August 12, 2004

EA-04-142

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

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INSPECTION REPORT 05000219/2004003; PRELIMINARY GREATER THAN GREEN FINDING

Dear Mr. Crane:

On June 30, 2004, 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 July 15, 2004 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 a safety significance that is preliminarily Greater Than Green. Section 4OA4 of the attached report describes that finding which involves the failure to follow procedures when replacing cooling fan drive belts of the No. 1 Emergency Diesel Generator (EDG) during a two-year overhaul in April 2004. This preliminary safety significance was based on a conservative internal and external initiating events risk analysis of the increase in core damage frequency (CDF) and large early release frequency (LERF). The Preliminary Greater Than Green characterization was due to the variability in the outcome of the analysis based on the potential ability of the #1 EDG to perform its safety function for a portion of its mission time.

Before the NRC makes a final decision on this matter, we are providing you an opportunity to (1) present to the NRC your perspectives on the facts and assumptions, used by the NRC to arrive at the finding and its significance, at a Regulatory Conference or (2) submit your position on the finding to the NRC in writing. This will also provide you an opportunity to submit any additional information concerning the significance of this finding, including the ability of the #1 EDG to perform its safety function, given the high vibrations and the loose cooling fan shaft bearing bolts. Further, we understand that you may plan to use the results of your July 28, 2004 testing of a diesel generator at an Exelon non-nuclear facility, as a consideration in your safety significance assessment of this finding at Oyster Creek. If you choose to do so, we would expect you would include a clearly articulated and effectively supported applicability analysis of that testing, since the diesel generator tested at the non-nuclear facility was not the same as the #1 EDG at Oyster Creek.

If you request a Regulatory Conference, it should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation 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 conference. If you decide to submit only a written response, such a submittal should be sent to the NRC within 30 days of the receipt 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.

This preliminary Greater Than Green finding also involves an apparent violation of NRC requirements for failing to follow procedures when replacing the cooling fan drive bolts for the specific EDG. This violation 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. No Notice of Violation is being issued for this inspection finding at this time, because the NRC has not made a final determination in this matter. 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.

The enclosed report also documents two NRC-identified findings and one self-revealing finding of very low safety significance (Green). These findings 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 (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 NCVs 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 U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Inspector at the Oyster Creek Generating Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Brian E. Holian Acting For/

A. Randolph Blough, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000219/2004003
w/Attachment: Supplemental Information

cc w/encl:
Chief Operating Officer, AmerGen
Site Vice President, Oyster Creek Nuclear Generating Station, AmerGen
Plant Manager, Oyster Creek Generating Station, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Manager Licensing - Oyster Creek, AmerGen
J. Fewell, Assistant General Counsel, Exelon Nuclear
Correspondence Control Desk, AmerGen
J. Matthews, Esquire, Morgan, Lewis & Bockius LLP
Mayor of Lacey Township
K. Tosch - Chief, Bureau of Nuclear Engineering, NJ Dept. of Env. Protection
R. Shadis, New England Coalition Staff
N. Cohen, Coordinator - Unplug Salem Campaign
W. Costanzo, Technical Advisor - Jersey Shore Nuclear Watch
E. Zobian, Coordinator - Jersey Shore Anti Nuclear Alliance

Distribution w/encl:
Region I Docket Room (with concurrences)
S. Collins, RA
J. Wiggins, DRA
P. Eselgroth, DRP
R. Barkley, DRP
R. Summers, SRI
C. Miller, OEDO
R. Laufer, NRR
P. Tam, PM, NRR
T. Colburn (Back up PM), NRR
D. Holody, EO, RI
D. Vito, ORA, RI

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML042260078.wpd:
After declaring this document "An Official Agency Record" it **will** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP		RI/DRS		RI/ORA		RI/DRP	
NAME	RSummers/*RJS*		PEselgroth/*PWE*		WSchmidt/*WLS*		DHolody/*DJH*		ABlough/*BeH for*	
DATE	08/11/2004		08/11/2004		08/11/2004		08/11/2004		08/12/2004	

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-219
License No.:	DPR-16
Report No.:	05000219/2004003
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	April 1, 2004 - June 30, 2004
Inspectors:	Robert Summers, Senior Resident Inspector Jeff Herrera, Resident Inspector Aniello L. Della Greca, Senior Reactor Inspector Suresh Chaudhary, Reactor Inspector Brice Bickett, Reactor Inspector Joseph Schoppy, Senior Reactor Inspector Thomas Hipschman, Reactor Inspector Steven Dennis, Reactor Inspector Ronald Nimitz, Senior Health Physicist Richard Barkley, P.E., Senior Project Engineer
Approved By:	Peter W. Eselgroth, Chief Projects Branch 7 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000219/2004003; 04/01/04 - 06/30/04; Oyster Creek Generating Station; Personnel Performance During Non-Routine Plant Evolutions, Operator Work-Arounds, Surveillance Testing, Event Follow-up.

This report covers a 13-week period of inspection by resident inspectors and announced inspections by a regional senior health physics inspector, a senior operations engineer, a senior reactor inspector, a reactor inspector and an emergency preparedness inspector. One preliminary Greater Than Green finding and apparent violation, and three green findings involving non-cited violations (NCV), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of the Oyster Creek Quality Assurance Program for failure to adequately assess and correct plant equipment that was the subject of operating experience information and prevent a transient event that resulted in the trip of the 'D' recirculation pump during four loop full power operations on April 2, 2004.

  This finding is greater than minor because it had an actual impact of tripping one of four operating reactor recirculation pumps, and therefore could be reasonably viewed as a precursor to a significant event. It increased the likelihood of a plant scram due to resultant power operation in the "buffer zone" of the core power to flow map where there is significantly reduced margin to the flow-biased high power scram setpoint. It also resulted in operation of the remaining reactor recirculation pumps at speeds that could result in pump damage without operator action to reduce the speed and resultant core flow. This condition affects the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding is of very low safety significance because it does not contribute to: a primary or secondary system LOCA initiator, or both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, or the likelihood of a fire or internal/external flood. This finding has a cross-cutting aspect of Problem Identification and Resolution (PI&R) in that engineering evaluation of External Operating Experience and corrective action implementation was inadequate to prevent a similar condition at the site. (Section 1R14)

Cornerstone: Mitigating Systems

- TBD. A self-revealing apparent violation, having potential safety significance greater than very low significance, was identified for failure to implement appropriate procedural requirements for maintenance on the #1 Emergency Diesel Generator (EDG) during an overhaul conducted April 26 - 30, 2004. Technicians failed to follow written procedures to torque the cooling fan shaft bearing bolts following fan belt replacement as prescribed by Technical Specification 6.8.1. The poor maintenance practices led to the high vibrations during surveillance testing and manual shutdown of the #1 EDG on May 17, 2004. The high vibrations increased the potential for a failure of the #1 EDG due to loss of the skid-mounted cooling system. The final determination of risk significance is pending potential additional information from AmerGen concerning the ability of the #1 EDG to perform its safety function, given the high vibrations and the loose cooling fan shaft bearing bolts.

  The finding was more than minor because it affected the mitigation system cornerstone objective to ensure the availability, reliability, and capability of systems (emergency AC power) that respond to initiating events to prevent undesirable consequences and the related attributes of equipment performance, human performance and procedure quality. The finding was determined to have a potential safety significance greater than very low significance using a conservative assumption, based on initial information, that the #1 EDG would have been unable to perform its safety function for 17 days (April 30 - May 17). This assumption was made due to the large degree of uncertainty associated with the ability of the #1 EDG to operate with the high vibrations and loose cooling fan shaft bearing bolts The Phase 1 screening identified that a Phase 2 analysis was needed because the #1 EDG would have been inoperable in excess of its Technical Specification Allowed Outage Time of 7 days. Preliminary Phase 2 and Phase 3 evaluations resulted in a preliminary Greater Than Green finding, considering the increase in both core damage frequency and large early release frequency for internally and externally initiated losses of offsite power. Also, this finding has a cross-cutting aspect of human performance in that technicians failed to follow written procedures. (Section 4OA4)

- Green. A self-revealing event involving an inadvertent loss of shutdown cooling resulted in a Green finding and non-cited violation (NCV) for failure to establish and maintain appropriate procedural requirements for the operation of the shutdown cooling system, as prescribed by Technical Specification 6.8.1 and the Oyster Creek Operation Quality Assurance Plan (Quality Assurance Topical Report) NO-AA-10, Rev. 72.

  This finding is greater than minor because the procedural control deficiency actually led to a loss of the normal shutdown decay heat removal capability and affected the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspector determined, in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," that the finding was of very low safety significance because: (1) the reactor coolant temperature rise was very low and not considered a loss of control event; and, (2) the deficiency did not: increase the likelihood that a loss of decay heat removal would occur due to failure of the system itself or support systems, or include decay heat removal instrumentation or vessel level instrumentation such that degraded core cooling could not be detected, or increase the likelihood of a loss of Reactor Coolant System (RCS) inventory or RCS level instrumentation, or involve a design or qualification deficiency; or result in an actual loss of safety function for risk-significant equipment with respect to internal or external events. (Section 4OA3)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Green finding and non-cited violation (NCV) for the licensee's failure to identify a condition adverse to quality in accordance with 10 CFR Part 50 Appendix B Criterion XVI and Oyster Creek Station Procedure LS-OC-125, Corrective Action Program Procedure, Rev. 4, when a secondary containment airlock door was found open, resulting in a momentary violation of Technical Specification 3.5.B and Procedure 312.10, "Secondary Containment Control," Rev. 8.

  This finding is greater than minor because the failure to timely identify the condition adverse to quality for the airlock door, if left uncorrected, could have led to a more significant event involving a failure of the airlock interlock. This condition affects the Reactor Safety Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases from accidents or events. Since the finding only adversely affects the radiological barrier function of the Standby Gas Treatment System, it was determined to be of very low safety significance. This finding also has a cross-cutting aspect of PI&R in that operators failed to properly initiate a Corrective Action Process (CAP) report when the degraded and open airlock door was discovered. (Section 1R16)

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. One licensee-identified violation and associated corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Oyster Creek began the integrated inspection period at 100% of Rated Thermal Power (RTP). On April 1, 2004, power was reduced to 50% RTP to repair the “D” reactor recirculation pump controls. On April 2, 2004, power was further reduced to 35% RTP to conduct a full closure test of the MSIVs due to a partial closure test failure. On April 4, 2004, Oyster Creek returned to 100% RTP. On May 26, 2004, a planned shut down was commenced to begin a mid-cycle maintenance outage to replace the “A” reactor recirculation pump motor. On May 27, 2004, a reactor scram occurred from about 2% RTP due to spiking events on several IRM detectors. Following completion of planned repairs to the reactor recirculation system and other off-line activities, reactor start-up commenced on June 1. Oyster Creek returned to 100% RTP on June 3, 2004, and remained at 100% for the remainder of the inspection period, except for minor power reductions for testing.

1. REACTOR SAFETY

Cornerstones: Initiating Events/Mitigating Systems/Barrier Integrity

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

a. Inspection Scope

The inspectors reviewed Oyster Creek’s seasonal readiness preparations to verify that safety-related equipment would remain functional when challenged by summer weather conditions. The inspectors reviewed the licensee’s seasonal readiness procedure (OP-AA-108-109, Seasonal Readiness, Revision 1), seasonal check lists, and performed walk downs to verify that the safety-related equipment would remain functional during adverse weather conditions. The inspectors evaluated the condition of the Emergency Service Water System, Service Water System, Recirculation Motor-Generator Set Coolers, Circulating Water System, Electrical Switchyard, and Emergency Diesel Generators prior to the onset of summer weather conditions.

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

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (IP 71111.02 - 21 Samples)

a. Inspection Scope

The inspectors reviewed six safety evaluations (SE) that were completed during the past three years. The SEs reviewed were distributed among initiating event, mitigating system, and barrier integrity cornerstones. These SEs were reviewed to verify that changes to the facility or procedures as described in the Updated Final Safety Analysis Reports (UFSAR) and changes to tests not described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59, and that the safety issues pertinent to the changes were properly resolved or adequately addressed. The reviews also included the verification that the licensee had appropriately concluded that the changes and tests could be accomplished without obtaining license amendments.

The following six safety evaluations were reviewed:

OC-2001-E-0007	UFSAR 3.5.1.3
OC-2001-E-0011	Addition of Maintenance Isolation Valves in a control rod drive (CRD) Hydraulic Drive and Cooling Water Line, Rev. 1
OC-2002-E-0001	1MNCR O2001-1839 Disposition/Operability Assessment, Rev. 0
OC-2002-E-0003	Oyster Creek Increased Core Flow Implementation, Rev. 0
OC-2002-E-0004	Evaluation of 6 Hour Service Water Outage During 1R19, Rev. 1
OC-2002-E-0005	Procedure to Flush Core Spray and Flood Reactor Cavity

The inspectors also reviewed 15 screen-out evaluations for changes, tests and experiments for which AmerGen determined that safety evaluations were not required. This review was performed to verify that the licensee's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The listing of screened-out evaluations reviewed is provided in the Attachment.

In addition, the inspectors reviewed the administrative procedures that were used to control the screening, preparation, and issuance of the safety evaluations to ensure that the procedure adequately covered the requirements of 10 CFR 50.59. In conjunction with this review, the inspectors also reviewed selected applicability review forms related to plant changes (primarily procedure changes) for which the requirements of 10 CFR 50.59 did not apply.

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 April 30, 2004, the inspectors walked down the #2 EDG during the system outage on the #1 EDG. On June 7, 2004, the inspectors walked down the #2 Core Spray (CS) system during in-service testing of the #1 CS system. The inspectors verified that the associated maintenance activities did not adversely affect redundant components. 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 system operating procedures, as well as applicable chapters of the UFSAR.

Complete System Walkdown. (71111.04S - 2 Samples)

This inspection activity represented two samples.

On June 10, 2004 the inspectors completed a detailed review of the alignment and condition of the Isolation Condenser (IC) system. The inspector used AmerGen's procedures and other documents listed below to verify proper system alignment:

- Drawing No. GE 148F262, "Emergency Condenser Flow Diagram," Rev. 51
- Procedure 307, "Isolation Condenser system," Rev. 84
- Drawing No. 1083-131-19. Rev. B, "Isolation Condenser Replacement Drywell Penetration Piping and Isolation Valves"
- Drawing No. 1083-131-19. Rev. A, "Isolation Condenser Replacement 10 Inch Penetration Pipe & Flued Collar Weldment"
- General Arrangement Drawing No. 3E-153-02-005, "Reactor Building Plan Floor Elevation 95' -3"," Rev. 7
- Drawing No. EI705, "Appendix R Safe Shutdown Circuit Routing Reactor Building El. 23' -6"," Rev. 0
- Drawing No. EI707, "Appendix R Safe Shutdown Circuit Routing Reactor Building El. 75' -3"," Rev. 0

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

- Piping and structural supports showed no visible signs of degradation or leakage;
- Cabling and conduits showed no visible signs of degradation;
- Component foundations were not degraded;
- Valves showed no visible signs of degradation; and

- All deficiencies properly identified and dispositioned.

On June 16, 2004, the inspector conducted a detailed review of the alignment and condition of the #1 Emergency Service Water (ESW) system. The inspectors used the licensee procedures and other documents listed below to verify proper system alignment:

- Drawing No. BR-2005, "Emergency Service Water Flow Diagram"
- Procedure 322, "Service Water System," Rev. 58
- Procedure 310, "Containment Spray System Operation," Rev. 82

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:

- Piping and structural supports showed no visible signs of degradation or leakage;
- Observed pump shaft packing lubricating water performed desired function;
- Observed interface with containment spray heat exchangers and that flow was adequate and expected for current configuration;
- Component foundations were not degraded; and
- Flow and pressure in system was indicating properly.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (IP 71111.05Q - 9 Samples)

a. Inspection Scope

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

- TB-FZ-11F, Feedwater pumps, Elevs. 0'-6" and 3'-6"
- RB-FZ-1D, 51' Elevation
- RB-FZ-1C, 75' Elevation
- RB-FZ-1B, 95' Elevation
- FW-FA-18, Fire water pump house
- RB-FZ-1F4, NE Corner Room -19' Elevation
- OB-FZ-6A&6B, 480V switchgear rooms

- OB-FZ-4, Cable Spread Room, 36’ Elevation
- OB-FZ-22A, New Cable Spread Room (Mechanical equipment Room) Elev. 74’-6"

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (IP 71111.06 - 2 Samples)

1. External

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

2. Internal

a. Inspection Scope

The inspector verified that operator actions to mitigate flooding described in section 10.7. of the Oyster Creek Internal Flooding Analysis, dated November 1991, were appropriately addressed in abnormal and emergency procedures. A walkdown of the northeast corner room, which contains the #1 Containment Spray/Emergency Service Water (CS/ESW) system pumps and heat exchangers, was also performed.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

This inspection activity represented one inspection sample. This inspection assessed the LORT provided to the SROs and the ROs and the evaluation conducted on the simulator on June 3, 2004. The inspectors assessed the proficiency of the operating crew and verified that the evaluations of the crew identified and addressed operator performance issues. The inspection activities were performed using NUREG-1021, Rev. 8, “Operator Licensing Examination Standards for Power Reactors,” and Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program.”

The training included three scenarios and about four hours of testing/evaluation. The inspectors assessed the simulator crew’s performance during each scenario. The inspectors also assessed the evaluator’s assessment of the crew, to verify that operator performance issues were identified and appropriate remediation was conducted to address identified weaknesses. The following “out-of-box” simulation tests were observed:

- SC-1D, RCS Flow Instrument Failure; Unisolable LOCA Outside Containment
- PC-5C, Electro-Magnetic Relief Valve Instrument Failure; RCS Leak; ATWS
- RPV-6D, Loss of CRD; ATWS; Loss of PCS

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (IP 71111.12Q - 3 Samples)

a. Inspection Scope

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

- Isolation Condenser system
- 125 VDC system
- Emergency Service Water system

The inspectors also reviewed the following documents:

- ER-AA-310-1003, “Maintenance Rule - Performance Criteria Selection,” Rev. 2
- ER-AA-310-1004, “Maintenance Rule - Performance Monitoring,” Rev. 1
- ER-AA-310, “Implementation of the Maintenance Rule,” Rev. 2
- OC-7 Functional Failure Definition for System 211 (Isolation Condenser system)
- Emergency Isolation Condenser System Health Report, 1st quarter 2004
- Topical Report 140, Rev. 0, “Emergency Service Water and Service Water System Piping Plan”
- 125 VDC Maintenance Rule Performance Assessment - dated December 12, 2003

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- Heavy load lift risk during the planned maintenance on the #1 EDG during the week of April 26, 2004

- #2 EDG protection during the planned maintenance on the #1 EDG during the week of April 26, 2004

- The concurrent planned outage of the #2 EDG, the #1-1 turbine building closed cooling water heat exchanger, and #1 combustion turbine on May 11, 2004

- Main Generator in manual voltage control due to the Amplidyne being removed from service because of improper generator voltage regulation on June 10, 2004.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (IP 71111.14 - 2 samples)

a. Inspection Scope

For the two sampled 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 April 1, 2004, the inspectors observed the operator response to the “D” Recirculation pump trip event. Operator actions consisted of reducing power to approximately 55% and allowing the plant to stabilize under those conditions. An investigation to the “D” Recirculation pump trip revealed that the 1-watt 4R resistor on the Motor-Generator voltage regulator card failed due to temperature effects from the power dissipation exceeding 1 watt. While at reduced power, AmerGen replaced the failed resistor in the “D” Recirculation Motor-Generator (MG) Set voltage regulator card in addition to replacing the other 4R resistors in the remaining recirculation loops. The “A” 4R resistor was not replaced because this pump was not in service and investigation of the “A” 4R resistor did not show signs of degradation. (Subsequently, the 4R resistor in the “A” recirculation loop MG set was replaced during the maintenance outage that began in May 27, 2004, with the upgraded 2-watt design in order to prevent future failures of the 4R resistor.)

- On May 27, 2004, the reactor scrammed during a plant shutdown. The Reactor Protection System (RPS) responded to spiking on intermediate range monitor channels 13 and 14 on RPS system 1 and channel 18 on RPS system 2, resulting in a plant scram. Due to the plant being at approximately two percent power, there was minimal impact on the plant from this scram. AmerGen formed a team to determine the specific problem that was causing excessive noise on the system, which resulted in the scram. This issue was documented in CAP report O2004-1314. An Unresolved Item (URI) was identified for the multiple channel spiking events affecting the intermediate range monitoring system during the plant shutdown on May 27, 2004. This item will remain unresolved pending the review of the licensee’s root cause analysis to determine if prior corrective actions for the IRM spiking problems were ineffective. **(URI 0500219/200400302**)

b. Findings

Introduction. A Green NCV was identified for failure to implement the Oyster Creek Operational Quality Assurance Plan (QA Topical Report, NO-AA-10, Rev. 72) to support the activity requirement to assess operating experience information that involved the recirculation pump Motor-Generator (MG) set voltage regulator card. This condition led to a subsequent failure of the "D" MG set voltage regulator card and a trip of the associated recirculation pump during four loop, full power operation on April 1, 2004.

Description. At 11:45 p.m. on April 1, 2004, the main control room received both a drive motor lockout and drive motor breaker trip alarm and the subsequent trip of the 'D' Recirculation pump. Local investigation revealed no apparent cause for the breaker trip in either the MG set room or the 4160V room. Power was reduced using recirculation flow and control rods in accordance with abnormal operating procedures. Operators inserted 16 CRAM array control rods to maintain power in the proper region of the power-to-flow map. Reactor power was stabilized at approximately 52% power.

The licensee's prompt investigation revealed that within the 'D' MG set voltage regulator control box, a "burning smell" was noticed and circuit card 1CB exhibited a discoloration around the area of the 4R resistor. The resistor was blackened, showing signs of surface cracking and the resistor color bands could not be read due to discoloration. When an extent of condition walk down was performed, the other voltage regulator boxes were inspected, and except for the 'A' MG set, the remainder showed similar signs. At the time of the event, the station had been running at 100% power, in a four loop operating mode since August 29, 2003. This was a result of the 'A' Recirculation pump tripping on that date due to a suspected ground within the drywell.

Further document investigation indicated that in 1996 work orders were initiated to refurbish all 5 voltage regulators and replace the field overload relays with a new style. As part of this refurbishment, the system manager indicated a need to procure new 1CB, Mag Amp boards with two watt, 4R resistors, based on a review of a GE SIL that was issued to all GE designed BWR 3/4s. The SIL directly calls out the marginal sizing of the 4R resistor when recirculation speeds are increased beyond normal levels. The licensee's review of the work order documentation indicated that the system manager believed that the cards purchased for the refurbishment had already been modified with the two watt resistors by GE. This action was never completed and the 1CB Mag Amp boards were installed on all five MG set voltage regulator cards with the original design, one watt resistors. In addition, the licensee's review of the procurement history indicated that ten additional resistors were purchased for the specified work order, but never installed on the circuit boards. The licensee's field inspection of the purchased resistors indicated that the purchased resistors were one watt and not two watt resistors as specified in the procurement documentation.

The GE SIL identified that these 4R resistors, having a tolerance of only ±5%, could fail at higher generator speeds due to excessive temperature. That failure of this resistor will cause the generator voltage to become unstable, and a subsequent MG set trip is likely. The normal speed of the recirculation pumps is around 47 Hz, which requires the 4R resistor to dissipate 0.98 watts. During routine recirculation pump maintenance, a higher speed demand would be required of the four remaining pumps. The four remaining pumps operate between the speeds of 50 and 53 Hz when the plant is at 100% power in four loop operation. Calculations have indicated that when the MG set is running at 50 Hz, the circuit is required to dissipate 1.1, watts and at speeds nearing 54 Hz, the circuit dissipates to 1.272 watts. On average, an MG set circuit would be required to dissipate more than one watt approximately 25 days per year based on the average time the plant is operated with only four MG sets. At the time of the failure, Oyster Creek had been operating in four loop configuration for about 7 months.

Analysis. AmerGen failed to replace the voltage regulator 4R resistors with two watt resistors as recommended by industry information and their engineering staff. This is a performance deficiency in that the assessment activity of the operating experience information provided by GE, required by Oyster Creek Quality Assurance Topical Report, NO-AA-10, Rev. 72, Appendix F, failed to identify that the recirculation pump MG sets could be continuously operated at a speed that could damage the voltage regulator cards and cause a plant transient. Traditional enforcement does not apply for this finding because it did not have any actual safety consequences or the potential for impacting the NRC’s regulatory function and was not the result of any willful violation of NRC requirements.

This finding is greater than minor because it had an actual impact of tripping one of four operating reactor recirculation pumps, and therefore could be reasonably viewed as a precursor to a significant event. It increased the likelihood of a plant scram due to resultant power operation in the “buffer zone” of the core power to flow map where there is significantly reduced margin to the flow-biased high power scram setpoint. It also resulted in operation of the remaining reactor recirculation pumps at speeds that could result in pump damage without operator action to reduce the speed and resultant core flow.

This condition affects the Initiating Events Cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during power operations. The finding was determined to be of very low safety significance (Green) using a Phase 1 analysis of the NRC Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations, in that, it does not contribute to: a primary or secondary system LOCA initiator, or both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, or the likelihood of a fire or internal/external flood. This finding has a cross-cutting aspect of PI&R in that the engineering evaluation of External Operating Experience and resultant corrective action implementation was inadequate to prevent a similar condition at the site.

Enforcement. Oyster Creek Quality Assurance Topical Report (QATR), NO-AA-10, Rev. 72, Appendix F states that operating experience assessment is an activity within the scope of the QATR. Contrary to these requirements AmerGen failed to properly evaluate the recommendations to replace the 4R resistors in the MG set voltage regulator reference board as stated in the GE SIL 586 Rev. 1. The AmerGen assessment failed to evaluate the significance of continuously operating in a four loop configuration that resulted in the 'D' recirculation pump trip transient on April 1, 2004. This was entered into the AmerGen corrective action program under CAPs O2004-0785 and O2004-0821. This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 0500219/200400301)**

1R15 Operability Evaluations (IP 71111.15 - 4 Samples)

a. Inspection Scope

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

- Main Steam Isolation Valve (MSIV) failure to pass 10% closure surveillance - On April 3, 2004 during the performance of the MSIV 10% closure test, NSO3A & NSO4A failed to respond as expected. The MSIVs were then declared inoperable and subsequently tested using the full closure test. The full closure test demonstrated the operability of the MSIVs. Engineering was directed to evaluate the status of the 10% closure circuit and its effect on the MSIVs. This issue was documented in CAP 2004-0795.

- #1 EDG #12 cylinder contamination - During the #1 EDG system outage, the #12 cylinder was observed to have signs of lube oil contamination. The system engineer and the GE technical representative examined this issue and had determined that the contamination had no effect on the performance of the #12 cylinder. The licensee and NRC inspectors examined the cylinder on April 28, 2004. The liner wall was visible through the air inlet ports; the exam revealed no visible significant wear. The cross-hatching was clearly visible with ample oil film. The results of chemical analysis of the film indicated a minute jacket water leak which was determined to not affect operability. The licensee established a monitoring plan to periodically sample, test, and trend the lube oil to determine if the leakage increases. In addition, the licensee has scheduled an examination of the #12 cylinder at the next EDG maintenance work window to reassess the observed conditions. This issue was documented in CAP O2004-1032.

- 'E' EMRV 125V DC ground - A "soft" electrical ground began to appear on the 'B' 125 VDC negative circuit in January 2004; over time, the ground faded. On May 17, 2004, the ground indication suddenly changed from about 2 to 20 milliamps. Troubleshooting narrowed the ground to a circuit associated with the 'E' Electro Magnetic Relief Valve (EMRV). The ground condition was determined to cause the 'E' EMRV to be degraded but operable. During the 1FO6 maintenance outage 1000 psig drywell inspection, the 'E' EMRV pilot valve was noted as having a plume of steam from the exhaust port that was impinging on the Patel connector for the solenoid operator. Both the 'C' and 'E' EMRVs had no steam deflection elbow installed on the exhaust port as is the case for the 'A,' 'B,' and

‘D’ valves. Both the ‘C’ and ‘E’ valves were replaced and steam deflecting elbows were installed on their exhaust ports. This issue was documented in CAPs O2004-0159, O2004-0311, and O2004-1189. The operability evaluation was documented in Oyster Creek Op Evaluation # OC-2004-OE-0002.

- ESW pipe leak - On May 12, 2004, while running ESW pump 52B, a leak was observed from the weld on the ESW outlet from the 1-2 Containment Spray Heat Exchanger. The engineering operability evaluation determined that a crack had propagated through the toe of the weld between the reinforcement plate at the ESW outlet nozzle and the pipe piece to the flange face. The apparent cause was determined to be weld fatigue. Based on Technical Evaluation A2088856 E02, the subject nozzle would have maintained structural integrity with the degraded condition, using the techniques of Code Case N-513. However, due to the configuration, volumetric examinations were not successful in characterizing the flaw. As a result, the seven day limiting condition for operation was entered and the crack repaired within the allowed outage time. This evaluation was documented in CAP O2004-1153.

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 operators in responding to initiating events. The inspectors reviewed the licensee’s implementation of procedure OP-AA-102-103, “Operator Work-Around Program.” The inspectors observed physical conditions of equipment in the plant during routine tours to identify conditions that may challenge operator actions in use of mitigating systems. The inspectors also reviewed the status of the corrective actions described in CAP Nos. O2003-2320 and O2004-1157 which identified specific problem resolutions relating to the reactor building airlock doors.

b. Findings

Operator Failure to Recognize Degraded Secondary Containment Airlock

Introduction. The inspector identified a Green finding and an NCV for failure to identify a condition adverse to quality when a secondary containment airlock door was found open resulting in a momentary violation of Technical Specification 3.5.B and Procedure 312.1, “Secondary Containment Control,” Rev. 8.

Description. On May 4, 2004, while on a tour of the Reactor Building Equipment Drain Tank (RBEDT) room, the inspector and an accompanying radiation protection technician discovered the secondary containment outer airlock door interlock unexpectedly activated, preventing entry. The control room was notified and approval was given for the technician to bypass the interlock to open the outer door. However, upon entry, the technician found the inner door open. One door was immediately closed restoring the secondary containment to a fully operable condition. The technician informed the control room of the observed conditions. The inspector and technician completed the tour of the RBEDT room and exited the area. Upon exit, the inner door was verified closed. On May 5, 2004 the inspector returned to the RBEDT room outer door and discovered the airlock door interlock again to be activated. This condition was reported to the control room and the inner door was again found open.

At the time of this event, the reactor was at about 100% power, the drywell was locked closed and inerted, no other degraded conditions were affecting primary or secondary containment, and fuel handling operations were in progress in support of the CY 2004 ISFSI campaign.

Technical Specification 3.5.B.1 specifies that the secondary containment integrity be maintained at all times when the reactor is not subcritical or when operations are being performed in, above or around the spent fuel storage pool, that could preliminary cause a release of radioactive materials during an event. Technical Specification 3.5.B.2 states that upon accidental loss of secondary containment integrity, restore integrity within 4 hours. As noted above, the technician immediately closed one of the two doors restoring secondary containment integrity as required by the technical specifications.

The secondary containment system includes the reactor building envelope, associated dampers and valves, the airlock doors, and the Standby Gas Treatment System (SGTS). The design of the secondary containment is to remain in a negative pressure condition for all analyzed accidents for both power operation conditions and fuel handling conditions. The airlock doors enable the SGTS to maintain a negative pressure in the reactor building. Procedure 312.10, “Secondary Containment Control,” specifies:

> “Operate Reactor Building Airlock doors as follows:
>
> - Maintain at least one door at each opening to the reactor building closed at all times.
>
> - If at any time both doors at an opening to the reactor building are unintentionally opened at the same time, then close at least one door and report the incident to the shift supervisor to initiate repair to the airlock interlock.”

On May 13, 2004, subsequent to the inspector identifying this concern, CAP O2004-1157 documented the failure to maintain secondary containment integrity for a short time while both airlock doors were open (for the May 4 entry). This CAP also documented the failure to note the condition in the operating logs and failure to document the condition in a CAP when initially identified on May 4, 2004. Oyster Creek procedure OP-OC-100, “Conduct of Operations,” requires, in part, that if it is discovered that a technical specification limiting condition for operation could be exceeded, then the Operations' Supervisor shall ensure specified actions are taken, including issuance of a CAP in accordance with procedure LS-OC-125. Procedure LS-OC-125, “Corrective Action Program Process,” Rev. 3, requires that a CAP be initiated for conditions adverse to quality, including unplanned entries for technical specification actions for limiting conditions for operation.

Subsequent review of the CAP and maintenance databases revealed that this door has had several documented failures of the inner door automatic closure device since CY2000. Several days after the inspectors identified the concern with the airlock door, the FIN team investigated the door and determined that, in addition to the previously identified condition of the door closing device, the door latching device was sticking, resulting in the door latch not fully closing.

While the licensee’s CAP review identified that technical specification requirements were met since at least one door was immediately closed, the review revealed that operations' personnel did not demonstrate the appropriate sensitivity to secondary containment control governed by procedure. Appropriate measures were not taken when personnel trying to get in through the outer airlock door were told to override secondary containment in order to open the door, nor was an appropriate log entry or CAP initiated when it was known that both airlock doors were open at the same time.

The latch was repaired by the FIN team. The degraded door closing device is scheduled to be repaired when parts are available. An Operator Challenge has been created to track resolution of the airlock door degraded conditions. In addition, each operating crew was briefed on the importance of maintaining secondary containment integrity and the need to make appropriate log entries when challenges to containment integrity are identified.

Analysis. The failure to document in the control room logs and in a CAP report that the inner airlock door for the RBEDT room was found open on May 4, 2004, was a violation of station procedures. Accurate record keeping is needed to identify degraded conditions and effect corrective actions in a timely manner. This is a performance deficiency. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC regulatory function, and was not the result of any willful violation of NRC requirements or AmerGen procedures.

This finding is greater than minor because the failure to timely identify the condition adverse to quality for the airlock door, if left uncorrected, could have led to a more significant event involving a failure of the airlock interlock. This condition affects the Reactor Safety Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases from accidents or events. Also, this finding has a cross-cutting aspect of PI&R in that operators failed to properly initiate a Corrective Action Process (CAP) report when the degraded and open airlock door was discovered.

The airlock doors function to ensure secondary containment integrity and to support the SGTS capability to maintain a negative pressure in the reactor building and minimize ground level releases of radioactive materials. This issue was evaluated using IMC 0609 Appendix A, “Significance Determination Process for At-Power Situations.” Since the finding only adversely affected the radiological barrier function associated with the SGTS function, it was determined to be of very low safety significance (Green).

Enforcement. 10 CFR Part 50 Appendix B Criterion XVI, station procedures, LS-OC-125, “Corrective Action Program Procedure,” and general operating procedures, OP-OC-100, “Oyster Creek Conduct of Operations,” and OP-AA-111-101, “Operating Narrative Logs and Records,” require timely identification and resolution of conditions that adversely affect the performance of safety-related equipment. Contrary to these requirements, a CAP report was not initiated for a condition adverse to quality that was identified on the RBEDT room inner airlock door when the door latching device failed open, causing a momentary loss of secondary containment integrity on May 4, 2004. Operators failed to make a control room log entry to track the associated technical specification required actions for the event. Because this condition is of very low significance and has been entered into AmerGen’s corrective action program (CAP O2004-1157), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368).
(NCV 05000219/200400303)

1R17 Permanent Plant Modifications (IP 71111.17B - 10 Samples)

a. Inspection Scope

The inspectors reviewed ten risk-significant plant modification packages selected from among the design changes that were completed within the past two years. The review was to verify that: (1) the design bases, licensing bases, and performance capability of risk significant structures, systems or components had not been degraded through modifications; and, (2) modifications performed during increased risk configurations did not place the plant in an unsafe condition.

The selected plant modifications were distributed among initiating event, mitigating systems, and barrier integrity cornerstones. For these selected modifications, the inspectors reviewed the design inputs, assumptions, and design calculations to determine the design adequacy. The inspectors also reviewed field change notices that were issued during the installation to confirm that the problems associated with the installation were adequately resolved. In addition, the inspectors reviewed the post-modification testing, functional testing, and instrument and relay calibration records to determine readiness for operations. Finally, the inspectors reviewed the affected procedures, drawings, design basis documents, and UFSAR sections to verify that the affected documents were appropriately updated. For accessible components, the inspectors also performed field observation of installed equipment to detect possible abnormal installation conditions.

The following modifications were reviewed:

- ECR OC-01-00621, Crosstie ESW to Service Water to Allow Repairs
- ECR OC-01-00627, Replace Degraded Voltage Relay Timers
- ECR OC-01-00628, 480V Circuit Breaker Undervoltage Device Replacement
- ECR OC-02-00487, Replace Degraded Instrument Air Drying Towers C/D
- ECR OC-02-01441, Oyster Creek Drywell Vessel Corrosion Assessment
- ECR OC-03-00028, H430 - Modify SPF Cooling Discharge Lines with Anti-Syphon Holes
- ECR OC-03-00088, J064 - ‘A’ & ‘B’ CRD Pump Breaker Modification, Rev. 3
- ECR OC-03-00364, Release Revision 1 of Calculation C-1302-810-5450-004
- ECR OC-03-00731, EDG Fuel Oil Priming Pump Addition for EDG2 CAP 02003-1735
- ECR OC-03-00169, Revision of DC MOV Voltage Drop Calculation, “C-1302-730-5350-008, R4,” Rev 0

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

- ‘D’ Recirculation pump MG set voltage regulator card 4R resistor - Post maintenance testing was performed per work order C2007667 on April 5, 2004, following failure of the ‘D’ Recirculation pump MG set voltage regulator card 4R resistor.

- #1 EDG cooling fan - Post maintenance testing was performed per work order C2008008-04 and procedure 636.4.003, “Diesel Generator #1 Load Test” on May 17, 2004, following failure of the EDG cooling fan shaft bolting.

- MSIV, NSO3A - Post maintenance testing was performed on June 1, 2004, per work order C2007828 and procedure 602.4.004 (10% partial closure test) following troubleshooting activities during 1FO6 as a result of a prior test failure in April 2004 (CAP O2004-0795). The valve again failed the partial closure test. The valve was immediately inspected by maintenance and engineering who recommended lubricating the guide rods and adjusting the air exhaust port to allow air to vent faster.

- MSIV, NSO3A - Post maintenance testing involving full-closure testing was conducted on June 2, 2004, per work order A2086259 Eval 02 and procedure 602.4.005, following maintenance to lubricate and adjust the air venting capability of the valve operator due to failing the 10% closure test on June 1.

- ESW piping tie in - Surveillance 607.4.016 and 607.4.017, “Containment Spray and Emergency Service Water System Operability and Quarterly In-Service Test” was performed after ESW piping tie into complete modification changes to ESW piping during the week of May 30, 2004.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (IP 71111.20 - 1 Sample)

1FO6: Maintenance Outage to Replace the “A” Reactor Recirculating Water Pump Motor

a. Inspection Scope

The inspectors observed outage maintenance activities for the 1FO6 maintenance outage and verified those activities were performed in accordance with plant procedures. In addition, during the outage, the inspectors reviewed the daily outage risk assessments and verified the equipment alignments used to support the assessments. The inspectors also monitored the availability of the decay heat removal system due to a high decay heat condition throughout the maintenance outage. The inspectors observed portions of the shutdown and cooldown on May 26 and 27, 2004. During the reactor startup, the inspectors physically observed portions of the primary containment (drywell) with plant operators to verify part of the RCS 1000 psig visual leakage inspection. During the plant startup, which began on May 27, 2003, the inspectors observed and verified adherence to procedure No. 201, “Plant Startup.” The inspectors continued to observe control room startup activities until full power was achieved on June 3, 2004.

b. Findings

On June 1, 2004, while restoring the shutdown cooling system to a normal, standby readiness, a momentary loss of shutdown cooling occurred. This issue was followed up by the inspector. Details of this event are described in Section 4OA3 of this report.

1R22 Surveillance Testing (IP 71111.22 - 6 Samples)

a. Inspection Scope

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

- Procedure No. 636.2.004, "Diesel Generator Battery Discharge Test," completed on April 26, 2004
- Procedure No. 619.3.008, "Low Pressure Main Steam Line Functional Calibration While Operating," completed on April 7, 2004
- Procedure No. 610.4.022, "Core Spray System Surveillance," completed on May 2, 2004
- Procedure No. 636.4.003, "Diesel Generator #1 Load Test," completed on May 17, 2004
- Procedure No. 602.4.003, "Electromatic Relief Valve Operability Test," completed on June 1, 2004
- Procedure No. 665.5.005, "Drywell Airlock Leak Rate Test," completed on June 2, 2004

b. Findings

During the testing of the #1 EDG on May 17, 2004, a human performance-related finding was identified. See Section 4OA4 for the details of this finding.

1R23 Temporary Plant Modifications (IP 71111.23 - 2 Samples)

a. Inspection Scope

Two samples were selected for review by the inspectors. The inspectors reviewed a Temporary Modification (TM) associated with the temporary routing of the signal cable for the air ejector off gas radiation monitor . The inspectors reviewed the associated implementing documents to verify the plant design basis and the system operability was maintained, which included CC-AA-112, "Temporary Configuration Changes," Rev. 6. The inspectors also reviewed a TM associated with a temporary air connection to purge the torus in support of the RCS 1000 psig inspection. The inspectors reviewed the associated implementing documents to verify the plant design basis and the system operability was maintained, which included CC-AA-112, "Temporary Configuration Changes," Rev. 6. The TM allowed for the inspection at 1000 psig reactor pressure by purging the torus of nitrogen. The inspectors verified that the temporary modification was removed in accordance with station procedures.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (IP 71114.06 - 1 Sample)

a. Inspection Scope

The inspectors observed an emergency preparedness (EP) drill from the control room simulator and the technical support center on May 6, 2004. The inspectors evaluated the conduct of the drill and AmerGen's performance related to emergency action level classifications, notifications, and protective action recommendations. The drill contained ten opportunities that are tracked by the NRC Drill/Exercise Performance (DEP) performance indicator. The inspectors also reviewed several condition reports (CAP Nos. 2004-1113, 2004-1114, 2004-1116, and 2004-1120) associated with EP areas for improvement identified during the drill.

The inspectors reviewed the following documents:

- Oyster Creek EP Drill 5/6/04 Scenario, Rev. 2
- EP-OC-1010, "Radiological Emergency Plan For Oyster Creek Generating Station," Rev. 1
- EP-AA-125, "Emergency Preparedness Self Evaluation Process," Rev. 2

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (IP 71121.01 - 1 Sample)

a. Inspection Scope

The inspector toured areas controlled as High Radiation Areas and reviewed the effectiveness of access control to these areas. The inspector physically inspected and challenged three locked High Radiation Area access points to determine if access controls were sufficient to preclude unauthorized entry.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation (IP 71122.02 - 1 Sample)

1. Inspection Planning/In-Office Inspection

a. Inspection Scope

The inspector reviewed the solid waste system description in the UFSAR and the recent radiological effluent release report (2003) for information on the types and amounts of radioactive waste.

b. Findings

No findings of significance were identified.

2. System Walkdown

a. Inspection Scope (71122.02 - sample not completed)

The inspector walked down selected accessible portions of the station's radioactive liquid and solid waste collection, processing, and storage systems and locations to determine if: systems and facilities were consistent with descriptions provided in the UFSAR; to evaluate their general material conditions; and to identify changes made to systems. Areas visually inspected were the high purity waste collection tank, floor drain collector tank, waste neutralizer tanks, concentrated liquid waste tanks, chemical waste de-watering filter (A&B), high purity waste filter and outdoor storage tanks (waste sample tanks, chemical waste distillate and floor drain sample tanks). Also reviewed and toured were various pump and building areas throughout the two radwaste facilities. In addition, the inspector toured outdoor yard storage areas and toured the low-level waste storage facility.

The inspector selectively reviewed the following topics:

- the status of non-operational or abandoned radioactive waste process equipment and the adequacy of administrative and physical controls for those systems;
- changes made to radioactive waste processing systems and potential radiological impact including conduct of safety evaluations of the changes, as necessary;
- current processes for transferring radioactive waste resin and sludge to shipping containers and mixing and sampling of the waste, as appropriate;
- radioactive waste and material storage and handling practices;
- sources of radioactive waste at the station, processing (as appropriate) and handling of the waste; and,

- the general condition of facilities and equipment and licensee actions on apparent deficient conditions, as appropriate.

The review was against criteria contained in the station's UFSAR, 10 CFR Part 20, 10 CFR 61, the Process Control Program (PCP), and applicable station procedures.

b. Findings

No findings of significance were identified.

3. Waste Characterization and Classification

a. Inspection Scope (71122.02 - sample not completed)

The inspector selectively reviewed the following topics:

- radio-chemical sample analysis results for radioactive waste streams and waste types;

- the development of scaling factors for difficult to detect and measure radionuclides;

- methods and practices to detect changes in waste streams;

- classification and characterization of waste relative to 10 CFR 61.55 and 10 CFR61.56;

- implementation of applicable NRC Branch Technical Positions (BTPs) on waste classification, concentration averaging, waste stream determination, and sampling frequency;

- current waste streams and their processing relative to descriptions contained in the UFSAR and the station's approved PCP (RW-AA-100, Rev. 2);

- current processes for transferring radioactive waste resin and sludge discharges into shipping/disposal containers to determine adequacy of sampling;

- revisions of the PCP and the UFSAR to reflect changes (as appropriate); and,

- implementation of Procedure RP-AA-605, 10 CFR61 Program, Rev. 0.

The review was against criteria contained in 10 CFR 20, 10 CFR 61, 10 CFR 71, the UFSAR, the Process Control Program, applicable NRC Branch Technical Positions, and AmerGen's procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (IP 71151)

a. Inspection Scope

The inspectors reviewed the Performance Indicator (PI) data from January 2003 through December 2003 for Emergency Diesel Generator Unavailability and for Scrams with a Loss of Normal Heat Removal to verify their accuracy. The inspectors reviewed AmerGen's process for identifying and documenting the PI data as described in OC procedures LS-AA-2040 Rev. 4, "Monthly PI Data Elements for Safety System Unavailability," and LS-AA-2003 Rev. 0, "Use of the INPO Consolidated Data Entry Database for NRC and WANO Data Entry," and compared the data using criteria contained in NEI 99-02 Rev. 2 to verify it was properly dispositioned in the PI reports.

The inspectors also reviewed operator log entries for EDG out of service time and interviewed the EDG system engineer to discuss the criteria used to determine EDG unavailability.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (IP 71152)

1. Routine Screening of Items Entered in the Licensee's CAP Program

a. Inspection Scope (71152)

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 AmerGen's corrective action program. This review was accomplished by attending daily screening meetings and management review meetings, and by accessing the licensee's computerized database.

b. Findings

No findings of significance were identified.

2. <u>Identification and Resolution of Problems Associated with Select 10 CFR 50.59 Issues and Plant Modifications</u>

a. <u>Inspection Scope</u>

The inspectors reviewed corrective action process (CAP) reports associated with selected 10 CFR 50.59 issues (Section 1R02) and plant modification issues (Section 1R17) to ensure that the licensee was identifying, evaluating, and correcting problems associated with these areas and that the planned or completed corrective actions for the issues were appropriate. The inspectors also reviewed four audits and self-assessment reports related to engineering activities, including 10 CFR 50.59 safety evaluation and plant modifications at Oyster Creek.

The listing of the condition reports and self assessments reviewed is provided in Attachment 1.

b. <u>Findings</u>

No findings of significance were identified.

3. <u>Identification and Resolution of Problems Associated with Radioactive Waste Handling</u>

a. <u>Inspection Scope</u> (71122.02 - sample not completed)

The inspector selectively reviewed assessments of the radioactive waste handling, processing, storage, and shipping programs including the PCP. The inspector also reviewed selected corrective action documents written since the previous inspection. The following documents were reviewed:

- Chemistry, Radwaste, and Process Control Program audit # NOSA-OYS-04-04 (AR00214001), dated April 28, 2004
- Various Radwaste audit templates
- April 2004 Focused Self-Assessment (Report 4017-RE-027)
- Corrective Action Documents (CAPs: 2004-0334, 2004-1571, 2004-1572, 2004-0429, 2004-0964, 2004-0968, 2004-0971, 2004-0972, 2002-1820).

The review used criteria contained in 10 CFR 20 Appendix G, 10 CFR 71.101, and applicable station audit and surveillance procedures.

b. <u>Findings</u>

No findings of significance were identified.

4. <u>Semi-Annual Review of Corrective Action Program Trends</u> (IP 71152)

a. Inspection Scope (IP 71152 - 1 sample)

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

- Oyster Creek Nuclear Safety Review Board Meeting, February 9 and 10, 2004
- Oyster Creek Nuclear Safety Review Board Observation Visit, October 22 thru 28, 2003
- Nuclear Oversight Quarterly Report, NOSPA-OC-04-1Q, January - March 2004
- Security Plan, FFD, Access Authorization, PADS Audit Report, NOSA-OYS-04-02, February 23 - 27, 2004
- NOS Maintenance Functional Area Audit Report, NOSA-OYS-04-01, March 8 - 19, 2004

b. Findings

No findings of significance were identified.

5. Annual Sample Review (IP 71152 - 1 sample)

a. Inspection Scope

Observations

The inspector reviewed AmerGen's efforts to identify and correct problems with the recirculation system to return the system to five (5) pump, automatic operation. These efforts were successfully completed during a maintenance outage in late May/early June 2004, although additional troubleshooting was undertaken at the end of this inspection period due to intermittent, small flow oscillations on the "C" MG set that prompted AmerGen to temporarily place that MG set in manual.

The inspector discussed the current condition of the system with the responsible system engineer as well as the basis for, and the adequacy of the corrective actions taken to date to return the system to five (5) pump, automatic operation. He reviewed twelve (12) corrective action program reports in 2002-2004 related to recirculation system equipment issues, three A/Rs related to the "A" recirculation pump motor failure in August 2003 and the replacement of the scoop tube positioner for the "C" MG set, and long-term plans to maintain and improve the reliability of the system. The inspector noted that AmerGen's long-term improvement plans are tentative, but involve refurbishing MG sets B, C, D & E between 2005 - 2008 ("A" was completed in June 2004) as well as replacing recirculation pump motors B & C by the end of 2008 ("D" & "E" were purportedly replaced in the last decade while "A" was replaced in June 2004.) AmerGen is also planning to replace the cooling coil inside the "A" and "E" recirculation pump motors by the end of 2008 ("B", "C" & "D" pumps had their cooling coils replaced in the recent past).

The inspector noted that pending the failure analysis and rebuilding of the "A" recirculation pump motor, AmerGen does not have a spare recirculation pump in the event of a future motor failure. Moreover, AmerGen also lacks a spare MG set or fluid coupling positioner in the event of a future MG set equipment failure. Thus it will be challenging to keep the recirculation system in five (5) pump, automatic operation until AmerGen's long-term reliability improvements are completed in 2008.

b. Findings

No findings of significance were identified.

6. PI&R Cross-cutting Aspects of Findings Described Elsewhere in the Report

The inspectors identified a non-cited violation of the Oyster Creek Quality Assurance Program for failure to adequately evaluate operating experience information and correct a condition affecting the recirculation pump voltage regulator card that resulted in the trip of the 'D' recirculation pump during four loop full power operation. This finding has a cross-cutting aspect of PI&R in that the engineering evaluation of external operating experience and corrective action implementation were inadequate to prevent a similar condition at the site. (Section 1R14).

The inspectors identified a non-cited violation for failure to identify a condition adverse to quality when a secondary containment airlock door was found open, resulting in a momentary violation of Technical Specification 3.5.B.2. This finding has a cross-cutting aspect of PI&R in that operators failed to initiate a CAP report for this condition as required by the CAP Process Procedure. (Section 1R16)

4OA3 Event Follow-up (IP 71153)

a. Inspection Scope

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

- ISFSI transfer truck hydraulic failure during spent fuel move on April 16, 2004
- 69 KV switchyard fire and voltage transient causing a loss of the 1E1 bus on April 20, 2004
- Loss of Shutdown Cooling event on May 31, 2004
- 230 KV switchyard fire and voltage transient causing a loss of the 1E1 bus and the "S1B" startup transformer on June 30, 2004

b. Findings

Inadequate Procedure Results in a Temporary Loss of Shutdown Cooling Capability

Introduction. A self-revealing event involving an inadvertent loss of shutdown cooling resulted in a Green finding and an NCV for failure to establish and maintain appropriate procedural requirements for the operation of the shutdown cooling system, as prescribed by Technical Specification 6.8.1 and the Oyster Creek Operational Quality Assurance Plan, NO-AA-10, Revision 72.

Description. On May 31, 2004, while the plant was in a cold shutdown condition and the shutdown cooling system was in service, technicians were performing Attachment 305-8 of procedure 305, "Shutdown Cooling System Operation." At the time, activities were in progress to remove the bypass jumpers for the reactor recirculation loop temperature shutdown cooling isolation logic in order to continue to restore the shutdown cooling system to a standby readiness condition prior to commencing a reactor startup and heat-up. The isolation logic bypass jumpers are normally installed during a plant maintenance or refueling outage to prevent an inadvertent trip of the shutdown cooling system due to a false trip signal. During the maintenance outage, the reactor recirculation loop temperature circuitry had maintenance conducted on it that introduced a trip signal into the logic. This trip signal was not reset as part of the maintenance. Also, there were no indications or alarms associated with this trip function being in an actuated state. When the technician removed the jumper as part of the system restoration process (per Attachment 305-8), the previously actuated logic was no longer bypassed and the shutdown cooling system tripped off.

Prior to commencing the activity to remove the bypass jumpers, operators verified that associated temperature instrumentation was operable and that the indicated RCS temperature was about 178 degrees with adequate margin to boiling and well below the logic actuation temperature of 350 degrees. During the pre-evolution brief, operators discussed the possibility that removing the jumpers could introduce an isolation trip of shutdown cooling. This was not an expected condition of the activity, however, recovery actions were discussed in case of the need. Operators responded to the trip of the shutdown cooling system by verifying that the isolation trip was inadvertent, reset the trip actuation logic, and then restored shutdown cooling to service. The system was restored in about 14 minutes and RCS temperature increased by about 4 degrees. The initial RCS temperature was 178 degrees, and the post-transient temperature was 181 degrees.

The loss of shutdown cooling resulted in an unplanned entry into a high outage risk condition as determined by the licensee's outage risk management program. At the time of the event, the licensee had calculated outage risk to be Yellow due to the high decay heat condition of the RCS. On May 31, 2004, the calculated time to boil was about 95 minutes. The loss of the shutdown cooling system resulted in a calculated Red shutdown risk condition as indicated by the licensee's ORAM-Sentinel Model. The licensee responded to this condition appropriately; however, certain communications about the event were not timely. This condition was documented by the licensee in CAP O2004-1392.

Analysis. Procedure 305, Attachment 305-8, "Bypassing Isolation Interlocks for the Shutdown Cooling System Isolation Valves," 'Restoration Section' did not include an appropriate step to ensure that the isolation logic was reset prior to removing the bypass jumpers. This resulted in an unexpected trip of the shutdown cooling system on May 31, 2004. This is a performance deficiency. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC regulatory function, and was not the result of any willful violation of NRC requirements or AmerGen procedures. The finding was more than minor because the procedural control deficiency actually led to a trip of the shutdown cooling system isolation actuation logic and a resultant loss of the normal shutdown decay heat removal capability.

In accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," the inspector determined that the finding was of very low safety significance (Green) because: (1) while it resulted in an actual loss of the shutdown cooling system, the resultant reactor coolant temperature rise was very low, and not considered a loss of control event since the temperature rise (about 4 degrees) relative to the margin to boil which was less than 0.2 times the final margin to boil (about 31 degrees); (2) per Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs," Checklist 6, "BWR Cold Shutdown or Refueling Operation; Time to Boil < 2 hours: RCS level < 23' Above Top of Flange," the deficiency involved an inadequate operating procedure for the decay heat removal function while shutdown, [Checklist 6 Event I.B (1)], that: (i) did not increase the likelihood that a loss of decay heat removal would occur due to failure of the system itself or support systems; (ii) did not include decay heat removal instrumentation or vessel level instrumentation such that degraded core cooling could not be detected; (iii) did not increase the likelihood of a loss of RCS inventory, or that could result in a loss of RCS level instrumentation; (iv) did not involve a design or qualification deficiency; and, (v) did not result in an actual loss of safety function for risk-significant equipment with respect to internal or external events.

The inspector noted that the operators prepared for the possible loss of the shutdown cooling system as part of the evolution and carried out the appropriate steps to recover the system with a minimal rise in RCS temperature while maintaining an adequate margin to boil. AmerGen entered this finding into their corrective action program as CAP O2004-1392.

Enforcement. Oyster Creek Technical Specification 6.8.1 requires that procedures be established, implemented, and maintained, in part, for applicable procedures recommended in Appendix “A” of Regulatory Guide 1.33 as referenced in the Oyster Creek Operational Quality Assurance Program, NO-AA-10, Rev. 72. Appendix “A” of Regulatory Guide 1.33 includes operating procedures for the Shutdown Cooling System. Contrary to the above, Oyster Creek Procedure 305, Shutdown Cooling System Operation, Rev. 83, Attachment 305-8, was not adequately established or maintained. As such, it did not include the required actions to prevent an inadvertent isolation of the Shutdown Cooling System while restoring the isolation interlock to a normal configuration. This led to a loss of shutdown cooling on May 30, 2004. Because this condition is of very low significance and has been entered into AmerGen’s corrective action program (CAP O2004-1392), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368). **(NCV 05000219/200400304)**

4OA4 Cross-Cutting Aspects of Findings

1. Human Error Involving Procedure Adherence Violation Results in a Loss of Emergency Diesel Generator Capability

a. Inspection Scope (IP 71111.22)

During a review of a problem with the cooling system during the conduct of the #1 EDG load test on May 17, 2004, the inspectors observed the physical condition of the cooling system fan shaft support and associated bolting, interviewed workers who had conducted maintenance on the diesel generator during a 2-year overhaul in April 2004, reviewed AmerGen’s root cause investigation conducted per CAP O2004-1184, and reviewed AmerGen’s technical evaluation of EDG functionality with the loose bolting as documented in action request (AR) A2089090.

b. Findings

Introduction. A self-revealing apparent violation, having potential safety significance greater than very low significance, was identified for failure to implement appropriate procedural requirements for maintenance on the #1 EDG during an overhaul conducted April 26 - 30, 2004, as prescribed by Technical Specification 6.8.1.

Description. On May 17, 2004, while the plant was at 100 % power, operators were performing a loaded, operational surveillance test of the #1 EDG. After completing the required loaded run, operators observed noise and vibration from the engine cooling fan while the diesel was in its post-run cool-down cycle. Based on the observed condition, the diesel was emergency stopped. An inspection revealed that the bolting on the engine cooling system fan shaft pillow-block assembly had loosened, with one bolt removed and the second nearly detached. Operators declared the diesel generator inoperable and repairs commenced to replace the bolting.

A root cause review of the failure determined that maintenance conducted during a two-year overhaul on April 26 - 30, 2004, had replaced the diesel cooling system fan drive belts. AmerGen's root cause investigation, as well as independent review by the resident inspectors, determined that the technicians failed to follow written procedures to torque the cooling fan shaft bearing bolts following fan belt replacement. The poor maintenance practices led to the high vibrations during surveillance testing and manual shutdown of the #1 EDG on May 17, 2004. The high vibrations increased the potential for a failure of the #1 EDG due to loss of the skid-mounted cooling system. The final determination of risk significance is pending potential additional information from AmerGen concerning the ability of the #1 EDG to perform its safety function, given the high vibrations and the loose cooling fan shaft bearing bolts.

Analysis. The performance deficiency was that AmerGen failed to implement appropriate procedural requirements for the maintenance on the #1 EDG with respect to the fastening of the cooling fan shaft bearing bolts. The finding was more than minor because it affected the mitigation system cornerstone objective to ensure the availability, reliability, and capability of systems (emergency AC power) that respond to initiating events to prevent undesirable consequences, and the related attributes of equipment performance, human performance, and procedure quality.

The inspectors conservatively assumed, based on initial information, that the #1 EDG would have been unable to perform its safety function for 17 days (April 30 - May 17). This assumption was made due to the large degree of uncertainty associated with the ability of the #1 EDG to operate with the high vibrations and loose cooling fan shaft bearing bolts.

Using the 17-day exposure time, the finding was preliminarily evaluated in accordance with Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The Phase 1 screening identified that a Phase 2 analysis was needed because the #1 EDG would have been inoperable in excess of its Technical Specification Allowed Outage Time of 7 days.

The inspectors conducted a bounding Phase 2 evaluation using the Risk-Informed Inspection Notebook for Oyster Creek Nuclear Generating Station, Revision 1. An exposure time of greater than three days and less than thirty days was used. Worksheet Table 3.5, LOOP, was evaluated using IMC 0609 Appendix A rule 1.6 for an EDG finding. Operator recovery of the emergency AC power function using the nearby First Energy Combustion Turbines was not credited (see note 1. on Table 3.5). The SDP Phase 2 resulted in an internal event delta CDF in the mid E-6 range.

The NRC Headquarters SRA conducted a preliminary Phase 3 analysis, as discussed below, for internal and external initiating events, indicated a delta CDF in the mid-E-6 range and a delta LERF in the mid-E-7 range. As such, the finding could be of low - moderate safety significance (WHITE). However, due to the variability in the outcome of the analysis based on the potential ability of the #1 EDG to perform for its safety function for a portion of its mission time, the significance of this performance deficiency is preliminarily characterized as Greater Than Green. Additional information from AmerGen concerning the ability of the #1 EDG to perform its safety function would facilitate more refined risk analysis. This information would include a clearly articulated and effectively supported applicability analysis of testing conducted on an EDG not exactly similar to the #1 EDG at Oyster Creek.

Delta CDF: The Oyster Creek SPAR model, updated with NUREG 5496 LOOP initiating event frequencies and associated offsite power non-recovery probabilities, was used. The #1 EDG was failed by setting the test and maintenance term, EPS-DGN-TM-DG1, to true for a 17-day period, resulting in a delta-CDF in the mid-E-6 range. The risk increase was dominated by a LOOP with failure of the other diesel (Station Black Out (SBO) sequence) and the failure to recover offsite power prior to core damage. The analysis also included a review of external initiating events, finding that a LOOP caused by a fire contributed in the low E-7 range to the delta CDF total. The other external events (flooding, high winds, and earthquakes) did not contribute significantly to the total increase in CDF.

Delta LERF: Revised Manual Chapter 0609, appendix H, "Containment Integrity Significance Determination Process," was used to evaluate the impact of this performance deficiency on the Large Early Release Frequency (LERF). The appendix H factors relevant to this issue involve SBO sequences that result in reactor vessel breach with a dry containment floor. For such SBO sequences Appendix H gives a multiplier of 1.0 at BWRs with a Mark 1 containment. However, through discussions with the licensee, the analyst discovered several factors that should be credited for LERF mitigation at Oyster Creek. Mitigation possibilities include AC recovery and injection via core spray prior to vessel breach, fire water injection, and a unique concrete berm in containment that could be effective in containing core debris. By taking these factors into consideration, the senior reactor analyst determined that a more appropriate LERF multiplier would be 0.1. Therefore, the increase in LERF was estimated at CDF * 0.1 or in the Mid-E-7 range.

Review of AmerGen Analysis: The SRA also reviewed the results of a preliminary AmerGen risk analysis, which assumed that the #1 EDG was inoperable for the 17 days. This review found that the SPAR results generally reflect higher risk for this condition. However, the licensee model was non-conservative relative to offsite power failure rates and recovery probabilities. Adjusting the AmerGen result with the higher LOOP frequencies from NUREG 5496 would result in a Low-E-6 increase in CDF. The adjustments for LOOP non- recovery probabilities would further increase the licensee results. Therefore, it was concluded that AmerGen's preliminary results were reasonably consistent with the preliminary SDP results.

Enforcement. Oyster Creek Technical Specification 6.8.1 requires the licensee to establish, implement, and maintain written procedures, in part, for maintenance that can adversely affect the performance of safety-related equipment and for surveillance and test activities of equipment that affects nuclear safety. Work Order (R2017655), the Diesel Generator Inspection (24 Month) Surveillance Test (Procedure 636.1.010), and maintenance instruction, M.I. 1200, were procedures required by TS 6.8.1, to control the activities conducted during the 2-year overhaul of the #1 EDG completed on April 30, 2004. On April 30, 2004, the technicians completed work on the #1 EDG without fully implementing Maintenance Instruction 1200 requirements when replacing the fan belts for the cooling system, in that specified torque values were not used when returning the associated pillow-block bolting to its required configuration. This led to a high vibration and imminent loss of the EDG cooling system on May 17, 2004. This condition was evaluated as a preliminary Greater Than Green finding, pending determination of the final safety significance. As such, this violation was treated as an apparent violation. **(AV 05000219/200400305)**

4OA5 Other Activities

Offsite Power System Operational Readiness (TI 2515/156)

a. Inspection Scope

The inspectors interviewed station personnel in order to confirm the operational readiness of offsite power systems in accordance with NRC requirements prescribed in General Design Criterion 17 of 10 CFR 50, Appendix A; 10 CFR 50, Appendix B Criterion XVI; plant technical specifications for offsite power systems, 10 CFR 50.63; and 10 CFR 50.65(a)(4). The inspectors also evaluated the licensee's response to various questions concerning the maintenance rule, station blackout, corrective action, and offsite power design robustness and quality.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On July 15, 2004, the resident inspectors presented the inspection results to Mr. C. N. Swenson, Senior Vice President and other members of licensee management. The licensee acknowledged the findings presented. In addition, the resident inspectors reviewed the findings and discussion from the visiting inspectors that were presented in exit meetings conducted April 2, June 23 and July 7, 2004. 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 violation of very low-safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- 10 CFR 20 Appendix G, requires that the licensee establish and implement a Quality Assurance (QA) program to assure compliance with the requirements of 10 CFR 61.55. Contrary to this requirement, as of April 2004, the QA program did not assure compliance with the waste radionuclide concentration determination provisions of 10 CFR 61.55. Specifically, as of April 2004, a non-representative method (single direct sampling) was used to determine radionuclide concentrations in resin liners containing stratified resin from different waste streams. In April 2004, a vendor audit identified that the use of the direct sampling method resulted in underestimation of radionuclide concentrations and thus curie content of spent resins shipped for shallow land disposal (e.g., resin shipment No. OC4003-04-04 shipped March 9, 2004). (CAP O2004-1572; O2004-1733)

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
P. Bloss, BOP Systems Manager
A. Farenga, Radwaste Shipping Coordinator
D. Fawcett, Licensing Engineer
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
J. Renda, Radiation Protection Manager
D. Slear, Manager, Regulatory Assurance
B. Stewart, Senior Licensing Engineer
C. Swenson, Site Vice President
G. Waldrep, Quality Assurance Manager
C. Wilson, Director, Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000219/200400302	URI	(Section 1R14) Review corrective actions for IRM spiking that led to scram event on May 27, 2004.
05000219/200400305	AV	(Section 4OA4) Human performance event - failure to follow procedures led to failure of cooling system for EDG #1 on May 17, 2004.

Opened and Closed

05000219/200400301	NCV	(Section 1R14) Ineffective evaluation of operating experience information for reactor recirculation system voltage control circuit - wrong wattage resistor installed.
05000219/200400303	NCV	(Section 1R16) Failure to identify condition adverse to quality for degraded latching mechanism on secondary airlock door on May 4, 2004.

05000219/200400304	NCV	(Section 4OA3) Inadequate procedure for restoration of the shutdown cooling system led to unexpected loss of shutdown cooling on May 31, 2004.

LIST OF DOCUMENTS REVIEWED
(not previously referenced)

10 CFR 50.59 Screened-out Evaluations

OC-2002-S-0038	ECR 01-00828 - Isolation Condenser Vent Valve Replacement
OC-2002-S-0156	Reactor Water Cleanup Modification Contingency
OC-2002-S-0357	ECR OC 02-00676, ESW Pump Anti-Fouling Coating
OC-2002-S-0427	Repair of Service Water Pipe Downstream of RBCCW HX
OC-2003-S-0039	ECR 03-00098 - Remove Internals from Check Valve V-16, Rev. 0
OC-2003-S-0070	Temp Mod for Isolating HC-1-1 for Maintenance, Rev. 0
OC-2003-S-0167	ECR/DCP 03-00220 - MG Set Room Ventilation Change, Rev 0
OC-2003-S-0172	RBCCW System Pipe Support 541-1022 Off-Center, Rev. 1
OC-2003-S-0216	ESW I Underground Pipe Bypass Modification
OC-2003-S-0511	NRW Chlorine Booster Pump Mechanical Seal Upgrade, Rev. 0
OC-2004-S-0009	Control Rod Drive Reactor Head Cooling System-Installation of Cross Connect Pipe to Reactor Head Vent System, Rev. 0,
OC-2004-S-0016	ECR 04-00020 Temporary Support of 8" DR-8 Pipe
OC-2004-S-0026	Replacement of Drain Tank 1-2 and 1-5 Instrumentation, Rev. 0
OC-2004-S-0036	Fuel Oil Transfer Line Tagged Out of Service More Than 90 Days, Rev. 0
OC-2004-S-0050	Deletion of Procedure 823.31, Rev 5

Action Requests

A2069334, A2008592, A2069438

Audits & Self-Assessments

NOSA-OC-03-05, NOS Engineering Design Control Audit Report, August 19, 2003
NOS Corporate Comparative Audit Report, 2003 Engineering Design Control, October 22, 2003
Focus Area Self-Assessment Report (Engineering Fundamentals), 3rd Quarter 2003
O2003-2497, Common Cause Analysis Report, February 16, 2004

Corrective Action Report Items

O1998-1274-1, O2000-07756, O2001-0711, O2001-0711-1, O2001-1839, O2002-0496, O2002-0711, O2002-1059, O2003-0473, O2003-1735, O2004-0772*, O2004-0789* O2002-1308, 02003-0616, 02003-1075, 02003-1270, 02003-1723, 02003-1930, 02003-2454, 02004-0008, 02004-0805, 02004-0821, 02004-1387, 02004-1438, 02004-1444

Calculations

C-1302-241-E310-111	Factor of Safety Against Failure for the Containment Spray Pumps 1-1 and 1-2, Rev. 0
C-1302-730-5350-008	Oyster Creek - Generic Letter 89-10 MOVs Voltage Drop Calculation, Rev. 4
C-1302-862-E310-007	Diesel Generator Fuel Transfer Pump Inlet Pressure, Rev. 0

Drawings

DWG-01-00621-1	ESW/SW Cross-Connect Mod., Rev. 0
GE 223R0173, Sh 15	4160V System Electrical Elementary Diagram - P. T. Undervoltage, Heater, D.C Supply & DG 1 Tie
GU 3W-241-A2-1000, Sh 2	ISI Configuration Drawing Containment Spray System, Rev. 5

Procedures

CC-AA-103, Rev. 5	Configuration Change Control
LS-AA-104, Rev. 3	Exelon 50.59 Review Process
LS-AA-104-1000, Rev. 1	50.59 Resource Manual
205.94.0, Rev. 0	RPV Floodup using Core Spray
Special 02-004, Rev. 0	SW 6 Hour Out Of Service Window in 1R19

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
AR	Action Request
CAP	Corrective Action Process
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CS	Core Spray
CS/ESW	Containment Spray/Emergency Service Water
DEP	Drill/Exercise Performance
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
EMRV	Electro Magnetic Relief Valve
EP	Emergency Preparedness
ESW	Emergency Service Water
IC	Isolation Condenser
IMC	Inspection Manual Chapter
IP	Inspection Procedure
LERF	Large Early Release Frequency
MG	Motor-Generator
MSIV	Main Steam Isolation Valve

NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PCP	Process Control Program
PI	Performance Indicator
PI&R	Problem Identification & Resolution
PSIG	Pounds per Square Inch Gauge
QATR	Quality Assurance Topical Report
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water
RBEDT	Reactor Building Equipment Drain Tank
RCS	Reactor Coolant System
RPS	Reactor protection System
RTP	Rated Thermal Power
SDP	Significance Determination Process
SE	Safety Evaluation
SGTS	Standby Gas Treatment System
SSCs	Systems, Structures and/or Components
ST	Surveillance Test
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order