U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Number: 95-13

Docket No.: 50-443

License No.: NPF-86

Licensee: North Atlantic Energy Service Corporation
Post Office Box 300
Seabrook, New Hampshire 03874

Facility: Seabrook Station

Dates: October 3, 1995 - November 13, 1995

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Approved By: John F. Rogge 12/28/95
John Rogge, Chief Date
Division of Reactor Projects, Branch 8

Inspection Summary: Inspections were conducted during normal and backshift hours in the areas of plant operations, maintenance, engineering, and plant support.

Routine inspections included feedwater isolation valve actuator hand rail modifications and diesel generator fuel oil transfer system surveillance. Initiative inspections included observation of new fuel receipt inspection, Main Steam Safety Valve lift setpoint verification and adjustments, and 18 month diesel generator integrated surveillance testing. Reactive inspections included inspector review of reactor thermal power exceedance and radiological practices in responding to an injured person within the Radiological Controlled Area (RCA).

Results: The results of the inspection are summarized in the Executive Summary. During the period one non-cited violation was identified regarding exceeding licensed thermal power by a small fraction on two separate occasions.

EXECUTIVE SUMMARY

SEABROOK STATION
NRC INSPECTION REPORT NO. 50-443/95-13

Plant Operations: During the period, on two separate occasions, the reactor was operated in excess of the licensed rated thermal power of 3411 MWt. The first event occurred when operators increased power following an indicated reactor power decrease when a steam flow transmitter was calibrated. The second event occurred following a restart of the Main Plant Computer System (MPCS) and the calorimetric defaulted to the steam flow mode, which had not been renormalized following the first event. The inspector found operators did not fully evaluate and completely understand the cause of the unexplained indicated power decrease prior to taking actions to increase reactor power. The first event also indicated a weaknesses in performing on-line maintenance. Station evaluations identified adequate root cause and corrective actions.

The requalification examinations witnessed and reviewed by the inspectors is written and performed in an acceptable manner. The requalification program for licensed operators was good and in keeping with 10 CFR 55 standards and is administered, updated and maintained in very good order by effective management oversight. Some areas were noted for minor improvement regarding communications, command and control, and control board attentiveness by the operators during simulator scenarios.

The NRC inspectors' review of plant operations past performance did not display any trends or patterns of unsafe operation of the facility. The NRC inspectors noted the very good cooperation between operations and training is reflected in the attitudes of operators in their approach to training and retraining.

Maintenance: Fuel receipt inspection activities were performed safely during the inspection period. The inspector observed personnel perform the activity in a slow, deliberate manner, following applicable procedures. Maintenance personnel performed the handrail modification around the feedwater isolation valves with the plant at power, which had the potential to cause a reactor trip, in a careful manner.

Engineering: The inspector observed main steam safety valve setpoint verification and adjustments with the plant in mode 1. The evolution was performed in a safe, controlled, and well coordinated manner. Personnel followed the procedure, with one exception, which was licensee identified. Four safety valves did not meet the as-found acceptance criteria. The occurrence was documented in an Adverse Condition Report (ACR). The licensee determined the condition was reportable per 10 CFR 50.73 The ACR evaluation demonstrated good safety focus in determining that secondary system overpressure limits would not have been exceeded.

The inspector observed portions of integrated testing of the station diesel generators and engineered safeguards systems. Overall, the test was conducted in a well coordinated manner. Involved personnel were focused during the complex activity. The inspector reviewed selected test results and noted the acceptance criteria was met. In one instance, due to an unexpected response, personnel performed action outside the procedure. The inspector found this did not entirely meet management expectations regarding procedural adherence and the licensee is evaluating the adequacy of pertinent station guidance.

Plant Support: During the period, a worker fell inside containment from an improperly secured step ladder. Health physics technicians determined the worker was not contaminated prior to exiting the radiologically controlled area and subsequent transportation to a local hospital. The inspector verified that proper radiological and emergency plan requirements were followed. The event disclosed the potential need for improved procedural guidance since emergency plan requirements were previously revised.

Safety Assessment/Quality Verification: The Licensee Event Reports (LERs) reviewed during the period were accurate, technically sound, and fulfilled the reporting requirements of 10 CFR 50.73.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the start of the report period, the reactor was at 100% of rated thermal power. On October 30, the licensee identified that on two occasions the plant was operated slightly in excess of the maximum rated thermal power, and reported the condition to the NRC in a 24 hour report (Section 2.2). On November 2 the operators began reducing reactor power in preparation for the upcoming refueling and inspection outage. Starting at 40% reactor power the licensee began setpoint verification and setpoint changes for the Main Steam Safety Valves (Section 4.1). The refueling outage began 11:59 p.m. on November 3 when the generator was disconnected from the grid. The plant reached mode 5, cold shutdown, at 8:42 p.m. on November 4. The 18 month integrated diesel generator surveillance testing was performed during the period (Section 4.2). The plant was in mode 5, day 10 of the refueling outage, at the conclusion of the inspection period.

2.0 PLANT OPERATIONS (71707,71750,92901,93702)

2.1 Plant Operations Review

The inspector observed the safe conduct of plant operations (during regular and backshift hours) in the following areas:

Control Room	Fence Line (Protected Area)
Primary Auxiliary Building	Residual Heat Removal Vaults
Diesel Generator Building	Turbine Building
Switchgear Rooms	Intake Structure
Security Facilities	

Plant housekeeping, including the control of flammable and other hazardous materials, was observed. During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication of equipment status. These records included various operating logs, turnover sheets, tagout, and lifted lead and jumper logs.

Control room instruments were independently observed by NRC inspectors and found to be in correlation amongst channels, properly functioning and in conformance with Technical Specifications. Alarms received in the control room were reviewed and discussed with the operators; operators were found cognizant of control board and plant conditions. Control room and shift manning were in accordance with Technical Specification requirements. Posting and control of radiation, high radiation, and contamination areas were appropriate. Workers complied with radiation work permits and appropriately used required personnel monitoring devices.

2.2 Plant Licensed Thermal Power Exceedance

On October 30, the licensee determined that the plant was operated in excess of the maximum licensed thermal power rating of 341' megawatts thermal (MWt) for the shift average on two separate occasions. The condition was reported to the NRC as a 24 hour report as required by the facility operating license.

On October 19, the plant exceeded licensed thermal power by a small fraction (3413 MWt or 100.06%) when operators increased power in response to an indicated thermal power decrease to 3408 MWt following calibration of one of the 'A' Steam Generator steam flow transmitters. The second power exceedance (3413 MWt or 100.06%) occurred on October 26, following a start of the Main Plant Computer System (MPCS) which reset the calorimetric to the steam flow mode. Previously on October 19 the calorimetric was placed in the feed flow mode while an evaluation of the first event was performed. From the time of the MPCS restart until the next day when Reactor Engineering discovered that the calorimetric was in the undesired steam flow mode, operators made adjustments that caused actual thermal power to exceed 3411 MWt.

Prior to the first event, the licensee was performing planned on-line maintenance to calibrate the various steam flow transmitters. This was the first time the calibration was performed with the plant at power. Although the transmitter was in tolerance, technicians adjusted the transmitter as allowed by procedure and placed the transmitter back in service. Shortly after the transmitter was returned to service the indicated thermal power decreased from 3411 MWt to 3408 MWt. Operators believed a slight unintended boration had occurred due to other evolutions that were in progress at the time. A containment building spray (CBS) pump was recirculating the refueling water storage tank (RWST) and a separate boric acid tank pump surveillance was being run. Operators diluted to achieve 3411 MWt to compensate for the apparent boration. Operators continued to evaluate the potential cause for the unexpected power decrease. Operators contacted Technical Support for assistance in evaluating the condition. Technical Support review indicated that the power decrease was caused by the steam flow calibration adjustment to the steam flow transmitter. At shift turnover the oncoming operators questioned the power increase and Technical Support informed Reactor Engineering of the transmitter calibration and resultant indicated power decrease. Reactor Engineering personnel reviewed the data and quickly realized the transmitter calibration invalidated the normalization constants used for the calorimetric. Planning and scheduling of the transmitter calibration activity failed to identify this potential. The calorimetric was switched to the feed flow mode and reactor power was maintained below 3411 MWt. Feed flow mode was to be maintained until final resolution of the steam flow calorimetric issue. Adverse condition report (ACR) 95-332 was written to evaluate the occurrence. The licensee preliminarily determined the plant did not exceed the licensed thermal power, but the ACR evaluation would make the final determination.

The second event followed a software modification to the Main Plant Computer System. After the modification was complete the MPCS was restarted. The MPCS, per design, returned the calorimetric to the steam flow mode automatically. Operators made adjustments to maintain reactor power at less than 3411 MWt. However, since the steam flow constants had not been re-normalized yet the steam flow calorimetric was approximately 2 MWt lower than the feed flow based calorimetric. When adjustments were made in the inaccurate steam flow mode, actual thermal power exceeded the limit. This condition lasted for approximately 19 hours until Reactor Engineering identified the problem. ACR 95-344 was written to evaluate the occurrence.

On October 30, the licensee completed their evaluation and determined that on both occasions maximum rated thermal power limit had actually been exceeded. After intensive recalculation using historical hourly calorimetric data, Reactor Engineering determined when the adjustments were made to maintain indicated reactor power at or near 3411 MWt, actual reactor power slightly exceeded licensed thermal power (3413 MWt) for the eight hour average. ACR 95-345 was written to evaluate both occurrences. The licensee also performed a Human Performance Enhancement System (HPES) evaluation to evaluate the operators response surrounding the October 19 event. The ACR evaluation determined the causes of both events and identified corrective actions aimed at preventing recurrence. The ACR evaluation concluded the determination whether power was exceeded could have been more timely. The evaluators determined personnel did not exhibit a questioning attitude when it was realized that the steam flow calorimetric was unreliable. The evaluation also identified the occurrence was another example of on-line maintenance weaknesses and would be captured in the ACR which is broadly addressing that issue.

Beginning on October 20 and through the final ACR evaluations the inspector held considerable discussions with involved personnel and station management and was significantly involved in the initial questioning of whether thermal power had been exceeded. Initially on October 20, the inspector questioned Operations management and Reactor Engineering personnel if the thermal power limit had been exceeded following the steam flow transmitter calibration. At that time, the licensee initially determined that thermal power limits had not been exceeded. The inspector specifically questioned Operations department on how this was done and whether Reactor Engineering was consulted to perform the determination. It was not clear to the inspector how the determination was done and whether Reactor Engineering had been consulted to determine if thermal power had been exceeded. The inspector, after discussions with Reactor Engineering supervisor, learned that a formal determination would be part of the ACR evaluation.

The inspector reviewed the completed ACR and HPES evaluations after conclusion of the inspection period. The inspector considered the root cause determinations were adequate and that recommended corrective actions appeared appropriate. The ACR evaluation was also appropriately critical. The HPES evaluation concluded operators responded appropriately to the October 19 occurrence. Although a good initiative, the evaluation did not appear to critically evaluate operator performance. The inspector found that although the planning and scheduling process missed the potential for the transmitter calibration to invalidate the calorimetric, the operators had a potential opportunity to prevent the power exceedance on October 19. Specifically operators did not fully understand why indicated reactor thermal power decreased to 3408 MWt, nor did they consult Reactor Engineering or I&C, prior to restoring power to 3411 MWt. The on-coming operators, who at shift turnover critically questioned the calorimetric, displayed a good questioning attitude. The inspector agreed with the licensee conclusion regarding timeliness of the thermal power exceedance determination. The inspector noted that ACR 95-332 was not written until October 23 when the event occurred on October 19. The inspector agreed with the licensee assessment of minimal actual safety significance since thermal power remained below 3479 MWt, which corresponds to 102% of rated thermal power and therefore is bounded by the plant accident analyses. Due to the corrective actions planned or taken, the thorough extensive thermal power recalculation performed by Reactor Engineering which identified the thermal power exceedance and the negligible safety consequence the occurrences are considered a non-cited violation. The inspector had no further questions.

3.0 MAINTENANCE AND SURVEILLANCE (61726,62703,92902, 60708)

3.1 New Fuel Receipt Inspection

The inspector directly observed new fuel receipt inspection activities (WR 95W000477), on October 12, 19 and 20. Licensee personnel performed the activity using procedure MS 0515.09, New Fuel Offloading, and RS 0722, New Fuel and Core Component Inspection. During the outage the licensee will load 80 new fuel assemblies into the core and change out all 57 removable Rod Control Cluster Assemblies (RCCAs). The inspector reviewed the procedures and held discussions with maintenance and reactor engineering personnel regarding the receipt inspection process and found them appropriately knowledgeable of the task and procedure. The inspector observed adequate foreign material exclusion (FME) controls, which consisted of general FME practices rather than a controlled FME area. A fuel vendor (Westinghouse) representative and a Yankee Nuclear Service Division (YNSD) representative also actively participated in the inspection activities. Movement of fuel from the shipping containers to the new fuel storage vault was slow, controlled and deliberate. Overall, the inspector found the activity was performed safely and according to procedure. Maintenance and Reactor Engineering personnel documented their activities well in the work package. Health Physics technicians provided good support for the activity. The inspector had no further questions.

3.2 Feedwater Isolation Valve Actuator Handrail Modification

On October 20, the inspector observed maintenance personnel modifying the handrails (Work request 95W001830) surrounding the feedwater isolation valves (FWIV) in the west pipe chase. The inspector reviewed the work plan and discussed the work with involved maintenance personnel. The modification was performed to better facilitate FWIV actuator replacement during the upcoming refueling outage. The licensee considered the work "trip avoidance" due to the physical location of the work relative to the valve actuators and the potential to trip the reactor. Maintenance personnel utilized careful work practices to avoid contacting the valve actuator. The inspector verified personnel were aware of the work plan requirements regarding notification of control room personnel when entering and exiting the pipe chase to perform this maintenance. The inspector had no further questions.

3.3 Ultrasonic Inspection Preparations

On October 17, the inspector observed maintenance personnel grinding welds (work request 95W000546) on the train B service water system above ground field welds to support ultrasonic (UT) inspections. The inspector verified that an ignition source permit was issued and that a firewatch was present during the grinding activity. The inspector observed the presence of combustible material below the work area and that sparks from the grinding were landing in the vicinity of the combustible materials. The inspector informed the firewatch who promptly removed the combustible material. The firewatch indicated that the presence of the combustible material was contrary to the station procedural guidance as well as not meeting supervisory expectations. The inspector discussed the occurrence with maintenance department management. The inspector found there was no actual safety consequence and had no further questions.

3.4 Diesel Generator Fuel Oil Transfer System Surveillance

On November 4, the inspector observed the performance the diesel generator fuel oil transfer pump flow verification 18-month surveillance (EX1804.023 and RTS 95RE00103001). The surveillance verifies the capability of either diesel generator fuel oil transfer pump to transfer fuel oil to either fuel oil day tank from either fuel oil storage tank as required by Technical Specification 4.8.1.1.2.F.11.

The inspector verified test personnel performed the procedure correctly. The test personnel used good communication during performance of the test. A noteworthy strength was the self-checking used during the multiple valve positioning required to perform the surveillance. The surveillance successfully demonstrated the fuel oil transfer capabilities required by the plant technical specifications. The inspector had no further questions.

4.0 ENGINEERING (71707,37551,92903,40500)

4.1 Main Steam Safety Valve Testing

The inspector witnessed insitu setpoint verification and setpoint changes for the Main Steam Safety Valves (MSSVs) during Mode 1 with the plant at approximately 40% of rated thermal power and the power range neutron high flux trip setpoints reduced to 48%. The testing was performed using procedure EX 1804.041, "Main Steam Safety Valve Inplace Setpoint Verification," and RTS 95RE00119001. The procedure provides instructions for setpoint verification and the lift setpoint adjustment when using the Furmanite Trevitest system. The procedure also provided instructions for adjusting lift setpoint for applicable MSSVs to the new setpoint values associated with the Technical Specification amendment values. The licensee submitted a Technical Specification amendment which revised the MSSV lift setpoints and the maximum Power Range Neutron Flux - High Setpoints with inoperable MSSVs to assure the consequences of postulated overpressure events will remain within the Basis of Technical Specification 3.7.1.1. This was necessary after a Westinghouse Nuclear Safety Advisory Letter, NSAL94-001 and NRC Information Notice 94-60 indicated there may be non-conservatisms with the Power Range Neutron Flux - High setpoints with inoperable MSSVs provided in Technical Specifications. The licensee initially issued Technical Clarification, TS-011, for interim guidance, that included the revised maximum allowable neutron flux high setpoints with inoperable MSSVs. NRC inspection reports 50-443/94-03, 50-443/95-08 and 50-443/95-15 pertain. The inspector observed portions of the testing from both the East and West pipe chases where the valves are located, held discussions with involved licensee personnel and the Furmanite test personnel, reviewed the procedure and completed data sheets. The inspector verified proper LCO action statements were entered.

The inspector found that the licensee performed the testing in a controlled and well coordinated manner. Communications for the most part were good, the inspector identified some instances in the field where less than formal communication were utilized. The informal communication had no impact on testing. Licensee and vendor personnel were thoroughly knowledgeable of the activity. The inspector noted that during post test calibration checks of the test equipment, the contract test personnel began the evolution without using the procedure and missed a step. This was identified by contract personnel and the procedure was subsequently used and post test calibration performed satisfactorily.

The inspector reviewed the test results and found that as-left acceptance criteria were met. Four safety valves did not satisfy the as-found criteria of $\pm$ 3% by a small percentage. After the second lift, without any setpoint adjustment, the valves tested within this range. The remaining 16 valves were within the as-found acceptance criteria. Each Steam Generator has five associated safety valves and three of the five failed the as-found acceptance criteria non-conservatively on Loop 2. Adverse Condition Report ACR 95-355 was written to document the as-found failures. The licensee determined the event was reportable per 10 CFR 50.73 as a condition prohibited by plant Technical Specifications and submitted an LER, since at some time in the operating cycle the lift setpoints were not within tolerance and therefore inoperable. The inspector reviewed both the completed ACR and LER which were issued after the close of the inspection period. The licensee determined contrary to Technical Specifications Seabrook Station operated at full power with up to three inoperable MSSVs in one loop for an indeterminate period of time. The ACR evaluation assessed the significance of the as-found condition for potential inadequate overpressure protection during the previous operating cycle. The licensee evaluation verified using the as-found MSSV lift setpoints would not have resulted in secondary system pressures exceeding the Condition II (Events of Moderate Frequency) pressure limit. The limiting Updated Final Safety Analysis (UFSAR) overpressure transient is Loss of Load/Turbine Trip. The inspector reviewed the supporting licensee calculation. The inspector found the ACR evaluation was technically sound and demonstrated a strong safety focus in evaluating the potential safety consequence of the non-conservative setpoints. The inspector had no further questions.

4.2 Diesel Generator And Engineered Safeguards Integrated Surveillance

On November 7, the inspector observed performance of Diesel Generator 1A 18-month Operability and Engineered Safeguards Pump And Valve Response Time Testing Surveillance (EX 1804.001). The purpose of the comprehensive test included testing of the diesel generator interlocks, start and standby functions upon receiving a safety injection signal. The test verified emergency core cooling system (ECCS) pump and valve response times, the capability of the diesel generator to withstand a full load reject, and the diesel generator response to Safety Injection (SI), Loss of Power (LOP), a combination of LOP/SI and Tower Actuation (TA) signals. This integrated test is considered a complex procedure by the licensee and as such requires additional administrative controls that govern the activity according to station policies contained in the Site Management Manual (SMM). The testing was performed in mode 5. The inspector reviewed the procedure and SMM, observed the pretest briefings, held discussions with involved test personnel, verified current test equipment calibrations, observed testing, and reviewed the completed test data and the complex test procedure critique.

The inspector found the preshift briefing was attended by all pertinent personnel and the briefing met the licensee administrative requirements for a complex procedure. The testing was performed in a coordinated and controlled manner. Roles and responsibilities were clearly discussed and defined including responsibilities for test control and plant operations.

The inspector noted the evolution was performed according to procedure. Some minor test discrepancies were identified and properly documented. The licensee evaluated the test discrepancies, and determined the surveillance acceptance criteria were met and presented the test exceptions to the Station Operation Review Committee (SORC).

The inspector observed at one point in the procedure test personnel identified the Startup Feed Pump (SUFP) would not start following the procedure step as written and the SUFP pre-lube pump was already running when the procedure directed that the pump be started. The test director held discussions with Operations Shift Supervision and the situation was evaluated. A consensus was reached and certain actions were taken that were not specifically governed by procedure. The course of action taken had no adverse impact or safety consequence, minimized the time the plant was in an abnormal lineup, and allowed the procedure to recommence. The inspector questioned whether this course of action was recognized by station management or procedures since the procedure did not work as written and no procedure change was implemented prior to taking the action which allowed the procedure to recommence.

The inspector reviewed selected test data and found acceptance criteria were met and discrepancies resolved. The inspector reviewed station guidance on procedural adherence and could not conclusively determine whether taking action outside the procedure was acceptable. The inspector held discussions with Station Management who indicated this did not entirely meet management expectations but is evaluating existing procedural adherence guidance. The inspector had no further questions.

5.0 PLANT SUPPORT (71707,71750)

5.1 Radiological Controls

On November 5, a worker inside containment fell from an improperly secured step ladder. Plant personnel quickly responded and were able to assist the individual who had walked to the step off pad (SOP). Health Physics (HP) technicians escorted the individual from the containment SOP to a more suitable location within the radiologically controlled area (RCA) to remove protective clothing and performed a whole body frisk. The HP technicians determined the individual was not contaminated. The person was then transported to the hospital where he was treated and released.

The inspector held discussions with HP supervision to determine if appropriate actions were taken regarding RCA exit procedures for potentially contaminated injured person as well as any Emergency Plan requirements regarding injured personnel. The inspector reviewed Operations Department Instruction (ODI) 32, Medical Emergency Response, Security Department Instruction (SDI) 25, Fire and Medical Emergency and emergency action levels (EALs). The EALs were reviewed to determine if any event classification was necessary. The inspector found through review of the Emergency Response Manual and discussions with HP supervision that previously the licensee revised the EALs to remove the event classification scheme for transporting a contaminated or potentially injured person offsite. Consequently, the licensee deleted the associated emergency plan procedure ER 4.4, RCA Medical Emergency, and incorporated RCA medical emergency provisions into ODI 32. The HP supervisor indicated that although the response to the injured person was handled appropriately the occurrence revealed some weaknesses regarding adequate guidance for health physics personnel in responding to a medical emergency within the RCA since the former emergency plan procedure had been deleted. The HP department had previously issued standing orders that contained the guidance of the deleted procedure, however, it appeared to HP supervision that the guidance should be formally proceduralized. The licensee is evaluating the need for specific procedural guidance. The inspector confirmed proper radiological practices were implemented throughout the response to the personnel injury inside the RCA. The inspector had no further questions.

5.2 Security

The inspectors noted good control and oversight of numerous station and contractor personnel entering the protected area during the beginning of the refueling outage. Specifically, the inspectors noted vigilant security officer observation and direction of plant personnel access through the various detection equipment when entering the protected area via the security gatehouse. Security management was also observed providing additional oversight.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (92700)

6.1 Licensee Event Report Review

The inspectors reviewed Licensee Event Reports (LERs) submitted to the NRC to verify accuracy, description of cause, previous similar occurrences, and effectiveness of corrective actions. The inspectors considered the need for further information, possible generic implications, and whether the events warranted further onsite followup. The LERs were also reviewed with respect to the requirements of 10 CFR 50.73 and the guidance provided in NUREG 1022 and its supplements.

6.1.1 LER 95-06, Reactor Thermal Power Exceedances

LER 95-06, "Reactor Thermal Power Exceedances," dated November 29,1995 documented the licensee identification that licensed reactor thermal power was exceeded on two separate occasions. Section 2.2 of this report reviewed the event in detail. The inspector found the LER accurately described the event and the associated root causes and corrective actions. The inspector had no further questions.

6.1.2 LER 95-07, Main Steam Safety Valve Setpoint Testing Failures

LER 95-07, "Main Steam Safety Valve Setpoint Testing Failures," dated December 3, 1995 documented the failure of four Main Steam Safety Valves (MSSVs) to meet the as-found criteria as required by Plant Technical Specifications and ASME Boiler and Pressure Vessel Code. Section 4.1 of this report reviewed the test failures in detail. The inspector determined the LER was an accurate description of the event and the reportability and safety consequence evaluations were good.

7.0 NRC MANAGEMENT MEETINGS AND OTHER ACTIVITIES (71707,40500)

7.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on December 28, 1995, summarizing the preliminary findings of this inspection. No proprietary information was identified as being included in the report.

7.2 Other NRC Activities

During the week of October 16-20, two NRC Region I operations engineers conducted an operator licensing requalification examination. The results of this inspection are documented in the attachment to this report.

NRC Inspection Report No. 50-443/95-13

ATTACHMENT 1

REQUALIFICATION TRAINING PROGRAM INSPECTION

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NO:	50-443/95-13
LICENSEE:	North Atlantic Energy Services Corporation
FACILITY:	Seabrook Station
INSPECTION AT:	Seabrook, New Hampshire
INSPECTION DATES:	October 16-20, 1995
INSPECTORS:	J. D'Antonio, Operations Engineer

Original Signed by: 10/30/95

T. J. Kenny, Senior Operations Engineer Date
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

APPROVED BY:

Original Signed by: 10/30/95

Michael C Modes, Acting Chief Date
Operator Licensing and
Human Performance Branch
Division of Reactor Safety

DETAILS

1.0 BACKGROUND AND SCOPE

During the week of October 16, 1995, the NRC conducted a performance-based inspection of the Seabrook Station (SS) requalification training program, using NRC Inspection Procedure 71001, "Licensed Operator Requalification Program Evaluation." The NRC inspectors reviewed the requalification examination materials for the biennial written examination, the annual operating tests administered during the week of the inspection, and selected examination materials from examinations administered earlier in the cycle. Interviews were conducted with licensed operators, training and supervisory personnel to assess the examination process. Selected administrative procedures related to the development and administration of the examinations were also reviewed.

SS utilizes the systematic approach to training (SAT) with the training broken down into "blocks." Each block is the training requalification session that each operating crew and inactive licensed operator must attend. There are six crews who attend training once every six weeks. When all six crews and all inactive licensed operators have received the training for that session the block is ended.

SS has been continually upgrading Emergency Response Procedures and was currently in revision 1B which is the current revision of the owners group.

2.0 FINDINGS

Overall, the SS requalification training was acceptable. The operators did not perform the simulator portion of the examination very well. Although acceptable, some of the scenarios showed a repeat in previously seen weaknesses such as communications decorum, command and control and attentiveness to the control board operation. The assessments of the simulator exercises, however, were performed very well. Job performance measures (JPM) were more challenging than the NRC inspectors normally see because of minor detractors added in order to challenge the operators. These were also assessed very well by the SS staff. The written examination was assessed as good, the questions were appropriate and at the proper technical level for a crew requalification examination. The inspectors also concluded that there is very good management oversight of the requalification program. There is sufficient self-assessment to continually evaluate the program and upgrade it to keep up with on-site and off-site changes that could influence operator retraining.

2.1 Review of Operating History

The NRC's review of the operational performance events, since the last inspection, show continued safe operations at SS. The NRC inspectors did not see any incidence of gross operator errors requiring changes to the operator requalification program. In one incidence, the inspectors noted the operations manager had inserted a reactor trip scenario, which had occurred in the plant, into training in order for all operators to be informed of the event and to practice it on the simulator.

2.2 Requalification Examination Development

All of the examinations met the requirements of NUREG/BR-0122 "Examiners' Handbook for Developing Operator Licensing Examinations." The written examination was appropriate and at the proper technical level for the crew being examined. The scenarios were less challenging than the NRC would administer; however, they met all of the Examiners' Standard criteria.

Written

The NRC inspectors reviewed the written examination given just prior to the week of the inspection and one earlier examination, along with two weekly quizzes. These exams were at an appropriate technical and comprehension level for requalification and contained little or no overlap between examinations.

Scenarios

The NRC inspectors observed the four scenarios administered during this inspection and reviewed four additional scenarios from the facility examination bank. Although these scenarios met examiner's standard criteria, they were less challenging than those typically seen at other facilities and did not go deeply into the Emergency Operation Procedures (EOPs), Functional Recovery Procedures (FRPs), and Emergency Contingency Actions (ECAs). The above procedures were entered and operator proficiency was demonstrated, then quickly exited, without extensive action required. The scenarios did, however, involve more demonstration of Abnormal Operation Procedure (AOP) depth than usual.

The detail level provided in the scenario write ups was sparse, and consisted of a brief outline of events with little detailed breakdown of expected operator actions or indication of where the critical tasks occurred. The inspectors were concerned this lack of detail in procedures can result in the quality of an evaluation being highly dependent on the individual evaluator's knowledge of procedures and operating standards. This was discussed with the training manager who indicated that he would consult with his peers from other utilities to find a better method of writing scenarios. The actual evaluations by the SS evaluator instructors after the scenarios were thorough and detailed (see section 2.3).

JPMs

The JPMs used for this examination were good with appropriately safety related tasks. A good feature of many of the JPMs was the extensive use of minor faults. In several of the observed JPMs, the more common use of a single fault was accompanied with some minor detractors to further challenge the candidate. Fifteen different JPMs were observed or reviewed by the inspectors.

2.3 Examination Administration

The facility evaluation of crew and individual performances is very good. The evaluations are thorough, detailed, and critical. Very good evaluations are assessed regarding control board manipulation errors, command and control deficiencies, communications decorum, and coordination and prioritization of tasks.

The facility tended to attribute operational errors or oversights to the unit shift supervisors failing to provide adequate command and control, control bands, or clear communications to the operators. The inspectors would allocate that the operators should as. .e some of the command and control blame as well. Operators should as. fr . the appropriate band for level control if it is not given. The basis for this comment is that the board operators are expected to know the prc edures well enough to anticipate what orders they would be getting in respor.e to a given event and to understand the purpose and intent of an order wh. it is given. After discussions with the facility evaluators they agreed t. t this should be a factor for future evaluations.

All crews passed the requalification .amination and their operational performance was adequate. The crews p.rform two scenarios each. Both crews, one a shift crew and the other a staf crew, performed well in the first of their scenarios; however, the other i..olving a steam generator tube rupture was performed poorly. Both crews fail d to take action to control pressurizer pressure after blocking the main steam ine isolation, resulting in the block resetting itself as the pressurizer fil ed, thus allowing the isolation which complicates event response by depriving the crew of steam dumps for cooldown. For the operating crew this was simply n error in command and control, discussed earlier. The RO knew he want d pressure to remain below 1950 psig but was not given a control band and became distracted thus taking no action to control pressure after the valve actuation block. The second crew saw pressure rising but took no action because pressurizer level was approximately 100% and they thought sprays would have no effect because of the full pressurizer. In both scenarios the crews recovered with no damage to the core. The training department representative stated that these errors would be factored into the next training sessions.

2.4 Licensed Operator Requalification Training Feedback System

SS has a very good feedback system. The NRC inspectors reviewed self-assessments in the form of: feedback from crews after classroom and simulator training; instructor feedback to upgrade lesson plans after design changes to the plant; feedback into the training program from outside sources, such as NRC, INPO, and others; and, internal audits conducted by quality assurance, and the training department. Through discussions with operators and instructors and a review of the lesson plans, the NRC inspectors verified changes to the training program in the form of upgrades to the simulator, changes to the lesson plans and changes to the methods of conducting the operations regarding command and control and operator feedback communications.

2.5 Remedial Training Program

The facility's remediation practices for past instances of examination failures have been performed very well. The NRC inspectors reviewed annual examination results and weekly quiz results for the past year. There were no operating examination failures. One annual exam failure and five weekly quiz failures occurred. Remediation for the failures was appropriate and documented. The retake exams were of an identical difficulty level to the original exam, with few overlapping questions.

Attendance and makeup records for the prior year were also reviewed. The training missed was documented as having been made up by attending the class during another requalification week. In a few instances, individual classes were made up by self-study.

2.6 Conformance With Operator License Conditions

The requirements of 10 CFR 55(e) and (f) regarding the reactivation of an inactive license are being met. SS has a procedure in place for maintaining the licenses current. Through discussions and a review of documentation the inspectors concluded that the hours on shift are being maintained and cataloged for all active licenses and the medical portions of the licenses are continuously upgraded in accordance with 10 CFR requirements.

2.7 Management Oversight and Self Assessment

Management oversight of the licensed operator requalification program was very good. The inspectors, through interviews and review of documents, concluded that management, including The Executive Director of Nuclear Operations, is visible to the training department and takes an active role in auditing and commenting on the program. The managers appear to be accessible to the training department and the operators for discussion and support on a continuing basis. For example, the review of "Line Observations for Training" documents that are used for observations of all the aspects of training, by SS management, was complete and had comments that were both praiseworthy and critical of the program and the participants.

Management endorsed two programs that were positive and seemed to be enhancing the conduct of the operators during operations: The first is STAR: stop, think, act, review. And second, BAG (before, after, going) where the supervisor or operator reviews what transpired before, what will transpire after and then where he/she is going next regarding any evolutions. The second program is new and is still being reinforced by the training department in their critiques after operations training.

The NRC inspectors witnessed a very good and positive working relationship between the training department and the operations department. The operations manager (OM) is part of the evaluation team that evaluates the requalification examination at the end of each training cycle. The inspectors verified through discussions and personal observations that the OM also works very closely with the training manager to see that the operators get the best and latest training available during their retraining cycle.

3.0 EXIT MEETING

An exit meeting was conducted on October 20, 1995, during which the NRC inspectors reviewed the scope of the inspection. The following personnel attended the meeting.

North Atlantic Energy Services Company

B. L. Drawbridge	Executive Director Nuclear Power
R. M. Cooney	Assistant Station Director
J. Grillo	Operations Manager
R. Hickok	Training Manager
L. Carlson	Operations Training Manager
S. Kessinger	Senior Operations Instructor
R. F. White	Licensed Operator Requal Coord
J. Sobotka	NAC Coordinator

Nuclear Regulatory Commission

T. Kenny	Senior Inspector
J. Macdonald	Senior Resident Inspector
J. D'Antonio	Operations Engineer
D. Mannai	Resident Inspector