January 31, 2005

Christopher M. Crane
President and Chief Executive Officer
AmerGen Energy Company, LLC
4300 Winfield Road
5th Floor
Warrenville, IL 60555

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000219/2004005

Dear Mr. Crane:

On December 31, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Oyster Creek Generating Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 13, 2005, with Mr. C. N. Swenson and other members of your staff.

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

Based on the results of this inspection, two self-revealing findings were identified as having very low safety significance (Green). Both findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, 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, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Oyster Creek.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

We appreciate your cooperation. Please contact me at 610-337-5234 if you have any questions regarding this letter.

Sincerely,

/RA/

Peter W. Eselgroth, Chief
Projects Branch 7
Division of Reactor Projects

Docket No. 50-219
License No. DPR-16

Enclosure: Inspection Report 05000219/2004005
w/Attachment: Supplemental Information

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Plant Manager, Oyster Creek Generating Station, AmerGen
Regulatory Assurance Manager Oyster Creek, AmerGen
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-219
License No.:	DPR-16
Report No.:	05000219/2004005
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	October 1, 2004 - December 31, 2004
Inspectors:	Robert Summers, Senior Resident Inspector Jeff Herrera, Resident Inspector Joseph Furia, Senior Health Physicist Ronald Nimitz, Senior Health Physicist Fred Jaxheimer, Resident Inspector, Susquehanna Richard Barkley, Senior Project Engineer
Approved By:	Peter W. Eselgroth, Chief Projects Branch 7 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000219/2004005; 10/01/04 -12/31/04; Oyster Creek Generating Station; Event Follow-up and Access Control to Radiologically Significant Areas.

This report covers a thirteen-week period of inspection by resident inspectors and announced inspections by regional senior health physics inspectors, a senior project engineer, and a resident inspector from the Susquehanna Station. Two Green findings involving non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3 dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- **Green**. A self-revealing event involving an inadvertent loss of the containment isolation function resulted in a Green finding and non-cited violation for not establishing and maintaining appropriate procedural requirements for the operation of the containment vent isolation valves, as prescribed by Technical Specification 6.8.1 and the Oyster Creek Operational Quality Assurance Plan.

  This finding is more than minor because it affects the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events and the related attributes of configuration control and procedure quality. In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 screening. The finding represented a degradation in the barrier integrity cornerstone objective, in that both drywell vent containment isolation valves were open and the primary containment isolation logic was bypassed for each valve, causing a loss of safety function for the containment barrier. This required an evaluation per Manual Chapter 0609, Appendix H, Containment Integrity Significance Determination Process for the containment barrier being degraded due to an actual open pathway in the physical integrity of reactor containment that can potentially increase Large Early Release Frequency (LERF) without affecting Core Damage Frequency (CDF), a Type B finding. Appendix H, Section 6, Table 6.1, Phase 1 Screening for Type B findings for BWR Mark 1 Containment Types required a Phase 2 assessment since the finding involved an open path through the containment vent system. Appendix H, Section 6, Table 6.2, Phase 2 Risk Significance for BWR Mark 1 Containment Types screened to Green because: although the finding resulted in the possible leakage rate from the drywell to the environment of >100% containment volume/day via the open pathway, the fault exposure time was very small (i.e., less than 2 hours). A cross-cutting aspect of human performance was identified in that: (1) the procedure development involved a human error in identifying the wrong switch listed in step 6.51.6, and (2) that the initial questioning of this action by an operator did not result in preventing the resultant loss of containment integrity.

Cornerstone: Occupational Radiation Safety

- **Green.** A self-revealing event involving an underestimation of airborne radioactivity for in-valve grinding work resulted in a finding and non-cited violation of 10 CFR 20.1501, in that AmerGen did not provide reasonable surveys to evaluate the magnitude of airborne radioactivity concentrations, and potential radiological hazards present, during work on main steam isolation valve (MSIV) NSO4A on September 18, 2004.

  This finding is more than minor in that it is associated with the program and processes for exposure control and monitoring attribute of the Radiation Safety Cornerstone attributes and did affect the objective of the Cornerstone. Specifically, analyses of airborne radioactivity sample concentrations for in-valve grinding work significantly underestimated airborne radioactivity due to incorrect assessment of radionuclides, relative to applicable exposure limits, and incorrect analysis of alpha airborne radioactivity concentrations. The finding was determined to be of very low risk significance using NRC Manual Chapter 0609, Appendix C, in that: 1) it did not involve an ALARA finding, 2) it did not involve an overexposure, 3) there was no substantial potential for an overexposure, and 4) the ability to assess dose was not compromised. AmerGen temporarily suspended work, implemented additional radiological controls, and modified sample analyses. No significant personnel dose was identified. (Section 2OS1)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Oyster Creek began the integrated inspection period at full power. On November 2, 2004, Oyster Creek began a maintenance refueling outage. The outage ended on November 22, 2004, lasting approximately 20 days in duration, at which time Oyster Creek returned to full power and remained at full power through December 31, 2004.

1. REACTOR SAFETY

Cornerstones: Initiating Events/Mitigating Systems/Barrier Integrity

1R01 Adverse Weather Protection (IP 71111.01 - 2 Samples)

a. Inspection Scope

The inspectors reviewed the licensee's seasonal readiness preparations to verify that safety-related equipment would remain functional when challenged by winter weather conditions. The inspectors reviewed the licensee's seasonal readiness procedure (OP-AA-108-109, Seasonal Readiness, Revision 1), seasonal check lists, and performed walk downs to verify that the safety-related equipment would remain functional during adverse weather conditions. The inspectors evaluated the condition of the Emergency Service Water System, Service Water System, Electrical Switchyard and fire diesels prior to the onset of winter weather conditions.

The inspectors also reviewed a sample of deficiencies associated with AmerGen's winter readiness action item list to verify that problems were entered into the corrective action program and appropriately addressed for resolution in a timely manner.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (IP 71111.04)

a. Inspection Scope

Partial System Walkdown. (71111.04Q - 3 Samples)

The inspectors performed three partial system walkdowns during this inspection period. To evaluate the operability of the selected systems, the inspectors checked for a correct valve lineup by comparing positions of valves with system drawings, as well as examining overall system material condition. The results of inspections as well as minor deficient equipment conditions identified by the inspectors, were discussed with the appropriate system engineers.

This inspection activity represented three samples of the following systems:

- Emergency Diesel Generator (EDG) #2 during EDG #1 six month inspection during week of October 27, 2004;
- Temporary EDG fuel oil tank and system during fuel oil tank cleaning conducted the week of November 4, 2004; and,
- Service Water System #2 during System #1 maintenance activities during the week of December 10, 2004.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (IP 71111.05Q - 7 Samples)

a. Inspection Scope

The inspectors walked down accessible portions of the seven fire zones listed below due to the potential to impact mitigating systems. Plant walkdowns included observations of combustible material control, fire detection and suppression equipment availability, and compensatory measures. As a part of the inspection, the inspectors had discussions with fire protection personnel, and reviewed procedure 333, "Plant Fire Protection System," and the Oyster Creek Fire Hazards Analysis Report to verify that the fire program was implemented in accordance with all conditions stated in the facility license.

C YARD, "Office Building Roof, Turbine Building Roof, Outside Areas within Power Block"

C DG-FA-15&17, "No. 1 & 2 Emergency Diesel Generator Room"

C RB-FZ-1H, "Trunnion Room, Elev. 23'-6"

C RB-FA-2, "Drywell and Torus"

C TB-FZ-11E, "Condenser Bay, Elev. 3'-6"

C OR-FA-19, "Old Radwaste Building"

C RB-FZ-1A, "119' Elevation"

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (IP 71111.06 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the Oyster Creek Individual Plant Examination of External Events, Section 5.2, “External Floods,” TS and the UFSAR, Section 2.4.2 concerning flood design considerations. The inspectors reviewed the procedure for Response to Abnormal Intake Level, 2000-ABN-3200.32, Rev. 19 and a walkdown of the following outside buildings was performed:

- Fire Diesel Pump Room
- Emergency Diesel Generator Rooms
- Intake Structure
- Standby Gas Treatment and Off-Gas Building

The inspectors also reviewed flood protection analyses for and features in the Reactor and Turbine buildings as well as the system health report for the reactor floor and equipment drains system. CAPs O2002-1284, O2004-0387 and O2004-1838 on this system were also reviewed and a tour conducted of three of the four Reactor Building corner rooms. Discrepant equipment conditions were noted and discussed with the responsible system engineer.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08 - 1 Sample)

a. Inspection Scope

The inspectors observed in-process non-destructive examination (NDE) activities and reviewed documentation of NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in an increase in risk of core damage. The direct observations and documentation reviews were performed to verify activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements. The inspectors reviewed a sample of inspection reports initiated to document the performance and record results of in-service inspection (ISI) examinations completed during the previous refueling outage. Also, the inspectors evaluated the licensee’s effectiveness in resolving relevant indications identified during ISI activities.

The inspectors observed in-process manual ultrasonic testing (UT) of the three reactor closure head penetration nozzle to pipe welds. Inspectors also reviewed data collection and data analysis of the automated N1C, N1D, N5A and N6B reactor vessel nozzles weld examinations, to verify the effectiveness of the licensee's program for monitoring degradation of risk significant piping systems, structures and components. The inspectors evaluated the activities for compliance with the requirements of ASME Section XI of the ASME Boiler and Pressure Vessel Code, the EPRI performance demonstration initiative (PDI) standards and station procedures.

The inspectors reviewed two ASME Section XI code repair/replacement activities from the 1R20 refueling outage. Specifically, the inspectors reviewed liquid penetrant (PT) records associated with repair activities on the disc and wedges for the isolation condenser condensate return valve, V-14-35. In addition, the inspectors witnessed PT examinations and reviewed data records associated with the replacement of the emergency service water (ESW) keepfill line pipe with corrosion resistant material. Inspectors verified that these activities were in accordance with the applicable construction code, ASME Sections IX and XI code requirements, and station procedures.

The inspectors performed a review of a portion of the remote In-Vessel Visual Inspections (IVVI) of the control rod guide tube (CRGT) welds and a sample of IVVI video tape of reactor steam dryer structural welds. Inspectors conducted these visual examination samples to evaluate test equipment performance, examination technique, inspection environment (water clarity), and to assess the licensee's contractor oversight activities, including corrective action. The inspectors examined the licensee's evaluation and disposition for continued operation without repair for the non-conforming conditions identified during these activities. This included review of the associated condition reports and material non-conformance engineering evaluations performed for the two new recordable indications found during the VT-1 visual examination of the reactor steam dryer. Inspectors also walked down the reactor building closed cooling water (RBCCW) piping inside the primary containment to assess the overall health of this system and to compare reported corrosion and overall system health with that documented in NRC Special Inspection Report 50-220/03-003.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (IP 71111.11Q - 1 Sample)

a. Inspection Scope

This inspection activity represented one inspection sample. This inspection assessed the LORT provided to the SROs and the ROs and the evaluation conducted on the simulator on November 30, 2004. The inspectors assessed the proficiency of the operating crew and verified that the evaluations of the crew identified and addressed operator performance issues. The inspection activities were performed using NUREG-1021, Rev. 8, "Operator Licensing Examination Standards for Power Reactors," and

Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program."

The training included three scenarios and about four hours of testing/evaluation. The inspectors assessed the simulator crew's performance during each scenario. The inspectors also assessed the evaluator's assessment of the crew, to verify that operator performance issues were identified and appropriate remediation was conducted to address identified weaknesses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (IP 71111.12Q - 2 Samples)

a. Inspection Scope

The inspectors selected two samples for review. The inspectors reviewed AmerGen's implementation of the maintenance rule as described in Oyster Creek procedure ER-AA-310, "Implementation of the Maintenance Rule." The inspectors verified that the selected Systems, Structures and/or Components (SSCs) were properly classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed Action Requests (ARs), Corrective Action Program reports (CAPs), (a)(1) corrective action plans and routine preventive maintenance activities. The inspectors also discussed the current system performance, associated issues and concerns, and planned activities to improve performance with the system engineers. In addition, unavailability data was compared with control room log entries to verify accuracy of data and compliance with (a)(1) goals. AmerGen trending data was also reviewed. The two SSCs reviewed during the inspection period were as follows:

- Core Spray System
- Startup Transformers

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (IP 71111.13 - 5 Samples)

a. Inspection Scope

The inspectors evaluated five on-line risk work activities and verified that the licensee evaluated the risk associated with the inoperability of the system along with other ongoing maintenance work. In addition, the inspectors reviewed work schedules, recent corrective action documents, troubleshooting plans, repair and retest results, and control room logs to verify that other concurrent planned and emergent maintenance or surveillance activities did not adversely affect the plant risk already incurred with the out of service components. The inspectors assessed AmerGen's risk management actions during shift turnover meetings, control room tours, and plant walkdowns. The inspectors also used AmerGen's on-line risk monitor (ORAM Sentinel) to evaluate the risk associated with the plant configuration and to assess AmerGen's risk management. When appropriate, the inspectors verified compliance with Technical Specifications (TS). The following five activities were reviewed:

- "B" Isolation Condenser valve, V-14-35, failure to pass surveillance testing during week of October 7, 2004;
- Reactor Building Closed Cooling Water check valve repair work conducted during week of November 15, 2004;
- Electro-Matic Relief Valve(s) "D" and "E" work conducted during the week of November 18, 2004;
- Electro-Matic Relief Valve "C" inadvertent opening during hydrostatic testing during the week of November 19, 2004; and,
- Service Water System pump 1-1 preventative maintenance activities conducted during the week of December 10, 2004.

b. Findings

Introduction

An Unresolved Item was identified for an issue involving pressure locking of the condensate return isolation valve (V-14-35) revealed during special testing and analysis on October 8 and 14, 2004, that could adversely affect the safety function of the Isolation Condenser System.

Description

During the troubleshooting and accelerated surveillance testing of the "B" Train IC System Condensate Return Isolation Valve, on October 8 and 14, 2004, the valve failed to open as required due to pressure locking.

As a result of a failure to open of the "B" Train valve, a detailed test and evaluation of the cause of the opening failure was conducted. The testing concluded that the "B" Train valve failed to open due to pressure locking of the double disk gates in the closed seat position. The evaluation also revealed that a likely contributor to the October 2004 failure was a maintenance activity conducted on October 7, 2004, that included tightening the packing to reduce or eliminate observed stem leakage. This activity was not considered to be intrusive maintenance since it was done in a controlled manner to restore the packing to a previously tested condition. Troubleshooting/testing of the valve on October 8, 2004, revealed the pressure lock condition.

On October 8, 2004, the pressure lock condition was not initially understood, and an operability evaluation concluded that the valve was degraded but operable, subject to accelerated surveillance testing to monitor the degraded condition. A subsequent test failure on October 14, 2004, resulted in additional troubleshooting and engineering evaluation that concluded in determining the causes of both valve testing failures to be the result of pressure locking.

The licensee determined that the October 7, 2004, packing adjustment likely reduced the bonnet volume leakage and thereby contributed to the pressure locking condition. The actual initiating condition resulted from testing which cycled the normally closed valve to an open position for test purposes and then restored it to a normally closed position. This testing evolution introduced relatively cooler water into the valve bonnet volume from the upstream piping. When the valve was subsequently closed, the cooler water trapped in the bonnet was heated by a thermal plume from the water in the downstream piping connected to the RCS recirculation system piping. The heated water in the bonnet over-pressurized the valve internals and created excessive frictional forces for the disks against the closed seats, such that the motor operator stalled during a subsequent opening cycle and tripped the power supply on thermal overload.

The motor operator thermal trip device is normally bypassed during standby readiness operation for valve, V-14-35, but is not bypassed during normal testing configuration to prevent damage to the valve during testing. The redundant train valve, V-14-34, thermal trip device is not normally bypassed during standby readiness. (The reason for the difference in the application of the thermal overload trip devices is due to the relative location of the motor control centers with respect to the IC system piping and associated HELB impact analysis on IC System availability.) Based on interviews with licensee engineering, the inspector concluded that valve, V-14-35, would have failed to open had it been demanded to open in response to an actual transient condition, even with the thermal protective relay bypassed, as happened on October 8 and 14, 2004. However, the power supply breaker would have likely tripped on an over-current condition. Since the valve had developed maximum torque during the two tests in which the motor stalled during the opening cycle, it was not likely that opening forces would have been greater from a standby readiness condition. Weak link analysis of the valve and motor operator for the two events in October identified that the excessive forces experienced would not adversely affect the component. Based on discussions with the system and component engineers, the inspector determined that the weak link analysis would apply for the case when the valve motor protective thermal relay was bypassed. The conclusions of the weak link evaluation were confirmed by visual inspection of the valve actuator and its internals during the refueling outage in November 2004, which did not identify as a result of the pressure lock events any failed components.

As a result of the licensee's test and evaluation for the October opening failures, the licensee restored the valve packing to a relaxed condition, resulting in observable leakage from the bonnet area during valve cycling. Accelerated testing continued until the refueling outage in November 2004, when the valve internals were inspected for other damage or conditions that could have affected the valve performance. An independent engineering evaluation based on the visual examination of the valve internals concluded that bonnet over-pressurization was the cause of the October failures.

During the November refueling outage, the licensee implemented a modification to both IC System condensate return isolation valves to provide a pressure vent path through the downstream disk to the valve's discharge piping. This modification should preclude over-pressurization of the bonnet volume water and prevent future pressure locking of the valves.

The inspectors concluded that the pressure locking condition was of a transient nature that was self-correcting with time. This resulted in the failure mechanism not being easily detected by the licensee's surveillance test program. The unusual testing that revealed this condition in October 2004 was a result of an operability evaluation for the initially unexplained failure of the valve to open during a troubleshooting maintenance activity on October 8, 2004. The accelerated testing revealed that pressure locking resulted from a thermal transient involving entrapped fluid in the valve bonnet subsequent to cycling the valve during power operations. Normal operational leakage from the bonnet area subsequently would relieve the condition, returning the component to an functional state. Repeated successful quarterly surveillance test results over several years indicated that the pressure lock condition was never previously detected. However, in this case, the condition appeared to occur within hours after cycling the valve from conditions of normal operating pressure and temperature. The special test sequence was identical to the normal quarterly surveillance test sequence, except that the frequency of the test was increased from quarterly to daily.

The inspectors noted that the licensee's normal maintenance and testing schedule resulted in testing both trains of the IC System on the same day or sequential days. This contributed to the inspector's concern that pressure locking could occur in a common-mode manner and be undetected.

At the close of the inspection period this matter was still under review. The inspectors noted that the pressure locking condition was previously evaluated as part of an evaluation per NRC Generic Letter 95-07. That evaluation failed to include the above described mechanism as a possible cause for pressure locking the IC System condensate return isolation valves. However, the inspectors could not complete a review of this event without additional information from the licensee to determine: (1) the possible fault exposure time for the inoperable valve; (2) the use of manual operation of the valve handwheel and operator recovery credit for the IC function; (3) the possible affect on the redundant component, valve V-14-34; and (4) an understanding of the extent of condition for the inadequate evaluation per Generic Letter 95-07. This matter is unresolved pending review of the above listed issues. **(URI 05000219/200400501)**

1R14 Personnel Performance During Non-routine Plant Evolutions (IP 71111.14 - 4 Samples)

a. Inspection Scope

For the four sampled non-routine events, the inspectors reviewed operator logs and plant computer data to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures. The inspectors also reviewed relevent Corrective Action Program (CAP) documentation associated with the events, including the following CAPs and associated evaluations-CAP Nos.: O2004-3920 and O2004-3994. The following four non-routine activities were selected for review:

- Shutdown for 1R20 refueling outage on November 2, 2004;
- Startup from 1R20 refueling outage on November 20, 2004;
- Inadvertent opening of “C” EMRV during RPV hydrostatic pressure test on November 19, 2004; and,
- RPV water level transient and half scram during reactor startup on November 22, 2004.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (IP 71111.15 - 4 Samples)

a. Inspection Scope

The inspectors reviewed operability evaluations in order to verify that they were performed as required by Oyster Creek procedure LS-AA-105, “Operability Determinations.” The inspector assessed the accuracy of the evaluations, the use and control of compensatory measures if needed, and appropriate action if a component was determined to be inoperable. The inspectors verified that the technical specification limiting conditions for operation were properly addressed. The four selected samples are listed below:

- V-14-35 “B” Isolation condenser condensate return valve failure to stroke closed in the time allowed during week of October 20, 2004;
- V-14-34 “A” Isolation condenser condensate return valve during the week of October 20, 2004;
- Main Steam Isolation Valve, NSO4A, local leak rate testing failure during week of November 15, 2004; and,

- Reactor vessel head potential flaws discovered through non-destructive examination activities during 1R20 refueling outage.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (IP 71111.16 - 1 Sample)

a. Inspection Scope

The inspectors reviewed the operator work-around database and a sample of the associated corrective action items to identify conditions that could adversely affect the operability of mitigating systems or impact human reliability in responding to initiating events. The inspector reviewed the licensee's implementation of procedure OP-AA-102-103, "Operator Work-Around Program." The inspector also reviewed the status of the corrective actions described in CAP Nos. O2000-0463 and O2003-2125 which identified specific problem resolutions relating to the operator work-around for difficulty in manual operation of valve, V-20-83, the Fire Water System alternate supply valve to the Core Spray System.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (IP 71111.19 - 5 Samples)

a. Inspection Scope

Five samples were selected for review by the inspectors. The inspector reviewed and observed portions of post maintenance testing associated with the below listed five maintenance activities because of their function as mitigating systems and their potential role in increasing plant transient frequency. The inspectors reviewed the post maintenance test documents to verify that they were in accordance with the licensee's procedures and that the equipment was restored to an operable state. The following post maintenance test activities were selected for review:

C Emergency Diesel Generator No.1 post maintenance testing for 6 month surveillance on October 27, 2004;

C Isolation Condenser "B" condensate return valve, V-14-35, testing on October 20, 2004;

C Emergency Diesel Generator No. 2 Loss of Offsite Power testing on November 20, 2004;

- Main Steam Isolation Valve Local Leak Rate Testing conducted during refueling outage 1R20; and,

- Isolation Condenser “A” & “B” Condensate return valves, V-14-34 & 35, testing after completing modifications to valves on November 15, 2004.

b. Findings

No findings of significance were identified.

1R20 Refueling/Forced Outage Activity (IP 71111.20 - 1 Sample)

a. Inspection Scope

The inspectors reviewed and/or observed various risk significant activities associated with the refueling outage, 1R20, which began on November 2, 2004, and ended on November 22, 2004. These inspections included:

- Reviewed the overall outage schedule risk assessment.

- Observed portions of the reactor shutdown and cooldown evolutions.

- Reviewed availability and adequacy of reactor water level and temperature instrumentation during transient and shutdown conditions.

- Reviewed availability of protected equipment as specified by the daily shutdown risk assessment.

- Reviewed adequacy of contingency plans as specified by the shutdown risk assessment.

- Verified a sample of tagged out equipment were in the correct position as described by the associated tag.

- Toured spaces normally inaccessible during power operation including the Trunnion Room on November 4 and 5, the Drywell on November 4, 5, 10, 19 and 22, the Condenser bay on November 4 and 5, the torus room on November 15, and the Main Turbine Deck on November 16 and 19, 2004.

- Observed portions of refueling activities including: reactor disassembly, core fuel movement and reactor vessel pressure hydrostatic testing.

- Observed portions of the reactor startup including approach to criticality and reactor heat up.

- Verified required reactor vessel internal inspections were completed and that deficiencies identified by AmerGen were entered into their corrective action program.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (IP 71111.22 - 4 Samples)

a. Inspection Scope

The inspectors observed and reviewed four Surveillance Tests (ST) concentrating on verification of the adequacy of the test as required by technical specifications to demonstrate operability of the required system or component safety function. The inspector observed pre-test briefings and portions of the ST performance for procedure adherence, and verified that the resulting data associated with the ST met the requirements of the plant technical specifications and the UFSAR. The inspector also reviewed the results of past tests for the selected STs to verify that degraded or non-conforming conditions were identified and corrected, if needed. The inspector reviewed CAP Nos. O2004-3948 and O2004-3442 for problems identified during the System 2 Loss of Offsite Power Test and the Main Steam Isolation Valve Local Leak Rate Test. The following four activities were reviewed (including a Local Leak Rate Test during the refueling outage):

- “B” Isolation Condenser Operability Test per procedure 609.4.001 conducted on November 20, 2004;
- System 1 Loss of Offsite Power Test per procedure 636.2.001 conducted on November 3, 2004;
- System 2 Loss of Offsite Power Test per procedure 636.2.001 conducted on November 20, 2004; and,
- Main Steam Isolation Valve, NS04A, Local Leak Rate Test per procedure 665.5.003 conducted on November 4, 2004.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (IP 71111.23 - 1 Sample)

a. Inspection Scope

One sample was selected for review by the inspectors. The inspectors reviewed a Temporary Modification (TM) associated with the emergency diesel generator fuel oil storage tank temporary oil supply during fuel oil tank cleaning activities. The inspectors reviewed the associated implementing documents to verify the plant design basis and the system or component operability was maintained, which included CC-AA-112, “Temporary Configuration Changes,” Rev. 6. The TM allowed for continued operability of the emergency diesel generators while the normal fuel oil supply system was unavailable. The inspectors verified that the temporary modification was removed in accordance with station procedures.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (IP 71114.06 - 1 Sample)

a. Inspection Scope

The inspectors observed an emergency preparedness (EP) drill from the control room simulator, the technical support center, and the emergency operations facility on October 5, 2004. The inspectors evaluated the conduct of the drill and AmerGen’s performance related to emergency action level classifications, notifications, and protective action recommendations. The drill contained eight opportunities that contributed to the NRC Drill/Exercise Performance (DEP) performance indicator. The inspectors also reviewed several condition reports (CAP Nos. O2004-2997, O2004-2998, O2004-2999, O2004-3000, O2004-3001, O2004-3002, O2004-3003, and 2004-3004) associated with EP areas for improvement identified during the drill.

The inspectors reviewed the following documents:

- Oyster Creek EP Drill October 5, 2004, Scenario, Rev. 0
- Oyster Creek October 5, 2004, Off-Year Exercise Findings and Observation Report

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 19 Samples)

a. Inspection Scope

The inspector reviewed selected activities, and associated documentation, in the below listed areas. The evaluation of AmerGen's performance was against criteria contained in 10 CFR 20, applicable Technical Specifications, and station procedures.

Inspection Planning - Performance Indicators

The inspector reviewed performance indicators (PIs) for the Occupational Exposure Cornerstone. The inspector also discussed and reviewed current performance, relative to the indicators, with cognizant AmerGen personnel. The inspector selectively reviewed external and internal occupational dose assessments relative to applicable performance indicators (PIs). (See Section 40A1.)

Plant Walkdowns, RWP Reviews, and Jobs in Progress Reviews

The inspector identified exposure significant work areas during station tours and walked down selected radiological controlled areas and made independent radiation surveys. The inspector reviewed housekeeping, material conditions, posting, barricading, and access controls to determine if radiological controls were acceptable. The inspector determined if prescribed RWP, procedure, and engineering controls were in place. The inspector conducted selective independent radiation surveys to evaluate ambient conditions.

The inspector reviewed previously completed out-board MSIV work conducted in September 2004. The review included evaluation of the adequacy of applied radiological controls including radiation work permits, procedure adherence, radiological surveys, job coverage, system breach surveys, air sampling and contamination controls.

The inspector toured outage work areas and reviewed ongoing radiologically significant work in the drywell, reactor building, turbine building, and refueling floor. The inspector conducted direct observation and review of ongoing refueling outage work activities. The inspector reviewed control rod drive replacement, inboard and outboard main steam line valve work, and reactor water clean-up work activities. The inspector also selectively reviewed other refueling outage work including turbine component sand blasting, control and stop valve work, reactor disassembly, and resin transfer activities. The inspector reviewed implementation of Technical Specification High Radiation Area controls, reviewed the adequacy of electronic dosimeter setpoints, and verified workers' new actions to take upon alarms. The inspector evaluated the adequacy of personnel monitoring in areas of potential dose rate gradients. The inspector conducted post-outage station tours and walked-down selected radiological controlled areas to evaluate housekeeping, material conditions, posting, barricading, and access controls, as appropriate.

The inspector reviewed and discussed internal dose assessments for 2004 (as of the time of the inspection), to identify apparent occupational internal doses greater than 50 millirem committed effective dose equivalent (CEDE). The review also included the adequacy of evaluation of selected dose assessments, as appropriate, and included selected review of the program for evaluation of potential intakes associated with hard-to-detect radionuclides (e.g., transuranics).

The inspector reviewed physical and programmatic controls for highly activated or contaminated (non-fuel) stored within spent fuel or other storage pools, as applicable.

Problem Identification and Resolution

The inspector selectively reviewed self-assessments and audits since the previous inspection to determine if identified problems were entered into the corrective action program for resolution. The inspector evaluated the database for repetitive deficiencies or significant individual deficiencies to determine if self-assessment activities were identifying and addressing the deficiencies. The review also included evaluation of data to determine if any problems involved performance indicator (PI) events with dose rates greater that 25 R/hr at 30 centimeters, greater than 500 R/hr at 1 meter or unintended exposures greater than 100 millirem total effective dose equivalent (TEDE), 5 rem shallow dose equivalent (SDE), or 1.5 rem lens dose equivalent (LDE).

The review also included a review of problem reports since the last inspection which involved potential radiation worker or radiation protection personnel errors to determine if there was an observable pattern traceable to a similar cause. The inspector reviewed radiological problem reports from the recent refueling outage. The review included an evaluation of corrective actions, as appropriate. (See Section 4OA2)

High Risk Significant, High Dose Rate HRA and VHRA Controls

The inspector discussed procedure changes for High Radiation Area Access controls since the last inspection with the Radiation Protection Manager and selected supervisors to determine if the changes resulted in a reduction in the effectiveness and level of worker protection. During station tours, the inspector selectively reviewed implementation of High and Very High Radiation Area controls and discussed High Radiation Area controls with in-field, lead radiological controls personnel. Posting, barricading, and locking of High Radiation Areas was reviewed.

Radiation Worker Performance

During station tours, the inspector observed radiation worker performance with respect to stated radiation protection work requirements. The inspector selectively questioned workers to determine if they were aware of the significant radiological conditions in their workplace, there were RWP controls/limits in place, and their performance took into consideration the level of radiological hazards present.

The inspector reviewed all radiological problem reports since the last inspection to identify radiation worker errors traceable to a similar cause. Corrective actions were reviewed, as appropriate.

Radiation Protection Technician Proficiency

The inspector observed radiation protection technician performance with respect to radiation protection work requirements to determine if they aware of the radiological conditions in their workplace and the RWP controls/limits, and if their performance was consistent with expectations for potential radiological hazards present.

The inspector reviewed all radiological problem reports since the last inspection to identify radiation protection technician errors traceable to a similar cause. Corrective actions were reviewed, as appropriate.

b. Findings

Introduction:

AmerGen's radiological survey program did not provide reasonable surveys to evaluate the magnitude of airborne radioactivity concentrations, and potential radiological hazards present, during work on main steam isolation valve (MSIV) NSO4A on September 18, 2004. Specifically, analyses of airborne radioactivity sample concentrations for in-valve grinding work underestimated airborne radioactivity due to incorrect assessment of radionuclides relative to applicable exposure limits, and incorrect analysis of alpha airborne radioactivity concentrations. This is self-revealing violation of 10 CFR 20.1501.

Description:

On September 18, 2004, AmerGen conducted in-valve grinding activities on main steam line isolation valve (MSIV) NSO4A. Breathing zone air samplers were provided to workers conducting the work. The air sample for one worker ran from 3:50 a.m. to 4:30 a.m., was analyzed at about 4:45 a.m. that day, and indicated 72.1 times the applicable derived air concentration (DAC)[1] (based on scaled gross beta analysis) and 8 DAC alpha based on alpha counting (i.e, a total concentration of 80.1 DAC). Work was stopped, additional radiological controls were provided such as: replacement of ventilation hoses, decontamination inside the valve, and use of respirators with a higher

[1]10 CFR 20.1003 defines the Derived air concentration (DAC), in part, as the concentration of a given radionuclide in air, which, if breathed by the reference man for a working year of 2,000 hours under condition of light work, results in an intake of one annual limit of intake (ALI).

protection factor. Welding work subsequently commenced in the valve starting at about 9:50 a.m. that day.

The inspector's review identified that the airborne radioactivity sample analyses for the in-valve grinding did not provide a reasonable estimate of actual airborne radioactivity concentrations encountered. Specifically, the gross beta analysis was not properly assessed (scaled) relative to the radionuclides expected to be present and the proper 10 CFR 20 DAC value was not used for purposes of assessment. The gross beta analysis of the air sample indicated a scaled airborne radioactivity concentration of 72.1 DAC. This sample was re-analyzed by gamma spectroscopy about eight hours later and determined to indicate an airborne radioactivity concentration of 152 DAC using different scaling factors and allowable DAC value. Subsequent review indicated an incorrect DAC values was used for the gross beta analysis.

The inspector's review also identified that calculated alpha airborne radioactivity concentrations for the air sample were incorrect. The initial alpha count was conducted at about 4:45 a.m. on September 18, 2004, and indicated an alpha airborne radioactivity concentration of 8 DAC. The sample was recounted, for alpha airborne radioactivity, about seven hours later and indicated an airborne alpha radioactivity concentration of 69.7 DAC. AmerGen determined the initial sample results had been incorrectly determined.

Based on the above, the initial results for the air sample, were underestimated by a factor of 2.75 (i.e., 80.1 DAC versus 221 DAC). The inspector concluded that use of incorrect scaling factors, including DAC values, and incorrect sample counting, resulted in incorrect characterization of potential airborne radioactivity hazards associated with the in-valve work activities. This situation was not recognized until subsequent reanalysis of the initial breathing zone airborne radioactivity sample about 7 hours later.

<u>Analysis:</u>

The inadequate survey (evaluation) of potential airborne radioactivity hazards is a performance deficiency in that a requirement to conduct radiological evaluations, as specified in 10 CFR 20.1501, was not met by AmerGen which was reasonably within its ability to foresee and correct, and which should have been prevented. Such assessment is important to ensure application of proper radiological controls for worker protection. In addition, proper assessment of potential airborne radioactivity hazards is important relative to evaluation of the adequacy of applied controls to support resumption of work activities and to evaluate potential occupational exposures.

The finding is not subject to traditional enforcement in that the finding did not have any actual safety consequence, did not have the potential for impacting the NRC's ability to perform its regulatory function, and there were no willful aspects.

The finding was greater than minor in that it was associated with one of the Radiation Safety/Occupational Radiation Safety Cornerstone attributes (program and processes for exposure control and monitoring) and did affect the objective of the Cornerstone.

Specifically, AmerGen did not conduct reasonable assessment of airborne radioactivity concentrations during the in-valve grinding activity due to incorrect sample assessment, including sample counting errors. The finding was evaluated against criteria specified in NRC Manual Chapter 0609, Appendix C, and determined to be of very low safety significance, in that: 1) it did not involve an ALARA finding, 2) it did not involve an overexposure, 3) there was no substantial potential of an overexposure, and 4) the ability to assess dose was not compromised. AmerGen temporarily suspended work in accordance with its program controls when elevated airborne radioactivity was detected, ventilation hoses were changed out, and decontamination inside the valve was performed. In addition, involved personnel dose was assessed by whole body counting and additional air sampling was conducted. No significant personnel dose was identified. The sample analysis methodology was subsequently modified.

Enforcement:

10 CFR 20.1501 requires that reasonable radiological surveys be conducted to evaluate potential radiological hazards including airborne radioactivity hazards. Contrary to this requirement, due to sample assessment errors, including sample counting errors, AmerGen did not conduct a reasonable assessment of potential airborne radioactivity hazards encountered during grinding work in MSIV NSO4A on September 18, 2004. This is a violation of 10 CFR20.1501. Because this finding was of very low safety significance and AmerGen entered this finding into its corrective action program, this violation is being treated as a Non-Cited Violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy NUREG-1600. **(NCV 05000219/2004005-02)**

AmerGen adjusted its DAC and scaling values used for assessment of airborne radioactivity samples. A corrective action document was written. (CAP No. O2004-2635)

2OS2 ALARA Planning and Controls (71121.02 - 10 Samples)

b. Inspection Scope

The inspector conducted the following activities to determine if AmerGen was implementing operational, engineering, and administrative controls to maintain personnel occupational radiation exposure as low as is reasonably achievable (ALARA). The review was against the criteria contained in 10 CFR 20 and applicable industry standards and station procedures.

Inspection Planning

The inspector selectively reviewed pertinent information regarding station collective dose history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges.

The inspector determined the site specific trends in collective exposures (using NUREG-0713 and plant historical data) and source-term (average contact dose rate with reactor coolant piping) measurements. The inspector selectively reviewed site specific procedures associated with maintaining occupational exposures ALARA and processes used to estimate and track work activity specific exposures. The inspector determined the plant's three-year rolling average collective exposure. The inspector reviewed site specific procedures associated with maintaining occupational exposures ALARA as well as the processes used to estimate and track activity specific exposures.

The inspector selectively reviewed planning and preparation for the maintenance outage to determine if the AmerGen had established procedures, engineering and work controls, based on sound radiation protection principles, to achieve occupational exposures that were ALARA.

Radiological Work Planning

For planning purposes, the inspector selected work activities likely to result in the highest personnel collective exposures and reviewed the planning and preparation for those work activities to determine if ALARA requirements were integrated into work procedure and radiation work permit documents.

The work activities selectively reviewed were:

a. under vessel work/control rod drive change-out;
c. in-service inspection;
d. scaffolding activities;
e. shielding activities;
f. various valve work activities;
g. refueling activities; and,
h. radiological controls coverage.

The inspector selectively evaluated interfaces between operations, radiation protection, and other work groups particularly in the area of source term controls. The use of shielding and other techniques (e.g., decontamination) to reduce exposures was reviewed.

The inspector attended the October 6, 2004, Station ALARA Committee meeting. The inspector observed discussions regarding outage ALARA Plan Nos. 2004 -17, Fuel Moves, and 2004 -18, Reactor Reassembly and observed that a review was conducted to ensure a quorum was present.

The inspector compared the results achieved (dose and dose rate reductions, person-rem expended) with the estimated occupational doses established in the initial ALARA plans for selected work activities conducted during the fall 2003 outage. The inspector also reviewed exposure tracking for ongoing outage activities.

Job Site Inspections and ALARA Control

The inspector selectively reviewed under vessel work/control rod drive change-out, in-service inspection, scaffolding activities, various valve work activities, refueling activities, and radiological controls coverage. The inspector evaluated the use of ALARA controls for these work activities by reviewing use of engineering controls, implementation of ALARA procedures and controls, and use of shielding.

The inspector observed workers to determine if workers were utilizing low dose waiting areas and to determine if workers received appropriate on-the-job supervision to ensure the ALARA requirements were met. The inspector also reviewed job supervisor oversight to ensure the work activities were conducted in a dose efficient manner (e.g., work crew size minimized, workers properly trained, proper tools and equipment were used, etc.).

The inspector reviewed exposures of individuals from selected work groups.

Source-Term Reduction and Control

The inspector selectively reviewed AmerGen's evaluations in the area of source term controls. In particular, the inspector reviewed AmerGen's Co-60 source term control efforts. Areas reviewed included: source term, chemical controls, shutdown methodology, and clean-up strategies. Also reviewed were primary system piping radiation measurements including trends and current status.

Radiation Worker/Radiation Protection Technician Performance

The inspector observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas. The inspector reviewed activities that presented the greatest radiological risk to workers (e.g., under vessel work, reactor refueling pool work). The inspector determined if workers demonstrated the ALARA philosophy in practice (e.g., were workers familiar with the work activity scope and tools to be used, were workers utilizing ALARA low dose waiting areas) and whether there were any procedure compliance issues (e.g., were work activity controls being complied with). Also, the inspector observed worker/technician performance to determine if performance was consistent with expectations considering potential radiological hazards and the work involved.

Declared Pregnant Workers

The inspector reviewed the exposure and monitoring controls employed by AmerGen for declared pregnant workers with respect to 10 CFR 20 requirements, as applicable.

Verification of Dose Estimates and Exposure Tracking Systems

The inspector reviewed overall ALARA performance for the fourth quarter 2004 refueling outage. The inspector inter-compared accrued occupational dose, for various work tasks, relative to initial task estimates. Outage tasks reviewed, relative to initial estimates, included: under vessel work/control rod drive change-out, in-service inspection, scaffolding activities, shielding activities, various valve work activities, and various refueling activities including reactor disassembly and reassembly.

The inspector evaluated assumptions and bases for current annual collective exposure estimates and reviewed the dose rate and person-hour estimates (versus actual sustained) for accuracy for the tasks reviewed.

The inspector reviewed methods used to adjust exposure estimates (e.g., work-in-progress reviews), or replanning work, when unexpected changes in scope or emergent work are encountered. Also reviewed was the level of tracking detail, exposure report timeliness, and exposure report distribution.

The review was against the criteria contained in 10 CFR 20, applicable industry standards and station procedures.

Problem Identification and Resolutions

The inspector reviewed self-assessments, audits, and special reports related to the ALARA program to determine if identified problems were entered into the corrective action program for resolution. The inspector reviewed dose significant post-job (work activity) reviews and post-outage ALARA report critiques of exposure performance to determine if identified problems were properly characterized, prioritized, and resolved in an expeditious manner. (See Section 4OA2)

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 2 Samples)

a. Inspection Scope

The inspector reviewed selected activities, and associated documentation, in the below listed areas. The evaluation of AmerGen's performance in these areas was against criteria contained in 10 CFR 20 and applicable Technical Specifications and station procedures.

Problem Identification and Resolution

The inspector reviewed audits and self-assessments in the area of protective equipment to determine if identified issues in this area were entered into the corrective action program. The inspector reviewed condition reports and action requests to evaluate AmerGen's threshold for identifying, evaluating, and resolving problems in this area.

The inspector also selectively reviewed condition reports generated during the outage in the area of radiation monitoring instrumentation and protective equipment, as applicable. (See Section 4OA2)

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety [PS]

2PS2 Radioactive Material Processing and Transportation

Waste Characterization and Classification

a. Inspection Scope

The inspector selectively reviewed the radio-chemical sample results (i.e., 10 CFR Part 61 analysis) for each of AmerGen's radioactive waste streams. The inspector also selectively reviewed "Waste Stream Analysis and Scaling Factor Determination," dated December 1, 2004, and selectively reviewed procedure RP-OC-605-1001, Rev. 0, "Oyster Creek 10 CFR 61 Program."

The review was with respect to criteria contained in 10 CFR 61.55.

b. Findings

No findings of significance were identified.

2PS3 Radiological Environmental Monitoring Program (REMP) (71122.03 - 9 Samples)

a, Inspection Scope

The inspector reviewed: the current Annual Environmental Monitoring Report and licensee assessment results to verify that the REMP was implemented as required by technical specifications (TS) and the offsite dose calculation manual (ODCM) and for changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, interlaboratory comparison program, and analysis of data; the ODCM to identify environmental monitoring stations; licensee self-assessments, audits, licensee event reports, and interlaboratory comparison program results; the final safety analysis report (FSAR) for information regarding the environmental monitoring program and meteorological monitoring instrumentation; and, the scope of the licensee's audit program to verify that it meets the requirements of 10 CFR 20.1101 c).

The inspector walked down 7 (of 9) air particulate and iodine sampling stations;

two (of 4) surface water collection locations; and, 38 (of 48) thermoluminescent dosimeter (TLD) monitoring stations and determined that they were located as described in the ODCM and determined the equipment material condition to be acceptable.

The inspector observed the collection and preparation of a variety of environmental samples (listed above) and verified that environmental sampling was representative of the release pathways as specified in the ODCM and that sampling techniques were in accordance with procedures.

Based on direct observation and review of records, the inspector verified that the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the FSAR, NRC Safety Guide 23, and licensee procedures. The inspector verified that the meteorological data readout and recording instruments in the control room and at the tower were operable.

The inspector reviewed each event documented in the Annual Environmental Monitoring Report which involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement for the cause and corrective actions. The inspector conducted a review of the licensee's assessment of any positive sample results.

The inspector reviewed any significant changes made by the licensee to the ODCM as the result of changes to the land census or sampler station modifications since the last inspection. The inspector also reviewed technical justifications for any changed sampling locations and verified that the licensee performed the reviews required to ensure that the changes did not affect its ability to monitor the impacts of radioactive effluent releases on the environment.

The inspector reviewed the calibration and maintenance records for all air samplers. The inspector reviewed: the results of the licensee's contractor interlaboratory comparison program to verify the adequacy of environmental sample analyses performed by the licensee's contractor; the licensee's quality control evaluation of the interlaboratory comparison program and the corrective actions for any deficiencies; the licensee's determination of any bias to the data and the overall effect on the REMP; and quality assurance (QA) audit results of the program to determine whether the licensee met the TS/ODCM requirements. The inspector verified that the appropriate detection sensitivities with respect to TS/ODCM are utilized for counting samples and reviewed the results of the vendor's quality control program, including the interlaboratory comparison program to verify the adequacy of the vendor's program.

The inspector observed several locations where the licensee monitors potentially contaminated material leaving the radiologically controlled area (RCA), and inspected the methods used for control, survey, and release from these areas, including observing the performance of personnel surveying and releasing material for unrestricted use, and verifying that the work is performed in accordance with plant procedures.

The inspector verified that the radiation monitoring instrumentation was appropriate for the radiation types present and was calibrated with appropriate radiation sources. The inspector reviewed the licensee's criteria for the survey and release of potentially contaminated material; verified that there was guidance on how to respond to an alarm which indicates the presence of licensed radioactive material; and reviewed the licensee's equipment to ensure the radiation detection sensitivities are consistent with the NRC guidance contained in IE Circular 81-07 and IE Information Notice 85-92 for surface contamination and HPPOS-221 for volumetrically contaminated material. The inspector also reviewed the licensee's procedures and records to verify that the radiation detection instrumentation were used at their typical sensitivity level based on appropriate counting parameters and verified that the licensee has not established a "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high radiation background area.

B. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (IP 71151)

a. Inspection Scope

The inspectors reviewed the Performance Indicator (PI) data from January 2004 through December 2004 for Safety System Unavailability, Safety System Functional Failures, RETS/ODCM Radiological Effluent Occurrences, and Occupational Exposure Control Effectiveness PIs to verify their accuracy. The inspectors reviewed AmerGen's process for identifying and documenting the PI data as described in OC procedures LS-AA-2040 Rev. 4 , "Monthly PI Data Elements for Safety System Unavailability," and LS-AA-2003 Rev. 0, "Use of the INPO Consolidated Data Entry Database for NRC and WANO Data Entry," and compared the data using criteria contained in NEI 99-02, Rev. 2 to verify it was properly dispositioned in the PI reports. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as Performance Indicators.

Specifically, the inspector reviewed operator log entries for the Shutdown Cooling System and Containment Spray System out of service time and interviewed the Shutdown Cooling System and Containment Spray System engineers to discuss the criteria used to determine unavailability.

The inspector selectively reviewed corrective action program records for occurrences involving High Radiation Areas, Very High Radiation Areas, and unplanned personnel radiation exposures since the last inspection in this area.

Also, the inspector selectively reviewed corrective action program records and projected monthly and quarterly dose assessment results due to radioactive liquid and gaseous effluent releases for the fourth quarter 2003 to the fourth quarter 2004 (to date). The inspector also reviewed the 2002 and 2003 annual effluent release report.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (IP 71152)

1. Routine Resident Review of CAP Documents

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems”, and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s corrective action program. This review was accomplished by attending daily screening meetings and accessing the licensee’s computerized database.

b. Findings

No findings of significance were identified.

2. Identification and Resolution of Problems - ISI review (IP 71111.08)

a. Inspection Scope

The inspectors reviewed various condition reports which identified deficiencies during non-destructive testing activities, including those generated from two new recordable indications for the reactor steam dryer and CAP No. O2004-3603 which was generated following a human error made during Control Rod Tube Guide Tube (CRGT) visual examinations. The inspectors verified that identified deficiencies were reported, characterized, evaluated, and resolved within the corrective action program.

b. Findings

No findings of significance were identified.

3. Identification and Resolution of Problems - Public Radiation Safety (IP 71122.03)

a. Inspection Scope

The inspector reviewed the licensee’s Licensee Event Reports, Special Reports, and audits related to the radiological environmental monitoring program performed since the last inspection. The inspector determined that identified problems were entered into the corrective action program for resolution. The inspector also reviewed corrective action reports affecting environmental sampling, sample analysis, or meteorological monitoring instrumentation. Eighteen condition action program reports (CAPs) related to the problems identified in the radiological environmental monitoring program during the audit were reviewed.

b. Findings

No findings of significance were identified.

4. Identification and Resolution of Problems - Occupational Radiation Safety (IP 71121)

a. Inspection Scope

The inspector reviewed corrective action documents from the most recent outage to determine if identified problems were entered into the corrective action program for resolution and to evaluate AmerGen's threshold for entering issues into the program. The review included a check of possible repetitive issues, such as radiation worker or radiation protection technician errors. Also reviewed were recent audits and assessments, as appropriate, and corrective action program documents. (CAP Nos. O2004-2004, O2004-2977, O2004-3031, O2004-3502, O2004-3916, O2004-3888, O2004-3861, O2004-3848, O2004-3804, O2004-3731)

The review was against the criteria contained in 10 CFR 20, Technical Specifications, and station procedures.

b. Findings

No findings of significance were identified.

5. Annual Sample Review (1 Sample)

a. Inspection Scope

The inspector selected two corrective action documents (CAP Nos. O2003-1259, O2003-1836) associated with apparent unnecessary radiation exposures sustained by operators during entry into an area posted as a High Radiation Area to check a specific door in the reactor building. The checks were being conducted to implement a technical specification surveillance requirement. The inspector reviewed the CAP documents, discussed potential aggregate occupational exposure associated with the activity, and reviewed radiological surveys for the area.

b. Findings and Observations

No findings of significance were identified.

AmerGen's corrective action program identified that operators may be sustaining unnecessary low-level radiation exposure during checks of a door within a posted High Radiation Area in the reactor building. AmerGen evaluated the situation and determined that the checks were being conducted at a frequency in excess of that required by the technical specification. In addition, Amergen identified a discrepancy associated with scheduling conduct of tours of the area. AmerGen relaxed the frequency of the checks, coordinated the conduct of the checks, and revised procedures to provide for checks at the technical specification frequency. AmerGen reviewed the potential exposures sustained and concluded no significant personnel exposure was sustained either prior to or following its actions (e.g., the relaxation of the frequency). The travel route used by the operators did not require passage through an actual High Radiation Area.

6. Semi-Annual Review of Corrective Action Program Trends (IP 71152)

a. Inspection Scope (IP 71152)

The inspectors performed a semi-annual review of common cause issues in order to identify any unusual trends that might indicate the existence of a more significant safety issue. This review included an evaluation of repetitive issues identified via the corrective action process. The results of the trending review were compared with the results of normal baseline inspections. In addition, the inspector reviewed the following documents to determine if trends were identified that were not documented in the CAP system:

C Oyster Creek Nuclear Safety Review Board Meeting, September 9 and 10, 2004

C Nuclear Oversight Quarterly Report, NOSPA-OC-04-2Q, April - June 2004

C Nuclear Oversight Quarterly Report, NOSPA-OC-04-3Q, July - September 2004

C Engineering Programs Area Audit Report, NOSA-OYS-04-05, June 28 - July 16, 2004

C Organization and Administration, Training and Staffing Audit Report, NOSA-OYS-04-06, July 12 - 16, 2004

Surveillance and Test Program Audit Report, NOSA-OYS-04-07, September 7 - 17, 2004

Procedures, Document Control and Quality Assurance Records Audit Report, NOSA-OYS-04-08, September 20 - 24, 2004

Fire Protection Program Audit Report, NOSA-OYS-04-09, October 4 - 8, 2004

Independent Spent Fuel Storage Installation Audit Report, NOSA-OYS-04-10, November 29 - December 3, 2004

b. Findings and Observations

No findings of significance were identified. The inspector noted that the NOS reviews of maintenance during the six month period identified an unusually high percent of rework and that this was on an adverse trend. Also, NOS reviews of engineering revealed that the corrective actions were completed for the adverse trend in technical rigor that was identified in late 2003. The NOS quarterly reviews continued to observe the CAP process was not fully effective, although on an improving trend, and specifically noted that the station does not consistently use CAP data to identify repeat/recurring issues and that department managers, supervisors and CAP Coordinators are not fully engaged in their responsibilities to assure quality and value in CAP input, products, and trend reviews.

4OA3 Event Follow-up (IP 71153 - 3 Samples)

a. Inspection Scope

The inspectors reviewed the following three events during the period. The review consisted of observing plant parameters and status, including mitigating systems/trains and fission product barriers; reviewing alarms/conditions preceding or indicating the event; evaluating the performance of mitigating systems and licensee actions; and confirming that the licensee properly classified the event in accordance with emergency action level procedures and made timely notifications to NRC and state/county governments, as required. The specific events reviewed included:

- Loss of containment integrity on November 22, 2004
- Inadvertent EMRV Opening During Cold Hydro Test on November 19, 2004
- Inadvertent RPV Water Level Transient on November 22, 2004

b. Findings

Introduction

A self-revealing event involving an inadvertent loss of the containment isolation function resulted in a Green finding and non-cited violation for not establishing and maintaining appropriate procedural requirements for the operation of the containment vent and purge isolation valves, as prescribed by Technical Specification 6.8.1 and the Oyster Creek Operational Quality Assurance Plan.

Description

At 10:42 am, on November 22, 2004, plant operators placed the “Containment Vent and Purge Isolation Bypass” keylock switch in the bypass mode as directed by Step 6.51.1 of Plant Startup Procedure 201. At the time, a plant startup was in progress with the Mode Switch in Startup and reactor power at about 14.5%. Also at the time, containment drywell ventilation valves, V-27-1 and V-27-2, were open. With the isolation bypass keylock switch in bypass and the drywell ventilation valves open, the primary containment isolation function was inoperable. This condition existed for about two hours, when operators questioned the accuracy of the procedural step that directed the action.

Initially, when the operators were about to place the isolation bypass switch into the bypass mode, a reactor operator questioned the action. The question about the accuracy of the Startup Procedure did not result in stopping the switch manipulation as expected to ensure that the actions taken were appropriate and fully understood. At about this same time a reactor vessel water level transient occurred, diverting attention to recovery from that event to prevent a possible plant scram. The switch position was questioned later during a panel walk-down by the Shift Manager. The system operating procedure requirements were then reviewed and operators concluded that Startup Procedure 201 step 6.51.1 was in error, in that the wrong switch was identified. Operators restored the Containment Vent and Purge Isolation Bypass keylock switch to normal at about 12:22 pm, on November 22, 2004.

A human performance cross-cutting aspect of this finding was identified in that: (1) the procedure development involved a human error in identifying the wrong switch listed in step 6.51.6, and (2) that the initial questioning of this action by an operator did not result in preventing the action and resultant loss of containment integrity.

Analysis

The finding was more than minor because it affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radionuclide releases caused by accidents or events and the related attributes of configuration control and procedure quality.

In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 screening. The finding represented a degradation in the barrier integrity cornerstone, because both drywell vent containment isolation valves were open and the primary containment isolation logic was bypassed for each valve, causing a loss of safety function for the containment barrier. The Phase 1 screening resulted in an evaluation per Manual Chapter 0609, Appendix H, Containment Integrity Significance Determination Process for the containment barrier being degraded due to an actual open pathway in the physical integrity of reactor containment.

The Appendix H entry condition for this performance deficiency was a degraded condition affecting containment barrier integrity that can potentially increase Large Early Release Frequency (LERF) without affecting Core damage Frequency (CDF), which is considered a Type B finding. Appendix H, Section 6 provides the analysis for Type B findings. Table 6.1, Phase 1 Screening for Type B findings for a BWR Mark 1 containment type required a Phase 2 assessment since the finding involved an open path through the containment vent and purge systems. Table 6.2, Phase 2 Risk

Significance for BWR Mark 1 Containment Types screened to Green because although the finding resulted in the possible leakage rate from the drywell to the environment of >100% containment volume/day through the open vent system, the exposure time was only 2 hours (< 3 days) - column 3 of table.

Enforcement. Technical Specification 6.8.1 requires in part that written procedures shall be maintained for procedures recommended in Appendix A of Regulatory Guide 1.33 as referenced in the Oyster Creek Operational Quality Assurance Program. Appendix A of Regulatory Guide 1.33 includes general plant operating procedures for startup of the plant. Contrary to these requirements procedure 201, Plant Startup, was not maintained properly in that step 6.56.1 erroneously required plant operators to place the Containment Vent and Purge Valves Isolation Bypass switch into bypass. This resulted in the reactor primary containment isolation system being inoperable on November 22, 2004, at a time when it was required to be operable per Technical Specification 3.5.A.3 because the reactor was critical. Because this condition is of very low safety significance and has been entered into AmerGen's corrective action program (CAP O2004-4012), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368).
(NCV 05000219/200400503)

4OA4 Cross-Cutting Issues other than PI&R

Section 4OA3 of the report describes a finding for an inadequate startup procedure that resulted from human error during procedure development. Also, this erroneous step was initially questioned during the implementation; however, operators did not fully comprehend the required action and did not avoid taking the action, resulting in a loss of primary containment integrity.

4OA6 Meetings, including Exit

Exit Meeting Summary

On January 13, 2005, the resident inspectors presented the inspection results to Mr. C. N. Swenson and other members of licensee management. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

P. Bloss, BOP Systems Manager
M. Button, Director, Maintenance
C. Connelly, Manager, Chemistry & Rad Protection
J. Derby, Radiological Engineer
R. Detwiler, Director, Operations
R. Ewart, Security Manager
D. Fawcett, Licensing Engineer
J. Freeman, Shift Operations, Superintendent
M. Godknecht, Maintenance Rule Coordinator
S. Hutchins, Electrical Systems Manager
E. Johnson, System Engineer
A. Judson, Radiological Engineer
J. Kandasamy, Manager, Regulatory Assurance
J. Magee, Director, Engineering
M. Massaro, Plant Manager
D. McMillan, Director, Training
L. Newton, Chemistry Manager
J. O'Rourke, Assistant Engineering Director
J. Renda, Radiation Protection Manager
G. Seals, Radiological Engineer
H. Shoap, Normandeau Associates
C. Swenson, Vice President
D. Weible, Environmental Chemist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000219/2004005-01	URI	(Section 1R13) Pressure Locking of Isolation Condenser Valve, V-14-35, Potential Finding

Opened and Closed

05000219/2004005-02	NCV	(Section 2OS1) Violation of 10 CFR 20.1501 for Erroneous Radiological Surveys Associated with Repairs to MSIV

05000219/2004005-03 NCV (Section 4OA3) Violation of TS 6.8.1 for Inadequate Written Startup Procedure Causing a Loss of Containment Integrity

LIST OF DOCUMENTS REVIEWED
(not previously referenced)

Section 1R08: Inservice Inspection Activities

NDT Examination Reports

INR No. OC1R20-04-01, Steam Dryer Tie Bar Indication report
INR No. OC1R20-04-02 Rev-1, Steam Dryer 45 & 135 Lifting Lug Indications
GE Reports # GE 065, VT-3 Exam on Isolation Condenser V-14-35
GE Reports # GE 066, PT Exam on Isolation Condenser V-14-35 valve seat
GE Reports # GE 067, VT-1 Exam on Isolation Condenser V-14-35 bonnet bolts
GE Reports # GE 068, VT-3 Exam on Isolation Condenser V-14-35 valve stem
GE Reports # GE 069, PT Exam on Isolation Condenser V-14-35 disc
GE Reports # GE 071, PT Exam on Isolation Condenser V-14-35 Valve wedges, disc seating area, and trunion
GE Reports # GE 091, VT-1 Exam on Isolation Condenser V-14-35 Socket head cap screws

Procedures

ER-AA-335-014 Rev 1, Station VT-1 Inspection Procedure
ER-AA-335-016 Rev 1, Station VT-3 Inspection Procedure
MA-AA-716-012, Station PT Procedure
GE-PT-100, Procedure for the Penetrant Examination of pipe welds
MA-AA-716-008, FME Exclusion Procedure
GE-PDI-UT-2, PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe welds
GE-PDI-UT-10, PDI Generic Procedure for the Ultrasonic Examination of Dissimilar Piping welds

In Vessel and Remote Visual Examinations

IVVI CRGT-3 structural weld - bottom of control rod guide tubes
VT-1 Steam dryer structural welds

Repair-Replacement

WO# C2008344, NDE Shop welds for ESW keep fill line
WO# C200834407, NDE ESW Keep fill line field welds
WO# C2008974, Open, Inspect, and Repair Isolation Condenser V-14-35

ECR OC 03-00454-002, Replacement of ESW keep fill line pipe with corrosion resistant material

Flaw Evaluation

GE-0000-0034-4166-R0, Evaluation of New Steam Dryer Indication

Condition Reports and Action Requests

CAP Nos.: O2004-1316, O2004-1367, O2004-3489, O2004-3498, O2004-3525, O2004-3536, O2004-3603, O2004-3640, O2004-3664

A/R 2090093, Clean and Re-coat RBCCW lines in the Drywell

Other Documents

Inspection Report 50-220/03-003, Nine Mile Point Station - NRC Special Inspection
IVVI Video Tapes, OC1R20-04-07 through 09
GE NDE Team Training Certification Records

Section 1R12: Maintenance Rule Implementation

Procedures

ER-AA-310-1003, “Maintenance Rule - Performance Criteria Selection,” Rev. 2
ER-AA-310-1004, “Maintenance Rule - Performance Monitoring,” Rev. 1
ER-AA-310-1005, “Maintenance Rule - Dispositioning between (a)(1) and (a)(2),” Rev. 1
ER-AA-310, “Implementation of the Maintenance Rule,” Rev. 3
Startup Transformer alarm response procedures 2000-RAP-3024.02, s-5-b and s-5-c

Evaluations

OC-7 Functional Failure Definition for System 724 (Startup Transformers) Supporting Operability Documentation CR O2002-0758
System Health Overview Report, dated September 2004

Corrective Action Programs Documents

CAP Nos.: O2004-1527, O2004-1201, O2004-0052, O2003-1093, O2002-0660, O2004-2658, O2004-3964, O2004-0751, O2003-2472, O2002-1966, and O2002-1699

Section 2OS1: Access Control to Radiologically Significant Areas

Procedures

RP-OC-4001, Rev 0, Oyster Creek Radiation Protection Elevated Dose Rate Contingency Plan
6630-ADM-4212.01, Rev. 10, Air Sample Collection and Analysis
RP-AA-441, Rev 2, Evaluation and Selection Process for Radiological Respirator Use

Section 2OS2: ALARA Planning and Controls

Procedures

RP-AA-401, Rev.2, Operational ALARA Planning and Control
2003-2005 Exposure Reduction Plan
CY-AB-120-130, Rev. 2, BWR Shutdown Chemistry
CC-AA-102, Rev. 8, Design Input and Configuration Change Impact Screening
RP-AA-403, Rev. 1, Administration of the Radiation Work Permit Program
OU-AA-101, Rev. 6, Refuel Outage Management
WC-AA-101-1002, Rev.3, On Line Scheduling Process

Other Documents

Engineering Standard, Rev. 2, Cobalt Reduction Standard
Radiation Protection Outage Preparation Checklist
Station ALARA Committee Minutes, September 24, 2004, October 1 and 2, 2004
Oyster Creek 1R20 Flush Plan

Section 2PS3: Radiological Environmental Monitoring Program

Procedures

CY-OC-170-301, Rev 1, Offsite Dose Calculation Manual for Oyster Creek Generating Station
RP-AA-03, Rev 0, Unconditional Release Survey Method
ER-OCGS-02, Rev 0, Collection of Thermoluminescent Dosimeters (TLD's) for Radiological Analysis
ER-OCGS-03, Rev 0, Collection of Aquatic Sediment Samples for Radiological Analysis
ER-OCGS-04, Rev 0, Collection of Food Products and Broadleaf Vegetation Samples for Radiological Analysis
ER-OCGS-05, Rev 0, Collection of Air Iodine and Air Particulate Samples for Radiological Analysis
ER-OCGS-06, Rev 0, Collection of Surface Water Samples for Radiological Analysis
ER-OCGS-10, Rev 0, Collection of Well Water Samples for Radiological Analysis
ER-OCGS-14, Rev 0, Collection of Fish Samples for Radiological Analysis
ER-OCGS-16, Rev 0, Collection of Clam and Crab Samples for Radiological Analysis

2400-SMI-3662.02, Rev 9, Meteorological System Calibration
6633-PMI-4224.41, Rev 2, Calibration of the Bicron-NE Small Articles Monitor
Oyster Creek Nuclear Generating Station 2004 Land Use Survey

Reports

Oyster Creek Generating Station Unit 1 Annual Radiological Environmental Operating Report, 1 January Through 31 December 2003 (May 2004)
Teledyne Brown Engineering Environmental Services 2d Quarter 2004 Quality Assurance Report
Teledyne Brown Engineering Environmental Services Annual 2003 Quality Assurance Report

Corrective Action Programs Documents

Nuclear Oversight Audit (NOSA)-OYS-03-08 (10/03)

Radiological Environmental Monitoring Program Functional Area Self-Assessment (9/04)

CAP Nos.: O2004-0144, O2004-0147, O2004-0219, O2004-0854, O2004-1228, O2004-1580, O2004-1690, O2004-1725, O2004-1854, O2004-2455, O2004-2463, O2004-2684, O2004-2797, O2004-2798, O2004-2801, O2004-2802, O2004-2803, O2004-2921

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
AR	Action Request
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ESW	Emergency Service Water
HRA	High Radiation Area
IMC	Inspection Manual Chapter
ISI	In-service Inspection
IVVI	In-Vessel Visual Inspection
JO	Job Order
LER	Licensee Event Report
LHRA	Locked High Radiation Area
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Testing

NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OHS	Office of Homeland Security
OS	Occupational Safety
PDI	Performance Demonstration Initiative
PI	Performance Indicator
PI&R	Problem Identification & Resolution
PMT	Post Maintenance Test
PSIG	Pounds per Square Inch Gauge
PT	Liquid Dye Penetrant Testing
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RCA	Radiologically Controlled Area
REMP	Radiological Environmental Monitoring Program
RHR	Residual Heat Removal
RO	Reactor Operator
RPS	Reactor protection System
RWP	Radiation Work Permit
SDP	Significance Determination Process
ST	Surveillance Test
TLD	Thermoluminescent Docimeter
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
WO	Work Order