December 12, 2000

EA 00-265

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

SUBJECT: NRC's SEABROOK INSPECTION REPORT NO. 05000443/2000-008

Dear Mr. Feigenbaum:

On November 18, 2000, the NRC completed an inspection at your Seabrook Nuclear Power Station. The enclosed report presents the results of this inspection. The results were discussed on December 1, with Mr G. St. Pierre and members of your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. The in-service inspection and radiological controls programs were also inspected during this period.

The NRC identified one finding involving the failure to obtain NRC approval prior to removing both EDGs from service to perform a maintenance activity. The issue was evaluated under the Reactor Safety Significance Determination Process as of very low significance (Green). Because of the very low safety significance and because the issue has been entered into your corrective program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Seabrook Station.

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Sincerely,

/RA/

James C. Linville Chief
Projects Branch 6
Division of Reactor Projects

Docket No. 05000443
License No: NPF-86

Enclosure: NRC Inspection Report No. 05000443/2000-008

cc w/encl:
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000443
License No.:	NPF-86
Report No.:	05000443/2000-008
Licensee:	North Atlantic Energy Service Corporation
Facility:	Seabrook Generating Station, Unit 1
Location:	Post Office Box 300 Seabrook, New Hampshire 03874
Dates:	October 1 - November 18, 2000
Inspectors:	Raymond Lorson, Senior Resident Inspector Javier Brand, Resident Inspector Antone Cerne, Senior Resident inspector, Millstone Unit 3 Michael Modes, Senior Reactor Inspector Chris Welch, Resident Inspector, Ginna Station Laurie Peluso, Health Physicist
Approved by:	James Linville, Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000443-00-08, on 10/01-11/18/2000; North Atlantic Energy Service Corporation; Seabrook Station; Unit 1. Engineering Evaluations.

The report covers a seven week period of resident and specialist inspection. The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED) and was determined by the Significance Determination Process (SDP) in draft inspection Manual Chapter 0609 (see Attachment 1).

Cornerstone: Mitigating Systems

- Green. The “A” emergency diesel generator (EDG) was removed from service while the “B” EDG was inoperable to perform a maintenance outage. The licensee’s 10 CFR 50.59 evaluation did not properly consider whether this activity increased the likelihood of the occurrence of malfunction of the emergency power supply to the spent fuel pool cooling system. This activity was considered to be of very low risk since all fuel was located in the spent fuel pool and the time to boil following a loss of cooling was over twelve hours, temporary EDGs and non-electric powered inventory sources were available and the licensee implemented appropriate measures to control the risk while the plant was in this configuration. The inspector reviewed NRC Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” and determined that the configuration described above did not exceed any of the criteria that would have required a Phase 2 analysis. Therefore, this finding was determined to be of very low significance (Green). The failure to properly evaluate this activity was considered to be a violation of 10 CFR 50.59 and entered into the licensee’s corrective action program. This low risk, violation is being treated as a non-cited violation consistent with the NRC’s enforcement policy **(NCV 00-08-01)** (Section R13).

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Report Details

Summary of Plant Status: The plant was operating at approximately 100% power at the beginning of the period. On October 21, the operators shutdown the plant to begin refueling outage seven (RFO7). On November 1, the "B' emergency diesel generator (EDG) failed during a post-maintenance test run. The NRC formed a special inspection team (SIT) to review this failure. The outage was on-going at the completion of the period.

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

R04 Equipment Alignment

a. Inspection Scope

The inspectors walked down critical portions of temporary modification (TMOD) 00-0017. This licensee installed TMOD 00-0017 to provide a source of backup electrical power to emergency bus six while both site EDGs were removed from service for maintenance. The inspector observed a nuclear systems operator (NSO) walk-through operating procedure, OS-001-01-06, "Establishing Alternate Power To Bus 6 - OR07," to assess the procedural adequacy and the operators' ability to place the temporary diesel generators in-service.

Prior to the removal of both EDGs from service, the inspectors reviewed two spent fuel pool (SFP) inventory make-up sources (i.e the refueling water storage tank, and the condensate storage tank) that did not require an electrical power source to function. The inspector confirmed that the expected inventory make-up supply rate from these sources would exceed the rate of SFP inventory loss following a loss of off-site power event.

The inspectors also performed a partial system walkdown inspection of the service water (SW) cooling tower system while the ocean SW system was secured and of the "A" EDG during the "B" EDG maintenance outage. During these walkdowns, the inspectors verified that the redundant systems were properly aligned in accordance with plant procedures and system drawings. The inspectors also observed whether any material deficiencies where present that could challenge the operability of the redundant train.

b. Findings

There were no findings identified.

R05 Fire Protection

a. Inspection Scope

The inspectors toured the fire zones listed below to assess, on a sampling basis, the condition of the fire detection and suppression equipment, fire barriers, and the presence of combustible materials. Station drawings and pre-fire strategy tables were used to verify that fire fighting equipment was available in the field where required and that applicable sections of the fire main were properly aligned and charged. The following areas were reviewed:

- Fuel storage building zone FSB-F-1-A
- Service water cooling tower zones CT-F-1C-A, CT-F-1D-A & CT-F-2B-A
- Primary auxiliary building zone PAB-F-2C-Z

Compensatory measures for the fuel storage building were verified in place due to a degraded fire detection capability. Completed test results for procedure CP-381 Service Water Cooling Tower Fire Detection Operational Test were reviewed.

b. Findings

There were no findings identified.

R08 In-service Inspection Activities

a. Inspection Scope

The inspector performed an inspection of non-destructive examination (NDE) at Seabrook by reviewing the eddy current activities of the steam generators (SGs). The inspector reviewed the data acquisition and analysis being performed using Westinghouse ANSER 8.3, Rev. 70. The inspector interviewed the Seabrook independent resolution analysts. The inspector also interviewed one of the two additional Seabrook independent analysts who were reviewing randomly selected eddy current data (one on day shift and one on night shift).

The inspector witnessed the acquisition of eddy current data taken simultaneously from two tubes. The inspector reviewed the landmarking and set-up of the system with the Westinghouse program/developer responsible for the robotics system. The inspector directly observed data taken from tubes: SG B R1 C60 taken simultaneously with R2 C59, and SG B R3 C98 taken simultaneously with R4 C97. The inspector also directly observed a probe change with subsequent calibration and data pull for SG C R3 C98 simultaneously with R4 C97.

The inspector reviewed "Steam Generator Eddy Current Data Analysis Guidelines Manual" Rev 0, Effective Date November 1, 2000. The inspector also reviewed "Seabrook Fall 2000 1R07 E/C Operator Information" and "IP2 Spring 2000 Outage U-Bend Plus Point Analysis Training".

The inspector reviewed "Westinghouse Guidelines for U-Bend RPC Noise Measurements Before and During Inspection" MRS-TRC-1139. The inspector discussed the method, its application, and the specific data with the author of the method. The inspector verified the data taken for SG C R1 C33 and SG C R1 C37 by repeating the the specific maximum voltage measurements at the tube apex originally used to calculate the noise level in the tubes. The inspector reviewed the average noise calculations of the tube data taken from the Cycle 6 Plus Point and RPC examinations at 300 kHz and 400 kHz for SG A and D compared against the Electric Power Research Institute qualification data set ETSS 96511.

The inspector reviewed, with the Seabrook independent oversite analyst, the bobbin inspection data and calibration set-up for wear depth sizing using calibration standard FMST-003-99 for: SG C R24 C7 with Anti-vibration bar (AVB) wear at AV1 of 17%, SG C R30 C11 with AVB wear at AV1 of 18%, SG C R20 C11 with AVB wear at AV1 of 29%, SG C R39 C17 with AVB wear at AV3 of 43%, and SG C R48 C34 with AVB wear at AV3 of 58%. The inspector reviewed rotating pancake coil (RPC) and Plus Point data taken of a small indication at the top-of-tube sheet in SG C R43 C28. This indication was interrogated by alternate qualified eddy current methods and sized as 11% through wall dimension.

The inspector interviewed the Chemistry Manager and discussed the 0.03 to 0.15 gallon per day leak that was being tracked in SG B during the last cycle. This leak was calculated by measuring the amount of Argon and Xenon in the condenser off gas system. These measurements were compared with samples grabbed from the SG blow down stream. The measurements used to calculate the leak were at the limit of detection for these isotopes. Seabrook had experienced a similar leak during Cycle 6 located in SG D. This leak was also calculated using similar techniques to those currently used and determined to be in a similar range. SG D was subsequently eddy current examined and a tube containing 71% AVB wear was plugged. During Cycle 7 this SG did not leak. The inspector reviewed the leak charts for both the SG D and SG B leak cycles and the data used to produce the plots.

The inspector reviewed radiographs on FW-4608 weld F0901 consisting of a sweepolet to a cap weld for a branch connection that was being abandoned. There were four radiographs consisting of three film each. These radiographs were completed in conformance with the requirements of ASME Section III, for Class 2, 1995 Edition with the 1996 Addenda. The inspector reviewed FW 4607 weld F1601; a similar weld and radiographic set up.

The inspector also reviewed the results of additional NDE. The inspector reviewed the results of the visual examination of SI 0202-02V17B, the liquid penetrant of RH 0157-10 03, the liquid penetrant of RH 0157-01 10, the magnetic particle of MS 4000-02 05, the ultrasonic testing of RH 0157-01 10, and the ultrasonic testing of MS 400302 08.

The inspector reviewed the ASME Section XI Repair/Replacement Plan Traveler for WR 00RE00860001 for the replacement of the diesel generator relief valve, DG-V-118. This plan was in conformance with the requirements of Section III, 1974 Edition with the Summer 1996 Addenda.

The inspector reviewed two CRs picked from a list of thirteen condition reports attributed to the in-service inspection (ISI) coordinator. Condition report 00-11363 was generated as a consequence of an unacceptable linear indication of FW 4608-03-1506-18, weld FW 4608-03 03B and condition report 00-11339 was generated because a Level III signed off on an ISI data report before completing the qualification guide.

b. <u>Findings</u>

There were no findings identified.

R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed problems involving selected in-scope systems, structures, and components (SSCs) to assess the effectiveness of the maintenance rule program. The reviews focused on proper maintenance rule scoping, characterization of failed SSCs, safety significance classifications, 10 CFR 50.65 (a) (1) and (a) (2) classifications, and the SSC performance criteria.

The inspector reviewed the June 2000 nuclear instrumentation (NI) system health report and the event evaluation report for CR 00-11011. The event evaluation report described a problem with source range NI detector channel N32 that resulted in the generation of a reactor trip signal while the plant was in Mode 3. The inspector also interviewed the NI system engineer and reviewed the licensee's plans for improving the NI system performance monitoring program.

The inspector reviewed the maintenance rule functional failure (MRFF) determination discussed in CR 00-12491 for a problem involving loose components internal to a containment ventilation system. The licensee concluded that this problem did not affect the containment ventilation function but did not evaluate whether the containment isolation function was affected. The inspector reviewed the MRFF determination for CR 00-13175 which the licensee subsequently initiated to evaluate this aspect of the problem. The licensee concluded that the loose components did not affect the containment isolation function.

The inspector also reviewed CR 00-11860-02 which included the MRFF determination and corrective actions for a degraded bearing support internal to main steam valve MS-V-394 actuator. This valve is required to open to supply a steam path to the turbine driven emergency feedwater pump. The licensee determined that the degraded bearing support was not a MRFF and implemented a vendor approved modification to improve the design of the bearing support to prevent recurrence of this problem.

b. Findings

There were no findings identified.

R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspector sampled, through direct observation and/or document review the scheduling and conduct of the selected maintenance activities performed during RFO7 to determine whether the licensee properly evaluated and controlled these activities to protect the key shutdown safety functions. The maintenance activities reviewed included: replacement of the "A" charging pump and "A" residual heat removal pump mechanical seals, inspection of the service water system intake structure and the "A" service water system strainer.

The inspector also reviewed the risk associated with the removal of the “A” EDG from service on November 11, while the “B” EDG was inoperable. The inspector observed the on-site safety review committee meeting that approved this activity and reviewed applicable sections of the Updated Final Safety Analysis Report (UFSAR), the licensee’s 10 CFR 50.59 safety evaluation and Standing Operations Order 00-013 that described the licensee’s activities to minimize the risk associated with this activity.

b. Findings

The “A” EDG was initially removed from service to perform an inspection that had been developed by the event team investigating the cause for the “B” EDG failure. The licensee determined, at the completion of the recommended inspection activity, that the material condition of the “A” EDG was acceptable. The licensee elected to defer restoration of the “A” EDG to an operable condition in order to perform an extensive vendor recommended maintenance outage designed to enhance the “A” EDG condition.

During this event all fuel was located in the spent fuel pool and all fuel handling activities were suspended. The licensee procured temporary EDGs (described in Section R23) to provide a redundant supply of electrical power in the event of a loss of off-site power. Additionally, the licensee restricted any activities that could challenge the off-site power system, ensured adequate inventory within the spent fuel pool, and verified the availability of non-electric powered SFP inventory make-up sources. The licensee determined that the SFP temperature would not reach the boiling point for over twelve hours following a loss of SFP cooling event. The inspector determined, based on the above, that the removal of the “A” EDG in this condition was of very low risk.

Section 9.1.3 of the UFSAR stated that, “the spent fuel pool pump motors are Class 1E motors and are supplied from separate emergency busses.” The licensee’s 10 CFR 50.59 evaluation for this activity determined that it did not increase the likelihood of failure of equipment important to safety and therefore did not require NRC approval. The licensee’s evaluation, however, did not discuss the above statement and therefore the inspector questioned the adequacy of the licensee’s evaluation. The NRC conducted an enforcement panel on November 27, to review this concern and determined that the removal of both emergency EDGs from service had increased the probability of failure of the emergency power supply for the SFP pump motors. The licensee entered this finding into the corrective action program as CR 00-13520.

The inspector reviewed NRC Manual Chapter 0609, Appendix G, “Shutdown Operations Significance Determination Process,” and determined that the configuration described above did not exceed any of the criteria that would have required a Phase 2 analysis. Therefore, this finding was determined to be of very low significance (Green) per the significance determination process. Title 10 to CFR Part 50.59, requires, in part, that NRC approval be obtained prior making changes to the facility as described in the UFSAR that result in more than a minimal increase in the likelihood of occurrence of a malfunction of a system important to safety. Contrary to the above, the licensee established a plant configuration that increased the likelihood of failure of the spent fuel pool cooling system as described in the UFSAR. This is a violation of 10 CFR 50.59. This low risk, violation has been entered into the licensee’s corrective action program and is being treated as a non-cited violation consistent with the NRC's enforcement policy. **(NCV 00-08-01)**

R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed corrective action program documents (CR 00-11860-02, and CR 00-13175) and interviewed design and system engineers to evaluate the licensee's basis for determining that the problems discussed in Section R12 associated with the containment isolation system, and the turbine driven emergency feedwater system did not challenge the system operability.

The inspectors also reviewed the licensee's evaluation of a condition, identified prior to the shutdown for RFO7, involving contact between safety injection (SI) valve SI-V-112 motor operator, and scaffolding that was in the process of being erected. The licensee determined that minimal force would have been transferred to the valve due to the interaction with the scaffolding.

The inspectors reviewed applicable TSs, technical requirements (TR), the updated final safety analysis report (FSAR), and CRs 00-10545, 10546, 10537 and 10563, to assess the technical adequacy of the licensee's response to problems identified on October 11, 2000, regarding the control building air handling (CBA) system. Specifically, the inspector evaluated the licensee's operability determination (OE 4.5) which concluded that the 5 second criteria specified in TR 2.2 did not apply to damper 1-CBA-DP-28, and therefore, the slower damper closure time of 5.87 seconds did not degrade the control room complex exhaust isolation function. The inspector also reviewed the licensee's conclusion that requirements specified in TS 4.3.2.2, Engineered Safety Features actuation System Instrumentation, does not require time response testing of the CBA fans and dampers on a high radiation signal, and therefore, the failure to satisfy the time response requirements for CBA fan/filter actuation specified in TR 2, Item 14, did not require entry into TS 4.0.3 or TS 3.0.3.

b. Findings

There were no findings identified.

R17 Permanent Modifications

a. Inspection Scope

The inspector reviewed design change MMOD 99-0527, which modified the ventilation air supply flow into the residual heat removal (RHR) and safety injection (SI) equipment rooms. The modification was designed to reduce the likelihood of spreading loose surface contamination within these rooms. The inspector reviewed the post-installation test scope, data and acceptance criteria, drawings, and the modification package to determine whether the primary auxiliary building emergency air handling system would remain capable of meeting design bases requirements.

The inspector reviewed CR 00-12184 which indicated that the modification was accepted “as is” without performing an engineering evaluation for the test results which did not meet the test acceptance criteria in several areas. Design engineering performed a subsequent evaluation and determined that the operability and functionality of any equipment located in the affected areas had not been challenged by the post-modification ventilation flowrates. The inspector reviewed the licensee’s plans to balance the system ventilation flowrates.

b. Findings

There were no findings identified.

R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed the scope of the post-maintenance test activities, reviewed the test data, and/or observed a portion of the test activities following the completion of several maintenance activities including: replacement of source range nuclear instrument channel N32, replacement of the “A” RHR pump mechanical seal, modification of the ventilation in the safety equipment vaults, and the “A” EDG maintenance outage.

b. Findings

There were no findings identified.

R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed diverse operational, maintenance and scheduling activities prior to and during RFO7 to evaluate the licensee’s activities to assess and manage the outage risk. Specific activities reviewed included:

- Control of the plant shutdown, examination of the reactor system cooldown data to ensure that TS requirements were met, plant shutdown parameters, independent determination of the decay heat load, reactor vessel dis-assembly and re-assembly, and fuel handling activities.

- Configuration management to ensure that adequate reactor process instrumentation, decay heat removal, electric power, inventory make-up, and containment systems were available to minimize plant risk.
- Review of CR 00-12229, which discussed a concern associated with the clarity of the Mode 6 TS definition, and Standing Operations Order 00-012 which provided operational guidance when all fuel was located to the spent fuel pool.
- Control of temporary systems and equipment, including scaffolding, to ensure that temporary installations did not adversely challenge mitigation systems.
- The identification and resolution of problems by the review of selected condition reports and corrective actions (as discussed throughout this report).
- Review of the controls and implementation details associated with two separate equipment isolation and tagging activities. The first (clearance order 1-1167-00) involved a nine-step isolation of the train "B" main feedwater header valve 1-FW-V-39. The other tagout (clearance order 1-0981-00) encompassed a 57-step isolation of the train "B" emergency diesel generator (EDG), 1-DG-1B, to include the issuance of two supplemental clearance order sheets. Both tagouts were evaluated with regard to the requirements and guidance specified in Seabrook Station Administrative Procedure, MA4.2 (Revision 18), for "Equipment Tagging and Isolation". In verifying the equipment tags in the field, the inspector examined main control board switch positions and toured applicable areas in the turbine building, switchgear rooms, and all elevations of the EDG building. To confirm the proper position and tagging of certain valves skid-mounted on 1-DG-1B, the inspector signed-in under the licensee's control of the "B" EDG area as a foreign material exclusion (FME) zone, additionally assessing the adequacy of the licensee's FME work control activities. As necessary to ascertain the acceptability of tagout sequence performance and control authorization, the inspector interviewed licensed operators, tagging control supervisors, and designated work contact personnel, as such were available at the tagging office or equipment locations.

Additionally, during the examination of the isolation and tagging of the train "B" EDG fuel oil transfer pump, associated with clearance order 1-0981-00, the inspector noted the use of a temporary equipment tag to mark replacement of a relief valve on the discharge piping for the fuel oil transfer pump with a similar valve, originally intended for Seabrook Unit 2 use. The inspector verified the replacement relief valve, 2-DG-V-124, had been subjected to a satisfactory setpoint/leak test, in accordance with the applicable engineering procedures and ASME Section XI requirements, prior to the start of RF07. During this review the inspector interviewed the cognizant maintenance valve group supervisor and toured the hot-test facility where the relief valve was successfully tested.

b. Findings

There were no findings identified.

R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed licensee performance related to the conduct of local leak rate test (LLRT) activities. The provisions delineated in the North Atlantic Technical Requirements Program (TRP) 5.3 document were discussed with the cognizant program supervisor, particularly with regard to Type C testing acceptance criteria, testing frequencies, and the documented rationale for exempting certain containment penetrations from LLRT performance. The inspector specifically examined surveillance test records for evidence that the pre-outage (RF07) compilation of combined leakage for all Type B and C penetrations (e.g., as a result of LLRT of penetration 1-MM-MM-29 for the containment equipment hatch airlock in July 2000) was maintained below prescribed Technical Specification and TRP 5.3 limits. The inspector also reviewed Engineering Procedure, EX1803.003 (Revision 06), for “Reactor Containment Type B and C Leakage Rate Tests”, and checked the Type C test conduct and results, in accordance with the procedural controls, for the following RF07 containment penetrations:

- X-14 (two train “A” containment spray valves)
- X-15 (two train “B” containment spray valves)
- X-35D (four reactor coolant loop 3 sample line valves)
- X-71C (two combustible gas control valves, including retest of one valve)

The inspector also reviewed the current summary of the Type B and C LLRT status for all the containment penetrations scheduled for testing during RF07. The inspector noted various dispositions, including plans for valve replacement, where certain leakage rates were observed. During the sample review of ongoing LLRT activities, the inspector identified no penetration where the administrative leak rate criteria of TRP 5.3 and EX 1803.003 were exceeded.

The inspectors observed several surveillance testing activities of safety related components to verify that the system and components were capable of performing their intended safety function, to verify operational readiness, and to ensure compliance with required TSs and surveillance procedures. The surveillance observations and documentation reviews included the following:

- Service water cooling tower make-up pump SW-P-329, 18 month surveillance testing per surveillance procedure EX1804.031. The inspector also reviewed CR-0010347 which discussed observed difficulties associated with the initial priming of the service water cooling tower makeup pump. The licensee planned to review this condition to determine the necessary corrective actions.

- Main steam safety valve in-place set point verification test per surveillance procedure EX1804.041.
- Testing of 3 primary component cooling water valves per operations procedure OX 1412.1.
- 18 month loss of power testing of the "A" EDG train.
- "B" Train SW testing per operations procedure OX 1416.13.

b. Findings

There were no findings identified.

R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the details of TMOD 00-0017, which established a contingent means to provide electrical power for spent fuel pool cooling in the event off-site power was lost and the emergency diesel generators were not available. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and Technical Specification (TS) requirements, compared the actual equipment configuration to the modification document and installation instructions contained in work request (WR) 00W002004, and reviewed associated drawings. The inspector also reviewed condition reports 00-12951, and 00-12962 which described minor problems associated with the TMOD drawings.

b. Issues and Findings

No findings were identified.

2. RADIATION SAFETY
Cornerstone: Public Radiation Safety (PS)

OS2 ALARA Planning and Controls

a. Inspection Scope

.1 Job Site Inspections and ALARA Control

The effectiveness of the ALARA planning and controls program was determined for the Unit 1 refuel outage (RF07) during October 30-November 03, 2000. The inspector reviewed the four highest exposure jobs (activities with estimated collective exposures greater than 1 person-rem) that were in-progress or completed during this inspection period. The jobs reviewed were: (1) reactor disassembly and reassembly; (2) Insulation; (3) Residual Heat Removal pump (8A removal); (4) Steam Generator Eddy Current Testing and Tube Plugging. Areas reviewed for these jobs included an evaluation of the use of engineering controls to achieve dose reductions; review of the use of low dose waiting areas; review of on-job supervision provided to workers; a review of individual exposures from selected work groups.

The inspector conducted observations of radiation worker and radiation protection technician performance during high dose rate and/or high exposure jobs, listed above, to determine if the training/skill level is sufficient with respect to the radiological hazards. The inspector also conducted a review to examine the assumptions and basis for the various job estimates, including the methodology utilized for estimating job-specific exposures.

b. Issues and Findings

No findings were identified.

4. OTHER ACTIVITIES [OA]

OA3 Event Follow-Up

(Closed) LER 50-443/00-006, and 00-006, Supplement 1: main steam safety valve (MSSV) lift pressure outside technical specification limits. The licensee identified, during testing on October 20, 2000 that the “as found” set pressure for MS-V6 exceeded the TS required value. The valve was tested two more times and met the TS required lift pressure setpoint. The licensee also tested an additional eight MSSVs and did not identify any additional problems. The licensee’s planned corrective actions for this event include: upgrading of the MS-V6 disc with a modified disc that is considered less susceptible to setpoint deviations, and increasing the MS-V6 test frequency to once per refueling outage. The inspector determined that the licensee’s actions were reasonable and complete. This LER is closed. This event did not constitute a violation of NRC requirements.

OA5 Other

(Closed) EA 98-165/98-338, 98-339: These violations involved a licensee contractor that discriminated against a contract employee for raising a safety concern regarding a safety-related wiring installation. The parties implemented several corrective actions to address the immediate problem and implemented additional follow-up corrective actions to improve the safety conscious work environment (SCWE). The follow-up actions included: overhaul of the corrective action program, development of a formal plan to enhance SCWE awareness through training and meetings, and the conduct of a site culture survey. The inspector determined that the parties actions were reasonable and complete and noted that no problems were found in this area during the September Problem Identification and Resolution Team Inspection. This violation is closed.

OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. G. St. Pierre and other members of licensee management on December 1, 2000 following the conclusion of the period. The licensee acknowledged the findings presented. Additionally, the inspectors met with members of licensee management following the conclusion of the in-service inspection and radiation protection inspections.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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J. Grillo, Assistant Station Director
G. StPierre, Operations Manager
T. Nichols, Technical Support Manager
D. Sherwin, Maintenance Manager
J. Pandolfo, Security Manager
M. Ossing, NRC Coordinator
R. Anderson, Work Contact and Outage Manager
M. Campbell, Radiological Technical Specialist
W. Cash, Health Physics Department Manager
W. Cox, Radiological Technical Specialist
M. DeBay, Asst. Operations Manager
P. Harvey, Chemistry Department Manager
W. Leland, Chemistry/Health Physics Manager
W. Meyer, Jr, Health Physics Technician
M. Perkins, Health Physics Technician
D. Robinson, Chemistry Technical Supervisor
J. Sobotka, Regulatory Compliance Supervisor
R. Sterritt, ALARA Coordinator

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened:

NCV 00-08-01: Failure to Develop an Adequate 10 CFR 50.59 Safety Evaluation Prior to Removing the “A” EDG From Service While the “B” EDG was Inoperable.

Closed:

NCV 00-08-01: Failure to Develop an Adequate 10 CFR 50.59 Safety Evaluation Prior to Removing the “A” EDG From Service While the “B” EDG was Inoperable.

LER 50-443/00-006, and 00-006, Supplement 1: Main Steam Safety Valve Lift Pressure Outside Technical Specification Limits.

EA 98-165: Discrimination Against a Contract Employee for Raising a Safety Concern.

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
AVB	Anti-Vibration Bar
CBA	Control Building Air
CR	Condition Report
EDG	Emergency Diesel Generator
FME	Foreign Material Exclusion
FSAR	Final Safety Analysis Report
ISI	In-service Inspection
LLRT	Local Leak rate Test
MSSV	Main Steam Safety Valve
MRFF	Maintenance Rule Functional Failure
NCV	Non-cited Violation
NDE	Non Destructive Examination
NI	Nuclear Instrumentation
NSO	Nuclear Systems Operator
RHR	Residual Heat Removal
RFO7	Refueling Outage 7
SCWE	Safety Conscious Work Environment
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SG	Steam Generator
SIT	Special Inspection Team
SSC	Structure, System, or Component
SW	Service Water
TMOD	Temporary Modification
TR	Technical Requirement
TRP	Technical Requirements Program
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report

Attachment 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

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

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

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

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

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

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

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