October 29, 2007

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

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000219/2007004

Dear Mr. Crane:

On September 30, 2007, 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 26, 2007, with Mr. J. Randich, Plant Manager, 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.

The report documents two NRC-identified findings and one self revealing finding each 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 three findings as non-cited violations (NCVs) consistent with Section VI.A of the NRC's Enforcement Policy. If you contest these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Oyster Creek.

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

We appreciate your cooperation. Please contact me at (610) 337-5200 if you have any questions regarding this letter.

Sincerely,

**/RA/**

Ronald R. Bellamy, Ph.D., Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000219/2007004
w/ Attachments A and B

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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
Vice President - Licensing and Regulatory Affairs, AmerGen
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**U. S. NUCLEAR REGULATORY COMMISSION**

REGION I

| | |
|---|---|
| Docket No.: | 50-219 |
| License No.: | DPR-16 |
| Report No.: | 05000219/2007004 |
| Licensee: | AmerGen Energy Company, LLC (AmerGen) |
| Facility: | Oyster Creek Generating Station |
| Location: | Forked River, New Jersey |
| Dates: | July 1, 2007 - September 30, 2007 |
| Inspectors: | M. Ferdas, Senior Resident Inspector<br>R. Treadway, Resident Inspector<br>J. Kulp, Resident Inspector (Acting)<br>R. Li, Reactor Engineer, NRR (In-Training)<br>P. Presby, Operations Engineer<br>R. Nimitz, Senior Health Physicist<br>P. Kaufman, Senior Reactor Inspector<br>J. Schoppy, Senior Reactor Inspector<br>M. Patel, Reactor Inspector |
| Approved By: | Ronald R. Bellamy, Ph.D., Chief<br>Projects Branch 6<br>Division of Reactor Projects |

CONTENTS

## SUMMARY OF FINDINGS

IR 05000219/2007004; 07/01/07 - 09/30/2007; AmerGen Energy Company, LLC, Oyster Creek Generating Station; Operability Evaluations, Post-Maintenance Testing, and Event Followup.

The report covered a 3-month period of inspection by resident inspectors and regional inspectors. 3 Green non-cited violations (NCV) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

**Cornerstone: Mitigating Systems**

Green. The inspectors identified that AmerGen did not properly identify that the remote shutdown panel (RSP) was not capable of performing its design function when the 'B' isolation condenser (IC) makeup valve control power indicating status light was not illuminated on June 21, 2007. This finding was of very low safety significance (Green) and determined to be a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." AmerGen's corrective actions included repairing the RSP and discussing this issue with operations personnel on the adequacy of operability evaluations.

The finding was more than minor because it was associated with the protection against external factors (fire and toxic hazard) attribute of the mitigating systems cornerstone and affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process (SDP)," the inspectors conducted a Phase I SDP screening utilizing Figure F.1 in Appendix F. Per the Phase I screening criteria, the finding was assigned the category of "post-fire safe shutdown." The inspectors assigned a low degradation rating in accordance with Attachment 2 of Appendix F. A low degradation rating was assigned, because procedures existed and operators were trained at operating the 'B' IC makeup valve locally; and operators have a significant amount of time to complete the local operation. Therefore, in accordance with Appendix F step 1.3.1, "Qualitative Screening for All Finding Categories," this finding screened as very low safety significance because the finding was assigned a low degradation rating. The performance deficiency had a cross-cutting aspect in the area of problem identification and resolution because AmerGen did not thoroughly evaluate a problem for operability [P.1.(c)]. (Section 1R15)

Green. The inspectors identified that AmerGen did not properly implement fire protection plan requirements on June 22, 2007. Specifically, AmerGen did not repair fire penetration 762 in accordance with procedures and resulted in an unqualified configuration of sealing materials being installed in the plant. This finding was determined to be a NCV of license condition 2.C(3), "Fire Protection." AmerGen's corrective actions involved evaluating the as-found penetration seal for effectiveness in preventing the spread of a fire and procuring a fire seal qualification test that qualified the installed configuration; and evaluating the process and programs used to repair fire penetration seals in the plant.

The finding was more than minor because it was associated with the external factors (fires) attribute of the mitigating systems cornerstone and affected the objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with IMC 0609, Apendix F, "Fire Protection Significance Determination Process," the inspectors conducted a Phase I SDP screening utilizing Figure F.1. Per the Phase I screening criteria the finding was assigned the category of "Fire Confinement." The inspectors assigned a "Moderate B" degradation rating to the installed fire penetration seal in accordance with Attachment 2, Table A2.2 of Appendix F, because the installed seal configuration was between 6 and 9 inches and there was no test or evaluation available to qualify its fire rating. Therefore in accordance with Appendix F, step 1.3.2, "Supplemental Screening for Fire confinement Findings," screening criteria 3, this finding screened as very low safety significance because both sides of the wall were protected by a non-degraded automatic water based fire suppression system. The performance deficiency had a cross-cutting aspect in the area of human performance because AmerGen did not assure that accurate work packages were available to ensure that a qualified fire penetration seal was installed in the plant [H.2(c)]. (Section 1R19)

Green. A self-revealing finding was identified when AmerGen personnel did not properly implement procedural guidance during a response to a reactor feedwater pump (RFP) trip and a reactor scram on July 17, 2007. Specifically, the operating crew did not properly reduce reactor power as directed by an abnormal operating procedure; and did not properly implement EOP support procedures which challenged reactor water level control during recovery activities. This finding was determined to be a NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." AmerGen's corrective actions included revising the abnormal procedure to provide enhanced instructions, providing all operations personnel remedial training sessions in the simulator on this event, and issuing a standing order communicating operation's management expectations on operator response.

The finding was more than minor because it was associated with the human performance attribute of the mitigating systems cornerstone and affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was assessed in accordance with IMC 0609, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors performed a Phase 1 screening and determined that a Phase 2 evaluation was required to assess safety significance because the failure to properly implement procedure guidance in response to and during the event affected both the initiating and mitigating cornerstones. A Region 1 senior reactor analyst (SRA) determined that a Phase 2 evaluation was not suited to assess this event. A Phase 3 analysis was performed by the SRA and the finding was determined to be of very low safety significance. The performance deficiency had a cross-cutting aspect in the area of human performance because the operating crew did not follow procedures during their response to the event [H.4(b)]. (Section 4OA3)

B. Licensee-Identified Violations

None.

# REPORT DETAILS

## Summary of Plant Status

The Oyster Creek Generating Station (Oyster Creek) began the inspection period operating at full power.

On several occasions during the summer months (July 10 and 11, August 3 and 4) operators performed unplanned power reductions in accordance with operating procedures for several hours to maintain the plant's circulating water discharge temperature below Oyster Creek's environmental discharge permit requirements.

On July 17, 2007, Oyster Creek experienced an automatic reactor scram due to a low reactor water level following a trip of the 'C' reactor feedwater pump (RFP). The RFP tripped due to an electrical ground fault in the pump's motor. AmerGen reported this event to the NRC in Event Notification 43495, "Automatic Reactor Scram on Low Reactor Vessel Water Level (RVWL)." Additional information on this event is contained in section 4OA3 of this report. AmerGen personnel performed maintenance activities during the outage, which included repairs to the 'B' high pressure feedwater heater and intermediate range nuclear instrumentation. Operators commenced a reactor startup on July 21, 2007, and established the reactor critical and synchronized the main generator to the grid on July 22, 2007. AmerGen continued with repairs to the 'C' RFP motor and plant power was limited to 70% until a refurbished motor was received onsite and installed. The plant reached full power on August 1, 2007.

On August 8, 2007, operators performed an unplanned power reduction to 70% to investigate a steam leak in the condenser bay room. AmerGen identified a steam leak on a non-safety related steam trap (S-1-11) on a drain line from main steam piping to the turbine. On August 9, 2007, AmerGen attempted to repair the steam trap, but was unsuccessful. On August 10, 2007, AmerGen personnel obtained measurements for repairing the steam leak and returned the plant to full power. On August 17, 2007, operators commenced a planned power reduction to approximately 25% power and removed the main generator from the grid to repair the steam leak on steam trap S-1-11. AmerGen performed a temporary repair (leak repair box and injection) to stop the steam leak. Operators subsequently synchronized the generator to the grid and the plant reached full power on August 18, 2007.

On September 4, 2007, operators performed an unplanned downpower to 80% in accordance with normal operating procedures after identifying a 10 gallons per minute (gpm) through wall leak in the inlet header for the south waterbox of the 'A' condenser. Once power was stable at 80%, operators isolated the south waterbox of the 'A' condenser and performed a temporary repair on the leak. The plant returned to full power on September 5, 2007.

On September 29, 2007, operators performed a planned power reduction to 60% for a rod pattern adjustment. Additionally, during the downpower evolution, AmerGen performed corrective maintenance on the 'A' RFP drain line, repaired two leaking hydraulic control units (HCU's) and repaired several minor steam leaks in the condenser bay. The plant returned to full power on later that same day on September 29, 2007.

Oyster Creek operated at full power for the remainder of the inspection period.

# 1. REACTOR SAFETY

## Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope (4 samples)

The inspectors performed three partial and one complete equipment alignment inspections. The partial equipment alignment inspections were completed during conditions when the equipment was of increased safety significance such as would occur when redundant equipment was unavailable during maintenance or adverse conditions; or after equipment was recently returned to service after maintenance. The inspectors performed a partial walkdown of the following systems, and when applicable, the associated electrical distribution components and control room panels, to verify the equipment was aligned to perform its intended safety functions:

- Containment spray system #2 on July 9, 2007;
- Core spray system #2 on September 4, 2007; and
- Station blackout (SBO) transformer and SBO control panel on September 13, 2007.

The inspectors performed a complete system alignment inspection of the control room heating, ventilation and air conditioning (CRHVAC) system to determine whether the system was aligned and capable of maintaining the required pressure in the control room in accordance with design basis requirements. The inspectors reviewed operating procedures, surveillance test procedures, piping and instrumentation drawings, and the applicable equipment lineup list. The inspectors also reviewed corrective action program condition reports documenting CRHVAC system deficiencies to verify identified problems were being evaluated and corrected.

Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope (8 samples)

The inspectors performed a walkdown of eight plant areas to assess their vulnerability to fire. During plant walkdowns, the inspectors observed combustible material control, fire detection and suppression equipment availability, visible fire barrier configuration, and the adequacy of compensatory measures (when applicable). The inspectors reviewed Oyster Creek's Fire Hazards Analysis Report and Individual Plant Examination for External Events (IPEEE) for risk insights and design features credited in these areas. Additionally, the inspectors reviewed corrective action program condition reports documenting fire protection deficiencies to verify that identified problems were being evaluated and corrected. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report. The following plant areas were inspected:

- Reactor building 75' elevation on July 11, 2007;
- Reactor building 95' elevation on July 11, 2007;
- RFP room on July 16, 2007;
- Turbine building operating floor on July 18, 2007;
- 'B' 480V room on July 30, 2007;
- 'A' emergency diesel generator (EDG) building on August 30, 2007;
- 'B' EDG building on August 30, 2007; and
- 'D' 4160V room on September 6, 2007.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope (2 samples)

The inspectors performed a biennial review of heat sink performance at Oyster Creek. Based on a plant specific risk assessment and previous NRC inspections, the inspectors selected the following two heat exchanger (HX) samples to review: 'B' containment spray HX (H-21-1B); and the '1-2' reactor building closed cooling water (RBCCW) HX. The containment spray HXs are cooled by the emergency service water (ESW) system. The service water (SW) system directly supplies cooling water to the RBCCW HXs. The ESW and SW pumps are located at the SW intake and draw from the Barnegat Bay intake canal (the ultimate heat sink).

The inspectors reviewed the methods (inspection, cleaning, maintenance, and performance testing) utilized by AmerGen to ensure the heat removal capabilities on the 'B' containment spray HX and '1-2' RBCCW HX. The inspectors compared the methods utilized by AmerGen to the commitments made in their response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspectors reviewed the performance test methodology and results to verify that the HX could function as designed. The inspectors independently reviewed heat transfer calculations for the HX to verify that the condition and operation were consistent with design assumptions.

The inspectors reviewed the chlorination system to ensure that chemical treatments were controlled, tested, and evaluated for adverse effects on containment spray and RBCCW HX performance. The inspectors also reviewed inspection results, trending data, and action plans associated with the SW system and intake structure.

The inspectors compared surveillance test and inspection data to the established acceptance criteria to verify that the results were acceptable and that operation was consistent with design. The inspectors walked down the HXs, intake structure, chlorination system, and accessible portions of the ESW system to assess the material condition and configuration control of these systems and components.

The inspectors also reviewed a sample of corrective action program condition reports related to HXs, the chlorination system, SW system, and the intake structure to ensure that AmerGen appropriately identified, characterized, and corrected problems related to these essential systems and components.

Documents associated with these reviews are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope (1 sample)

The inspectors observed one simulator training scenario on August 29, 2007, to assess operator performance and training effectiveness. The scenario involved an electrical pressure regulator (EPR) failure, offsite fire, and a SBO with a stuck open electromatic relief valve (EMRV). The inspectors assessed whether operator performance met AmerGen's procedural requirements, and the simulator instructor's critique identified crew performance problems. The inspectors also assessed if the simulator adequately reflected expected plant response during the scenario. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope (2 samples)

The inspectors performed two maintenance effectiveness inspection activities. The inspectors reviewed the following degraded equipment issues to assess the effectiveness of maintenance:

- 'C' RFP motor failure on July 17, 2007 (IR 650654); and
- Bank 6 startup transformer 'B' phase voltage regulator failed upscale on July 17, 2007 (IR 650702).

The inspectors verified that the systems or components were monitored in accordance with AmerGen's maintenance rule program requirements. The inspectors compared documented maintenance preventable functional failure (MPFF) determinations and unavailable hours to those being tracked by AmerGen to evaluate the effectiveness of AmerGen's monitoring activities and determine whether performance goals were being met. The inspectors reviewed completed maintenance work orders and procedures to determine if inadequate maintenance contributed to equipment performance issues. The inspectors reviewed applicable work orders, corrective action program condition reports, preventive maintenance tasks, vendor manuals, and system health reports. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified. An unresolved item (URI) was identified to review AmerGen's corrective action program evaluation (IR 650654) regarding the 'C' RFP motor failure on July 17, 2007. The inspectors plan to review this evaluation after it is completed, which had not occurred by the end of this inspection period. **(URI 05000219/2007004-01, 'C' Reactor Feedpump Motor Failure)**

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope (5 samples)

The inspectors reviewed five on-line risk management evaluations through direct observation and document reviews for the following plant configurations:

- Bank 6 startup transformer unavailable due to emergent maintenance on July 10, 2007;
- 'A' EMRV unavailable due to planned maintenance; and the #1 air compressor and the #2 combustion turbine (CT) unavailable due to unplanned maintenance on August 2, 2007;
- '1-1' SW pump unavailable due to a failed motor, and containment spray system #2 unavailable due to planned maintenance on August 16, 2007;

- Bank 5 startup transformer, ‘1-1' condensate transfer pump, standby gas treatment (SBGT) system #1, and ‘1-1' diesel driven fire pump unavailable due to planned maintenance on August 23, 2007; and
- SBO transformer and ‘1-2' diesel driven fire pump unavailable due to planned maintenance on September 12, 2007.

The inspectors reviewed the applicable risk evaluations, work schedules, and control room logs for these configurations to verify the risk was assessed correctly and reassessed for emergent conditions in accordance with AmerGen’s procedures. AmerGen’s actions to manage risk from maintenance and testing were reviewed during shift turnover meetings, control room tours, and plant walkdowns. The inspectors also used AmerGen’s on-line risk monitor (Paragon) to gain insights into the risk associated with these plant configurations. Additionally, the inspectors reviewed corrective action program condition reports documenting problems associated with risk assessments and emergent work evaluations. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope (5 samples)

The inspectors reviewed five operability evaluations for degraded or non-conforming conditions associated with:

- ‘A/B’ ESW electrical conduit supports corroded on June 20, 2007 (IR 642467);
- Remote shutdown panel (RSP) control power indicating light not illuminated on June 21, 2007 (IR 643046);
- Discrepancy in feedwater flow measurements on July 11, 2007 (IR 649140);
- ‘A’ and ‘E’ EMRV incorrect resistor sizing on July 27, 2007 (IR 653354); and
- 4160V electrical power cables in the turbine building component cooling water (TBCCW) sump submerged in water on August 29, 2007 (IR 645011).

The inspectors reviewed the technical adequacy of the operability evaluations to ensure the conclusions were technically justified. The inspectors also walked down accessible portions of equipment to corroborate the adequacy of AmerGen’s operability evaluations. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

Introduction. The inspectors identified that AmerGen did not properly identify that RSP was not capable of performing its design function when the ‘B’ isolation condenser (IC) makeup valve control power indicating status light was not illuminated on June 21, 2007. This finding was of very low safety significance (Green) and determined to be a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action.”

Description. On June 21, 2007, operations personnel identified that the normally “on” indicating light for control power to the ‘B’ IC makeup valve (V-11-34) on the RSP was not illuminated. Operations personnel replaced the light bulb for the indicator, however that did not correct the problem. Operations personnel wrote corrective action program condition report IR 643046 to document the issue and requested repairs for a possible light socket issue. Operations personnel determined the RSP was operable because the issue was an indication problem and that repairs were expected to be made by July 30, 2007.

On June 28, 2007, the inspectors reviewed electrical scheme drawing BR E1108, “Elementary Diagram Remote Shutdown Panel Transfer Scheme” and noted that a potential cause for the indicating light not working could be due to a failed fuse in the circuit. A failure of a fuse in the circuit would prevent operation of ‘B’ IC makeup valve from the RSP. The inspectors informed operations personnel that they believed the RSP could not perform its design function because a fuse could have potentially failed; and that a potential condition adverse to quality existed. Specifically, the ‘B’ IC makeup valve could not be operated from the RSP if operators were required to evacuate the control room due to habitability issues such as from a fire or toxic gas release.

Operations personnel evaluated the inspectors concerns and requested maintenance to performed troubleshooting on the RSP in accordance with work order A2170430 on June 28, 2007. Maintenance personnel identified that a fuse (FU-10) had failed which resulted in the indicating light not being illuminated and the fuse failure would have prevented operation of the valve from the RSP. The inspectors noted that AmerGen maintained the ability to operate the ‘B’ IC makeup valve by manual operator action in accordance with operating procedure 307, “Isolation Condenser System.” In addition, the inspectors noted that AmerGen’s initial bases that the capability of the RSP was not impacted was not technically justified.

The performance deficiency associated with this finding involved not properly identifying that the RSP was not capable of performing its design function when the ‘B’ IC makeup valve control power indicating status light was not illuminated. AmerGen’s corrective actions included repairing the RSP and discussing this issue with operations personnel on the adequacy of operability evaluations.

Analysis. The finding was more than minor because it was associated with the protection against external factors (fire and toxic hazard) attribute of the mitigating systems cornerstone and affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process (SDP)," the inspectors conducted a Phase I SDP screening utilizing Figure F.1 in Appendix F. Per the Phase I screening criteria, the finding was assigned the category of "post-fire safe shutdown." The inspectors assigned a low degradation rating in accordance with Attachment 2 of Appendix F. A low degradation rating was assigned, because procedures existed and operators were trained at operating the 'B' IC makeup valve locally; and operators have a significant amount of time to complete the local operation. Therefore, in accordance with Appendix F step 1.3.1, "Qualitative Screening for All Finding Categories," this finding screened as very low safety significance (Green) because the finding was assigned a low degradation rating.

The performance deficiency had a cross-cutting aspect in the area of problem identification and resolution because AmerGen did not thoroughly evaluate a problem for operability and resulted in not identifying a degraded condition on the RSP [P.1.(c)].

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, from June 21 thru June 28, 2007, AmerGen did not promptly identify and correct a condition adverse to quality associated with the RSP. Specifically, operators did not recognize that they were not able to operate the 'B' IC makeup valve from the RSP due a fuse failure in the RSP. However, because the finding was of very low safety significance (Green) and has been entered in the corrective active program in condition report IR 643046 and IR 654058 this violation is being treated as a NCV, consistent with section IV.A of the NRC Enforcement Policy. **(NCV 05000219/2007004-02, Degraded Condition on the Remote Shutdown Panel Not Properly Identified)**

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope (2 samples)

The inspectors reviewed two permanent plant modifications which were installed when Oyster Creek was on-line in 2007. The inspectors verified that the design bases, licensing bases, and performance capability of risk significant structures, systems, and components (SSC) had not been degraded by the installed modifications. The inspectors reviewed the following permanent plant modifications:

- Replacement of mecatiss fire wrap on the #2 EDG cabling and conduit with 3M interam fire wrap (ECR OC-07-00428)
- Installation of refrigerated air dryers in the service/instrument air system (ECR OC-05-00672)

The inspectors performed a walk down of accessible components associated with the modifications to assess the adequacy of the modification. The inspectors reviewed the design assumptions to verify they were technically appropriate and consistent with the

Updated Final Safety Analysis Report (UFSAR). The inspectors reviewed the modification/design change documents of each of the permanent modifications. The inspectors verified that procedures, calculations, and other documents were properly updated with revised design information and operating guidance associated with the modification. The inspectors verified that post-modification testing was adequate to ensure the SSC would function in accordance with design assumptions. Documents reviewed are listed in the Supplemental Information attachment to this report.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope (7 samples)

The inspectors observed portions of and/or reviewed the results of seven post-maintenance tests for the following equipment:

- Lower cable spreading room fire penetration 762 on June 22, 2007 (WO A2170265);
- ‘1-2' traveling intake screen on July 5, 2007 (WO A2169992);
- ‘B’ EMRV on July 20, 2007 (WO C2015459);
- ‘E’ EMRV on July 31, 2007 (WO A2170799);
- ‘A’ IC steam isolation valve (V-14-30) on August 21, 2007 (WO A2154491);
- ‘1-1' SW pump and motor replacement on August 17, 2007 (WO A2067048); and
- ‘B’ control rod drive (CRD) pump (WO C2014549) on September 27, 2007.

The inspectors verified that the post-maintenance tests conducted were adequate for the scope of the maintenance performed and that they ensured component functional capability. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

Introduction. The inspectors identified that AmerGen did not properly implement fire protection plan requirements on June 22, 2007. Specifically, AmerGen did not repair fire penetration 762 in accordance with procedures and resulted in an unqualified configuration of sealing materials being installed in the plant. This finding was of very low safety significance (Green) and determined to be a NCV of license condition 2.C(3), “Fire Protection.”

Description. On June 22, 2007, AmerGen personnel completed repairs on fire penetration seal 762 which was found degraded during routine fire penetration seal inspections (IR 644539). On July 6, 2007, the inspectors performed a review of AmerGen’s repair and identified a small gap in the silicon sealant portion of the fire penetration which was allowing a small amount of air leakage through the wall between the lower cable spreading room into the lube oil bay. This issue was documented by AmerGen in corrective action program condition report IR 647756. AmerGen evaluated this issue and determined that the fire penetration seal remained operable with this condition.

AmerGen replaced the original grouted penetration seal with one consisting of kaowool (mineral fiber) and silicon caulk in accordance with work order A2170265 and procedure 2400-SMM-2900.55, “Fire Barrier Penetration Seal Installation and Repair.” The inspectors noted that the procedure specified that installation of fire penetration seals consisting of kaowool and silicon caulk should consist of 9 inches of kaowool sealed with 1 inch of silicon caulk. The inspectors determined maintenance personnel installed approximately 7 inches of kaowool (in an eight inch thick wall), sealed on one side of the penetration with a continuous layer of caulking of approximately one-half inch thickness and on the other with an incomplete layer of caulking of varying thickness. AmerGen based their repair procedure on a test of a fire penetration seal consisting of 8 inches of Kaowool with one inch of silicon caulk sealant which produced a qualified fire rating of 3 hours. Additionally, the qualified configuration specified that the results of the test were valid for a penetration with up to a 0.5 inch gap between the object and the opening in the wall. The inspectors determined that the actual gap was 0.75 inch (2 inch conduit in a 3.5 inch hole) based on review of AmerGen procedure 645.6.017 “Fire Barrier Penetration Surveillance” (page E4–236). The issue was documented by AmerGen in corrective action program condition report IR 653351. AmerGen investigated the issue and determined that the configuration of the tested fire penetration did not bound that of the installed fire penetration and therefore the test results could not be used to qualify the fire rating of the installed fire penetration seal. The inspectors noted that no other test results that bounded the installed configuration were immediately available to AmerGen in order to qualify the fire rating of the penetration seal.

The performance deficiency associated with this finding involved the installation of a non-qualified fire seal penetration in the plant. AmerGen did not properly repair fire penetration seals in accordance with procedures in order to maintain the fire rating required by the fire protection plan. AmerGen’s corrective actions involved evaluating the as-found penetration seal for effectiveness in preventing the spread of a fire and procuring a fire seal qualification test that qualified the installed configuration; and evaluating the process and programs used to repair fire penetration seals in the plant.

Analysis. The finding was more than minor because it was associated with the protection against external factors (fires) attribute of the mitigating systems cornerstone and affected the objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with IMC 0609, Apendix F, “Fire Protection Significance Determination Process,” the inspectors conducted a Phase I SDP screening utilizing Figure F.1. Per the Phase I screening criteria the finding was assigned the category of “Fire Confinement.” The inspectors assigned a “Moderate B” degradation rating to the installed fire penetration seal in accordance with Attachment 2, Table A2.2 of Appendix F, because the installed seal configuration was between 6 and 9 inches and there was no test or evaluation available to qualify its fire rating. Therefore, in accordance with Appendix F, step 1.3.2, “Supplemental Screening for Fire confinement Findings,” screening criteria 3, this finding screened as very low safety significance (Green) because both sides of the wall were protected by a non-degraded automatic water based fire suppression system.

The performance deficiency had a cross-cutting aspect in the area of human performance because AmerGen did not assure that accurate work packages were available to ensure that a qualified fire penetration seal was installed in the plant [H.2(c)].

Enforcement. License Condition 2.C(3), "Fire Protection," requires that Oyster Creek implement and maintain in effect all the provisions of the approved fire protection plan. The purpose of procedure 2400-GMM-2900.55, "Fire Barrier Penetration Seal Installation and Repair," is ". . . to provide instructions for installation and repair of fire rated barrier penetration seals, internal conduit smoke and water seals, and penetration seals to meet the technical requirements of Procedure 101.2, Fire Protection Program, Attachment 1012-3, Section 5.A and 5.B." Contrary to the above, AmerGen did not implement their fire protection program as required by ensuring that fire penetration seal 762 was repaired in accordance with the repair procedure to meet the desired fire rating qualification on June 22, 2007. However, because the finding was of very low safety significance (Green) and has been entered into their corrective action program (IR 646890 and 653351), this violation is being treated as an NCV, consistent with section IV.A fo the NRC Enforcement Policy. **(NCV 05000219/2007004-03, Improper Repair of a Fire Rated Penetration Seal)**

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope (1 sample)

The inspectors monitored AmerGen's activities associated with outage activities described below. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

On July 17, 2007, operators completed a plant shutdown to support an unplanned maintenance outage after a failure of the 'C' RFP motor and an automatic reactor scram. The inspectors observed portions of the shutdown from the control room, and reviewed plant logs to determine that technical specification requirements were met for placing the reactor in "hot shutdown" and "cold shutdown." The inspectors also monitored AmerGen's controls over outage activities to determine whether they were in accordance with procedures and applicable technical specification requirements. The inspectors verified that cooldown rates during the plant shutdown were within technical specification requirements.

The inspectors performed a walkdown of portions of the drywell (primary containment) on July 21, 2007, to verify there was no evidence of leakage or visual damage to passive systems contained in these areas. During the walkdown of the drywell the inspectors observed that the drywell trenches discussed in PNO-1-06-012, "Preliminary Notification of Event of Unusual Occurrence," dated November 9, 2006 (Agency-Wide Documents Access and Management System (ADAMS) Accession Number:

ML063130424), did not contain water. The inspectors noted that the observed conditions were similar to those contained in pictures of the trenches taken on April 29, 2007 (ADAMS Accession Number: ML071240314).

The inspectors verified that AmerGen assessed and managed the outage risk. The inspectors confirmed on a sampling basis that tagged equipment was properly controlled and equipment configured to safely support maintenance work. During control room tours, the inspectors verified that operators maintained reactor vessel level and temperature within the procedurally required ranges for the operating condition. The inspectors also verified that the decay heat removal function was maintained through monitoring shutdown cooling (SDC) parameters and performing a walkdown of the system on July 17, 2007. The inspectors observed Oyster Creek's plant onsite review committee (PORC) startup review meetings.

The inspectors performed an inspection and walkdown of portions of the drywell prior to containment closure on July 21, 2007, to verify there was no evidence of leakage or visual damage to various systems and components, and verify that debris was not left which could affect drywell suppression pool performance during postulated accident conditions. The inspectors monitored restart activities to ensure that required equipment was available for operational condition changes, including verifying technical specification requirements, license conditions, and procedural requirements. Portions of the startup activities were observed from the control room to assess operator performance. The inspectors further verified that unidentified leakage and identified leakage rate values were within expected values and within technical specification requirements.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (6 samples)

The inspectors observed portions of and/or reviewed the results of six surveillance tests:

- Containment electrical penetration nitrogen blanket functional test on July 1, 2007;
- EMRV operability/functional test on July 22, 2007;
- Containment spray/ESW system #2 operability and in-service test (IST) on August 1, 2007;
- Scram discharge instrument volume vent and drain exercise and IST on August 16, 2007;
- Core spray system #2 pump operability and IST on August 29, 2007; and
- 'B' IC valve operability and IST on September 25, 2007.

The inspectors verified that test data was complete and met procedural requirements to demonstrate that systems and components were capable of performing their intended function. The inspectors also reviewed corrective action program condition reports that documented deficiencies identified during surveillance tests. Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified.

**Cornerstone: Emergency Preparedness [EP]**

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope (1 sample)

The inspectors observed one emergency preparedness (EP) drill that was included as an input into the NRC's emergency drill and exercise performance indicator. This observation was made from the Oyster Creek full-scope plant reference simulator and the technical support center (TSC) on August 22, 2007. The inspectors observed AmerGen's critique of the training activity to verify that weaknesses and deficiencies were adequately identified. The inspectors specifically focused on ensuring AmerGen identified performance issues associated with event classification, notification activities, and protective action recommendations.

b. Findings

No findings of significance were identified.

**2. RADIATION SAFETY**

**Cornerstone: Occupational Radiation Safety**

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope (14 Samples)

The inspectors reviewed activities and associated documentation in the area of access control to radiologically significant areas.

The inspectors walked down radiological controlled areas at Oyster Creek and verified that radiological controls (postings, barricading, and access controls) were acceptable. During walk downs, the inspectors conducted independent radiation surveys to evaluate ambient conditions and adequacy of applied radiological controls.

The inspectors evaluate the adequacy and effectiveness of radiological controls utilized during two recently completed radiologically significant work activities. The work activities involved clean-out of the spent fuel pool (RWP-07-125) and reactor recirculation pump seal replacement (RWPs-07-0713, 0722). The review included evaluation of the adequacy of applied radiological controls including radiation work permits, procedure adherence, radiological surveys, job coverage, system breach surveys, airborne radioactivity sampling, contamination controls, and barrier integrity and associated engineering control performance. The inspectors also reviewed the use of electronic personnel dosimetry (EPDs) by Oyster Creek personnel.

The inspectors reviewed internal dose assessments for 2006 and 2007 (to date) to identify apparent occupational internal doses greater than 50 millirem committed effective dose equivalent (CEDE). The review involved an evaluation of dose assessments performed by AmerGen. The inspectors also reviewed the program which monitors potential intakes associated with hard-to-detect radionuclides (e.g., transuranics).

The inspectors reviewed self assessments and audits related to access control to radiological areas to determine if identified problems were being entered into the corrective action program for resolution. The inspectors reviewed corrective action program condition reports to determine if repetitive issues were occurring that could lead to more significant problems. The review also included an evaluation of condition reports to determine if any problems involved NRC performance indicator (PI) events with dose rates greater that 25 R/hr at 30 centimeters, greater than 500 R/hr at 1 meter or unintended exposures greater than 100 millirem total effective dose equivalent (TEDE), 5 rem shallow dose equivalent (SDE), or 1.5 rem lens dose equivalent (LDE).

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope (5 Samples)

The inspectors reviewed activities and documentation associated with radiological planning and controls to determine if AmerGen was implementing operational, engineering, and administrative controls to maintain personnel occupational radiation exposure as low as is reasonably achievable (ALARA). The inspectors evaluated AmerGen's performance against criteria contained in 10 CFR 20, "Standards For Protection Against Radiation," AmerGen procedures, and applicable industry standards..

The inspectors, as part of their inspection planning, reviewed information regarding Oyster Creek's collective dose history and current exposure trends. The inspectors evaluated Oyster Creek's collective exposures (using NUREG-0713 and plant historical data) and source-term (average contact dose rate with reactor coolant piping) measurements. The inspectors also evaluated Oyster Creek's three-year rolling average collective exposure. The inspectors reviewed AmerGen's procedures associated with maintaining occupational exposures ALARA.

The inspectors reviewed Oyster Creek's 2006 outage report to assess the results achieved (dose and dose rate reductions, person-rem expended) versus the estimated occupational doses established in the initial ALARA plans for various work activities conducted during the fall 2006 refuel outage. The inspectors reviewed implementation of program requirements for re-evaluation of dose estimates including re-review of work plans by the station ALARA committee. The inspectors also reviewed exposure tracking for ongoing outage activities.

The inspectors reviewed AmerGen's "Five Year ALARA Plan" to evaluate AmerGen's source term controls at Oyster Creek. The review included source term, chemical controls, shutdown methodology, and clean-up strategies. The inspectors also reviewed primary system piping radiation measurements including trends and current status. The inspectors discussed longer term source term reduction plans and efforts with radiation protection personnel.

The inspectors reviewed self assessments, audits, and special reports related to the ALARA program to determine if identified problems were being entered into the corrective action program for resolution. The inspectors also reviewed dose significant post-job (work activity) reviews and post-outage ALARA report critiques involving exposure performance to determine if identified problems were being properly characterized, prioritized, and resolved in the corrective action program.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03)

a. Inspection Scope (5 Samples)

The inspectors reviewed activities and associated documentation in the area of radiation monitoring instrumentation and protective equipment.

The inspectors reviewed the radiological source term data based on 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," in order to identify potential changes in radiation types and energies that could impact calibrations or analyses. The inspectors reviewed calibrations and operability determinations for instruments used for job coverage, laboratory counting, and personnel monitoring. The calibrations of the following instruments was reviewed:

- Survey meters: RO-2 -73359, RO-2-78801, RO-7-1026, RM-14-7417, telepole - 334713, and AMP 100- 5097-016;
- Laboratory instruments: Ludlum-700504, Ludlum-700509, Ludlum-25361, Ludlum-5361, SAC-78908, SAC-242, and Ge-Li -25361;

- Continuous air monitors: 70029 and 342;
- Personnel contamination monitor: 700535 and 702450; and
- Personnel electronic dosimeters: 28285, 28568, 28050, and 37714.

The inspector also reviewed the calibration records on high-range, containment radiation monitors, and other risk significant area radiation monitors.

The inspectors reviewed the functional testing of self-contained breathing apparatus (SCBA) to ensure equipment was being maintained in an operable condition. The components on three in-plant SCBA units were checked against approved component lists published by the SCBA manufacturer and the National Institute for Occupational Safety and Health (NIOSH). The inspectors reviewed periodic testing of the three SCBA units' components (i.e., hydro testing of tank, maintenance and testing of regulators, low pressure alarms) to ensure testing was being performed in accordance with published certification lists. The inspectors reviewed training and qualifications associated with use of the SCBAs by Oyster Creek personnel.

The inspectors reviewed self assessments and audits related to radiation monitoring equipment and protective equipment to determine if identified problems were being entered into the corrective action program for resolution. The inspectors also reviewed condition reports to determine if identified problems were being properly characterized, prioritized, evaluated, and resolved in the corrective action program.

b. Findings

No findings of significance were identified.

**Public Radiation Safety Cornerstone**

2PS1 Radiological Environmental Monitoring Program (REMP) And Radioactive Material Control Program (71122.03)

a. Inspection Scope (1 sample)

The inspectors reviewed activities and associated documentation in the area of unrestricted release of material from the radiologically controlled areas (RCA).

The inspectors performed a walk down of several locations where potentially contaminated material could leave the RCA to evaluated the adequacy of AmerGen's radiological monitoring in these areas. The inspectors reviewed the methods used for control, survey, and release from these areas. The inspectors observed radiation protection personnel surveying and releasing material for unrestricted use to verify it was being performed in accordance with AmerGen procedures. The inspectors verified that the radiation monitoring instrumentation being used was appropriate for the radiation types present on the material and was calibrated with appropriate radiation sources. The inspectors reviewed AmerGen's criteria used to survey and release potentially contaminated material; and verified that there was guidance on how to respond to an alarm which indicates the presence of licensed radioactive material.

The inspectors verified that equipment and procedures used to ensure that radiation detection sensitivities were consistent with NRC and industry guidance for surface contamination (IE Circular 81-07 and IE Information Notice 85-92) and volumetrically contained material (HPPOS-221). The inspectors reviewed portable and stationary monitors. The inspectors also verified that the AmerGen performed radiation surveys to detect radionuclides that decay via electron capture.

b. Findings

No findings of significance were identified

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope (2 samples)

The inspectors reviewed AmerGen's program to gather, evaluate, and report information on two performance indicators (PIs) associated with the barrier integrity cornerstone. The inspectors used the guidance provided in Nuclear Energy Institute (NEI) 99-02, Revision 4, "Regulatory Assessment Performance Indicator Guideline" to assess the accuracy of AmerGen's collection and reporting of PI data. The inspectors reviewed operating logs and corrective action program condition reports.

The inspectors verified the accuracy and completeness of the reported data between July 1, 2006 through June 30, 2007 for the following PIs:

- Reactor Coolant System Activity; and
- Reactor Coolant System Leakage.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered Into the Corrective Action Program

The inspectors performed a daily screening of items entered into AmerGen's corrective action program to identify repetitive equipment failures or specific human performance issues for follow-up. This was accomplished by a variety methods including, reviewing hard copies of each condition report, attending daily screening meetings, or accessing AmerGen's computerized database.

.2 Annual Sample Review

a. Inspection Scope (1 sample)

The inspectors reviewed AmerGen's evaluation and corrective actions associated with the following issue:

Refueling Outage 1R21 Snubber Inspection and Testing Deficiencies. The inspectors conducted a review of a common cause analysis (CCA) report (IR554069) which addressed snubber inspection and testing deficiencies. Specifically, the CCA report evaluated the 18 snubber inspection and testing deficiencies that were identified during a refueling outage in October 2006 (1R21). During the refueling outage a total of 47 snubbers were visually inspected and 27 snubbers were functionally tested.

The inspectors verified that the snubber testing and inspection deficiencies were appropriately evaluated and operability issues dispositioned in accordance with technical specifications 3.5.A.8 and 4.5.M, "Oyster Creek Generating Station Snubber Program Plan," the inservice inspection (ISI) program, and Oyster Creek procedures.

The CCA report identified that two Pacific Scientific PSA-10 mechanical snubbers, 212-0163 and 411-0026, failed the functional testing acceptance criteria due to hardened grease during the October 2006 refueling outage. The inspectors reviewed the operability evaluations for the two failed mechanical snubbers, including the piping systems operability evaluations, common causes and previous occurrences of mechanical snubber functional testing failures due to hardened grease at Oyster Creek.

The inspectors reviewed two previous mechanical snubber (212-0155 and 411-0003) functional testing failures that occurred at Oyster Creek due to hardened grease during the October 2002 refueling outage (1R19). In addition, the inspectors evaluated operating experience related to hardened grease issues in mechanical snubbers. The inspectors verified that corrective actions were developed to resolve the snubber inspection and testing deficiencies. The inspectors specifically reviewed the identified corrective actions to address the common cause failures of mechanical snubbers due to hardened grease.

Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings and Observations

No findings of significance were identified.

The CCA report concluded that Oyster Creek fully evaluated the snubber issues and corrective actions were developed to address the deficiencies. The CCA was detailed, thorough, and adequately addressed the causes of the snubber inspection and testing deficiencies identified during the Oyster Creek 1R21 outage.

The inspectors noted that AmerGen appropriately replaced the two deficient PSA-10 mechanical snubbers, 212-0163 and 411-0026, that failed to meet the functional testing acceptance criteria with operable snubbers; and performed engineering operability evaluations of the effects of the snubber failures on their supported piping system and structural supports. The evaluations which concluded the affected supports and piping systems remained operable and capable of performing their intended safety function were found acceptable by the inspectors. Therefore, there was no safety significance to the two mechanical snubber failures. The inspectors also concluded that the proposed corrective actions to address the various inspection and testing deficiencies were identified, assigned, tracked, and appropriately focused to correct the issues identified.

However, based on Pacific Scientific Technical Division 1992 information letter addressing lubricity of NRRG -159 grease, NRC Information Notice 94-48, "Snubber Lubricant Degradation in High-Temperature Environments," and Oyster Creek operating experience which included the two previous mechanical snubber functional testing failures due to hardened grease in 2002; the inspectors concluded that Oyster Creek did not fully utilize operating experience related to hardened grease in Pacific Scientific PSA-10 mechanical snubbers and this resulted in two additional mechanical snubbers failing functional testing during the 2006 refueling outage. The inspectors concluded this was a minor finding and will not be cited per NRC IMC 0612, "Power Reactor Inspection Reports."

4OA3 Event Followup (71153) (3 samples)

.1 'C' Feedwater Regulating Valve Air Line Shear

a. Inspection Scope

On July 5, 2007 during a plant walkdown, the inspectors identified an air leak due to a sheared air line to 'C' feedwater regulating valve air line (FWRV). The inspectors informed operations department personnel of the issue of the failed air line and observed AmerGen's investigation and operator actions in response to this issue. AmerGen personnel placed 'C' FWRV in local manual operation in accordance with operating procedure 317, "Feedwater System", and replaced a failed carbon steel pipe nipple which was the source of the air leak. Critical operating parameters were not impacted and no plant transients were observed while this condition existed. The plant remained at full power throughout the repair activity. The operators performed a small downpower (to 99.5 % power) prior to placing 'C' FWRV in automatic operation and returning the plant to full power.

The failed pipe nipple was quarantined and sent off-site for analysis. AmerGen performed an evaluation (IR 647498) and determined that the pipe nipple failure mechanism was high cycle fatigue due to normal system vibrations.

Documents reviewed for this inspection activity are listed in the Supplemental Information attachment to this report.

b. Findings

No findings of significance were identified.

.2 ‘C’ Reactor Feedwater Pump Trip and Reactor Scram

a. Inspection Scope

On July 17, 2007, Oyster Creek experienced an automatic reactor scram due to a low reactor water level following a trip of the ‘C’ RFP. The RFP tripped due to an electrical ground fault in the pump’s motor. Operators attempted to reduce reactor power in accordance with abnormal operating procedure ABN-17, “Feedwater System Abnormal Conditions,” by reducing reactor recirculation flow in an effort to reduce the magnitude of mismatch between steam flow and feedwater flow. However, reactor power was not reduced quickly enough to prevent an automatic reactor scram on low reactor water level (138 inches top of active fuel (TAF)). In order to mitigate the reactor scram and stabilize the plant, operators implemented abnormal operating procedure ABN-1, “Reactor Scram,” emergency operating procedure (EOP) EMG-3200.01A, “RPV Control - No ATWS”, EOP support procedure 2, “Feedwater/Condensate,” and EOP support procedure 3, “CRD.”

Following the reactor scram, reactor water level dropped to approximately 86 inches TAF which is below the reactor protection system (RPS) low-low water level setpoint (90 inches TAF). This resulted in a reactor primary containment isolation (reactor water cleanup (RWCU) system isolation and main steam isolation valve (MSIV) closure), reactor recirculation pump trips, and actuation of several other safety systems (IC, EDG, core spray, and SBGT) per the plant’s design. The primary containment isolation caused the MSIVs to close and initiation of the ICs to control reactor pressure. After the scram, reactor water level increased which caused a high reactor water level and the need for operations personnel to remove the ICs from service and cycle the EMRVs open to control reactor pressure. The steam relief from the EMRVs caused an increase in torus temperature that was controlled by using the containment spray system in the torus cooling mode in accordance with EOP EMG-3200.02, “Primary Containment Control.” Operators stabilized reactor water level and pressure and used the ICs to cool down the plant after the reactor scram. Operators placed the plant in cold shutdown using the shutdown cooling system. Attachment B of this report contains a time line of the event.

During the event, a small expected amount of activity was released to the environment from the ICs which were steaming off condensate water that contains trace amounts of radioactive tritium. AmerGen estimated the activity released during the event was about 1.05 curies, which would result in a worst case offsite dose of about 0.005 millirem. This worst case offsite dose is a very small fraction (approximately 1/3000th) of the 15 millirem annual organ dose limit and the 25 millirem annual total body dose limit. A senior regional inspector, specializing in health physics, reviewed the gas and liquid sample results as well as calculations performed by AmerGen personnel to verify that they accurately characterized the potential occupational and public dose impacts. The regional specialist also walked down the area where the release occurred. The regional specialist determined that AmerGen conservatively calculated projected public and occupational doses, and the doses calculated were well below applicable limits and no reporting criteria were met. The regional specialist concluded that AmerGen's dose assessments did not identify any significant public or occupational doses associated with the steam release.

AmerGen reported this event to the NRC in Event Notification 43495, "Automatic Reactor Scram on Low Reactor Vessel Level (RVWL)." Additional information on this event is contained in the NRC's "Preliminary Notification of Event or Unusual Occurrence Report" PNO-I-07-006A, dated July 23, 2007 (ADAMS Accession Number: ML072040203).

The resident inspectors responded to the control room following the reactor scram and observed the response of AmerGen personnel to the event, including operator actions in the control room. At the time of the event the inspectors verified that conditions did not meet the entry criteria for an emergency action level (EAL) as described in the Oyster Creek EAL matrix. In addition, the inspectors reviewed 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," to verify that AmerGen properly notified the NRC during the event. The inspectors also reviewed technical specification requirements to ensure that Oyster Creek was operated in accordance with its operating license.

The resident inspectors observed the PORC meeting prior to plant startup to evaluate whether AmerGen appropriately resolved issues identified during the event. The resident inspectors, with assistance from a regional inspector (in-office) who specializes in operational performance, reviewed AmerGen's post-trip report to gain additional information pertaining to the event; and ensure that human performance and equipment issues were properly evaluated and understood prior to plant startup.

In accordance with NRC IMC 0309,"Reactive Inspection Decision Basis for Reactors," the Region 1 Senior Reactor Analyst (SRA) performed an estimate of the event's risk significance to determine the type of inspection that should be performed to assess AmerGen's performance. The SRA estimated that the conditional core damage probability (CCDP) given this event was in the range of 1 core damage accident in 400,000 such transients. This CCDP estimate was conducted in accordance with NRC Inspection Manual Chapter 0309, using the Oyster Creek 3.31 Standardized Plant Analysis Risk (SPAR) model, which was changed to address the fact that operators would not use the isolation condensers if reactor water level was above 160 inches and could use them if level was returned to below 160 inches. The event was best represented by a plant transient (TRANS) with: failure of the 'C' RFP to run; closure of the MSIVs, and inability to use the ICs without operator action if level increased above 160 inches. The initiating event (IE) and the dominate core damage sequence safety function successes (S) and failures (F) included the following:

| | |
|---|---|
| IE | Failure of “C” RFP without a subsequent rapid power reduction caused a low level Reactor scram and subsequent low-low water level isolations (MSIV closure) |
| S | Reactor protection system |
| S | EMRVs closed |
| S | Feedwater - two of three pumps remained and only one was needed |
| F | Both IC - because of operator failure to take action to return the water level to below 160 inches and reinitiate the ICs |
| F | Main Condenser - MSIVs were closed |
| S | Depressurization of the reactor coolant system using EMRV |
| F | Suppression pool cooling (SPC) and SDC - because of operator error |
| F | Containment - Operators fail to vent prior to rupture |
| F | Late injection sources after containment rupture |

Based on the SRA’s analysis and NRC Region 1 management review it was determined that it would be appropriate to evaluate the event in accordance with inspection procedure 71153, “Event Followup.”

After reactor startup on July 22, 2007 the resident inspectors, with assistance from a regional inspector who specializes in operational performance, performed additional onsite inspections. The inspectors collected data from the plant computer to further evaluate plant conditions prior to, during, and following the event. They reviewed plant computer data, control room logs and conducted interviews with plant personnel to gain an understanding of how operations personnel and plant equipment responded during the event. The inspectors evaluated AmerGen’s program and process associated with event response to ensure they adequately implemented station procedures OP-AA-108-114, “Post Transient Review” and OP-AA-106-101-1001, “Event Response Guidelines”. In addition, the inspectors verified that the simulator adequately reflected the plant’s response.

The inspectors reviewed AmerGen’s evaluation (IR 653321) on how operations personnel performed prior to, during, and after the reactor scram. The inspectors will review AmerGen’s evaluation (IR 650654) into the cause of the ‘C’ RFP motor failure after it is completed (see section 1R12 for inspection details).

b. Findings

Introduction. A self-revealing finding was identified when AmerGen personnel did not properly implement procedural guidance during a response to a RFP trip which resulted in a reactor scram on July 17, 2007. Specifically, the operating crew did not properly reduce reactor power as directed by an abnormal operating procedure, and did not properly implement EOP support procedures which challenged reactor water level control during recovery activities. This finding was of very low safety significance (Green) and determined to be a NCV of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.”

<u>Description.</u> During review of this event, the inspectors identified two performance issues related to how the operating crew responded and mitigated the consequence of a RFP trip on July 17, 2007. Both issues involved inadequate implementation of procedures. The inspectors determined that the performance issues were isolated to the operating crew on duty because training examinations in the Oyster Creek's simulator after the event demonstrated acceptable performance in responding to a "simulated" RFP trip by the other operating crews.

The performance deficiency associated with this self-revealing finding involved the failure of the operating crew during the event to properly implement procedural guidance in abnormal and emergency operating procedures during response to a RFP trip and reactor scram. Therefore, the performance deficiency involves two examples of the operating crew not properly implementing procedures during the event and are described below. AmerGen's corrective actions for this issue included revising the abnormal procedure to provide enhanced instructions, providing all operations personnel remedial training sessions in the simulator on this event, and issuing a standing order communicating management's expectations on operator response.

<u>Abnormal Operating Procedure Not Properly Implemented</u>

After the RFP trip the reactor operator (RO) received alarms in the control room, informed the senior reactor operator (SRO) of the pump trip, and made a plant announcement. The SRO entered ABN-17, "Feedwater System Abnormal Conditions" and immediately ordered a rapid power reduction to 70% reactor power. The RO made an initial adjustment on the master reactor recirculation flow controller to begin the power reduction approximately 11 seconds into the event. The initial rate of speed reduction on the recirculation flow controller caused a 1.2 Hz reduction over 13 seconds. Recirculation pump speed was lowered at a faster rate by another 2.9 Hz over the next 6 seconds. At that time, the reactor automatically scrammed on low RPV water level.

The operator's action to rapidly reduce reactor power resulted in a reduction of 4.1 Hz which corresponded to the recirculation flow controller reading of approximately 42 Hz and an approximate power reduction of 20%, as indicated by average power range monitor (APRM) readings of 80% reactor power. Based on simulator observations and review of operational training material by the inspectors, reactor power needed to be rapidly reduced in a timely manner to below 70% in order to mitigate the transient and minimize likelihood of a reactor scram. Further details are provided below.

AmerGen's training personnel re-enacted the event on the Oyster Creek full-scope plant reference simulator and verified the accuracy of the re-enactment through comparison with plant data recorded on the plant computer alarm summary and parameter trend plots. The inspectors observed the re-enactment, which showed that a delay of 11 seconds from the time of event until an operator's initial action on the master recirculation flow controller had negligible effect on the ability of operations personnel to properly mitigate the event. The inspectors noted during the demonstration that the use of proper human performance error prevention methods (communication, self-checking, and use of redundant instrumentation) resulted in 10 seconds elapsing from the time of

RFP trip until the flow reduction was initiated. The inspectors also noted that proper operation of the master recirculation flow controller to promptly reduce the controller to approximately 30 Hz was successful at reducing reactor power to 70% with only a minor perturbation in reactor water level. In this demonstration, level dropped to 150 inches TAF from a normal level of 161 inches TAF. In contrast, during the event, reactor water level dropped to the low level trip set point of 138 inches TAF within 30 seconds of the 'C' RFP trip.

The inspectors reviewed AmerGen's operator simulator scenario exercise guides for both the initial and the re-qualification training programs. The inspectors noted that multiple exercise guides included training for response to the trip of a single RFP during power operation. The training material provided details on the proper and expected response by operators to this type of transient. The material stated that operators are to rapidly reduce power to within the capacity of two feed pumps (approximately 70% power) using recirculation flow to maintain water level above the low level trip setpoint. This performance standard is mirrored in the guidance contained in abnormal operating procedure ABN-17. AmerGen's evaluation into operator performance associated with this event concluded that the operator's response in reducing power during the transient was inadequate and that one of the contributing factors was a failure to rapidly lower the set point of the master recirculation controller.

<u>EOP Support Procedures Not Properly Implemented</u>

Following the reactor scram, reactor water level dropped to below the reactor protection system (RPS) low-low water level setpoint (90 inches TAF) which resulted in a reactor primary containment isolation, recirculation pump trips, and actuation of several safety systems (including the ICs) per the plant's design. Reactor water level recovered rapidly due to the feedwater supplied by 'A' and 'B' RFPs and reactor pressure dropped to less than 800 psi. Operators closed the condensate return valves on both ICs approximately two minutes into the event (from time of the trip of 'C' RFP) to stop the depressurization. This action caused water level to drop into the desired control band between 138 inches and 160 inches TAF. The inspectors determined that a combination of factors, including main feedwater regulating valve/low flow feedwater control valve leakby (~150 gpm), control rod drive injection flow (~150 gpm), heatup of the reactor coolant system (RCS) and no reactor steaming path subsequently resulted in level going high, out of the control band established by the senior reactor operator.

The inspectors noted that operating procedure 307, "Isolation Condenser System," does not allow placing an IC in service with water level above 160 inches TAF. Therefore, as reactor pressure continued to increase due to decay heat from the core, the operating crew manually operated EMRVs to control pressure. Level rose above 180 inches TAF when the 'A' EMRV and when the 'D' EMRV was opened. The crew took appropriate action to close the isolation condenser steam inlet valves in response to the high level condition approximately 13 minutes into the event. Much later, at 90 minutes into the event, level again rose above 180 inches TAF and operators again took appropriate actions to isolate the steam path into the ICs.

EOP support procedures (SP) SP-2, “Feedwater/Condensate” and SP-3, “CRD" provide operators with a graduated, symptom-based approach to reactor water level control. These procedures direct the control of reactor water level, and provide specific actions at established threshold levels. For example, when reactor water level cannot be maintained below 160 inches TAF, SP-2 directs closure of heater bank outlet valves on the operating RFPs. For the same level condition, SP-3 directs throttling closed the CRD bypass valve (V-15-30) and flow control valve (NC-30). If level reaches 170 inches TAF, these procedures direct tripping all operating RFPs and CRD pumps.

The inspectors noted that reactor water level rose above 170 inches TAF at 7.5 minutes into the event and remained greater than 170 inches TAF for about 2 minutes. At 10.5 minutes level again rose above 170 inches TAF. Contrary to the requirements of SP-2, the operating crew did not stop ‘A’ RFP until 13.5 minutes into the event, approximately 6 minutes after first reaching the action threshold level; and the requirements of SP-3, to stop the CRD pumps was also not taken. The inspectors noted that CRD pumps continued to inject into the reactor vessel at a relatively high flow rate until the scram signal was reset at 35 minutes into the event.

The inspectors reviewed plant computer plots/graphs of reactor water level and concluded that the operating crew controlled level high out of the desired level band provided in EOP EMG-3200.01A for an extended period of time due to excessive feedwater and CRD injection flow during the event. This resulted in the ICs being unavailable for a period of time during the event.

The performance deficiency associated with this self-revealing finding involved the failure of the operating crew during the event to properly implement procedural guidance in abnormal and EOP procedures during response to a RFP trip and reactor scram.

Analysis. The finding was more than minor because it was associated with the human performance attribute of the mitigating systems cornerstone and affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. AmerGen’s corrective actions for this issue included revising the abnormal procedure to provide enhanced instructions, providing all operationis personnel remedial training sessions in the simulator on this event, and issuing a standing order communicating management’s expectations on operator response.

The finding was assessed in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors performed a Phase 1 screening and determined that a Phase 2 evaluation was required to assess safety significance because the failure to properly implement procedure guidance in response to and during the event affected both the initiating and mitigating cornerstones. Specifically, the failure to properly implement the abnormal procedure resulted in an increase in the chance of a reactor scram initiating event and the failure to properly implement the EOP support procedure resulted in the unavailability of the ICs as a mitigating system, when reactor vessel water level was above 160 inches. A Region 1 SRA determined that the Oyster Creek site specific pre-solved Phase 2 SDP worksheets were not suited to assess this event because of the multiple conditions that needed to be evaluated and that a Phase 3 analysis should be performed.

The SRA conducted a Phase 3 analysis using the Oyster Creek 3.31 SPAR model, which was changed to address the fact that operators would not use the isolation condensers if reactor water level was above 160 inches TAF and could use them if level was returned to below 160 inches TAF. The SPAR analysis assumptions applied were:

- Over a year, the loss of any one of the three feedwater pumps would result in a low level reactor scram, because all crews would not have quickly reduced reactor power. The initiating event frequency estimated for the loss of a single feedpump was 0.1 per reactor year.
- The ICs would be unavailable, without additional operator actions, because all crews would not control reactor vessel water level below 160 inches.

The analysis estimated an increase in core damage frequency, because of the finding, in the range of 1 core damage accident in 5,000,000 years of reactor operation, low E-7 per year. This estimate was conservatively high because there was some indication that other operating crews would have been able to prevent the low level reactor scram by quickly reducing reactor power. The IE and the dominate core damage sequence safety function successes (S) and failures (F) included the following:

| | |
|---|---|
| IE | Over a reactor year failure of any one of the three feedwater pumps would have resulted a low level Reactor scram and subsequent low-low water level isolations (MSIV closure) ( Frequency 0.1 per reactor year). |
| S | RPS |
| S | EMRVs closed |
| S | Feedwater - two of three pumps remained and only one was needed |
| F | Both IC - because of operator failure to take action to return the water level to below 160 inches and reinitiate the ICs |
| F | Main Condenser - MSIVs were closed |
| S | Depressurization of the reactor coolant system using EMRV |
| F | SPC and SDC - because of operator error |
| F | Containment - Operators fail to vent prior to rupture |
| F | Late injection sources after containment rupture |

The finding was determined to be of very low safety significance (Green) using the Phase 1, Phase 2 and Phase 3 of the SDP.

The performance deficiency had a cross-cutting aspect in the area of human performance because the operating crew did not follow procedures during their response to the event [H.4(b)].

<u>Enforcement.</u> 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures and Drawings" states in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions,

procedures, or drawings. Contrary to the above, on July 17, 2007, AmerGen did not properly accomplish procedurally directed actions contained in abnormal procedure ABN-17, "Feedwater System Abnormal Conditions," EOP SP-2, "Feedwater/Condensate, " and EOP SP-3, "CRD." Specifically, operators did not perform a rapid power reduction as required by ABN-17, and they did not stop the RFP and CRD pumps as required by EOP SP-2 and SP-3 respectively. However, because the finding was of very low safety significance and has been entered into AmerGen's corrective action program in IR 653321, this violation is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy. (**NCV 05000219/2007004-04, Inadequate Procedure Implementation During Response to a Reactor Feedpump Trip and Reactor Scram**)

.3 (Closed) LER 05000219/2007-001-00, Automatic Reactor Scram Following Trip of Reactor Feed Pump.

This license event report (LER) described an event involving an automatic reactor scram due to a low reactor water level following a trip of the 'C' RFP on July 17, 2007. The cause of the 'C' RFP trip was attributed to an electrical fault internal to the motor. In section 1R12 and 4OA3.2 of this report, the inspectors evaluated how station equipment and AmerGen personnel performed during the event described in this LER. The inspectors reviewed this LER and determined that the information in the LER was accurate. This LER is closed.

4OA6 Meetings, Including Exit

Regional Administrator Site Visit. On September 18, 2007, a site visit was conducted by Mr. S. Collins, Regional Administrator, for the Region 1 office. During Mr. Collins' visit, he toured the plant and met with AmerGen managers. Mr. J. Clifford, Deputy Director Division of Reactor Projects for the Region 1 office, accompanied Mr. Collins on his visit.

Resident Inspector Exit Meeting. On October 26, 2007, the inspectors presented their overall findings to members of AmerGen's management led by Mr. J. Randich, Plant Manager, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information reviewed during the inspection period was returned to AmerGen.

4OA7 Licensee-Identified Violations

None.

ATTACHMENTS: A. SUPPLEMENTAL INFORMATION
B. SEQUENCE OF EVENTS

# ATTACHMENT A

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

<u>Licensee Personnel</u>
J. Dent, Director, Work Management
J. Dostal, Shift Operations, Superintendent
S. Dupont, Regulatory Assurance Specialist
R. Gayley, Program Manager, GL 89-13 Program
S. Hutchins, Senior Manager Design Engineering
T. Keenan, Manager Security
D. Kettering, Director, Engineering
J. Kandasamy, Manager, Regulatory Assurance
G. Ludlam, Director, Training
J. Magee, Director, Maintenance
J. Makar, Senior Manager System Engineering
D. Olszewski, Snubber Program Manager
P. Orphanos, Director, Operations
D. Peiffer, Manager Nuclear Oversight
J. Randich, Plant Manager
T. Rausch, Site Vice President
H. Ray, Manager, Engineering Programs
J. Renda, Manager Radiation Protection
T. Schuster, Manager Environmental/Chemistry Manager
S. Schwartz, System Engineer
T. Sexsmith, Manager Corrective Action Program
L. Velez, System Manager

<u>Others</u>:
R. Penny, State of New Jersey, Bureau of Nuclear Engineering
P. Schwartz, State of New Jersey, Bureau of Nuclear Engineering

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

| | | |
|---|---|---|
| Opened | | |
| 05000219/2007004-01 | URI | 'C' Reactor Feedpump Motor Failure (Section 1R12) |
| Opened/Closed | | |
| 05000219/2007004-02 | NCV | Degraded Condition on the Remote Shutdown Panel Not Properly Identified (Section 1R15) |
| 05000219/2007004-03 | NCV | Improper Repair of a Fire Rated Penetration Seal (Section 1R19) |
| 05000219/2007004-04 | NCV | Inadequate Procedure Implementation During Response to a Reactor Feedpump Trip and Reactor Scram (Section 4OA3) |
| 05000219/2007-001-00 | LER | Automatic Reactor Scram Following Trip of Reactor Feed Pump (Section 4OA3) |

## LIST OF DOCUMENTS REVIEWED

In addition to the documents identified in the body of this report, the inspectors reviewed the following documents and records:

**Section 1R04: Equipment Alignment**

Procedures
310, "Containment Spray System Operation"
610.4.021, "Core Spray System 1 Pump Operability and Quarterly In-Service Test"
RAP S7d, "SBO XFMR Trouble"

Drawings
GE 148F740, "Containment Spray System Flow Diagram"
GE 885D781, "Core Spray System Flow Diagram"
BR 2010, "Control Room and Cable Spreading Room HVAC System"

Condition Reports (IR)
661885, 662245, 611138, 577158, 348082, 350627, 350629, 350860, 350866, 350901,371866, 372418, 382311, 442416, 469882, 485153, 485337, 517643, 579375, 579383, 606494, 636337, 639208, 642147, 645374, 657037, 656992, 657029, 653146, 654529, 671918, 672305, 672397, 672597, 672850, 667327, 652302

Work Orders (AR)
A2118414, A2119029, AR2067359, A2119021, A2119020, A2108137, A2125339, A2168769

Other Documents
Maintenance rule system performance and functional failure definitions for CRHVAC system
CRHVAC system health reports

**Section 1R05: Fire Protection**
Procedures
ABN-29, “Plant Fires”
101.2, “Oyster Creek Site Fire Protection Program”
CC-AA-211, “Fire Protection Program”
333, “Plant Fire Protection System”

Drawings
GU 3D-911-02-012, “Fire Area Layout- Emergency Diesel Generator Vaults”
GU 3D-911-02-002, “Fire Area Layout- “D” 4160V Switchgear Room”

Condition Reports (IR)
658522, 654354, 646890, 676306

Other Documents
OC Fire Risk Analysis-Compartment Fire Scenario Development Report (R0467050033.04)
Oyster Creek Nuclear Generating Station Fire Hazard Analysis Report (990-1746)
Oyster Creek Fire Plans (OP-OC-210-008)

**Section 1R07: Heat Sink Performance**
Procedures
607.4.014, “Containment Spray and ESW System 2 Pump Operability, IST and Containment Spray Pumps Trip”
125.1, “Containment Spray Heat Exchanger Temperature Acquisition for Testing”
309.2, “Reactor Building Closed Cooling Water System”
310, “Containment Spray System Operation”
322, “Service Water System”
326, “Chlorination System”
2400-SMM-3214.02, “Containment Spray Heat Exchanger Cleaning and Assembly”
ABN-18, “Service Water Failure Response”
ABN-19, “RBCCW Failure Response”
ABN-31, “High Winds”
ABN-32, “Abnormal Intake Level”
CY-AA-120-410, “Circulating/Service Water Chemistry”
ER-AA-310, “Implementation of the Maintenance Rule”
ER-AA-340-1002, “Service Water Heat Exchanger and Component Inspection Guide”
ER-OC-450, “Structures Monitoring Program”
EP-OC-1010, “Radiological Emergency Plan for Oyster Creek Generating Station”

Drawings
BR 2005, Sheet 4, “Emergency Service Water System Flow Diagram”
BR 2006, Sheet 1, “Reactor Building Closed Cooling Water System Flow Diagram”
D-7799, “Reactor Building Closed Cooling Water Heat Exchangers Assembly 4 Details”

GE 148F740, “Containment Spray System Flow Diagram”

Condition Reports (IR)
354648, 360630, 373040, 374744, 389621, 436012, 442848, 452658, 466759, 502308, 509865, 517632, 530344, 549773, 554823, 556160, 550910, 564656, 581523, 587791, 587849, 588200, 600161, 605048, 606131, 613061, 615021, 618930, 623203, 630894, 636880, 637125, 642712, 642719, 646120, 648618, 650147, 660972, 661021

Work Orders (AR)
R0805542, R2089325 (PM16820M), R2103850 (PM24104I), A0782124, A2014243, A2060289, A2120853, A2140269, A2168987

Other Documents
EPRI NP-7552 Project 3052-1, “Heat Exchanger Performance Monitoring Guidelines”
EPRI TR-107397, “Service Water Heat Exchanger Testing Guidelines”
NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Rev. 2
Exchange Specification Sheet, Oyster Creek Emergency Containment Cooling System, dated 11/9/65
Self Assessment (FASA 00600161-04),”GL 89-13, Safety-Related Service Water Program”
Self Assessment (FASA AT 354289), “Oyster Creek GL 89-13 Program Focused Area Self-Assessment Report”
Calculation C-1302-214-E310-045, “OCNGS SDC Appendix R RELAP5 Analysis”
Calculation C-1302-241-E120-085, “Containment Spray System Heat Exchanger Performance Evaluation”
Calculation C-1302-241-E610-080, “Calculation of Torus Pool Temperature as NPSH Input”
607.4.014, “Containment Spray and ESW System 2 Pump Operability, IST and Containment Spray Pumps Trip,” dated 7/10/07 (completed)
607.4.016, “Containment Spray and Emergency Service Water System 1 Pump Operability and Quarterly Inservice Test,” dated 5/16/07 (completed)
Oyster Creek Generating Station Generic Letter 89-13 Program Description, dated 8/9/07
Oyster Creek Nuclear Generation Station Generic Letter 89-13 Response Letters, dated 1/30/90 and 1/13/92
SDBD-OC-241, “System Design Basis Document for Containment Spray System”
SDBD-OC-532, “Design Basis Document for Emergency Service Water System”

Evaluation A0703677, Containment Spray Heat Exchanger Performance Evaluation, dated 1/15/04, 1/21/05, 2/27/06, 4/17/07, and 5/16/07
Evaluation IR452658-02, (a)(1) Determination for System 168 (Intake Structure and Canal), dated 3/9/06
OpEval OC-2007-OE-0007
Specification S-2299-41 Addendum No. 1, “Heat Exchangers for Emergency Containment Cooling System for Jersey Central Power and Light Company Oyster Creek”
Top Ten Equipment Reliability Issues, dated 7/16/07
Topical Report 140, “Emergency Service Water and Service Water System Piping Plan”
NRC Information Notice 2006-17, “Recent Operating Experience of Service Water Systems

Due to External Conditions"
Chlorination PI Data, July 2006 – July 2007
Generic Letter 89-13 Program Health Report, 2nd Quarter 2007
Form # R2090364-10-01, Intake Structure, dated 11/14/06
Intake Staff Level, 8/1/06 – 8/7/06 and 7/24/07 – 7/31/07
RBCCW HX SW D/P Data, 6/1/07 – 8/16/07
RBCCW HX SW D/P Data, 6/1/06 – 8/31/06
Service Water Chlorination Flow and Total Residual Chlorine, 8/2/06 – 8/15/07
Service Water Chlorination Update, May 2007
System 180 Plant Structure Walk Down/Monitoring Report, dated 6/8/06
System Health Overview Report, Chlorination Systems, June 2007
System Health Overview Report, Containment Spray System, June 2007
System Health Overview Report, Emergency Service Water System, June 2007
System Health Overview Report, RBCCW & RBCCW Chem Addition, June 2007
System Health Overview Report, Service Water System, June 2007
Torus Cooling Trend with Containment Spray HXs, 7/17/07 – 7/20/07

**Section 1R11: Licensed Operator Requalification Program**
Procedures
ABN-1, "Reactor Scram"
ABN-37, "Station Blackout"
2000 EMG 3200.01A, "Reactor Pressure Vessel Control With No ATWS"
2000 GLN-3200.03, "Plant Specific Technical Guidelines for the Severe Accident Guidelines"
EP-OC-1010, "Oyster Creek Emergency Action Level (EAL) Matrix"

Other Documents
EOP User's Guide (2000-BAS-3200.02)

**Section 1R12: Maintenance Implementation**
Procedures
ER-AA-310, "Implementation of Maintenance Rule"
ER-AA-310-1005, "Maintenance Rule - Disposition Between (a)(1) and (a)(2)"

Condition Reports (IR)
650702, 655469, 650654, 672744, 672752

Work Orders (AR)
C2015525, A2172012

Other Documents
NEI 93-01, "Industry Guideline for monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
Vendor Manual:"Siemens MJ-X L(TM) Voltage Regulator Control Panel Installation and Operations Manual"

**Section 1R13: Maintenance Risk Assessments and Emergent Work Control**

Procedures
ER-AA-600-1042, "On-line Risk Management"
ER-AA-600-1021, "Risk Management Application Methodologies"
ER-AA-600-1014, "Risk Management Configuration Control"
ER-AA-600-1011, "Risk Management Program"
WC-OC-101-1001, "On-line Risk Management and Assessment"
WC-AA-101, "On-line Work Control Process"

Drawings
BR 3000, "Electrical Power System"

Condition Report (IR)
648480

Work Orders (AR)
C2015486

Other Documents
OC-2007-SUT-02, "Risk Assessment for Bank 6 SUT OOS Risk Color Adjustment to Yellow"
Risk Evaluation of Startup Transformer SB Maintenance Outage, dated May 10, 2007

**Section 1R15: Operability Evaluations**

Procedures
OP-AA-108-115, "Operability Determination"
MA-OC-773-001, "Testing/Condition Monitoring of Inaccessible Medium Voltage Cables not Subject to 10CFR50.49 Environmental Qualification"
EP-059, "Conduit Support Design and Installation"
610.4.022, "Core Spray System 2 Pump Operability and Quarterly In-Service Test"
LS-AA-120, "Issue Identification and Screening Process"
OP-AA-108-115, "Operability Determination"
WC-AA-106, "Work Screening and Processing"
346, "Operation of the Remote and Local Shutdown Panels"

Drawings
BR E1108, "Elementary Diagram Remote Shutdown Panel Transfer Scheme"

Condition Reports (IR)
642467, 653354, 652162, 652460, 653894, 651499,651916, 652008, 649140, 654413, 654428, 654433, O2001-0606, O2000-0830, O2000-0747, 351420, 645011, 643046, 654058, 657184, 653894

Work Orders (AR)
A2164080, A2172691, A2170430

Other Documents
NRC Inspection Manual - Part 9900 Technical Guidance, "Operability Determinations &

Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety
NRC Letter - “Oyster Creek - Issuance of Amendment No. 199 re: Automatic Depressurization System (TAC No. MA3413), dated October 14, 1998
NRC Letter - “Oyster Creek Nuclear Generating Station - Correction to Amendment No. 208 (TAC No. MA4145 and MA6074)”, dated October 28, 1999
NRC Letter - “Oyster Creek Nuclear Generating Station”, dated August 27, 1984
Risk-Informed Inspection Notebook for Oyster Creek Nuclear Generating Station, Revision 2.01
Operability Evaluation OC-2007-OE-0009, “Conduit and Supports for P-3-3A and P-3-3B”
ACIT 649140-02, “Technical Evaluation to determine if Plant Computer System Core Thermal Power is still valid for maintaining plant operation within the Licensed Thermal Power Limit”, dated July 12, 2007
ACMP, “Potential High Pressure Feedwater Heater Degradation Due to Tube Leakage”, dated July 13, 2007
OC-2007-OE-0010, “A and E Electromatic Relief Valves”
SDBD-OC-212, “Automatic Depressurization System”
4160V Cable DTE Test Matrix
FSAR Section 3.1.15, “Criterion 19 - Control Room”
FSAR Section 9.5.1, “Fire Protection Programs”
Technical Specification 3.12, “Alternate Shutdown Monitoring Instrumentation”

**Section 1R17: Permanent Plant Modifications**
Procedures
334, “Instrument & Service Air System”
ABN-35, “Loss of Instrument Air”

Drawings
BR2013, “Service Air Flow Diagram”

Condition Report (IR)
591923, 664305

Work Order (AR)
R2069963, A2161442

Other
3M Interam vendor manual
ECR OC 07-00428, “Replacement of mecatiss fire barrier for conduit 86-71”
ECR OC-05-00672, “Air Dryer Improvement Changes Per Vendor Recommendation”
VM-OC-2844, “Ingersoll-Rand ThermoStar TS1-13 Operations and Maintenance Manual”
FSAR Section 9.3.1, “Service and Instrument Air System”

**Section 1R19: Post-Maintenance Testing**
Procedures
MA-AA-716-012, “Post Maintenance Testing
OP-MA-109-101, “Clearance and Tagging”
CC-AA-102, “Design Input and Configuration change Impact Screening”

CC-AA-10, “Configuration Control Process Description”
120.6, “Fire Barrier Penetration Data Base Maintenance”
2400-GMM-3900.55, “Fire Penetration Seal Installation and Repair”, Rev. 12
2400-SMM-3150.16, “Mixing and Placement of Grout”
645.6.017, “Fire Barrier Penetration Surveillance”,
125.3, Leak Reduction Program Administration”
665.4.015, “Isolation condenser system leakage reduction”
OP-AA-103-105,”Limitorque Motor-Operated Valve operations”
2400-GMM-3917.51, “Installation and Use of Chesterton Packing”
641.43.001, “Service water pump operability and in-service test”
602.3.004, “Electromatic Relief Valve Pressure Sensor Test and Calibration”
2400-SME-3915.15, “Low Voltage Power Circuit Breakers Corrective/Preventive Maintenance for AKF-1B-10 Breakers”
2400-SME-3915.01, “Motor Control Center Preventive Maintenance”
2400-SME-3915.08, “Low Voltage Power Circuit Corrective/Preventative Maintenance for AK50, AKS50, AK75, AKR75 and AK100 Breakers”

Condition Report (IR)
647756, 646890, 642498, 644539, 553984, 618204

Work Order (AR)
A2169992, A2170265, A2122129, A2170799, A2138377, R2083927, A2154491, PM21115M, R2060725, A2165966, A2173914, R2039464, C2014549

Other
V-14-30 valve packing data sheet
Maintenance rule performance and functional failure definition for the isolation condenser system
Bisco Report 748-41, “Fire Test Configurations for a Three Hour Rated Fire Seal utilizing Bisco SF-20 Silicone Foam (Dow-Corning 3-6548), dated April 17, 1981
101.2, “Oyster Creek Site Fire Protection Program”, Rev. 55
990-1746, “Oyster Creek Nuclear Generating Station Fire Hazards Analysis Report”
CC-AA-209, “Fire Protection Program Configuration Change Review”, Rev. 1
CC-AA-211, “Fire Protection Program”, Rev. 2
602.3.004, “Electromatic Relief Valve Pressure Sensor Test and Calibration”, Rev. 44

**Section 1R20: Refueling and Outage Activities**
Procedures
ABN-3, “Loss of Shutdown Cooling”
201, "Plant Startup"
203, “Plant Shutdown”
305, “Shutdown Cooling System Operation”
OP-AA-108-108, “Unit Restart Review”
ABN-17, “Feedwater System Abnormal Conditions”
RAP-J1f, “Feed Pump Trip C”

Condition Report (IR)
654994, 654882, 652904

Other
1F12 Shutdown Risk Assessment

**Section 1R22: Surveillance Testing**
Procedures
SA-AA-129, “Electrical Safety”
MA-AA-1000, “Conduct of Maintenance”
602.4.003, “Electromatic Relief Valve Operability Test”
610.4.022, “Core spray system 2 pump operability and quarterly in-service test”
619.4.011, “SDIV vent and drain exercise and in-service test”
665.3.021, “Containment Electrical Penetration Nitrogen Blanket Surveillance”, Rev. 9
607.4.017, “Containment Spray and Emergency Service Water Pump System 2 Operability and Quarterly Inservice Test”,
609.4.001, “Isolation Condenser Valve Operability and In Service Test”

Drawings
GE 885D781, “Core spray system flow diagram”
GE 197E871, “SDIV flow diagram”
846D989, “Penetration Seal”

Condition Reports (IR)
656323, 512078, O2001-1759, 646222, 452744, 487415, 505882, 524147, 547175, 579485, 579677, 601643, 646222, 653219

Work Orders (AR)
R2102712, R2104036, R2106059

Other Documents
Technical specification 3.4, “Emergency cooling”
Jersey Central Power and Light Ltr, “I E Bulletin No. 77-06", dated December 19, 1977
Jersey Central Power and Light Ltr, “I E Bulletin No. 77-06", dated December 2, 1977
VM-OC-6379, “GEK-13903 Penetration Seals 216 x 752, G1, G2, G4, G6-16 and G19-22"
GE SIL 259, “Containment Electrical Penetration Integrity”, dated December 30, 1977
NRC IE Bulletin 77-06, “Potential Problems with Containment Electrical Penetration Assemblies”, dated November 22, 1977
SDBD-OC-243(MPR) Section 3.4.1.1.1.2, “Electrical Penetrations”, Rev. 0

**Section 4OA1: Performance Indicator (PI) Verification**
Procedures
LS-AA-2090, “Monthly Data Elements for NRC Reactor Coolant System (RCS) Specific Activity”
LS-AA-2100, “Monthly Data Elements for NRC Reactor Coolant System (RCS) Leakage”

Condition Reports (IR)
653982

Other Documents
Reactor Coolant System Leakage PI data and verification record, July 1, 2006 - June 30, 2007
Reactor Coolant System Activity PI data and verification record, July 1, 2006 - June 30, 2007
Adverse Condition Monitoring and Contingency Plan (ACMP) for Increasing Unidentified Leak Rate
ACMP for Potential Seal Failure for ‘C’ Recirc Pump

**Section 4OA2: Identification and Resolution of Problems**
Procedures
LS-AA-104-1001,”Oyster Creek Generating Station Snubber Program Plan”
ER-AA-330-004, “Visual Examination of Snubbers”
ER-AA-330-010, “Snubber Functional Testing “
ER-AA-330-011, “Snubber Service Life Monitoring“
ER-AA-330, “Conduct of Inservice Inspection Activities”
2400-GMM-3921.52, “Removal, Inspection and Installation of Mechanical Snubbers”
675.1.001, “Hydraulic Snubber Inspection and Replacement”

Condition Reports (Irs)
548520, 2152818, 2153712, 554473, 554069

Other Documents
GPU Nuclear SP 1302-52-045, “Requirements for Functional Testing of Snubbers”
Basic-PSA, Inc. Design Report 1319, “Mechanical Shock Arrestors Standard Design Specification”
Basic-PSA, Inc. Design Report 3020, “Mechanical Shock Arrestors Service Life Extension Program”
NRC Information Notice 94-48, “Snubber Lubricant Degradation in High-Temperature Environments”
Plant Health Committee Meeting Minutes, dated April 16, 2007
Pacific Scientific HTL/Kin Technical Division Letter, dated November 17, 1992

**Section 4OA3: Event Followup**
Procedures
OP-AA-108-114, “Post Transient Review”
OP-AA-101-111, “Roles and Responsibilities of On-Shift Personnel”
OP-AA-106-101-1001, “Event Response Guidelines”
OP-OC-100, “ Oyster Creek Conduct of Operations”
203.4, “ Plant Cooldown Following Reactor Scram”
307, “Isolation Condenser System”
305, “ Shutdown Cooling System Operation”
317, “Feedwater System”
ABN-1, “Reactor Scram”
ABN-17, “Feedwater System Abnormal Conditions” (Revision 6)
ABN-17, “Feedwater System Abnormal Conditions” (Revision 7)
ABN-17, “Feedwater System Abnormal Conditions” (Revision 8)
EMG-3200.01A, “RPV Control - No ATWS”

EMG-3200.02, “Primary Containment Control”
Support Procedure 2, “Feedwater/Condensate”
Support Procedure 3, “ CRD”
Support Procedure 11, “Alternate Pressure Control Systems Isolation Condensers”

<u>Condition Reports (IR)</u>
653321, 652904, 651298, 651251, 654882, 654994, 650860, 651540, 650910, 650654, 650907, 651271, 650702, 657526, 650910, 662815

<u>Work Orders (AR)</u>
A2171939

<u>Other Documents</u>
NUREG-1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73"
NEI 99-02, Rev 4, “Regulatory Assessment Performance Indicator Guideline”
Post Trip Report (IR 650654), Automatic Scram on July 17, 2007, dated July 18, 2007.
Plant Computer Data, dated July 17, 2007
Startup Plant On-Site Review Committee (PORC) Meeting 07-13 Report, dated July 17, 2007
Startup Plant On-Site Review Committee (PORC) Meeting 07-14 Report, dated July 17, 2007
Just-In-Time Training Material, “Crew Response to Feed Pump Trip (1F12 Scram Transient Training)
Shift Training Brief, “Expectations for Transient Mitigation,” dated July 20, 2007
Licensed Operator Requal Training simulator Exercise Guide 2621.885.0054C
Licensed Operator Requal Training simulator Exercise Guide 2621.885.0911
LER 05000219/1996-005-00, “Reactor Scram on Low Reactor Water Level Due to Operator’s Difficulty with Feedwater Level Control,” dated June 11, 1996
LER 05000219/1991-005-00, “Automatic Reactor Scram Due to Lose of Feedwater Caused by a Grounded Condensate Pump Motor,” dated September 23, 1991
LER 05000219/1992-009-00, “ Reactor Scram on Low Water Level Due to Feedwater Control Component Failure,“ dated September 15, 1992
NRC Preliminary Notification of Event or Unusual Occurrence, “ Reactor Scram on Low Reactor Water Vessel Water Level Due to a Trip of the ‘C’ Reactor Feed Pump” (PNO-I-07-006), dated July 17, 2007 (ADAMS Accession Number: ML071990428)
NRC Preliminary Notification of Event or Unusual Occurrence, “ (Update) Reactor Scram on Low Reactor Water Vessel Water Level Due to a Trip of the ‘C’ Reactor Feed Pump” (PNO-I-07-006A), dated July 23, 2007 (ADAMS Accession Number: ML072040203)
Oyster Creek Operations Narrative Logs, dated July 17, 2007
NRC Event Notification Report (Event Number 43495), “Automatic Reactor Scram on Low Reactor Water Level,”dated July 17, 2007
EOP User’s Guide (2000-BAS-3200.02)
Simulator Work Request 10241
Simulator Work Request 10325

**LIST OF ACRONYMS**

| | |
|---|---|
| ADAMS | Agency-Wide Documents Access and Management System |
| AmerGen | AmerGen Energy Company, LLC |
| ALARA | As Low As Reasonably Achievable |
| APRM | Average Power Range Monitor |
| CCDP | Conditional Core Damage Probability |
| CCA | Common Cause Analysis |
| CEDE | Committed Effective Dose Equivalent |
| CRD | Control Rod Drive |
| CRHVAC | Control Room Heating, Ventilation, & Air Conditioning |
| CT | Combustion Turbine |
| EAL | Emergency Action Level |
| EDG | Emergency Diesel Generator |
| EMRV | Electromatic Relief Valve |
| EOP | Emergency Operating Procedure |
| EP | Emergency Preparedness |
| EPD | Electronic Personnel Dosimetry |
| EPR | Electrical Pressure Regulator |
| ESW | Emergency Service Water |
| EOP | Emergency Operating Procedure |
| F | Failures |
| FWRV | Feedwater Regulating Valve |
| GPM | Gallons Per Minute |
| HCU | Hydraulic Control Unit |
| HX | Heat Exchanger |
| IC | Isolation Condenser |
| IE | Initiating Event |
| IPEEE | Individual Plant Examination External Events |
| IMC | Inspection Manual Chapter |
| ISI | In-service Inspection |
| IST | In-service test |
| IR | Condition Report |
| LDE | Lens Dose Equivalent |
| LER | Licensee Event Report |
| MPFF | Maintenance Preventable Functional Failures |
| MSIV | Main Steam Isolation Valve |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NIOSH | National Institute for Occupational Safety and Health |
| NRC | Nuclear Regulatory Commission |
| Oyster Creek | Oyster Creek Generating Station |
| PARS | Publicly Available Records |
| PORC | Plant Onsite Review Committee |
| RBCCW | Reactor Building Closed Cooling Water |
| RCA | Radiologically Controlled Area |
| RCS | Reactor Coolant System |

| | |
|---|---|
| RFP | Reactor Feedwater Pump |
| RO | Reactor Operator |
| RPS | Reaction Protection System |
| RSP | Remote Shutdown Panel |
| RWCU | Reactor Water Cleanup |
| S | Successes |
| SBO | Station Blackout |
| SCBA | Self-Contained Breathing Apparatus |
| SDC | Shutdown Cooling |
| SDE | Shallow Dose Equivalent |
| SDP | Significance Determination Process |
| SP | Support Procedure |
| SPC | Suppression Pool Cooling |
| SPAR | Standardized Plant Analysis Risk |
| SRA | Senior Reactor Analyst |
| SRO | Senior Reactor Operator |
| SSC | Structures, Systems, Components |
| SW | Service Water |
| TAF | Top of Active Fuel |
| TEDE | Total Effective Dose Equivalent |
| TBCCW | Turbine Building Closed Cooling Water |
| TRANS | Plant Transient |
| TSC | Technical Support Center |
| UFSAR | Updated Final Safety Analysis Report |
| URI | Unresolved Item |
| WO | Work Order |

## ATTACHMENT B

### SEQUENCE OF EVENTS
### Oyster Creek Automatic Reactor Scram On July 17, 2007

All entries were obtained from control room logs, interviews conducted by the licensee and the NRC, plant process computer, and corrective action condition reports. Entries that appear in italics are notes or observations made by the NRC inspectors.

Initial Plant Conditions (Pre-Event) - 100% Reactor Power

| **Time** | **Event** |
|---|---|
| 0520 | 'C' Reactor Feed Pump (RFP) trip. |
| 0520 | Control room operator starts reducing reactor recirculation pump flow per abnormal procedure ABN-17 approximately 11 seconds after the alarm received. *Initial rate of lowering recirculation pump speed causes 1.2 Hz reduction in 13 seconds. During next 6 seconds reactor recirculation pump speed is lowered at a faster rate another 2.9 Hz. A total of 4.1 hz reduction is performed which corresponds to a power reduction to 80%.* |
| 0521 | Automatic reactor scram occurs due to reactor protection system (RPS) reactor water level low condition. Main Generator trip occurs and the startup transformers in service. Operators enter abnormal operating procedure ABN-1, "Reactor Scram," emergency operating procedure (EOP) EMG-3200.01A, "RPV Control - No ATWS", EOP support procedure 2, "Feedwater/Condensate," and EOP support procedure 3, "CRD." |
| 0521 | RPS reactor water level low-low condition and the following reactor protection isolations and equipment initiations occur: (1) reactor water cleanup system isolation, MSIVs closure, reactor recirculation pumps trip; and (2) automatic actuation/start of emergency diesel generators, core spray system, isolation condensers (IC), and standby gas treatment system (SBGT) auto starts, |
| 0522 | 'A' and 'B' IC taken out of service (OOS) by operators due to lowering reactor pressure. |
| 0523 | 'B' reactor feed pump (RFP) removed from service by operators and 'A' RFP placed in manual. |
| 0524 | SBGT, EDGs, and core spray system taken OOS by operators and placed in standby. |
| 0529 | 'A' electromatic relief valve (EMRV) manually opened by operators for pressure control. |

0530 ‘A’ EMRV manually closed by operators after 67 seconds of operation.

0533 ‘D ‘ EMRV manually opened by operators for pressure control.

0534 ‘A’ and ‘B’ IC steam inlet valves closed due to reactor water level reaching 180 inches TAF.

0534 Operators remove the ‘A’ RFP from service for reactor water level control.

0535 ‘D’ EMRV manually closed by operators after 144 seconds of operation.

0536 ‘A’ RFP placed in service for level control.

0537 ‘A’ and ‘B’ IC steam inlet valves opened by operators after reactor water level goes below 180 inches TAF.

0539 ‘A’ IC placed in-service by operators for pressure control.

0551 ‘A’ IC taken OOS by operators for pressure control.

0556 RPS reactor scram signal reset by operators.

0557 ‘B’ IC placed in service by operators for pressure control.

0600 ‘A’ RFP removed from service for level control due to reactor water level swell when IC place in-service.

0606 ‘B’ IC taken OOS by operators for pressure control.

0613 ‘A’ IC placed in-service by operators for pressure control.

0627 ‘A’ IC taken OOS by operators for pressure control.

0650 ‘B’ EMRV manually opened by operators for pressure control. *IC could not be used due to reactor water level greater than 160 inches TAF.*

0651 ‘A’ and ‘B’ IC steam inlet valves closed by operators due to reactor water level reaching 180 inches TAF.

0652 ‘B’ EMRV manually closed by operators after 128 seconds of operation.

0654 A’ and ‘B’ IC steam inlet valves opened by operators after reactor water level goes below 180 inches TAF.

0656 Operators enter EOP EMG-3200.02, “Primary Containment Control” due to suppression pool/torus temperature greater than 95°F.

0703 ‘A’ IC placed in-service by operators for pressure control.

0705 Containment spray system 1 (in the torus cooling mode) placed in-service to reduce suppression pool temperatures. *Containment spray in the torus cooling mode does not involve spraying of containment/drywell. The water in the suppression pool is cooled by flowing through the containment spray heat exchangers and returned back to the suppression pool.*

0722 ‘A’ IC taken OOS by operators for pressure control.

0724 ‘A’ RFP placed in service for level control.

0727 ‘B’ IC placed in service by operators for pressure control.

0735 ‘B’ IC taken OOS by operators for pressure control.

0754 ‘B’ condensate pump taken OOS by operators in accordance with procedures.

0758 ‘B’ IC placed in service by operators for pressure control.

0800 ‘B’ IC taken OOS by operators for pressure control.

0808 ‘C’ condensate pump taken OOS by operators in accordance with procedures.

0809 ‘A’ IC placed in service by operators for pressure control.

0812 ‘A’ IC taken OOS by operators for pressure control.

0823 ‘B’ IC placed in service by operators for pressure control.

0826 ‘A’ condensate pump taken OOS by operators in accordance with procedures.

0828 ‘B’ IC taken OOS by operators for pressure control.

0839 ‘B’ IC placed in service by operators for pressure control.

0842 ‘B’ IC taken OOS by operators for pressure control.

0902 ‘A’ reactor feed pump (RFP) removed from service by operators.

0902 ‘B’ IC placed in service by operators for pressure control.

0922 Bank 6 startup transformer placed in manual due to voltage regulator not properly controlling within expected voltage band (voltage was high out of band). Operator stationed at voltage regulator in order to make voltage adjustments as needed.

| | |
|---|---|
| 0924 | 'B' IC taken OOS by operators for pressure control. |
| 0935 | 'B' IC placed in service by operators for pressure control. |
| 0939 | 'B' IC taken OOS by operators for pressure control. |
| 1005 | Shutdown cooling (SDC) system placed in-service. All pumps trip on high suction temperature. *Per the Oyster Creek Final Safety Analysis Report (FSAR) SDC is not considered an engineering safety feature therefore not reportable event.* |
| 1014 | Operators re-establish IC for pressure control. |
| 1035 | IC taken OOS and SDC system placed in-service by operators. |
| 1400 | Plant placed in a cold shutdown condition. |
| 1530 | Containment spray system 1 (in torus cooling mode) taken OOS by operators. |