U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-443
License No.:	NPF-86
Report No.:	50-443/99-02
Licensee:	North Atlantic Energy Service Corporation
Facility:	Seabrook Generating Station, Unit 1
Location:	Post Office Box 300 Seabrook, New Hampshire 03874
Dates:	March 21 - May 9, 1999
Inspectors:	Ray K. Lorson, Senior Resident Inspector Javier Brand, Resident Inspector Antone C. Cerne, Senior Resident Inspector, Millstone Joseph Carrasco, Reactor Engineer Harold Gray, Senior Reactor Engineer Tom Moslak, Radiation Specialist
Approved by	Clifford Anderson, Chief Projects Branch 5 Division of Reactor Projects

EXECUTIVE SUMMARY

Seabrook Generating Station, Unit 1
NRC Inspection Report 50-443/99-02

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers an eight week period of resident and specialist inspection.

Operations:

- Plant materials conditions were acceptable. The inspector noted a poor radiological work practice inside the containment. In addition, the licensee did not identify a minor valve packing deficiency during the pre-outage walkdown. (Section O2)

- The plant power reduction and cooldown, and the turbine volumetric testing were performed well. (Sections O4.1, and O4.3)

- The operators performed the emergency diesel generator testing generally well; however, the licensee identified a few test and configuration control deficiencies during the "A" train testing. These issues were properly entered into the licensee's corrective action program. (Section O4.2)

Maintenance:

- The licensee failed to establish adequate controls in June 1997 to ensure that the K-85 relay met the required calibration criteria prior to installation. This is a non-cited violation **(NCV 99-02-01)**. The event team review, and corrective actions for the relay failures during testing were adequate. The risk associated with this event appeared minimal since the operators could have taken manual actions to compensate for the relay failures. (Section M1.1)

- The installation of a freeze seal to support replacement of the spent fuel pump discharge valve was performed well. The inspector identified an industrial safety hazard in that personnel involved with the work activities failed to recognize that a local oxygen monitor indicated a low oxygen condition. The licensee implemented appropriate corrective actions for this finding. (Section M1.2)

- Excellent performance was observed during removal of the reactor vessel core barrel. The planning, and execution of this activity allowed the move to be completed without any personnel exposure. (Section M1.3)

- The licensee responded well to evaluate the extent of damage, and to repair a minor leak from the primary component cooling heat exchanger upper head assembly. (Section M1.4)

Executive Summary (cont'd)

- The "B" train 4.16kv bus outage, and the "B" EDG outage and cylinder/liner replacement, were well controlled, governed by adequate procedural guidance, and documented in a way that appeared to provide retrievable quality information. The replacement of the #11 cylinder assembly on the "B" EDG appeared to be a prudent action by the licensee. The installation was partially observed by the inspector and found to be adequately handled and documented. (Section M1.5)

- Inservice inspection (ISI) activities including examination of the piping welds, reactor vessel, steam generator tubes, and completion of the first 10 year ISI interval were well planned and implemented by qualified personnel in accordance with approved procedures. Observation of nondestructive testing activities in progress showed that the ISI work was conducted with proper oversight by the Seabrook staff and the results were well documented. The inspections observed were thorough and of sufficient extent to determine the integrity of the components inspected. Indications, when identified, were evaluated and addressed in accordance with the ASME Code and regulatory requirements. No issues of concern were identified. (Sections M8.1, and M8.2)

- The maintenance oversight group identified multiple procedural and documentation deficiencies associated with the installation of electrical splices on safety-related solenoid valves by construction services electricians. The licensee's planned and completed corrective actions for this finding appeared adequate. (Section M8.3)

Engineering:

- The licensee properly identified and evaluated two potential plant issues during the cooldown. (Section E2.1)

- The licensee responded well to fuel assembly upper nozzle bolt integrity issues. The inspector noted that the licensee did not identify the gap problem during the initial RFO6 inspections. (Section E2.2)

Plant Support:

- ALARA program requirements were well developed, integrated in the work control process and effectively implemented with respect to reactor disassembly and steam generator inspection/cleaning activities. The final cumulative personnel outage exposure was below the licensee's projected estimate, indicating that the ALARA measures were effectively implemented. (Section R1.1)

- Radiological controls were effectively implemented as evidenced by a qualified staff properly implementing procedures to minimize external and internal exposure, by developing detailed RWPs, appropriately monitoring personnel exposure, and adequately maintaining radiologically controlled areas. (Section R1.2)

- The Nuclear Oversight Group and Health Physics management effectively monitored radiation protection program implementation, worker practices, and procedural compliance through close and frequent observations. Prompt actions were taken to evaluate and correct factors that could degrade performance. (Section R7)

- The inspector reviewed an licensee event report involving an individual who was granted temporary unescorted access based on incomplete pre-employment documentation. The licensee subsequently revoked the individual's access, and concluded that the individual did not adversely affect any vital plant equipment. The inspector discussed this event with the Regional Security Specialist, and determined that the licensee's access control program was consistent with NRC requirements. (Section P8.1)

TABLE OF CONTENTS

Table of Contents (cont'd)

ATTACHMENTS

Report Details

Summary of Plant Status:

The facility began the period at approximately 100% of rated thermal power. On March 27, the operators shutdown the plant to begin refueling outage six (RF06). The outage was on-going at the end of the period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, routine operations were performed in accordance with station procedures and plant evolutions were completed in a deliberate manner with clear communications and effective oversight by shift supervision. Control room logs accurately reflected plant activities and observed shift turnovers were comprehensive and thoroughly addressed questions posed by the oncoming crew. Control room operators displayed good questioning perspectives prior to releasing work activities for field implementation. The inspectors found that operators were knowledgeable of plant and system status.

O2 Operational Status of Facilities and Equipment

a Inspection Scope (71707, 62707):

The inspectors routinely conducted independent plant tours and walkdowns of selected portions of safety-related systems during the inspection report period. These activities consisted of the verification that system configurations, power supplies, process parameters, support system availability, and current system operational status were consistent with Technical Specification (TS) requirements and UFSAR descriptions. The inspectors reviewed material conditions, housekeeping, and general work practices inside the containment during RF06, and following completion of the major outage maintenance activities.

b. Observations and Findings:

Material conditions inside the containment were acceptable, however, several minor material deficiencies were observed and identified to the licensee for correction. The deficiencies included items such as minor packing leaks on three safety injection (SI) system valves. The system engineer (SE) promptly evaluated these leaks and initiated an adverse condition report (ACR) to review the need for additional corrective action. This response was appropriate, however, the inspectors noted that the licensee had not identified these leaks during their pre-outage containment walkdown.

The inspector noted, during a containment tour, two nuclear system operators (NSOs) who did not appear to be actively involved in a work activity. The inspector questioned the individuals and learned that they had been waiting inside the containment for approximately 45 minutes to begin a test procedure. The inspector noted that the containment was posted as a high radiation area, and as a contaminated area, and questioned whether this practice was acceptable. The licensee indicated that the individuals were safe from a radiological exposure standpoint since they were waiting in a "low dose" area. The licensee also indicated that the NSOs received minimal dose during this activity. The inspector reviewed the licensee's radiological procedures, and the applicable radiation work permit for this activity, and concluded that the individuals demonstrated poor radiological work practices; however, they did not violate any radiological procedures, nor did their performance result in an adverse radiological consequences.

Conclusions:

Plant materials conditions were acceptable. The inspector noted a poor radiological work practice inside the containment. In addition, the licensee did not identify a minor valve packing deficiency during the pre-outage walkdown.

O4 Operator Knowledge and Performance

04.1 Plant Power Reduction, and Cooldown:

The inspectors observed selected portions of the plant power reduction on March 26, and the plant cooldown on March 27. The operators performed these activities safely and in accordance with the applicable operating procedures. The shift manager (SM), and unit shift supervisor (USS) maintained good command, and control of these evolutions. Crew briefings were conducted at appropriate times, and good control room communications were observed. The inspectors noted proper management and quality assurance oversight of these evolutions.

04.2 Emergency Diesel Generator (EDG) Surveillance Testing

a. Inspection Scope:

On March 29 and 30, while the plant was in cold shutdown (Mode 5), the inspector observed the operators perform the "B" EDG 18 month surveillance test in accordance with test procedure EX1804.015. On April 1, the inspector observed the operators perform the testing on the "A" EDG. The test was designed to verify that the EDGs would start and load onto their associated safety bus in response to a simulated safety injection signal (SI), a simulated loss of power (LOP) to the safety bus, and a concurrent SI/LOP condition.

b Observations and Findings:

The operators performed this complex procedure well. The pre-test briefing and control room activities were performed well. The operators maintained good control of key parameters during the test and enforced three-way communications with field support personnel. The licensee identified a few configuration and test control deficiencies during the "A" EDG test, and initiated an ACR to review each issue.

c. Conclusions:

The operators performed the emergency diesel generator testing generally well; however, the licensee identified a few test and configuration control deficiencies during the "A" train testing. These issues were properly entered into the licensee's corrective action program.

04.3 Turbine Volumetric Test

a. Inspection Scope

On March 25, the inspectors observed the operators perform turbine volumetric testing. The inspector attended the pre-test briefings, reviewed the applicable procedure, and observed portions of the test activities.

Observations and Findings

The test involved fully opening the turbine inlet valves at near rated power (100% power), while measuring key primary and secondary parameters. Opening the turbine inlet valves increased the steam flow to the main turbine which increased the rate of heat removal from the reactor. The operators performed periodic boron additions to maintain reactor power constant during the test.

The pre-test briefing and control room operations were performed well, and all test objectives were met. The inspectors observed the proper use of three-way communications, peer checking, and management oversight. There were no deficiencies identified during this testing.

b. Conclusions

Operators performed the turbine volumetric testing well. The inspector did not identify any issues during this observation.

O8 Miscellaneous Operations Issues

O8.1 Event Reports

The licensee appropriately made several non-emergency event reports during the period per 10 CFR 50.72 to report several issues involving the failure of two emergency power sequencer relays during testing (Section M1.1), the discovery of dead seals in the plant intake structure, and the granting of temporary unescorted access based on incomplete pre-employment screening documentation (Section P8.1). The inspector reviewed these reports and determined that they adequately described each event, and met the NRC reporting requirements.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Emergency Power Sequencer Relay Failures

a. Inspection Scope

The inspector reviewed the failure of a Westinghouse Model AR relay (K85) to operate properly during an eighteen month emergency diesel generator (EDG) surveillance test on March 29, 1999. The K85 relay failure prevented the unit auxiliary transformer (UAT) supply breaker to the Train "B" emergency bus from opening automatically in response to a simulated loss-of-power condition. This prevented the "B" EDG output breaker from closing onto its emergency bus during the test.

The inspector also reviewed a second AR relay (K77) failure which occurred during a subsequent EDG surveillance test on March 30, 1999. The K-77 relay failure prevented the "B" emergency power sequencer (EPS) from automatically starting the "B" containment building spray (CBS) pump as required.

b. Observations and Findings:

The licensee determined that the relay failure most likely occurred prior to the surveillance test, and therefore concluded that the "B" EDG was inoperable for an indeterminate period of time during the last operating cycle. In addition, the licensee identified that there may have been periods of time during the last operating cycle when the "A" EDG was inoperable for maintenance. Based on this information, the licensee concluded that both EDGs may have been simultaneously inoperable, and properly reported this condition to the NRC on March 31, 1999, per 10 CFR 50.72 (b) (2) (iii).

The K-85 relay failure would have prevented the EDG output breaker from automatically closing onto the emergency bus during a loss of power event, but it would not have prevented the EDG from starting. The inspector reviewed emergency operating procedures, E-O, and ECA-0.0, "Loss of All AC Power", and noted that the procedural guidance would have directed the operator to take manual action to power the emergency bus from the EDG, and also to start the containment building spray pump, if necessary. The inspector then reviewed the EDG output breaker schematic diagram, and confirmed that the relay failure would not have prevented the operator from successfully loading the EDG onto the emergency bus.

The licensee reviewed this event, and concluded that it presented minimal risk, based, in part, on the operators' ability to manually power the emergency bus with the EDG. An NRC Region I reactor risk specialist reviewed the licensee's analysis, and did not identify any basis for disagreeing with the licensee's conclusion.

The licensee formed an event team to identify the root cause(s), and corrective actions for the relay failures. The event team concluded that the K85 relay failure occurred due to inadequate contact pressure as a result of improper relay calibration, combined with the build-up of a resistive impurity on the relay contact surfaces. The failed relay was initially factory calibrated, and provided to the licensee through a qualified Appendix B supplier during the initial plant construction. The vendor apparently did not verify the relay calibration settings prior to supplying the relays to the site, nor did the licensee verify that the relay calibration setting was correct prior to installation of the relay into the "B" emergency power sequencer cabinet during the June 1997 refueling outage. The inspector noted that the manufacturer's instructions recommended that the relay calibration settings be checked prior to installation.

Appendix B, Criterion XV, requires, in part, that measures be established to prevent the use of non-conforming components. Contrary to the above, in June 1997, the licensee did not establish adequate measures to ensure that the Westinghouse Model AR relays met all required calibration settings prior to installation. This contributed to the failure of the K85, and K77 relays during testing. The relay failures resulted in the "B" EDG, and "B" containment building spray pumps being in an inoperable condition for greater than their allowed outage times. This is a violation of 10 CFR 50 Appendix B. The licensee initiated ACR 99-1160 to review this issue. This low risk, non-repetitive, non-willful violation is being treated as a non-cited violation (**NCV 99-02-01**) consistent with Appendix C of the NRC Enforcement Policy.

The inspector reviewed the corrective actions for this event, which included, installation of brand new relays into the emergency power sequencer cabinets, verification of the relay calibration settings prior to installation, bench, and field testing of the new relays, and enhancement of the relay preventive maintenance procedure. Additionally, the licensee replaced the relay housing covers. The failure mechanism attributed to the buildup of impurities on the relay contact surfaces was due to breakdown of the housing cover gasket. The inspector concluded that the corrective actions were adequate to address the identified relay deficiencies.

The licensee submitted LER 99-001-00 to describe this event and planned corrective actions. The inspector reviewed the LER, and concluded that it adequately described the event. This LER is closed.

c. <u>Conclusions:</u>

The licensee failed to establish adequate controls in June 1997 to ensure that the K-85 relay met the required calibration criteria prior to installation. This is a non-cited violation **(NCV 99-02-01)**. The event team review, and corrective actions for the relay failures during testing were adequate. The risk associated with this event appeared minimal since the operators could have taken manual actions to compensate for the relay failures.

M1.2 Freeze Seal For Replacement of Spent Fuel Discharge Valve (SFPV-2)

a. Inspection Scope

On March 22, the inspector reviewed the implementation of a freeze seal that was installed to support replacement of the spent fuel pool discharge check valve (SFP-V2). The inspector performed a field walkdown of the freeze seal, reviewed the work package and applicable procedures, interviewed the work supervisor and observed portions of the work activities.

b. Observations and Findings

The freeze seal was performed using nitrogen as the cooling medium. The mechanical supervisor conducted an excellent pre-job briefing. The work package was thorough, and included an on-line maintenance assessment, required precautions, and system lineup contingencies to prevent or mitigate the consequences of a freeze seal failure.

The inspector observed proper oversight by health physics (HP) technicians, management, and the quality assurance group. The area was properly controlled as potentially contaminated. A quality assurance inspector performed a liquid penetrant test on the affected pipe section before the freeze seal was installed, and after it was removed to ensure that the freeze seal did not adversely affect the pipe integrity. The licensee replaced the freeze seal jacket after identifying a nitrogen leak. The freeze seal installation and valve replacement were completed satisfactorily.

The inspector identified an industrial safety hazard prior to the replacement of the leaking freeze seal jacket. Specifically, workers in the area failed to recognize that the local oxygen monitor had alarmed on a low oxygen condition. The inspector immediately informed a technician who promptly isolated the nitrogen supply to the freeze seal, and all personnel evacuated the area. The mechanical supervisor determined that the technician responsible for watching the oxygen monitor had become distracted by the jacket repair activities. The licensee counseled the mechanic, added another mechanic to assist with the job, and initiated an ACR to document and trend this issue.

c. Conclusions

The installation of a freeze seal to support replacement of the spent fuel pump discharge valve was performed well. The inspector identified an industrial safety hazard in that personnel involved with the work activities failed to recognize that the local oxygen monitor indicated a low oxygen condition. The licensee implemented appropriate corrective actions for this finding.

M1.3 Reactor Vessel Core Barrel Move

a. Inspection Scope

On April 13, the inspectors observed the removal of the reactor vessel core barrel (also known as the lower internals) to support the scheduled ten year reactor vessel weld inspection. The inspectors also observed the return of the core barrel back into the reactor vessel on April 20. The inspectors attended the pre-job briefings, reviewed the applicable procedures and the safety evaluation, met with plant personnel prior to the evolutions, and observed portions of the moves.

b. Observations and Findings

The maintenance supervisor and the project manager performed an excellent pre-evolution briefing, and controlled the move well. All personnel demonstrated a questioning attitude, and were observed to be attentive during the move. The licensee's preparations for this activity were extensive and included: a practice move prior to the initial plant start-up, bench-marking and an operational experience review, and the extensive use of remote monitoring equipment.

The licensee decided to control the polar crane remotely to minimize the radiation dose from the move. The move was monitored and controlled successfully from outside the containment wall by underwater and dry video cameras. The move was completed without any personnel exposure, and the core barrel was safely placed into the specially designed storage stand located in the refueling cavity.

c. Conclusions

Excellent overall performance was observed during removal of the reactor (Rx) vessel core barrel. The licensee's planning, and execution of this activity allowed the move to be completed without any personnel exposure.

M1.4 Primary Component Cooling Heat Exchanger Leak

a. Inspection Scope

The inspector reviewed the licensee's response to a minor leak from the upper head of the "B" primary component cooling water (PCCW) heat exchanger.

b. Observations and Findings

The licensee identified the leak during post-maintenance testing following re-assembly of the "B" PCCW heat exchanger. The licensee formed a team to review this condition, and to co-ordinate the repair activities. Maintenance technicians removed the upper head assembly, inspected the gasket seating surfaces, and noted minor damage on the flange seating surface. The licensee attributed the damage to improper landing of the heat exchanger head assembly during the initial re-assembly.

The licensee developed a minor modification (MMOD 99-0554) to machine the upper head seating surface to remove the fit-up interference caused by the surface defects. The inspector review this MMOD and did not identify any deficiencies. Additionally, the licensee contacted the heat exchanger vendor who recommended that the structural weld for the upper flange be inspected to ensure that the improper fit-up did not cause any damage. The licensee inspected the weld, and did not identify any damage. The inspector performed an independent visual inspection of this weld and also did not identify any damage. The licensee successfully re-assembled the heat exchanger without any further problems.

c. Conclusions

The licensee responded well to evaluate the extent of damage, and repair a minor leak from the primary component cooling heat exchanger upper head assembly.

M1.5 Refueling Outage/Maintenance Activities

a. Inspection Scope (62707, 92902)

The inspector observed field maintenance activities, including work related to the "B" train safety-related bus E6 outage and the "B" train emergency diesel generator (EDG) cylinder replacement, performed during the de-fueled portion of operations refueling outage (ORO) # 6. The inspector interviewed cognizant system engineers and craft supervisors, as well as work control and operations personnel, to assess the process controls and assure TS and procedural requirements were being met. Both "A" and "B" train equipment that remained in an operating condition with power available was spot-checked to assure proper component status, personnel safety, and tagging controls.

b. Observations and Findings

(1) "B" train 4.16kv bus (1-EDE-SWG-E6) outage:

The inspector checked each electrical breaker cubicle to determine whether the breaker had been removed, was installed and locked open, or was racked in and energized. Danger tags and locking devices were examined, along with special personnel safety aids where ground cables had been installed, and caution tags were noted as applicable to control specific TS operability provisions. Bus E6 remained energized from an incoming offsite power supply through a reserve auxiliary transformer. This allowed continued energization of the 480v bus substation, 1-EDE-US-E62, supplying the control room emergency makeup air and filtration subsystems through a 460v motor control center, 1-EDE-MCC-621.

The inspector also examined bus E6, substation E62, and motor control center (MCC) 621 for the controls established by the breaker refurbishment program. For refurbished and re-installed breakers, tags were found to be affixed to the cubicles, identifying the breaker serial number, the refurbishment work order or purchase order, and appropriate retest and date information. For the energized power supply nodes on MCC 621, the inspector noted proper in-process work controls; including danger and test tags and timely phone communications for the node D40 breaker alignment, where the conduct of motor operated valve testing on a "B" train valve was in progress.

With most of the "B" train equipment unavailable as a result of the bus E6 outage, the inspector checked the status of the safety-related equipment being powered by the "A" train 4.16kv bus E5. The inspector noted a service water (SW) system, cooling tower pump and fan in operation as ultimate heat sink equipment with both the "A" and "B" train SW pumps out of service for pump house bay outage work. The applicable breakers were racked out, locked, and danger tagged. The inspector also spot-checked the "A" train unit substations and MCCs for appropriate breaker positions, tagging, and the existence of locking devices, including those in place on power panel switches.

Since the "A" train charging pump (CS) breaker was found to be racked open and locked, with the "B" CS pump breaker pulled for refurbishment, the inspector reviewed the procedural and schedule controls that would restore a charging pump to an operable status prior to plant entry into mode 6, in accordance with TS 3.1.2.1 and 3.1.2.3 requirements. Discussion with work control and operations personnel revealed that the emergency core cooling check valve and flow balance schedule detailed the operation of each CS centrifugal charging pump in accordance with engineering surveillance procedure (EX1804.063) controls. Additionally, the inspector examined the mode change checklist to be used by operations personnel in taking the plant from de-fueled to mode 6 conditions; noting the use of operations surveillance procedures (OX1408.02 & 04) to verify boron injection flow paths and borated water source availability prior to a plant entry into mode 6.

While inspecting the "A" train electrical switchgear and safety-related equipment, the inspector observed the in-process operation of the "A" train boric acid transfer, residual heat removal, and spent fuel pool cooling pumps, consistent with existing plant conditions.

(2) "B" train EDG outage and cylinder replacement:

The licensee identified some abnormal wear conditions on the #11 cylinder liner on the "B" EDG, and decided to replace the entire assembly with a similar unit from a Unit 2 EDG. During an inspection of the "B" train EDG work area, the inspector reviewed the work request (99RM22404002) and discussed with the system engineer and maintenance personnel the material control and testing details for both the new cylinder installation and the old cylinder analysis. The inspector witnessed rigging and installation of the new cylinder liner into its water jacket. Hydro-testing of the new water jacket, installed as an assembly in the "B" EDG, was confirmed.

The inspector noted proper component controls with the retag request and transfer numbers for use of the Unit 2 equipment in the Unit 1 EDG. The licensee had performed measurements on the new cylinder liner in accordance with the instructions and acceptance criteria delineated in the applicable maintenance procedure, MS0539.18 (Revision 3). The inspector verified the availability of the appropriate maintenance and test equipment calibration data and confirmed the existence of post-maintenance testing and new cylinder "break-in" run time instructions.

Discussion with the cognizant engineering personnel indicated that the licensee intended to perform additional analyses of the old cylinder seal ring and liner wear conditions with the assistance of vendor technical representatives. While initial licensee boroscopic examination of the "B" EDG cylinders had identified the abnormal wear, there was no evidence of EDG inoperability as a result of the identified conditions. The licensee's determination that the cylinder assembly should be replaced appeared to be an appropriate corrective and preventive maintenance decision.

c. Conclusions

The electrical maintenance activities inspected during ORO #6, involving the "B" train 4.16kv bus outage and the "B" EDG outage and cylinder/liner replacement, were well controlled, governed by adequate procedural guidance, and documented in a way that appeared to provide retrievable quality information. The removal of power from "B" train equipment, as well as the continued energization of "A" train components, were found to be in compliance with the plant technical specifications, with equipment restoration planned in accordance with procedural controls. The replacement of the #11 cylinder assembly on the "B" EDG appeared to be a prudent action by the licensee. Implementation was partially observed by the inspector and found to be adequately handled and documented.

M8 Maintenance and Material Condition of Facilities and Equipment

M8.1 First Ten-Year Interval In-service Inspection Program

a. Inspection Scope (73753)

This inspection was conducted to assess the licensee's first ten-year interval in-service inspection (ISI) program and the implementation of the ISI scheduled activities for this outage.

b. Findings and Observations

The ISI work in progress during the sixth refuel outage included ultrasonic testing (UT) of the reactor vessel welds, eddy current examination of the tubes in two of the four steam generators, and nondestructive testing of other pressure retaining components such as piping welds to complete the ASME Code Section XI first 10 year ISI interval scope.

The inspectors reviewed portions of the first ten-year interval ISI Program and noted that it complied with regulatory requirements of the Code of Federal Regulations, Title 10, Part 50.55a(g). This interval ISI Program initiated upon commencement of commercial operation (8/19/90) and is in effect for an interval of ten calendar years of plant service.

Piping Welds to Complete the ASME Code Section XI First Ten-Year Interval

The Seabrook ISI Program Plan presents information regarding performance of nondestructive examinations of Code Class 1, 2, and 3 components and their supports. The requirements were obtained from ASME Section XI. The Program Plan was formatted such that each ASME Section XI Code Category was listed. The ISI Program listed the identification of the items selected for examination, the Period in which the examinations are planned to be conducted and any associated Relief Requests.

The inspector verified that the personnel performing nondestructive examinations were qualified in accordance with SNT-TC-1A or ANSI N45.2.6 for the applicable examination technique as modified by the ASME Code, Section XI, IWA-2300. The licensee selected examination vendors from the approved vendor list. Examination personnel certifications were reviewed and approved by the Seabrook Station NDE Level III and the Authorized Nuclear Inservice Inspector (ANII).

This refueling outage (RF06) marked the last scheduled outage prior to the end of the First ten-year Interval at Seabrook Station. To ensure that the ASME Section XI requirements were met, the licensee performed a reconciliation of the entire first ten-year Interval for ASME Class 1 and 2 welds and components. This reconciliation incorporated a twelve-step review process which included: all nonexempt ASME lines for each system from the latest controlled piping and instrumentation drawings, all welds on those lines from the latest controlled isometric drawings, reviewing ISI Program population and percentages, and then scheduling additional examinations as required to meet the requirements of ASME Section XI. In addition, the licensee maintains an ISI database (Data check) which contains all ISI Program items, breaks them down by Code Category and lists the overall percentage and the percentage examined each period.

The inspector observed a demonstration of the ISI program database and noted that the licensee maintains the entire population of ISI welds and components. In addition, it also provides information on each weld/component such as Code Category, Code Item Number, required examinations, procedures, and outage schedule. The approved personnel certifications, examination procedures, and equipment inventory are entered into the database. A template is generated and supplied to vendor inspection personnel to ensure that only approved procedures are used for examinations. Although hard copy data sheets are the official plant ISI record, data is also entered electronically into the system so the database self-check feature can verify the use of approved equipment and consumables.

The inspector verified that the licensee had an effective system for dispositioning ISI findings. Seabrook Station procedure ES1802.006, "Disposition of Inservice Inspection Anomalies" provides the method to communicate examination failures. The procedure provides a flow path and instructions on disposition of examination failures. Operations personnel are notified through the ACR which is generated as required by this procedure. The Inservice Inspection Coordinator then reviews the examination failure to determine whether the condition is an Inservice condition which would require a sample expansion in accordance with the ASME Code.

The inspector noted that the Nuclear Oversight Group (NOG) was adequately involved in the ISI activities associated with piping welds, supports, and the ten-year reactor vessel ISI. The inspector verified by record examination that some of the NOG activities included reviews of NDE procedures and NDE personnel certifications for acceptability prior to the initiation of NDE activities. The NOG also assessed the NDE ongoing activities, and as of April 22, 1999, the NOG informed the inspector that no conditions adverse to quality were identified which was consistent with the findings of the inspector.

M8.2 Inservice inspection (ISI) Work in Progress

The inspector assessed the licensee ISI progress on ultrasonic examination of the reactor coolant system (RCS) piping. The RCS piping consists of wrought stainless steel, carbon steel, and cast stainless steel welded to cast stainless steel fittings. To aid in the examination of the RCS piping, the licensee developed a supplemental ultrasonic technique for use to evaluate an indication during the conduct of the ASME Section XI Code examination.

The approach was to determine the optimum search unit frequency, angle, wave propagation, and focal depth characteristics for examination of the material in question. It was determined that a reliable examination result was achieved using the supplemental technique. This was demonstrated on the Westinghouse cracked stainless steel piping samples at the EPRI NDE Center. The technique relies on the fact that minimizing the beam path will also minimize the effects of the cast material on the UT process. It was shown during the EPRI session that when a crack was approximately 50 percent through wall (on sample thickness ranging from 2.5 to 3 inches) there was reasonable reliability of detecting the crack tip.

Prior to observing the UT activities of the RCS piping, the inspector reviewed the applicable parts of procedures No. ES98-1-17, revision 0, "Ultrasonic Testing - General Requirements," and No. ES98-1-18, revision 0, "Ultrasonic Testing of Welds." The acceptance criteria were based on the Section XI of the ASME Code.
The inspector performed direct observations of the in-progress NDE activities inside the containment. Specifically, the inspector observed the UT examination performed on RCS Loop 2, weld No. RC-6-1-1. The inspector noted that the equipment and the manner in which the UT was conducted followed the procedures and the work plan. The NDE individuals performing the UT test were currently qualified. During this UT examination no indications were detected. The inspector randomly selected other welds on the RCS piping and reviewed the Liquid penetrant Examination Data sheets and determined that the PT examinations were conducted in accordance with an approved procedure by qualified personnel.

The inspector concluded that the volumetric and surface examinations for the RCS piping were conducted in accordance with approved procedures, using currently calibrated equipment and by qualified personnel. The inspector noted that the licensee was proactive in taking steps to enhance flaw detection capability on the RCS cast stainless steel piping welds.

Reactor Vessel Ultrasonic Testing

The inspector assessed the licensee's reactor pressure vessel (RPV) ISI activities scheduled for this outage. The RPV shell circumferential welds, longitudinal welds, inlet/outlet nozzle to shell welds, nozzle to safe end welds, safe end to pipe welds, and nozzle inner radius areas were examined using automated computer based UT. The extent of coverage for each weld was calculated and where the extent of coverage was found to be less than 90 percent, provision to request relief from the ASME Code as permitted by 10CFR 50.55a(g) via relief request was initiated for later submittal to NRC by the Seabrook staff. Preliminary coverage calculations determined that two vessel welds and four outlet nozzle to shell welds were expected to have less than 90% coverage of the volume by examination with UT.

The UT method was a computer based, robot driven system that uses multiple transducers of various types. This computer application provides for the acquisition of extensive quantities of UT data and permits processing of the data to provide a visual representation of reflectors in the volume being examined. The extent of recordable indications found by UT was compared to the ASME Code Section IX criteria (reference IWB 3200) and found to be acceptable to the applicable criteria. The recordable indications identified were compared to the fabrication records and preservice UT results. The inspector observed a portion of the data analysis performed on longitudinal weld 101-124-0 and reviewed parts of the Reactor Vessel Examination Plan, revision 0, the procedure for Remote Inservice Examination of Reactor vessel Welds ES 99-1-42, revision 00 and related drawings and sketches. No items of concern with the RPV Examination Plan, its performance or the in process documentation were identified.

Eddy Current Examination of Steam Generator Tubes

The inspector assessed the Eddy Current Examination of the Steam Generator (SG) tubes with the following observations. Seabrook has Model F type steam generators with thermally treated Inconel 600 tube material. At Seabrook there are a minimal number of tubes plugged. The steam generator tubes in SGs "B" and "C" were eddy current tested (ECT) during the previous refuel outage. The ECT inspection scope of the "A" and "D" SGs for the 1999 outage included 100% of the tubes by the bobbin coil technique, rotating probe (RPC) examination of 50% of the row 1 and row 2 U-bends, 50% of the tubes at the top of the hot-leg tubesheet, 40% of dents and dings and all of the indications identified by the bobbin coil examination. The only steam generator degradation mechanism identified was the wear of some tubes in the U-bend region by the antivibration structural bars. This resulted in the need to plug approximately 23 tubes in the steam generators. The work package for tube plugging provided controls on the plugging process and steps to identify each tube to be plugged and to verify the proper tubes were plugged. The inspector reviewed portions of the Steam Generator Eddy Current Data Analysis Guidelines Manual and noted it to be consistent with current industry practice for steam generator tube examination. The inspector found the ECT activities to be well planned and conducted by qualified personnel. No items of concern with the SG ECT examination plan, performance or documentation were identified.

c. Conclusion

Inservice inspection activities including examination of the piping welds, reactor vessel, steam generator tubes, and completion of the first 10 year ISI interval were well planned and implemented by qualified personnel in accordance with approved procedures. Observation of nondestructive testing activities in progress showed that the ISI work was conducted with proper oversight by Seabrook staff and the results were well documented. The inspections observed were thorough and of sufficient extent to determine the integrity of the components inspected. Indications, when identified, were evaluated and addressed in accordance with the ASME Code and regulatory requirements. No issues of concern were identified.

M8.3 Oversight Inspection of Electrical Splice Activities

The inspector reviewed quality assurance (QA) surveillance report QASR 99-0096 which identified multiple procedural adherence and documentation deficiencies by construction services (CS) electricians during installation of electrical splices on safety-related solenoid valves. The licensee implemented appropriate corrective actions for the QA findings including an engineering review to confirm that the components remained operable in the "as left" condition, and developing a self-assessment team to review CS work practices, and requirements. The inspector concluded that this was an excellent maintenance oversight finding, and indicated that QA personnel were critical of outage work activities.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Pressurizer Surge Line Cooldown

a. Inspection Scope

The inspector reviewed the licensee's evaluation of two separate issues raised by plant personnel during the plant cool-down for RFO6. The first concern was identified by operators on March 31, and involved a cooldown of the pressurizer surge line in excess of 100°F in one hour. The second issue was identified by a health physics technician who heard loud banging noises originating from the pressurizer cubicle. The licensee indicated that both issues had been identified during previous plant heat-ups or cool-downs.

b. Observations and Findings

The pressurizer is a vertical, cylindrical vessel designed to control the pressure of the reactor coolant system (RCS). The pressurizer is connected by a 14 inch surge line to the hot leg of one of the four RCS loops. The surge line is equipped with a temperature detector and a low temperature alarm. The surge line computer data indicated that the measured temperature decreased from 432°F to 280°F (152°F) in 15 minutes. Following the event, the licensee performed a walkdown of the affected piping and supports and did not identify any damage. (Additionally, the licensee noted that the 152°F surge line cooldown was acceptable and not detrimental to the surge line integrity.)

The licensee attributed the surge line temperature changes to thermal stratification of the surge line. This condition had been previously evaluated by Westinghouse in response to pressurizer surge line stratification issues discussed in NRC Bulletin 88-11. The review indicated that the maximum allowed pipe stratification temperature for the surge line was 320°F. Therefore, the licensee determined that the 152°F cool-down experienced during this event, was bounded by the analysis.

The licensee revised two operating procedures in accordance with Westinghouse recommendations made in WCAP 13588, "Operating Strategies for Mitigating Pressurizer Insurge and Outsurge Transients." Although these changes appeared appropriate, the inspector noted, a minor weakness, in that they were not incorporated until this latest event.

The licensee evaluated the loud noises heard from the pressurizer cubicle during the plant cooldown, and noted that the acoustic monitor located on the common pressurizer discharge line to the relief tank from both the power operated relief valves (PORVs) and the safety valves had alarmed. This monitor is installed to indicate the opening of these valves. The licensee verified that none of the valves had opened, and also did not identify any problems with the piping, related supports, or components.

The licensee's evaluation concluded that the noises were caused by expected movement of the pressurizer safety line ball joints in response to the temperature change. The licensee stated that the ball joints have never required any corrective maintenance. Additionally, the vendor reported that the ball joints should not require any maintenance for "twenty to thirty years" since they are not frequently exercised.

b. Conclusion

The licensee properly identified and evaluated two potential plant issues during the plant cooldown.

E2.2 Fuel Assemblies Upper Nozzle Bolt Integrity

a. Inspection Scope

The inspector evaluated the licensee's response to a Westinghouse notification on April 14th, regarding a gap between the fuel assembly upper nozzle spring holdown block and the upper nozzle forging. The inspector also observed the subsequent inspection of several fuel assemblies by reactor engineering personnel, and reviewed applicable documentation.

b. Observations and Findings

The Westinghouse notification was developed following investigation of recent latching problems during fuel movements at another facility. During RFO6 Seabrook transferred 193 fuel assemblies from the reactor into the spent fuel pool, without experiencing any latching problems. Reactor engineering personnel determined that a total of 28 fuel assemblies (Model "G") were potentially affected by this issue.

On April 20th, the reactor engineers re-evaluated the video recordings of the Model "G" fuel assembly inspections that were initially completed on April 13th, and identified that a gap appeared to exist in at least two of the assemblies. Westinghouse subsequently reviewed the video, and confirmed that the gaps appeared similar to those identified at the other facility.

Westinghouse attributed the latching problem to a 1/8 inch gap between the spring holdown block and the nozzle, and clamp rotation due to a failure of the Inconel 600 top nozzle bolt. The Westinghouse evaluation concluded that the failed screws could present a fuel handling problem, but would not adversely affect the reactor core. Additionally, the evaluation noted that the effects of a failure of all the four top nozzle screws would be minimal since it would result in a minimal displacement (0.8 inches) of the fuel assembly. Westinghouse also concluded that if one or more spring screws on the top nozzle were to fail, all parts would remain on the top nozzle. Westinghouse did provide contingencies to address fuel handling problems should they arise.

Westinghouse was reviewing this issue and may issue a Part 21 report. This concern was limited to only those assemblies manufactured in the 1996 time frame. Seabrook determined that the 28 "G" assemblies, which were scheduled to be reloaded during RFO6, were potentially affected. Seabrook inspected these assemblies using a high resolution camera, and concluded that all had the gap problem, and also that four to six of the assemblies had exhibited significant clamp rotation. No evidence of fractured bolting was observed. The licensee re-designed the core and replaced the 28 "G" assemblies with assemblies that did not have the gap problem.

c. Conclusions

The licensee responded well to fuel assembly upper nozzle bolt integrity issues. The inspector noted that the licensee did not identify the gap problem during the initial RFO6 inspections.

IV. Plant Support

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Outage Exposure Reduction Efforts

a. Inspection Scope (83750, 83728)

The implementation of the ALARA program, relative to work planning and control in support of the refueling outage, was reviewed for the period April 5 - 9, 1999. The inspection included evaluation of performance related to implementing radiological controls as contained in job-specific ALARA reviews, associated procedures, and records. The inspector interviewed staff and selected workers and directly observed radiological controls established for tasks performed in containment and other radiologically controlled areas. Tasks observed included steam generator upper bundle hydraulic cleaning/inspection, simplified head assembly modifications, and preparations for defueling the reactor.

Performance was evaluated relative to the applicable requirements contained in 10 CFR 20 and related licensee procedures.

b. Observations and Findings

The overall planning and preparations to minimize dose and to limit the spread of contamination when performing outage work activities were comprehensive. The Health Physics Department provided effective ALARA oversight through various program controls including the Radiation Safety Committee. Specific ALARA reviews detailed the radiological controls for dose intensive activities, including steam generator upper bundle hydraulic cleaning/inspection, simplified head assembly modifications, refueling preparations, and in-service inspection activities.

System flushes, installation of temporary shielding, teledosimetry, remote cameras, use of mock-up training, and integrating controls in the overall project planning were effective ALARA measures. ALARA reviews were comprehensive; incorporating industry experience, operational/engineering input, and lessons learned from past outages. The pre-job ALARA briefing for transferring the reactor upper internals appropriately addressed departmental coordination, individual responsibilities, completion of pre-requisites, and contingency measures. ALARA controls were included in daily job planning; emergent issues regarding minimizing dose were promptly addressed. Management closely monitored ALARA progress.

The cumulative exposure was maintained below the projected estimate during power operation. The outage ALARA goal was conservatively established. Work and transient activities in certain plant areas having elevated dose rates, such as the mechanical penetration area and ECCS equipment vaults, also challenged the ALARA program in achieving the goal. The final cumulative personnel dosage resulting from outage activities was about 94.3 person-rem. This collective exposure was below the projected estimate of 120 person-rem, indicating that the ALARA measures were effectively implemented.

No unplanned exposures or significant personnel contamination events occurred.

c. Conclusions

ALARA program requirements were well developed, integrated in the work control process and effectively implemented with respect to reactor disassembly and steam generator inspection/cleaning activities. However, the outage ALARA goal was challenged by outage work scope and activities in certain plant areas.

R1.2 Applied Radiological Controls

a. Inspection Scope (83750)

At various times, the inspector accompanied the Health Physics Department management and staff, and independently toured site areas, including the Containment Building, Primary Auxiliary Building, Waste Processing Building, and Instrument Calibration Facility to observe radiological practices, postings, and access controls. Technicians were interviewed to assess their knowledge of routine health physics procedures.

Performance was evaluated relative to the requirements contained in 10 CFR 20 and applicable licensee procedures.

b. Observations and Findings

For the site areas toured, Radiologically Controlled Areas (RCA) were properly posted and access was appropriately controlled. Contamination control measures were conscientiously carried out at the job sites. Radiological surveys were thorough and accurately characterized the radiological conditions in the work areas. Daily source checks of survey instruments and portal instrumentation were appropriately performed. Issuance of instruments was adequately controlled. Survey instrument calibration records were current and complete. Sealed source inventories and leakage tests were performed within the required frequency. Access to the Instrument Calibration Facility was properly controlled and areas within the facility properly posted.

High and locked high radiation areas (LHRA) in the Containment Building and the Primary Auxiliary Building were properly posted, physical barriers were in-place, and warning devices were installed, when appropriate, and were operational. Keys to LHRAs were appropriately controlled. Low dose waiting areas were conspicuously posted and conscientiously used by workers.

Housekeeping in all buildings was satisfactory. Receptacles containing potentially contaminated materials were properly labeled. Workers were observed performing proper contamination control measures.

Dosimetry was appropriately worn in RCAs. Teledosimetry, extremity dosimetry, and multibadging were appropriately designated for various tasks commensurate with the radiological conditions at the job site. Dosimetry records were current and properly maintained. Whole body counting was conservatively performed.

Radiation Work Permits (RWP) were complete with current survey data referenced, appropriate dosimetry designated, and protective clothing/equipment requirements stated. Pre-job RWP briefings were adequately detailed. Through interviews, laborers and technicians were knowledgeable of RWP requirements and current radiological and plant conditions.

Contractor radiation technicians were appropriately screened, trained, and qualified to perform their responsibilities. Shift turnovers between health physics supervision and technicians were comprehensive with current job status and emergent issues thoroughly discussed.

The Respiratory Protection Program was appropriately administered. Worker medical qualifications and fit test data for selected individuals were current and readily retrievable. Respiratory protection devices were properly maintained. Air sampling equipment was currently calibrated and appropriately placed in containment work areas.

c. Conclusions

Radiological controls were effectively implemented as evidenced by a qualified staff properly implementing procedures to minimize external and internal exposure, by developing detailed RWPs, appropriately monitoring personnel exposure, and adequately maintaining radiologically controlled areas.

R7 Quality Assurance in RP&C Activities

a. Inspection Scope (83750)

A Quality Assurance audit, selected surveillance reports, departmental self-assessments, and various management oversight activities were reviewed to determine the adequacy of identifying, evaluating, and correcting deficiencies related to the implementation of the radiation protection program.

b. Observations and Findings

The Radiation Protection Program (RPP) Audit (98-A01-01) was a comprehensive assessment that included observations of supervisory and technician job performance, verification that regulatory requirements were addressed by procedure, and evaluation of the effectiveness of interdepartmental controls. Factors that could degrade program effectiveness were identified and appropriately resolved.

Surveillances by the Nuclear Oversight Group were regularly conducted to evaluate the effectiveness of various elements of the radiation protection program, including instrument calibration and dosimetry programs. Prior to the outage, a dedicated surveillance evaluated the implementation of lessons learned from the previous outage regarding improving ALARA measures. During the current outage, surveillances were frequently performed for tasks involving radiologically significant conditions; e.g. the transfer of the reactor upper internals, and for routine jobs to critique worker practices and evaluate the effectiveness of ALARA controls. Issues were promptly communicated to management.

Several Adverse Condition Reports (ACRs) were reviewed (Nos. 99-1177, 99-1276, and 99-1325). The ACRs were initiated at a conservative threshold to address off-normal practices or trends. Probable causes were reasonably developed for each incident and issues were resolved in a timely manner.

Management observations of in-progress jobs were routinely conducted as part of the site-wide Human Performance Monitoring Program. The quality of pre-job briefings, field activity performance, and turnovers were systematically evaluated. Additionally, during plant tours, Health Physics management and supervision frequently challenged technician knowledge on RWP content and work area radiological conditions using a standardized questionnaire.

Departmental self-assessments of the Radiation Protection Program effectiveness Respiratory Protection Program adequately addressed current performance, the status of initiatives, and identified areas for improvement.

c. Conclusions

The Nuclear Oversight Group and Health Physics management effectively monitored radiation protection program implementation, worker practices, and procedural compliance through close and frequent observations. Prompt actions were taken to evaluate and correct factors that could degrade performance.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comment (71707, 71750)

The inspectors observed security force performance during inspection activities. Protected area access controls were found to be properly implemented during random observations. Proper escort control of visitors was observed. Security officers were alert and attentive to their duties.

P8 Miscellaneous Security and Safeguards Issues

P8.1 Closed LER 99-S01-00: incomplete employee screening records. The LER discussed an event involving an individual who provided false information on his Unescorted Access Authorization Affidavit. Based on this information, the licensee granted the individual temporary unescorted access to the protected area on February 17, 1999. On April 8, 1999, the licensee revoked the individual's access after receiving unfavorable information regarding the individual's criminal history. The licensee reviewed the individual's access history and concluded that he did not adversely impact any vital plant equipment.

The inspector performed an in-office review of this event, and discussed it with the NRC Regional Security Specialist who indicated that the licensee's program was consistent with NRC Requirements. The inspector noted that this event was similar to an earlier event reported by the licensee in December 1998, but did not identify any violation of NRC requirements. This LER is closed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management, following the conclusion of the inspection period on May 13. The licensee acknowledged the findings presented.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. Diprofio, Unit Director
J. Grillo, Assistant Station Director
J. Hill, Operations Supervisor
G. StPierre, Operations Manager
B. Seymour, Security Manager
T. Nichols, Technical Support Manager
D. Sherwin, Maintenance Manager

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observation
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 83750	Occupational Exposure
IP 73753	Inservice Inspection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:	NCV 99-02-01, Failure of Emergency Power Sequencer Relays During Surveillance Testing.
Closed:	NCV 99-02-01, Failure of Emergency Power Sequencer Relays During Surveillance Testing LER 99-001-00, Emergency Diesel Generator Inoperable due to Westinghouse AR Relay Failures. LER 99-S01-00, Incomplete Pre-Employment Screening Records

LIST OF ACRONYMS USED

ACR	Adverse Condition Report
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CAS	Central Alarm Station
CBS	Containment Building Spray
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
FME	Foreign Material Exclusion
gpd	gallons per day
gpm	gallons per minute
LCO	Limiting Condition for Operation
MOV	motor operated valve
MPCS	Main Plant Computer System
NSARC	Nuclear Safety and Audit Review Committee
ODCM	Offsite Dose Calculation Manual
psig	pounds per square inch gauge
QA	Quality Assurance
QC	Quality Control
REMP	Radiological Environmental Monitoring Program
RESL	Radiochemical and Environmental Sciences Laboratory
RHR	Residual Heat Removal
SG	Steam Generator
SORC	Station Operations Review Committee
SUFP	Startup Feedwater Pump
SW	Service Water
TDEFW	Turbine Driven Emergency Feedwater Pump
TLD	Thermoluminescent Dosimeter
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
WR	work request