February 12, 2004

EA-04-033

Mr. Christopher M. Crane
President and CEO
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000219/2003005 AND PRELIMINARY WHITE FINDING

Dear Mr. Crane:

On December 31, 2003, the U.S. 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 22, 2003, with Mr. C. N. Swenson and other members of your staff.

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

Based on the results of this inspection, a finding was identified that appears to have low to moderate safety significance. As described in Section 4OA5 of this report, the finding involves the failure to identify and take prompt and adequate corrective action for a significant condition adverse to quality involving the 4160 volt feeder cable from the No. 1 emergency diesel generator (EDG #1) output breaker to the 1C 4160 volt electrical bus. Specifically, AmerGen did not take appropriate actions to evaluate, inspect, test or replace this cable following prior failures of similar cables at your facility, including the 1996 cable failure to emergency bus "1D" and a 2001 cable failure to the 1B2 480 volt substation. As a result, the cable failed on May 20, 2003, causing a loss of the 1C 4160 volt bus that was not recoverable, and forcing a plant shutdown due to the loss of the bus and multiple pieces of safety-related equipment.

This finding was assessed using the reactor safety Significance Determination Process (SDP) as a potentially safety significant finding that was preliminarily determined to be White (i.e., a finding of low to moderate significance). The finding appears to have low to moderate safety significance because the failure of the cable associated with the 1C 4160 volt emergency bus resulted in an unrecoverable loss of the bus, which increased the likelihood of a reactor transient and caused the mitigating equipment powered from the bus to be unavailable to perform its safety function. However, the safety significance of the finding was limited by the prompt shutdown and cooldown of the plant and subsequent replacement of the cable in a timely manner.

Following identification of the failed cable and the misclassification of that cable in engineering documentation, you implemented corrective actions to replace the cable in question and reverify the manufacturer of the other medium voltage cables onsite. Long-term corrective actions include a schedule to replace the remaining medium voltage cables that are potentially vulnerable to similar failures.

This preliminary White finding also involves an apparent violation of NRC requirements that is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's Website at http://www.nrc.gov/what-we-do/regulatory/enforcement.html.

We believe that we have sufficient information to make our final risk determination for the performance issue regarding this failed cable. However, before the NRC makes a final decision on this matter, we are providing you an opportunity to either submit a written response or to request a Regulatory Conference where you would be able to provide your perspectives on the significance of the finding, the bases for your position, and whether you agree with the apparent violation. If you choose to request a Regulatory Conference, we encourage you to submit your evaluation and any differences with the NRC evaluation at least one week prior to the conference in an effort to make the conference more efficient and effective. If a Regulatory Conference is held, it will be open for public observation. The NRC will also issue a press release to announce the Regulatory Conference. If you choose a written response, such a submittal should be sent to the NRC within 30 days of the date of this letter.

Please contact Mr. Peter Eselgroth at (610) 337-5234 within 10 business days of the date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision and you will be advised by separate correspondence of the results of our deliberations on this matter.

Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

This report also documents one self-revealing finding of very low safety significance (Green). The finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. Additionally, licensee-identified violations which were determined to be of very low safety significance are listed in this report. If you contest the apparent violation or the NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Inspector at the Oyster Creek Generating Station.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by the order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year 2002, and the remaining inspection activities for Oyster Creek were completed during calendar year 2003. The NRC will continue to monitor overall safeguards and security controls at Oyster Creek.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if you choose to provide one) will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction.

Sincerely,

/RA/

A. Randolph Blough, Director
Division of Reactor Projects

Docket No. 50-219
License No. DPR-16
ISFSI License No. 72-15

Enclosure: Inspection Report 05000219/2003005
w/Attachment: Supplemental Information

<u>cc w/enc</u>:
Chief Operating Officer, AmerGen
Site Vice President, Oyster Creek Nuclear Generating Station, AmerGen
Plant Manager, Oyster Creek Generating Station, AmerGen
Regulatory Assurance Manager Oyster Creek, AmerGen
Senior Vice President - Nuclear Services, AmerGen
Vice President - Mid-Atlantic Operations, AmerGen
Vice President - Operations Support, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Director Licensing, AmerGen
Manager Licensing - Oyster Creek, AmerGen
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R. Shadis, New England Coalition Staff
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Distribution w/encl: **(VIA E-MAIL)**
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-219
License No.:	DPR-16
ISFSI License No.:	72-15
Report No.:	05000219/2003005
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	September 29, 2003 - December 31, 2003
Inspectors:	Robert Summers, Senior Resident Inspector Jeff Herrera, Resident Inspector Steve Dennis, Resident Inspector Richard Barkley, Senior Project Engineer Jason Jang, Senior Health Physicist Joseph M. D'Antonio, Operations Engineer John McFadden, Health Physicist Nancy McNamara, EP Inspector Julian Williams, NRC Contractor/Operations Engineer Aniello Della Greca, Senior Reactor Inspector John Wray, Health Physicist
Approved By:	Peter W. Eselgroth, Chief Projects Branch 7 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000219/2003005; 09/29/03-12/31/03; Oyster Creek Generating Station; Other Areas.

This report covers a thirteen-week period of inspection by resident inspectors and announced inspections by regional senior health physics inspectors, operations engineers, an Emergency Preparedness (EP) inspector, and a senior reactor inspector. One preliminary White finding was identified as well as one Green non-cited violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3 dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems and Barrier Integrity

- Preliminary White. A finding was identified by the NRC involving the failure to comply with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, specifically the failure to take appropriate corrective actions following medium voltage cable failures in 1996 and 2001. As a result, on May 20, 2003, a ground fault on the cable running from the No. 1 EDG output breaker to the 4160 volt emergency bus "1C" caused an unrecoverable loss of the "1C" electrical bus. The loss of bus "1C," as well as the mitigating system and barrier integrity equipment powered from this bus, forced a Technical Specification required shutdown.

  This finding was more than minor because it was associated with the equipment performance attribute of the initiating events and mitigating systems cornerstones. Specifically, the failure of the cable associated with the 1C 4160 volt emergency bus resulted in the loss of the bus, which increased the likelihood of a reactor transient and caused the mitigating equipment powered from the bus to be unavailable to perform its safety function. A significance determination process (SDP) Phase 1 screening of this finding was performed and determined that the finding degraded both the initiating event and mitigating systems cornerstones. Therefore, an SDP Phase 2 evaluation was performed, which determined that the finding was of low to moderate safety significance (White). (Section 40A5)

- Green. A self-revealing finding was identified involving a non-cited violation of Technical Specification (TS) 6.8.1 because an operator failed to follow a surveillance procedure for test and calibration of the Core Spray System I Instrument Channel while at power. This resulted in the inadvertent tripping of the 480 Volt Alternating Current (VAC) Vital Bus 1A2 when the operator opened the wrong breaker during the test activity.

  This finding is greater than minor because it had an actual impact of de-energizing the 480 VAC Vital Bus, rendering inoperable mitigating systems and barrier integrity support system equipment, and therefore could be reasonably viewed as a precursor to a significant event. Both the mitigating systems and barrier integrity cornerstones were affected because the error led to core spray and containment spray subsystems being made inoperable. The finding is of very low safety significance because no plant transient occurred, operators were able to determine the cause of the failure and restored power to the vital bus in 1.25 hours, and an NRC risk analyst determined a final significance of less than 1×10^{-7} change in core damage frequency based on a Phase 3 significance evaluation. (Section 40A5)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Oyster Creek began the period at 98% Rated Thermal Power (RTP) due to ongoing leak repair maintenance on the second stage feedwater heater valves. Full RTP was achieved on October 3, 2003. Power was reduced again to 96% RTP on October 23, 2003, to support leak repair to a second stage heater drain tank level transmitter isolation valve and full RTP operation was achieved later that same day. Power was reduced to about 50% RTP on November 13, 2003, due to low intake water level and grass accumulation on the circulation water traveling screens caused by unusual high wind conditions. On November 16, 2003, full RTP was achieved and the plant operated at or near full RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (IP 71111.01 - 1 Sample)

a. Inspection Scope

On October 20, the inspectors walked down portions of the condensate system, the emergency service water (ESW) system, emergency diesel generators (EDG), the ultimate heat sink, fire water system, and switch yard. These systems were selected because their safety-related functions could be affected by adverse weather. The inspectors observed plant conditions, reviewed the up-to-date winter readiness work checklist and reviewed Corrective Action Process (CAP) report Nos. O2003-1578, O2003-2156, O2003-1790, O2003-2002, O2003-1920, and O2003-2048, evaluating those conditions using criteria documented in OP-AA-108-109, “Seasonal Readiness,” Rev. 1.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (IP 71111.04)

a. Inspection Scope

Partial System Walkdown. (71111.04Q - 2 Samples)

The inspectors performed two partial system walkdowns during this inspection period. On November 14, 2003, the inspectors walked down the control rod drive system (CRD), while the “B” CRD pump was out of service for maintenance. On November 19, 2003, the inspectors walked down Core Spray (CS) system 1 while the CS system 2 was out of service for maintenance and testing. To evaluate the operability of the selected train or system when the redundant train or system was inoperable or out of service, the inspectors checked for correct valve and power alignments by comparing positions of valves, switches, and electrical power breakers to the procedures listed below as well as applicable chapters of the Updated Final Safety Analysis Report (UFSAR):

- 302.1, “Control Rod Drive System,” Rev. 82
- 308, “Emergency Core Cooling System Operation,” Rev. 72

Complete System Walkdown. (71111.04S - 1 Sample)

On November 5, 2003, the inspectors conducted a detailed review of the alignment and condition of the EDG battery starting system. The inspectors used the licensee procedures and other documents listed below to verify proper system alignment:

- Drawing No. BR-3001, Emergency Power System One-Line Diagram, Rev. 4
- Procedure 341, “Emergency Diesel Generator Operation,” Rev. 69

The inspectors also verified electrical power requirements, labeling, hangers and support installation, and associated support system status. The walkdowns also included evaluation of system supporting structures against the following considerations:

- Battery cell terminal posts and connections showed no visible signs of degradation.
- Battery reservoir levels appeared normal.
- Component foundations were not degraded.
- Battery compartment ventilation equipment was functional.
- Battery cell temperatures were normal.

A review of outstanding maintenance work orders was performed to verify that the deficiencies did not significantly affect the EDG system function. In addition, the inspectors monitored the diesel fuel oil storage tank particulate removal activity, and reviewed the CAP database to verify that EDG fuel oil TS requirements and equipment alignment problems were being identified and appropriately resolved.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- OB-FZ-6A, “A” 480 V Switchgear Room
- OB-FZ-6B, “B” 480 V Switchgear Room
- OB-FZ-8C, A&B Battery Room, Tunnel & Electric Tray Room, 35'-0" Elevation

- LL-FA-32, Low level Radwaste Storage Facility Yard area - Storage of Radioactive Materials (Non-waste)
- MT-FA-12, Main Transformer and Condensate Area
- OB-FZ-22A, New Cable Spread Room (Mechanical Equipment Room), Elevation 63'-6"
- OB-FZ-5, Control Room 46'-6" Elevation

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Training (LORT) (IP 71111.11B - 1 Sample; IP 71111.11Q - 1 Sample)

1. IP 71111.11B - Biennial LORT Inspection

a. Inspection Scope

The licensee event report history was reviewed for events related to licensed operator performance and training. No events of significance were noted for individual followup.

The inspectors reviewed two examples of the 2003 comprehensive reactor operator (RO) and senior reactor operator (SRO) written examinations. The facility generated CAP No. O2003-2240 to address NRC comments resulting from this review.

The inspectors observed the administration of operating examinations to two crews and one staff crew. The operating examination consisted of two or three simulator scenarios and one set of five job performance measures per crew. The inspectors verified these examination materials satisfied the criteria of the examination standards and 10 CFR 55.59.

The inspectors observed simulator performance during the conduct of the examinations and reviewed performance testing and discrepancy reports to verify compliance with the requirements of 10CFR55.46. The inspectors noted that this simulator is used to meet experience requirements for operator license applicants. The inspectors verified that the facility evaluates and updates the simulator core model to replicate performance of the most recent plant core. Performance data from plant surveillance and testing activities are used for evaluation where such data is available, most recent core engineering data is used where plant data is not available (such as for transients testing). The following tests and data were reviewed:

Steady State and Core Performance Testing:

- TQ-AA-302-101, Rev 0, 100% Steady State Accuracy Test (includes estimated critical position, heat balance, and traversing incore probe traces), April 6, 2002 and October 16, 2003
- TQ-AB-303-103, Rev 0, Boiling Water Reactor Power Coefficient of Reactivity, July 25, 2003
- NOT 09, Shutdown Margin Measurement Test, March 24, 2002

Simulator Transient Tests:

- TTS-70 Simultaneous Trip of All Feedwater Pumps, August 15, 2003
- TTS-71 Simultaneous Closure of MSIVs, August 4, 2003
- TTS-78 Simultaneous Main Steam Isolation Valve Closure, Stuck Open Electro Magnetic Relief Valve (EMRV), Loss of Iso Condenser, Loss of Feedwater, July 6, 2003
- TTS-75 Maximum Power Ramp 100% to 75% to 100% RTP, August 5, 2003

Normal Evolution Tests:

- NOT 01, Approach to Critical, February 20, 2001
- NOT 20, Average Power Range Monitor Surveillance Test, January 7, 2002
- NOT 14, Recirc Pump Trip Circuitry Test, January 15, 2002
- NOT 18, CRD Pump Operability Test, January 10, 2002

Conformance with operator license conditions was verified by review of the following records:

- Attendance records for the most recent year training cycle
- 50% of all licensed operator medical records.
- Proficiency watchstanding and reactivation records. A 100% sample of RO and 50% sample of SRO watchstanding documentation was reviewed for the current and prior quarter to verify currency.

Remediation training records for the prior two years were reviewed. Fifteen instances of evaluation failures were reviewed and found to be adequately remediated.

Feedback forms for the prior two years were reviewed, and operators were interviewed. The personnel interviewed provided several examples of training responding to written or verbal feedback or requests for training.

A review was conducted of final licensee requalification exam results for the annual operating testing cycle. The inspection assessed whether pass rates were consistent with the guidance of NUREG-1021, Revision 9, “Operator Licensing Examination Standards for Power Reactors” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance SDP.”

The inspector verified that:

- Crew failure rate was on the dynamic simulator examination was less than 20% (Failure rate was 0%).

- Individual failure rate on the comprehensive biennial written exam was less than 20% (Failure rate was 0%).

- Individual failure rate on the walk-through (JPMs) was less than 20% (Failure rate was 0%)

- More than 75% of the individuals passed all portions of the exam (100% of the individuals passed all portions of the exam).

b. Findings

No findings of significance were identified.

2. IP 71111.11Q - Quarterly LORT Inspection

a. Inspection Scope

This inspection activity represented one inspection sample. This inspection assessed the LORT provided to the SROs and the ROs and the evaluation conducted on the simulator on November 4, 2003. The inspectors assessed the proficiency of the operating crew and verified that the evaluations of the crew identified and addressed operator performance issues. The inspection activities were performed using NUREG-1021, Rev. 8, “Operator Licensing Examination Standards for Power Reactors,” and Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program.”

The training included two scenarios and about four hours of classroom instruction. The inspectors assessed the simulator crew’s performance during the scenario. The inspector also assessed the evaluator’s assessment of the crew, to verify that operator performance issues were identified and appropriate remediation was conducted to address identified weaknesses.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (IP 71111.12Q - 1 Sample)

a. Inspection Scope

One sample was selected for review by the inspectors. 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 system, structure and/or component (SSCs) was properly classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed action requests (ARs), CAPs, (a)(1) corrective action plans and routine preventive maintenance activities. The inspectors also discussed the current system performance, associated issues and concerns, and planned activities to improve performance with the system engineer. In addition, the inspector compared unavailability data with control room log entries to verify accuracy of data and compliance with (a)(1) goals. AmerGen trending data was also reviewed. The following additional documents were reviewed: Maintenance Rule Technical Basis Document, TDR-1196, Rev. 1; NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The SSC reviewed during the inspection period was the following:

- 480 VAC Electrical Distribution System

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (IP 71111.13 - 4 Samples)

a. Inspection Scope

Four samples of emergent work were selected for review by the inspectors. The inspectors verified that the licensee evaluated the risk associated with the inoperability of the system along with other ongoing maintenance work. The documents associated with the troubleshooting plan, repair, and retest of the system were also reviewed. When appropriate, the inspectors verified compliance with TSs. Risk assessments were reviewed for the following activities:

- Biocide and filtration of Main Fuel Oil Storage Tank on September 16, 2003. The main fuel oil storage tank was found to have substantial water and sediment, exceeding industry recommended values. The fuel oil storage tank was treated with biocide for biologic growth and filtered to remove sediment and water as corrective actions described in CAP 02003-1865.

- Filtration of the Emergency Diesel Generator Fuel Oil Storage Tank on October 8, 2003. The EDG fuel oil storage tank "bottoms" were found to have high sediment and water. During the process of sediment removal of the main fuel oil storage tank, the EDG fuel oil storage tank was sampled and found to have high sediment content, although still within Oyster Creek operational guidelines. The fuel oil was filtered in order to remove the particulate and water and was subsequently treated with a stabilizer to arrest fuel oil degradation. The inspector reviewed CAP O2003-2076 and CAP O2003-2225 generated during the licensee's review. The inspector also reviewed Apparent Cause Evaluation, "Excessive Water and Sediment in EDG Fuel Oil Storage Tank," dated December 9, 2003; Operability Evaluation OC-2003-OE-0008, Diesel Generator Fuel Oil Storage and Transfer System, dated September 23, 2003; and Operability Evaluation OC-2003-OE-0011, dated December 31, 2003. There was no evidence of biological contamination of the fuel oil, therefore no biocide treatments were recommended. The inspectors verified that the licensee evaluated the fuel oil condition, as well as reviewed the actions taken to restore fuel oil quality.

- On September 28, 2003, the "C1" battery charger failed as indicated by a main control room alarm, leading to declaring the associated Direct Current (DC) bus inoperable. The spare battery charger for the bus was put into service, restoring the bus to operability. The inspector reviewed CAP No. O2003-1962 and Action Request (AR) A 2071497 generated during the licensee's review of the risk from the emergent work for the issue. The inspectors verified that the licensee evaluated the risk associated with the inoperability of the system along with other ongoing maintenance work. The inspectors reviewed Apparent Cause Evaluation, "C1 Battery Charger Failed to Maintain DC C Bus Voltage," dated October 30, 2003. The licensee's engineering evaluation of the failure mechanism determined that this event involved a maintenance preventable functional failure since the 125 VDC distribution center voltage fell below the TS required minimum. As a result the system was placed in Maintenance Rule (a)(1) status.

- On November 12, 2003, a modification to the "A" CRD breaker was performed. The inspectors verified that the licensee evaluated the risk associated with the inoperability of the system along with other ongoing maintenance work.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

For the non-routine events described below, the inspectors reviewed operator logs and plant computer data to determine what occurred and how the operators responded, and to determine if the response was in accordance with plant procedures:

- On November 13, 2003, the inspectors observed the site response to a loss of a 34.5 kV line due to an offsite fault during high wind conditions. The Q-121 breaker failed to open one of three phases upon demand, leaving one phase of the 34.5 kV line closed and providing power to a fault in the Whiting Switchyard. The Whiting feed was subsequently damaged and lost. TS minimum offsite power lines were met due to the remaining offsite power lines provided to the site. No significant plant transient occurred.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (IP 71111.15 - 4 Samples)

a. Inspection Scope

The inspectors reviewed four of the operability determinations the licensee had generated that warranted selection on the basis of risk insights. The inspectors assessed the accuracy of the evaluations, the use and control of compensatory measures if needed, and compliance with the TS. The inspector's review included a verification that the operability determinations were made as specified by Procedure LS-AA-105, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to the TS, UFSAR, and associated design-basis documents. The selected samples are listed below:

- No. 2 EDG - fast start exceeded acceptance criteria on December 15, 2003, (CAP No. O2003-2577) and AR A207012801, "Evaluate EDG2 Fast Start Time Outside Procedural Limit," dated December 15, 2003.

- Core Spray System 2 DP Alarm Switch out of calibration on November 18, 2003, (CAP No. O2003-2394) and AR A2075821, "Replace DP Alarm Switch for DPIS-RV30B," dated November 18, 2003.

- C1 Battery Charger failed a full load (500 amp) test on October 2, 2003, (CAP O2003-2017) and operability evaluation "125 VDC Distribution System," dated October 2, 2003.

- EDG Fuel Oil Storage Tank unacceptable sample results for water and sediment on October 27, 2003, (CAP O2003-2225) and operability evaluation "Diesel

Generator Fuel Oil Storage and Transfer System," Rev 2, dated December 31, 2003.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the operator work-around database and a sample of the associated corrective action items to identify conditions that could adversely affect the operability of mitigating systems or impact human reliability in responding to initiating events. The inspector reviewed the licensee's implementation of procedure OP-AA-102-103, "Operator Work-Around Program." The inspector also reviewed the status of the corrective actions described in CAP No. O2003-1622 which identified specific problem resolutions relating to the operator work-around associated with Safety Relief Valve and EMRV acoustic monitors becoming saturated during the August 14, 2003, electrical grid transient.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (IP 71111.17 - 1 Sample)

a. Inspection Scope

Annual. The inspectors reviewed one permanent plant modification, Engineering Change Request (ECR) 02-00531, "Design Package for ESW 1, ESW 2 and SW Piping Non Tie-in Work" that provided the directions for the installation of service water (SW) and ESW system 1 re-routed piping in the trenches between system tie-in-points. The inspectors verified that the existing design bases, licensing bases, and performance capability of the ESW system was not degraded by this modification.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Condensate return valve V-14-34 - surveillance procedure 609.4.001, "Isolation Condenser Valve Operability & IST," Rev. 49, performed on October 6, 2003, following trending measurements of the motor current from the Motor Control Center.

- 1-2 Fire Diesel - surveillance procedure 645.4.018, "Fire Pump Monitoring Test," Rev. 49, performed on October 15, 2003, following work performed during a scheduled system maintenance outage.

- Standby Liquid Control - surveillance procedure 612.4.001, "Standby Liquid Control Pump and Valve Operability and In-Service Test," Rev. 24, performed on September 29, 2003.

- "C" EDG battery - surveillance procedure 636.2.005, Diesel generator weekly battery surveillance, Rev. 20, performed on November 5, 2003.

- Emergency diesel generator #1 load test - surveillance procedure 636.4.003, Diesel generator #1 load test, Rev. 71, performed on November 4, 2003.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (IP 71111.22 - 4 samples)

a. Inspection Scope

The inspectors observed and reviewed four Surveillance Tests (ST) concentrating on verification of the adequacy of the test as required by Technical Specifications to demonstrate operability of the required system or component safety function. The inspector observed pre-test briefings and portions of the ST performance for procedure adherence, and verified that the resulting data associated with the ST met the requirements of the plant TSs and the UFSAR. The inspector also reviewed the results of past tests for the selected STs to verify that degraded or non-conforming conditions were identified and corrected, if needed. The following four STs were observed:

- 607.4.015, “Containment Spray and ESW System 2 Pump Operability, IST and Containment Spray Pumps Trip,” Rev. 13, conducted October 16, 2003
- 609.3.002, “Isolation Condenser Isolation Test and Calibration - A1/B1 Sensors First,” Rev 51, conducted on October 8, 2003
- 636.2.005, “Diesel Generator Weekly Battery Surveillance,” Rev. 20, conducted on November 5, 2003
- 602.3.004, “EMRV Pressure Sensor Test and Calibration,” Rev. 42, conducted on October 29, 2003

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

One sample was selected for review by the inspectors. The inspectors reviewed a Temporary Modification (TM) associated with the sediment filtration of the EDG fuel oil storage tank. The inspectors reviewed the associated implementing documents to verify the plant design basis and the system or component operability was maintained, which included CC-AA-112, “Temporary Configuration Changes,” Rev. 6. The TM was installed on the EDG fuel oil storage tank during filtration of the fuel oil storage tank for sediment removal. This modification allowed for the enhancement of the recirculation capability of the fuel oil in order to achieve a higher sediment removal by returning post filtered fuel oil to the top of the fuel oil tank to improve mixing.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes (IP 7111404 - 1 Sample)

a. Inspection Scope

A regional, in-office review was conducted of licensee-submitted revisions to the emergency plan, implementing procedures and EALs which were received by the NRC during the period of August - December 2003. A thorough review was conducted of plan aspects related to the Risk Significant Planning Standards, such as classifications, notifications and protective action recommendations. A cursory review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.47(b) and the requirements of Appendix E and they are subject to future inspections to ensure that the combination of these changes continue to meet NRC regulations. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (IP 71114.06 - 0 Samples)

“No Sample was available” this quarter due to completion of the LORT for all licensed personnel during the period.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (IP 71121.01 - 3 Samples)

a. Inspection Scope

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 Exelon/Oyster Creek’s access controls to radiologically significant areas. This inspection activity represents the completion of three (3) samples relative to this inspection area (i.e., inspection procedure sections 02.05.a and 02.07.a and b) and fulfills the annual inspection requirements.

High Risk Significant, High Dose Rate HRA and VHRA Controls

On December 8, 2003, the inspector met with the Radiation Protection Manager and discussed the controls and procedures for high-dose-rate high radiation areas (HRAs) and for very high radiation areas (VHRAs). The inspector reviewed the subject procedures (as listed in the List of Documents Reviewed section) to verify that the level of worker protection was adequate.

Radiation Protection Technician Proficiency

On December 9, 2003, the inspector observed radiation protection technicians performing pre-job briefs for two evolutions in HRAs. One involved opening a system to remove and clean a strainer on a valve in the old radioactive waste building; the other addressed inspecting, troubleshooting, and hydrolazing a hubdrain on the 75-foot elevation in the reactor building. Later, the inspector observed a radiation protection technician providing job coverage for the hydrolazing work activity. Throughout the inspection, the inspector observed the shift radiation protection technician at the main control point for the radiologically-controlled area (RCA) providing RCA entry briefings and, on one occasion, conducting a personnel decontamination. The observed radiation protection technicians were aware of the radiological conditions in the workplace and of the controls and limits on the applicable Radiation Work Permits (RWPs). The radiation protection technician performance was found to be consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspector reviewed four CAP reports (CAP Nos. O2003-1505, -1683, 1843, and -2153) which were generated since the last inspection. These reports were ones which involved radiation protection technician error. The corrective action approach taken by the licensee to resolve the reported problems appeared to be appropriate based on the inspector's assessment of the causal factors. The inspector performed a selective examination of 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), Technical Specifications, and appropriate plant procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (IP 71121.02 - 1 Sample)

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). This inspection activity represents the completion of one (1) sample relative to this inspection area (i.e., inspection procedure section 02.05.b) and fulfills the annual inspection requirements.

Source-Term Reduction and Control

The inspector reviewed several exposure reduction documents and plans generated by the licensee to reduce their elevated source term. Regarding plant source-term reduction and control, these documents discussed removal of hot spots, tracking of stellite-containing components for cobalt reduction initiatives, hydrogen injection management, noble metals treatment, effect of zinc injection, permanent shielding in the drywell, and guide tube vacuuming during outages. Based on this review, the licensee has developed an understanding of the plant source term, including knowledge of input mechanisms to reduce the source term, and has developed a source-term strategy.

The inspector performed a selective examination of documents (as listed in the List of Documents Reviewed section) for regulatory compliance and for adequacy of control of radiation exposure.

The review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701 (Use of process or other engineering controls), and appropriate plant procedures.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (IP 71121.03 - 1 Sample)

a. Inspection Scope

The inspector reviewed the program for health physics instrumentation to determine the accuracy and operability of the instrumentation. This inspection activity represents the completion of one (1) sample relative to this inspection area (i.e., inspection procedure section 02.02) and fulfills the annual inspection requirements.

Identify Additional Radiation Monitoring Instrumentation

During the inspection, the inspector made tours of the reactor, turbine, old radioactive waste, and new radioactive waste buildings and open areas inside and outside the RCA. For the observed HRA work, the radiation protection technicians were using medium range, hand-held radiation survey meters. The inspector noted the use of numerous continuous air monitors in the reactor and the operational radioactive waste buildings. The licensee reported that thus far in 2003, there had not been any internal exposures approaching 50 millirems of committed effective dose equivalent. The inspector reviewed the most recent whole body counter calibration and observed its use for performing whole body counts on personnel on December 11, 2003, in Building 14. Since the last inspection, the licensee installed gamma-sensitive portal monitors at the RCA exit points to augment the beta-sensitive full-body p-personnel contamination monitors which were already in place.

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

The review was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, Technical Specifications, and appropriate plant procedures.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety [PS]

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring (71122.01 - 9 Samples)

a. Inspection Scope

The inspector reviewed the following documents to evaluate the effectiveness of the licensee's radioactive gaseous and liquid effluent control programs. The requirements of the radioactive effluent controls are specified in the Technical Specifications/Offsite Dose Calculation Manual (TS/ODCM). This inspection activity represents the completion of nine (9) samples relative to this inspection area (i.e., inspection procedure sections 02.01a - d and 02.02d - k).

- the 2002 Radiological Annual Effluent Release Reports including projected public dose assessments;
- the most recent Offsite Dose Calculation Manual (ODCM) and technical justifications for ODCM changes;
- implementation of IE Bulleting 80-10, Contamination of Non-Radioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to environment;
- selected 2003 analytical results for radioactive liquid, charcoal cartridge, particulate filter, and noble gas samples;
- selected 2002-2003 monthly radioactive gaseous release permits;
- trending evaluations of the relationship between the Augmented Offgas System (AOG) availability and the amount of radioactive gaseous releases to the environment;
- implementation of the compensatory sampling and analysis program when the effluent radiation monitoring system (RMS) is out of service;
- trending evaluations of the availability for effluent RMS;
- calibration records for chemistry laboratory measurements equipment (gamma and liquid scintillation counters);
- implementation of the measurement laboratory quality control program, including control charts;
- implementation of the interlaboratory comparisons by the licensee and the contractor laboratory;
- the 2003 Quality Assurance (QA) Audit (NOS Audit # NOSA-OYS-03-08) and corrective actions;
- most recent Channel Calibration Results and Channel Functional Test Results for the radioactive liquid and gaseous effluent RMS, and most recent Channel Calibration Results for Flow Measurement Devices which are listed in the ODCM Tables 4.3.3.10-1 and 4.3.3.11-1

  <u>RMS</u>:

  - Reactor Building Service Water Monitor;
  - Turbine Building Sump No. 1-5 Monitor;
  - Turbine Building Ventilation Noble Gas Monitor (Low and High Ranges);
  - Main Stack Noble Gas Monitor (Low and High Ranges); and
  - AOG Building Exhaust Noble Gas Monitor.

  <u>Flow Rate Measuring Device</u>:

  - Main Stack Effluent Flow Measuring Device; and
  - Turbine Building Ventilation Effluent

- most recent surveillance testing results (visual inspection, delta P, in-place testings for High-Efficiency Particulate Air (HEPA) and charcoal filters, air capacity test, and laboratory test for iodine collection efficiency) for the following air treatment systems:

  - TS 4.5.H — Standby Gas Treatment System;
  - UFSAR 9.4.4.2.2 — New Radwaste Building Ventilation System (HEPA filter test only); and
  - UFSAR 9.4.4.2.3 — Offgas Building Ventilation System (HEPA filter test only).

The inspector also toured and observed the following activities to evaluate the effectiveness of the licensee's radioactive gaseous and liquid effluent control programs.

- walkdown for determining the availability of radioactive liquid/gaseous effluent RMS and for determining the equipment material condition; and
- walkdown for determining operability of air cleaning systems and for determining the equipment material condition.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

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

a. Inspection Scope

The inspectors sampled licensee submittals for Performance Indicators (PIs) listed below for the period October 2002 through September 2003. To verify the accuracy of the PI date reported during that period, PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Rev. 1, were used to verify the basis in reporting each data element.

- Safety System Unavailability - Residual Heat Removal
- Safety System Functional Failures

The inspector reviewed a selection of LERs, portions of the control room operator log entries, daily plant status meeting reports, the monthly operating reports, and PI data sheets to determine whether the licensee adequately identified the availability/unavailability times for the systems affecting the Safety System Availability - Residual Heat Removal and Safety System Functional Failure PIs for the previous four quarters. This number was compared to the number reported for each PI during the current quarter. In addition, the inspectors also interviewed licensee personnel associated with the PI data collection, evaluation, and distribution.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

1. Annual Sample Reviews (1 Sample)

a. Inspection Scope

The inspectors reviewed CAP Nos. O2003-0423 and O2003-0483 regarding failure to implement all requirements of the operator work-around program, and verified that corrective actions were adequate.

b. Findings and Observations

No findings of significance were identified. The inspector verified that the root cause evaluations and associated corrective actions were appropriate and timely. Therefore, no violation of regulatory requirement or findings were identified.

2. Corrective Action Program Record Reviews

a. Inspection Scope

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

During the inspections in the Reactor Safety and Radiation Safety strategic areas discussed in this report, the inspectors reviewed selected calendar year 2002-2003 CAP reports to evaluate the effectiveness of the licensee's problem identification and resolution processes in the cornerstone areas of mitigating systems, barrier integrity, occupational radiation safety, and public radiation safety. The following CAPs were reviewed:

- CAPs for mitigating systems and barrier integrity (O2003-1578, O2003-1622, O2003-1790, O2003-1865, O2003-1920, O2003-1962, O2003-2002, O2003-2048, O2003-2156, O2003-2225, O2003-2240, O2003-2394, and O2003-2577).

- CAPs for public radiation safety (O2002-0025, O2002-0146, O2002-0182, O2002-0185, O2002-0397, O2002-0557, O2002-0806, O2002-0673, O2002-0688, O2002-0895, O2002-0897, O2002-1008, O2002-1210, O2002-1233, O2002-1844, O2002-1851, O2002-1964, O2003-0426, O2003-0513, O2003-0548, O2003-0630, O2003-0777, O2003-0792, O2003-0817, O2003-1261, O2003-1533, O2003-1564, O2003-1320, O2003-1742, O2003-1779, O2003-

1980, O2003-2191, O2003-2226, O2002-0684, O2002-1982-6, O2003-0119, O2003-0119, O2003-0403, O2003-0548, O2003-0562, O2003-0629, O2003-1893, O2003-2174, O2003-2179, and O2003-2206).

- CAPs for occupational radiation safety (O2003-1505, O2003-1512, O2003-1683, O2003-1772, O2003-1843, O2003-2153, and O2003-2172)

b. Findings and Observations

No findings of significance were identified. For the documented reports reviewed, the inspectors observed that the full extent of the issues were identified, an appropriate evaluation was performed, and appropriate corrective actions were specified and prioritized.

4OA3 Event Follow-up (IP 71153)

1. (Closed) Licensee Event Report 05000219/2002003-01, Insufficient Appendix R Electrical Separation due to Sand Void, Revision 1

On September 19, 2003, the licensee submitted a revision to LER 05000219/2003003. The only significant change to the revised LER was a long-term corrective action to install a three-hour rated fire barrier between the affected 4160 VAC feeder conduits to vital buses 1A2 and 1B2 to meet Appendix R, Section III.G.2 requirements. This action was expected to be completed by April 30, 2004. The revised LER was reviewed by the inspectors and no findings of significance were identified. The original LER was previously reviewed by the NRC in Inspection Report 50-219/02-011, Section 4OA3, and was determined to be a violation of minor significance that was not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented this event in CAP O2002-1551. This LER is closed.

2. (Closed) Licensee Event Report 05000219/2003003-00, Actuation of Reactor Protection System due to Grid Transient

On August 14, 2003, an electrical disturbance in the off-site electrical power grid (Northeast Blackout) caused an overexcitation of the main generator that resulted in a main generator lockout, a turbine trip, and a reactor scram from 100% RTP. Operators placed the plant in a cold shutdown condition in accordance with plant procedures. While the Northeast grid disturbance caused voltage fluctuations on the local grid that caused the plant trip, off-site power was available throughout the event. An unrelated equipment malfunction during the event resulted in operators using the isolation condensers to cool down the reactor. This event is discussed in NRC Inspection Report 50-219/2003-004, Section 1R14. The licensee completed repairs to correct the equipment malfunctions and restarted the plant on August 16, 2003. The LER was reviewed by the inspectors and no findings of significance were identified. No violations were identified. The licensee documented this event in CAP O2003-1616. This LER is closed.

3. (Closed) Licensee Event Report 05000219/2002004-00, Actuation of Reactor Protection System due to Instrument Malfunction

On August 22, 2003, a turbine trip was caused by a spurious actuation of the moisture separator Hi-Hi level switch. This caused a reactor scram from 100% power. Operators placed the plant in a cold shutdown condition in accordance with plant procedures. All four moisture separator Hi-Hi level switches were replaced and the plant restarted. The LER was reviewed by the inspectors and no findings of significance were identified. No violations were identified. The licensee documented this event in CAP O2003-1674. This LER is closed.

4OA5 Other Activities

1. Follow Up of Unresolved Items

a. (Closed) Unresolved Item 05000219/2003004-01, Failure to Implement Surveillance Test Procedure Required by Technical Specifications

Introduction. A self-revealing Green finding and NCV was identified for failure to comply with TS 6.8.1.B, in that a written procedure implementing a surveillance test of the Core Spray System 1 was not followed as prescribed, resulting in a loss of a 480 VAC Vital Bus, 1A2. The complete analysis of this finding is described in NRC Inspection Report 50-219/2003004, Section 1R22, Surveillance Testing. The finding was left unresolved in that report because the significance was not yet determined.

Description. On September 10, 2003, at about 2:00 p.m., a non-licensed operator was performing portions of procedure 610.3.115, Core Spray System 1 Instrument Channel and Level Bistable Calibration and System Operability Test. Step 6.11.10 of the procedure requires placing the keylock switch at the breaker for Core Spray Pump 1A in the trip position. The operator erroneously placed the keylock switch in the trip position at the breaker feeding the 1A2 480 VAC Vital Bus, deenergizing the bus and its associated equipment. The equipment deenergized included two of the four loops of Core Spray and Containment Spray (each loop provides 100% system capability). Once the reason for the power loss to the bus was known, power was restored, per procedure, at about 3:15 p.m. A half reactor scram occurred, but no plant transient was caused by this event.

Analysis. The inspectors determined that the operator had failed to implement a TS required procedure as written. Specifically, the operator tripped the feeder breaker to Bus 1A2 as opposed to tripping the breaker for Core Spray Pump 1A. This failure led directly to the deenergization of Bus 1A2 and its associated equipment. Therefore, the finding was considered to be a licensee performance issue.

In accordance with Inspection Manual Chapter 0612, Appendix B, "Issue Disposition Screening," the inspectors also determined that the finding was more than minor because it was similar to Example 4.b in Inspection Manual Chapter 0612, Appendix E, and there were risk consequences to deenergizing this switchgear while the reactor was at power.

In accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Phase 1 screening. The finding represented a degradation in both the mitigating systems and barrier integrity cornerstones, because both core spray and containment spray systems were affected. As a result, the Phase 1 screening identified the necessity for a Phase 2 approximation since two cornerstones were degraded.

The inspectors conducted a Phase 2 evaluation of this finding using the Risk-Informed Inspection Notebook for Oyster Creek Nuclear Generating Station, Revision 1. An exposure time of less than three days was used. All worksheets were evaluated and the following functions were adjusted to reflect mitigating system components unavailable from lack of power: Control Rod Pumps; Low Pressure Injection; Containment Heat Removal; Late Inventory. Additionally, the likelihood of a transient was increased because the loss of Bus 1A2 results in a half scram.

The regional Senior Reactor Analyst reviewed the risk-informed notebook requirements for crediting operator recovery credit. It is known that it took the operators 1 hour and 15 minutes to recover the bus under non-emergency conditions. Therefore, if any credible accident sequence had occurred concurrent with the failure, the operators would have been able to recover power prior to core damage. Operators used procedures that they had trained on to conduct the recovery, conditions would not dramatically change in the switchgear rooms during an initiating event scenario, and equipment was readily available to perform the actions necessary to restore the bus. Therefore, the analyst determined that operator recovery credit was appropriate for the recovery of Bus 1A2.

The result from this estimation indicated that the finding was of very low safety significance (Green). According to the counting rule worksheet, the change in core damage frequency was estimated to be on the order of 10^{-7}. Therefore, the Senior Reactor Analyst conducted a Phase 3 evaluation.

The analyst determined that the performance deficiency represented a finding of very low risk significance. This was based on a Phase 3 evaluation using the NRC's Standardized Plant Analysis Risk (SPAR) model for Oyster Creek. The Revision 3i model was revised to include updated loss of offsite power recovery curves. The model provided a change in core damage frequency of 4.0×10^{-8}. The major factors ensuring that the risk was very low were the short period of time that the bus was deenergized and the availability of the bus and two sources of supply power. Because the results of the internal events evaluation indicated that the change in core damage frequency was less than 1×10^{-7}, no external events nor large early release frequency evaluation was required.

Therefore, the inspectors concluded that the performance deficiency was of very low risk significance (Green).

Enforcement. Because this failure to comply with TS 6.8.1.B is of very low safety significance and has been entered into the corrective action program (CAP O2003-1814), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000219/2003005-01, Failure to Implement a Surveillance Test Procedure required by Technical Specifications 6.8.1.B.

b. (Closed) URI 05000219/2003003-06; Loss of 4.16 kV Emergency Bus 1C Due to Ground Fault in Normally Energized Underground Cable.

Introduction. A Preliminary White finding and apparent violation was identified for failing to comply with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, related to a cable ground fault and subsequent loss of 4.16kV emergency bus 1C.

Description. The Oyster Creek emergency diesel generators (EDGs) provide onsite power to the emergency buses through underground cables located within a concrete vault connecting the EDG building and the Turbine Building (TB). The EDG output breaker is located in the EDG building, resulting in the cables that connect the EDG to the vital bus being normally energized, but not carrying any load. On May 20, 2003, with the plant operating at 100%, the emergency bus 1C normal supply breaker tripped and locked out, de-energizing the bus. Subsequent investigation by AmerGen determined that the breaker trip was due to a ground fault in one of the cables that connect the “1C” bus to the associated, No. 1 EDG output breaker. The loss of bus “1C” resulted in the loss of certain system components required for normal operation and accident mitigation. The loss of one of two emergency buses also led to a technical specification required shutdown of the plant. This was achieved via a controlled down-power and a manual scram at 9:43 a.m., approximately 9 hours after the event. Cold shutdown was achieved at 7:13 p.m. of the same day. This event is recorded in Licensee Event Report (LER) No. 05000219/03-002.

The independent failure analysis and the root cause analysis prepared by AmerGen determined that the May 20, 2003, cable failure was the result of water intrusion into the “1C” cable between the insulation and the insulation shield. Apparently, underground water filtered through the conduit and cable jacket, degrading the cable insulation and causing the cable to short to ground. The cable that failed in May 2003 was manufactured by Anaconda. Prior to this event, Oyster Creek had experienced an additional ten inservice, medium voltage cable failures. Of these, six were failures of Anaconda Unishield type cables. A seventh failed cable was also Unishield type manufactured by Cablec, after their purchase of the Anaconda company. Based on a chart prepared by the licensee, the Anaconda and Cablec cable failures were either due to water intrusion or manufacturing defects that, over time, caused the cables to fail.

Water intrusion and breakdown of the cable insulation was the cause of a 1996 failure of a cable associated with the No. 2 EDG output breaker to 4160 volt emergency bus “1D”. This cable, like the “1C” cable, was located within an underground cable vault structure between the EDG building and the TB. Water intrusion was also the failure mode that caused a cable supplying power to the 1B2 480V unit substation to short to ground and the supply breaker to trip on November 11, 2001. This cable was also manufactured by Anaconda and had only been in service for about five years, but in this case, the cable was “buried” in a sand bed that runs under the TB basement floor.

Oyster Creek’s cable testing program instituted in 1991 was ineffective in identifying insulation degradation and predicting failures of Anaconda Unishield type cables as evidenced by the failures that occurred in 1996 and 2001. Therefore, following the November 2001 cable failure, AmerGen had planned to replace the remaining Anaconda cables located within the sand-bed of the plant with Okonite Okoguard type cables. During the 2001 extent of condition review of the remaining cables, AmerGen used a chart that had been developed during an earlier walkdown of underground cables rather than using official design documents. An erroneous entry on this chart led the licensee to believe that the manufacturer of the cables that supply emergency power from the No. 1 EDG to emergency bus “1C” was Cablec. Therefore, the licensee did not evaluate the bus “1C” cables nor assess the need for appropriate corrective actions, such as additional testing or replacement, to ensure their continued functionality during the ensuing operating cycles.

Statement of Performance Deficiency

AmerGen failed to take appropriate corrective actions, following medium voltage cable failures in 1996 and 2001, to prevent the unisolable grounding of the cable supplying power to 4160 volt emergency bus “1C”. The failure of the bus led to a forced plant shutdown with a loss of a number of balance-of-plant controls that required a manual scram from a higher power level than a normal shutdown. In addition, a significant portion of the mitigating systems and barrier integrity support systems were among the components no longer functional because of the loss of power.

Including the May 2003 failure, Oyster Creek has had 11 medium voltage, in-service cable failures. Of the earlier 10 failures, six involved Anaconda cable and three of the six Anaconda failures were associated with cables that supply power from the two EDGs to the 4160 V emergency busses through the underground concrete vaults. In addition, two of the 10 failures (one in 1975 and the other in 1977) involved cables running in these same vaults from the two EDGs. However, these cables were manufactured by GE Vulkene and the failures were attributed to lightning strikes. The 4160 volt “1C” cable that failed in May 2003 had been installed in 1977 after one of these failures.

During their analysis of the May 2003 cable failure, AmerGen’s engineering organization assessed whether water entered into the cable vault structure. They concluded that, prior to 1991, rain water could enter the cable vault from an opening in the floor at the switchgear in the EDG building. (Note: Much of the roof of the EDG building is grating open to the environment to permit air flow into the building to support EDG operation and cooling.) Rainwater spillage into the TB basement was clearly evident from the vault associated with the 1D bus. In 1991, a modification installed a concrete dam around the power conduit at the floor line in the EDG building to prevent water entrance into the conduits from either diesel. However, AmerGen’s 2003 apparent cause analysis noted that by the time the rain water intrusion was corrected, there had been many years of water exposure to the cables. The licensee's apparent cause review stated that this "likely was the contributory factor for the EDG 1 cable failure."

In the same analysis, the licensee also noted that, in recent years, they had observed some water leakage into the TB basement from the No. 2 EDG power conduit duct bank and controls conduit for both EDGs (the analysis did not mention observed leakage from the No. 1 EDG power conduit). While noting that the leakage source was not from the EDG switchgear floor area, AmerGen concluded that the recent water intrusion into the affected conduits was associated with rain water pooling in the area of the TB west wall. The source was from perforated steel conduit stubs at the TB west wall as a result of long-term corrosion and original construction defects (i.e., voids in the surrounding vault/wall joint interfacing concrete).

In the May 2003 failure analysis of faulted sections of the "1C" cable, the licensee's vendor, Cable Technology Labs, Inc., stated that the underground cable (i.e., in the cable vault) had been operating in a wet environment, based on direct measurement of the insulation water content. As a result, moisture had permeated into the cable. The accumulated moisture created localized pockets between insulation shield and insulation, leading to voids and initiating partial discharge. Water also contributed to the corrosion and oxidation of the neutral (drain) wires leading to partial discharge occurrence between the wires and the insulation shield. This partial discharge in time resulted in insulation erosion, regions of stress enhancement, and possible dielectric failure.

Based on the inspection results stated above, the NRC concluded that AmerGen's extent of condition review for the November 2001 Anaconda cable failure did not adequately assess the "1C" bus cable that later failed in May 2003. This was a deficiency in the licensee's analysis which relied, in part, on an inaccurate cable list that included erroneous manufacturer identification for the "1C" bus cables. In addition, the NRC concluded that AmerGen's 2001 extent of condition review and assessment of corrective actions were incorrect for the following reasons:

- Prior to the 2003 cable failure, in-service failures of Anaconda cable had occurred in the cable vaults as well as in conduits buried in the sand-bed. Therefore, the 2001 evaluation was too narrow in scope in that it only considered Anaconda cables in the sand bed for additional corrective action/replacement.

- The 1996 failure of the cable from the No. 2 EDG to the 4160 volt emergency bus "1D" involved Anaconda cable and water intrusion in a cable vault. Engineering personnel have known since 1990 that water could intrude into the cable vaults from the EDG building. While that source was addressed in 1991, other water intrusion sources were known to exist in 2001 in at least two of the raceways between the EDGs and the TB.

- The cable that failed in May 2003 was installed in 1977 and had been exposed to a wet environment. The 1996 failure of the 4160 volt emergency bus "1D" cable that had been similarly exposed to a wet environment and for which a modification had been installed, should have prompted AmerGen to evaluate the impact of such conditions on the “1C” emergency bus cable.

As previously stated, the above facts support the conclusion that the “1C” emergency bus cable failure was attributable to ineffective corrective actions following the November 2001 Unit Substation 1B2 cable failure and the 1996 failure of the “1D” emergency bus cable. Since 1996, four separate root cause analyses have indicated that the Anaconda cables were subject to failure from a combination of manufacturing defects and environmental attributes, such as moisture or water intrusion. Such failures were not limited to cables in the sand-bed region. Therefore, Oyster Creek engineering should have known that the “1C” emergency bus cable was of Anaconda manufacture, should have assessed the non-sand-bed cable runs, including the “1C” underground vault, for water intrusion and other environmental conditions that made the Anaconda cable vulnerable to failure, and should have included these factors in the extent of condition review that was performed in November 2001.

Analysis. In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Disposition Screening,” the inspectors determined that this issue was more than minor because it was associated with the equipment performance attribute of the initiating events and mitigating systems cornerstones. Specifically, the failure of the cable associated with the 1C 4160 volt emergency bus resulted in the loss of the bus, which increased the likelihood of a reactor transient and caused the mitigating equipment powered from the bus to be unavailable to perform their safety functions.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a significance determination process (SDP) Phase 1 screening and determined that the finding degraded both the initiating event and mitigating systems cornerstones. Therefore, an SDP Phase 2 evaluation was required.

The inspectors conducted an SDP Phase 2 evaluation of the risk significance of the performance deficiency and determined that the finding was of low to moderate safety significance (White). The inspectors used the following assumptions in the Phase 2 evaluation:

- The time of the failure was known because the cable was normally energized and its failure directly resulted in a loss of bus “1C”. The licensee operated the unit at power with bus “1C” unavailable for approximately 9 hours and 13 minutes before initiating a manual scram to comply with the plant’s Technical Specifications. Therefore, an exposure period of less than three days was used.

- The cable failure increased the likelihood of a reactor transient due to a loss of the 4160 volt emergency bus “1C”, but it did not directly result in a reactor transient. Therefore, the initiating event likelihood of the loss of 4160 volt bus “1C” initiating event was increased by one order of magnitude.

- The unavailability of bus “1C” resulted in the following equipment being unavailable to perform their safety functions.

    - 1 of 2 core spray pumps
    - 1 of 3 shutdown cooling pumps
    - 1 of 2 standby liquid control pumps
    - 2 of 3 turbine building closed cooling water pumps
    - 1 of 2 reactor building closed cooling water pumps
    - 1 of 2 subsystems of containment spray

- Emergency electrical bus 1C and the mitigating equipment powered from the bus were not recoverable following the cable failure due to the location of the cable fault.

The dominant core damage sequences involved transients with loss of the power conversion system and small-break loss of coolant accidents when the low pressure injection function failed.

The risk significance of this finding due to fire events was dominated by fires that induced a loss of offsite power. However, the increase in core damage frequency due to fire events was substantially less than the risk due to internal events. The risk significance of this finding due to seismic events was dominated by seismic events that result in losses of offsite power. However, the frequency of seismically induced loss of offsite power events was orders of magnitude less than the frequency of loss of offsite power events used in the internal event analysis. Therefore, the risk contribution due to fire and seismic events was negligible.

Because Oyster Creek is a boiling water reactor with a Mark I containment and the increase in core damage frequency attributable to the finding was dominated by core damage sequences that contributed to large early release frequency (LERF), the inspectors evaluated the finding using the existing draft IMC 0609, Appendix H, “Containment Integrity SDP.” The inspectors applied the LERF factors for each core damage accident sequence which contributed to LERF and determined that the increase in LERF for this performance deficiency was greater than 1.0E-7 and less than 1.0E-6 per year. Consequently, the inspectors determined that the performance deficiency was of low to moderate safety significance (White) based on the increase to core damage frequency and large early release frequency.

This result was validated by the Senior Reactor Analyst who performed an independent calculation and verified that the SDP result (White) was appropriate for the performance deficiency. In addition, the licensee’s evaluation independently concluded that the significance of the cable failure was of low to moderate safety significance.

<u>Enforcement</u>. 10 CFR 50, Appendix B, Criterion XVI, requires in part, that for significant conditions adverse to quality, measures shall assure that the cause of the condition is determined and corrective action taken to prevent recurrence.

Contrary to the above, on May 20, 2003, with the plant operating at 100% power, a ground fault in an Anaconda cable running from EDG #1 output breaker to 4160 volt emergency bus “1C” caused a loss of power to this emergency electrical bus, a significant condition adverse to quality, whose recurrence was not prevented by corrective actions taken for previous, similar Anaconda cable faults. Specifically, corrective actions taken for Anaconda cable ground faults that occurred in October 1996 and November 2001, caused by water intrusion and breakdown of the cable insulation, did not prevent recurrence of the Anaconda cable ground fault caused by water intrusion and insulation breakdown on May 20, 2003. (AV 05000219/2003005-02, Failure to comply with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, related to a cable ground fault and subsequent loss of 4.16kV emergency bus “1C”).

2. Operation of an Independent Spent Fuel Storage Installation (ISFSI) (IP 60855)

a. Inspection Scope

The inspector observed preparations for installation of new Horizontal Storage Modules (HSMs) and discussed with cognizant licensee representatives and contract personnel controls to prevent unexpected personnel exposures due to radiation streaming through air vents in adjacent empty modules. Procedure E-20518 Rev 1, “HSM Installation,” dated October 31, 2003, was reviewed. The inspector verified that the procedure had been revised to include a requirement for placement of temporary shielding in the adjacent HSM vents prior to workers accessing the area. ALARA Plan #2003-18, “Install Eight HSMs at ISFSI” and Radiation Work Permit (RWP) 00093 Rev 00 were reviewed for adequacy of physical barriers, security oversight, and radiation protection controls. Station Work Order C2006310, Independent Spent Fuel Storage Installation (ISFSI) was reviewed and compared to the requirements of Transnuclear specification entitled “Field Erection of NUHOMS Precast HSM Array.”

The inspector observed receipt of the first Precast HSM on November 5, 2003. The temporary shields were examined by the inspector and preparations for pre-job surveys were observed. The inspector reviewed the pre and post shield installation surveys completed on November 11 and 12, 2003, and discussed with licensee representatives controls enacted when an 8-inch gap was created during installation. The inspector verified that dose rates did not exceed 100 mr/hr as anticipated and the work area was properly posted and controlled. Dose rates with temporary shielding in place were less than 1 mr/hr providing effective dose reduction. The inspector reviewed personnel dose records for the task and determined that no worker received unexpected radiation exposure.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The results of the Biennial LORT inspection were discussed with facility management at an exit meeting on October 23, 2003. Final requalification exam results were reported by the facility in a telephone call on November 10, 2003.

The results of the public radiation safety (effluent monitoring) inspection was discussed with facility management at an exit meeting on November 21, 2003.

The results of the occupational radiation safety inspection were discussed with facility management at an exit meeting on December 12, 2003.

On January 22, 2004, the resident inspectors presented the inspection results to Mr. C. N. Swenson and other members of licensee management. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following two violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- Section 2.3.3 of UFSAR requires that the licensee maintain the onsite Meteorological Monitoring Program as specified in Regulatory Guide 1.23 (RG 1.23). The required percentage for annual data recovery per RG 1.23 is greater than 90%. The licensee's data recovery percentage for 2002 was about 83% for the 33-ft level. The licensee has identified corrective actions (CAP Nos. O2003-2212 and O2003-2403) to enhance the system. This violation is of very low significance because the instrumentation does not affect actual releases of radioactive materials.

- A violation involving the failure to meet administrative requirements for radioactive effluent release or monitoring reporting described in Section 6.9 of the Technical Specifications was identified by AmerGen. The violation involved the untimely or incomplete submittal of information in the 2002 Annual Radioactive Effluent Release Report, the 2002 Annual Radiological Environmental Operating Report, and the 2002 Annual Radioactive Effluent Release Report. The licensee has identified corrective actions (CAP No. O2003-2206) to correct the administrative reporting problems. This violation was of very low significance because the administrative errors did not involve an actual exposure to members of the public.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

P. Bloss, BOP System Manager
M. Godknecht, Maintenance Rule Coordinator
E. Harkness, Vice President, Projects
S. Hutchins, Electrical Systems Manager
J. Magee, Director, Engineering
M. Massaro, Plant Manager
D. McMillan, Director, Training
L. Newton, Manager, Chemistry & Rad Protection
J. O'Rourke, Assistant Engineering Director
D. Slear, Manager, Regulatory Assurance
B. Stewart, Senior Licensing Engineer
C. Swenson, Vice President
C. Wilson, Director, Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000219/2003005-02	AV	A Preliminary White finding and apparent violation involving the failure to comply with 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, related to a cable ground fault and subsequent loss of 4.16kV emergency bus 1C. (Section 4OA5)

Opened and Closed

05000219/2003005-01	NCV	Failure to Implement a Surveillance Test Procedure required by Technical Specifications 6.8.1.B. (Section 4OA5)

Closed

05000219/2003003-00	LER	Actuation of Reactor Protection System due to Grid Transient (Section 4OA3)
05000219/2003004-00	LER	Actuation of Reactor Protection System due to Instrument Malfunction (Section 4OA3)

05000219/2002003-01	LER	Insufficient Appendix R Electrical Separation due to Sand Void, Revision 1 (Section 4OA3)
05000219/2003004-01	URI	Failure to Implement a Surveillance Test Procedure required by Technical Specifications 6.8.1.B. (Section 4OA5)
05000219/2003003-06	URI	Loss of 4.16 kV Emergency Bus 1C Due to Ground Fault in Normally Energized Underground Cable (Section 4OA5)

LIST OF DOCUMENTS REVIEWED
(not previously referenced)

TQ-AA-106, Rev. 2, Licensed Operator Requal Training Program
OP-AA-105-102, Rev. 2, NRC Active License Maintenance
OP-AA-105-101, Rev. 2 Administrative Process for NRC License and Medical Requirements.
OP-AA-102-103, Rev. 0, Operator Workaround Program
TDR No. 3000, Rev. 0, Cycle 18 Preliminary Fuel Cycle Design Report
Simulator Work Request #4187, Adjust the EXITECH AMON Core Model to Simulate the Cycle 19 Fuel Load.

Section 2OS1: Access Control to Radiologically Significant Areas:

RWP OC-1-03-00058, Rev. 00, Observation and inspection
RWP OC-1-03-00075, Rev. 00, Remove and clean strainer on a valve
RWP OC-1-03-00107, Rev. 00, Inspect, troubleshoot, and hydrolaze hubdrain on 75-foot elevation of reactor building
Procedure RP-AA-460, Rev. 2, Controls for high and very high radiation areas
Procedure RP-OC-460-1001, Rev. 0, Additional high radiation exposure controls
Procedure RP-OC-460-1002, Rev. 0, Radiation protection controlled keys
Procedure RP-AB-401-1301, Rev. 0, Writer's guide for preparation of an LPRM exchange ALARA plan
Procedure RP-AB-401-1302, Rev. 0, Writer's guide for preparation of a CRD replacement ALARA plan
Procedure RP-AB-401-1303, Rev. 0, Writer's guide for preparation of IRM/SRM replacement ALARA plan
Procedure RP-AB-460, Rev. 0, TIP area access controls

Section 2OS2: ALARA Planning and Controls:

Detailed dose estimate for week of December 8, 2003 (Week E-0) as of November 24, 2003 (Week E-1)(12-week planning process)
Schedule of work activity for week of December 8, 2003 as of December 5, 2003
Exposure summary reports for December 8, 10, 11, and 12, 2003

ALARA plan 2003-017 (RWP OC-1-03-00026, Rev. 01) Leak repair activities in steam-affected areas at power
Procedure RP-AA-400, Rev. 2, ALARA program
Procedure RP-AA-401, Rev. 2, Operational ALARA planning and controls
Agenda for Station ALARA Committee meeting on December 8, 2003
Exposure reduction plan for 2003 -2005
Radiation protection department excellence plan and action item tracking
Listing of ALARA reviews performed in 2003

Section 2OS3: Radiation Monitoring Instrumentation and Protective Equipment:

Whole body counter calibration records (November 3, 2003)(energy, efficiency, and quality assurance verification)
Whole body counter source check and out-of-range test reports for December 11, 2003
Certificate of Calibration for standard radionuclide source No. 66890-79 used for whole body counter calibration
Procedure RP-AA-220, Rev. 1, Bioassay program
Procedure RP-AA-222, Rev. 1, Methods for estimating internal exposure from in vivo and in vitro bioassay data
Procedure RP-AA-440, Rev. 4, Respiratory protection program
Procedure RP-OC-2001, Rev. 0, Dosimetry investigation reports
Procedure 6633-PMI-4222.06, Rev. 5, Calibration of the Canberra whole body counting system
Focused area self-assessment report on the respiratory protection program including respirator storage, issuance, maintenance, repair, testing, and QA, October 27, 2003
Operating Experience Report on Supplied Air Hood Air Line Connection, May 2003

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
AOG	Augmented Offgas
AR	Action Request
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CS	Core Spray
DC	Direct Current
EAL	Emergency Action Levels
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
EMRV	Electro Magnetic Relief Valve
EP	Emergency Preparedness
ESW	Emergency Service Water
HEPA	High-Efficiency Particulate Air (filter)

HRA	High Radiation Area
HSM	Horizontal Storage Module
ISFSI	Independent Spent Fuel Storage Installation
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LORT	Licensed Operator Requalification Training
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PI	Performance Indicator
PMT	Post Maintenance Testing
QA	Quality Assurance
RCA	Radiologically Controlled Area
RMS	Radiation Monitoring System
RO	Reactor Operator
RTP	Rated Thermal Power
RWP	Radiation Work Permit
SDP	Significance Determination Process
SRO	Senior Reactor Operator
SSC	Structure, System, and Component
ST	Surveillance Test
SW	Service Water
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VAC	Volts Alternating Current
VHRA	Very High Radiation Area