U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee:	North Atlantic Energy Service Corporation Post Office Box 300 Seabrook, New Hampshire 03874
Facility:	Seabrook Station
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Inspection Summary: This inspection report documents the safety inspections conducted during day shift and back shift hours. The inspections assessed station performance in the areas of plant operations, maintenance, engineering, plant support, and safety assessment/quality verification.

Results: No violations were identified. One unresolved item was identified involving inconsistencies between the final safety analysis report and operation procedures for remote safe shutdown system components and labeling. See the executive summary for a general ssessment of licensee performance.

EXECUTIVE SUMMARY

SEABROOK STATION
NRC INSPECTION REPORT NO. 50-443/94-05

Plant Operations: The operators performed well during routine and non-routine activities. Operators and plant management closely monitored an increasing reactor coolant pump seal leak-off trend. Plant management ordered that the unit be shutdown one week early to commence the third refueling outage. This decision reflected a safety conscious approach to plant operations. Operators shutdown and cooled down the unit to cold shutdown in a well controlled manner. Some component and labeling inconsistencies exist between the updated safety analysis report and the procedures involving remote safe shutdown system operation. Excellent use of shutdown risk management was observed, as evidenced by the decision to fully off-load the core before entering into mid-loop operations. Operations management also briefed operators on the lessons learned from refueling activities at other facilities.

Maintenance: With one notable exception, routine and non-routine maintenance and surveillance activities were performed well. Improper procedural adherence and poor oversight controls were in evidence during the performance of the maintenance activity to open both containment airlock doors after the plant entered a cold shutdown status. Consequently, several workers received minor injuries from the air flow through the hatch caused by a pressure differential that was not properly relieved. Additionally, the lack of a comprehensive set of controls for scaffolds built near safety-related equipment was identified as a weakness. Licensee changes to the scaffold controls program addressed these concerns during this period.

Engineering: Engineering personnel properly evaluated and resolved the following emergent issues: site specific tornado analysis, 120MB incandescent lamp failures, and polar crane modifications. Particularly noteworthy was the engineering manager's involvement in addressing the concerns raised with respect to the adequacy of existing scaffold controls.

Plant Support: Health physics personnel reacted well to an unexpected plant system crud burst and were involved in the root cause investigation. The health physics and chemistry groups institutionalized a self-assessment process. The security force continued to perform routine and non-routine duties in accordance with program requirements. The emergency preparedness staff performed an off-hours drill in a professional manner and identified opportunities for improvement with a comprehensive self-critique.

Safety Assessment/Quality Verification: The new occurrence review committee process provides management with timely and meaningful, reactive performance trends. The process assigns a significance level, an assessment, and is significantly better than previous trend processes. The quality group interface meeting with line management appears to provide a format more conducive to effective communications than the previous finding review board format. Opportunities exist to further improve the information exchange between quality personnel and line management.

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DETAILS

1.0 PLANT OPERATIONS (71707,60705,93702)

1.1 Plant Activities

At the beginning of this inspection period, the reactor was operating at 100% power. On April 4, the operators commenced a normal shutdown due to increasing leak-off from the number one seal in the "D" reactor coolant pump (RCP). During the shutdown, the leak-off rate decreased and stabilized. The operators terminated the shutdown with the reactor at approximately 87% power.

On April 7, plant management decided to commence a normal plant shutdown to start the third refueling outage (OR03). The main generator breaker was opened shortly after midnight on April 9, signaling the start of OR03. The refueling outage had been originally scheduled to start on April 16. The operators shutdown and cooled the plant down to operational mode 4, (hot shutdown) on April 9. On April 10, the operators further cooled down the plant to operational mode 5 (cold shutdown) where the plant was maintained for the remainder of this inspection period.

1.2 Routine Plant Operations

The inspector conducted daily control room tours, observed shift turnovers, attended the morning station manager's meeting, and monitored plan-of-the-day meetings. The inspector checked and confirmed that operational activities were being performed in accordance with technical specification requirements. The inspector conducted tours in the primary auxiliary building, the emergency diesel generator rooms, the residual heat removal vaults, the electrical switchgear rooms, the emergency feedwater building, the fuel storage building, and the condensate storage tank area. During the tours and attendance at the various meetings, the inspector noted generally good implementation of work controls over plant activities and an overall good performance, including cognizance of the current plant configuration, by the operations staff.

During control room tours, the inspector observed that operators promptly responded to alarms and had only a few hard wired and video annunciator system (VAS) alarms remain lit. The inspector verified the adequacy of two tagging orders.

1.3 Abnormal Reactor Coolant Pump Seal Leak-off Flow

During the early part of this inspection period, the seal leak-off flow by the #1 seal for all four reactor coolant pumps started increasing at a slow trend. This phenomenon had been observed during other similar pre-refuel periods at Seabrook, with the reactor core at end of life and boron concentrations decreasing to adjust for the reactivity changes resulting from fuel depletion. The "D" reactor coolant pump (RCP) #1 seal leak-off rate was noted to be higher than the other three pumps and trending up at a higher rate. The licensee projected that preemptive measures would be necessary to prevent the flow from exceeding off-normal operating limits, requiring a power reduction and stoppage of the RCP. This, in turn, would require a reactor shutdown in accordance with the Technical Specifications, with a reactor coolant loop out of service.

The licensee evaluated the effects of certain plant conditions and variables (e.g., seal injection temperature and filter delta-pressure, volume control tank pressure) on the "D" RCP seal leak-off flow rate. With two parallel seal injection filter flow paths to the RCP seals, periodic swapping of the injection filters, with replacement of the isolated filter, was determined by the licensee to be the most effective mechanism for minimizing the increasing trend in the seal leak-off flow rate. Procedurally, the "RCP Malfunction" procedure, OS1201.01, discusses the criteria for RCP #1 seal leak-off operating ranges and controls. While the normal operating range at full power condition is 1 to 5 gpm, continued operation is allowed up to 6 gpm. The licensee, in consultation with the RCP manufacturer (Westinghouse), also processed a change to procedure OS1201.01 allowing the upper limit for the "D" RCP #1 seal leak-off to be raised to 7 gpm. This was accomplished in conjunction with implementation of a temporary modification, 94TMOD0011, to allow for local monitoring of the "D" RCP seal return flow up to 8 gpm and was coordinated with the issuance of Standing Operating Order (SOO) No. 94-12.

The inspector reviewed SOO 94-12, discussed it with operations personnel, and observed the briefing of different operator shift crews on the implementation details. The inspector noted that even though the upper limit for seal leak-off flow was extended to 7 gpm, operator contingency actions were written to commence at a leak rate of 5.5 gpm in accordance with the SOO. The inspector also reviewed 94TMOD0011 and attended the Station Operation Review Committee meeting on March 25, which approved the temporary modification. NRC Information Notice 93-84 describes operating procedure limitations related to the determination of Westinghouse RCP seal failures. Attached to IN 93-84 is Westinghouse Technical Bulletin NSD-TB-93-01-R0 that provides revised procedural guidelines for RCP shutdown with #1 seal leak-off outside specified operating limits. The inspector questioned whether this guidance had been considered by the licensee in evaluating current seal leak-off conditions. The licensee provided evidence of evaluation of the Westinghouse recommendations in 1993. However, a Westinghouse preference to have RCP shutdown precede #1 seal leak-off isolation had not been incorporated into the current Seabrook RCP abnormal response procedure. The licensee discussed this sequence of operations with Westinghouse personnel and subsequently included in a change to procedure OS1201.01 the preferred order of steps for stopping a RCP and isolating the leak-off line.

The licensee continued to monitor "D" RCP #1 seal leak-off flow, swapping seal injection filter flow paths as necessary, to control the leak rate within the normal operating range of the RCP. The inspector observed control room evolutions during periods of filter train swap-over. On March 24, a spike near 5 gpm was observed as the filter flow paths were shifted because of bypass flow in the isolated filter train. The operators quickly responded to this situation by again changing filter flow paths and bringing the leak rate down to 4 gpm. Plant operators were able to control the seal flow below the off-normal leak-off range level of 5 gpm until April 4 when a spike slightly above 6 gpm was observed.

In accordance with directions provided from the plant manager, operators commenced a controlled plant shutdown at a reduction of approximately 2.5% thermal power per hour in response to the observed leak-off conditions. The inspector noted that while SOO No. 94-12 provisions were followed, the plant manager had conservatively set the 6 gpm limit for initiation of a plant shutdown in order to preclude any need for performing a rapid power reduction, as would be required by OS1201.01 if a 7 gpm leak-off rate was reached. The inspector witnessed the initiation of the plant shutdown on April 4, confirming compliance with the power decrease requirements of procedure OS1000.06. The inspector also observed that with the initial boron addition to the reactor coolant system for power reduction, in conjunction with another filter train swap, the "D" RCP leak-off flow rate decreased to approximately 5.6 gpm very quickly and continued to trend down as the power reduction continued.

During the evening of April 4, the shutdown was suspended at approximately 87% thermal power with the seal leak-off rate stabilized at approximately 5.3 gpm. This decision was made by plant management based upon an evaluation of current seal conditions, in consideration of the plant readiness for commencing refuel outage activities. The power reduction resumed, as planned, on April 7 with the opening of the generator breaker (signaling the start of the refuel outage) shortly after midnight on April 9.

The inspector witnessed shutdown evolutions on April 7 and 8 and continued cooldown activities on April 10, verifying operator cognizance and coordination of power reduction targets with Xenon concentrations and boric acid additions to the reactor coolant system. The inspector witnessed control rod (Bank "D") insertion, as necessary, to maintain the core power distribution within the axial flux difference (AFD) target band. Overall, the plant shutdown proceeded on schedule with evidence of good controls. With regard to the licensee response to the "D" RCP #1 seal leak-off problems, the inspector determined that licensee operations and technical support personnel properly trended and reacted to changing flow conditions. Additionally, plant management assessed the competing variables and plant impact considerations in deciding to shutdown the plant one week early to begin OR03. This indicated not only a safety conscious attitude, but also evidence that the licensee intended to gain control of the anomalous leakage in such a way to avert the need for a rapid power reduction, and thus conduct a methodical, controlled plant shutdown, as was observed by the inspector.

During OR03, the "D" RCP seal assembly will be replaced. The licensee intends to contract for additional analysis of the as-found #1 seal conditions to determine the cause for the observed abnormal leak-off flows. Additionally, two design coordination reports (DCRs 94-13 & 94-14) are being developed to permanently install an expanded RCP #1 seal leak-off flow measurement (i.e., 0-10 gpm) capability and an enhanced #2 seal leak-off flow measurement indication. These DCRs are being considered for installation during OR03 and would support further usage of the guidelines of the Westinghouse Technical Bulletin, NSD-TB-93-01-R0. These improvements are intended to facilitate the overall plant and operations response to abnormal RCP seal leak-off conditions that might arise in the future.

1.4 Engineered Safety Feature System Walkdown - Remote Safe Shutdown (URI 50-443/94-05-01)

The inspector reviewed section 7.4 of the updated final safety analysis report (UFSAR), which discusses the Seabrook systems required for safe shutdown and maintenance of the reactor in cold shutdown conditions. Shutdown of the plant from outside the control room is controlled from specifically designated remote safe shutdown (RSS) locations. Additionally, certain manual operations (e.g., valve manipulations) are required locally at the field location of designated equipment. Table 7.4-1 of the UFSAR lists the equipment required for safe shutdown, delineating the safety function (e.g., decay heat removal) provided by each component, as well as the RSS location designated for component manipulation and control.

The inspector examined each of the six major control panels, designed for RSS functions, as well as the electrical switchgear and distribution equipment assigned an RSS role. The field locations, inside and outside the radiologically controlled area where RSS components were designated, were spot-checked. In accordance with the UFSAR description and procedural provisions, the inspector verified it possible to manually trip the reactor, isolate the main steam lines, and trip the reactor coolant pumps from RSS locations, if these actions were not accomplished, as intended, prior to control room evacuation. The inspector also checked the RSS capability to defeat the solid state protection system from circuit breakers located in power panels in the separate "A" and "B" train switchgear rooms. During these field walk-downs, the inspector examined the designated components for the placement of specifically designed, RSS color coded labels.

The inspector also reviewed the abnormal operating procedures, OS1200.02, 02A & B, describing the safe shutdown and cooldown from the remote safe shutdown facilities. The inspector checked for consistency between the different procedures (e.g., train "A" vs. train "B") and compared the equipment prescribed in the procedural steps with the components listed in UFSAR Table 7.4-1. The RSS panels, MM-CP-108 A & B, were checked in detail to confirm the clarity of procedural directions for operations from these primary RSS locations. Certain field equipment locations were also spot-checked. While no procedural-related violations or unacceptable RSS equipment conditions were identified, the inspector did raise certain questions in areas relative to the UFSAR/Procedure/Tagging interface of RSS components, as follows:

- inconsistencies between the UFSAR and OS1200.02 series procedures, e.g.,
    - intermediate head ECCS pumps, SI-P-6 A & B, are listed in UFSAR Table 7.4-1, but are not procedurally required as RSS equipment.
    - spent fuel cooling pumps, SF-P-10 A & B, and thermal barrier cooling pumps, CC-P-322 A & B, are operationally required in the RSS procedures, but are not listed in UFSAR Table 7.4-1.
- an inconsistency between the abnormal RSS procedures, in that OS1200.02 requires checking and/or repositioning the emergency boration flow-path valve, CS-V-426, but OS1200.02B does not list it as a component checked or placed in local control.
- instruments listed in OS1200.02 for checking condensate storage tank level (i.e., LIS-4052, FW-PI-4208 & 4209) are not marked with color coded RSS labels at their field locations.

Pending licensee clarification of the identified inconsistencies, determination whether the labeling of additional RSS equipment is required and further review of these questions by the NRC, the above issues involving the RSS system are collectively classified as an unresolved item. **(URI 50-443/94-05-01)**

1.5 Refueling Outage Preparations

The inspector performed a review of licensee activities to determine if the lessons learned during refueling operations at other facilities, as documented in NRC Information Notice (IN) 94-13 (dated February 22, 1994), were intended to be applied to OR03 operations and planning. During this review, the inspector evaluated operations department refueling procedures, the use of shutdown risk management applied to the upcoming refueling outage, and the final OR03 scope reduction effort, which was implemented to limit schedular impact, while improving overall work quality.

NRC IN 94-13 describes recent refueling operational events that occurred at four other facilities, which suggest that increased licensee management attention is warranted. These other facilities banged, dropped, and/or improperly positioned fuel assemblies. The inspector held discussions with the operation department manager. The manager knew of the industry issues and made a videotape that captured the lessons learned. All Seabrook operating crews viewed the videotape. The inspector also watched the videotape and confirmed that it covered all the issues contained in IN 94-13. The operations manager indicated that North Atlantic has never dropped or banged a fuel assembly, but was cognizant of the need for vigilance in this area. Currently, North Atlantic does not utilize contractors to operate the refueling equipment. A senior reactor operator (SRO) supervises the refueling operations locally and reports to the SRO who is stationed "at the controls" in the control room. The inspector determined that the operations department manager applied the lessons learned described in 94-13.

The inspector reviewed the adequacy of ten refueling procedures used by the operations personnel. The procedures contained the necessary prerequisites, cautions, surveillance requirements, and communication requirements. The inspector identified no concerns with the procedures.

The inspector held discussions with the reliability and safety engineering manager and reviewed the outage risk management program guidance. The manager demonstrated the computer software used to define the different levels of safety and identify where contingency plans are appropriate. The inspector held several discussions with the outage manager. Most significantly, the inspector noted that in an effort to reduce outage risk, North Atlantic does a full core off-load before entering into reactor coolant system mid-loop operations. The inspector noted frequent discussions at the plant manager's morning meeting concerning outage risk as various jobs progress. The inspector also noted the presence of a representative from the engineering and safety engineering group in the shift outage manager room. A daily shutdown equipment status sheet specifies the required path, alternate path, and upcoming activities within the next 24 hours affecting the following: boration flow path, primary cooling, primary make-up, spent fuel pool cooling and make-up, and electrical sources. The inspector determined that the licensee has taken appropriate measures at institutionalizing the concept of outage risk management.

The inspector reviewed the work deleted from the scope of OR03. At the end of the last routine inspection period, plant management directed that the work scope of OR03 be reduced to ensure work quality and maintain the same outage duration. Plant management reduced the outage work by approximately 10%. On March 7, plant management removed a total of seventeen design changes and minor modifications from OR03 work planning. On March 16, plant management added five of the seventeen work tasks back into the outage scope, in order to improve the performance or control of safety-related components. The inspector judged that the addition of the five work activities back into the outage scope demonstrated an excellent safety perspective.

The inspector witnessed the movement of new fuel assemblies from the new fuel storage vault into the spent fuel pool. Procedural controls (RS0722) were verified, as was health physics coverage, material controls and operator double verification activities. Although not procedurally required, a licensed operator supervised activities of the spent fuel pool manipulator crane. The inspector also noted good practices related to the grid strap inspections of new fuel by Westinghouse technicians prior to placement in the spent fuel pool and the use of grid storage maps and fuel identification number verification and logging. The transfer of the new fuel was completed without incident.

The inspector concluded that the licensee has conducted excellent preparation for the upcoming refueling outage with an appropriate emphasis on safety.

2.0 MAINTENANCE (61726, 62703, 92701,37700)

2.1 Routine Maintenance and Field Observations

During this inspection period, the inspector witnessed maintenance activities in progress, completed field work and various component lineup and system configurations intended to support specific preventive and corrective maintenance functions. At times, the inspection was preplanned to observe certain key maintenance activities, while in other cases, random field work was observed during plant inspection-tours. In all cases, cognizant licensee personnel were interviewed to determine the adequacy of licensee work controls and of the criteria delineated to establish successful work completion. The following represent some of the maintenance/work control areas examined:

- **Service Water Piping Support Modifications**

The inspector observed contract maintenance workers modify service water (SW) diesel generator piping supports in accordance with design coordination report (DCR) 93-03. The contract workers closely monitored the applicable plant operational mode for which work on each support was authorized in the DCR. The contract workers and contract supervisor were also noted to be very experienced. The modifications required cutting, welding, bolting, and fabrication. The inspector confirmed close oversight of the maintenance workers by site services and the system engineer. The inspector therefore concluded that the modifications were being performed properly by contract personnel with the appropriate oversight being provided by cognizant licensee personnel.

- **Instrument and Control Corrective Maintenance**

The inspector observed instrument and control (I&C) technicians perform a two point calibration check of the refueling water storage tank and ambient room temperature detectors. The corrective maintenance resulted when a piece of measuring and test equipment (M&TE) failed a calibration check after being used to calibrate the subject temperature detectors. The inspector observed that the I&C technicians closely adhered to procedural instructions. The inspector concluded that the activity was properly performed.

2.2 Surveillance Activities

The inspector observed portions of the following safety-related surveillances to assess the adequacy of the procedural acceptance criteria, calibration of test instruments, qualification of personnel, interdepartmental communications, and the evidence of administrative approvals.

- Turbine Driven Emergency Feedwater Pump Test
- Cooling Tower Temperature Surveillance Test
- Main Steam Safety Valve Setpoint Test

The inspector witnessed the "Trevitesting" of some main steam safety valves (MSSV) intended to verify operability of the valves in accordance with inservice testing (IST) program and technical specification requirements. The Furmanite Company utilizes a "Trevitest" process, which consists of a hydraulic lift device to boost the existing main steam line pressure to the MSSV lift setpoint ratings, in order to verify as-found valve set-points without removal of the MSSVs from the steam lines. The licensee controls the MSSV in-place setpoint verification activities through the implementation of an engineering surveillance procedure, EX1804.041. This procedure was recently revised (Revision 1) effective March 29, requiring the plant to be below 50% rated thermal power during the conduct of testing. The inspector confirmed that Technical Clarification, TS-011, discussing operation at reduced power levels with inoperable MSSVs was considered in the revision to EX 1804.041, since the Trevitesting process renders a MSSV inoperable during testing.

A total of seven MSSVs, of the 20 total, were tested, but not more than one valve was tested in any one main steam line at a time. Two Trevitest rigs were available, with one in use in each of the two main steam pipe chases. The inspector observed testing, examined the MSSVs, and/or discussed the Trevitest data with cognizant personnel at both work locations. The inspector noted that the seven MSSVs chosen for testing were selected representative of all four main steam lines, and based upon either known seat leakage (i.e., deficiency tags written) or challenge to the valves during the "A" main steam isolation valve closure and reactor trip event in January (Reference: NRC inspection report 50-443/94-03.) The inspector reviewed the as-found and as-left MSSV setpoint test data with cognizant licensee personnel.

The inspector also discussed with licensee technical support personnel how the Seabrook Trevitesting conduct and accuracy might be affected by problems with the results of this Furmanite test process, which have been identified at other nuclear sites (e.g., Palo Verde, Braidwood). These problems, as documented in Event Reports and Furmanite correspondence, relate to the Trevitest setpoint equation and calculation of the mean seat area for each MSSV tested. At the other sites, where Dresser 3707R series valves are used, the Furmanite approximation for the valve main seat sealing area was sufficiently different than the valve's actual main seat area so as to create setpoint offset errors. At Seabrook, however, Crosby MSSVs are used. Using Crosby data, the licensee was able to calculate what penalty would be taken for worst case situations with the Furmanite model and method applied to the Seabrook MSSVs. In all cases, it was determined that the worst case error was bounded by the Technical Specification and ASME Code tolerances. Thus, the generic issue of comparative problems found with Dresser MSSVs at other sites was determined to raise no component operability or safety impact questions relative to the use of the Trevitesting process at Seabrook.

The inspector held discussions with licensee personnel regarding the analysis of Trevitesting accuracy for the MSSVs and interviewed a Furmanite Company project engineer regarding the problems identified at other sites. Licensee evaluation of this issue was both complete and comprehensive. Similarly, the conduct of Trevitesting for the MSSV setpoint verification was well controlled. No unresolved safety concerns were identified relative to the observation and follow-up of this surveillance testing.

2.3 Containment Personnel Hatch Event

On April 10, with the plant in cold shutdown and containment not required, an event involving personnel injury occurred while maintenance personnel were performing a safety-related activity for the simultaneous opening of both containment personnel hatch doors. Due to a 0.5 psid between containment and the environment outside containment, an air flow of sufficient velocity to cause flying debris and knock people down was encountered. The licensee's preliminary investigation determined that the inner door was full open vice cracked open, as specified by Procedure MS0535.07, Personnel Hatch Airlock Mechanical Interlock Disconnect. Three personnel were pushed out of the airlock onto a platform. Eleven plant workers received minor medical attention and two were sent to the local hospital for further treatment and later released.

The plant manager ordered that a formal event evaluation team be formed to determine the root cause of this incident and implement corrective actions, as necessary. The inspector discussed this event with an OSHA official. OSHA responded by sending two inspectors to review the industrial safety aspects of the event. The inspector reviewed MS0535.07 and inspected the airlock outer door. A Preliminary Notification (PNI-9424) was issued by the NRC on April 11 to describe the known details of this occurrence. At the end of this inspection period, the licensee's event evaluation team investigation was still in progress. The inspector will assess this event in the next routine inspection period.

2.4 Scaffold Controls For Safety-Related Equipment

During a routine tour of the primary auxiliary building, the inspector observed a scaffold erected around the "B" primary component cooling water (PCCW) heat exchanger, 1-CC-E-17B. After inspecting the scaffolding, the inspector identified that the scaffolding did not appear to meet the intent of the specific MA 4.10, Installation and Removal of Temporary Equipment, scaffolding evaluation. The evaluation stated, in part, that," the failure of the staging could be of some consequence to safety-related equipment..., All staging associated with the work shall be removed immediately upon completion of the work." The comments section of the scaffold tag specified that the plant be in mode 5 before scaffold erection begins. The inspector noted that the plant was in operational mode 1.

The inspector reviewed the licensee's program for the evaluation and use of scaffolding erected near safety-related equipment. The controls are specified in MA 4.8, Control Of Temporary Equipment, and in MA 4.10. MA 4.8 evaluates the scaffold while MA 4.10

contains the general guidelines for the erection and use of scaffolds. MA 4.8 allows the maintenance technicians who build the scaffold around safety-related equipment to perform a self-evaluation. If the maintenance technicians judge that the scaffold cannot impact safety-related equipment during a seismic event, a temporary modification is not required. The inspector noted that the scaffold that engulfed 1-CC-E-17B contacted small bore piping attached to the heat exchanger. Further, there were other safety-related components that had the potential to be adversely affected by the collapse of the scaffold during a seismic event. The inspector discussed the noted concerns with the mechanical maintenance department manager, a maintenance support engineer, and the engineering department manager.

The licensee promptly removed the scaffold erected around 1-CC-E-17B. The plant manager initiated a walkdown to review the adequacy of all other scaffolds erected around safety-related equipment. The licensee identified one scaffold erected in the mechanical service water cooling tower that needed to be removed. Two others in the pipe chase were either reinforced or removed. The plant manager placed a hold on the erection of any other scaffolds until the controls contained in MA 4.8 were enhanced.

The inspector reviewed engineering evaluation 87-03 entitled "Guidelines For Erecting Temporary Structures And Performing Rigging Operations Adjacent To Safety-Related Equipment." The existing MA 4.8 controls were not fully consistent with the provisions of engineering evaluation 87-03. The maintenance staff indicated that a substantive change to MA 4.8 was made in December, 1993 to address concerns identified by the nuclear quality group in March, 1993. In December, 1993 and February, 1994 the licensee identified safety-related scaffold concerns and initiated operational information reports 93-127 and 94-15. The licensee realized that the change made to MA 4.8 in December, 1993 required improvement. The inspector considered it positive that the licensee self-identified these scaffold control weaknesses.

On April 6, the maintenance support department issued change 2 to revision 4 of MA 4.8. The change required a more thorough individual scaffold evaluation performed by engineering personnel, when necessary. The evaluation process was consistent with engineering evaluation 87-03. The inspector attended the training sessions held with maintenance technicians and with technical support engineers. The inspector judged that the training effectively described the changes made to MA 4.8.

In summary, although a weakness in the control of scaffolds was identified, the inspector confirmed that the licensee was aware of the program shortcomings and had initiated corrective action documents to enhance the process. The inspector concluded that the enhancements made to MA 4.8 and the training given during this inspection period addressed the inspector's concern.

3.0 ENGINEERING (71707,37828,92700)

3.1 Site-Specific Tornado Analysis

An attachment to NRC inspection report 50-443/93-05 documents the history of the development of a site-specific design basis tornado (DBT), intended to be used in the design analysis of a limited number of safety-related systems, structures and components at Seabrook. The characteristics of the site-specific tornado were delineated in a revision the Seabrook UFSAR (reference: section 2.3.1.2b.2). The original question of component adequacy centered on six tornado doors that were identified in 1990 to not meet the differential pressure criteria for the original DBT described in the UFSAR. Since that time a seventh door and some diesel generator building components had also been identified to require site-specific DBT review to verify design adequacy.

On March 14 and 15, NRC personnel from the Office of Nuclear Reactor Regulation visited Seabrook Station to inspect the tornado doors and other equipment in question and to review the design calculations justifying current component adequacy. The inspector attended a meeting between licensee and NRR personnel, at which time the origin and justification for a site-specific DBT were discussed. During this NRR staff visit, the licensee indicated that as a result of structural modifications, in conjunction with further design analysis, all of the components in question meet the original DBT criteria. Therefore, the licensee committed to revise the updated UFSAR to remove any reference to site-specific tornado characteristics; thus, rendering the original DBT characteristics to be the only governing tornado criteria for Seabrook Station.

The Office of NRR plans to issue a Safety Evaluation of the tornado design basis upon receipt of the licensee's letter that confirms the position that all equipment satisfies the original DBT loading criteria and provides analyses demonstrating that no new unresolved safety concerns have been raised. The licensee correspondence on this subject (NYN-94044) was submitted after the conclusion of this inspection period. The inspector verified that prior licensee actions, to include both design review and component rework activities, in this area of DBT review did not constitute an unreviewed safety question and determined that the Office of NRR appears satisfied with current licensee commitments. No further inspection follow-up of this issue is planned, as the remaining licensee actions involve updated FSAR changes and analyses that are subject to NRR review. However, the licensee submitted on April 15 an LER (No. 94-006-00) addressing the unanalyzed condition of the diesel generator building components which have since been confirmed to meet original DBT criteria. The inspection review of LER 94-006-00 will be addressed in a future inspection report.

3.2 120MB Incandescent Lamp Failure

The inspector reviewed the evaluation of a potential 10 CFR 21 concerning the failure of a 120MB lamp that occurred at Wolf Creek, which resulted in the loss of control power to a motor control center. The evaluation indicated that Seabrook Station does not have any of the subject lamps in safety-related applications. The bulbs are used in status monitor displays. The licensee previously installed voltage dropping transformers or voltage dropping resistors to reduce the operating voltage in such applications at Seabrook. The licensee performed qualification testing to establish seismic qualification of the lamps. The inspector determined that the licensee had thoroughly reviewed the industry experience related to this issue.

3.3 Polar Crane Modification

Based upon correspondence, dated March 2, from the Whiting Corporation, the licensee was informed of safety concerns related to the potential over-stressing of bolted connection points of the trolley main hoist components of the two polar cranes supplied to Seabrook Station. This notification was considered a potential 10 CFR 21 report for further licensee evaluation. The licensee removed both cranes from immediate service and commenced planning for the repair/modification of the Unit 1 polar gantry crane. The other polar crane is located in Seabrook Unit 2, for which the construction permit has expired and no NRC licensing docket is currently active. For personnel safety considerations, the Unit 2 polar gantry crane has been danger tagged to preclude use until this problem is resolved.

The inspector reviewed the Whiting Corporation letter and evaluated the identified issue in relation to similar Whiting crane bolting problems addressed in the 1990 time frame. (reference: open item 90-88-03, closed in NRC inspection report 50-443/91-22). The inspector also examined internal licensee correspondence related to the Unit 1 polar crane rework and the 10 CFR 21 evaluation process. Since the polar crane is required for heavy lifts inside containment during OR03, the inspector met with cognizant licensee maintenance and scheduling personnel to discuss the plans for the Unit 1 crane inspection, rework and release for use. Minor modification, MMOD 94-520 was issued to specify the scope of bolting changes required for repair.

Work commenced on the polar crane on April 13, as controlled by work request 94W001169. The inspector reviewed the work request, verifying appropriate quality control involvement, proper torque considerations and torque wrench calibrations. The inspector noted that all new and replacement bolts were designated to be high-strength (i.e., ASTM A-325) bolting material with the appropriate matching nut and washer material. The inspector also reviewed MMOD 94-520, checking for design criteria consistent with the repair of the problems identified by the Whiting Corporation. The inspector noted that the scope of work involved not only bolt replacement to correct the original identified problem, but also additional inspection to verify other bolt material was correct and not in need of replacement. The polar gantry crane rework was completed on April 15 and the crane returned to service to support OR03 lift activities within the Unit 1 containment.

The inspector determined that this repair activity was well controlled and proceeded in a deliberate manner, with evidence that work scope changes were properly processed to resolve unexpected field conditions. Additionally, since completion of this rework was required to support critical path activities during the refuel outage, adequate manpower and time were allotted to satisfactorily accomplish the job, even with the added work and scope changes. The licensee continues to evaluate this issue for reportability in accordance with 10 CFR 21. The inspector has no questions or unresolved safety concerns regarding the engineering and maintenance field work that has been completed.

4.0 PLANT SUPPORT (71707,71709,82205,82301)

4.1 Radiological Controls

The inspector observed the reaction and response of the health physics (HP) staff to a crud burst resulting from an operations activity on the chemical volume and control system letdown line radiation monitor booster pump. The inspector determined that the HP staff performed comprehensive radiological surveys to bound the increasing dose rates. As a precautionary measure, the HP staff extended the high radiation area boundary encompassing the volume control tank room. No release to the environment occurred from this event. The licensee generated a corrective action document to investigate the root cause of the crud burst. The inspector noted that HP personnel led the investigation. The inspector determined that the HP staff reacted promptly and implemented both ALARA principles and 10 CFR 20 requirements.

The inspector observed flushes performed subsequent to the resin sluicing of a boron thermal regeneration system demineralizer. Excellent teamwork between HP, operations, and radwaste personnel was evident. The resin sluicing discharge piping was mechanically agitated with a rubber mallet to increase the effectiveness of the flushing. The inspector noted excellent HP supervisor oversight of the flushing activity. The HP technicians utilized a special long handle probe to measure the dose rate. The HP technicians directed plant workers to a low dose waiting area. The inspector determined that the HP personnel maintained strict control of the resin sluicing activity.

The inspector reviewed the self-assessment process utilized by the chemistry and health physics group. Procedure JD0999.902 dated March 3, 1994 established a formal self assessment process. The inspector reviewed the procedure and applicable observation sheets, and observed HP and chemistry managers and supervisors in the field. Opportunities for improvement that are identified are tracked and corrective actions implemented. The group also started an initiative to formally verify the effectiveness of previous corrective actions. The inspector observed HP personnel maintaining close oversight of plant radiation workers. The inspector determined that the implementation of a structured self-assessment process in the chemistry/HP group is a significant strength.

The inspector concluded that the HP and chemistry group performance in the areas inspected was appropriately directed and achieved excellent results. The HP staff reacted well to a significant challenge and were intimately involved in the investigation of the root cause.

4.2 Security

The inspector toured the protected area, observed security guards on patrol, and monitored activities at guard island. The security force properly dispositioned fitness-for-duty test failures. Guards posted at the control room door as a compensatory watch during door modification remained alert and understood their responsibilities. The inspector assessed that the security force conducted routine activities in a meticulous manner. No unresolved security concerns were identified.

4.3 Emergency Preparedness

On March 16, the licensee conducted an unannounced, back shift mobilization drill of the Emergency Response Organization (ERO), testing the capability of the ERO notification system (ERONS) and the response time of ERO personnel reporting to their assigned emergency facilities. In accordance with predetermined drill objectives, licensee responders were expected to respond within 60 minutes, as directed, to either the emergency operational facility (EOF), the technical support center (TSC) or the operations support center (OSC). Other Seabrook facilities participating in this drill included the main control room, where gaitronics announcements and pager actuations were implemented, and the protected area guard island where the ERONS was initiated.

One objective of this ERONS drill was to demonstrate the ability to mobilize the station ERO to support the activation of the designated station emergency response facilities in a timely manner and to ensure the availability of adequate manpower to support 24-hour coverage. Another objective involved the mobilization of key ERO personnel during back shift hours. The conduct of a successful drill is required during 1994 to satisfy a commitment to demonstrate the above objectives every six years, as delineated in the planning standards of NUREG-0654.

The resident inspectors observed the initiation and conduct of the unannounced, back shift ERONS drill. Discussion with cognizant licensee EP personnel confirmed limited dissemination of the date and drill start time, on a need-to-know basis. An inspector witnessed the gaitronics drill announcement and pager actuation from the control room and noted that operations personnel also authorized the initiation of the ERONS equipment at guard island. As the drill progressed, the inspectors checked ERO personnel response at the EOF, TSC and OSC. Some problems with ERONS telephone call-out of certain maintenance staff personnel were identified, as well as the lack of timely response of some personnel required for the full complement of all ERO positions at each emergency facility. These problems were recognized by the licensee, as it was determined that the drill objectives were only partially satisfied.

The inspector reviewed licensee reports evaluating the overall system and player drill performance and an Operational Information Report (OIR 94-058) documenting that the mobilization drill was not fully successful. The inspector determined that the licensee self-assessment of this drill appropriately identified the problem areas, as well as the need for additional review and corrective measures. The licensee plans to conduct another unannounced, back shift mobilization drill prior to the end of CY94, in order to demonstrate the required objectives in accordance with EP program commitments.

5.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500,35702)

5.1 Occurrence Review Committee

The inspector reviewed the licensee's new process for identifying and evaluating performance trends. NRC inspection report 50-443/93-13 documented a negative performance trend involving multiple examples where plant equipment important to safety were unnecessarily challenged. At that time, several different trending programs had not been effective in identifying the negative trend. During the current inspection, the inspector attended meetings, held discussions, observed work performance, and reviewed various reports to evaluate the effectiveness of the licensee's revised process controls in this area.

The quality assurance (QA) department developed the occurrence review committee (ORC) to identify and evaluate performance trends in a timely manner. The ORC is chaired by a QA trend engineer. The ORC membership consists of an individual representing the following functional areas: human performance, operation support, maintenance, site services, reliability and safety engineering, technical projects, training, and the independent safety engineering group. The ORC meets weekly to review all newly generated corrective action documents including station information reports, operational information reports, and condition deficiency reports.

Using a significance factor worksheet, the ORC assigns a significance level to each occurrence. The worksheet has examples of the degree of challenge to reactor plant safety, industrial safety, repeat occurrences, and economic impact. From lowest to highest, the significance levels are none (no perceived consequence), low (minor), moderate (important), and high (significant). For those occurrences directly caused by personnel error, the occurrence is attributed to one of the following categories: omission, extraneous act, untimely act, transposition, out of sequence, and quantitative deficiency. The ORC leader notifies the plant manager of any concerns identified during each meeting. The ORC issues a monthly and quarterly report, which provide trends, evaluation, and recommendations. Since the ORC reviews the occurrence before completion of the root cause evaluation, the process is later validated by comparing the root cause to the cause assigned by the ORC. The inspector determined that the process provides valuable performance trending information in a timely manner.

The inspector reviewed the January and February monthly and the quarterly trend reports. The quarterly report indicated that the number of personnel errors being reported has significantly increased because of the lower reporting threshold. The report identified the "omission" category as the dominant factor. The performance of each department was trended. Several occurrences that involved personnel safety issues were identified as an area in need of further management attention. The plant manager met with the department managers to discuss the various performance trends.

The inspector concluded that the ORC trending and evaluation data provided plant management with better performance information than was in evidence during the time period of NRC inspection report 50-443/93-13. The concept of assigning a significance factor with a multi-disciplinary review is viewed as a strength in the process.

5.2 Quality Programs Interface Meeting

The inspector attended a quality programs interface meeting where quality assurance (QA) personnel briefed line management with performance assessments for a four month period. The inspector reviewed the data, assessments, and held a discussion with the nuclear quality manager. The interface meeting replaced the finding review board meeting. The assessments were presented in the seven functional categories of the old NRC systematic assessment of licensee performance process. Line management appeared to be receptive to the assessments.

Quality control inspectors and auditors identified numerous issues during the four month time period. The various department personnel self-identified numerous issues. The inspector considered this to be a strength that station personnel possessed a questioning attitude. However, the inspector judged that the packaging of the individual issues into broader performance assessments appears to need improvement. Also, since the meeting discussed a four month time period, the inspector considered it a program shortcoming that the report or a report summary was not available before or at the meeting. The report was not available at the end of this inspection period.

The inspector discussed the above observations with the nuclear quality manager. The manager indicated that the plant manager has requested that the meeting be held monthly. The quality manager also indicated that this interface meeting was only the third one and that the meeting format and presentation style were still evolving. The inspector will continue to monitor these activities during future inspections.

6.0 MEETINGS (30702)

Two resident inspectors were assigned to Seabrook Station throughout the period. The inspectors conducted back shift inspections on March 14, 16, 28 and April 6 and 7 and deep back shift inspections on March 9, 10 and 14 and April 4, 5, 6 and 10.

Throughout the inspection, the inspectors held periodic meetings with station management to discuss inspection findings. At the conclusion of the inspection, the inspector held an exit meeting with the Executive Director of Nuclear Production and his staff to discuss the inspection findings and observations. No proprietary information was covered within the scope of the inspection. No written material regarding the inspection findings was given to the licensee during the inspection period.

Two region based inspectors conducted an independent measurements inspection from February 28 to March 4, using the NRC Region I Mobile Laboratory to verify the licensee's capability to analyze radioactive effluents. The results of this inspection are documented in NRC inspection report 50-443/94-04.

Another region based inspector conducted an inspection of the Seabrook Station effluent monitoring program during the period, March 28 - April 1. The results of this inspection are documented in NRC inspection report 50-443/94-06. Additionally, three NRC personnel from the Office of NRR visited Seabrook Station on March 14-15 to discuss the licensee's site-specific tornado analysis and tour plant areas where the design calculations include consideration of tornado loadings other than is documented in the original criteria of the updated FSAR. The NRR review is discussed further in section 3.1 of this inspection report. Also, on March 21-22, a senior operations engineer from the NRC Office of Analysis and Evaluation of Operation Data (AEOD) visited Seabrook to review the licensee's operations experience feedback program. No concerns were identified during this review.

From March 28 - April 4, an NRC operations officer from the Office of AEOD visited Seabrook Station to observe control room activities and overall station operations. Along with the accompaniment of nuclear system operators on plant rounds and other site tours, the operations officer witnessed the conduct of system testing (e.g., emergency feedwater system), observed routine plant evolutions and reviewed technical and training manuals relative to the design features of various plant components. Some licensee event reports (LER) were also examined with respect to the applicable plant operating procedures. No issues of safety concern were identified during this AEOD visit.