two gravity feed paths must be operable and capable of adding NaOH solution.

The SS reviewed the CAT Tank Low Level Alarm Response Procedure (OP 3309 Section 8.16) and completed the applicable initial and subsequent action procedure steps. The SS incorrectly assumed that the level decrease was caused by transmitter or instrument calibration error and took no further action throughout the 3:30 pm to 11:30 pm shift on 11/17. The SS did not further isolate the CAT tank because he did not believe the tank could have been leaking for so long without being tagged. The SS stated that the CAT tank level decrease apparently began on 11/13 when RWST level was reduced to fill the Reactor Cavity. The inspector noted that the licensee's failure to promptly isolate the CAT tank increased the Na contamination in the RWST and RCS.

During shift turnover, the offgoing SS on the 3:30 pm to 11:30 pm shift noticed that CAT tank level had decreased by 100 to 200 gallons during his shift and informed the oncoming SS. The oncoming SS decided to further isolate the tank by closing the manual valve (3QSS*V29) upstream of the two parallel motor-operated valves (3QSS*MOV29A & 29B) that were checked closed in accordance with OP 3309 Section 8.16.

PIR 220-87 was written on 11/18 to document the high Na. The event was reviewed and determined not to be reportable. It was forwarded to the unit superintendent for routine review and investigator assignment. The inspector reviewed event reportability event and identified no inadequacies.

The abnormal concentration of Na was caused by leakage past valves that connect the CAT to the RWST. There are 4 valves in the line connecting the CAT tank to the RWST. From the CAT Tank, the first is a manual valve (3QSS*V29), then 2 valves in parallel (3QSS* MOV 29A & 29B) that open during accident conditions; the last valve is a manual valve (3QSS*V32).

There is a drain valve, between 3QSS*V32 and the 3QSS*MOV29A and 29B, that drains to the Liquid Radioactive Waste System (LWS). All of these valves were repositioned by various operating shifts in an attempt to stop the CAT tank level decrease.

The high Na in the RWST and RCS was presented to the plant staff in the 11/19 Unit 3 Superintendent Meeting during PIR Review. The Operations Assistant reviewed the CAT Tank Valve lineup with the midnight shift SS who had isolated the system on the previous day. The plant equipment operators (PEOs) had closed 3QSS*V32 and opened the drain to LWS. The SS mistakenly thought that 3QSS*V29 had been closed to prevent leakage past 3QSS*MOV29A and 29B. The Operations Assistant had the SS close 3QSS*V29 to provide additional assurance that the leakage from the CAT Tank would be stopped. The drain to LWS was left open to collect future leakage.

This lineup was presumed adequate to stop the RWST and RCS sodium contamination. However, RWST Na concentration continued to rise until it peaked at 25 ppm over the next several days. The licensee reviewed the incident and determined that the leakage into the RWST was stopped when all interconnecting valves were closed. The license concluded that the reason RWST Na concentration continued to increase was because a slug of 2% NaOH was trapped in the quench spray subsystem (QSS) spray pump suction which communicates directly with the RWST. One of two RWST Recirculation (Recirc) Pumps runs continually, recirculating the RWST. It takes suction from a different area of the tank than the QSS spray pumps but would promote the dissolution of NaOH in the QSS suction lines. During RWST drain down for cavity fill, a sufficient pressure differential (DP) was developed to cause leakage past 3QSS*MOV 29A and 29B. This leakage mingled with RWST water that was sent to the cavity. The NaOH in this water freely mixed with RCS water after the head was removed during cavity fill. Since RHR was running, the Na concentration of RCS built up similarly to the RWST Na concentration. The RCS Na concentration was reduced by normal ion exchange in the letdown system demineralizer. The licensee also removed the RWST Na by running a special skid mounted demineralizer system to cleanup both the RWST and the RCS. The Na concentration decreased as expected and the Na concentrations of the RWST and RCS were less than 1 ppm and 10 ppb, respectively, by 12/4.

The licensee promptly responded to the inspector's questions. However, the licensee's followup and root cause identification are still in question in regard to preventing recurrence of this situation. The inspector will continue to monitor future licensee corrective actions. (UNR 87-24-02)

10.0 ESF and Waste Disposal Building Contamination During Cavity Drain Down

On 12/1, a Plant Equipment Operator (PEO) answered a fire alarm at the 4.5 foot elevation of the Waste Disposal Building. He identified that the alarm had been caused by vapor coming from the sump in the Waste Disposal Building. Hot condensate was entering the sump from the condensate return dump valve from the radwaste evaporator. The dump is repositioned to the sump when a high conductivity reading is sensed on the waste evaporator return to the auxiliary condensate system.

This vapor was also noticed by a security guard stationed at the door between the Waste and Auxiliary Buildings at the 4' 6" elevation because it was coming from an area that was posted as a contaminated area. The guard, concerned about potential radioactivity, left his post to report the problem from a safer location. This action required the Security Shift Supervisor (SSS) to report that the guard had left his post and the SSS called the Control Room for further information. No further information was known since the event was still being investigated. Shortly after the SSS call, the inspector entered the control room, reviewed the event with the SS, and noted that the guard's activities were prudent considering the situation.

The licensee stated that the waste evaporator condensate valve was dumping to the sump because of a high conductivity alarm. When the conductivitiy alarm cleared, the valve would not return to its normal position as required. Sufficient water was put into the sump to cause the sump to overflow. The radwaste PEO requested help and together with other PEOs was able to return the valve to its proper position The limit switch on the valve was stuck and had to be manually repositioned so that the valve would return to its correct position. The operators reported to the control room that the problem was corrected in the waste disposal building and that Health Physics (HP) was surveying the area to ensure there was no spread of contamination.

Later, during the HP surveys, technicians reported that the sump in the waste disposal building was overflowing again. The operators returned to the waste disposal building, expecting to find the condensate dump valve again putting water into the sump. The valve was not dumping and an immediate source of the water could not be located. Different HP technicians reported that the ESF building sump was also overflowing and a PEO was sent to investigate. Three valves were found open on the line that was being used to pump the Rx cavity to the RWST. Two of the valves were caution tagged for Tagout #3952 and had hoses attached for draining to the sump. The valves were 3QSS*V932 and 3CHS*V930, which are drain valves that were open for previous work activities. The third drain valve, 3SIL*V882, did not have a hose attached. The PEO closed the three valves to stop the flow of water into the ESF room sump. HP surveyed the area after the water level receded and determined that the areas that had been flooded were contaminated up to levels of 90,000 dpm per 100 square centimeters (cm). The highest levels were near the sump proper and the general area levels ranged from 1000 dpm per 100 square cm to 40,000 dpm per 100 square cm.

The reason the ESF sump caused the second overflow of the waste disposal building sump was that approximately 25,000 gallons of water was pumped to the high level LWS tanks from the ESF sump when the tanks did not have the capacity to hold that additional volume. Portions of the south and middle rooms of the 4'6" elevation of the ESF building have been decontaminated to allow limited access. Further decontamination will be necessary to release the areas for unrestricted use.

The event apparently occurred during cavity drain down. Repair activities on the QSS return valve to the RWST (3QSS*V044) were underway per work order AWO M3-87-2938 to replace a bad rubber gasket on this butterfly valve. Prior to cavity drain down, the flow path was investigated to ensure its availability. Additionally, plant tagouts of boundary valves were cleared, as appropriate, for cavity drain down. Tagout 3952-87-18 which was hung to support AWO M3-87-2938 was reviewed and the valve was closed. However, 3QSS-V932 on the same tagout was not cleared and it was directly in line with the cavity drain down flowpath. It contributed the major portion of the total volume released to the sump.

The licensee was questioned regarding the event and responded quickly to the inspector's questions. However, the licensee's lack of boundary valve control during reactor cavity drain down led to ESF and Waste Disposal Building Contamination. The inspector will continue to monitor the licensee's corrective action. The lack of boundary valve control is of concern to the NRC and this item is unresolved pending further NRC review of the adequacy of equipment configuration control (UNR 87-24-03).

11.0 Mechanical and Hydraulic Snubber inspection

In accordance with TS 3/4.7.10, hydraulic and mechanical snubbers were visually inspected and functionally tested during this first refueling outage. By letter dated October 27, 1987, the licensee notified NRC Region I of their snubber inspection program for the first refueling outage. One facet of this program was the removal of 212 mechanical snubbers under the snubber reduction program and the resistance temperature detector (RTD) bypass loop removal project. The number of snubbers and the snubber population, divided unto four functional test groups, was as follows.

Group	Size/Type	Quantity in Plant Start	Removed	Remaining
A	PSA 1/4 and 1/2	402	169	233
B	PSA 1, 1L, 3, and 10	583	43	540
C	PSA 35 and 100	115	0	115
D	All Hydraulic Snubbers*	44	0	44
	Totals	1144	212	932

* (plant has large hydraulic snubbers supporting the Steam Generator and Feedwater systems)

Although the letter stated the snubbers to be removed under the snubber reduction program would be visually inspected prior to removal, the inspection would have no affect on future inspection intervals. The inspector informed plant management that he could not concur with this approach if the snubber problem is independent of the snubber location. In other words, if failure is due to mechanical problems, the failure should be counted in determining the schedule for the next visual inspection. The plant staff agreed and the plant manager issued instructions that snubbers being removed were to be visually inspected.

The licensee plan was to visually inspect 1043 safety-related snubbers (broad definition including snubbers that could affect safety) plus 101 snubbers supporting the RCS bypass loops. The RCS bypass loop snubbers were inspected after removal because the inspector's concern and discussion with plant management was after the bypass loop snubbers had physically been removed. Of the 1043 snubbers inspected (the 101 RCS bypass loop snubbers were still being inspected at the time of this review), 895 snubbers clearly passed the visual inspection, and 106 required engineering evaluation. Of these, 96 engineering evaluations had been completed. The major reason engineering evaluations were required was the "C" dimension (as-found piston stroke position) was outside the 1/2 inch tolerance allowed. This was true for 94 of the completed engineering evaluations and 9 of the 10 evaluations remaining to be completed. Other problems were one broken lock washer and some of snubbers out of alignment with their clevis by more than the allowed 5 degrees.

Of the four groups functionally tested, there were 6 failures, all in mechanical Group B (PSA 1, 1L, 3, and 10). The original sample size was 36 PSA Sizes 1, 3, and 10. There were 2 failures: one was frozen at the 2-inch position; the other had more than the 5 % drag allowed by the manufacturer. For each failure, the TS required an additional 5 % sample tested. In all, six samples of Group B size snubbers (a total of 145 snubbers) were tested. The failures were:

Snubber No.	Location	Reason for Failure
19458	3MSS4PSSP0486	Locked-up (frozen in place)
22541	3RCS1PSSP0134	Did not lock up in either direction
23850	3MSS4PSSP0489	Exceeded 5 % drag in tension
23853	3MSS4PSSP0490	Exceeded 5 % drag in compression
5689	3RHS1PSSP0420	Locked up
8797	3RHS1PSSP0414	Exceeded 5 % drag then locked up

All three main steam snubbers that failed were on the 3-inch bypass line around the main steam line isolation valves. The licensee believes considerable wear of these snubbers resulted from continuous vibration of the bypass line. They are investigating system support changes to reduce bypass line vibration. The licensee had not finished the evaluation of snubber testing failures at the end of the report period.

The inspector reviewed the procedures used for snubber visual inspection and for functional testing and observed some inspection and testing during the outage. Snubbers with minor visual problems (rust, dirt, etc.) were selected for functional testing. The inspector will continue to review the snubber visual inspection and functional test program during future inspections.

12.0 (Open) IE Bulletin 85-03, Motor Operated Valve (MOV) Common Mode Failure During Plant Transients Due to Improper Switch Settings

The licensee's June 11 and October 28, 1986 responses addressed the six specified actions related to the subject bulletin. This response was reviewed for timeliness and content in Inspection Report 50-245/86-17 for Millstone 1.

IE Bulletin 85-03 specifies that, for MOVs in the high pressure coolant injection/core spray and emergency feedwater systems (RCIC for BWRs) that are required to be tested for operational readiness in accordance with 10 CFR 50.55a(g), licensees develop and implement a program to ensure that components are selected, set, and maintained properly. The licensee's reply concluded that, for Millstone 3, this includes the Auxiliary Feedwater (AFW), Safety Injection (SI), and the charging portion of the Chemical and Volume Control (CVC) systems. The bulletin provisions are addressed as follows.

a. Design Bases for Motor Operated Valves (MOVs)

The licensee was to review and document the design basis for each MOV including the maximum valve differential pressure expected during both opening and closing for both normal and abnormal events. The licensee identified 4 AFW system MOVs, 13 SI system MOVs, and 13 CVC system MOVs to be included in the program. In addition, the licensee informed the inspector that 3 additional MOVs (3SIH*MV8802A&B and 3SIH*MV8806) have been included in the program. The specified design differential pressures for all 33 valves were equal to or greater than the normal and abnormal event maximum differential pressures. Therefore, this bulletin action was satisfied.

b. Translate the Specified Design Differential Pressure to the Correct MOV Switch Settings

The licensee's response outlines a plan to determine the proper switch settings using a combination of analytical and empirical data. The inspector found that, although this engineering had been performed, it was difficult to review during the outage. The review will be performed after the final report is submitted.

c. Testing of MOVs to Ensure Valve Switch Settings

The licensee committed to stroke test each valve, to the extent practical, to verify that the settings defined have been properly implemented. To this end, NNECO has purchased Motor Operated Valve Analysis and Test System (MOVATS) equipment and initiated MOV testing at Millstone 3. The inspector reviewed Special Procedure 87-3-3, Procedure for Testing Limitorque MOVs using MOVATS and observed the physical testing of several MOVs. The licensee has two testing crews working to complete the testing of 33 MOVs including flow testing of 10 MOVs during the current refueling outage. The licensee stated that important MOV findings/resettings will be provided in the final report.

d. Review/Revise Procedures to Ensure Correct MOV Switch Settings

The licensee committed to review and revise procedures as necessary to ensure that correct switch settings are determined and maintained throughout the life of the plant. The licensee is working to have the necessary procedure changes completed a few months after Unit 3 returns to power operation.

e. Report the Results of MOV Design Basis Review and Provide the Schedule for Corrective Actions

The licensee's June 11 and October 28, 1986 letters provides the results of MOV design basis review and commits to complete the other actions for Millstone 3 a couple of months after the end of the current refueling outage. he licensee has established a coordinated and comprehensive MOV testing program that addresses the concerns of IEB 85-03. They now plan a composite report for all 4 plants that will be submitted to the NRC about 60 days after end of the Millstone Unit 2 outage. This schedule is acceptable to NRC Region I. The NRC will review the final report when it is received.

13.0 Review of Licensee Event Reports (LERs)

Licensee Event Reports (LERs) submitted during the report period were reviewed to assess LER accuracy, the adequacy of corrective actions, compliance with 10 CFR 50.73 reporting requirements and to determine if there were generic implications or if further information was required. Selected corrective actions were reviewed for implementation and thoroughness as documented elsewhere in this report. The LERs reviewed were:

LER 87-35-00, Surveillance Test Method not in Accordance with Technical Specification (TS)

On 10/16, at 3:15 p.m., while operating at 98% power in Mode 1, Operations Department personnel performing a biannual procedure review identified a difference between the testing method specified by plant Technical Specifications and the method utilized by the surveillance procedure. The surveillance test for containment air lock integrity requires the use of a pressure decay test. The method utilized by the plant surveillance was a constant pressure makeup test. The inspector noted that either method provides valid indication of seal integrity.

Initial operator actions upon discovering the discrepancy were to enter the Limiting Condition for Operation (for 2 doors inoperable) and perform an air lock leakage surveillance verifying the overall air leakage was within limits. The test was satisfactory. The surveillance was updated to ensure it complied with TS.

There have been two similar incidents, LER 86-047-00 and LER 86-053-00, in that a surveillance value was not updated from the value in preliminary Technical Specifications. Additionally, the licensee has identified 2 other surveillance-related events that will result in LERs. The inspector has noticed an increased frequency of these types of events and will address this trend in future inspections.

LER 87-036-00, Setpoint Drift on Main Steam Safety Valves

At 1:00 p.m. on 10/31, with the plant in hot standby, eleven Main Steam Safety Valves failed the +/- 1% tolerance band set by Technical Specification Table 3.7-3. No operator action was required since the plant was in hot standby at the time the valves were being tested.

The valves were being tested per an approved Maintenance Surveillance Procedure. Of the fifteen valves that were tested, eleven valves failed.

Six tested high, five tested low, and four were satisfactory. The remaining untested valves will be replaced or tested. The licensee determined that there were no safety issues since the valves all lifted within the design pressure of the main steam system (1200 psi).

The inspector observed this activity during his monthly surveillance activity and noted that the licensee followed procedures. as required. The acceptable band is small at plus or minus 12 psi and the valves are tested in the as-found condition. The licensee assumed that the setpoint drift occurred on these valves sometime during plant operation as it was discovered soon after going into hot standby. The licensee stated that the valves have been tested and setpoints had been calibrated previous to this incident.

The root cause of the event is undetermined and the inspector noted that LER 87-009-00, Early lifting of Pressurizer Safeties, identified setpoint drift as a potential root cause. The inspector will continue to monitor the licensee's corrective action.

LER 87-038-00, Loss of Emergency Bus 34C

The inspector reviewed the event and documented his findings in section 3.1 of this report.

14.0 Committee Activities

The inspector attended meeting 3-87-162 of the Plant Operations Review Committee (PORC) on 11/11 and reviewed the minutes for PORC meetings 87-142 and 87-144. The inspector noted by observation and/or from the written record that committee administrative requirements were met for the meetings, and that the committees discharged their functions in accordance with regulatory requirements. The inspector observed a thorough discussion of matters before the PORC and a good regard for safety in the issues under consideration by the committee. No inadequacies were identified.

15.0 Observation of Maintenance

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

-- Replacement of the Refueling Cavity Pit Seal

-- "B" Diesel Generator Injector Pop Test

-- Steam Generator Handhole Repair

No inadequacies were identified.

16.0 Observation of Surveillance Testing

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

-- Snubber Inspection, Procedure for Inservice Inspection, 11/3

-- Loss of Power (LOP) Test, 11/29

-- ESF with LOP Test, 12/2

No inadequacies were noted.

17.0 Review of Refueling Activities

The inspector reviewed refueling activities from the control room and in the containment to verify that fuel handling was completed safely and in accordance with regulatory requirements.

Observations of the licensee's activities in the control room indicated that: special nuclear material was accounted for as it was moved from the spent fuel pool and the reactor vessel; continuous communications were established and maintained between the control room, the spent fuel pool and the Sigma refueling machine; and the licensee was aware of source range nuclear instrumentation operability requirements and verified the requirements were met during core alterations.

Observations of licensee activities inside the containment indicated that: containment integrity was maintained as required; cavity water level was maintained greater than 23 feet above the vessel flange; restrictions on polar crane movement over the open vessel were observed; radiation protection contols were established and observed; the SROs in charge of refueling activities completed a turnover upon completion of a work period; and inspector review of fuel movement confirmed that fuel moves were completed in accordance with the schedule established by procedure. The inspector reviewed radiation levels on the refueling bridge with a fuel assembly loaded in the refueling bridge and with the mast in the full up position. Radiation levels were acceptable. The inspector noted that caution and care were used when moving fuel, and that the SRO in charge of refueling actively monitored and directed the activities from the refueling bridge whenever core alterations were in progress.

The inspector noted the licensee's regard for the importance of safety and following proper procedures and administrative controls during refueling activities was very high. No inadequacies were noted.

18.0 Management Meetings

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