March 27, 2001

Mr. Ted C. Feigenbaum
Executive Vice President and Chief Nuclear Officer
Seabrook Station
North Atlantic Energy Service Corporation
c/o Mr. James M. Peschel
P.O. Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION - NRC INSPECTION REPORT 05000443/2001-002

Dear Mr. Feigenbaum:

On February 17, 2001, the NRC completed an inspection at the Seabrook nuclear power station. The enclosed report documents the inspection findings which were discussed on February 23, 2001, with Mr G. St. Pierre and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert Summers, Acting Chief
Projects Branch 6
Division of Reactor Projects

Docket No. 05000443
License No: NPF-86

Enclosure: NRC Inspection Report No. 05000443/2001-002

Attachments: (1) NRC Revised Reactor Oversight Process

cc w/encl:
B. D. Kenyon, President and Chief Executive Officer
J. M. Peschel, Manager - Regulatory Programs
G. St. Pierre, Station Director - Seabrook Station
D. Roy, Training Manager - Seabrook Station
D. E. Carriere, Director, Production Services
L. M. Cuoco, Esquire, Senior Nuclear Counsel
D. A. Smith, Manager of Regulatory Affairs, Northeast Nuclear Energy Company
W. Fogg, Director, New Hampshire Office of Emergency Management
D. McElhinney, RAC Chairman, FEMA RI, Boston, Mass
R. Backus, Esquire, Backus, Meyer and Solomon, New Hampshire
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R. Hallisey, Director, Dept. of Public Health, Commonwealth of Massachusetts
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000443
License No.:	NPF-86
Report No.:	05000443/2001-002
Licensee:	North Atlantic Energy Service Corporation
Facility:	Seabrook Generating Station, Unit 1
Location:	Post Office Box 300 Seabrook, New Hampshire 03874
Dates:	December 31, 2000 - February 17, 2001
Inspectors:	Raymond Lorson, Senior Resident Inspector Javier Brand, Resident Inspector Jason Jang, Senior Radiation Specialist
Approved by:	Robert Summers, Acting Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000443-01-02, on 12/31/2000 - 2/17/2001; North Atlantic Energy Service Corporation; Seabrook Station; Unit 1. Licensee Identified Violation.

The report covers a seven-week period of inspection conducted by the resident staff and a regional radiation specialist. The inspection identified no significant findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP) . The significance of findings for which the SDP does not apply are indicated by “no color” or by the severity level of the applicable violation.

A. Inspector Identified Findings

No significant findings were identified.

B. Licensee Identified Violations

Violations of very low significance which were identified by the licensee have been reviewed by the inspector. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in section 4OA7 of this report.

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Report Details

Summary of Plant Status: The plant began the period in a refueling outage. The operators performed the reactor start-up on January 28, 2001. The plant reached 100% power on February 1, 2001 and was operated at approximately 100% power for the duration of the inspection period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a partial system walkdown of the "A" emergency diesel generator while the "B" emergency diesel generator (EDG) was removed from service for a maintenance outage. During this walkdown the inspectors verified that the redundant system was properly aligned in accordance with plant procedures and system drawings. The inspector also observed whether any material conditions were present that could challenge the operability of the of the "A" emergency diesel generator.

b. Findings

There were no significant findings identified during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured two areas important to plant safety to assess the condition of the fire detection and suppression equipment, fire barriers, and the presence of combustible materials. Station drawings and pre-fire strategy tables were used to verify that fire fighting equipment was available in the field where required and that applicable sections of the fire main were properly aligned and charged. The following areas were reviewed: primary containment, and the emergency feedwater pump-house.

b. Findings

There were no significant findings identified during this inspection.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector reviewed the licensee's flood protection program and performed walkdowns of areas important to safety to access the condition of the internal and external flood protection barriers and procedures. Station drawings and other applicable documentation were used to verify that flood protection equipment and barriers were in good condition and installed in the field where required. The inspector also reviewed several engineering evaluations, the applicable design basis document, condition reports, and the UFSAR to verify that the licensee had implemented measures to protect safety-related equipment from flooding events.

The inspector reviewed the licensee’s corrective actions for condition report (CR)-00-0514 which evaluated a licensee identified condition involving the incorrect positioning of the water/oil separator vault #1 drain valve DF-V156. The inspector also reviewed CR 01-01128 which evaluated two NRC identified issues related to the revision of the alarm response procedure (D5477) for an abnormal oil/water separator vault #1 level, and the updating of a plant sign (operator aid) to ensure correct positioning of the DF-V156 valve. Additionally, the following areas were toured and the following documents were reviewed during this inspection:

Areas toured:

- east and west main steam pipe chases
- “A” and “B” residual heat removal pump vaults
- primary component cooling water exchangers
- “A” and “B” safety injection pump rooms
- “A” , “B”, and “C” charging pump rooms
- the electrical cable spreading room
- service water and cooling tower pump buildings
- emergency feedwater pump room
- the oil/water separator vault #1

Documents reviewed:

- UFSAR Sections 9.3.3 and 9.3.3.4, Equipment and Floor Drainage System
- Drawing, DBD-PB-01, “Plant Barriers”
- Report, AR-98019498, Evaluation Of SER 3-98, Rev.1, Recurring Event, Flooding of ECCS Rooms Caused by Fire Protection System Water Hammer
- Engineering Evaluation, SS-EE-97-002, Rev.00, Plant Drainage System Guidelines
- Engineering Evaluation, 90-50, Internal Flooding Potential Through Plant Drain and Sump Systems.
- Procedure, FR-Z.2, “Response To Containment Flooding"
- Procedure, ES0802.001, "Revetment Surveillance Program"
- Procedure, ON1044.02, "Oil/Water Separation System Operation"
- Procedure, OS1025.01, Rev. 10, "Floor and Equipment Drain System Operation"
- Procedure, RTS 97RE00325001, "5 Year Inspection Of Revetments"
- Procedure, OS0243.02, "Fire Main Break"
- Procedure, OS1200.03, "Severe Weather Conditions"
- Procedure, OS1290.01, "Response To HELB Systems Actuation or Malfunction"
- Procedure, MA 5.7, "Station Barriers, Penetration Seals, and Fire Barrier Wrap"

- Alarm response procedures for abnormal oil/water separator level, EDG building level, electrical tunnel sump level, east and west pipe chases level, containment sump level, "A" and "B" equipment vaults sump level, primary auxiliary building sump level and condenser pit flood level
- Condition Report, CR-00-0514, Engineering Evaluation of incorrect as-found position of valve, DF-V156

b. Findings

There were no significant findings identified during this inspection.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed problems involving selected in-scope systems, structures, and components (SSCs) to assess the effectiveness of the maintenance rule program implementation. The reviews focused on proper maintenance rule scoping, characterization of failed SSCs, safety significance classifications, 10 CFR 50.65 (a) (1) classifications, and the SSC performance criteria. The inspector examined corrective action program documentation (condition reports, 01-000940 and 01-000942), the system performance report and interviewed the system engineer to review the planned and completed corrective actions. The following systems were reviewed:

- Digital rod position indication
- Rod control system

b. Findings

There were no significant findings identified during this inspection.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspector reviewed through direct observation, document review and/or interviews with plant engineering and operations personnel three emergent maintenance activities to determine whether the licensee properly evaluated and controlled these activities to minimize risk. The inspector also performed an independent evaluation of the effectiveness of the licensee’s trouble-shooting activities and the planned and completed corrective actions as described in the applicable on-line maintenance assessments, condition reports (CRs 01-00796, 01-00114, 01-00131), and design change request 95-012. The following activities were reviewed:

- A through wall pipe leak identified at the welded joint in the discharge side piping of the “B” service water (SW) pump, on January 3, 2001.

- Seat leakage through two of the three pressurizer (Pz) safety relief valves RC-V-115 and 117, during the January 25, 2001, start-up and the subsequent installation of a gagging device to seat the valves to stop the leakage.

- The loss of the digital rod position indication (DRPI), and rod misalignments in shutdown bank E-5 that occurred during rod drop testing on January 26, 2001, and shutdown bank D that occurred on January 27, 2001, during start-up.

b. Findings

There were no significant findings identified during this inspection.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspector observed operators perform the plant start-up and power ascension performed on January 25 through 27, 2001, following the seventh refueling outage. The review included pre-startup preparations, reactivity controls, crew performance and procedural compliance. Also, the inspector in part verified that technical specification (TS) requirements, commitments, or pre-requisites for mode changes were met prior to changing modes. The inspector observed portions of the rod drop testing performed per procedure IX1666.911, and the operators' response to the rod mis-alignment events discussed in Section R12 (condition reports 01-000940 and 01-000942).

b. Findings

There were no significant findings identified during this inspection.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed and observed portions of the post-maintenance test activities performed following the overhaul of the "B" emergency diesel generator (EDG). The activities included: several "break-in" runs and the 24-hour endurance test. The inspector also reviewed the licensee's response to metal debris particles that were found in the in the vicinity of the # 5 main bearing after the initial one minute break-in run.

b. Findings

There were no significant findings identified during this inspection.

1R22 Surveillance Testing

a. Inspection Scope

The inspector observed portions of surveillance testing of the turbine driven emergency feedwater (EFW) pump, performed on January 25, 2001. The inspector attended the pre-job brief, performed system and control room walk-downs, observed operators perform test evolutions, and interviewed applicable personnel. Additionally, the following surveillance procedures were reviewed:

- EX1804.032, “Emergency Feedwater Turbine Driven Pump 18 Month Auto Actuation Surveillance”
- OX1436.02, “Turbine Driven Emergency Feedwater Pump Quarterly and 18 Month Surveillance Test and Monthly Valve Alignment”
- OX1436.13, “Turbine Driven Emergency Feedwater Post Cold Shutdown or Post Maintenance Surveillance and Comprehensive Pump Test”

b. Findings

There were no significant findings identified during this inspection.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope

Radiological Effluent Technical Specification/Offsite Dose Calculation Manual (RETS/ODCM) - Radiological Effluent Occurrences

The inspector reviewed the following documents to ensure the licensee met all requirements of the performance indicator from the first quarter 1999 to the fourth quarter 2000 (8 quarters):

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- associated procedures

Reactor Coolant System Activity

The inspector observed a reactor coolant system iodine analysis performed on January 31, 2001, per chemistry procedure, CX0901.02, “Determination Of Dose Equivalent I-131.” The inspector also compared the sample results to the technical specification (TS) limits and previously reported reactor coolant system activity performance indicator data.

Reactor Coolant System Leakage

The inspector reviewed the design documents, procedures, interviewed personnel and reviewed input data to confirm that the licensee was appropriately monitoring and reporting data for the reactor coolant system leakage performance indicator.

b. Findings

There were no significant findings identified during this inspection.

4OA3 Event Follow-Up

The inspectors reviewed an event that occurred on January 8, 2001, which involved the lifting of the “B” residual heat removal (RHR) pump suction relief valve (RH-V-89) during pressurization of the standby “B” RHR train per operations procedure, OS1013.04, “Residual Heat Removal Train B Start-up and Operation.” Relief valve, RC-V-89, remained open for about eight minutes and approximately 320 gallons of reactor coolant were transferred from the reactor coolant system (RCS) to the primary relief tank (PRT).

The plant was in Mode 5 at the time of the event, and the reactor decay heat load was low as indicated by the licensee’s reported time to boil of 180 hours. The RCS temperature was approximately 145 degrees F.; RCS pressure was about 350 psig; and the pressurizer was full. Solid plant pressure control was in effect at the time. The reactor operator (RO) was alerted to the RCS inventory loss by a decreasing RCS pressure indication and by an increasing PRT level indication. The RO adjusted the flow through the charging and letdown systems and isolated the “B” train RHR system from the RCS to stop the inventory loss and stabilize RCS pressure. Relief valve, RC-V-89, reseated and RCS pressure was restored.

The licensee initiated condition report (CR) 01-00193 and formed an event team to review this event. The inspectors observed the operators’ initial response to the event, interviewed operations, maintenance and engineering personnel, and reviewed plant drawings, system operating data, the licensee’s event team report, and operating and maintenance procedures to assess the event significance, as well as the adequacy of the corrective actions.

The event team determined that RC-V-89 lifted due to a problem with operations procedure OS1013.04 associated with pressurization of the standby “B” RHR train. The procedural steps improperly directed the operator to establish a configuration that aligned the “B” RHR pump suction relief valve to the discharge pressure of the operating “A” RHR pump. The relief valve opened since its lift pressure of 450 psig was less than the approximate 500 psig discharge pressure of the operating RHR pump. The inspectors determined that the root cause for this event was appropriate and noted that operations procedure, OS1013.04, was revised to prevent recurrence of this type of event.

During the event, the relief valve remained open longer than expected and allowed RCS pressure to drop below 200 psig before re-seating. The licensee investigated this anomaly and determined that the relief valve did not properly reseat since the nozzle ring setting for valve, RC-V-89, had been misadjusted during maintenance. This resulted in an increased loss of RCS inventory during the event. The licensee inspected additional relief valves that had been adjusted during refueling outages six and seven and identified two such valves (CS-V-148 and CS-V-173) where the ring settings were not properly set. The inspector did not identify any safety concerns related to the improper ring settings on relief valves, CS-V-148 and CS-V-173. The licensee planned to conduct additional training for the maintenance technicians regarding this event and revised relief valve maintenance procedures, MS0519.12, MS0519.13, and MS0519.54 to provide better guidance to the maintenance technicians.

The license identified that the two inadequate procedures were root causes of the event. This licensee identified violation is discussed in section 4OA7 of this report. The inspector discussed this event with and provided plant operating data to a Region I Senior Reactor Analyst (SRA) to determine the significance of this event. The SRA determined that this event had very low risk significance and should be categorized as (Green) in accordance with phase 2 of the shutdown significance determination process (SDP). This conclusion was reached based on consideration of several factors including the: low reactor decay heat load, availability of redundant RCS makeup systems, the availability of the secondary system to remove decay heat, the relatively large amount of time (~ 18 hours based on a 42 gpm leak and a minimum of 47,000 gallons of RCS inventory available prior to reaching mid-loop conditions) available to the operators to identify and isolate the leak, and the ability to isolate the primary containment.

4OA6 Meetings, including Exit

The inspectors presented the inspection results to Mr. G St. Pierre on February 23, 2001, following the conclusion of the period. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials evaluated during the inspection were considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations. The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
(1) NCV 443/2001-02-01	Technical Specification 6.7.1, Appendix A of Regulatory Guide 1.33 - failure to properly develop an adequate operating procedure for pressurization of the standby shutdown cooling train, and failure to properly implement the maintenance procedure for adjusting the ring setting on safety-related relief valves.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

G. StPierre, Station Director
J. Grillo, Assistant Station Director
B. Plummer, Operations Manager
T. Nichols, Technical Support Manager
D. Sherwin, Maintenance Manager
J. Pandolfo, Security Manager
R. Hickok, NRC Coordinator
J. Sobotka, Regulatory Compliance Supervisor

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed: NCV 01-02-01; failure to properly develop an adequate operating procedure for pressurization of the standby shutdown cooling train, and failure to properly implement the maintenance procedure for adjusting the ring setting on safety-related relief valves.

LIST OF ACRONYMS USED

CR	Condition Report
DCR	Design Change Request
DRPI	Digital Rod Position Indication
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
NCV	Non-Cited Violation
PRT	Pressurizer Relief Tank
psig	pounds per square inch gage
PZ	Pressurizer
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RO	Reactor Operator
SRA	Senior Reactor Analyst
SDP	Significance Determination Process
SSC	Structure, System, or Component
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.