U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-423/86-33

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: October 7, 1986-November 17, 1986

Inspectors: J. T. Shedlosky, Senior Resident Inspector
F. A. Casella, Resident Inspector
E. M. Conner, Project Engineer

Approved by: Elo Mc Cabe 11/20/86
E. C. McCabe, Chief, Reactor Projects Section 3B Date

Inspection Summary:

Areas Inspected: Routine on-site resident inspection (132 hours) of plant operations, radiation protection, physical security, fire protection, surveillance and maintenance.

Results: This inspection identified satisfactory performance in all areas. A licensee identified violation involving inability to station a fire watch in containment was noted (Detail 2b). Corrective actions were found acceptable at present.

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DETAILS

1. Summary of Facility Status

The plant operated at full power except for brief periods for surveillance testing and/or preventive maintenance at reduced power.

2. Review of Specific Activities

Although plant operators and equipment performed as expected during this inspection period, several problems occurred and are summarized below.

a. Pressurizer Power Operated Relief Valves

PORV leakage problems developed during the beginning of this operating cycle. The valves were re-built during a maintenance outage which ended on August 16. However, leakage through both PORVs recurred. There are three postulated leakage paths within a PORV. Of these, leakage was found through damaged main seats and under the main seats. Repairs were made to both PORVs in this area during August. The third potential leakage path, through the valve solenoid operator, was not examined because internal replacement parts were not available.

One PORV, 3RCS*PCV 455A, was isolated on August 19, 1986 due to excessive through valve leakage. The PORVs are isolated from the pressurizer when their motor-operated blocking valves are shut. The blocking valves are remotely operated from the control room main control board. However, this requires operator action.

Leakage through the second PORV, 3RCS*PCV 456, reached an excessive rate and caused the licensee to isolate the valve at 1400 on October 29, 1986. The valve leakage, which had reached a rate of about fifty (50) gallons per hour, was causing an excessive amount of influent water to the radioactive liquid waste systems. This was due to the amount of makeup water to the pressurizer relief tank (PRT) to which the pressurizer safety valves and PORVs discharge. The tank is cooled by makeup of cool water and is then drained to the liquid waste system.

PORVs are not required in order to meet code requirements, and the Technical Specifications do not require PORV operability. The resident inspectors will continue to follow the licensee's actions in addressing this problem.

b. Turbine-Driven Auxiliary Feedwater Pump Steam Supply Valve

Two inadvertent trips of the turbine-driven auxiliary feedwater pump steam supply valve occurred on October 15 and November 8. In each case, the pump was not operating but was in a standby condition. Control room operators were alerted to this condition through a control board annunciator. Plant equipment operators were dispatched to the pump and re-set the steam supply valve. No personnel were found in the area and there was no apparent cause for the trip. Security data confirmed that no one had been in the area at the time in question.

The licensee is continuing to investigate these occurrences. Recent testing of the pump has been successful; the last was conducted on November 10. The resident inspectors will continue to follow auxiliary feedwater system performance during routine inspection.

c. Loss of All Turbine Plant Component Cooling Water

Both running Turbine Plant Component Cooling Water (TPCCW) pumps tripped at 0544, October 19, on low suction pressure. This occurred as system flow increased when valves were opened to an idle heat exchanger when the "A" isolated phase bus cooling coils were being placed in service. This transient condition was complicated by additional problems. The pumps continued to trip on low suction pressure due to the pressure changes created by pump start in a system without flow. The operators successfully restarted the pumps after isolating the low suction pressure instrument. Cooling water flow was restored before reaching a high temperature on generator stator cooling. Although TPCCW services systems and components which are not safety-related, the loss of cooling to systems such as stator cooling could result in a turbine trip within several minutes.

This event is an example of operations personnel effectively dealing with a problem caused by a design weakness. Resolution of design weaknesses will be addressed incident to routine NRC inspection.

d. Steam Generator Feedwater Flow Oscillations

At 3:00 p.m., 10/21, the loss of level within the "B" Fourth Point Feedwater heater caused the heater drain pumps to trip and resulted in feedwater flow oscillations on the No. 3 Steam Generator. Control room operators stabilized steam generator level by taking manual control of the feedwater regulating valve and restarting the heater drain pumps. The cause of the low heater level was identified to be erratic operation of the "B" First Point Heater normal level control valve. That was stopped by placing the heater drain valve controller in manual with the valve about 75% full open. First Point heater level is maintained at the set point of this high level drain valve. This has resulted in stable heater drain flow and has eliminated a source of steam generator level oscillations. However, until final corrective actions are taken with the normal level control valve, the plant suffers a loss in turbine steam cycle efficiency.

This event is an example of operations personnel effectively dealing with a steam generator feedwater transient.

Additionally, the licensee has identified condensing pot size as a plant-unique cause of past level oscillations at Millstone Unit 3, and is planning a design change to correct it. Oversized level instrument reference volumes will be replaced with units of a conventional size. The pipe connection to these devices were evaluated as insufficient for the condensate from the larger steam volume of the larger condensing pots.

3. Failure to Comply with a Technical Specification (TS) Action Statement

A portion of the station fire water system was out of service for one hour and forty-five minutes on October 30 for repairs after excavation damage to a fire main isolation valve. The licensee exceeded the one hour TS Action Statement for stationing a continuous fire watch at the containment cable penetration area, which is located within the sub-atmospheric containment. All other TS action statements were met through compensatory actions. The licensee made immediate reports to the NRC.

Action statement 3.7.12.2.a, applicable to Limiting Condition for Operation 3.7.12.2.k, states that if the Containment Cable Penetration Area Sprinkler is inoperable a continuous fire watch with back-up fire suppression equipment is required within one hour. The normal time required to enter the containment during reactor operation exceeds this. The licensee plans to address this, possibly through an application to change the license requirements.

During the incident, the inspector confirmed that the electrical penetration area was monitored by a fire detection system independent of the sprinkler system. Overall, this matter was classified by the inspector as a licensee identified item of minor safety significance, with corrective actions acceptable at this time. Resolution will be examined incident to routine inspection.

4. Licensee Action on Previous Inspection Findings

(Closed) Unresolved Item (UNR) 50-423/84-23-04, Effect of Corrosive Battery Room Atmosphere on Exposed Inter-Cell Cable Assemblies

This item resulted from inspector observation that the safety-related inter-cell battery cables and terminal lug assemblies did not meet the standard commercial practice of inserting the cable insulation within the lug shank and crimping the lug connection. The concern was that acid vapor could result in cable-to-terminal corrosion. Inspection report (IR) 423/85-23 documented that the inter-cell connections would be re-terminated after covering with Type WCSF-1000N Raychem heat shrink tubing in accordance with engineering and design coordination report (E&DCR) T-E-05027. The inspector observed the corrective actions taken in the battery rooms and confirmed that this work was completed.

(Closed) Inspector Follow Item (IFI) 85-21-01, Fully Develop Maintenance Training Program

When Inspection 85-21 was performed, the licensee was just beginning to develop the maintenance training program (MTP). The inspector has now reviewed the implemented MTP at the training center. The program, involving approximately 1000 hours per person, will take five years for all 132 maintenance personnel at the four Northeast Utilities (NU) facilities to be fully trained. INPO accreditation, requiring 70% of the lesson plans be developed and 20% of the training be completed for 20% of the staff, has been applied for.

NU maintenance uses many ex-Navy and NU system people with 5 to 10 years prior experience. The training labs are equipped with items such as a complete diesel (the same make as the Unit 1 and 2 diesel generators), a full size reactor coolant pump seal mock-up for Unit 2 (one for Unit 3 is on order), and extensive test equipment. This is in addition to a variety of pumps and valves.

MTP instruction is performed by six mechanical, six electrical, eight instrument and control, four chemistry, four health physics, and two quality assurance instructors. Instructors have been recruited from the operations/maintenance plant staff. In addition, four instructional technicians (may be reduced to one at a later time) are presently available to assist the instructors in material preparation.

The inspector found the maintenance training program to be acceptably established. INPO accreditation will be addressed separately as a matter of licensee performance.

(Open) UNR 50-423/85-35-03, Containment High Range Monitor Qualification

This item was left unresolved in IR 85-34 pending NRC review of qualification test data for the Kaman mineral-insulated cable/connector assembly. In IR 86-09, our review of the licensee's data contained in Qualification Report No. 460036-002 was documented. Seven examples of problem areas were listed in the report. The inspector found no new data available for NRC review. The item remains unresolved.

(Closed) UNR 50-423/85-53-08, Redundant Control Room Fuses

During a fire protection inspection (IR 85-53), redundant fuses to protect the transfer circuits between the main control panels and the auxiliary shutdown panels were recommended. The licensee committed to provide this fusing under E&DCR T-C-05936. The inspector confirmed that the redundant fuse installation work had been completed and documented.

(Closed) IFIs from 50-423/85-71, Surveillance Procedure Adequacy

The subject inspection, conducted just prior to licensing, left a number of minor items to be corrected by the licensee. NRC review had concluded that none of these items adversely affected the safe operation of the facility. During this inspection, the inspector selected 13 of the more significant of these items to confirm that appropriate licensee corrective actions had been taken. One of these, IFI 50-423/85-71-34, involving four typographical errors in surveillance procedure 3442A01 (Delta-T/Tavg Channel Calibration), had not yet been corrected. The licensee added the noted corrections to the in-process procedure revision. No discrepancies were found in the other 12 items. The failure to correct the typographical errors was evaluated as an isolated and minor discrepancy. These inspector follow items are closed.

(Closed) Violation 50-423/85-74-03, Failure to Maintain Escorts for Workers

As addressed in Region 1 letter dated June 26,1986 responding to NU's letter of April 7, 1986, in addition to the stated corrective action, the licensee should also consider strengthening the General Employee Training (GET) with respect to escort procedures. No further occurrences of this problem have been observed. The resident inspectors will evaluate GET and will monitor the use of escorts under the routine inspection program.

(Closed) Violation 50-423/85-74-04, Unlocked, Unattended Vehicle in Protected Area

This violation is resolved based on information provided by the licensee in their letter dated April 7, 1986 as supplemented by telephone conversations documented in Region 1 letter of June 26, 1986. No subsequent failures to lock unattended vehicles in the protected area have been observed. The resident inspector will continue to monitor this aspect under the routine inspection program.

(Closed) UNR 50-423/86-02-02, Tracking Technical Specification LCOs

The subject item involves failure to implement 12 hour grab sampling of the plant vent with the reactor plant ventilation discharge radiation monitor out of service, and failure to verify every seven days that containment protective backup breakers were tripped or the inoperable breaker racked out when containment penetration overcurrent protection devices were inoperable. As a result of these failures, in addition to continuing the logging of LCOs in the Shift Supervisor's log, the licensee added the listing of all active LCOs on the daily morning meeting notes. When a department other than operations is responsible for the required action, the Shift Supervisor notifies that department of entry into the LCO. In addition, LCO status is now an entry in the shift turnover log.

Since the occurrence of these failures to perform required surveillances, a surveillance required by TS 4.7.12.4. for Halon bottle weight was missed. However, this was due to misinterpretation of the TS and not to inadequate tracking. The inspector found the licensee's corrective actions (surveillance tracking) acceptable.

(Closed) UNR 50-423/86-07-02, Deletion of Two Startup Tests Committed to in FSAR Table 14.2-2

As addressed in IR 86-07, the licensee deleted the ejected rod test at 30% power level and the dropped rod test at 50% full power. IR 86-20 documented the acceptability of deleting the ejected rod test at 30% power level. Attachment 1 of licensee's March 12, 1986 letter provides the required safety analysis for deleting the rod drop/negative rate trip test at 50% power level, substituting an NRC recommended 10% load swing test. These changes are the subject of a proposed FSAR revision. The inspector had no further questions on this issue.

(Closed) UNR 50-423/86-08-01, Failure to Completely Document and Retest a Modification Done Under an E&DCR Results in Partial Safety Injection

This issue resulted from a partial safety injection during surveillance testing. Design change documents, implementing a modification, had been inadequately reviewed during work order preparation. That resulted in not making appropriate procedure changes. The licensee was to perform further review of other changes and Region I was to evaluate the licensee's design control process.

In LER 86-027, the licensee reported ten more missed procedure changes due to the same design change document. Their review showed no problems with other design changes. Inspection Report 50-213/85-15 (Haddam Neck) documents NRC review of NU's design control process. That report identified problems with NU's design review and safety evaluation procedures, onsite review committee activities, and implementation of prompt and effective corrective actions. NU's efforts to improve and refine the PDCR process were discussed in correspondence with the licensee. The inspector reviewed the records related to this issue and found corrective actions have been taken and that the NU design change program has been found acceptable. Region I inspectors will continue to monitor the E&DCR program under the routine inspection program.

(Closed) IFI 50-423/86-08-03, PORV Controller Modification

IR 86-08 documents the acceptability, per TMI TAP Item II.K.3.9., of PORV operation with the derivative time constant set "off." The possibility remains that the controller could keep PORV 455A closed if the pressurizer pressure is below 2250 psia for a short time prior to a condition when PORV opening is desirable.

The inspector reviewed Plant Modification Request 3-86-64 and a 5/8/86 Westinghouse Memo to the licensee (from J. Moore to J. Crockett) on the subject. Licensee corrective action was, per the Westinghouse recommendation, to add a step to EOP 35 ES-0.2 (Natural Circulation Cooldown Procedure) requiring the master pressure controller to be put in manual. Since the master controller only controls one PORV (the other valve operates on sensed pressure directly), automatic overpressure protection remains available during natural circulation cooldown. In addition, PORV 455A can be opened manually from the control room. The inspector had no further questions on this matter.

<u>(Closed) Violation 50-423/86-09-02, Administrative Controls of QA/QC Personnel Qualification</u>

This issue involved the assignment of a QC inspector to witness a startup test on which he was not trained or briefed. IR 86-20 documents that further review of this issue found draft controlling procedures QA 1312A (Guidelines for QA Surveillance Activities) and ACP-QA-9.07 (QA Surveillance Program) adequately respond to the violation. SORC approval was, however, needed. The inspector confirmed SORC approval and the issuance and implementation of the above procedures.

<u>(Closed) IFI 50-423/86-13-01, Nonradiological Chemistry and Measurement Control</u>

On completion of analyses of water samples by the licensee and Brookhaven National Laboratory, an evaluation was performed.

<u>Millstone Unit 3 - Split Samples</u>

		BNL	Millstone
Boron (ppm)	RWST	2130 +/- 10	2033 +/- 0
Ammonia (ppb)	Feedwater #3	112 +/- 6	153
	Feedwater #4	117 +/- 11	151
Hydrazine (ppb)	Feedwater 2A	33 +/- 1	24
	Feedwater 2B	34 +/- 0	23
Sulfate (ppb)	Steam Generator #5	57 +/- 4	13
	Steam Generator #6	35 +/- 7	13.1

The analytical comparisons were acceptable.

<u>(Closed) NRC Information Notice No. 85-45 - Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Designed Plants</u>

This Information Notice concerned potential seismic interactions between components within the flux mapping systems. This was received and analyzed by the licensee who reported their findings in accordance with 10 CFR 50.55(e) in letters dated July 26 and September 18, 1985. The flux mapping equipment supports were included within the original plant design to ensure that it did not become a gravity missile and interact with the seal table pressure boundary. However, evaluation of the original installation indicated that the anchors were not sufficient to limit the displacement of the tubing connection at the seal table. Because the reaction at the tubing to seal table interface was difficult to analytically predict, during plant construction additional supports were added by the licensee to ensure that the relative displacement between the flux mapping equipment and the seal table is limited to acceptable levels. The inspector had no additional questions.

(Open) Bulletin 85-03, Motor-Operated Valve Common Mode Failure Due to Improper Switch Settings

The licensee has revised their June 11, 1986 response to this Bulletin by letter dated October 28. This correspondence provided additional information which addressed the original procurement procedures and preoperational startup test program results. The licensee has committed to complete work on the Bulletin action items by two months after the end of the first refueling outage. That outage is scheduled to begin in the first calendar quarter of 1988. No problems with the licensee's program were identified. The NU actions on the concerns expressed in Bulletin 85-03 will be reviewed further at a later date.

5. Review of Events Occurring at Other Reactor Facilities

The inspector reviewed several events at other reactor facilities for applicability to Millstone Unit 3.

a. Single Failure Rendering a BWR Standby Gas Treatment System Inoperable

An engineering review of the Pilgrim Nuclear Station Standby Gas Treatment System (SGTS) identified a design deficiency which could render the system inoperable in the event of a single failure.

The system design weakness involves cross-tie valves between the redundant filter units. These were intended to provide cooling to a standby filter. A single failure in the filter fire detection and suppression system could render one such system incapable of removing radioactive iodine from the gaseous effluent. The inspector reviewed the design of the Millstone Unit 3 Reactor Plant Ventilation and Containment Structure Ventilation System Filters. There were no similar design weaknesses identified.

b. Failure of General Electric Type AK-F-2-25 Circuit Breakers to Trip

Four failures of General Electric Type AK-F-2-25 breakers to open upon demand have occurred at the Pilgrim Nuclear Station during the last three years. Mechanical binding of the trip shaft prevented the breakers from tripping and resulted in shunt trip coil failure.

The inspector found that the AK-F-2-25 breaker used at Unit 3 as the Main Generator Exciter Field Breaker. In addition, a type AK-F-2E is in use as the Generator Field Breaker. Type AK-R-30 breakers are used throughout the D.C. distribution system. There have been no Pilgrim-type failures recorded for Millstone Unit 3 equipment. The licensee was provided with detailed information concerning the failures at the Pilgrim Station. The inspector had no further questions at this time.

c. Seismic Qualification of General Electric Type HGA Relays

During seismic qualification testing of the General Electric Type HGA relays for Susquehanna Nuclear Station, a problem concerning contact chatter was identified. This relay is in use at the Millstone Station. As follow-up on the information provided by the inspector, the licensee located all of these devices installed in the plant through a computer data base. There are 288 HGA relays at Millstone 3, with 115 of these in Class 1E circuits (mostly annunciators). This licensee is still assessing this matter, which will be readdressed during routine inspection.

6. Review of Licensee Event Reports (LERs)

LERs submitted during this report period were reviewed. The inspector assessed LER accuracy, whether further information was required, if there were generic implications, adequacy of corrective actions, and compliance with the reporting requirements of 10 CFR 50.73 and Administrative Control Procedure ACP-QA-10.09. Selected corrective actions were checked for thoroughness and implementation as documented elsewhere in this report. The LERs reviewed were:

86-052-00, Failure to Perform Surveillance Within the Required Frequency

The licensee discovered that a surveillance was not performed within the required interval. This surveillance verifies the operability of the Instrument Rack Room under-floor area fire suppression system. Surveillance 4.7.12.4 requires that the system Halon storage tank weight and pressure be verified to be at least at 95% of full charge and 90% of full pressure, respectively. The six-month interval for this requirement ended on August 26 but the surveillance was not done by September 18. At that time, the licensee declared the system inoperable and stationed a fire watch in the area, which is adjacent to the control room. The surveillance was completed on September 19; the system was found to meet the Technical Specification. The licensee identified, as a contributing cause, the unusual circumstance of the transfer of responsibility for this surveillance between two departments being in process. This item was evaluated by the inspector as licensee identified and appropriately corrected.

There were no other problems identified.

86-053-00, Incorrect Intermediate Range Monitor (IRM) Setpoints

As the result of a third party review of Reactor Protection System (RPS) setpoints, the licensee discovered that the setpoints for the IRM were less conservative than allowed by Technical Specification 2.2.1, Table 2.2-1.5. This review was conducted as the result of earlier problems in establishing the overtemperature differential temperature trip setpoint. These were addressed by LER 86-047-00 and NRC Inspection Report 50-423/86-28 paragraph 2.a. As a result of those discoveries, the licensee committed to this third party review of all RPS and Engineered Safety Feature setpoints.

The IRM discrepancy was the only third party review item for which a trip system setpoint less conservative than that allowed by the Technical Specifications. In this isolated case, the IRM setpoints were established on October 31, 1985 based on information provided by the Nuclear Steam Supply System (NSSS) vendor, Westinghouse. These setpoints were not verified during the power ascension test program as required by the NSSS Startup Procedures. At the time, it would have been possible to determine the proper detector current which corresponds to 25% of Rated Thermal Power and recalibrate the instruments. This error was discovered on October 15, 1986. At that time, the reactor was operating at full power. Technical Specification Action Statement 3 for Table 3.3-1.5 does not apply above 10% rated thermal power. This allowed the licensee to implement changes to the IRM calibration procedures which now establish Trip Setpoints within the allowed range of Specification 2.2.1. The licensee's analysis indicates that, in their original configuration, the two IRM Channels would have reached their high flux trip setpoint at 32% and 37% (instead of 25%). The safety significance was addressed in LER 86-053-00, which identified a redundant trip as being the power range trip set for 25% power.

The inspector reviewed the results of this setpoint study and attended the Plant Operations Review Committee meeting at which it was presented. This study recalculated the RPS and ESF trip setpoints and determined any differences between values stated in the Technical Specifications, Westinghouse documents and the calibration procedures, including those in a conservative direction. This study also included a verification of the Westinghouse Setpoint Methodology for Protection Systems (WCAP-10991) for Section 9, "Overtemperature Differential Trip Setpoint". Corrections needing to be made to this document have been provided to Westinghouse by the licensee.

The inspector found that this setpoint study was a detailed licensee effort to establish setpoint calculation documents and provide accurate and uniform data based on design, calibration and regulatory documents. There were no unacceptable conditions identified.

7. Safety Committee Meetings

The inspector attended Plant Operations Review Committee (PORC) meetings on October 30 and 31 and also a Nuclear Review Board meeting on October 22. Technical Specification requirements for attendance were met. The meetings were characterized by frank discussions and questioning of causes and corrective actions. Individual members were encouraged to provide their opinions. Both Committees demonstrated a good ability to track internal commitments and followup specified corrective actions. No deficiencies in Committee performance were observed.

8. Management Meetings

During this inspection, periodic meetings were held with senior plant management to discuss the inspection scope and findings. No proprietary information was identified as being in the inspection coverage. No written material relating to inspection findings was provided to the licensee by the inspector.