U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Number: 96-02

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: February 13, 1996 - April 1, 1996

Inspectors: David J. Mannai, Resident Inspector
Peter J. Habighorst, Resident Inspector, Haddam Neck
Edward B. King, Physical Security Inspector, Safeguards Section, DRS

Accompanied by: Glenn T. Dentel, Reactor Intern

Approved By: Eben L. Conner for 4/29/96
John Rogge, Chief Date
Division of Reactor Projects, Branch 8

Scope: Resident inspection and safety assessment of plant activities including plant operations, maintenance, engineering, and plant support.

Overview: See executive summary.

9605060362 960429
PDR ADOCK 05000443
Q PDR

EXECUTIVE SUMMARY

SEABROOK STATION
NRC INSPECTION REPORT NO. 50-443/96-02

Plant Operations: Operations were performed safely and effectively for the majority of activities throughout the inspection period. Operator response to the inadvertent high energy line break (HELB) actuation was prompt, effective and minimized the impact on plant operation. However, the NRC identified several areas of concern. The inspector identified lack of foreign material exclusion control during performance of turbine driven emergency feedwater surveillance testing. Additionally, initial resolution of reactor vessel level indicating system (RVLIS)/HELB trouble alarms was not rigorous. Communications between operations and system engineering were not thorough and complete. The basis for continued system operability was not well understood and the system was incorrectly considered operable when the numerous system trouble alarms were received. Separately, inspectors found operators had become desensitized toward a safety-related spray additive tank (SAT) level indicator, in the containment building spray system (CBS), that was indicating below the technical specification minimum level. Subsequently, it was learned that SAT level indicator was out of tolerance, which had been a longstanding station problem.

Maintenance: During the period, the inspector identified a chainfall hoist was attached to the containment recirculation sump valve CBS-V-008 enclosure with the plant in mode 1 at 100% of rated thermal power. CBS-V-008 is considered risk significant in the Seabrook Station Probabilistic Risk Assessment. Additionally, maintenance performance of troubleshooting activities on the RVLIS/HELB system resulted in an inadvertent HELB systems actuation, which was considered an unplanned challenge to both plant systems and operators.

Engineering: The operability determination, performed to address the potential to clog emergency core cooling system (ECCS) throttle valves, concluded systems were operable. The determination was technically sound and thorough. Actual verification of sump screen strainer size and condition was considered a strength.

Plant Support: Overall, performance within the various plant support disciplines was considered good. A containment entry to verify ECCS sump screen size was well controlled with a few minor exceptions. The Vehicle Barrier System (VBS) was effectively implemented by the required date. A violation, related to failure to implement its procedure to address the continual behavioral observation program requirements, was cited.

The inspector identified housekeeping conditions in the west steam and feedwater pipe chase and containment hatch area that did not meet the guidance of station procedures and management expectations. The same conditions were identified in a quality assurance surveillance following OR04 and were never fully addressed.

Executive Summary (Continued)

Safety Assessment/Quality Verification: Nuclear Safety Audit Review Committee (NSARC) activities were conducted and documented properly and according to technical specification and station administrative requirements.

TABLE OF CONTENTS

DETAILS

1.0 Summary of Facility Activities

At the start of the report period, a slow reactor power ascension to 90% of rated thermal power was in progress. Reactor power was held stable at 90% while generator stator cooling water conditions improved by chemical flushing of the system. On February 21, 1996, the plant resumed the power ascension and reached 100% of rated thermal power on February 22, 1996 at 10:22 a.m.. On March 5, the reactor vessel level indicating system was declared inoperable (Section 2.3). On March 7, 1996, the High Energy Line Break (HELB) system actuated resulting in letdown isolation, steam generator blowdown isolation, and auxiliary steam isolation. The HELB system actuation was attributed to instrumentation and control work on the train 'B' RVLIS/HELB cabinet (Section 3.2). Operators responded quickly and effectively to minimize the effects of the transient on the plant (Section 2.4). The reactor remained at approximately 100% power through the conclusion of the report 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 Emergency Feedwater Quarterly Flow Surveillance (VIO 50-443/96-02-01)

On February 27, the inspector observed the turbine driven emergency feedwater (EFW) quarterly flow surveillance (procedure OX1436.02 and RTS 96R003054001). The inspectors reviewed the surveillance procedure, the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), attended the preevolution brief and held discussions with operations personnel involved with the surveillance.

The surveillance procedure was performed satisfactorily with TS surveillance acceptance criteria satisfied. The inspector observed good communication and coordination between personnel in the control room, east and west steam and feedwater pipe chases and the EFW pump house. The inspector independently verified system flow, turbine speed, pump suction pressure and other acceptance criteria. Measuring and test equipment (M&TE) used during the test were in current calibration.

The inspector independently identified that temporary jumpers, which are used to facilitate inservice testing (IST) of the check valves in the instrument air supply piping to the EFW steam admission valve FW-V-395, did not have foreign material exclusion (FME) cleanliness covers installed. Consequently, during performance of check valve testing within the surveillance test, debris via the jumper can be introduced to the instrument air system associated with the valve actuator for FW-V-395, the steam admission valve for the EFW turbine. Similar situations existed for valves FW-V-393 and FW-V-394, EFW steam isolation valves, which are located in the east and west steam and feedwater pipe chases. The inspector notified the control room of this observation and FME covers were promptly installed.

The inspector reviewed licensee procedures and requirements for system cleanliness and found that this portion of instrument air system was designated level "C" as such requires FME controls. The inspector observed that in this instance no foreign material was introduced to the instrument air lines for FW-V-395 valve actuator, however, the potential clearly existed. The particular jumper, which were attached by lanyard to the instrument air piping, was permanently stowed on the floor of the EFW pumphouse. Additionally, the lack of FME covers in these jumpers appears to have existed for extended period. The surveillance procedure did not contained instructions regarding use of FME covers. Chapter 2, Station Policies Section 7, Housekeeping/Cleanliness/Foreign Material Exclusion, of the Station Management Manual (SMM) requires that controls for foreign material exclusion from systems and components shall be implemented per Procedure MA 3.4, Foreign Materials Exclusion. Procedure MA 3.4, Section 4.1 General Requirements, requires critical equipment important to safe operation and shutdown of the plant shall receive FME considerations and that openings shall be appropriately covered to prevent entry of dust, dirt, or other foreign objects.

Overall, the inspector determined the EFW surveillance was performed satisfactorily. However, the failure to implement required FME controls created the potential to affect proper operation of all three safety-related steam supply valves to the risk significant turbine driven EFW pump. Additionally, the inspector was concerned that once notified of the lack of FME controls, operations personnel did not initiate an adverse condition report (ACR) to formally document the problem. Consequently, the formal corrective action process was not used to determine the reason for the lack of FME controls, identify similar occurrences and implement appropriate comprehensive corrective actions. The inspector reviewed selected quality assurance audits and surveillances and found similar FME findings had been identified. The failure to implement required FME controls on safety-related equipment is considered a violation. (VIO 50-443/96-02-01)

2.3 RVLIS/HELB System Trouble

On March 3, 1996 at 7:43 a.m., the control room operators received control board alarm F5898, "RVLIS/HELB System Trouble" associated with the reactor vessel level indicating system (RVLIS) and high energy line break (HELB) actuation system. These two systems share circuity and software and are located in the same cabinet in the control room. Local display indication showed that Train A was satisfactory and Train B had a diagnostic error for subcooling margin and a diagnostic message "error in NVRAM." The non-volatile random access memory (NVRAM) circuit card is considered a safety-related portion of the system. The alarm reset and alarmed three minutes later. All HELB diagnostics were satisfactory based on procedure OS1290.01 "Response to HELB system actuation or trouble." The RVLIS is accident monitoring instrumentation and subject to technical specification requirements. Operations personnel contacted the system engineer for the Reactor Vessel Level Indication System at home, who determined that the subcooling margin message indicated that the 'B' train subcooling value, toggling above and below the low alarm setpoint of 40°F, is normal and consistent with plant conditions. Analog critical safety functions were found satisfactory. The licensee considered the equipment operable based on the system engineer's recommendation as long as the system reset clears the "RVLIS/HELB System Trouble" alarm. However, in later discussions with the system engineer, the inspector learned the system is considered inoperable until the alarm is reset satisfactorily. According to the March 3 operating log entry, the problem was not considered high priority.

Control room operators continued to receive "RVLIS/HELB System Trouble" alarms. On Tuesday, March 5, the first attempt to eliminate the repeated trouble alarms was unsuccessful. Instrumentation and Control technicians attempted to "refresh NVRAM" which consists of installing a computer maintenance terminal to the system cabinet and following appropriate vendor procedural guidance for the computerized system. This computer refresh was unsuccessful in stopping the alarms and operations personnel declared RVLIS inoperable and Technical Specification (TS) 3.3.3.6 limiting condition for operation (LCO) action statement was entered. The licensee preliminarily determined that the probable cause of the alarm was a circuit card malfunction. On March 7, trouble shooting confirmed a faulty circuit board and the system was returned to an operable status on March 8 (Sections 2.4 and 3.2).

The inspector held discussions with system engineering and operations personnel, reviewed technical specifications, operating logs, print out of alarms and developed an understanding of a sequence of events regarding resolution of the repeated alarms. The inspector identified that the alarm was received a total of 25 times between the initial alarm on March 3 and when the system was declared inoperable at 12:18 p.m., on March 5. Until the alarm was reset, RVLIS was considered inoperable by the system engineer. There were no corresponding LCO action statements entered on March 3 or March 4 in the control room operating logs. Although the alarms were quickly reset, technical specification LCO action statements should be logged regardless of the action statement entry duration. After declaring the system inoperable on March 5, the day shift operators, after discussions with the system engineer, entered LCO action statement entry and exit times in the operating logs for several previous alarms received that morning. The system was restored to an operable status on March 8, within the required action statement time frame conservatively using March 3, 7:43 a.m.

The inspector concluded the problem was ultimately corrected within the required technical specification action statement requirements. However, initial problem resolution by station personnel was not commensurate with the safety significance of the system. The initial problem resolution focused heavily on corrective actions used in the past. The time frame between initial symptoms and unsuccessful software "refresh" with numerous repeated alarms did not represent aggressive timely resolution of the safety-related accident monitoring instrumentation equipment problem. The communications from the time of initial identification until the system was declared inoperable were not fully effective. This was evidenced by the operators not appropriately entering TS LCO actions statements in the operating log and that the system engineer was not made aware of the subsequent numerous alarms. In addition, through discussions with operators on the morning of March 5, the inspector found the basis for RVLIS operability was not well communicated between operating shifts. Decision making during the initial problem resolution did not reflect station management expectations regarding conservative decision making. Once it was determined the refresh would not correct the problem, the LCO action statement log entry was non-conservatively dated that day vice the initial day of discovery on March 3.

2.4 High Energy Line Break (HELB) Systems Actuation

On March 7, at 9:48 a.m., the HELB systems actuated resulting in letdown, steam generator blowdown, and auxiliary steam systems isolations. The HELB system actuation was attributed to instrumentation and control (I&C) work on the train 'B' RVLIS/HELB cabinet (Section 3.2). The inspector observed prompt and effective operator response to the HELB actuation. Charging flow was reduced to minimize the pressurizer level increase with the letdown path isolated. Operators initially entered procedure OS1202.01, "Loss of Letdown," then transitioned to procedure OS1290.01, "Response to HELB systems actuation or malfunction." Operators did not initially enter OS1290.01 since the HELB alarm was not received. The operators communicated with I&C personnel to ensure stable conditions before reestablishing letdown flow. The operations personnel reestablished steam generator blowdown according to procedure OS1227.01, "Recovery from steam generator blowdown system isolation." The inspector noted good control room communication and coordination in response to the transient. Off-normal procedures were correctly followed. The inspector had no further questions.

2.5 Spray Additive Tank Level Indicator Problem

During a main control room tour, the inspector identified that level indicator (CBS-LI-2331) for the Spray Additive Tank (SAT), which is part of the

Containment Building Spray System (CBS), was reading approximately 500 gallons low and below the Technical Specification (TS) minimum value. The inspector observed the other channel (CBS-LI-2333) level and the main plant computer point (CBS-L-2338) were within the technical specification allowable range. Control room personnel were questioned concerning TS required channel check of the SAT level. The operators perform a channel check once per shift using a main plant computer point to satisfy the technical specification surveillance criteria. Control room personnel indicated that the instrument in question was not density compensated and are subject to temperature variations. The inspector questioned operator personnel again on two additional separate occasions when the level was still indicating below TS allowable range.

Following the additional questioning of the level by the inspector, a work request (96W000206) was initiated to perform the calibration for level transmitter CBS-LT-2331. Using procedure IS1622.210, the transmitter was found out of tolerance and was subsequently calibrated to within tolerance. The level transmitter was placed in service and a 200 gallon difference was noted between the just calibrated CBS-LT-2331 and the other channel (CBS-LT-2333) level. The licensee performed a scope change to check the other channel and determined the indicator was incorrectly reading 200 gallons high. Subsequently, the indicator (CBS-LI-2333) was replaced.

The inspector reviewed the work request, calibration procedures, TS requirements and questioned maintenance personnel on the work. The licensee identified that the computer point is calibrated via a repetitive task sheet (RTS) and not through a formal procedure. The licensee also identified that the tolerance permitted by channels CBS-LT-2331 and CBS-LT-2333 calibrations (+/- two percent) are greater than the TS range (two percent total range). Through review of previous calibrations and the nature of repetitive channels out of tolerance, the inspector determined that the drift in the channel is a long standing problem. Thus from discussions with I&C personnel, the cause of the error in level was not due to temperature variations.

The licensee is creating a formal procedure for the calibration of the computer point (CBS-L-2338). The instrumentation and controls department is performing a review of all high priority items associated with a TS commitment to determine if procedures are written to meet the requirements. In addition, a request for engineering services (RES) was written to review the tolerance limits for channels CBS-LT-2331 and CBS-LT-2333 and for the limited TS range. The inspector found the out of tolerance instrument had no actual safety significance. The indicator is used for indication only and the main plant computer point remained operable during the time period in question. However, if the main plant computer was unavailable, then the remaining channels would be in disagreement. The inspector found operators became overly reliant on the main plant computer point and did not critically question the indication. Additionally, it was apparent, through discussions, that the drift occurred repeatedly and the operators had come to accept the result as normal, attributing the drift incorrectly to temperature variations.

3.0 MAINTENANCE (61726,62703,92902)

3.1 Chainfall Hoist Connected to Safety-related Equipment (URI 50-443/96-02-02)

On February 26, 1996, while touring the -26'0 elevation mechanical penetration area in the primary auxiliary building, the inspector identified that a chainfall hoist was attached to the recirculation sump suction valve CBS-V-008 enclosure. Maintenance was performed on the valve during refueling outage OR04. The inspector promptly notified control room personnel, specifically the shift manager. The inspector voiced concern regarding the potential impact on the operability of the valve during a design basis accident (DBA) or safe shutdown earthquake (SSE). The chainfall was promptly removed. The licensee documented the problem on adverse condition report ACR-96-154. The ACR resolution was not complete at the end of the inspection period. The inspector reviewed the Engineering Evaluation 95-04, which contained the 1994 Risk Management Summary Report. The report provides a perspective of current risk assessment results that cover all modes of operation at Seabrook Station.

The safety-related CBS sump valves are important from a plant specific probabilistic risk assessment (PRA) perspective which is evidenced by the CBS sump valves inclusion in Risk Management Summary Report Table ES-1, Risk Significant Systems Ranking, which list systems considered to be risk significant. Additionally, in Table 3-5, System Importance Ranking, the CBS sump valves are given an at-power core damage frequency system importance ranking of 2. The breakdown in work controls which allowed the chainfall hoist to remain attached, the missed opportunity for station personnel to have identified this earlier and the potential impact on system operability for a risk significant system represented areas of NRC concern. This item will remain unresolved pending final NRC review of the licensee's completed ACR evaluation. (URI 50-443/96-02-02)

3.2 RVLIS Troubleshooting Activities

On March 5, 1996, attempts to correct the repeated RVLIS/HELB system trouble alarms were unsuccessful and control room operators declared the system inoperable and Technical Specification (TS) 3.3.3.6 limiting condition for operation action statement was entered, as previously discussed in section 2.3 of this report. The licensee preliminarily determined that the probable cause of the alarm was a faulty circuit card. The inspector attended the prebrief meeting, reviewed the work package, and observed the card replacement. To facilitate circuit card replacement, the system was deenergized. Following replacement and reenergization of the train B RVLIS/HELB system, letdown, steam generator blowdown, and auxiliary steam systems isolated due to an inadvertent HELB system actuation. Operator response to this evolution was discussed in Section 2.4. ACR 96-153 was initiated to document the system actuation. The suspected cause of the inadvertent HELB actuation was the replacement card actuation setpoint was not the correct value. The circuit card can not be bench tested and must be set up installed. The card replacement was performed later in the day using the same work request with a scope change. The scope change had the HELB leads lifted prior to powering the system to prevent actuation. The inspector reviewed the scope change package and determined that the licensee took the proper precautions to prevent another HELB actuation.

The inspector concluded the activity was performed adequately, with the exception of the inadvertent HELB system actuation. Although prompt operator response minimized the impact on plant operation, the actuation resulted in an unplanned challenge to plant systems and operators. Troubleshooting activities, with appropriate immediate corrective actions to prevent another system actuation, were resumed and completed successfully. The inspector learned the card was previously changed in this manner without causing a system actuation. Another undesired consequence of the actuation was the Main Plant Computer System locked out the sub-routine for RVLIS/HELB system, which presented an additional challenge to the operators since the HELB actuation alarm was not received. This issue will also be addressed in the completed ACR evaluation.

3.3 Emergency Feedwater System Corrective Maintenance

On February 27, 1996 the inspectors observed portions of the replacement of the pressure regulator (WR 96W000085) that supplies air to valve 1-MS-F-395A. 1-MS-FY-395A supplies air to actuator of valve MS-V-395, the steam admission valve to the turbine driven emergency feedwater pump. The pressure regulator was replaced due to diaphragm air leakage. The inspector found the authorized safety-related maintenance activity was performed according to the work plan, with technicians properly documenting action taken. The inspector had no further questions.

3.4 Diesel Generator 1B Monthly Operability Surveillance

On February 21, 1996, the inspector observed a nuclear system operator perform portions of OX1426.05. The nuclear system operator (NSO) was knowledgeable of the procedural guidance actions. Appropriate communications with the control room were maintained. The diesel was started on a simulated SI signal. The surveillance was completed at 10:17 p.m. on February 21. Based upon a review of procedural steps, the inspector noted that some of the steps lacked detail. For example, no guidance is stated about the designated method to start the engine (either manual or SI actuation test signal). Notwithstanding, both control room operators and the NSO were aware of the procedural step basis. Additionally, no guidance is stated about the need to bar the diesel over.

In preparation for the surveillance observation, the inspector verified a sampling of various commitments to the procedure from either the Updated Final Safety Analysis Report (UFSAR), past NRC inspection reports (IRs), or past NRC Information Notices (INs). The past commitments were adequately incorporated into surveillance procedure OX1426.05. Specifically, the inspector verified UFSAR section 9.5.8.4 that documented an action to visually inspect the flanged joints of the intake air and exhaust system during EDG operation. This commitment was documented on procedural step 8.1.29. Additionally, the inspector verified the commitment to NRC IR 50-443/93-80 to verify that the dirty fuel oil reservoir is empty during each surveillance (Step 8.1.46), and action in regards to NRC IN 84-69 and IN 85-28 to verify the correct position of the voltage regulator after completion of the surveillance. Following the close of the inspection period, the licensee identified some examples where procedure enhancements could be made to better reflect UFSAR wording. Overall, the inspector found the surveillance was performed correctly and had no further questions.

4.0 ENGINEERING (71707,37551,92903,40500)

4.1 Emergency Core Cooling System

On February 20, the potential inoperability of the emergency core cooling (ECCS) systems at Seabrook Station was raised when Northeast Utilities Millstone Unit 2 plant declared the high pressure safety injection (HPSI) system inoperable due to the potential to clog HPSI throttle valves with debris passing through the sump screens during the containment sump recirculation phase. The valve internal clearances in the throttled position are potentially smaller than the particles that can pass through the containment recirculation sump screen. Seabrook Station initiated adverse condition report ACR 96-109 and performed a preliminary operability determinations which was completed on February 21. The licensee's final operability determination was completed on March 22. The inspector reviewed the preliminary and final operability determinations. Additionally, the inspector accompanied the licensee inside containment to verify sump screen size. The Updated Final Safety Analysis Report (UFSAR) and procedure OE 4.5, Operability Determination, were reviewed.

During the recirculation phase of a loss of coolant accident (LOCA), the ECCS is aligned to take suction from the recirculation sumps via the residual heat removal system (RHR). The RHR pumps' discharge supplies the charging system (CS) (high head safety injection) and safety injection (SI) (intermediate head safety injection) pumps. The SI pumps inject into all four RCS cold legs (cold leg recirculation) or hot legs (hot leg recirculation). The charging pumps inject into all four cold legs. Each of these injection lines has a manual throttle valve that is used to balance flow and prevent pump runout. RHR system flow does not flow through the throttle valves and therefore not subject to flow blockage. For a large break LOCA, adequate core cooling can be maintained with no flow through charging or safety injection systems. For a small break LOCA, charging and safety injection systems provide flow to the RCS that exceed required makeup for decay heat considerations. Once the reactor coolant system (RCS) has been depressurized below the residual heat removal (RHR) pump shutoff head, the RHR system provides the majority of the flow.

The licensee's preliminary and final operability determinations found the ECCS systems remained operable and thus able to fulfil the intended safety function. The determination concluded that the potential for clogging did exist. The sump screen size consists of 8x8 openings per inch with 0.028 wire size yielding and individual opening size of 0.097 inches. The throttle valves, in their flow balanced position, result in internal clearances that are smaller than the particles that can pass through the sump screen (0.060 inches for the CS valves and 0.044 inches for the SI valves) and consequently are potentially susceptible to clogging. However, the licensee concluded with the small, low density, low mass particle size, low sump approach velocity (0.2 ft/sec), and flow path through the charging and safety injection pump (11 stage centrifugal pumps will likely cause particle fragmentation) minimize the potential for clogging of the throttle valves. Any material of sufficient size to become lodged within the flow passage of the throttle valve would then become subjected to a high differential pressure (800 psid) across the throttle valve. The high differential pressure across the throttle valves result in a high likelihood that any material lodged would be forced through the valve. Also, the available flow far exceeds that required to fulfill the safety function. The licensee verified the strainer size was per design and in overall good condition.

The inspector agreed with the licensee's operability determination and concluded the initial and final determinations met the requirements of OE 4.5. The initial determination was prompt and comprehensive. The final operability determination was thorough and well supported. The ECCS sump strainer screen size was independently verified by the inspector and reflected the description contained in the UFSAR. The prompt response to the issue demonstrated a good safety perspective. The inspector had no further questions.

5.0 PLANT SUPPORT (71707,71750)

5.1 Radiological Controls

The inspector observed implementation of radiological controls during tours in the radiologically controlled area (RCA). Random sampling of portable hand held friskers and portal monitors demonstrated that they were calibrated as required by station procedures. The inspector determined by observation of several tasks in the radiologically controlled area that the licensee was effectively implementing radiological controls to minimize the spread of contamination and incorporating as-low-as-is-reasonably-achievable principles.

5.1.1 Containment Entry

The inspector determined Health Physics coverage of the containment entry for ECCS sump screen size verification activities was well controlled and focused on minimizing radiological exposure and ensuring personnel safety. The requirements for the containment pre-entry brief, contained in section 5.14 of procedure ON1090.04, Containment Entry, were properly discussed. The inspector observed sound radiological practices throughout the containment entry activities.

One minor exception to otherwise well controlled activity, occurred during actual entry into containment. Specifically, when the airlock outer door was opened it was discovered that containment lighting circuits had not been energized as required by procedure ON1090.04 prerequisites. The control room was promptly notified. While waiting for the lighting circuits to be energized, the outer airlock door remained open. The inspector questioned the health physics technician operating the door and found the individual was unaware of the control room alarm that actuates when the door is open for ten minutes. A few minutes later control room operators called the technician to inquire why the alarm was received. Technical Specification 3.6.1.3 for the primary containment airlock requires both the inner and outer doors be closed, except when the air lock is being used for normal transit entry and exit through the containment, then at least one door shall be closed. The alarm serves to alert operators that the containment door is open. When the containment lights were turned on, the containment entry resumed.

The inspector determined this part of the containment entry could have been coordinated better between control room personnel and personnel entering containment since all prerequisites were not completed when the entry was commenced. There was minimal impact since the result was an insignificant time delay. The lack of knowledge regarding the airlock door alarm suggested the need for improved procedural guidance and training for operation of the personnel hatch including better awareness and sensitivity to containment airlock and containment integrity technical specification requirements. The inspector considered these minor weaknesses and had no further questions.

5.2 Security

The inspector observed security force performance during the course of routine inspection activities. Protected area access controls were noted to have been properly implemented during random observations. Individuals with visitor badges were noted to have been properly in the control of designated escorts. Additionally, alarm station officers were observed to be attentive to alarm and surveillance stations and aware of the status of security systems.

5.2.1 Vehicle Barrier System Installation

During the period, regulation 10 CFR 73.55 required a Vehicle Barrier System (VBS), which is a part of vehicle control measures, be installed and operational by February 29, 1996. On April 1, the inspector held discussions with security personnel concerning the VBS and performed a walkdown of portions of the system.

The inspector confirmed the VBS was implemented and made operational by the required date. Personnel interviewed were knowledgeable of the VBS, associated design requirements and operational aspects of the VBS. Appropriate licensee procedures were changed to reflect installation of the new system. A more detailed inspection will take place in the future by NRC Region I specialist inspectors using Temporary Instruction 2515/132, Malevolent Use Of Vehicles At Nuclear Power Plants. The inspector had no further questions.

5.2.2 Security Access Authorization (VIO 50-443/96-02-03)

During an NRC integrated performance assessment inspection conducted on February 5-16, 1996, the team reviewed the 1995 combined audit of the security, access authorization, and fitness-for-duty programs, No. 95-A03-01, conducted March 20-31, 1995. The audit identified one finding and seven recommendations. The audit finding addressed a weakness in the licensee Continuous Behavior Observation Program relative to infrequent visitor access into the protected area.

Specifically, the licensee's NRC-approved Physical Security Plan (the Plan), Revision 19, dated April 26, 1995, Section 3.1, states, in part, that all elements of NRC Regulatory Guide 5.66 have been implemented to satisfy the requirements of 10 CFR 73.56. One of the requirements of 10 CFR 73.56, as stated in Section (b)(2)(iii), is that the unescorted access authorization program must include behavioral observation, conducted by supervisors and management personnel, designed to detect individual behavioral changes which, if left unattended, could lead to acts detrimental to the public health and safety. Additionally, one of the elements of Regulatory Guide 5.66, as noted in Section 3, under the "Clarification to the Guidelines," is that prior to the reinstatement of an employee's access authorization, it is reasonable to expect that the licensee will ascertain that the activities the employee was engaged in during his or her absence would not have the potential to affect the employee's trustworthiness and reliability.

To satisfy the licensee commitments as described in the Plan, the licensee's continual behavioral observation program requires, as documented in the Seabrook Station Security Program (SSSP), Revision 16, Section 3.9, titled "Reinstatement of Unescorted Access Authorization," that if more than 30 days have lapsed since an individual was at Seabrook Station, conduct an interview with the individual to ascertain that the activities of the individual during his or her absence would not affect his or her trustworthiness and reliability. However, the licensee failed to implement its procedure to address the continual behavioral observation program requirements. The failure to implement required behavioral observation program is considered a violation. (VIO 50-443/96-02-03)

5.3 Housekeeping Tour

On March 19, 1996 the inspector toured the west main steam and feedwater pipe chase, which contains safety-related portions of the main steam, main feed system and emergency feedwater (EFW) systems and in the vicinity of the containment personnel hatch. Major components within the pipe chase area include the main steam isolation valves, atmospheric steam dump valve, main steam safety valves, emergency feedwater system steam isolation valves. The inspector reviewed station procedures MA 3.3, Housekeeping and MA 4.8, Control of Temporary Equipment.

The inspector found debris, bags of piping insulation, combustible material, temporary drain hoses, temporary lighting and trash on various levels within the west pipe chase area. The temporary lighting, which was laying on the floor contained a broken bulb with exposed filaments, represented a potential fire hazard. The inspector found ladders improperly stored in the vicinity of the containment personnel hatch. Also found near the hatch area were nitrogen gas bottles temporarily secured. The pre-staged nitrogen bottles used for recharging MSIV accumulators were stored according to temporary storage requirements, however the inspector questioned the anticipated duration of temporary storage.

The inspector informed shift management of the housekeeping conditions in the pipe west chase and the containment hatch areas. Through discussions the inspector learned the conditions likely existed since completion of the refueling outage. On March 26, Quality Assurance (QA) initiated an adverse condition report (ACR 96-205) on the housekeeping conditions in the west pipe chase. The conditions had been initially identified by QA at the completion of the refueling outage OR04 in December 1995 and documented in Quality Assurance Inspection Report (QAIR) 95-0671. QA noted in the ACR that followup walkdowns found some of the items were previously corrected, however, unsatisfactory conditions remained. Ladders were properly stored according to station requirements. The nitrogen bottles are expected to remain in place and a request for engineering service (RES) was initiated to establish a permanent bottle storage location.

The inspector considered general housekeeping conditions in the aforementioned areas did not meet the requirements of MA 3.3 or management expectations. However, there was no actual adverse impact to equipment or plant operation and overall was considered a minor performance weakness. The temporary equipment control requirements of procedure MA 4.8 for ladders were not followed in some cases, also with no impact on safe plant operation. The inspector found the conditions, which existed for sometime, represented examples where management expectations were not met nor were they enforced proactively in a timely manner. Additionally, the line organization did not respond adequately to the previous QA identification of housekeeping conditions. The licensee review of housekeeping responsibilities and accountability was initiated. The inspector had no further questions.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (92700)

6.1 Nuclear Safety Audit Review Committee Meeting

The inspector directly observed portions of the Nuclear Safety Audit Review Committee (NSARC) meeting (96-02) convened on March 13, 1996. The North Atlantic Management Manual procedure NM 11250, Nuclear Safety Audit Review Committee Operation, and Technical Specification (TS) Section 6.4.3 and the meeting minutes were reviewed. The inspector verified a quorum of NSARC members was present. The NSARC activities were conducted and documented properly and according to TS and station administrative requirements. The inspector had no further questions.

6.2 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.

6.2.1 LER 96-001-00

LER 96-001-00, "Automatic Reactor Trip," dated February 23, 1996, documented the automatic reactor trip due to high pressurizer pressure when a random failure of an electro-hydraulic control (EHC) speed control circuit card resulted in closure of the turbine control and combined intercept valves. This event was previously discussed in NRC inspection report 50-443/96-01. No new issues were revealed in the LER.

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 April 23, 1996, summarizing the preliminary findings of this inspection. No proprietary information was identified as being included in the report.

7.2 Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Updated Final Safety Analysis Report (UFSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the UFSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspectors verified that the UFSAR wording was consistent with the observed plant practices, procedures and/or parameters.

7.3 NRC Activities

During the week of February 12-16, 1996, the second week of a two consecutive week, onsite Independent Performance Assessment Process (IPAP) Team inspection was performed. Results of the inspection was documented in NRC Inspection Report No. 50-443/96-80.