January 28, 2003

Mr. Jack Skolds
Chairman and CEO of AmerGen
AmerGen Energy Company, LLC
4300 Winfield Road
5th Floor
Warrenville, IL 60555

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 50-219/02-08

Dear Mr. Skolds:

On December 28, 2002, 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 23, 2003, with Mr. E. Harkness and other members of your staff.

This 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.

This report documents four NRC-identified findings and three self-revealing findings of very low safety significance (Green), six of which were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were 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, DC 20555-001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at Oyster Creek.

Since the terrorist attacks on September 11, 2001, the NRC has issued two Orders (dated February 25, 2002, and January 7, 2003) and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the Order dated February 25, 2002. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) 2002, and the remaining inspections are scheduled for completion in CY 2003. Additionally, table-top security drills were conducted at several licensee facilities to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigative strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY 2003, the NRC will continue to monitor overall safeguards and security controls, conduct inspections, and resume force-on-force exercises at selected power plants. Should threat conditions change, the NRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial power reactors.

In accordance with 10 CFR 2.790 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 Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch No. 7
Division of Reactor Projects

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

Enclosure: Inspection Report 50-219/02-08
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-219
License No.:	DPR-16
Report No:	50-219/02-08
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	September 29, 2002 - December 28, 2002
Inspectors:	Robert Summers, Senior Resident Inspector Steve Dennis, Resident Inspector Steve Shaffer, Reactor Engineer Tim O'Hara, Reactor Inspector George Morris, Reactor Inspector John McFadden, Health Physicist Tom Burns, Reactor Inspector Jennifer Bobiak, Reactor Engineer Gregory Smith, Senior Physical Security Inspector
Approved By:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000219/2002-008; AmerGen Energy Company, LLC; 09/29/02-12/28/02; Oyster Creek Generating Station; inservice inspection activities, maintenance risk assessment and emergent work evaluation, personnel performance during non-routine evolutions, post-maintenance testing, surveillance testing, and problem identification and resolution.

The report covered a 13-week period of inspection by resident and region-based inspectors. Seven Green findings, six of which with non-cited violations (NCVs), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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. Inspector Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- **Green**. An inadequate maintenance procedure resulted in the inadvertent ignition of Hydrogen gasses contained in the offgas system during air in-leakage testing. The procedure failed to provide instructions to properly isolate and vent the test device sample chamber from the process stream before ionizing the test sample chamber.

  A self-revealing finding was identified. This finding is greater than minor because it had an actual impact of igniting the offgas system hydrogen gas causing the main condenser offgas system to be isolated, and therefore could be viewed as a precursor to a significant event. If the offgas system could not have been quickly restored, it would have caused a reactor scram transient. The finding is of very low significance because all mitigation systems were available during this event. The hydrogen ignition did not result in damage to the plant and was contained within a system designed for such events, and operators restored the offgas system before main condenser vacuum degraded to a trip condition. In addition, this finding had a human performance aspect, in that plant technicians proceeded to perform the test without a plant specific procedure and they did not fully adhere to the guidance provided with the equipment, which had a direct causal affect on the event initiation. (Section 1R14)

Cornerstone: Mitigating Systems

- **Green**. A non-cited violation of 10 CFR 50 Appendix B, Criterion V was identified for failure to implement engineering instructions provided in an engineering change request document.

  This finding is greater than minor because it affected the design control attribute of the Mitigating Systems Cornerstone and could have affected the reliability of the isolation condenser system. AmerGen personnel installed three pipe fittings in the isolation condenser system using material that was specifically prohibited from use by the engineering document. Oyster Creek personnel had not adhered to procedural requirements governing the control of materials used for the installation of piping in the isolation condenser system. The finding is of very low safety significance because the plant was not operational at the time and subsequent analysis verified the vent line modification was in compliance with the applicable Code and design requirements. In addition, this finding had a human performance aspect, in that plant technicians did not adhere to installation guidance provided in the modification package. (Section 1R08)

- **Green**. A non-cited violation of Oyster Creek Technical Specification 6.8, Procedures and Programs, was identified for failure to have an adequate surveillance procedure for the emergency service water pump. AmerGen failed to maintain appropriate acceptance criteria in the quarterly emergency service water pump inservice test procedure.

  The finding is considered more than minor because it is associated with the Mitigating Systems cornerstone attribute of procedure quality and affects the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The finding is of very low safety significance because the finding was a qualification deficiency confirmed to not result in the loss of the safety function of the Emergency Service Water System. (Section 1R19.1)

- **Green**. A non-cited violation of Oyster Creek Technical Specification 6.8, Procedures and Programs, was identified for failure to adequately implement a Control Rod Drive system procedure. AmerGen declared a control rod operable, following maintenance work, without performing post-maintenance testing as required by the procedure.

  The finding is considered greater than minor in that the issue was associated with the Mitigating System Cornerstone and potentially affected the scram function of a control rod in response to an initiating event. The finding is of very low safety significance because the control rod remained at the full in position (notch 00) throughout the performance of the maintenance work and no other control rods were concurrently inoperable. (Section 40A2.1)

Cornerstone: Barrier Integrity

- **Green**. A non-cited violation of 10 CFR 50 Appendix B Criterion XVI, Corrective Actions, was identified for failure to adequately identify and correct a condition adverse to quality involving the continued operability of the No. 2 Standby Gas Treatment System (SGTS) charcoal filter.

  The finding is considered greater than minor because it had an actual impact in that the No. 2 SGTS was inoperable. The finding is of very low safety significance because the finding only represented a degradation of the radiological barrier function provided for by the standby gas treatment system.

In addition, this finding had a corrective action performance aspect, in that degraded or non-conforming conditions adverse to quality had not been identified in a timely manner to ensure appropriate corrective actions were taken. (Section 1R13.1)

- **Green**. A non-cited violation was identified during the performance of the primary containment isolation valve test on October 22, 2002, for failure to maintain the secondary containment configuration in accordance with Technical Specification 3.5.B, when the trunnion room door was opened and not administratively controlled, which resulted in a temporary loss of secondary containment.

  A self-revealing finding was identified. The finding is considered more than minor because the reactor safety barrier integrity cornerstone attribute of human performance was involved and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events. The finding is of very low safety significance since the finding involved a BWR in a Cold Shutdown condition with time to boil being greater than two hours and reactor coolant system level less than 23 feet above the top of reactor flange and the inspector verified that secondary containment closure could be accomplished in sufficient time before a release of fission products, including the unavailability of AC power and the expected environmental condition in containment. This finding had a human performance aspect, in that plant operators did not adhere to directions provided to ensure that the trunnion room door was maintained closed and only opened for the short time for passage through the area as required by the licensee's administrative controls. (Section 1R22)

Cornerstone: Occupational Radiation Safety

- **Green**. A non-cited violation of Technical Specification 6.13 was identified for failure to establish fully effective problem resolution relative to recurring problems involving personnel failing to hear the integrated dose alarm of their electronic self-reading personnel dosimetry equipment and to promptly respond to such an alarm. Specifically, on July 19, 2001, September 18, 2002, and October 7, 2002, workers did not hear the dosimetry alarms and consequently did not promptly exit the area.

  A self-revealing finding was identified. Repeat events in violation of Technical Specification 6.13 were more than minor in that worker safety could be impacted in similar circumstances if workers failed to properly respond to alarming dosimeters in situations with the potential for unplanned radiation dose. Following the Occupational Radiation Safety Significance Determination Process, it was identified that the issues were not ALARA findings, did not result in any overexposure, did not create a substantial potential for any overexposure, and did not compromise the licensee's ability to assess dose to workers. None of these three recurring incidents resulted in any significant unintended exposure. Therefore, the issues were determined to be of very low safety significance. (Section 4OA2)

B. Licensee Identified Violations

None.

Report Details

Summary of Plant Status:

Oyster Creek began the inspection period at approximately 96 percent power in an end-of-cycle power coast down due to fuel depletion. A refueling outage began on October 4, 2002, and ended on October 27, 2002. Oyster Creek returned to 100% power on October 29, 2002, and remained there for the duration of the inspection period with the exception of several occasions during which reactor power was decreased for a brief period of time for control rod and reactor recirculation flow adjustments.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection

Winterization

a. Inspection Scope

The inspectors reviewed plant procedures, system readiness reviews, action requests, and the status of corrective actions to verify the ability of risk significant systems to function in the winter climate. The inspectors walked down portions of the Emergency Service Water, Service Water, Fire Water, Circulating Water, and Dilution systems to verify seasonal readiness was performed as described in the winter readiness plan. Additionally, the inspectors reviewed a sample of maintenance work performed on heat tracing, insulation and other portions of those systems to verify the work was completed as documented in associated work packages. The inspectors also reviewed the readiness status of the house heating boilers and the plant heating steam system. The following documents were reviewed.

- Exelon Seasonal Readiness Procedure - OP-AA-108-109 Rev.1
- Winter Readiness Plan - 2000-PLN-3000.02
- Corrective Action Program Document -CAP No. 02002-1735 - Review of Winter Readiness.
- Winter Readiness Matrix of Work Requests - dated November 18, 2002

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial Equipment Alignments

a. Inspection Scope

The inspectors performed three partial walkdown inspections during this inspection period on the systems listed below. Each walkdown included a random sampling of valve and breaker positions in the field that were verified to be properly aligned in accordance with associated system operating procedures. Control room indications and controls were verified to be appropriate for the standby or operating status of the system and system maintenance action requests were reviewed to verify that no degraded conditions existed adversely affecting operability.

- Liquid Poison System - October 31 - November 1, 2002
- DC Distribution and Main Battery Banks A, B, and C - November 6 - 8, 2002
- Standby Gas Treatment System 1 and 2 - December 13 - 20, 2002

b. Findings

No findings of significance were identified.

.2 Full Equipment Alignment - Core Spray System

a. Inspection Scope

The inspectors conducted a detailed review of the alignment and conditions of the Core Spray system from October 5, 2002, through November 21, 2002. The inspectors reviewed operating and surveillance procedures associated with the system and performed a walkdown to verify normal system alignment was maintained in accordance with procedural checklists. Additionally, valve and electrical breaker positions in the field, including those accessible on drywell elevation 23', were verified to be properly aligned in accordance with electrical prints and piping diagrams. Control room indications and controls were also verified to be appropriate for the standby or operating status of the system and consistent with technical specification requirements and the Updated Final Safety Analysis Report. The inspectors reviewed and evaluated the potential impact on Core Spray system operation from open work orders, design modifications, engineering change requests and corrective action process (CAP) reports. The inspectors also reviewed and discussed the Core Spray System maintenance health report with the system engineer. The documents reviewed are listed below.

- CAP Nos. O2002-0769, 1466, 1636, 1669, 1699, 1719, 1764, 1932, and 1966
- Pipe Stress and Support Analysis Calculation - C-1302-21-E310-122
- Core Spray System Health Report
- Core Spray Elementary Diagram - GE 718E644 SHT.2
- Core Spray System Flow Diagram - GE 885D781
- Core Spray System Electrical Diagram - GE 116B8328 SHT. 15A, B, C, D
- Core Spray Valve Operability and In-service Test - 610.4.003
- Core Spray Pump Operability Test - 610.4.002
- Core Spray Component Health Indicator Program

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Area Inspection

a. Inspection Scope

The inspectors conducted fire protection inspection activities consisting of plant walkdowns, discussions with fire protection personnel, and reviews of 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. Plant walkdowns of the accessible portions of the six areas described below were completed to assess the licensee’s control of combustible material, fire detection and suppression equipment capability, and any related compensatory measures.

- RB-FA-2, Drywell
- TB-FZ-11E, Condenser Bay, Elev. 3'-6"
- OB-FZ-8C, A and B Battery Rooms
- TB-FZ-11C,Switchgear Rooms West Side of Turbine Building
- TB-FA-26, 4160V Switchgear Room South Side of Turbine Building
- TB-FZ-11B and 11F - Turbine Building West Side (includes entry doors to Reactor Building Corner Rooms containing the Core Spray Pumps)

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (ISI)

a. Inspection Scope

The inspector reviewed selected samples of nondestructive examination (NDE) activities completed this outage. Also, the inspector reviewed selected samples of repair, replacement and modifications to safety-related systems. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems. The documentation review was performed to verify the activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspector reviewed a sample of examination and deviation reports from the previous outage wherein recordable indications had been accepted by the licensee for continued service.

The inspector reviewed ultrasonic testing (UT), penetrant testing (PT), and remote in-vessel visual examinations to verify effectiveness of the process in identifying degradation of risk significant systems, structures, and components and to evaluate the activities for compliance with the requirements of ASME Section XI. The inspector examined the licensee’s disposition of non-conforming conditions identified during inservice inspection activities (Corrective Action Reports O2000-1631, O2002-1465, 1533, 1633, and 1643) and verified evaluations were performed for continued operation without repair or rework. The inspector reviewed the UT results performed on recirculation system weld NG-D-0018, core spray weld NZ-3-0072 and the PT results of control rod drive weld NC-4-0001B.

The inspector reviewed a sample of video recordings of the remote in-vessel visual inspection (IVVI) of the steam dryer and core spray piping base material and selected fittings and welds. In addition, for stub tubes 42-43 and 46-39, the inspector reviewed the results of the visual examination of exterior weld cladding on the tubes, the circumferential weld of the stub tubes to the vessel lower head and the weld of the stub tubes to the control rod drive housings. This review was conducted to verify the test conditions enabled the performance of the examination and that test results were accurately characterized and recorded. Also, the inspector confirmed that for the recordings evaluated, the visual examination was in compliance with the requirements of ASME Section XI.

The inspector reviewed installation activities associated with the modification, repair and replacement of selected components in the isolation condenser and the service water systems to verify the activities were performed in accordance with the requirements of ASME Section III, IX, XI and ANSI B 31.1, as applicable. The inspector reviewed Engineering Change Request (ECR) 01-00828, Isolation Condenser Vent Valve Replacement and ECR 01-00621, Crosstie Emergency Service Water to Service Water. The inspector reviewed the engineering specifications, process control instructions, welding instructions, NDE requirements and the test results of the completed modifications.

b. Findings

Introduction. The inspector identified a non-cited violation of 10 CFR 50 Appendix B, Criterion V, having very low safety significance (Green). On October 17, 2002, AmerGen Energy failed to follow engineering instructions provided in ECR 01-00828 and, as a result, Oyster Creek personnel installed three pipe fittings using material which was specifically prohibited from use in the vent line in the isolation condenser system.

Description. The inspector identified the use of base material and weld filler metal not in accordance with the engineering specification governing the replacement of vent piping in the isolation condenser system. Engineering specification ECR 01-00828 specified the new materials shall comply with the requirements of ANSI B 31.1-1983 with Winter 1984 Addenda, and the replacement valves shall be designed to ASME Section III, Class 2 (NC).

The Engineered Material section of ECR 01-00828 states: “the piping is 3/4 inch, schedule 80, seamless stainless steel ASME/ASTM SA/A 312 Type 316. The “L” grade material is not permitted unless the material is dual grade in that it complies with both specifications and is so marked (316/316L). Fittings are socket weld ASME/ASTM SA/A 182 F316."

In the review of welding and fabrication documents, the inspector observed that, contrary to ECR 01-00828, Oyster Creek personnel installed components of Type 316L material in some portions of the modification. The inspector also noted that the ANSI/ASME Power Piping Code establishes the maximum stress allowable limits for “L” grade stainless steel at a significantly lower level than the non “L” grade.

In response to the inspectors observations, Oyster Creek personnel determined that three pipe couplings of the “L” grade material were installed and, therefore, CAP O2002-1643 was initiated. The reduced stress allowable values of the “L” grade material installed during the modification would invalidate Oyster Creek stress analysis calculation, C-1302-211-E310-134, which supports the design and fabrication of the modification as meeting the applicable Code requirements. As a result, Oyster Creek personnel retrieved the certified material test reports and reviewed the actual mechanical property levels. They determined that for the materials used, the strength levels were significantly higher than the nominal Code required minimums. Consequently, the stress analysis calculations were performed again using the actual stress allowable per the certified material test reports with acceptable results. On this basis, Oyster Creek personnel accepted the modification as built, and were revising the engineering specification to allow the use of the installed “L” grade materials.

Analysis. AmerGen’s failure to prevent the installation of material prohibited by engineering specification in a safety-related system is considered a performance deficiency. The finding adversely impacted the stress analysis used in-part to approve the modification. In addition, this finding had a human performance aspect, in that a direct cause of the event was a failure to adhere to engineering guidance during the implementation of the modification.

The issue is more than minor since failure to install the correct safety-related material as specified in engineering modification documents could be a precursor to a more significant event. The integrity of the vent line could only be ensured through an analysis utilizing the stress allowable as determined from the actual installed material strength levels. The issue affects the Mitigating System cornerstone since a failure of the isolation condenser vent line would require isolation of the condenser(s). However, the issue was determined to have very low safety significance using Phase 1 of the NRC significance determination process described in NRC Inspection Manual Chapter (IMC) 0609, Appendix A, because the plant was not operational at the time of the material substitution and subsequent analysis verified the integrity of the vent line. Therefore, the isolation condenser system remained operable and the issue did not represent an actual loss of system function.

Enforcement. Failure to adhere to the specified material requirements for installation of safety-related components is a violation of 10 CFR 50 Appendix B, Criterion V. Criterion V of 10 CFR 50, Appendix B, “Instructions, Procedures, and Drawings” requires that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances, and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to this requirement, in October 2002, the licensee did not install vent line piping in the isolation condenser system, an activity affecting quality, in accordance with documented engineering instructions. As a result, material specifically prohibited from use by the engineering specification was installed in the system. However, because of the very low safety significance of this issue, and because it was entered into the Oyster Creek CAP (CAP O2002-1643), the issue is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-219/02-08-01, Failure to Adhere to Specified Material Requirements for Installation of Safety Related Components.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator simulator training on November 20, 2002, to verify that the Oyster Creek (OC) operator requalification program adequately evaluated how well operators have mastered the training objectives, including training on high-risk operator actions. The inspectors reviewed the critical tasks associated with the simulated control room exercise, observed the operators performance during the exercise and observed the post-exercise critique to assess the licensee's effectiveness in evaluating and correcting any observed deficiencies. The inspector also reviewed licensee conformance with procedure 2611-PGD-2612, OC Licensed Operator Requalification Training Program.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee'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) in accordance with 10 CFR 50.65. The inspectors reviewed action requests (ARs), corrective action program reports (CAPs), engineering change requests (ECRs) and (a)(1) corrective action plans. The inspectors also compared unavailability data with control room log entries to verify compliance with (a)(1) goals. AmerGen's trending data was also reviewed. The SSCs reviewed during the inspection period included:

- Containment Spray/Emergency Service Water System 1
- Core Spray System 2

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

.1 Standby Gas Treatment System #2 Charcoal Bed Replacement

a. Inspection Scope

On December 13, 2002, the #2 Standby Gas Treatment System was declared inoperable due to a reported failure of a charcoal bed test sample in which the technical specification acceptance criteria of greater than or equal to 95% removal efficiency was not achieved. The inspectors reviewed the risk assessment describing the overall effects of the failure on the system and the plant. Additionally, the inspectors verified that replacement charcoal filters were appropriately installed and tested and that the associated technical specifications were addressed and documented in the CAP (CAP O2002-1951).

b. Findings

Introduction. A Green NCV was identified for failure to identify a condition adverse to quality in order to implement timely corrective actions to ensure the continued operability and availability of the Standby Gas Treatment System (SGTS) charcoal bed in accordance with Technical Specification 4.5.H.1.A and 10 CFR 50 Appendix B Criteria XVI, Corrective Action, which resulted in unnecessary unavailability time to replace the failed charcoal bed.

Description. On December 13, 2002, a vendor test report of the charcoal efficiency in the No. 2 Standby Gas Treatment System indicated that the charcoal bed efficiency was 92.39%, which is below the 95% minimum allowed by plant technical specification 4.5.H.1.A. The sample of the charcoal bed was taken on October 11, 2002 and was analyzed per ASTM D3803-1983. It was later determined by the licensee that the vendor had not used an Oyster Creek specific required face velocity flow of 45.72 feet/minute during the test. While identifying that an incorrect test method had been used, the licensee concluded that the charcoal efficiency was still below the acceptance criteria even when accounting for the erroneous test method. Operators declared the No. 2 SGTS inoperable and verified that the No. 1 SGTS was operable, including verifying the acceptability of the most recent charcoal bed efficiency test completed in May 2001.

During a review of recent test results for the SGTS, the inspectors noted the following. In December 1999, the No. 2 SGTS charcoal filter was determined to be 95.21 % efficient. In May 2001, the No. 2 SGTS charcoal filter efficiency was determined to be 95.03 % efficient, just slightly above the allowable minimum of 95%. The inspector determined that the licensee had not identified this condition to be degraded at that time. As such, actions were not taken to ensure the continued operability of the charcoal bed, such as through timely replacement of the filter or accelerated testing to further analyze the efficiency. The No. 2 SGTS charcoal filter bed was not tested again until October 2002, when the efficiency was found below the required technical specification acceptance criteria.

On October 11, 2002, the licensee sent a charcoal sample to their vendor for analysis and testing in accordance with technical specification surveillance requirement 4.5.H. This test is required every 18 months. While the test sample was taken within the specified surveillance period of 18 months (actually 17 months), the test results were not reported to the site until December 13, 2002. This further delayed corrective action until a replacement charcoal filter could be procured, received, and installed on December 17, 2002.

On December 16, 2002, a new charcoal filter bed was received at the plant and installed. However, the licensee's Nuclear Oversight group identified that the supply vendor had not ensured appropriate testing was completed prior to sending the new filter beds to the plant. The licensee identified that an old standard of the test procedure had been used which resulted in the wrong face velocity flow rate being used during the test. This discrepancy was resolved prior to the No. 2 SGTS being declared operable on December 17.

The inspectors noted that the licensee's Final Safety Analysis Report section 6.5 states that the offsite doses calculation still remains well within the 10 CFR 100 guidelines even with charcoal efficiencies as low as 78% efficiency. In addition, during a review of CAP reports associated with the SGTS charcoal filters, the inspectors noted that the No. 1 SGTS charcoal filter was replaced in June 2000. Efficiency testing had been conducted in December 1999, which was found acceptable at that time. However, the licensee determined on June 1, 2000, that they had failed to implement a committed action to use new acceptance criteria required by Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal." This resulted in operating for about six months with the No. 1 SGTS charcoal filter inoperable in 2000 due to efficiency being below 95%. This event was reported to the NRC in LER 50-219/2000-005 and is described in CAP O2000-0713. At that time, the licensee's corrective actions failed to include measures to identify degraded charcoal efficiency in order to prevent operation with inoperable SGTS filters.

Analysis. The deficiency associated with this finding is an inadequate identification of a problem so that timely corrective actions could be taken, which led to an unexpected failure of the charcoal filter bed when tested in October 2002 and led to an extended unavailability of the No. 2 SGTS. The finding was greater than minor because it had an actual impact in that the No. 2 SGTS was inoperable. The finding affects the barrier integrity cornerstone due to the impact on one of two trains of SGTS. The inspectors used IMC 0609 Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," regarding barrier integrity and determined that the finding only represented a degradation of the radiological barrier function provided for by the standby gas treatment system, and therefore screened to Green, very low safety significance.

This finding also had human performance and corrective action program deficiencies that are discussed in Sections 4OA2 and 4OA4 of this report. The human performance aspect involved a failure to ensure that the appropriate qualification test method was implemented for the SGTS charcoal efficiency tests in October and December 2002. The corrective action program deficiency involved the performance finding itself, failure to timely identify and correct a condition adverse to quality.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action requires in part that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, on May 1, 2001, the licensee failed to identify that the No. 2 SGTS charcoal filter efficiency could not be assured to remain operable for the next surveillance interval, a condition adverse to quality, and no actions were taken to either replace the filter or ensure its continued operability. Because the failure to identify and timely correct this condition is of very low safety significance and has been entered into the CAP (CAP O2002-1951), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy:

NCV 50-219/02-08-02, Failure to Timely Identify and Correct a Condition Adverse to Quality Regarding the Charcoal Filter Efficiency of the SGTS.

.2 Average Power Range Monitor #4 Flow Control Trip Reference Card Inoperability

a. Inspection Scope

During performance of the daily core power checks, on September 26, 2002, the licensee found, that Average Power Range Monitor (APRM) #4 Flow Control Trip Reference Card had an INOP status light blinking red (CAP O2002-1405). Based upon the INOP indication, the APRM was bypassed and declared inoperable by the control room operators. The licensee then assembled a prompt investigation team to evaluate, troubleshoot, and repair the problem. The inspectors verified the operability of the other APRMs and reviewed the licensee's risk assessment of the issue. Additionally, the inspectors interviewed the system engineer regarding the evaluation and reviewed the plans for troubleshooting, repair, and post-maintenance testing of the affected component. Adherence to technical specifications and abnormal procedures was also verified by the inspectors.

b. Findings

No findings of significance were identified.

.3 1A2 480V Breaker Trip

a. Inspection Scope

On October 9, 2002, with the plant shutdown and in the refueling mode of operation, the feeder breaker to Vital Motor Control Center 1A2 tripped open. This resulted in a trip of Reactor Protection System Channel 1, a loss of Turbine and Reactor Building Ventilation Systems, and closure of various primary containment isolation valves. The inspectors verified that the operators used the appropriate alarm response and plant operating procedures, responded to alarms in a timely manner, and communicated clearly during the transient and recovery. Additionally, the inspectors verified that technical specifications and reportability requirements were properly addressed, and the transient and its effects were appropriately analyzed and documented in CAP O2002-1536.

b. Findings

No findings of significance were identified.

.4 'A' Control Rod Drive Pump Failure and Repair

a. Inspection Scope

On October 24, 2002, the 'A' Control Rod Drive pump failed while in service. The licensee evaluation determined the failure was due to failure of the pump bearings. When the failure occurred, material from the failed bearing passed through one of three

40 micron pump discharge filters due to the filter collapsing at the bottom. This resulted in some bearing material passing into the Control Rod Drive System. The inspectors reviewed the operability evaluation (discussed in section 1R15) and risk assessment describing the overall effects of the failure on the system and the plant. Additionally, the inspectors verified that technical specifications were properly addressed, and the transient effects were appropriately analyzed and documented in the associated CAP, O2002-1709.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Offgas System Air In-Leakage Testing

a. Inspection Scope

On November 6, 2002, a loss of main condenser offgas event occurred. For this non-routine event, the inspectors observed operator actions, reviewed operator logs, plant computer data, and strip charts to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures.

b. Findings

Introduction. A Green self-revealing finding was identified involving an inadequate testing procedure that resulted in the inadvertent ignition of Hydrogen gasses contained in the offgas system during air in-leakage testing. The procedure failed to provide instructions to properly isolate and vent the test device sample chamber from the process stream before ionizing the test sample chamber.

Description. On November 6, 2002, maintenance troubleshooting testing was in progress to determine the source of air in-leakage into the “A” Main Condenser. The testing activity was being performed using test equipment on loan from another station and had not been used previously at Oyster Creek. The technicians conducting the test relied on guidance provided in their normal procedure, 2400-SMM-3302.05, “Main Condenser Leak Test.” However, this guidance was not revised to account for the use of the new test instrument. In addition, in lieu of a vendor manual/general operating guidance being sent to Oyster Creek with the instrument, a site specific procedure from the station that loaned the test instrument was sent with the equipment to be used as guidance by Oyster Creek staff. During setup, the technicians noted that the test instrument would vent to the general area and decided to close the vent path. This prevented the instrument from establishing an appropriate vacuum condition in the sample chamber. When the technician subsequently placed the instrument in test mode a flash was noted that ignited the offgas Hydrogen gasses. This was due to the test chamber then having a mixture of Hydrogen and Oxygen exposed to a high temperature heater element. The ignited gas traveled back through the sample line to the main offgas line, which then ignited causing a loss of the offgas system. The loss of the offgas system resulted in degrading main condenser vacuum conditions that the plant operators responded to in order to prevent a reactor scram.

The resident inspector was in the control room at the time of the event and observed operator response to these conditions. The inspector noted that operators were successful in following abnormal operating procedures and properly restored the offgas system, avoiding the need to manually shutdown the plant or risk a reactor scram event. Also, the inspector determined that plant operators appropriately followed the emergency plan (EP) and declared an Unusual Event at 2:28 p.m. per the EP event classification guide and made timely notification to external response organizations, including the State of New Jersey and the NRC. The inspector observed additional activities undertaken by the licensee in response to the offgas ignition to ensure that plant equipment had not been damaged. The licensee found that the offgas radiation monitoring system had been adversely affected involving a loss of sample flow due to the buildup of moisture resulting from the hydrogen/oxygen combination. This was considered a functional failure of the offgas radiation monitoring system and a 72 hour limiting condition for operation was entered. The offgas radiation monitors were not capable of continually monitoring plant effluents for about 3.5 hours. During that time the stack monitor was available to monitor plant effluents. After the licensee completed its assessment of the structural integrity of the offgas system piping, the Unusual Event was terminated at 6:55 p.m.

Analysis. A self-revealing, Green finding was identified due to use of equipment without appropriate procedures. This finding is greater than minor because it had an actual impact of igniting the offgas system hydrogen gas, causing the main condenser offgas system to be isolated and therefore could be viewed as a precursor to a significant event. If the offgas system could not have been quickly restored, it would have caused a reactor scram. The inspectors assessed the significance of this finding using IMC 0609 Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations, Phase 1 screening for the initiating events cornerstone. The finding is of very low significance because the event did not contribute to the likelihood of a LOCA initiator, all mitigation systems were available during this event to respond to any resultant reactor scram condition, the hydrogen ignition did not result in significant damage to the plant noting that the offgas system design was intended to successfully mitigate the consequences of this type of hydrogen ignition, and operators restored the offgas system before main condenser vacuum degraded to a reactor scram condition.

This finding also has a related causal factor of human performance, in that technicians proceeded with the test performance without an appropriate station procedure. In addition, the procedure guidance used from the station supplying the equipment was not adhered to resulting in the equipment being operated without the appropriate vacuum condition in the test sample chamber. The inspectors reviewed CAP O2002-1775 and verified that the licensee had properly evaluated the cause of the event and the related human performance causal factor.

Enforcement. The inadequate test procedure was associated with equipment that is not subject to the procedural administrative controls listed in Technical Specification 6.8.1, nor was the associated equipment under test subject to the quality assurance requirements in 10 CFR 50 Appendix B. Therefore, no violation of regulatory requirements occurred. Therefore, this issue is being treated as a Finding (FIN): FIN 50-219/02-08-06, Inadequate Procedural Guidance and Personnel Performance Resulting in a Plant Event.

1R15 Operability Evaluations

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 where a component was determined to be inoperable, the inspectors verified that the Technical Specification limiting condition for operation were properly addressed. The selected samples are listed below.

- Standby Gas Treatment (SBGT) - during the performance of surveillance test 651.4.001, SBGT System Test, on October, 14, 2002, the licensee found that reactor building vacuum could not be maintained within the surveillance acceptance criteria. The operability evaluation found the system to be operable and is described in Action Request (AR) A2043351.

- Emergency Diesel Generator (EDG) No.1 - during the performance of surveillance test 636.2.001, EDG Auto Actuation Test, on October 21, 2002, the licensee found that the EDG output breaker closing time could not meet the surveillance acceptance criteria. The operability evaluation found the system to be operable but degraded and is described in AR A0708054 and CAP O2002-1662.

- Control Rod Drive System (CRD) - on October 24, 2002, the ‘A’ CRD pump bearing failed causing foreign material from the pump to collect at the pump discharge filter. Some of the foreign material passed through a collapsed portion of the filter and into the CRD system. The operability evaluation of the CRD system found that the debris would have no effect on the scram function of the control rods and is described in CAP O2002-1709.

- EDG No. 2 - during the performance of surveillance test, 636.4.013, “Emergency Diesel Generator No. 2 Load Test,” the load increased from 2662 to 3272 KW in 10 seconds, dropped to 936 KW in eight seconds, and returned to 2662 KW in the next 14 seconds. The licensee determined that the cause of the load swing was a momentary failure of the speed signal from the magnetic pickup. This degraded speed circuit would not adversely affect the EDG performance during emergency start conditions, as the mechanical governor would then ensure EDG load and frequency as designed. The operability evaluation found that the electronic speed circuit was degraded but operable and would not affect the safety function of the EDG, as described in CAP O2002-1798.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

.1 Emergency Service Water Pump 52B

a. Inspection Scope

The inspector reviewed the surveillance testing data associated with procedure 607.4.004, “Containment Spray and Emergency Service Water (ESW) System 1 Pump Operability Test” conducted November 13, 2002. The inspectors verified that component operability was reestablished following a necessary revision to the procedure pump flow acceptance criteria limits by AmerGen engineering personnel.

b. Findings

Introduction. A Green, self-revealing NCV was identified for failure to have an adequate surveillance procedure in accordance with technical specification 6.8.1, which resulted in the 52B pump being incorrectly declared inoperable.

Description. On November 13, 2002, AmerGen conducted quarterly surveillance test 607.4.004, “Containment Spray and Emergency Service Water System 1 Operability and Inservice Test.” During the test, pump flow on ESW pump 52B exceeded the upper action limit value and was declared inoperable. AmerGen entered Technical Specification 3.4.C.4 due to the inoperable pump and issued CAP O2002-1808 to address the issue. An evaluation by AmerGen engineering determined that a recent revision to the test procedure erroneously changed the acceptance criteria that required use of precision test gages to satisfy the biennial comprehensive test requirements of the ASME code. The test performed on November 13, 2002, was not intended to satisfy the biennial ASME code requirements. Also, the technicians did not use the necessary precision test gages during the test. Therefore, the acceptance criteria was incorrect for the quarterly test per procedure 607.4.004. As a result, licensee engineering provided an operability determination to the plant operators and then revised the acceptance criteria in the procedure reflecting the correct acceptance criteria. Once verified by operations personnel that ESW pump 52B flow values were within acceptable limits, the Limiting Condition for Operation was exited.

Analysis. Procedure 607.4.004 was inaccurate with respect to ESW pump flow acceptance criteria, leading to the 52B ESW pump being incorrectly declared inoperable. The pumps remained available during the period of technical specification inoperability and the system remained capable of fulfilling its safety function. The inspector reviewed the engineering documentation and verified that the revised acceptance criteria was met. The finding is considered more than minor because it is associated with the Mitigating Systems cornerstone attribute of procedure quality and affects the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. The inspectors used IMC 0609 Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” regarding mitigating systems and determined that the finding was a qualification deficiency confirmed to not result in the loss of the safety function of the

Emergency Service Water System, and therefore screened to Green, very low safety significance.

Enforcement. Technical Specification 6.8.1 in part states that written procedures shall be established, implemented, and maintained for surveillance and test activities of equipment that affect nuclear safety. Contrary to the above, the licensee's failure to prescribe the correct ESW pump flow acceptance criteria in Procedure 607.4.004, could have resulted in increased unavailability of the pump to perform unwarranted corrective maintenance due to the test failure. Because the failure to maintain the ESW pump surveillance procedure is of very low safety significance and has been entered into the CAP (O2002-1808), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-219/02-08-03, Failure to Maintain a Technical Specification Surveillance Test Procedure.

.2 Primary Containment Isolation Valves

a. Inspection Scope

The inspectors reviewed and observed portions of the post-maintenance testing associated with Primary Containment Isolation Nitrogen Purge Valves, V23-14 and V23-15. The valves were restored to operability in accordance with procedure, 312.9-9, "Primary Containment Control," performed on October 1, 2002. The inspectors reviewed the post-maintenance test document to verify that component operability was reestablished as specified by the procedure and that test data was complete and valid.

b. Findings

No findings of significance were identified.

.3 'A' Main Steam Isolation Valve

a. Inspection Scope

The inspectors reviewed and observed portions of the post-maintenance testing associated with repair (Work Order #C20022648) of the Inboard Main Steam Isolation Valve (MSIV), V-1-7, on October 22, 2002. The test was performed during the refuel outage and the valve was initially declared inoperable on October 11, 2002, following the failure to meet the Local Leak Rate Test (LLRT) requirements specified in surveillance test procedure, 665.5.003, "Main Steam Isolation Valve Leak Rate Test." The inspectors verified that the associated outboard MSIV was operable. The inspectors reviewed the post-test data to verify compliance with technical specifications and ensure operability requirements were met. The inspectors also reviewed the reportability requirements of the test failure and verified that AmerGen submitted a Licensee Event Report in accordance with 10 CFR 50.73 (LER 05000219/2002-002). The LER is discussed in report Section 40A3.

b. Findings

No findings of significance were identified.

.4 Primary Containment Isolation Logic

a. Inspection Scope

During the performance of surveillance test, 610.3.115, “Core Spray System Instrument Channel and Level Bistable Calibration,” on November 4, 2002, an unexpected primary containment isolation signal occurred for torus vent valves, V-28-18 and 47. The valves were closed (normal position) when the signal occurred and could not be reopened due to the containment isolation signal. AmerGen I&C personnel performed troubleshooting on the logic circuitry (Action Request A2047440) and determined one of the relay contacts in the circuit had high resistence. The relay was burnished and the circuit was retested under surveillance test procedures, 610.3.115 and 312.9, “Primary Containment Control.” The inspectors reviewed the post-maintenance test documents to verify that component operability was reestablished as specified by the procedure and that test data was complete and valid.

b. Findings

No findings of significance were identified.

.5 <u>"B" Control Room Heating, Ventilation, and Air Conditioning (HVAC)</u>

a. <u>Inspection Scope</u>

On November 13, 2002, the No. 1 compressor for the "B" train of Control Room HVAC was found tripped off by operations personnel. AmerGen entered Technical Specification 3.17.B.1 due to the inoperability of the equipment and notified the I&C department to begin troubleshooting the problem. Maintenance personnel replaced the No. 1 compressor under Action Request, A2041849, and also repaired a ground on the No. 2 compressor #2 motor winding. Work was completed on November 30, 2002, and the "B" Control Room HVAC was declared operable following the post-maintenance test. The inspectors reviewed the results of procedure, 654.4.003, "Control Room HVAC System Operability Test," to verify that component operability was reestablished as specified by the procedure and that test data was complete and valid.

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Refueling and Outage Activities</u>

.1 <u>Routine Outage Activities</u>

a. <u>Inspection Scope</u>

The inspectors reviewed and/or observed various risk significant activities associated with the refueling outage, which began on October 4, 2002, and ended on October 27, 2002. 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 was in the correct position as described by the associated tag.
- Toured spaces normally inaccessible during power operation including the Trunnion Room on October 8 and 18, the Drywell on October 14, 15, 17, and 23, the Main Turbine Deck on October 10 and 22, and the Condenser Bay on October 7, 8, 10, and 18.
- 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.

.2 Nuclear Steam Supply System Leak Test (Reactor Vessel Hydrostatic Test)

a. Inspection Scope

On October 4, 2002, Oyster Creek performed test procedure, 602.4.001, “Nuclear Steam Supply System Leak Test.” The inspectors reviewed previous test results and observed pre-test briefings, control room communications, and verified procedural adherence. In addition, the inspectors performed a drywell inspection on 23' elevation and an under vessel inspection during the test.

During the under vessel inspection, the inspectors noted several leaks, which were also identified and documented by licensee maintenance personnel. The leakage was identified and quantified as Control Rod Drive Mechanism (CRDM) flange leakage on 15 CRDMs. This problem was described in CAP O2002-1465, which was reviewed for accuracy by the inspectors. The inspectors also verified that the leakage was within technical specification limitations and that maintenance requests were written to disposition and correct the leakage (ARs A20444386, A2013991). Additionally, no CRDM ”above flange” leakage or other under vessel leakage was noted by the licensee during the test.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector observed pre-test briefings and portions of surveillance test (ST) performance for procedural adherence, and verified that the resulting data associated with the test met the requirements of the plant technical specifications and the OC Updated Final Safety Analysis Report. The inspector also reviewed the results of past test performance of the selected STs to verify that degraded or non-conforming conditions were identified and corrected, if needed. The following STs were observed:

- Station Blackout Functional Test - surveillance procedure 678.4.005, performed on October 5, 2002.
- Primary Containment Isolation Valve Test - surveillance procedure 619.3.004, performed on October 22, 2002
- Isolation Condenser Isolation Test - surveillance procedure 609.3.002, performed on October 31, 2002
- Grid Undervoltage Test - surveillance procedure 632.2.002, performed on November 14, 2002

b. Findings

Introduction. A Green, self-revealing, NCV was identified during the performance of the primary containment isolation valve test on October 22, 2002, for failure to maintain the secondary containment configuration in accordance with Technical Specification 3.5.B.2, which resulted in a temporary loss of secondary containment.

Description. On October 22, 2002, with the unit in a shutdown condition and secondary containment in an operable condition per the plant technical specifications, primary containment isolation valve testing was in progress at about 12:30 p.m. per procedure 619.3.004. As part of the test procedure setup, the No. 1 Standby Gas Treatment System was in service maintaining reactor building negative pressure. Operators noted that the standby gas treatment system was not maintaining the reactor building negative pressure per the design of about - 0.25 inches (water). An investigation was conducted and the licensee determined that the trunnion room door was in an open position. Per the technical specifications, the trunnion room door is administratively controlled and maintained in a closed position whenever secondary containment is required to be operable. It was further determined by the licensee that the trunnion room door had been propped open with a broomstick handle at about 12:45 p.m., on October 22. This was done by a non-licensed operator during the removal of a safety tag (clearance) on equipment located in the trunnion room.

Analysis. When the plant operators noted that the standby gas treatment system could not maintain reactor building negative pressure properly, they immediately began an investigation. Since it was known at the time that plant operators were removing a clearance in the trunnion room, an operator was dispatched to ascertain the trunnion room door position. Finding it opened, the operators closed the trunnion room door and restored secondary containment to a fully operable condition. The inspector verified that the operators restored secondary containment within the four hours permitted by the plant technical specifications. The door apparently was only propped open for about 15 minutes. The licensee took additional actions, including posting both sides of the trunnion room door to ensure that plant personnel would notice that the door was to be maintained closed, and both plant operators involved in the clearance operation were briefed about the human performance aspects of this event, since the pre-evolution briefing included a discussion about ensuring that the door was maintained closed.

The inspector determined that the surveillance test revealed a configuration control problem with secondary containment that was caused by a human performance deficiency in the conduct of field operations to restore equipment to a normal condition. The trunnion room door lineup deficiency was considered self-revealing because it was outside the scope of the primary containment isolation valve surveillance test, even though the licensee identified the problem. The inspector verified that the licensee was capable of closing the trunnion room door even if conditions in the containment environment had degraded due to any radiological release.

The inspectors determined that the performance deficiency was more than minor because the finding involved the reactor safety barrier integrity cornerstone attribute of human performance for the functionality of the containment, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events. The inspectors used IMC 0609 Appendix G, “Significance Determination of Reactor Inspection Findings for Shutdown Operations,” for Containment Guidelines involving a Boiling Water Reactor (BWR) in a Cold Shutdown condition with time to boil being greater than 2 hours and reactor coolant system level less than 23 feet above the top of reactor flange. Using this guidance the inspector verified that the finding involved a situation in that secondary containment closure can be accomplished in sufficient time before a release of fission products, including unavailability of AC power and the expected environmental condition in containment. Therefore, the finding screened to Green, very low safety significance.

This finding also had a human performance aspect that is discussed in Section 4OA2 of this report.

Enforcement. Technical Specification 3.5.B requires that the trunnion room door is to be maintained closed except that it can be opened for a short time under administrative control. Contrary to the above, plant operators opened the trunnion room door and left it open without establishing appropriate administrative controls to ensure that it would be closed. This problem was only revealed because the SGTS was in operation at the time and could not maintain reactor building negative pressure per its design, since the trunnion room door was opened, allowing the reactor building to communicate directly with the turbine building environment. Because this failure to maintain the trunnion room closed is of very low safety significance and has been entered into the CAP (O2002-1680), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-219/02-08-04, Failure to Maintain Secondary Containment Configuration - Trunnion Room Door Open Without Administrative Control.

1R23 Temporary Plant Modifications

a. Inspection Scope

On October 20, 2002, the inspectors reviewed a temporary modification installed under AR, A2040178. The temporary modification was to seal a small hole, approximately 2" X 1½," on the Service Water System piping downstream of the Reactor Building Closed Cooling Water System heat exchangers. The hole in the piping caused in-leakage of reactor building air into the Service Water System due to a siphon effect and resulted in a degradation of secondary containment. The inspectors walked down the temporary modification installation and reviewed the associated 10 CFR 50.59 screening, system procedures, technical specifications, the associated sections of the UFSAR, and the evaluation package (CAP O2002-1059). Additionally, the inspectors verified that the modification was performed in accordance with OC Procedure, 108.8, “Temporary Modification Control.”

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 <u>Access Control to Radiologically Significant Areas</u>

a. <u>Inspection Scope</u>

The inspector reviewed radiological work activities and practices and procedural implementation during observations and tours of the facilities and inspected procedures, records, and other program documents to evaluate the effectiveness of Oyster Creek's access controls to radiologically significant areas.

The inspector conducted tours in various parts of the facility to verify the adequacy of the radiological controls which were being implemented during the ongoing refueling outage (1R19). On October 14, 2002, the inspector toured the protected area outside the Radiologically-Controlled Area (RCA) and observed RCA entries and exits being made by radiation workers for outage work activities at the drywell processing facility. The inspector reviewed and discussed, with the radiological-protection technicians, the use of a drywell access sheet implemented to provide for radiological control. At 7:30 a.m., on October 15, the inspector observed the radiological-protection technician shift turnover at the drywell processing facility. Also, the inspector observed a radiological pre-job brief for radiation workers preparing to enter the drywell to perform demobilization activities. Afterwards, the inspector entered the drywell and observed selected work activity and associated radiological controls for compliance with Radiological Work Permit (RWP) requirements on the 13-foot, 23-foot, and 46-foot elevations of the drywell. In the afternoon, the inspector went to the refueling floor of the reactor building and observed work activities at the control point and on the refueling floor. At 7:00 p.m., the inspector observed the radiological protection management shift turnover. On October 16, the inspector toured inside the condenser bay of the turbine building including the trunnion room and the Health Physics (HP) control point for this area. Later, the inspector toured the turbine building operating floor and observed work activities there, including work on the 'B' low pressure turbine with its outer shell removed and the lowering of a portable sand blast enclosure over a removed low pressure turbine rotor. On October 17, the inspector visited the HP radiological instrument calibration trailer.

During these observations and tours the inspector reviewed, for regulatory compliance, the posting, labeling, barricading, and level of radiological access control for locked high radiation areas (LHRAs), high radiation areas (HRAs), radiation and contamination areas, and radioactive material areas. The inspector observed activities, at the main RCA access control point and at satellite RCA access control points, to verify compliance with requirements for RCA entry and exit, including proper wearing of record dosimetry, and issuance and use of alarming electronic radiation dosimeters. The inspector reviewed observed work activities for compliance with the RWP requirements.

The inspector performed a selective examination of RWPs, procedures, and other program documents (as listed in the List of Documents Reviewed section) to evaluate the adequacy of radiological controls.

The review in this area was against criteria contained in 10 CFR 19.12, 10 CFR 20 (Subparts D, F, G, H, I, and J), site Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed the effectiveness of the licensee's program to maintain occupational radiation exposure as low as is reasonably achievable (ALARA).

On October 16, 2002, the inspector reviewed selected ALARA plans which had been generated for various outage job activities. Based on the observations made on tours described in Section 2OS1 of this report, the inspector verified that selected controls, enumerated in the ALARA plans, had been adequately implemented.

Based on outage-dose and annual-dose tracking data available at the time of the inspection and through discussions with the Oyster Creek Radiation Protection Manager, the inspector noted that actual outage and annual cumulative doses (year-to-date) were running below the original estimates.

The inspector performed a selective examination of procedures, records, and documents (as listed in the List of Documents Reviewed section) for regulatory compliance and for adequacy of control of radiation exposure. The review in this area was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701 (Use of process or other engineering controls), and site procedures.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed the program for health physics instrumentation to determine the accuracy and operability of the instrumentation.

During the plant tours described in Section 2OS1, the inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity and radiation levels, including portable field survey instruments, hand-held contamination frisking instruments, whole-body friskers, and portal monitors. The inspector conducted a selective review of the instruments observed in the toured areas, specifically for verification of current calibration, of appropriate source checks, and of proper function. The inspector also reviewed the calibration process for the Siemens electronic personal dosimeters which had recently been placed into service and also reviewed selected calibration records for these dosimeters.

The inspector performed a selective examination of procedures, records, and documents (as listed in the List of Documents Reviewed section) for regulatory compliance and adequacy in this area.

The review in this area was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, site Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

2PS2 Radioactive Material Processing and Transportation

a. Inspection Scope

The inspector reviewed the radioactive material processing and transportation work activities and practices during tours of the facilities, discussed observations and issues with Exelon representatives, and inspected procedures, procedural implementation, records, and other program documents to evaluate the effectiveness of the performance in this area.

Radioactive waste system walkdown

On December 10, 2002, the inspector conducted a walkdown of accessible portions of the station’s radioactive liquid and radioactive solid waste collection, processing, and storage systems/locations with the radioactive material/waste shipping supervisor. The purpose of this walkdown was to selectively verify that the current system configuration and operation agreed with descriptions contained in the Updated Final Safety Analysis Report (UFSAR) and in the Process Control Program (PCP). This tour included the reactor building (RB), the old radioactive waste building (ORW), the new radioactive waste building (NRW), and the low level radwaste storage facility (LLRWSF). On December 12 and 13, 2002, the inspector again toured the above-mentioned facilities.

During these walkdowns and discussions with Exelon representatives, the inspector reviewed the status of non-operational or abandoned in-place radioactive waste process equipment and administrative and physical controls for the systems, evaluated any changes made to radioactive waste processing systems and the potential radiological impact, and reviewed the current processes for transferring radioactive waste resin and sludge to shipping containers and for resin/sludge dewatering.

Waste characterization and classification

The inspector reviewed applicable procedures on December 9 -12 and analysis and scaling factor records on December 10, 2002. The inspection included a selective review of the waste characterization and classification program for regulatory compliance, including the following items:

- the radio-chemical sample analysis results for radioactive waste streams,
- the development of scaling factors for difficult-to-detect-and-measure radio nuclides,
- the methods and practices to detect changes in waste streams as described in the PCP, and
- the methods and practices to determine waste classification (10 CFR 61.55) and to determine DOT shipment subtype (49 CFR 473).

Shipment preparation

The inspection included a review of radioactive waste program documents, shipment preparation procedures, and activities for regulatory compliance, including the following:

- observation on December 9, 2002 of a dewatering evolution on a high integrity container (HIC) in the NRW building,
- observation on December 11, 2002 both of testing for the status of HIC dewatering and of the capping of a HIC,
- observation and verification on December 12, 2002 of the proper marking and labeling on two drums (Type A containers) prepared for shipment (shipment no. OC-0402-02),
- review on December 12, 2002 of the certification and documentation for a specification Department of Transportation (DOT) Type A container and verification of appropriate NRC license authorization of shipment recipients for the shipments selected for inspection,
- review on December 10, 2002 of the radioactive material shipping logs for the calendar years of 2001and 2002,
- review on December 11 and 12, 2002 of the training provided to appropriate personnel in accordance with NRC Bulletin 79-19 and 49 CFR Part 172 Subpart H, and
- discussions concerning regulatory requirements with the radioactive waste shipping supervisor.

Shipping records

On December 10 and 12, 2002, the inspection involved a review of the six non-excepted package shipment records (as listed in the List of Documents Reviewed section) for compliance with NRC and DOT requirements, including shipment papers and description requirements, shipper’s certification, proper use of forms, package marking and labeling, vehicle placarding, emergency response information, and packaging requirements.

Identification and resolution of problems

In the area of identification and resolution of problems, the inspection included a selective review of audit/surveillance activities by Nuclear Oversight and self-assessments (as listed in the List of Documents Reviewed section). This review involved material related to the radioactive material and transportation programs since the previous inspection and a determination of whether identified problems were entered into the corrective action program for resolution. The review of related issues in the corrective action program is covered in Section 4OA2 of this report.

During the review of the five areas listed above under inspection scope, the inspector performed a selective examination of procedures, records, and documents (as listed in the List of Documents Reviewed section) for regulatory compliance and adequacy.

The above review in this section was against criteria contained in 10 Code of Federal Regulations (CFR) Part 20 Subpart F (Surveys and monitoring), 20.1902 (Posting requirements), Subpart I (Storage and control of licensed material), Subpart K (Waste disposal), Appendix G to Part 20 (Requirements for transfers of low-level radioactive waste intended for disposal at licensed land disposal facilities and manifests), 10 CFR 61.55 Waste classification, 61.56 Waste characteristics, 61.57 Labeling, 10 CFR 71 Packaging and transportation of radioactive material, 49 CFR Part 172 (Hazardous materials table, special provisions, hazardous-materials communications, emergency response information, and training requirements), Part 173 (Shippers-general requirements for shipments and packaging), Subpart I (Class 7 (radioactive materials)), Part 177 (Carriage by public highway), NRC Bulletin 79-19, and site procedures.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection (PP)

3PP3 Response to Contingency Events

The Office of Homeland Security (OHS) developed a Homeland Security Advisory System (HSAS) to disseminate information regarding the risk of terrorist attacks. The HSAS implements five color-coded threat conditions with a description of corresponding actions at each level. NRC Regulatory Information Summary (RIS) 2002-12a, dated August 19, 2002, “NRC Threat Advisory and Protective Measures System,” discusses the HSAS and provides additional information on protective measures to licensees.

a. Inspection Scope

On September 10, 2002, the NRC issued a Safeguards Advisory to reactor licensees to implement the protective measures described in RIS 2002-12a in response to the Federal government declaration of threat level “orange.” Subsequently, on September 24, 2002, the OHS downgraded the national security threat condition to “yellow” and a corresponding reduction in the risk of a terrorist threat.

The inspector interviewed licensee personnel and security staff, observed the conduct of security operations, and assessed licensee implementation of the threat level “orange” protective measures. Inspection results were communicated to the region and headquarters security staff for further evaluation.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed the Oyster Creek performance indicator (PI) data against applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, to verify that all conditions that met the NEI criteria were recognized and identified as PI occurrences. The inspectors verified the accuracy of the reported data through reviews of monthly operating reports, shift operating logs, Licensee Event Reports (LERs), security event log, corrective action program records involving the Residual Heat Removal (RHR) System, protected area security equipment, personnel screening, and fitness for duty, and records of occurrences involving high radiation areas, very high radiation areas, unplanned personnel exposure, and monthly and quarterly gaseous and liquid effluent release reports (significant records reviewed by the inspector are listed in the Attachment to this report). Except where noted below, the inspectors reviewed 12 months of reported data (from October 2001 - September 2002) for the following six PIs:

Safety System Unavailability - Residual Heat Removal Systems
Protected Area Security Equipment Performance Index
Personnel Screening Program Performance
Fitness-For-Duty/ Personnel Reliability Program Performance
Occupational Exposure Control Effectiveness (July 2001 - September 2002)
RETS/ODCM Radiological Effluent

The inspector’s review of records did not identify any significant problems with the PI accuracy or completeness and thus verified the reported PI data.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Annual Sample Review

a. Inspection Scope

The inspectors selected five corrective program reports for detailed review (CAPs O2001-1447, O2001-1718, O2001-0690, O2001-1024, and O2001-1155). These reports were associated with: an inadequate post-maintenance test for control rod 42-27, a medium voltage cable failure, an error in the 125 Volt DC voltage drop calculation, an inadequate assessment of plant risk for emergent work involving the isolation condensers, and a failure of plant staff to properly respond to an alarming self-reading dosimeter (SRD). The reports were reviewed to ensure that the full extent of the issue was identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized, and timely, effective resolution was taken. The inspectors evaluated the reports against the requirements of the licensee's CAP as described in procedure, LS-OC-125, "Corrective Action Program (CAP) Procedure," and 10 CFR 50, Appendix B.

b. Findings and Observations

There were two findings identified during this review associated with ineffective corrective actions for personnel response to alarming SRDs, and for inadequate performance of post-maintenance testing on control rod 42-27, as described in detail below.

Finding 1:

Introduction. A Green, self-revealing NCV was identified for failure to establish fully effective problem resolution relative to recurring problems involving personnel failing to hear the integrated dose alarm of their electronic self-reading personnel dosimetry equipment and to promptly respond to such an alarm.

Description. Corrective Action Program reports, identified as CAP 2001-1155 (dated July 18, 2001), CAP 2002-1365 (dated September 18, 2002), and CAP 2002-1505 (dated October 6, 2002), were documented to address licensee-identified non-conformance with the Technical Specification requirement to provide personnel with a radiation monitoring device which alarms and alerts personnel when a pre-set integrated dose is received. The proposed corrective actions for CAP 2001-1155 (dated July 18, 2001) included, in part, a) the establishment and use of a Locked High Radiation Area/ALARA Briefing Checklist with a check off item for high noise areas and b) an action item to identify high noise areas and other tasks impacting audible alarms of electronic self-reading dosimeters and to look at options for supplementing the alarm volume of the current electronic self-reading dosimeter in use. However, as described in CAP 2002-1365 (dated September 18, 2002), a station worker failed to hear the dose alarm on his electronic self-reading dosimeter while working in a high radiation area. Further, as described in CAP 2002-1505 (dated October 6, 2002), a contracted worker failed to hear the dose alarm on his electronic self-reading dosimeter while working in an area conservatively posted as a locked high radiation area. The highest dose rate in this area was less than 1000 but greater than 100 millirem per hour.

Analysis. These issues constituted performance deficiencies in that they resulted in requirements not being met, and the issues should have been prevented. These issues were not of the type to be addressed by traditional enforcement. These repetitive events were more than minor in that worker safety could be impacted in similar circumstances if workers failed to properly respond to alarming dosimeters in situations with the potential for unplanned radiation dose. The inspectors used IMC 0609, Occupational Radiation Safety Significance Determination Process, and identified that the issues were not ALARA findings, did not result in any overexposure, did not create a substantial potential for any overexposure, and did not compromise the licensee's ability to assess dose to workers. None of these three incidents resulted in any significant unintended exposure. Therefore, the issues were determined to be of very low significance (GREEN).

Enforcement. Oyster Creek Technical Specification 6.13 requires that an individual, permitted to enter a High Radiation Area, shall be provided with one or more of three options, namely: a) a radiation monitoring device which continuously indicates the radiation dose rate in the area, b) a radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received, or c) a health physics-qualified individual with a radiation dose rate monitoring device who is responsible for providing positive exposure control over the activities within the area and who will perform periodic radiation surveillance at the frequency in the RWP.

Contrary to the above, in this violation, demonstrated by the three examples described in CAPs 2001-1155, 2002-1365, and 2002-1505, the viable option provided (i.e., a radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received) was not effective since the alarm was not heard due to the noise levels/hearing difficulties in the work areas. And, the corrective actions, taken as a result of CAP 2001-1155, were not fully effective based on the occurrence of two similar incidents (CAPs 2002-1365 and 2002-1505) more than one year after the first incident (CAP 2001-1155).

AmerGen documented this issue in CAPs 2001-1155, 2002-1365, and 2002-1505. Additional corrective actions have been taken. Because this self-revealing violation was of very low safety significance and because AmerGen entered these issues into its corrective action program, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-219/02-08-07, Ineffective Resolution of Identified Problems with Personnel Response to Alarming SRDs.

Finding 2:

Introduction. A Green NCV was identified for failure to adequately implement post-maintenance testing for the Control Rod Drive System, as prescribed in procedure, 302.1, "Control Rod Drive System" Attachment 8.

Description. On September 9, 2001, Control Rod Drive System Hydraulic Control Unit (HCU) 42-27 was isolated from Control Rod drive system pressure and removed from service for maintenance to replace the operators on the scram inlet and outlet valves. The work performed on those valves impacts the scram function of the control rod. The control rod was fully inserted (notch 00) and was declared inoperable when the maintenance started. Oyster Creek Technical Specification 3.4.B.4. in part, states, control rods which cannot be moved with control rod drive pressure shall be considered inoperable. On September 14, 2001, the maintenance work was completed and HCU

42-27 was returned to service, and declared operable. No post-maintenance testing, as required by Control Rod Drive System procedure attachment 302.1-8, was performed prior to declaring the control rod operable.

On September 16, 2001, approximately 35 hours after the HCU was initially declared operable and following a review of the maintenance work performed on the HCU, it was determined by AmerGen that scram time testing and demonstration of control rod movement should have occurred as a post-maintenance test, prior to declaring the HCU operable (CAP No. 02001-1447, 02002-1778). Once identified by AmerGen that the required post-maintenance test was not done, the HCU was declared inoperable, the appropriate test was completed successfully, and the HCU was returned to an operable status. The inspectors verified that the post-maintenance test was performed in accordance with procedures and the results of the test were satisfactory.

Analysis. AmerGen failed to implement procedures to perform the post-maintenance test on control rod drive system HCU 42-47 prior to declaring it operable per technical specifications on September 14, 2002. The inspectors noted that the control rod remained at the full in position (notch 00) before, during, and after the performance of the maintenance work and no other control rods were inoperable during the 35 hour period before demonstrating the operability of control rod 42-47. Also, inspectors verified the post-maintenance test was satisfactory when performed 35 hours later.

The finding is considered greater than minor in that the issue was associated with the Mitigating Systems cornerstone and potentially affected the scram function of a control rod. The inspectors used IMC 0609, "Significance Determination Process," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," regarding mitigating systems and determined that:

- the finding did not represent an actual loss of the safety function for any mitigating system and did not result in a loss of function of a single train of any mitigating systems for greater than its TS allowed outage time;

- the finding did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant per 10 CFR 50.65 for greater than 24 hours;

- the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic, flooding or severe weather initiating event; and

- the finding did not involve the loss of a safety function that contributed to the external event initiated core damage accident sequences.

Therefore, the finding screened as Green, very low safety significance.

Enforcement. Oyster Creek Technical Specification 6.8.1 states, in part, that written procedures shall be implemented, as recommended in Appendix "A" of Regulatory Guide 1.33.The Control Rod Drive System procedure is listed in Appendix "A" of Regulatory Guide 1.33. Contrary to the above, post-maintenance test requirements, as specified in Oyster Creek procedure No. 302.1 attachment 8, Control Rod Drive System, were not implemented for HCU 42-27 prior to declaring the HCU operable. The procedure attachment specifically states that scram time testing is the required test to demonstrate component operability following the maintenance work which was performed. The scram time test was not successfully completed until 35 hours after the HCU was returned to service.

AmerGen documented this issue in CAPs O2001-1447 and 02002-1788.Because this violation was of very low safety significance and AmerGen entered this finding into its corrective action program, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-219/02-08-05, Failure to Perform a Required Post-Maintenance Test.

Observations Not Involving Findings:

In addition to the above two findings, the inspectors identified minor examples of inadequate or untimely evaluations to ensure continued equipment operability.

The first example is associated with the replacement of medium voltage cable. The inspector had previously identified an inadequate cable pulling tension calculation (reference IR 50-219/2002006) on cable 14-25. The calculation had not been revised prior to the actual cable pull that occurred during this inspection when the licensee experienced pulling tensions almost 300% higher than predicted. While the licensee's evaluation, documented in CAP O2002-1551, justified that no adverse affect on the copper conductor had occurred, it failed to consider the higher pulling tension affects on the cable insulation at the conduit bends. In response to this concern, AmerGen evaluated the higher side wall pressures per Action Request, A2021871, which determined its acceptability. In addition, although the post-installation insulation resistance polarization indices on cable 14-25 were below the acceptance criteria, the dielectric strength tests were acceptable indicating no immediate concern. Since the root cause evaluations and associated corrective actions appeared to be acceptable upon completion of the additional evaluations, no violation of regulatory requirements or findings were identified.

The second example involved an inadequate calculation for the 125 Volt battery. The inspector noted that the licensee had not updated the voltage drop calculation subsequent to the battery discharge testing conducted during the 19R outage on October 15, 2002. As a result, the licensee added a new corrective action item to CAP O2001-0690 to provide an updated evaluation for continued operability with the lower battery capacity. The evaluation was completed and showed that there was sufficient battery capacity to maintain operability. Since the root cause evaluations and associated corrective actions appeared to be acceptable upon completion of the additional evaluations, no violation of regulatory requirements or findings were identified.

.2 Cross-References to PI&R Findings Documented Elsewhere

Section 1R08 describes a finding for failure to identify that three pipe fittings installed in the isolation condenser system per modification ECR 01-00828 did not meet the required material specifications. The licensee had a prior opportunity to identify this finding, which may be indicative of a potential deficiency in the licensee's design control process.

Section 1R13 describes a finding for failure to identify that the No. 2 SGTS charcoal filter efficiency had degraded in May 2001. As a result, corrective actions were not scheduled. In October 2002, during the next regularly scheduled charcoal efficiency test, it was determined that the filter was inoperable. The licensee had prior opportunity to identify and correct this degraded condition, which may be indicative of a potential deficiency in the licensee's component monitoring process.

Section 4OA2.1 describes a finding for failure to effectively resolve a problem associated with workers not adhering to the requirements to leave the RCA whenever their SRD alarms. In October 2002, the inspector identified that corrective actions implemented for an event in 2001 had not been fully effective since repeat events had occurred after completing the initial corrective measures. The licensee had prior opportunity to resolve this condition, which may be indicative of a potential deficiency in the licensee's CAP resolution process.

4OA3 Event Follow-up

.1 (Closed) LER 50-219/02-002-00, Local Leak Rate Test Results in Excess of Technical Specification Limits

On October 11, 2002, during local leak rate testing, Inboard Main Steam Isolation Valve (MSIV) V-1-7 exceeded the technical specification leak rate limit. The risk significance of the test failure was considered minimal by AmerGen because the other MSIV in the same steam header met the leak rate requirements of technical specifications and the leakage provided adequate margin between projected offsite dose and 10 CFR 100 guidelines. AmerGen repaired the valve under Work Order #C2002648. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the test failure in CAP No. O2002-1557. This LER is closed.

4OA4 Cross-cutting Issues Involving Human Performance

AmerGen's failure to prevent the installation of material prohibited by engineering specification in a safety-related system is considered a performance deficiency. The finding adversely impacted the stress analysis used in-part to approve the modification. In addition, this finding had a human performance aspect, in that a direct cause of the event was a failure to adhere to engineering guidance during the implementation of the modification, which was determined to be a non-cited violation of 10 CFR 50 Appendix B Criterion V. (Section 1R08)

AmerGen's failure to ensure that appropriate procedural guidance was available and adhered to for the air in-leakage testing of the main condenser, which precipitated an ignition of the hydrogen gas in the offgas system is considered a performance deficiency. This finding had a human performance aspect, in that a direct cause of the event was that the technicians proceeded to perform the test without an appropriate procedure. (Section 1R14)

AmerGen's failure to ensure that the Trunnion Room door is maintained closed, except for short durations to allow passage resulted in a loss of secondary containment configuration control and is considered a performance deficiency. This finding had a human performance aspect in that facility operators were briefed about the need to maintain the door closed except for passage prior to implementing a safety tag clearance that necessitated entry into the trunnion room. Normally this room is locked closed. The operators propped the door open in order for the activity to be completed more easily, which included a second verification of the tags being released and restored to normal condition. During this time reactor building negative pressure could not be maintained appropriately by the SGTS in service. (Section 1R22)

4OA5 Other Activities

a. Inspection Scope

An audit of the licensee's performance of the interim compensatory measure imposed by the NRC's Order Modifying License, issued February 25, 2002 was completed in accordance with the specifications of NRC Inspection Manual Temporary Instruction (TI) 2515/148, Revision 1, Appendix A, dated September 13, 2002.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 23, 2002, the resident inspectors presented the inspection results to Mr. E. Harkness 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 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Licensee (in alphabetical order)
V. Aggarwal, Director, Engineering
E. Harkness, Vice President
M. Massaro, Plant Manager
R. Hillman, Manager, Chemistry & Radwaste
J. Magee, Director, Maintenance
D. McMillan, Director, Training
M. Newcomer, Senior Manager, Design
D. Slear, Manager, Regulatory Assurance
C. Wilson, Senior Manager, Operations

b. List of Items Opened, Closed, and Discussed

Opened and Closed

05-219/2002-002-00	LER	Local Leak Rate Test Results in Excess of Technical Specification Limits for Inboard Main Steam Isolation Valve V-1-7 (Section 4OA3.1)
05-219/02-08-01	NCV	Failure to adhere to specified material requirements for installation of safety related components. (Section 1R08)
05-219/02-08-02	NCV	Failure to timely identify and correct a condition adverse to quality regarding the No. 2 SGTS charcoal filter efficiency. (Section 1R13)
50-219/02-08-03	NCV	Failure to maintain surveillance test procedure 607.4.004, Containment Spray/Emergency Service Water System 1 Pump Operability. (Section 1R19.1)
50-219/02-08-04	NCV	Failure to maintain secondary containment configuration - Trunnion Room Door open without Administrative Control. (Section 1R22)
05-219/02-08-05	NCV	Failure to conduct required post-maintenance testing for Control Rod Hydraulic Control Unit 42-27. (Section 4OA2.1)
05-219/20-08-06	FIN	Inadequate procedural guidance and personnel performance during testing resulting in a plant event. (Section 1R14)

05-219/02-08-07	NCV	Ineffective Resolution of Identified Problems with Personnel Response to Alarming SRDs.(Section 40A2.1)

c. List of Acronyms

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ARs	Action Requests
BWR	Boiling Water Reactor
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
DOT	Department of Transportation
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
EP	Emergency Plan
ESW	Emergency Service Water
HCU	Hydraulic Control Unit
HIC	High Integrity Container
HP	Health Physics
HRA	High Radiation Area
HSAS	Homeland Security Advisory System
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation & Controls
IMC	Inspection Manual Chapter
IVVI	In-Vessel Visual Inspection
JO	Job Order
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LLRT	Local Leak Rate Test
LLRWSF	Low Level Radioactive Waste Storage Facility
LSA	Low Specific Activity
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRW	New Radioactive Waste building
OC	Oyster Creek
ODCM	Offsite Dose Calculation Manual
OHS	Office of Homeland Security
ORW	Old Radioactive Waste building

OS	Occupational Safety
PARS	Publicly Available Records
PCP	Process Control Program
PI	Performance Indicator
PS	Public Radiation Safety
PT	Penetrant Testing
RB	Reactor Building
RCA	Radiologically Controlled Area
RETS	Radiological Effluent Technical Specification
RHR	Residual Heat Removal
RIS	Regulatory Information Summary
RWP	Radiation Work Permit
SCO	Surface Contaminated Object
SGTS	Standby Gas Treatment System
SRD	Self-Reading Dosimeter
SSCs	Structures, Systems, and Components
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing

d. List Of Documents Reviewed

Procedure 101.3, Rev. 42, Shipment of radioactive materials, nonwaste and waste shipment
Procedure 351.68, Rev. 11, Low level radwaste storage facility (LLRWSF) - Receipt, transfer, storage, and shipment of radioactive waste/reusable equipment
Procedure 351.69, Rev. 5, Transport cask handling procedure
Procedure 351.72, Rev. 3, Control rod drive (CRD) packaging for offsite processing or burial
General Electric CRDM rebuild procedure/CRDM shipping box loading, Rev. 0
Procedure 352.0, Rev. 34, Process control plan for processing filter media and resins using SEG supplied equipment
RP-AA-100, Rev. 2, Process control program for radioactive wastes
RP-AA-600, Rev. 5, Radioactive material/waste shipments
RP-AA-600-1001, Rev. 0, Exclusive use and emergency response information
RP-AA-600-1002, Rev. 0, Highway route controlled quantity/advance notification for radioactive/ waste shipments
RP-AA-600-1003, Rev. 0, Radioactive waste shipments to Barnwell and defense consolidation facility (DCF)
RP-AA-600-1004, Rev. 0, Radioactive waste shipments to Envirocare
RP-AA-600-1005, Rev. 0, Radioactive material and nondisposal site waste shipments
RP-AA-602, Rev. 5, Packaging of radioactive material shipments
RP-AA-602-1001, Rev. 0, Packaging of radioactive material/waste shipments
RP-AA-603, Rev. 2, Inspection and loading of radioactive material shipments
RP-AA-603-1001, Rev. 0, Inspection and loading of radioactive material/waste shipments
RP-AA-605, Rev. 0, 10CFR 61 Program

<u>Records for shipment nos.</u>

OC-0122-02, radioactive material, metal container, LSA-II
OC-0235-02, radioactive material, sea-land container, SCO-II
OC-0402-02, radioactive waste, Type A container, Yellow-II
OC-1003-02, radioactive waste, metal containers, LSA-II and SCO-II
OC-1004-02, radioactive waste, metal containers, SCO-II
OC-8002-02, radioactive waste, Type A containers, Yellow-III
Memo to file titled RADMAN DAW Database and dated November 20, 2002
Memo to file titled Designation of Staff Qualified for Transfer, Packaging, and Shipping of Low-Level Radioactive Material and dated November 26, 2002
Department of Transportation (DOT) 7A Type A Certificate for General Electric specification container no. 1056-S
NOA-OC-01-4Q, NOS field observation of removal, packaging, and shipping of old laundry trailers from the RCA yard, October 22 - 26, 2001
NOA-OC-01-4Q, NOS field observation of radwaste shipping/radiation protection, October 23-25, 2001
NOA-OC-01-4Q, NOS field observation of radwaste shipping activities for January 17, 2002, January 17, 2002
NOA-OC-01-4Q, NOS field observation of corrective actions for radiation protection and chemistry/radwaste/environmental, March 26, 2002
NOA-OC-02-1Q, NOS field observation of radwaste processing activities, March 27, 2002
NOA-OC-02-2Q, NOS field observation of radwaste processing activities, May 10 and 13, 2002
NOA-OC-02-4Q, NOS field observation for evaluation of plant isotopes (10 CFR 61 analysis) and beta energy analysis, November 13, 2002
Assessment and characterization of the Oyster Creek solid low level waste management program, First quarter 2001
Exelon Nuclear fleet-wide radwaste program improvement initiative/Electric Power Research Institute, Inc.-sponsored program assessment, February 2002
Assessment report (April 8 - 9, 2002) titled Material Condition/Oyster Creek New Radwaste Building
Memo titled Oyster Creek Radwaste Material Condition Improvement and dated July 2, 2002
Procedure 820.4, Rev. 4, Operation of SEEDS software
Procedure 829.2, Rev. 25, Stack effluent: sampling and analysis
Procedure 829.6, Rev. 29, AOG building: effluent and process sampling and analysis
Procedure 829.11, Rev. 21, Turbine building ventilation system: sampling monthly gaseous effluent data from October 2001 up to and including September 2002
Quarterly summary gaseous effluent data from the fourth quarter of 2001 up to and including the third quarter of 2002
RWP OC-1-02-00057, Rev. 00, Mechanical and electrical maintenance/NMD
RWP OC-1-02-00058, Rev. 00, Observation and inspection
RWP OC-1-02-00402, Rev. 00, 1R19 refueling activities on 119-foot elevation of the reactor building
RWP OC-1-02-00415, Rev. 01, 1R19 HCU maintenance on the 23-foot elevation in reactor building
RWP OC-1-02-00422, Rev. 00, Installation of noble metals monitoring system tie-in on

51-foot elevation of reactor building
RWP OC-1-02-00526, Rev. 00, 1R19 drywell observation and inspection
RWP OC-1-02-00614, Rev. 00, 1R19 minor maintenance in the condenser bay

<u>LHRA/ALARA briefing checklist for OC-1-02-00422</u>

Radiological surveys for clean-up system heat exchanger room on 51-foot elevation of reactor building dated November 9, 2000, August 21, 2002, September 17, 2002, and October 5 and 13, 2002
Memorandum titled “Required site stand down for RP incidents” dated September 20, 2002
Significance Determination for CAP O2002-1365
Event Response Team Charter for CAP O2002-1365
Apparent Cause Evaluation for CAP O2002-1365
Dose and dose rate vs. time profile for CAP O2002-1365
CAP O2002-1505, Worker exceeded his electronic dosimeter dose alarm
Apparent Cause Evaluation for CAP O2002-1505 (Draft 1C)
Dose rate vs. time profile for CAP O2002-1505
Location listing for annual noise level surveys at Oyster Creek
Siemens electronic personal dosimeter (EPD Mk2) technical handbook
Exelon Nuclear Industrial Safety Pocket Guide, 2002
Procedure RP-AA-376-1001, Rev. 0, Radiological posting, labeling, and marking standard
Procedure RP-AA-403, Rev. 1, Administration of the radiation work permit program
Procedure RP-MA-403-1001, Rev. 1, Radiation work permit processing
Procedure RP-AA-460, Rev. 2, Controls for high and very high radiation areas
Procedure RP-OC-401, Rev. 0, Conduct of radiological work
Procedure RP-AA-4002, Rev. 0, Radiation protection refuel outage readiness
Procedure LS-AA-2140, Rev. 3, Monthly Performance Indicator (PI) data elements for occupational exposure control effectiveness
Exelon Nuclear Fundamentals of Radiation Protection
Oyster Creek Radiation Protection Department management review meeting for RP Fundamentals roll-out on August 14, 2002
Nuclear Oversight continuous assessment report, Oyster Creek Nuclear Generating Station, NOS-OC-02-3Q, July - September 2002
1R19 exposure report by RWP dated October 18, 2002
Outage ALARA Plan tracking dated October 14, 2002
RP outage preparation checklist (RP-AA-4002, Attachment 1)as of October 3, 2002
Station ALARA Committee Meeting Minutes for August 5, 15, and 22 and for October 11, 2002

ALARA Plan No. 2002-016A, Rev. 0, 1R19 drywell miscellaneous valve maintenance (RWPs OC-1-02-00504)
ALARA Plan No. 2002-018A, Rev. 0, 1R19 drywell insulation removal and reinstallation uncoupling (RWPs OC-1-02-00506)
ALARA Plan No. 2002-020A, Rev. 0, Drywell scaffolding installation and removal (RWPs OC-1-02-00508)
ALARA Plan No. 2002-023A, Rev. 0, 1R19 drywell CRD exchange and uncoupling (RWPs OC-1-02-00511)

ALARA Plan No. 2002-031A, Rev. 0, 1R19 drywell ISI/IGSCC/FAC inspections uncoupling (RWPs OC-1-02-00519)
ALARA Plan No. 2002-057E, Rev. 2, Refueling floor activities including reactor disassembly, defuel/refuel, in-vessel inspections and repairs, and reactor reassembly (RWPs OC-1-02-00404, -00406, and -00407)
ALARA Plan No. 2002-077A, Rev. 0, 1R19 disassemble/open/inspect and repair as necessary drywell MSIV 1-7 (RWPs OC-1-02-00529)
Procedure RP-OC-1002, Rev. 0, Evaluation of plant radioisotopes and energies
Procedure 6630-ADM-4221.42, Rev. 12, Background efficiency and operational check determination and performance of the scaler counting system
FRA-ANP/Siemens US Office EPD Calibration Methodology, October 4, 2002
Siemens Certificate of Calibration for electronic personal dosimeter (EPD) Mark 2 (Serial No. 27487) and for electronic personal dosimeter (EPD-N) (Serial No. 7000347)
OCR 19-028, Penetrant Examination, Control Rod Drive Safe End to Safe End, NC-4-0001B
OCR 19-098, Ultrasonic Examination, Core Spray Elbow to Pipe, NZ-3-0072
OCR 19-060, Ultrasonic Examination, Recirculation Pipe Weld NG-D-0018
01-C-176, Ultrasonic Examination, Weld 23-1-16
01-C-174, Ultrasonic Examination, Weld 23-1-17
QA 20.03, Visual Examination of Drywell Head, Torus Exterior and Various Drywell Elevations
INR#OC-2002-01, Linear Indications on Steam Dryer
PDI-UT-2, Rev C, Ultrasonic Examination of Austenitic Pipe Welds
PT-OCK-100, Liquid Penetrant Examination
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