


U. S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee: North Atlantic Energy Service Corporation
Post Office Box 300
Seabrook, New Hampshire 03874
Facility: Seabrook Station
Dates: January 1, 1996 - February 12, 1996
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Approved By:


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3/6/96
Date

Inspection Summary: Inspections were conducted during normal and backshift hours in the areas of plant operations, maintenance, engineering, and plant support. Reactive inspections of operational response to the January 27, 1996 automatic reactor trip and subsequent recovery activities were performed. Additionally, reactive inspection of the changing chemistry in the main generator stator water cooling system was conducted. Followup inspections of previous inspector concerns and Licensee Event Reports were accomplished.

Results: The results of the inspection are summarized in the Executive Summary. One previously identified unresolved item regarding cracked emergency diesel generator turbocharger housing bolts was closed (UNR 50 - 443/95-04-01).

EXECUTIVE SUMMARY

SEABROOK STATION NRC INSPECTION REPORT NO. 50-443/96-01

Plant Operations: Control room operators responded appropriately to the January 27, 1996, automatic reactor trip. Emergency Operating Procedures were properly entered and transitioned through to normal station recovery. The shift manager demonstrated excellent plant and personnel safety perspectives in the initial response, evaluation of, and repair controls to a main generator hydrogen cooler gas leak.

Maintenance: Comprehensive troubleshooting work plans were developed to address each component anomaly encountered during and following the reactor trip. The electro-hydraulic control system plan was particularly well developed and was supported by turbine generator control system supplier expertise. Additionally, routine maintenance observed was effectively implemented.

Engineering: A modification design change was effectively developed and implemented to resolve concerns regarding emergency diesel generator turbocharger housing bolt cracking. Additionally, design and system engineers effectively supported the January 27, 1996, post trip evaluations as well as the main turbine generator stator water cooling system oxygen concentration concerns.

Plant Support: Radiological controls were properly implemented throughout the report period. Radiation workers were observed to be wearing proper dosimetry and protective clothing. The security staff effectively implemented the security plan requirements during the report period.

The licensee maintained excellent radioactive liquid and gaseous effluent control programs, with capabilities to protect the public health and safety and the environment. The licensee also upgraded the Offsite Dose Calculation Manual (ODCM) and radioactive liquid and gaseous effluent control procedures since the last inspection, conducted in March 1994. The upgraded ODCM and procedures were well developed and effectively integrated. The Chemistry Staff demonstrated excellent knowledge in the effluent control programs. The licensee's staff demonstrated excellent ability to understand and implement the applicable technical specifications for air cleaning systems. The Chemistry Staff responded to QA audit findings in a timely manner, and with good corrective actions.

The upgrading efforts for the ODCM and effluent procedures were well done and provided for clear and concise instructions and documentation of program specifications.

Safety Assessment/Quality Verification: A supplemental Licensee Event Report addressing area temperature requirements was very broad based and demonstrated good self critical analysis. Additionally, plant management displayed outstanding plant and reactor safety perspectives during the ongoing evaluations of the stator water cooling system.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

At the start of the report period, the reactor was at 100% of rated thermal power. On January 27, 1996, at 9:45 a.m., the reactor automatically tripped in response to high pressurizer pressure caused by electro hydraulic control malfunction (Section 2.2). On January 30, the reactor entered Mode 1. Reactor power was increased and reached 100% power on January 31 at 9:05 p.m.. On February 1, at 3:06 a.m., reactor power was decreased in response to low generator stator cooling flow. At 7:22 p.m., the reactor power was reduced to 18% and the main turbine was tripped for generator stator cooling troubleshooting (Section 2.3). On February 2, at 5:16 a.m., the main generator breaker was closed and a slow power increase commenced. The reactor was held at 40% power from February 5 to February 9 due to generator stator cooling concerns. Reactor power ascension was in progress at the end of the inspection period.

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

2.1 Plant Operations Review

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

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

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

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

2.2 Automatic Reactor Trip

On January 27, 1996, at 9:45 a.m., the reactor automatically tripped from approximately 100% of rated thermal power. The reactor protection system initiated in response to high pressurizer pressure in excess of the Technical Specification (TS) trip setpoint of 2,385 psig, as stated in TS TABLE 2.2-1.

All safety-related systems functioned as designed in response to the reactor trip. Both power operated relief valves (PORVs) actuated, limiting the ensuing transient to a maximum pressure of approximately 2,387 psig. The pressurizer relief tank, which receives the discharge from the PORVs, experienced a maximum pressure of approximately 12 psig. The feedwater system isolated, as designed, due to the reactor trip with a coincident low reactor coolant system average temperature (Tave). The emergency feedwater system initiated, as designed, in response to the feedwater system isolation. Following the reactor trip, control room operators entered Emergency Response Procedure (ERP), E-0, "Reactor Trip and Safety Injection," and transitioned to ERP ES-0.1, "Reactor Trip Response" following verification that a safety injection had not occurred nor were valid initiation signals received for a safety injection. Operators promptly stabilized the reactor in Mode 3, HOT STANDBY, with the reactor subcritical and average coolant temperature stable above 350 degrees Fahrenheit.

The post trip report investigation revealed that less than a minute before the automatic reactor trip, an electro-hydraulic control (EHC) system circuit board experienced a component failure that introduced a false high speed error signal, causing the main turbine control intercept valves and control valves to close. Control room operators were alerted to the failure by receipt of an "EHC Malfunction" alarm and indication that the combined intercept valves had received a fast closure signal. Additionally, operators also observed that the generator output meter was indicating that generator load was rapidly decreasing and control rods were stepping in to the reactor core in response to the mismatch between Tave and the programmed reference temperature (Tref). Upon receipt of this information, the unit shift supervisor (USS) directed that a manual reactor trip be initiated. However, due to the speed of the transient, the automatic reactor trip on high pressurizer pressure occurred before the control room operator could initiate the manual reactor trip.

Several minor equipment malfunctions were experienced by the licensee during plant response to the reactor trip. Initially, closed indication for the "A" feedwater isolation valve was not received in response to the feedwater system isolation. The plant computer indicated the valve was fully closed, however it was not until approximately 15 minutes after receipt of the isolation signal and following exercise of the isolation valve limit switch that full closure indication was attained.

Additionally, at 12:23 p.m. on January 27, 1996, following the plant shutdown, operations personnel identified a hydrogen gas leak at a flanged joint on the generator hydrogen cooler. The USS directed that; the turbine building be evacuated of non-trip response related personnel; the hydrogen supply be isolated; turbine building ventilation be maximized; the fire brigade leader assess the potential for igniting the vented hydrogen; and emergency action levels be monitored for applicable entry condition criteria. The fire brigade leader reported that all samples for the presence of hydrogen were less than detectable with the exception of one sample taken within 18 inches of the leakage source. The brigade leader advised that there was little concern for the hydrogen igniting. The shift supervisor concluded the leak did not pose a risk to the plant and directed that the generator be evacuated of the remaining hydrogen and be purged with carbon dioxide consistent with existing

station procedures. The hydrogen cooler repair is documented in section 3.2 of this report.

Following the reactor trip, control room personnel also identified that main condenser vacuum had degraded to approximately 27 inches of mercury (in. hg.). Ultimately, the licensee identified leakage at the tubeside manway covers on the "A", "C", and "D" moisture separator reheaters (MSRs). The MSRs were isolated and condenser vacuum performance improved. The manway cover bolts were torque checked and a hot torque value of up to 600 foot pounds was authorized.

The inspector observed portions of the post trip recovery activities in the control room, reviewed selected instrument recorder traces, attended related management meetings, and observed various troubleshooting activities. Control room operators responded appropriately to the reactor trip. The symptoms of the EHC failure were quickly identified, however due to the speed of the transient the automatic reactor trip occurred before operators could initiate a manual trip that had been directed by shift supervision. Additionally, operators properly entered and transitioned through emergency operating procedures. The shift supervisor displayed good personnel and equipment safety perspectives in evaluation of the gas leak that developed on the main generator hydrogen cooler. Overall, the causal analysis of the trip was thorough and properly addressed each aspect of the trip. The inspector had no questions regarding this issue.

2.3 Generator Stator Cooling Water System

Following plant restart from the January 27, 1996, the licensee experienced increased oxygen levels in the generator stator water cooling system, which in turn caused accelerated formation of copper corrosion. The increased copper corrosion caused system strainers to become clogged, reducing system flow, and increasing system conductivity. Although not safety-related, generator stator water cooling system process controls initiate power runbacks on decreased system flow. Additionally, proper system operation is essential to the equipment well being of the main generator. The licensee conservatively reduced reactor thermal power and generator electrical output to minimize the potential for a transient initiated within the stator cooling system. An issue investigation team was formed to thoroughly evaluate the evolving status of the system. The investigation team was extensively supported by several turbine generator vendors and independent offsite power plant chemistry expertise.

Following several days of investigation, the licensee endorsed the General Electric Company recommendation to provide a forced oxygen injection modification to the stator water cooling system to ensure an oxygen saturated environment. Oxygen saturation supports the formation of cupric oxide which creates a stable protective layer of the copper components in the stator water cooling system. The licensee developed temporary modification, 96-TMOD-0007, that implemented the nonsafety-related change. The modification was installed consistent with the controls established in work request, 96W000199.

A monitoring program was established to trend system flow, temperature, and chemical status during a controlled power ascension following installation of the oxygen injection system. The modification has evidenced good initial results. System parameters had stabilized and reactor power was being gradually increased at the end of the report period.

The licensee demonstrated excellent plant and equipment safety perspectives throughout the evaluation of this issue. Reactor power was conservatively decreased throughout the period of evaluation and the main generator was removed from service during certain aspects of system maintenance. The event team employed diverse system and chemistry expertise available from onsite and independent offsite sources. The licensee clearly and deliberately evaluated the options for managing oxygen concentrations within the stator water cooling system before selecting a forced oxygen injection option. The inspector had no concerns at this time and will continue to monitor system status during routine inspection activities.

3.0 MAINTENANCE (61726,62703,92902)

3.1 Electro-Hydraulic Control System Troubleshooting

The licensee initiated work request, 96W000158, to troubleshoot the electro-hydraulic control (EHC) system failure that caused the January 27, 1996, automatic reactor trip. The inspector reviewed the work package and observed portions of the troubleshooting activities. Good configuration control was noted throughout the lifting and landing of test jumpers. Additionally, the troubleshooting steps and in-process work were well controlled and thoroughly documented in the work package. The troubleshooting was supplemented by a control system specialist from the General Electric Company, the turbine generator supplier. As stated above, the EHC system fault was isolated to the low value gate or the backup gate on one of two speed error circuit boards. Because the fault could not be consistently reproduced during diagnostic testing of each card, the licensee conservatively replaced both cards. Subsequent post maintenance testing of the newly installed circuit boards was performed satisfactorily.

Previously the licensee had performed a single failure vulnerability study entitled "Operational Reliability of Seabrook Station." This study had identified the nonsafety-related EHC system as having numerous sub-components, whose single failure could result in a plant trip. The failed primary and backup low value gates for the speed error EHC circuit cards were identified in the study, which established a failure rate of 0.00002 percent probability per hour of operation. The current control system design provides little option to limit single failure circuitry. As a result, the licensee made an internal decision to upgrade the control system during the sixth refueling outage (approximately 1999) to a newer generation system that includes significantly more design redundancy. In light of this recent failure the licensee is evaluating the ability to expedite procurement and installation of the new system. The inspector had no further questions regarding this maintenance activity.

3.2 Hydrogen Cooler Leak Repair

As documented above, the turbine generator hydrogen cooler developed a hydrogen leak following the January 27, 1996 automatic reactor trip. Initially, operations personnel responded effectively to isolate the leak, purge the cooling system, and ensure plant personnel and equipment safety. Subsequently work request, 96W000157, was initiated to repair the leak. Maintenance personnel confirmed that the leak was caused by the degradation of gasket material on the east side hydrogen cooler end bell flange.

During the gasket replacement, maintenance personnel demonstrated good personnel safety practices by requesting fire protection personnel to perform hydrogen detection testing as the cooler was disassembled. The gasket was replaced and the end bell was retorqued to design values. The torque values for the remaining end bells were also verified. Additionally, a post maintenance integrity check was performed when the system was returned to service. The inspector reviewed the work request, discussed the event with operations and maintenance personnel, and had no concerns with this maintenance activity.

3.3 Feedwater Level Transmitter Replacement

On February 1, 1996, steam generator "A" narrow range level loop feedwater transmitter, FW-L-519, spuriously failed low for approximately 20 seconds and recovered to normal operation. The transmitter was providing input to the controlling steam generator level channel at the time of the failure. Operators promptly swapped controlling channels and work request, 96W000196, was initiated to troubleshoot the instrument channel.

The troubleshooting plan was well detailed. Trip avoidance requirements such as detailed pre-evolution briefings were included. The inspector reviewed portions of the instrument circuit troubleshooting in instrument cabinet, CAB-02. The troubleshooting identified a circuit card, NLP2, that was out of specification and was replaced. Unrelated to this maintenance effort, control room operators were preparing to reduce reactor power in anticipation of removing the main generator from service to facilitate work on the stator water cooling system. The operators stopped the transmitter maintenance activity following replacement of the circuit card to ensure no high risk maintenance was ongoing during the power reduction. After the reactor was stabilized at a reduced power level, a second briefing was conducted for the containment entry to replace the transmitter. The inspector observed the briefing, which was thorough and attended by all involved disciplines. The transmitter was replaced and calibrated in accordance with station procedures and the instrument was returned to service. The inspector reviewed plant computer traces of the instrument failure, as well as, the completed work package which was determined to be well developed with good documentation of in-process work activities. The inspector had no questions regarding this activity.

4.0 ENGINEERING (71707,37551,92903,40500)

4.1 (Closed) UNR 50-443/95-04-01, Failed Emergency Diesel Generator TurboCharger Housing Bolts

Previously, the licensee identification of failed emergency diesel generator (EDG) turbocharger housing bolts was documented in NRC Inspection Report Nos. 50-443/95-04, section 3.2 ("B" EDG, 2 bolts) and 50-443/95-08, section 3.3 ("A" EDG, 1 bolt). Licensee technical support personnel performed an evaluation that concluded the EDGs remained operable with the broken bolts in place. The licensee, with support from an independent material properties laboratory, concluded the bolts failures were most likely attributable to loads applied to the specific bolt locations by the bolt torque sequences and torque initiation points. Metallurgical analysis had confirmed that the bolts were of the specified chemical composition and various destructive testing methodologies confirmed consistent material properties (for Grade 12.9). The corrective action recommended replacement of the existing bolts with upgraded bolts of higher tensile strength.

Minor modification, MMOD 95-0532, was developed to replace the bolts with new M16x2.0 bolts of 4340, 17-4Ph, or equivalent material. Installation of the bolting modifications was governed by work requests 95W002029 for the "A" EDG and 95W002031 for the "B" EDG. The work requests established torque values and passes, as well as torque initiation points. Additional controls were established, with good limitations and controls, to maintain the EDGs in an operable status throughout the bolt modifications.

The modifications were implemented in early November 1995, just prior to the refueling outage. The inspector reviewed the modification and the completed work requests. Additionally, the inspector discussed the work activities with the responsible system engineer. The inspector concluded the licensee disposition of the failed turbocharger bolts was appropriate. Causal analysis for the bolt failures was augmented by an independent material properties laboratory. The upgraded bolt modification was well developed and the work requests were properly implemented. The inspector had no further questions. This item is closed.

5.0 PLANT SUPPORT (71707,71750)

5.1 Radiological Controls

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

5.2 Security

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

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (92700)

6.1 Licensee Event Report Review

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

6.1.1 LER 95-003-001

LER 95-003-001, "Inadequate Area Temperature Monitoring Surveillances," dated October 30, 1995, supplemented the initial event report (LER 95-001, dated July 31, 1995) regarding the June 30, 1995 discovery that the primary component cooling water (PCCW) pump area temperature monitoring was not being monitored in accordance with Technical Specification requirements. The initial LER was documented in NRC Inspection Report No. 50-443/95-08, section 6.1.3.

The licensee recalibrated the PCCW area temperature switch (PAH-TISH-5397) and reset the alarm setpoint to 99.5 degrees Fahrenheit, as established in the standard instrument schedule (SIS). The immediate root cause of the incorrect alarm setpoint identified that instrumentation and controls (I&C) personnel improperly referenced the instrument calibration folder for the temperature switch which is an unapproved document rather than the SIS. As a result of this event, the licensee performed a broad-based review of the area temperature monitoring requirements. Several additional discrepancies were identified during the review that indicated temperature monitoring program weaknesses, but did not impact Technical Specification (TS) operability requirements.

Specifically, an inconsistency was identified between the TS Bases statement B.3.7.10, which indicates the associated TS TABLE 3.7-3 includes an allowance of plus or minus 4.5 degrees Fahrenheit for instrument error, and actual instrument setpoints relative to the Service Environment Chart (SEC). The SEC establishes calculated maximum hypothetical area temperatures for a given set of credible scenarios. The TS TABLE 3.7-3 area temperature values as listed equal the SEC calculated values, without accounting for the TS BASES described 4.5 degree instrument error tolerance. The licensee reviewed the as-left

alarm setpoints from the most recent calibration for each area TS TABLE 3.7-3 temperature switch and concluded the setpoints were less than the stated TS value. Additionally, the licensee review identified inconsistent tolerances for the switches had been established in the repetitive task sheets that implement the switch calibrations. Ten temperature switches had tolerances established that could have allowed as left settings to exceed the TS limits. The review discussed above had previously confirmed that the as-left settings were within TS requirements. Further, the licensee review of calibration records indicated that four temperature elements had no record of having been calibrated. The temperatures in these areas were recorded locally until the four elements were calibrated. The calibrations indicated the elements were less than 3.5 degrees Fahrenheit out of calibration. Finally, the licensee review identified that a functional test of the alarm function had been performed on one temperature switch rather than the required calibration, which ensures element calibration in addition to the alarm function. The switch was calibrated with acceptable results.

The inspector discussed the supplemental event report with cognizant licensing and engineering personnel. The additional information, root cause, and corrective actions described in the LER fulfilled the 10 CFR 50.73 reporting criteria. The licensee displayed good causal analysis awareness by performing a broad-based review of the area temperature monitoring program. Although several programmatic weaknesses were identified, TS requirements were not impacted. The corrective actions were appropriate to resolve the contributing factors to this event review. The inspector verified that the stated corrective actions were accomplished as described. The inspector had no further questions.

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

7.1 Routine Meetings

At periodic intervals during this inspection, meetings were held with senior plant management to discuss licensee activities and areas of concern to the inspectors. At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting on March 05, 1996, summarizing the preliminary findings of this inspection. No proprietary information was identified as being included in the report.

7.2 Other NRC Activities

During the week of January 9-11, 1996, a Region I Senior Radiation Specialist conducted a routine safety inspection of the radioactive liquid and gaseous effluent control programs. The results of the inspection are included as Attachment 1 to this report.

During the week of February 5-10, 1996, the first week of a two consecutive week onsite Independent Performance Assessment Process Team Inspection was performed. Results of the inspection will be documented in NRC Inspection Report No. 50 443/96-80.

NRC Inspection Report No. 50-443/96-01

ATTACHMENT 1

Radioactive Liquid and Gaseous Effluent Controls

Jason C. Jang
January 4 and January 9-11, 1996

(Procedure 84750)

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

DOCKET/REPORT NO.: 50-443/96-01
LICENSEE: North Atlantic Energy Service Corporation (NAESCO)
Seabrook, New Hampshire
FACILITY: Seabrook Station
DATES: January 4, 1996 and January 9 - 11, 1996

INSPECTOR:

Jason C. Jang
Jason Jang, Sr. Radiation Specialist
Radiation Safety Branch
Division of Reactor Safety

1-25-96
Date

APPROVED BY:

James D. Wolfe for
John R. White, Chief
Radiation Safety Branch
Division of Reactor Safety

1-25-96
Date

AREAS INSPECTED: Announced safety inspection of the radioactive liquid and gaseous effluent control programs including: management controls, audits, air cleaning systems, calibration of effluent radiation monitoring systems, and implementation of the Offsite Dose Calculation Manual (ODCM) and the above programs.

REPORT DETAILS FOR SEABROOK FEEDER REPORT NO. 96-01

1.0 PURPOSE

The purpose of this inspection was to review the licensee's ability to control and quantify effluent radioactive liquids, gases, and particulates during normal and emergency operations.

2.0 MANAGEMENT CONTROLS

2.1 Program Changes

The inspector reviewed the organization and administration of the radioactive liquid and gaseous effluent control programs for changes made since the last inspection, conducted in March 1994. The inspector determined that there were no changes. The Chemistry Department has primary responsibility for conducting the radioactive liquid and gaseous effluent control programs. The System Engineering, Operations, Radwaste Operations, and Instrumentation and Controls (I&C) Departments also have responsibilities to support effluent control programs, such as air cleaning systems, radwaste discharges, and radiation monitoring system calibrations (radiological and electronic calibrations).

2.2 Quality Assurance (QA) Audits

The inspector reviewed the 1994 QA audit report (Report No. 94-A09-03) and the 1995 QA audit report (Report No. 95-A08-01). These audits were conducted by Nuclear Safety Assessment (NSA) personnel and covered the radioactive liquid and gaseous effluent control programs. The inspector noted that the audits were conducted by members of NSA with assistance from other technical personnel. The 1994 audit team identified two findings and one observation. The 1995 audit team identified four findings and one observation. The inspector determined that these findings and observations were not safety significant, but were intended for the enhancement of the effluent control programs. The corrective actions for 1994 audit findings were completed in a timely manner. The responses to the 1995 findings were very good, and corrective actions are expected to be completed in the latter part of 1996. The inspector noted that the scope and technical depth of the audits were very good in assessing the radioactive liquid and gaseous effluent control programs. The inspector had no further questions in this area.

2.3 Review of Annual Radioactive Effluent Report

The inspector reviewed the 1994 annual radioactive effluent release report. The report provided data indicating total released radioactivity for liquid and gaseous effluents. This annual report also summarized the assessment of the projected maximum individual and population doses, resulting from routine radioactive airborne and liquid effluents. The inspector determined that there were no obvious anomalous measurements, omissions or trends in the report.

Projected doses to the public, resulting from radioactive liquid and gaseous effluents from the site during 1994, were acceptably calculated and reported in the 1994 Annual Radioactive Effluent Releases Report by the licensee. The projected doses are well below the TS limits as shown in the following table.

Radioactive Liquid Effluents	Dose	TS Limit	% of TS
Whole Body dose	7.6E-4 mrem	3 mrem	2.5E-2
Organ Dose	3.1E-3 mrem	10 mrem	3.1E-2
Iodine, H-3, and Particulates			
Organ Dose	1.4E-3 mrem	15 mrem	9.3E-3
Noble Gases			
Beta Air Dose	3.1E-5 mrad	20 mrad	1.6E-4
Gamma Air Dose	8.3E-5 mrad	10 mrad	8.3E-4

3.0 REVIEW OF OFFSITE DOSE CALCULATION MANUAL (ODCM)

The inspector reviewed the licensee's ODCM, Revision 15, effective January 5, 1995. The ODCM contained the following parts and its sections:

Part A:

- 1.0 Introduction;
- 2.0 Responsibilities for Part A;
- 3.0 Liquid Effluent Sampling and Analysis Program;
- 4.0 Gaseous Effluent Sampling and Analysis Program; and
- 5.0 Radiological Environmental Monitoring.

Part B:

- 1.0 Introduction/Responsibilities for Part B;
- 2.0 Method to Calculate Offsite Liquid Concentrations;
- 3.0 Offsite Dose Calculation Methods;
- 4.0 Setpoint Calculation Methods for Liquid and Gaseous Effluents;
- 5.0 Bases for Dose Calculation Methods; and
- 6.0 Bases for Liquid and Gaseous Effluent Monitors Setpoints.

The ODCM provided the sampling and analysis programs used for quantifying radioactive liquid and gaseous effluent concentrations, and for calculating projected doses to the public. All necessary parameters, such as effluent radiation monitor setpoint calculation methodologies and dose factors, were listed in the ODCM.

Based on the above review, the inspector determined that the licensee's ODCM contained all necessary information and instruction to acceptably implement and maintain the radioactive liquid and gaseous effluent control programs and the Radiological Environmental Monitoring Program.

4.0 RADIOACTIVE LIQUID AND GASEOUS EFFLUENT CONTROL PROGRAMS

Inspection of this area consisted of: (1) physical walkdown of facilities and equipment, including the control room; (2) review of the following selected licensee's procedures; and (3) review of selected radioactive liquid and gaseous discharge permits with respect to TS and the ODCM requirements.

The procedures reviewed included:

- o CX 0917-01, Liquid Effluent Release
- o CS 0917-02, Gaseous Effluent Release
- o CS 0917-03, Unmonitored Plant Release

During the review of the above radioactive liquid and gaseous effluent procedures, the inspector noted that these procedures were upgraded. These upgraded procedures were more detailed and easier to follow than previous versions. The inspector also noted that these procedures were written to reflect the ODCM requirements. The inspector noted that all effluent radiation monitoring systems (RMS) were operable at the time of this inspection. The inspector also determined that the reviewed discharge permits were complete, and met the ODCM requirements for sampling and analyses at the frequencies and lower limits of detection established in the ODCM.

During discussion with the Chemistry Department staff, the inspector noted that the responsible individuals had maintained and continually enhanced their excellent knowledge in the areas of: (1) radioactive liquid and gaseous effluent controls; (2) Effluent/Process Radiation Monitoring Systems (RMS); (3) the application of procedures designed to protect the public health and safety, and the environment; and (4) the ODCM requirements.

Based on the above reviews/discussions, the inspector determined that the licensee maintained excellent radioactive liquid and gaseous effluent control programs.

5.0 CALIBRATION OF EFFLUENT/PROCESS RADIATION MONITORING SYSTEMS (RMS)

The inspector reviewed the most recent calibration results for the following effluent RMS to determine the implementation of the Technical Specification (TS) and ODCM requirements.

- o Waste Liquid Test Tanks Radiation Monitor (R-6509)
- o Steam Generator Blowdown Flash Tank Radiation Monitor (R-6519)
- o Turbine Building Sump Pump Radiation Monitor (R-6521)
- o Primary Component Cooling Water Radiation Monitors (R-6515 & 6516)
- o Containment Purge Radiation Monitors (R-6527 A & B)
- o Plant Vent Wide Range Noble Gas Monitor (R-6528)
- o Condenser Air Evacuators Discharge Monitor (R-6505)

- o Waste Gas Compressor Inlet Radiation Monitor (R-6503)
- o Waste Gas Compressor Discharge Radiation Monitor (R-6504)

The I&C Department had the responsibility of performing electronic and radiological calibrations for the above radiation monitors. All reviewed calibration results were within the licensee's acceptance criteria.

During the review of the above RMS calibration results, the inspector independently verified several calibration results, including linearity tests and conversion factors. The inspector had no further questions in this area. Based on the above reviews, the inspector determined that the licensee implemented a very good calibration program.

6.0 AIR CLEANING SYSTEMS

The inspector reviewed the licensee's most recent surveillance test results to determine the implementation of TS requirements for the following air cleaning systems:

- o Containment Purge Exhaust System
- o Primary Auxiliary Building Exhaust System
- o Fuel Storage Building Exhaust System (Trains A & B)

The inspector reviewed the following surveillance test results:

- o Visual Inspection
- o In-Place HEPA Leak Tests
- o In-Place Charcoal Leak Tests
- o Air Capacity Tests
- o Pressure Drop Tests
- o Laboratory Tests for the Iodine Collection Efficiencies

All reviewed test results were within the licensee's TS acceptance criteria. During the review of the above test results, the inspector noted that the responsible individual had very good knowledge, not only about the TS requirements, but also about implementing TS correctly with regard to the technical bases. The inspector had no further questions in this area.