November 10, 2004

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

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000219/2004004

Dear Mr. Crane:

On September 30, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Oyster Creek Generating Station. The enclosed integrated inspection report documents the inspection findings, which were discussed on October 14, 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, four self-revealing findings were identified as having very low safety significance (Green). All four findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at Oyster Creek.

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

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000219/2004004
w/Attachment: Supplemental Information

cc w/encl:
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Site Vice President, Oyster Creek Nuclear Generating Station, AmerGen
Plant Manager, Oyster Creek Generating Station, AmerGen
Regulatory Assurance Manager Oyster Creek, AmerGen
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Vice President - Operations Support, AmerGen
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-219
License No.:	DPR-16
Report No.:	05000219/2004004
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	July 1, 2004 - September 30, 2004
Inspectors:	Robert Summers, Senior Resident Inspector Jeff Herrera, Resident Inspector Richard Barkley, Senior Project Engineer Joseph M D'Antonio, Operations Engineer Suresh Chaudhary, Reactor Inspector
Approved by:	Peter W. Eselgroth, Chief Projects Branch 7 Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000219/2004004; 07/01/04 - 09/30/04; Oyster Creek Generating Station; Operability Evaluations, Event Follow-up.

This report covers a 13-week period of inspection by resident inspectors, a reactor inspector and a senior project engineer. Four 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. A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, was identified for failure to adequately correct a condition adverse to quality affecting the Intermediate Range Monitor (IRM) System, resulting in a reactor scram while at 2% power operations. The reactor protection system processed IRM Hi-Hi/INOP on channels 13, 14 and 18 IRMs, while operators were driving the Source Range Monitor (SRM) detectors into the core. AmerGen initiated an investigation into the issue and CAP O2004-1314 was written in order to document the associated corrective actions to prevent recurrence.

  This finding was more than minor because it resulted in a plant scram while the reactor was critical and can reasonably be viewed as a precursor to a significant event. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspector determined that the finding was of very low safety significance (Green) using a Phase 1 Significance Determination Process evaluation, because all mitigating system equipment functions remained available. This finding had a cross-cutting aspect of Problem Identification and Resolution in that prior corrective actions were ineffective. (Section 4OA3.2).

Cornerstone: Mitigating Systems

- Green. A self-revealing non-cited violation of Technical Specification 6.8.1 was identified because procedures for restoration of the shutdown cooling system were not adequate. This resulted in the loss of shutdown cooling while removing trip logic bypass jumpers in order to restore the shutdown cooling system to power operation standby readiness requirements. Upon realization of the loss of shutdown cooling system, plant operators returned the shutdown cooling system to operation.

This finding is more 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 (Green) 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 4OA3.4)

Cornerstone: Barrier Integrity

C Green. A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, was identified for failure to adequately correct a condition adverse to quality affecting Main Steam Isolation Valve (MSIV), NS04A, which resulted in the failure of the MSIV to close during testing. Oyster Creek operators immediately closed the inboard MSIV, NS03A, in order to maintain technical specification compliance.

This finding is more than minor because if left uncorrected could have resulted in a more significant safety concern. This finding is associated with the Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radio nuclide releases caused by accidents or events and the cornerstone attribute of design control to maintain the operational capability of the containment isolation function. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspector determined that the finding was of very low safety significance (Green) using a Phase 1 Significance Determination Process evaluation, because the finding did not represent an actual open pathway in the physical integrity of reactor containment. This finding had a cross-cutting aspect of Problem Identification and Resolution in that the resolution was not timely or effective to prevent the occurrence. (Section 1R15)

C Green. A self-revealing non-cited violation of Operating License No. DPR-16, Section 2.C.(1) was identified because operators exceeded the licensed thermal power limit of 1930 megawatt thermal (MWt) by approximately 0.4% for a period of approximately 19 hours. When identified, Oyster Creek operators reduced power until steady state core thermal power was below 1930 MWt.

This finding is more than minor because if left uncorrected, the finding could become a more significant safety concern. If left uncorrected, reactor core thermal power could have exceeded the initial power level of 102% for certain analyzed plant events. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspector determined that the finding was of very low safety significance (Green), using a Phase 1 Significance Determination Process evaluation, because there were no plant events that could have resulted in a breach of the fuel barrier during the overpower condition. This finding had a cross-cutting aspect of Human Performance in that plant operators failed to recognize an alarming condition in a timely manner that contributed to the event. (Section 4OA3.3)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Oyster Creek began the inspection period at 100% Rated Thermal Power (RTP). On September 12, 2004, operators reduced power to 40% RTP in order to troubleshoot the outboard Main Steam Isolation Valve (MSIV), NS04A, which had failed to close during a routine surveillance test. On September 14, 2004, the plant was shutdown to a cold shutdown condition to make repairs to MSIV, NS04A. A reactor startup was commenced on September 23, 2004, and full power operations were achieved on September 24, 2004.

1. REACTOR SAFETY

Cornerstones: Initiating Events/Mitigating Systems/Barrier Integrity

1R04 Equipment Alignment (IP 71111.04)

a. Inspection Scope

Partial System Walkdown. (71111.04Q - 4 Samples)

The inspectors performed four partial system walkdowns during this inspection period. To evaluate the operability of the selected system(s), the inspectors checked for a correct valve lineup by comparing positions of valves with system drawings, as well as examining overall system material condition. The results of recent cleaning and NDE inspections of the air receivers, as well as minor deficient equipment conditions identified by the inspector, were discussed with the appropriate system engineers.

This inspection activity represented four samples of the following systems:

- Control Rod Drive System during operability testing on August 11, 2004
- Standby Gas Treatment System 2 during System 1 maintenance on August 23, 2004
- Service and Instrument Air System on September 22, 2004
- Service Water System during System 1 testing on September 23, 2004

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 below-listed fire areas due to the potential impact to associated mitigating systems equipment in the areas. 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, including reporting a minor deficiency with a fire door, reviewed procedure 333, “Plant Fire Protection System,” and reviewed 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. The inspector selected the following areas for direct observation of conditions:

• OB-FZ-8C, “A&B Battery Room, Tunnel & Electric Tray Room, 35' Elevation”

C OB-FZ-10B, “Chemical Lab, Laundry, Instrument Shop, 35' Elevation”

C OG-FA-21, “Augmented Off-Gas Building”

C OB-FZ-8A, “MG Set Room”

C TB-FA-26, “Battery Room South of 4160V Switchgear”

C TB-FZ-11A, “Turbine Operating Floor, 46' Elevation”

C RB-FZ-1E, “Reactor Building, 23' Elevation”

C TB-FZ-11B, “Turbine Lube Oil Storage, Pumping & Purification Areas, 0' and 27' Elevations

C TB-FZ-11D, “Turbine Building Basement Floor South End, 3'-6" Elevation”

b. Findings

No findings of significance were identified.

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

1. Quarterly Inspection Requirements

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 August 12, 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 an emergency preparedness scenario and consisted of four hours of testing/evaluation. The inspectors assessed the simulator crew’s performance during the scenario in response to both abnormal and emergency conditions. The inspectors assessed the evaluator’s assessment of the crew to verify that operator performance issues were identified and appropriate remediation was conducted to address identified weaknesses.

b. Findings

No findings of significance were identified.

2. Biennial Inspection Requirements

a. Inspection Scope

On July 20, 2004, the inspector conducted an in-office review of licensee annual operating test results for 2004. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance Significance Determination Process (SDP)”. The inspectors verified that:

- Crew failure rate was less than 20%. (Crew failure rate was 0%)
- Individual failure rate on the dynamic simulator test was less than or equal to 20%. (Individual failure rate was 0%)
- Individual failure rate on the walk-through test was less than or equal to 20%. (Individual failure rate was 0%)

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors selected two samples for review. The inspectors reviewed AmerGen’s implementation of the maintenance rule as described in Oyster Creek procedure ER-AA-310, “Implementation of the Maintenance Rule.” The inspectors verified that the selected Systems, Structures and/or Components (SSCs) were properly classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed Action

Requests (ARs), Corrective Action Program reports (CAPs), (a)(1) corrective action plans and routine preventive maintenance activities. The inspectors also discussed the current system performance, associated issues and concerns, and planned activities to improve performance with the system engineers. In addition, unavailability data was compared with control room log entries to verify accuracy of data and compliance with (a)(1) goals. AmerGen trending data was also reviewed. The two SSCs reviewed during the inspection period were as follows:

- Heater Drains and Vents System
- 480 VAC Electrical Distribution 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
- Common cause analysis performed for the Heater Drains and Vent System documented under CAP O2004-0142
- Heater Drains and Vent System maintenance rule performance assessment - Dated January 20, 2003
- System Manager red/yellow summary report for Heater Drains and Vents System
- Preventative Maintenance activities (PMs) for System 431 as of August 9, 2004
- Open Corrective Maintenance/Elective Maintenance (CM/EC) Action Requests (ARs) for System 431
- Heater Drains and Vents System monitoring basis form and system walkdown templates
- SHIP report for Heater Drains and Vents System, dated June 2004

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- 4160 VAC cable testing during the week of July 26, 2004
- Feedwater Heater, 1-B-3, level oscillation decreases feedwater temperature, causing a small power increase on July 27, 2004
- Screen wash emergency cleaning work during week of August 17, 2004
- Main generator and exciter brush inspection during week of August 18, 2004
- Loss of shutdown cooling during shutdown operations during week of September 20, 2004

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (IP 71111.15 - 5 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 five selected samples are listed below:

- Standby Gas Treatment System 1 charcoal filter not meeting technical specifications for radioactive methyl iodide removal efficiency during week of August 17, 2004

- Isolation Condenser System valve leakage on August 4, 2004

- Standby Gas Treatment System 1 fan mounting plate and uni-strut seismic support on August 25, 2004

- Control Rod Drive System, pump “A” gearbox incorrect gear teeth potentially affecting operability of the control rod drive pump on September 8, 2004

- Main Steam Isolation Valve (MSIV) poppet buffeting and failure to close due to guide rib wear on September 20, 2004

b. Findings

Introduction

A self-revealing Green NCV was identified for failure to adequately correct a condition adverse to quality affecting MSIV, NS04A, which resulted in the failure of the MSIV to close during testing.

Description

On September 11, 2004, the MSIV, NS04A, failed to pass a 10% closure surveillance test and a subsequent full closure test. The valve was declared inoperable and the inboard MSIV was closed in order to maintain compliance with Technical Specifications. An orderly shutdown was commenced in order to further investigate the cause of the failure of valve NS04A and to make necessary repairs. AmerGen’s investigation revealed that the outboard MSIV, NS04A, had accumulated excessive rib guide wear which prevented the valve from fully closing. This issue had been identified previously as documented in an August 10, 1993, General Electric Services Information Letter (SIL).

The 1993 GE SIL identified that Atwood and Morrill MSIVs had failed to close because the valve poppet (main disk) hung up on grooves worn into the guide ribs. These grooves were created by constant steam flow buffeting, creating vibration and or rocking of the poppet. The poppet guide rings rest on guide ribs while in the fully open position and the SIL stated degradation of the guide ribs could prevent the MSIV from closing. The SIL recommended two basic actions for owners of Atwood and Morrill MSIVs. The first recommendation was to modify the quarterly testing valve stroke to ensure that valve travel was greater than the pilot disc travel alone. General Electric recommended a value of 85% open to allow for tolerances in setting valve operator limit switches. In addition to 85% valve travel, criteria for test termination were recommended, including terminating the test if the valve travel time exceeded 15 seconds, or when a reduction of main steam line flow occurs. This test criteria allowed for valid indication of poppet valve travel and a determination whether the poppet may be stuck. The second recommendation from the SIL consisted of valve modifications to prevent or minimize damage to the guide ribs due to vibration during normal operation. Two main modifications were recommended in the SIL, including installing an anti-rotation device for the stem and poppet to reduce or prevent rocking and cutting of the guide ribs, and a retrofit poppet and cover to allow the poppet to be back-seated against the cover when the valve is fully open to minimize poppet vibration. The anti-rotation modifications were fully implemented at Oyster Creek. The back-seat modification, however, had only been completed on one of the four MSIVs, not including valve, NS04A.

The initial response to the SIL took credit for performing a full closure stroke test of the MSIVs and the back-seat modifications were to be evaluated for possible future installation. On February 1, 1994, the NRC issued Information Notice 94-08 addressing the potential for 10% closure surveillance testing to fail to detect an inoperable main steam isolation valve due to the poppet being stuck in the full open position from rib guide wear. This Information Notice identified that the 85% open limit switch would provide positive identification of poppet movement. Oyster Creek reviewed the Information Notice and determined no actions were required due to use of quarterly full closure testing employed at the time. On March 1, 2001, Oyster Creek submitted a Technical Specification change request to permit performance of full closure testing every cold shutdown in lieu of quarterly testing and replaced the quarterly test with a 10% closure test requirement. This change request did not re-evaluate the adequacy of the 10% closure testing to incorporate the SIL recommendations or the actions described in the NRC Information Notice 94-08.

Oyster Creek engineering originally intended to install the back-seat modifications during shutdown opportunities in which other MSIV valve maintenance was performed. This schedule was consistent with the SIL recommendations. However, these modifications were not implemented during such opportunities. Further, AmerGen's investigation revealed that there were no currently scheduled plans to implement the backseat modification, and no evaluations or other corrective actions to prevent or minimize guide rib wear, or to positively monitor or trend for this adverse condition.

During the 1R19 outage in October 2002, Oyster Creek installed strain gauges on two MSIVs in order to support valve actuator replacement. At that time, strain gauge tests had revealed a resistance to valve travel in valve, NS04A. However, since this was the initial use of this testing method, the anomaly was not understood nor the cause determined.

On September 11, 2004, valve NS04A, the outboard MSIV on the 'A' main steam line failed to pass a 10% closure test. A subsequent full closure test was performed in order to verify operability and the valve failed to close once again. AmerGen operations staff closed the inboard MSIV, NS03A, in order to comply with technical specifications and subsequently began reducing power in order to investigate and repair valve NS04A.

Analysis

AmerGen failed to adequately implement the recommendations provided in General Electric SIL No. 568 resulting in the failure of the MSIV to close, a repeat of the type of event at the River Bend Station described in the 1993 SIL and the 1994 NRC Information Notice. This is a performance deficiency in that 10 CFR 50, Appendix B, Criterion XVI, specifies that measures shall be established to assure that conditions adverse to quality such as deficiencies are promptly identified and corrected. This condition was reasonably within the licensee's ability to foresee and prevent. 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 more than minor because if left uncorrected the finding could result in a more significant safety concern. This finding is also associated with the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers (containment) protect the public from radio nuclide releases caused by accidents or events and the cornerstone attribute of design control to maintain the operational capability of the containment isolation function.

The finding was determined to be of very low safety significance (Green) using phase 1 analysis of the SDP for Reactor Inspection Findings At-Power, in that, the finding: does not represent a degradation of the radiological barrier function provided for the control room, or auxiliary building, spent fuel pool, or Standby Gas Treatment System (SGTS); does not represent a degradation of the barrier function of the control room against smoke or a toxic atmosphere; does not represent an actual open pathway in the physical integrity of reactor containment, or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the reactor containment.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality, such as deficiencies and non-conformances are promptly identified and corrected. Contrary to these requirements, AmerGen failed to timely implement the installation of the back-seat modification provided in the 1993 SIL 568 and take proper action to reduce the MSIV susceptibility to rib guide wear and subsequent failure to close. This was entered into the AmerGen corrective action program under CAP O2004-2499. Because of the low safety significance and since the issue has been entered into the licensee's corrective action program, this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000219/200400401)**

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

a. Inspection Scope

The inspectors reviewed the operator work-around database and a sample of the associated corrective action items to identify conditions that could adversely affect the operability of mitigating systems or impact human reliability in responding to initiating events. The inspector reviewed the licensee's implementation of procedure OP-AA-102-103, "Operator Work-Around Program." The inspector attended an Operator Work-Around Review Board meeting on September 8, 2004, that reviewed all operator work-around issues that are currently listed in the licensee's database to assess individual corrective action prioritization, as well as the aggregate impact of these equipment performance issues.

The board was widely attended by station senior management. The board concluded that the current issues are being corrected with appropriate priorities within the engineering and maintenance organizations, and that the aggregate impact of the work-arounds were being reasonably managed by the operations organization.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Annual. The inspectors reviewed one permanent plant modification, Engineering Change Request (ECR) Number OC 04-00163, "Installation of Permanent Repair Clamp on Service Water Piping." This ECR provides for a leak enclosure clamp on the 20 inch service water discharge piping downstream of the reactor building closed cooling water system and replacement of the fiberglass wrap previously installed as a temporary modification in order to seal the leak. This area of the service water piping was evaluated to have an impact on secondary containment integrity due to the fact that the water at this point "falls" to the seal well, creating a vacuum in the pipe. An existing hole in the pipe due to erosion would draw air into the pipe and release it to the environment outside the reactor building. The new clamp is considered a permanent design change to the plant pipe. The ECR provided a determination of the adequacy of the seismic qualification levels for the piping, including the additional weight supplied by the piping clamp. The inspectors verified that the clamp was installed per instructions and specifications outlined in the associated work order, and that the modification has maintained the system availability, reliability, and functional capability of the service water system and secondary containment.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (IP 71111.19 - 6 Samples)

a. Inspection Scope

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

- 665.3.003, "MSIV Local Leak Rate Test," performed on September 21, 2004
- 602.4.002, "MSIV Closure and In Service Test," Rev. 29 performed on September 21, 2004
- 645.4.018, "Fire Pump Monitoring Test," Rev. 50, performed on July 7, 2004, after system maintenance work
- 609.4.001, "Isolation Condenser Valve Operability and In Service Test," Rev. 50, performed on September 3, 2004, after back-seating associated valves to reduce stem packing leakage
- 654.4.003, "Control Room Heating, Ventilation, and Air Conditioning (HVAC) System Operability Test - "B" System," performed on August 10, 2004, after planned maintenance
- 651.4.001, "Standby Gas Treatment System Operability Test - System 1," performed on August 25, 2004, after a planned system overhaul

b. Findings

No findings of significance were identified.

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

1FO7 Maintenance Outage to Repair Main Steam Isolation Valve (MSIV), NS04A

a. Inspection Scope

The inspectors observed outage maintenance activities for the 1FO7 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 September 11, 2004. The inspectors physically observed portions of the MSIV repairs conducted in the trunnion room of the reactor building. During the plant startup, on September 23, 2004, 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 September 24, 2004.

b. Findings

On September 20, 2004, while preparing to restore the shutdown cooling system to a normal, standby readiness condition, a momentary loss of shutdown cooling occurred. This issue was followed up by the inspector. Details of this event are described in Section 4OA3.4 of this report.

1R22 Surveillance Testing (IP 71111.22 - 6 samples)

a. Inspection Scope

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

- 645.4.001, “Fire Pump No. 1 Operability Test,” performed July 10, 2004
- 621.3.033, “Air Ejector Offgas Radiation Monitor - “A” Train Test,” performed July 21, 2004
- 634.2.002, “Main Station Battery Weekly Surveillance Test,” performed July 27, 2004
- 641.4.001, “Service Water Pump Operability and In Service Test,” performed July 29, 2004
- 642.4.001, “Reactor Building Closed Cooling Water In Service Test,” performed July 30, 2004
- 602.3.004, “Electromagnetic Relief Valve Pressure Sensor Test and Calibration,” performed August 27, 2004

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

One sample was selected for review by the inspector. The inspector reviewed the work order package for the installation of a temporary charger to be connected to both the A1 and A2 24 VDC chargers in order to maintain power to the associated batteries while the chargers are replaced. Completion of this work package is planned for the 1R20 refueling outage. A temporary modification package was not required since the activity was to be controlled by a work order. The inspector also reviewed the evaluation documented in Action Request (AR) A07080074 Evaluation 04 that documented the 50.59 review for this temporary modification.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (IP 71151 - 2 Samples)

a. Inspection Scope

The inspectors reviewed the Performance Indicator (PI) data from August 2003 through August 2004 for: (1) Reactor Coolant System Leakage Rate and (2) Reactor Coolant System Activity. 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 discussed with the responsible system engineer alternate calculation methods in the event the computer program for calculating a leak rate was unavailable, and reviewed AmerGen's procedure for responding to small step changes in RCS unidentified leak rate as observed in August and September 2004.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (IP 71152)

1. Daily Screening for Repeat Equipment Failures and Human Performance Issues for Follow-up (IP 71152)

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

2. Annual PI&R Sample Review (IP71152 - 2 Samples)

a. Inspection Scope

The inspectors reviewed two annual samples of selected issues for detailed evaluation of the resolution determined by the licensee’s corrective action program. The first sample involved pipe support system deficiencies and the second sample involved fuel oil contamination with water and sediment.

The first sampled item selected for review was documented in several CAPs, O2001-0480, O2001-0499, O2003-1213, O2004-0045, and O2004-0142. These CAPs documented a variety of pipe support problems, e.g. non-conforming and/or missing pipe supports/snubbers. The inspection included the review of the troubleshooting efforts, engineering analysis/evaluation, the root cause determination, the corrective action plan, implemented design modification and the post modification test or inspection. Also, the inspection included a walk-through inspection and observation of the accessible portions of the plant structures affected by the identified deficiencies. The design and licensing bases of the affected system was also reviewed.

The second sampled item selected for review was documented in three CAPs. The inspector reviewed AmerGen’s efforts to prevent a recurrence of fuel oil water and sediment contamination in the main fuel oil storage tank that occurred in the Fall of 2003. Specifically, the inspector reviewed CAPs O2003-1865, O2003-2225, and O2004-1241, as well as the apparent cause evaluation (ACE) of the event. Individual corrective actions proposed in the ACE were discussed with the responsible system engineers and representatives from the Chemistry department. For the few corrective actions proposed that were not ultimately implemented, justification for the decision made was provided by AmerGen personnel or in CAP documentation. Trending of water and sediment levels in the EDG fuel oil storage tank was reviewed from January 2004 to mid-September 2004. No trend in contamination levels was evident - all readings were less than detection limits. In addition, plans to drain and clean the EDG fuel oil tank in the upcoming 1R20 outage were reviewed and discussed with system engineering.

b. Findings

No findings of significance were identified.

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

The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," for failure to adequately correct a condition adverse to quality affecting the Intermediate Range Monitors (IRMs) that resulted in a plant scram from 2% power. This finding has a cross-cutting aspect of PI&R in that the engineering evaluation and identified corrective actions failed to adequately address the susceptibility of the IRMs to EMI induced spikes. (Section 4OA3.2)

The inspectors identified a non-cited violation for 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," for failure to adequately correct a condition adverse to quality affecting the MSIVs that resulted in a failure to satisfy a Technical Specification surveillance requirement to be capable of isolating the reactor containment. 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 1R15)

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:

b. Findings

1. (Closed) Licensee Event Report 05000219/2004-002-00, Change in Methodology Used by General Electric and Global Nuclear Fuels to Demonstrate Compliance with Emergency Core Cooling System Performance Criteria. This event report described a concern identified by the licensee's fuel vendor of a postulated new heat source that affected the calculation of the Peak Clad Temperature and maximum local cladding oxidation required by 10 CFR 50, Appendix K and 10 CFR 50.46(b)(2). The postulated heat source is the recombination of hydrogen and oxygen within the fuel bundles during core heatup in a Loss of Coolant Accident (LOCA) event. This issue had not been

addressed by the analysis models used by the fuel vendor for the condition when reactor power is greater than 25% and the primary containment is not inerted. Oyster Creek operators immediately implemented corrective actions to prevent postulated local fuel clad oxidation levels from exceeding 10 CFR 50 limits.

2. (Closed) Licensee Event Report 05000219/2004-003-00, Actuation of Reactor Protection System due to Spurious Hi-Hi Trip Signals on Intermediate Range Monitors Caused by Electromagnetic Interference (EMI).

Introduction

A self-revealing Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, was identified for failure to adequately correct a condition adverse to quality affecting the Intermediate Range Monitor System, resulting in a reactor scram while at 2% power.

Description

At 12:31 a.m., on May 27, 2004, the reactor scrammed during a plant shutdown from about 2% reactor power. The reactor protection system processed IRM Hi-Hi/INOP on channels 13, 14 and 18 IRMs, while operators were driving the SRM detectors into the core. During subsequent troubleshooting activities, AmerGen determined that excessive electrical noise spiking appeared on channels 13, 14 and 18. AmerGen assembled a root cause team to troubleshoot the matter. The material condition issues found during the AmerGen investigation were repaired. CAP O2004-1314 was written to document this issue and the associated corrective actions to prevent recurrence. After the completion of the 1F06 outage, and after startup, CHAR services was contracted to perform additional testing and determine the cause of SRM switch activation noise generation.

The licensee's root cause investigation report concluded that an EMI induced spike caused IRM channels 13, 14 and 18 to spike simultaneously, resulting in a scram from 2% power. IRMs 13 and 14 were found to have loose cable connections at the drawer and nicks in the outer surface of the cable. AmerGen determined that this was the entry point of the noise intrusion and the reason the channels spiked. The root cause investigation concluded that the under vessel connectors and jumpers for the SRM detectors were the entry point for noise generated locally by the drive mechanisms. This allowed a direct transmission pathway to the control room and eventually to a cross connection coupling to the IRM cables. Previous CAP actions have been directed at reducing noise generators in an effort to reduce the susceptibility to noise, but no active effort was directed at reducing the susceptibility of the IRM instruments to become an antenna/receiver for the EMI. Less than adequate maintenance work practices, which resulted in damage to cables, contributed to the cause of the event by allowing the instruments to be receptors to the EMI initiators.

The contributing causes outlined in the root cause investigation report included the material condition of the installed plant equipment. This led to the decline in Nuclear Instrumentation (NI) System performance. An opportunity to correct these SRM/IRM material deficiencies during refueling outage 1R18 had been planned, but was subsequently descoped from the outage.

Historically, Oyster Creek technical staff has been challenged with numerous spiking events resulting in half scram conditions and numerous CAPs had been written to address the issues associated with SRM/IRM spiking in an attempt to resolve the problems associated with the NI System. In CY 2003, CAP O2003-1097 documented a common cause analysis conducted by AmerGen that concluded that issues associated with IRM/SRM spiking had not been adequately addressed.

Analysis

AmerGen failed to adequately correct the SRM/IRM EMI induced spiking issues identified by the engineering staff that resulted in a reactor scram on May 27, 2004. This is a performance deficiency in that 10 CFR 50, Appendix B, Criterion XVI, specifies that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. This was reasonably within the licensee's ability to foresee and prevent. 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 more than minor because it is associated with a transient initiator contributor affecting the initiating event cornerstone. This finding also affects the attributes of equipment performance in reliability and maintenance for the initiating events cornerstone. This finding did not affect the mitigating systems or containment barrier cornerstones.

The finding was determined to be of very low safety significance (Green) using phase 1 analysis of the SDP for Reactor Inspection Findings for At-Power Situations, in that, the finding does not: contribute to a primary or secondary system LOCA initiator; contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available; or, contribute to the likelihood of a fire or internal/external flood.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, requires that measures shall be established to assure that conditions adverse to quality, such as deficiencies and non-conformances, are promptly identified and corrected. Contrary to this requirement, AmerGen failed to take proper actions to reduce SRM/IRM EMI induced noise as identified in CAPs generated by AmerGen technical staff. Because this issue is of very low safety significance and has been entered into the AmerGen corrective action program (CAP No. O2004-1314), this violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 0500219/200400402)**

3. <u>(Closed) Licensee Event Report 05000219/2004-004-00</u>, Operation Slightly in Excess of the Thermal Power Limit Due to Plant Computer Isolator Power Supply Degradation Affecting Input Values to the Heat Balance Calculation. This LER describes an event where the licensed thermal power limit of 1930 megawatt thermal (MWt) was exceeded by a maximum of 0.4%, for a period greater than eight hours, prior to recognition and taking action to restore power below the license limit.

<u>Introduction</u>. A self-revealing Green non-cited violation (NCV) of Section 2.C.(1), "Maximum Power Level" of Operating License No. DPR-16, was identified because Oyster Creek operators failed to timely recognize and respond to alarms indicating a low flow for the control rod drive system computer input to the plant computer heat balance calculation. This condition occurred when the control rod drive PC-1 power supply output voltage began to slowly degrade, causing the CRD flow input to the plant computer system (PCS) to trend lower. This resulted in Oyster Creek exceeding the licensed thermal power limit of 1930 MWt for approximately 19 hours.

<u>Description</u>. On August 29, 2004, minor recirculation flow adjustments termed "load maintenance" were performed to maintain the PCS indicated core thermal power (CTP) near the Maximum Power Level limit of 1930 MWt. The frequency and total recirculation flow changes were not abnormally high as the PC-1 power supply output voltage began to slowly degrade. This degradation in output voltage caused the control rod drive flow input to the PCS to slowly trend lower. CRD flow is a dynamic input to the PCS heat balance calculations from which the instantaneous CTP, 15-minute average, 1-hour average, and 8-hour average CTP are computed and displayed. Indicated CTP began to non-conservatively trend lower at a slight rate indistinguishable from that associated with fuel depletion and normal control board indications. At 8:03 p.m., a plant computer system alarm was generated for CRD flow Lo-1 at 55 gallons per minute (gpm). Actual CRD flow was 60 gpm.

The isolator power supply continued to slowly degrade through August 30, 2004. On August 31, 2004, at 7:17 a.m., the plant computer system CRD Flow Lo-2 alarm (30 gpm) was generated while actual CRD flow remained at 60 gpm. Operators continued making load adjustments, maintaining indicated power at 100%. On September 1, 2004, at 1:11 a.m., the PCS CRD Flow Lo-3 (10 gpm) alarm was generated while actual CRD flow remained at 60 gpm. At 9:38 a.m., the isolator power supply failed, resulting in numerous PCS point failures and alarms. An administrative rod block was inserted per procedure, and repair to the PCS was requested. At 1:50 p.m., the PCS heat balance indicated 1936 MWt after installation of a spare isolator power supply. Operators lowered core thermal power until indicated power was below the 1930 MWt limit.

Oyster Creek's investigation concluded that, although operators review the sequence of events PCS displays once each shift per procedure OP-OC-100, they failed to recognize the CRD flow alarms due to the volume and frequency of display changes. In addition, exceeding a PCS alarm setpoint does not result in direct audible or visual annunciation to alert the panel control room operator. The number of "sequence of events" display changes desensitized control room operators to the importance of the PCS indications, and as a result, the significance of the control rod drive Lo Flow alarms were not timely recognized. Calculations showed that the maximum power level limit was initially exceeded at 6:31 p.m., on August 31, 2004. The calculated peak core thermal power attained on September 1, 2004, was 1937 MWt, or approximately 100.4% power.

Analysis. The finding is a performance deficiency because plant operators did not identify that Oyster Creek operated slightly in excess of its licensed thermal power limit of 1930 MWt for approximately 19 hours. This was reasonably within the licensee's ability to foresee and correct. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and was not the result of any willful violation of NRC requirements or Exelon procedures. This finding is more than minor because if left uncorrected, it could become a more significant safety concern. The overpower condition did not reach a level that compromised the integrity of the fuel barrier since operators took action to reduce power. However, the increasing rate of thermal power, if left uncorrected, could have resulted in reactor power exceeding the radiological consequence accident analysis initial power condition of 102%. The inspectors concluded that this issue is associated with the Design Control attribute of the Barrier Integrity cornerstone, and affected the cornerstone objective to maintain functionality of Fuel Cladding thermal limits.

This finding was assessed using Phase 1 of the Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations. The finding was determined to be of very low safety significance (Green), because while the reactor power exceeded the license condition limit of 1930 MWt, the radiological consequence accident analysis initial condition of 102% was not exceeded.

Enforcement. Oyster Creek Nuclear Generating Station facility Operating License, DPR-16, section 2.C.(1), limits the reactor core thermal power to 1930 MWt. Contrary to the above, thermal power exceeded 1930 MWt from August 31, 2004 through September 1, 2004. During this period, the reactor core thermal power exceeded the limit by 0.1 - 0.4%. Because this issue is of very low safety significance and has been entered into the corrective action program (CAP No. O2004-2384), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000219/200400403)**

4. Inadequate Procedure Results in a Loss of Shutdown Cooling Capability

Introduction. A self-revealing event involving an inadvertent loss of shutdown cooling resulted in a Green finding and non-cited violation 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.

Description. On September 20, 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. 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 ‘B’ temperature circuitry failed, causing an isolation system logic trip. Due to the failed high temperature input, the shutdown cooling system tripped off while the operators performed the steps per the attachment 305-8 to remove the jumper as part of the system restoration process.

Prior to commencing the activity to remove the bypass jumpers, operators did not verify that associated temperature instrumentation was operable. The operators did verify that the indicated reactor coolant system (RCS) temperature had adequate margin to boil and was well below the logic actuation temperature of 350 degrees F. During the pre-evolution brief, operators discussed the possibility that removing the jumpers could introduce an isolation trip of shutdown cooling, but that 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, and then restored shutdown cooling to service using the bypass jumpers. The system was restored in about 15 minutes and RCS temperature increased about 4 degrees to 151 degrees F.

The loss of shutdown cooling resulted in an unplanned entry into a high outage risk condition. Outage risk had been Yellow due to the high decay heat condition of the RCS. On September 20, 2004, the calculated time to boil was about 95 minutes. The loss of the shutdown cooling system resulted in a Red shutdown risk condition as indicated by the ORAM-Sentinel Model.

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 verify that the isolation logic was not in a tripped condition prior to removing the bypass jumpers. This resulted in an unexpected trip of the shutdown cooling system on September 20, 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. Therefore, this deficiency affected the availability of the decay heat removal function during shutdown operational conditions.

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 was less than 0.2 times the final margin to boil; (2) per Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both (Pressurized Water Reactors (PWRs) and Boiling Water Reactors (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 did not increase the likelihood that a loss of decay heat removal would occur due to failure of the system itself or support systems; did not include decay heat removal instrumentation or vessel level instrumentation such that degraded core cooling could not be detected; did not increase the likelihood of a loss of RCS inventory, or that could result in a loss of RCS level instrumentation; did not involve a design or qualification deficiency; and, 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 O2004-2657.

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. 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 maintained, in that it did not include the required actions to prevent an inadvertent isolation of the Shutdown Cooling System while restoring the isolation interlocks to a normal configuration. This led to a loss of shutdown cooling on September 20, 2004. Because this condition is of very low significance and has been entered into AmerGen's corrective action program (CAP O2004-2657), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, issued May 1, 2000 (65FR25368).
(NCV 05000219/200400404)

4OA4 Cross-Cutting Aspects of Findings Other Than PI&R

A self-revealing Green non-cited violation of Section 2.C.(1), "Maximum Power Level" of Operating License No. DPR-16, was identified because Oyster Creek operators failed to timely recognize and respond to alarms indicating a low flow for the control rod drive system computer input to the plant computer heat balance calculation. This finding has a cross-cutting aspect of human performance in that the operators failed to timely identify a PCS alarming condition that would have alerted them to condition affecting the heat balance calculation and thus reactor thermal power. (Section 4OA3.3)

4OA6 Meetings, including Exit

Exit Meeting Summary

On October 14, 2004, the resident inspectors presented the inspection results to Mr. C. N. Swenson and other members of licensee management. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Oyster Creek Personnel
J. Karkoska, Mid-Atlantic EP Manager
K. Poletti, EP Manager
J. Cohen, Lead Auditor
P. Bloss, BOP Systems Manager
M. Godknecht, Maintenance Rule Coordinator
J. Hackenberg, Operations Training Manager
E. Harkness, Vice President, Projects
S. Hutchins, Electrical Systems Manager
J. Magee, Director, Engineering
M. Massaro, Plant Manager
D. McMillan, Director, Training
C. Connelly, Manager, Chemistry & Rad Protection
J. O'Rourke, Assistant Engineering Director
J. Kandasamy, Manager, Regulatory Assurance
B. Stewart, Senior Licensing Engineer
C. Swenson, Site Vice President
R. Detwiler, Director, Operations
G. Waldrep, Manager, Nuclear Oversight
J. Renda, Radiation Protection Manager
D. Fawcett, Licensing Engineer

New Jersey State Department of Environmental Protections
R. Russell, Nuclear Engineer, Bureau of Nuclear Engineering (BNE)
D. Zannoni, Supervisor, Nuclear Engineering, BNE

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000219/200400401	NCV	A self-revealing Green NCV was identified for failure to adequately correct a condition adverse to quality affecting a Main Steam Isolation Valve. (Section 1R15)
05000219/200400402	NCV	A self-revealing Green NCV was identified for failure to correct a condition adverse to quality affecting IRMs, causing a reactor scram. (Section 4OA3.2)

05000219/200400403	NCV	A self-revealing Green NCV was identified for failure to maintain the core thermal power below the licensed limit. (Section 4OA3.3)
05000219/200400404	NCV	A self-revealing Green NCV was identified for an inadequate procedure that resulted in a loss of shutdown cooling. (Section 4OA3.4)

Closed

05000219/2004002-00	LER	Change in Methodology Used by General Electric and Global Nuclear Fuels to Demonstrate Compliance with Emergency Core Cooling System Performance Criteria. (Section 4OA3)
05000219/2004003-00	LER	Actuation of Reactor Protection System due to Spurious Hi-Hi Trip Signals on Intermediate Range Monitors Caused by Electromagnetic Interference. (Section 4OA3)
05000219/2004004-00	LER	Operation in Excess of the Thermal Power Limit Due to Plant Computer Isolator Power Supply Degradation Affecting Input Values to the Heat Balance Calculation. (Section 4OA3)

LIST OF DOCUMENTS REVIEWED
(not previously referenced)

CAP Nos. O2004-1120, O2004-1099, O2004-1817, O2004-1057, O2004-1042, O2004-1168, O2004-2296, O2004-2357, O2004-2399, O2001-0480, O2001-0499, O2003-1213, O2004-0045, O2004-0142, O2002-0627, O2002-0635, O2002-0694, O2002-1663, O2003-2017, O2004-1355, O2004-0449, O2003-2225, O2003-2660, O2003-1865, O2003-2454, O2003-2454, O2003-1930, and O2003-1723
A2057287e01, “Molded case circuit breaker with undervoltage fitted device - electrical loading evaluation”
Apparent Cause Evaluation (LS-AA-125-1003), “Excessive Water & Sediment in EDG Fuel Oil Storage Tank”
Apparent Cause Evaluation (CAP 2004-2111), “Control Rod Drive 38-27 Apparent Flange Leak”
Maintenance Rule (a)(1) evaluation 2002-001, Rev. 1, on the 480 VAC Distribution System
RCS Leakage PI Data and Verification Record, August 2003 - August 2004
Station Procedure 312.9, “Primary Containment Control,” Rev. 31
System Health Report for the 480 VAC Distribution System, June 2004

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
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CTP	Core Thermal Power
DER	Deviation/Evaluation Report
EC	Elective Maintenance
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
ESW	Emergency Service Water
FA	Fire Area
FAQ	Frequently Asked Question
FZ	Fire Zone
gpm	gallons per minute
HVAC	Heating, Ventilation and Air Conditioning
IFI	Inspector Follow-up Item
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Plant Owners
IRM	Intermediate Range Monitor
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LORT	Licensed Operator Requalification Training
MWt	megawatt thermal
MSCRWL	Minimum Steam Cooling RPV Water Level
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ORAM	On-line Risk Assessment Monitor
PI	Performance Indicator
PI&R	Problem Identification & Resolution
PM	Preventative Maintenance
PMT	Post Maintenance Test
PSIG	pounds per square Inch gauge
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water
RCA	Radiologically Controlled Area
RO	Reactor Operator

RP	Radiation Protection
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
RWP	Radiation Work Permit
SDP	Significance Determination Process
SGTS	Standby Gas Treatment System
SIL	Service Information Letter
SRO	Senior Reactor Operator
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VAC	Volts Alternating Current
VDC	Volts Direct Current
WO	Work Order