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
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Inspectors:	Antone C. Cerne, Senior Resident Inspector Richard A. Laura, Resident Inspector William T. Olsen, Resident Inspector, Maine Yankee Joseph Sebrosky, AEOD Operations Officer (visiting)

Approved By: John Rogge 6/6/94
John Rogge, Chief — Date
Reactor Projects Section 4B, DRP

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, and plant support.

Results: North Atlantic performed refueling activities in a safety conscious manner. One violation, that involved inadequate maintenance training, was identified. See the executive summary for an assessment of licensee performance.

EXECUTIVE SUMMARY

SEABROOK STATION
NRC INSPECTION REPORT NO. 50-443/94-08

Plant Operations: The operators performed refueling activities in a safety conscious manner by off-loading fuel from the reactor core and draining the reactor coolant system to mid-loop conditions without incident. Several examples of the incorporation of PRA insights were observed that provided additional equipment availability beyond regulatory requirements. An inconsistency between two operations refueling procedures that involved the availability of a reactor cavity water level transmitter was identified. Three minor boric acid leaks from mechanical joints located inside containment were found to have been properly entered into the work control system for corrective actions.

Maintenance: The maintenance staff generally performed work in a safe and controlled manner. Two unrelated deficiencies were identified involving a work request scope change and also a locked-up pipe support strut that had been inspected during this outage by technical support. The deficiencies were isolated cases; but nonetheless, indicated lapses in attention-to-detail. One violation was identified concerning inadequate maintenance worker training and job task analysis for operation of the containment personnel air-lock. Tube degradation in the PCCW heat exchangers was properly evaluated and corrective actions initiated.

Complex surveillance tests were generally performed in a controlled manner by properly resolving test anomalies. The improper installation of an electrical test jumper resulted in a test interruption. Although the jumper was re-installed correctly and the surveillance test completed, neither technical support nor maintenance personnel initiated a corrective action document to identify the cause of the misplaced jumper and to develop further corrective measures.

Engineering: The engineering staff properly resolved the safety injection pump run-out flow issue in the long term by conducting a special test and setting manual SI throttle valves to reduce flow to less than the pump manufacturer's run-out flow rate limit. The interim plant response did not include a reportability determination and could have been more timely. The quality of a 1992 operability determination was adequate, but did not fully address the nuclear steam system supplier recommendations. Additionally, two temporary modifications were reviewed, which were properly implemented.

Plant Support: Adequate support of operational and refuel outage activities was identified to exist in the radiological controls and security areas. The health physics staff and the security guard force responded well to the additional workload challenges resulting from increased personnel processing and expanded radiological work activities. The Emergency Preparedness staff initiated actions to correct problems with a recent Emergency Response Organization Notification System drill.

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DETAILS

1.0 PLANT OPERATIONS (60710, 71707, 93702)

1.1 Plant Activities

At the beginning of this inspection period, the operators maintained the plant in operational mode 5 (cold shutdown) in preparation for the third refueling outage (OR03). On April 23, the plant entered operational mode 6 (refueling) when the first reactor pressure vessel head stud was loosened. On May 3, the operators finished off-loading the fuel assemblies from the reactor core. On May 5, the operators drained the reactor coolant system (RCS) to mid-loop level (-85 inches, with zero referenced to the reactor pressure vessel flange level) to facilitate access to the primary side of the steam generators for eddy current testing. For the remainder of this inspection period, the plant remained defueled in mode 6 with the RCS drained to mid-loop.

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 containment, the penetration areas, the main steam chases, the primary auxiliary building, the emergency diesel generator rooms, the turbine building, and the service water pump house. During the tours and attendance at the various meetings, the inspector noted an adequate implementation of operational controls over plant activities and an overall good performance, including cognizance of the current plant configuration, by the operations staff.

During the implementation of major plant evolutions (e.g., RCS draindowns, cavity fills, system isolations) and other refuel outage activities, the operators on shift and the work control group maintained both cognizance and responsibility for the overall coordination of safe job performance. A deliberate approach was observed with respect to shift turnovers, evolution pre-briefs, and adherence to step-by-step procedural requirements while performing routine operations and tests. Of particular note was the continued excellent use of a probabilistic risk assessment (PRA) methodology in planning for outage contingencies and providing a defense in depth for safety-related functions and tasks. Examples of this conservative philosophy, observed by the inspector, include:

- ensuring that multiple RCS active injection pathways were available and in service prior to draining the reactor water level to remove the vessel head for core off-load.

- providing a backup 50Kw diesel generator to supply temporary power to a spent fuel pool coolant pump, as a contingency to the loss of all site a-c power during the refuel outage.

- maintaining the protected train concept and at least a single train of emergency diesel generator power capability at all times, even beyond the technical specification requirements.

The inspector examined various equipment availability, configuration controls, and abnormal and special operating procedures related to the above examples and other instances where the licensee maintained a conservative operating philosophy in system line-ups and emergency equipment availability. Overall, the PRA usage and defense-in-depth approach to work planning continues to be a strength in the licensee's control of safety significant OR03 activities.

During the main plant computer outage, the inspector observed that the licensee utilized a temporary computer to provide trend and alarm functions. Operations management promulgated guidance on how to use the temporary computer. The inspector observed operators use the temporary computer to closely track plant parameters. Excellent teamwork between operations support and technical support engineers was evident.

During tours of the containment, the inspector observed boric acid leaks from valves RH-V-111 & SI-V-117, and from a mechanical joint on a blank flange downstream of valve SI-V-2. The inspector observed that no carbon steel bolts were used on these stainless steel components. The inspector performed a review to determine whether or not these leaks were entered into the work control system for corrective maintenance. Work requests existed for RH-V-111 and SI-V-117. A work request for the SI-V-2 flange leak was not initiated; however, the blank was subsequently removed to support an operational activity. The inspector had no further questions or concerns relative to these leaks.

1.3 Fuel Movement

The inspector observed core off-load activities from the control room and within the containment building, including the witness of fuel movement during periods of deep back shift inspection. The inspector attended the briefing of station personnel, held prior to the commencement of core off-loading, and verified proper coverage of such areas as the chain of command, communications and control, and procedural requirements. A Westinghouse specification (F-5) providing instructions, precautions and limitations on the handling of fuel assemblies was reviewed, as was the Refueling Operation procedure, OS1000.09. The inspector confirmed the existence of adequate coordination controls between the plant, reactor engineering section, the operations staff and health physics personnel and noted precise status and record-keeping relative to the movement of each fuel assembly.

One issue identified by the inspector, as a result of observing fuel movement preparations, was an inconsistency in certain station operating procedures regarding the status of a specific reactor vessel level instrument (LI-9405) connected off level transmitter RC-LT-9405. The inspector noted that an abnormal operating procedure (OS1215.05), addressing a loss of refueling cavity water, used LI-9405 to check reactor vessel level to confirm the integrity of the RCS. This was highlighted during the core off-load briefing. However, the inspector determined that this level indicator had been valved out of service with RC-LT-9405 during the fill of the reactor refueling cavity in accordance with other procedural (OS1015.02) requirements. The inspector notified the operations personnel of this discrepancy and corrective actions were initiated to restore the level indication capability. The level instrument had been removed from service prior to cavity fill because under such conditions, it is over-ranged. The licensee, however, checked with the component vendor to determine whether over-ranging would adversely affect the instrument. Upon learning that the level indication could remain in service with the cavity filled, corrective measures were taken to revise OS1015.02. Thus, the abnormal operating procedure did not have to be revised. The inspector concluded that the licensee took immediate and comprehensive actions to resolve this procedure discrepancy.

Core off-load commenced at 1:30 a.m. on April 30 and was completed at 0:35 a.m. on May 3. All 193 fuel assemblies were moved from the reactor pressure vessel to the spent fuel pool without incident, although core alterations were suspended for a period of time to work on the fuel handling machine. During the period of core offload, the inspector spot-checked containment conditions for compliance with technical specification requirements for containment integrity during core alterations. No unresolved safety concerns were identified. The inspector noted that the core off-load was well planned, deliberately executed and safely conducted.

2.0 MAINTENANCE (61726, 62703, 92701)

2.1 Routine Maintenance and Field Observations

During this inspection period, the inspector witnessed maintenance activities in progress, completed field work and various component line-up 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:

- Reassembly of a containment air purge (CAP) 36" butterfly valve, CAP-V-2, inside containment. This work, accomplished in accordance with work request 93WR0401, was required to provide a functioning CAP system in support of OR03.

- Replacement of Fisher I/P transmitters with Rosemount models inside containment, in accordance with minor modification, MMOD 91-601.

- Replacement of service water (SW) system piping for the emergency diesel generator jacket water heat exchangers, in accordance with design coordination report, DCR 93-003.

- Disassembly and eddy current examination of a primary component cooling water (PCCW) heat exchanger, in accordance with work request 94W0520.

- Preventive maintenance of the "B" train emergency diesel generator (EDG), in accordance with work request 94WR1155 and repetitive task sheet 94RM04711001.

- Replacement of General Electric (GE) right angle lockout relays (86 device) with equivalent Electroswitch relays, in accordance with work request 94W0514, DCR94-008 and station procedure LS94-1-3. The 86 device relays had exhibited malfunctions in the past during the degraded voltage testing conducted as part of required surveillance activities.

The inspector observed that the above work activities were generally well controlled. The work packages were reviewed in the field and proper authorization, quality control and documentation were verified. The inspector also spot-checked various maintenance procedures (i.e, MS series), referenced in the work packages, to determine if the activities were being performed in accordance with the approved guidelines and established criteria.

While no violations were identified, the inspector did note that the PCCW heat exchanger tubing eddy current examination was being repeated without a scope change to work request 94W0520. The reason for the repeated nondestructive examination (NDE) was to determine if the NDE results would be affected by a change in the eddy current coil to a saturation probe. The inspector had no concerns regarding the repetition of the NDE or the qualification of the new technique. However, since heat exchanger tubes that had already been plugged had to be reopened for this inspection, the inspector indicated to the cognizant licensee engineer that the issuance of a scope change to control such additional work would have been prudent. The additional NDE was completed and the tubes replugged, but this work was subsequently superseded by the decision to replace all tubes in both PCCW heat exchangers (see section 2.4 of this inspection report).

An additional field observation by the inspector involved the questionable physical condition of a pipe strut, used in combination with a snubber, to support the SW piping for the "B" train EDG heat exchanger. The inspector noted corrosion on the strut bushings had adversely impacted the strut's freedom of rotational movement. No problems were identified with respect to the snubber condition. Discussion with licensee technical support personnel revealed that the licensee's prior visual inspection of the subject support (1-4417-RM-11) had adequately examined the snubber, but neglected to inspect the strut assembly. Operational Information Report (OIR) no. 94-119 was issued to document the incomplete support inspection activity and work request 94W1878 was issued to repair the strut. The inspector confirmed that the licensee also took additional corrective actions to reinspect other supports examined during OR03 by the NDE inspector involved in the incomplete inspection. Also, other pipe supports in the general area were examined for similar unacceptable corrosion. In both cases, no other nonconforming pipe support conditions were identified.

The above inspection issues represent isolated cases of work process problems in the maintenance/technical support areas where routine NRC field observations have otherwise revealed an adequate system of work controls. Licensee response to the identified concerns has been comprehensive and appropriately directed. The inspector has no further questions in this area, but will continue to monitor field maintenance and work control adequacy on a routine inspection basis.

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, evidence of administrative approvals, and the overall acceptability of procedural implementation and test conduct:

- EDG Fuel Oil Train "B" Piping Pneumatic ISI 10 Year Test
- EDG 24 Hour Load Test and Hot Restart Surveillance
- 18 Month Emergency Diesel Generator Test
- ECCS Check Valve Full Flow Verification Test
- Safety Injection Pump Runout Test

During the safety injection pump runout test performed in accordance with procedure ES94-1-1, the inspector observed that the "A" pump run-out flow rate exceeded the maximum specified flow rate of 675 gpm. The licensee generated a station information report (SIR) to evaluate reportability, evaluate the cause, and to develop corrective actions. Subsequently, the licensee adjusted the manual throttle valves to reduce the pump run-out flow to less than 675 gpm. The test director and operators performed these tests in a well controlled manner. The significance and history of the pump run-out issue is further evaluated in section 3.1 of this report.

The inspector witnessed portions of the ECCS check valve testing performed in accordance with Procedure EX1804.039. A test interruption occurred when valve CS-V196 (mini-flow isolation valve) failed to automatically close. Licensee investigation determined that an electrical jumper had been improperly installed. Two qualified instrument and control (I&C) technicians inadvertently installed the electrical jumper on the de-energized side of a slide link. After proper reconnection of the jumper, the operators recommenced the surveillance test and valve CS-V196 automatically closed, as expected. The inspector determined that the test director had taken the appropriate immediate corrective actions to complete the test.

Subsequently, the inspector performed a review to determine the root cause of the misplaced electrical jumper and corrective actions taken to preclude repetition. The inspector identified that technical support, maintenance, and operations personnel did not generate a STAR sheet (NOTE: STAR represents the "Stop-Think-Act-Review" program acronym) or an operational information report (OIR). The inspector discussed this with the technical support manager and the I&C department supervisor. The inspector determined that upper levels of plant management were unaware of the cause of the test interruption. The unit journal logged the test interruption, but did not document the cause. In response to the inspector's concern, the licensee generated a STAR sheet and an OIR to evaluate the misplaced electrical jumper. The inspector concluded that the lack of the initiation of a corrective action document represented a missed opportunity for the licensee to evaluate and develop lessons learned. The inspector noted, however, that this example appears to be an isolated case, based upon the NRC review of STAR Program activities and the follow-up of licensee identified problems.

The inspector also observed the hydrostatic testing of the EDG "B" fuel oil system storage tank and piping. Station personnel performed the required testing as directed by station procedure EX 1811.325. The testing is required by ASME Section XI and the station technical specifications. The inspector determined, through review of the procedure and interviews, that all personnel were familiar with the procedural provisions. All required test equipment was available and calibrated in accordance with station requirements. The test supervisor demonstrated positive control of the testing and ensured that all prerequisites and initial conditions were satisfied prior to actual test performance. The test supervisor made a decision to nitrogen purge all the fuel lines prior to actual testing. This was done to prevent any situations where fuel oil could be ignited during pressurization. After verification of the valve line-up and establishment of communications with test personnel, the test was commenced. The test pressure of 66-69 psig was reached and stabilized for a ten minute hold time, and the test was completed without any problems identified. Station quality control personnel inspected all the valves, piping and tank for leakage and found no deficient component conditions.

Additionally, the inspector noted that on April 15, a technical support engineer had observed service water pump SW-P-41D start at an earlier step in the emergency power sequencer (EPS) operation than expected during the conduct of technical specification required surveillance testing in accordance with procedure EX 1804.015. This concern was documented in station OIR 94-82.

After investigation by engineering department personnel, the licensee determined that an Engineering Change Authorization (ECA 03/104209A), issued in June of 1984, directed removal of wiring associated with the service water pumps at step 5 of the EPS and relocated them to step 8. The ECA allowed the service water pump discharge valve more time to fully close prior to a SW pump start signal. The relocation of the wiring was performed satisfactorily, however, the wire at step 5 for the SW pumps was not removed causing the anomalous EPS operation.

Station engineering personnel directed that the improperly spared wiring for the EPS "B" sequencer be removed as required by the original ECA in accordance with station work request 94W1295 and functionally tested as directed by the attached retest.

The inspector questioned the acceptability of this method of retesting without Station Operations Review Committee (SORC) review and approval prior to testing. In a meeting with the engineering manager and cognizant engineering personnel it was determined that the completed test data would be submitted to the SORC for final review and approval prior to closure of this item. The inspector also verified that the licensee conducted a review of similar ECAs performed during the time period when this error occurred and determined that no other similar problems existed. The inspector determined that the licensee aggressively resolved these issues taking prompt correct reaction. The inspector had no further questions regarding this issue.

Overall, the inspector concluded that complex testing activities were performed in a controlled and safe manner. The inspector identified no unresolved safety issues or concerns and verified that the licensee demonstrated an appropriate approach to the resolution of all test anomalies.

2.3 Containment Personnel Hatch Event (VIO 50-443/94-08-01)

Background and NRC Inspection Methodology

An event that occurred on April 10, documented in the last routine inspection report, where workers were blown out of the containment personnel air-lock. Plant workers were in the process of opening the inner and outer containment personnel air-lock doors, which is a safety-related activity. At the end of the previous inspection period, the licensee formed an event evaluation team to identify the cause and corrective actions. The licensee made an NRC four hour notification per 10 CFR 50.72 (b), newsworthy event. OSHA performed an investigation to review the industrial safety aspects of the event. During this inspection period, OSHA conducted a closing conference with licensee management. OSHA will issue their findings at a later date.

The inspector assessed this event by reviewing the procedure, interviewing the mechanic in charge of the activity, inspecting the damage that occurred to the outer air-lock door, attending the OSHA closing conference, discussing the event with licensee management, and reviewing the event evaluation team findings. The inspector also reviewed the mechanic's training qualifications. Technical specifications and the updated safety analysis report were reviewed to review the equipment design and to determine the level of safety significance.

Chronology and Event Evaluation Team Findings

The inspector reviewed the event evaluation team findings documented in station information report (SIR) 94-27. The lead mechanical supervisor assigned a senior mechanic to defeat the interlocks and open both air-lock doors in accordance with procedure MS0535.07. The lead supervisor held a job briefing with the mechanic that focused on the door interlocks. The mechanic had difficulty executing the procedural steps due to unfamiliarity with the work task. At this point, an I&C technician and the containment coordinator provided assistance in opening the air-lock doors. The three workers inadvertently deviated from step 8.2.2 that specified to open the inner door one to two feet to insert a wooden wedge. The inner door was fully opened.

Immediately after the I&C technician depressed the outer door open button, the outer door rapidly flung open. A differential pressure of 0.5 psid between the containment and containment enclosure space existed, which exerted a significant force on the outer door. In essence, the procedure methodology uses the air-lock to depressurize the containment atmospheric pressure. The three workers in the air-lock were blown out onto the platform. Material and debris were blown due to the wind gust, which the team calculated to be 127 miles/hour. The outer door hinge locking bar was damaged. Several minor personnel injuries occurred.

The event evaluation team attributed the root cause to failure to follow procedural instructions. Other contributing factors included: the lead supervisor should have assigned two workers to perform the task, the job briefing did not cover the potential consequence of incorrect actions (the lead supervisor knew of a previous event in 1991 where a worker got injured), and various procedural inadequacies. Human performance enhancement report 94-SIR-27 identified that certain managerial inadequacies contributed to this event. Too few workers were assigned, no workers were experienced with the task, and insufficient supervisory resources were available for the work scheduled for that day, no training or qualification existed for the task, and the lessons learned from the 1991 occurrence were not fully applied.

North Atlantic developed and implemented short and long term corrective actions. On April 12, the plant manager ordered that all physical plant work be stopped. Senior management held briefings with all station personnel to emphasize the importance of following procedures and doing the job right the first time. The inspector attended one of the meetings. The plant manager clarified the plant motto, "Do it right and on-time, every time." Several plant workers commented that the cost competitive initiatives seem to conflict with doing the job right philosophy. The plant manager explained that doing the job right was more important than doing it on-time. After the briefings were held, plant physical work resumed.

Some of the corrective actions include:

- Procedure MS0535 was supplanted by operation procedures OS1058.03 and OS1058.04. The new procedures utilize a safer methodology to open the air-lock doors and better warning statements. A temporary modification installed at the air-lock provides local atmospheric pressure indication. The differential pressure limit of 0.5 psid has been lowered to 0.098 psid. The operators are responsible to lower containment pressure to minimize the differential pressure.

- Management held the workers involved with this event accountable with disciplinary actions.

- Management training was improved to better match worker qualification and experience to the task being performed.

- Maintenance repaired the door hinge locking bar on the outer door.

NRC Determination of the Air-lock Event Safety Consequence and Significance

The inspector performed a review to determine the actual safety consequence and safety significance of this event. At the time of the event, the plant remained in operational mode 5 (cold shutdown). Technical specifications (TS) 3.6.1.1 specifies that primary containment integrity is not required to be OPERABLE in mode 5. Based on this, the inspector determined that no actual reactor safety consequence resulted from this event. The inspector concluded that defeating the interlocks and opening both air-lock doors was allowed by TS.

The inspector performed a review of this event to determine the level of safety significance. Defeating the air-lock door interlocks and opening both doors are a safety-related activity. The door hinge locking bar on the outer door was damaged. Several plant workers were injured as a result of this event with the potential for more serious personnel injuries. The inspector determined that the event did involve a degree of safety significance based on the equipment damage and personnel injuries. The inspector concluded that the defense-in-depth approach to this activity failed.

NRC Inspection and Findings

Based on interviews, procedure reviews, and review of the evaluation report, the inspector determined that the event evaluation team correctly identified the root cause as failure to follow procedure. However, the inspector identified one significant contributing cause that needed further evaluation and further corrective actions. No formal or informal training had been given to workers for the hatch interlock removal and door opening task. Other than giving supervisors additional training on assigning work tasks, the evaluation report did not develop any corrective actions to address the lack of training for this critical task. The inspector noted that NRC unresolved item 94-03-01 identified several concerns in the maintenance training area that warranted management attention. The lack of worker training and familiarity with the air-lock equipment directly contributed to this event. This is a violation of 10 CFR 50, Appendix B Criterion II, which specifies that personnel performing activities affecting quality shall be provided with the necessary indoctrination and training to assure that suitable proficiency is achieved and maintained. **(VIO 50-443/94-08-01)**

The inspector also identified two minor errors contained in the report. The evaluation report indicated that the procedure contained no statements indicating any potential safety hazards, when, in fact, procedure steps 8.2.3 and 8.2.5 did contain caution statements. Also, the report indicated that the event team reviewed revision 1/change 2 when revision 1/change 3 was in effect. The inspector informed a regulatory compliance engineer of these two minor errors.

The inspectors reviewed the associated atmospheric and containment pressure monitoring temporary modification and the two new operational procedures. The inspector passed through the air-lock several times and witnessed successful operation of the interlocks and doors. The inspector judged that the new procedure methodology substantially reduces the consequence of a personnel error. The operational procedures specifies that an operations manager direct this evolution. The inspector concluded that, with the exception of the assessment for maintenance training/job task analysis adequacy, the licensee generally performed a thorough review of this event. The inspector has no additional questions in this area.

2.4 Primary Component Cooling Water Heat Exchanger

During the current outage, the primary component cooling water (PCCW) heat exchangers in both trains were inspected by eddy current testing (ECT) of the tubes. The "B" train heat exchanger (CC-E-17B) was examined first with the results of the ECT indicating a general pitting/erosion problem on the inner surface of the tubes. Approximately half of the 3,120 tubes exhibited some percentage of wall thinning in the localized areas of the pitting. While the number of tubes which required plugging, using 50% or greater wall loss as the plugging criteria, was only on the order of 40 tubes, the extent of the problem led the licensee to initiate an intensive corrective action analysis not only to determine the root cause, but also to evaluate the various options for corrective measures.

The plugged tubes were re-examined with a different ECT coil; tubes were pulled, sectioned and sent to independent materials laboratories for analysis; and hydrolasing of the inside surfaces of the remaining tubes, with repeated ECT, was implemented to properly characterize both the nature and extent of the overall problem. An engineering evaluation (no. 94-013) was documented to restore CC-E-17B to an operable status on an interim basis to support the removal from service and similar examination and ECT of the "A" train heat exchanger (CC-E-17A). The licensee issued a station information report (SIR 94-32) to document and control the corrective action progress. Since all of tubing in both heat exchangers had been replaced with similar 90-10 copper nickel tubes during OR01 in 1991, the licensee's assessment also evaluated events (e.g., ocean storms affecting service water turbidity) during the last two cycles, which may have adversely impacted the passivity of the protective layer on the tubing inner surface.

The ECT on CC-E-17A revealed similar degraded tubing results. Therefore, licensee management made the decision to replace all of the tubes in both heat exchangers. Results from the material analysis from the independent laboratories confirmed that flow induced erosion represented the dominant cause of the wall thinning pits. The licensee also initiated design change work to reduce the turbulence of the service water flow at both the inlet (i.e., flow straightener baffle plates in the heat exchanger upper channel head space) and outlet (i.e., throttle downstream valves, SW-V-19 and SW-V-20 per MMOD 94530 to sustain a positive pressure to avoid a vacuum condition) of the heat exchangers. In the longer term the licensee is considering changing the size of an installed orifice areas. The same contractor that was involved with the retubing in 1991 was brought in to again to perform the work. The licensee extended the duration of OR03 to accommodate this additional scope of work and an alternate spent fuel pool cooling system (see section 3.2 of this inspection report) was again put in service to allow both trains of PCCW to be taken down and worked concurrently.

The inspector reviewed the events in progress and work activities related to the above sequence of ECT, results review, and licensee analysis and decision-making processes. The condition of tubes removed from CC-E-17B were examined and discussed with cognizant system and materials engineers. The inspector reviewed SIR 94-32 and engineering Evaluation 94-013, as well as the related documents regarding the setup and use of the alternate spent fuel pool cooling system. Meetings were held with licensee technical support and management personnel, as necessary, to discuss both the status of the evaluation methodology and the criteria involved with the final decision process. The inspector also reviewed the PCCW heat exchanger vendor maintenance manual and specification sheets to confirm the licensee's proper consideration of design data in its problem analysis.

The inspector verified that a significant margin existed between the nominal wall thickness (0.049") and the minimum wall requirements (0.006") for the heat exchanger tubing. This information was useful in assessing the content and substance of the licensee's engineering evaluation and in evaluating the actual safety impact of the identified degraded tube conditions. Based upon this information, the inspector concluded that operability of both PCCW heat exchangers was not in question during the previous operating cycle.

Thus, the final licensee decision to retube both PCCW heat exchangers, considering the delay incurred in OR03 completion, represents a significant and conservative position by licensee management to ensure a safe and uninterrupted fourth cycle of operation. Given the uncertainty involved with the exact cause of pitting/erosion initiation, the NRC considers this decision to be prudent and to also exemplify the judicious, safety conscious approach with which emergent problems identified during this refuel outage are being handled.

3.0 ENGINEERING (37551, 37700, 92700)

3.1 SI Pump Run-out Issue

The inspector performed an assessment of how the engineering staff evaluated and resolved the safety injection pump run-out issue. On October 1, 1991, Westinghouse (W) Energy Systems Electric Corporation issued an informational letter to North Atlantic. In this letter, W, identifies potential safety issues with emergency core cooling system pump run-out limits.

The technical concern of the SI pump run-out issue involves the ECCS operation in the hot recirculation phase when the residual heat removal (RHR) pumps provide suction boost to the suction of the SI pumps. The suction boost increases the suction pressure and causes the SI pumps to run-out further, which may cause SI pump damage if the system balancing did not account for this boost effect. While no report pursuant to 10 CFR 21 was issued by W relative to this potential deficiency, the letter does recommend actions to be considered by North Atlantic. The W letter indicates that the SI pumps at Seabrook had a maximum continuous run-out limit of 675 gpm. W recommends that if the potential to exceed the run-out limit exists, then the licensee should modify the system configuration to assure that pump operability will not be challenged.

In the short term, the engineering staff performed an operability evaluation dated November 19, 1992. Engineering calculations revealed that during hot leg recirculation, a failure of one SI pump could result in exceeding the 675 gpm limit by as much as two percent. The evaluation provided justification that the ECCS subsystem remain fully operable. The hot leg recirculation phase is initiated approximately 18 hours after a postulated accident, post-LOCA configuration. With zero backpressure, the RHR pumps are capable of providing the necessary decay heat removal flow. In the longer term, the licensee developed two special test procedures to check, and adjust as necessary, the actual SI pump run-out flow rates.

During this inspection period, the licensee performed a special test procedure to determine the actual SI pump run-out flow rates under worst case conditions. The inspector observed the conduct of this test. The licensee used a clamp-on sonic flow meter, which indicated, after post-calibration adjustment, that the "A" SI pump ran-out at 680 gpm (not acceptable), while the "B" SI pump ran-out at 647 gpm (acceptable). The operators initiated a station information report. The regulatory compliance manager indicated that North Atlantic will submit a voluntary licensee event report describing the safety significance of this issue and to alert other plants of the potential run-out problems. The licensee performed Procedure ES94-1-2, Safety Injection Hot Leg Flow Balance, which adjusted the manual SI hot leg throttle valves. After adjusting the throttle valves, the "A" and "B" SI pump flow rates were lowered to 624 and 622 gpm, respectively.

The inspector discussed the method that the licensee resolved the SI pump run-out issue with NRR reactor systems branch engineers. The inspector discussed the issue with the engineering manager and technical support personnel. The inspector reviewed the accident analysis and the system design contained in the updated safety analysis report. The inspector concluded that the licensee addressed the longer term aspect of the SI pump run-out concern by measuring the actual run-out flow and adjusting the manual throttle valves, as necessary. The submittal of a voluntary LER will inform the NRC and other plants of the possibility and safety significance of exceeding the SI pump run-out flow.

The inspector concluded that the engineering staff properly resolved the SI pump run-out issue in the long term; however, certain aspects of the interim response could have been better. First, North Atlantic could have resolved this issue in a more timely manner. The licensee received the W letter on October 8, 1991 and did not perform an operability evaluation until one year later. Further, the licensee missed a SI pump run-out testing window of opportunity during the second refueling outage, which occurred September 7, 1992 to November 11, 1992. Second, the engineering staff did not perform a reportability determination when engineering calculations showed the SI run-out flow could exceed the manufacturer's limit. Third, while the operability determination was of adequate quality, the evaluation did not incorporate a W recommendation to add cautions to emergency operating procedures to give the operators guidance to limit SI pump run-out. The inspector discussed these concerns with the engineering manager who acknowledged the inspector's assessment. The inspector has no additional questions or unresolved safety concerns relative to the licensee's response to this potential safety issue identified by W.

3.2 Temporary Modifications

The inspector reviewed the following Temporary Modifications (TMOD) for proper consideration of 10 CFR 50.59 criteria and the documentation of an appropriate safety evaluation. Equipment modified or installed by the implementation of these modifications was inspected for configuration and control provisions delineated in the TMOD work request packages. Additionally, the inspector verified the existence of interdepartmental coordination for TMOD effectiveness where the temporary installations impacted different organizational responsibilities and departmental work controls.

- 93TMOD023, Temporary containment penetrations in support of steam generator sludge lancing and eddy current testing operations

- 94TMOD022, Alternate spent fuel pool cooling (ASFPC) system operation

The inspector checked that the temporary valves associated with 93TMOD023 had been correctly incorporated into the Operations Department Instruction ODI.33, governing the containment integrity capability status. Valve position and tagging were coordinated to ensure the required containment integrity during core alterations. The inspector examined the affected spare containment penetration (ES9) from both the inside and outside of containment and determined that appropriate controls had been established for TMOD installation.

With respect to 94TMOD022, the inspector noted that the NRC had previously reviewed the plan to implement ASFPC system operation, in conjunction with DCR 90-42, in support of OR01 work. The previous related inspection activities are documented in inspection reports 50-443/90-23, 91-01, 91-04 and 93-14. During this inspection, additional design considerations (e.g., spent fuel pool heat load conditions) and operational contingencies were revised to ensure that the temporary cooling towers, allowed by DCR 90-42, would satisfactorily provide a heat sink for the current OR03 conditions. Successful implementation of this TMOD allows both PCCW trains to be removed from service to support heat exchanger retubing activities as discussed in section 2.4 of this report. The inspector checked the licensee's evaluation of safety margins while utilizing the ASFPC system and confirmed that a limited system operation period was assumed to probabilistically deal with external design basis events. The inspector verified the Operations Department Action Status Tracking Log documents a limiting condition for operation (LCO) of the ASFPC system for a 56 day period, ending July 10. The Operations Department also has published a standing operating order (no. 94-014) and an operating procedure (OS94-1-3) dealing with the ASFPC system normal system operations and contingent line-ups. Both the TMOD and procedural controls related to the ASFPC system were approved by the SORC prior to the initiation of the temporary cooling system operations.

With regard to the above temporary modifications reviewed and field inspected, the inspector has no remaining questions. No unresolved safety issues or concerns involving the adequacy of licensee TMOD evaluation process were identified.

4.0 PLANT SUPPORT (71707, 81700, 82701, 84523)

4.1 Radiological Controls

The inspector observed general radiation worker practices and other radiation program controls within the radiologically controlled area (RCA). Because of extensive OR03 activities and the increased number of personnel, including contractors, working within the RCA, the licensee's Radiation Protection Program was significantly challenged during this inspection period. Based upon the licensee's ALARA goals established for the refuel outage, as well as other management objectives (e.g., source term reduction, RCS shutdown chemistry initiatives), licensee performance in the radiological controls area has been excellent.

Where emergent work was determined to require unscheduled entries into high radiation fields, ALARA planning and health physics controls were strongly evident. This is exemplified by the preparations for a reactor pressure vessel (RPV) head inspection, where the Seabrook Unit 2 RPV head was used as a mockup for planning, practice and preparation to achieve minimum personnel radiation exposure. The inspector attended a planning meeting where technical support and health physics personnel developed a plan to inspect a surface indication on the underside of the reactor vessel head. Discussions focused on reducing the amount of stay time needed to perform the inspection in order to minimize radiation exposure. This example is discussed further in NRC inspection report 50-443/94-09. Additionally, the inspector observed comprehensive health physics (HP) coverage of work activities and routine surveys with the containment building, where HP checkpoints were established on two separate floor elevations and closed circuit television (CCTV) capabilities were used in support of radiological controls. As an example, the inspector noted plant workers closely following radiation work permit requirements while cleaning the threads on the reactor pressure vessel studs. Also, generally clear and adequate radiological protection postings were observed by the inspector throughout the RCA.

The inspector identified no unresolved safety concerns relative to the radiation protection program, or implementation of its provisions during plant outage conditions, during random inspection-tours within the plant RCA, including several checks within containment. Good controls were in evidence, particularly in support of any special evolutions, e.g., core alterations.

The inspector also reviewed the Station Chemistry Manual procedure, CP 9.1, describing the effluent limits and conditions governed by the National Pollutant Discharge Elimination System (NPDES) permit number NH0020338, issued by the Environmental Protection Agency to Seabrook Station. Discussions were held with cognizant chemistry department personnel, regarding discharge points and NPDES sampling criteria. The inspector noted that the Brown's River discharge point has been terminated, but that periodic observation and chemical grab sampling continues at other places on the site, as such would affect discharges through the circulating water tunnels to the Atlantic Ocean. A new revision to CP 9.1 is being processed to reflect the elimination of the Brown's River discharge point and sample criteria.

The inspector has previously reviewed effluent discharge criteria and permit provisions in the control room while releases were in progress. The inspector noted close coordination between the operations and chemistry departments in monitoring the temperature difference between the circulating water intake and discharge structures, in accordance with NPDES permit requirements. Overall, the licensee monitoring program for effluent discharges appears to be effective and well controlled.

4.2 Security

The inspector toured the protected area and several vital area zones, observing security guards on patrol and at stations where compensatory measures were posted. The inspector noted several situations where the work requirements of the refuel outage required security guard force coverage to control access to specific plant locations. In all cases, discussions with the guards revealed personnel knowledgeable in their assigned duties and the required controls.

The inspector also witnessed personnel access into the protected area, noting appropriate security force directions (e.g., visitor escort) and actions (e.g., body frisking) when required. A situation involving the identification of an unattended badge/keycard was handled correctly by termination of the keycard, check for unauthorized transactions and documentation in an Incident Report, and as a logable safeguards event.

The inspector also verified, through discussion with security force managers, the existence of an active fitness-for-duty program. This was confirmed by the Nuclear Safety Assessment of the Security Program, Audit Report No. 94-A03-02. The inspector reviewed the security audit results, noting a generally strong and effective implementation of the Seabrook Security Plan. The Audit itself evidenced a comprehensive evaluation of program implementation to the criteria delineated in 10 CFR 73, 10 CFR 26, and the Seabrook Technical Specifications.

No unresolved safety issues or concerns were identified as a result of the routine inspection of the security activities observed or reviewed during the refuel outage.

4.3 Emergency Preparedness

The inspection of the Seabrook Emergency Response Organization notification system (ERONS) drill, conducted on March 16, is documented in NRC inspection report no. 50-443/94-05. At that conclusion of the unannounced, back shift mobilization drill, the licensee determined that the drill objectives had only been partially satisfied.

During the current inspection period, the inspector discussed with licensee emergency preparedness (EP) personnel the corrective measures initiated to ensure a capable ERONS while the program is further reviewed, and until another off-hours, unannounced drill is conducted. The inspector also reviewed the completed OIR 94-058, documenting the EP corrective actions, as well as a self-assessment of the last ERONS drill problems and potential causes. Among the corrective measures that have been implemented were the issuance of pagers to compensate for uncertainty in the ERONS equipment capabilities, revisions to the Emergency Response Organization stating levels for "key responders" and the development of additional administrative controls regarding EP responders assignments and emergency reporting details.

The inspector verified that the licensee submitted to the NRC, in accordance with 10 CFR 50.54q, a report of the resulting revisions to the Seabrook Station Radiological Emergency Plan (SSREP). North Atlantic in a letter (NYN-94041), dated April 15, provided the NRC with the implemented changes and determined that the revisions do not decrease the effectiveness of the SSREP. The inspector verified that the licensee intends to conduct another off-hours, unannounced drill after the completion of OR03.

The inspector has no further questions at the present time regarding the ERONS implementation plans and SSREP revisions. This area will be the subject of additional NRC inspections, as the effectiveness of the changes will have to be demonstrated at the drill planned for the future.

5.0 MEETINGS (30702)

Two resident inspectors were assigned to Seabrook Station throughout the period. An additional resident inspector provided augmented inspection coverage of electrical maintenance and testing activities for OR03 during a two-week period in April. Other NRC inspections conducted over the course of this routine resident inspection are listed below. The inspectors conducted back shift inspections on April 28 and May 9, 16, 19 and 23, and deep back shift inspections on April 19, 22, 23, 24, 25, 27 and 30, and May 7.

- April 18-21, Physical Security (two inspectors), IR 50-443/94-07
- April 25-29 and May 7-13, Safety Assessment and Quality Verification (five inspectors), IR 50-443/94-80
- April 25-29, May 2-6 and May 16-20, Inservice Inspection Program (two inspectors), IR 50-443/94-09
- May 2-9, Operator Licensing (OL) Examinations (four OL examiners), IR 50-443/94-10-OL
- May 9-13, Radiological Protection Program (one inspector), IR 50-443/94-12

Throughout the current resident 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.

Additionally, from April 11-22, 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 diesel generator), 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.

On April 14, North Atlantic management personnel met with Region I managers in King of Prussia, Pennsylvania to discuss overall Seabrook Station performance issues, plans for OR03 and actions related to the Personnel Error Response Team activities, as documented in a North Atlantic letter (NYN-94036), dated April 8, to the NRC. The NRC response to this licensee correspondence, which was issued on April 28, documents the conduct of this meeting and both the NRC and licensee attendees.