U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NOS.: 50-245/93-33
50-336/93-29
50-423/93-30

LICENSE NOS.: DPR-21
DPR-65
NPF-49

LICENSEE: Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

FACILITY NAME: Millstone Nuclear Power Station, Units 1, 2 and 3

INSPECTION AT: Berlin and Waterford, CT

INSPECTION DATES: November 29 - December 3, 1993

INSPECTOR: Aniello L. Della Greca 1/8/94
Aniello Della Greca, Sr. Reactor Engineer — Date
Electrical Section, EB, DRS

APPROVED BY: William H. Ruland 1/8/94
William H. Ruland, Chief — Date
Electrical Section, EB, DRS

<u>Areas Inspected</u>: Engineering staffing and performance related to the plant modification process and root cause analysis.

<u>Results</u>: The inspector's review of the engineering staff found that the licensee had been undergoing a reorganization and, therefore, the effectiveness of the new organization could not be fully evaluated. However, the move of the technical staff to the plant sites and the establishment of system engineering groups were viewed as positive steps toward improving the efficiency of the staff and the safety of the plant. The engineers interviewed were considered knowledgeable and capable of providing technical support to the operations organization.

Based upon the plant design change request packages reviewed, the design change and plant modification process was considered acceptable. Evaluations were thorough and supported by adequate documentation. No safety concerns were identified in this area. However, the settings of the DWST level and other TS limits require further NUSCo review to ensure that they are not exceeding TS limits. The inspector also identified no major concerns in the corrective action and root cause analysis programs. However, he found that none of the five licensee event reports reviewed was supported by a formal root cause analysis. Benefits could have been drawn from a formal analysis at least in the case of the missed surveillance.

DETAILS

1.0 PURPOSE AND SCOPE

The purpose of the inspection was to review the current organization of the Northeast Utilities Service Company (NUSCo) engineering staff and to determine their effectiveness in providing technical support to the safe operation of the Millstone Nuclear Power Station. The inspection focused primarily on the plant modification process and on the corrective action program for licensee event reports (LERs). The inspection included interviews of management and engineering personnel and direct observation of licensee program results.

2.0 ENGINEERING ORGANIZATION AND STAFFING (37700)

To address various concerns regarding the performance of the nuclear organization, Northeast Utilities initiated a review that resulted in the formulation of their Performance Enhancement Program, a six year plan with four specific goals: (1) safe operation of the nuclear plants; (2) operational excellence; (3) category 1 rating in the NRC systematic assessment of licensee performance (SALP); and (4) cost efficient operation. The plan, divided in three phases, included a reorganization of the engineering staff directed to improve the utilization of human resources and provide better technical support of the plant operations.

At the time of the inspection, the reorganization was undergoing its final stages. The organizational chart provided indicated five directors of engineering, one for each plant and the corporate staff, reporting to the Vice President of Nuclear Engineering Services. Also reporting to the Vice President were several managers of special projects. The plant organizations were similarly structured, with the engineering and technical staff divided into various groups, each headed by a discipline supervisor and reporting to two managers, the Design Engineering Manager and Technical Support Manager. The organizations included newly established system engineering groups, both in the Design Engineering and in the Technical Support Sections. The corporate staff, responsible for programs applicable to all plants, was also divided into groups, each headed by a supervisor, and reporting to four managers, Nuclear Fuel Engineering, Radiological Assessment, Safety Analysis, and Engineering Support. The staff of each plant currently consisted of approximately one hundred engineers and technicians with Unit 3 having the largest staff. The groups were manageable, typically composed of less than ten engineers with a few larger groups.

Because of its new structure, the effectiveness of the new organization was not evaluated. However, discussions with NUSCo management personnel indicated that methods were or would be in place to measure the performance of the engineering organization and to make the necessary adjustments. The inspector viewed the move of most of the technical staff to the plant sites and the establishment of system engineering groups as positive steps toward improving the efficiency of the staff and the safety of the plant. Individually, the engineers interviewed were considered to be knowledgeable in their area of expertise and capable of providing the necessary technical support to the plant operations organization.

3.0 DESIGN CHANGES AND PLANT MODIFICATIONS (37700)

To ascertain that design changes and plant modifications were performed in accordance with the requirements of the Code of Federal Regulations and of plant licensing documents, the inspector reviewed the current procedures and selected modification packages from each of the three plants. The review also evaluated the technical quality of the modifications, the thoroughness of the design analysis, design input, technical reviews and safety evaluations, and management involvement in the resolution of problems.

The Millstone design modification process is described in Administrative Control Procedure ACP-QA-3.10, also identified as Nuclear Engineering and Operations Procedure NEO 3.03. This procedure establishes a uniform method for performing plant design change records (PDCRs) at all the Millstone plants. It includes the detailed method by which PDCRs are prepared, reviewed, and dispositioned. The process by which safety evaluations ensure that proposed plant changes do not compromise the plant safety and satisfy the requirements of 10 CFR 50.59 is included in Procedure NEO 3.12.

The above procedures are applicable to all plant changes, both minor and major. For simple changes of limited scope, Procedure NEO 3.03 includes a short form, Form C, which simplifies the modification process by asking appropriate questions regarding the impact of the modification on systems and components important to safety. The instructions, questions and references provided within the procedure were considered adequate for a proper review, processing, and closure of all design changes and plant modifications.

To evaluate the implementation of the procedure, the inspector reviewed the following PDCRs from the three plants: (1) No. 1-83-92, pertaining to the replacement of the low pressure coolant injection pump motors with larger ones; (2) No. 1-18-93, involving the dedication of a heating oil storage tank to the storage of reserve diesel fuel and the installation of necessary piping; (3) No. 1-44-93, regarding the replacement of diesel generator lube oil pressure switches with similar ones; (4) No. 2-123-91, related to the replacement of six inverters and associated static switches with new ones; (5) No. 2-028-92, pertaining to the addition of a high containment pressure signal to the main steam line isolation logic; (6) No. 2-140- 92 involving the replacement of various diesel generator components to support the conversion of the engine from a parallel to a series air intake system; (7) No. MP3-91-181, pertaining to the replacement of inverter capacitors with similar capacitors; (8) MP3-91-216 related to the setpoint change of the demineralized water storage tank (DWST) level alarms; and (9) No. MP3-92-108 regarding the installation of two Agastat timers into motor control centers to improve the operation of the auxiliary building filtration system.

A review of the above PDCRs, involving both major and minor modification processes, determined that: the changes were well documented; the packages contained adequate design details and pertinent calculations; the design analyses adequately addressed the reason for the change; technical reviews and safety evaluations, where applicable, were thorough; and procedural requirements had been followed. No areas of concern were identified, except as follows:

PDCR MP3-91-216 was initiated to change the DWST high level alarm and, thus, prevent the recurrence of water spillage previously experienced. The spillage was the result of the alarm setpoint not accounting for normal loop inaccuracy of ± 11.6 inches. The lowering of the high level alarm setpoint caused a corresponding reduction of the range to which the tank level could be kept (less than one inch) without causing a low level alarm. To avoid a nuisance alarm, the low level setpoint was also lowered. However, in so doing, the modification potentially challenged the technical specifications (TS) limit for minimum demineralized water storage. Specifically, Calculation NSP-098-FWA determined the minimum required volume to be 324,794 gallons, corresponding to a level of 45'-1.5" above the tank bottom. Taking into account the loop error (11.6"), Section 4.7.1.3.2 of the TS set the minimum volume at 334,000 gallons of water, corresponding to a level of 46'-5". For the same reason, the original calculation set the low level alarm at 47'-8.4". The modification lowered this setpoint to 46-9". The following table describes the applicable old and new settings in height and volume units.

	OLD SETTINGS		NEW SETTINGS	
Overflow	48'-9"	350,854 g	48'-9"	350,854 g
High Level Alarm	48'-7"	349,654	47'-9"	343,657
Low Level Alarm	47'-8.4"	343,297	46'-9"	336,460
Tech. Spec. Minimum	46'-5"	334,000	46'-5"	334,000
Minimum Required Volume	45'-1.5"	324,794	45'-1.5"	324,794
Instr. Loop Error	± 11.6"	6,957	± 11.6"	6,957

The modification had considered the effect of lowering the alarm setpoint and recognized that, with the water level just above the alarm setpoint, the tank volume could be below the TS limit. However, the evaluation concluded that the condition was acceptable because that level was still above the calculated safety limit by a margin of approximately 5000 gallons. Discussions with the NUSCo engineering indicated that the case may be applicable to other instrument loops but felt that the method used met the intent of the requirements of ISA standard S67.04-1982 and the recommendations of Regulatory Guide 1.105. The licensee also indicated that, for the system in question, additional demineralized water was available from the condensate storage tank. Because the old low level alarm setpoint included the calculated instrument loop error twice, the inspector concurred that its setting might have been too conservative. However, it's lowering to a level that could challenge the TS limit, although not safety-significant, was considered unacceptable. Similarly, the licensee reliance on the control room level indicator to establish minimum acceptable normal operating level was considered unacceptable because the indicator was subject to a similar instrument loop error. This issue is unresolved pending NUSCo's evaluation of the acceptability of the TS limits for the DWST level, as well as other safety-significant parameters set using the same methodology (50-423/93-30-01).

4.0 CORRECTIVE ACTION AND ROOT CAUSE PROGRAM

A vehicle used by NUSCo to correct programmatic or quality issues is the corrective action requests (CAR). The CAR method is used when available procedures are not adequate to prevent recurrence of the identified problem, or no procedure is available to correct it. The process to document, evaluate and resolve a condition adverse to quality is described in Nuclear Engineering and Operations Procedure NEO 2.18.

The procedure that describes the root cause investigation process is ACP-QA-10.12. This process can be initiated by a plant information report (PIR), an audit finding, or any other condition where a root cause investigation is considered necessary by the station management. The PIR is the means to identify, document, investigate, and correct a condition that requires the involvement of plant management or reporting to an external agency. The PIR process is described in ACP-QA-10.01. For conditions requiring an independent root cause investigation, NUSCo developed Nuclear Safety Engineering Procedure NSE 7.01. The inspector's review of the above procedure found the instructions to be clear with adequate charts and references.

To determine the efficacy of the procedures, the inspector reviewed several licensee event reports from the three Millstone units and discussed the resulting corrective actions with responsible engineers. The LERs were selected because they constitute one of the conditions that initiate the PIR process. The LERs reviewed included 91-004 and 92-030 from Unit 1; 91-009 and 93-014 from Unit 2; and 91-030 from Unit 3. Corrective actions to resolve the issues described by the LERs were adequate. However, the inspector made the following observations:

1. None of the LERs reviewed resulted in a formal root cause investigation.

2. Following the issuance of LER 91-009, the Unit 2 engineering continued its investigation of the problems experienced in conjunction with the diesel generator 12U load control and determined that the root cause to be different from the one identified in the LER. Yet, no revision of the LER was issued.

In the LER, the cause of the problem was attributed to large resistance value changes in the droop potentiometer associated with the governor controller, based on observations during the plant tests. As stated in the LER, the voltage swings observed by the licensee could not be repeated in laboratory tests by the manufacturer. Subsequent NUSCo investigation determined that the load control problems were due to oxidation of relay contacts in the governor controller unit. The wiping action during the opening and closure of the contacts had rendered the problem difficult to detect and repeat in laboratory experiments. Further discussions with NUSCo engineering indicated that they had replaced the governor of one diesel and that they would be replacing the other. The new governor used an enclosed type relay. The licensee also indicated that the investigation of this issue was still ongoing and that the LER would be revised.

3. LER 93-014 was issued to report a missed quarterly surveillance of a recirculation fan and a block valve required by the Unit 2 Technical Specifications, Section 4.0.2. A review of this LER by the inspector revealed that the TS required surveillance had been missed five more times in the last three years. A review of each LER involved did not appear to suggest a common cause. However, the inspector expressed a concern that root cause may not have been found. Discussions with the licensee scheduling personnel suggested that proper mechanisms were in place to identify schedules and that the missed surveillance could have been the result of oversight or human error.

5.0 EQUIPMENT MAINTENANCE AND TESTING

During the electrical distribution system functional inspection of the Millstone, Unit 3, facility, the inspection team also reviewed the results of battery discharge tests for the Units 1 and 2 station batteries. This review revealed that, at the end of the May 4, 1991, performance service test, the voltage of Unit 1 station battery 18B was 106.2 Volts, only 1.2 Volts above the TS required minimum. The team raised a concern regarding the capability of the battery to provide, two years later, minimum voltage to the its safety-related loads.

During the current review, the inspector discussed the finding with NUSCo engineering and determined that the low voltage had been recorded after an eight hour discharge test. Because of concerns regarding the capability of the batteries to deliver the required load for an eight hour period, the licensee had reevaluated the Technical Specifications requirements and had previously submitted to the NRC a TS change request, reducing the battery availability period from eight to two hours. The performance service test should have been conducted for two, not eight hours. A review of the TS confirmed the required availability period to be two hours. The inspector had no further concerns in this area.

During a walkdown of the Unit 2 Control Room, the inspector observed that the low range pressurizer pressure indicator, PI-103-1B, appeared to be reading slightly below 1600 psi (approximately 1580 psi). Since the normal operating pressurizer pressure is well above 1600 psi, the inspector discussed his observation with the senior control room operator who indicated that the instrument was calibrated for the low pressure range and, therefore, the reading was within tolerances. He also reviewed the computer output for the instrument loop and found the pressure reading to be 1590 psi.

To ensure that the observed indication did not adversely impact any safety-related component, the inspector reviewed the instrument loop diagram and the calibration records for the applicable devices. The inspector determined that both the indicator and the voltage-to-current converter had been recently found out of calibration. The meter itself was replaced in April 1993, because it could not be adjusted. The inspector also determined that the indicator itself was not safety-related and that it received its signal from a signal limiter and a voltage-to-current converter. These components, although classified as category I, did not provide an output to safety-related instruments and, therefore, performed only an isolation function. Based upon the review, the inspector had no further concerns with the observation.

6.0 EXIT INTERVIEW

At the conclusion of the inspection, on December 3, 1993, the inspector met with the licensee personnel denoted in Attachment 1. At that time he summarized the purpose and scope of the inspection and identified to the licensee the findings discussed within the body of this report. The licensee acknowledged the findings without comments.

ATTACHMENT 1

Persons Contacted

Northeast Nuclear Service Company

	D. Basler	Plant Engineer
	G. Bohn	Plant Engineer
	P. Cassidy	Unit 2 Operations Technician
	S. Cohen	System Engineer
	J. DeLaCruz	Maintenance Engineer
	R. Ewing	Senior Engineer, MP1 PSD
*	K. Hannon	Associate Analyst, Nuclear Licensing
*	J. Hickman	Senior Engineer, MP3 PSD
	E. Lindsay	Lead Reactor Engineer
	M. Martell	Senior Reactor Engineer
*	R. Necci	Director, MP2 Engineering
	G. Olsen	Electrical Engineering Supervisor
	G. Pitmans	Director, MP3 Engineering
	J. Plourde	Plant Engineer
	R. Poole	Unit 2, Maintenance Planner
	H. Risley	Director, MP1 Engineering
	R. Sholler	System Engineer
	S. Sudigala	Engineering Supervisor
*	G. VanNordennen	Supervisor, Nuclear Licensing

* Indicates personnel attending the exit meeting on December 3, 1993.