May 6, 2005

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 05000219/2005002

Dear Mr. Crane:

On March 31, 2005, 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 April 8, 2005, with Mr. C. N. Swenson and other members of your staff.

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

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

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

We appreciate your cooperation. Please contact me at 610-337-5069 if you have any questions regarding this letter.

Sincerely,

/RA/

Arthur Burritt, Acting Chief
Projects Branch 7
Division of Reactor Projects

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

Enclosure: Inspection Report 05000219/2005002
w/Attachment: Supplemental Information

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Plant Manager, Oyster Creek Generating Station, AmerGen
Regulatory Assurance Manager Oyster Creek, AmerGen
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-219
License No.:	DPR-16
Report No.:	05000219/2005002
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	January 1, 2005 - March 31, 2005
Inspectors:	Robert Summers, Senior Resident Inspector Jeff Herrera, Resident Inspector Suresh Chaudharry, Senior Reactor Engineer Andrew Rosebrook, Reactor Engineer Ronald Nimitz, Senior Health Physicist Nancy T. McNamara, EP Inspector
Approved By:	Arthur Burritt, Acting Chief Projects Branch 7 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000219/2005002; 01/01/05 - 03/31/05; Oyster Creek Generating Station; Maintenance Risk Assessment and Emergent Work Evaluation, Operability Evaluations, Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems, Other.

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

A. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

C Green. A self-revealing finding and non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, was identified for a February 15, 2005, event involving an inadvertent trip of the #1 Emergency Diesel Generator during troubleshooting repairs to the area lighting system.

This finding was more than minor because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems (emergency AC power) that respond to initiating events to prevent undesirable consequences and the related attributes of equipment performance, human performance and procedure quality. The finding is of very low safety significance because the redundant train of AC power was available and the affected train safety function was lost for less than its Technical Specification allowed outage time. This finding also has a cross-cutting aspect of PI&R in that corrective actions for similar prior events were not effective at preventing a repeat condition. (Section 1R13)

C Green. A self-revealing finding and non-cited violation of Technical Specification 3.4.D was identified for failure to adequately restore the "A" control rod drive (CRD) pump to standby readiness after testing and maintenance on February 17, 2005.

This finding was more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (high pressure decay heat removal water makeup). The specific attributes of equipment performance, human performance, and procedure quality were adversely impacted for the CRD system, which functions as a high pressure injection makeup source for decay heat removal for transient event sequences. The finding is of very low safety significance because the redundant CRD pump was available and the condition was identified and corrected within 30 days. In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the inspectors conducted a Significance Determination Process (SDP) Phase 1 screening and determined that this finding required a Phase 2 approximation based upon the loss of a safety function of a single train for greater than its Technical Specification allowed outage time (AOT). The inspectors conducted a Phase 2 SDP evaluation and the issue screened to Green. The most dominant core damage sequences involved the transients without power conversion system (TPCS) and the failure of make-up to the isolation condensers and either failure of the low pressure injection system or failure to depressurize the reactor vessel. This finding was of very low risk significance because of the availability of the redundant CRD pump and the relatively short period of time the “A” CRD pump was inoperable.

This finding involved the cross-cutting aspect of PI&R, in that troubleshooting actions were not sufficient to identify the problem that caused the “A” CRD pump to fail to start on several occasions during testing on February 17, 2005. This issue also involved the cross-cutting aspect of human performance in that maintenance and surveillance personnel did not identify that the drive motor did not charge the breaker closing springs, and plant procedures also failed to include appropriate steps to ensure that breaker closing springs charged at the end of surveillance and maintenance activities to confirm the standby readiness configuration of the system. (Section 1R15)

C Green. A self-revealing finding and non-cited violation was identified for failure to comply with 10 CFR 50, Appendix B, Criterion XVI, related to the evaluation of Generic Letter 95-07, “Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves,” in that the “B” train Isolation Condenser condensate return isolation valve was pressure locked and failed to open on October 8, October 12, and again on October 14, 2004, during testing.

This finding was more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (Decay Heat Removal). The specific attributes of design control and equipment performance were adversely impacted for the isolation condenser system which functions to remove post-shutdown decay heat. The finding is of very low safety significance because the redundant train was not similarly affected by the pressure locking condition and remained available, and the pressure locking condition was detected and corrected in sufficient time such that the affected train safety function was lost for less than its Technical Specification allowed outage time. This issue involved the cross-cutting aspect of PI&R, in that the evaluation of Generic Letter 95-07 was insufficient to recognize the susceptibility of the Isolation Condenser System condensate return isolation valves to pressure locking from an at power initiating condition due to thermal binding. (Section 4OA5)

Cornerstone: Public Radiation Safety

C Green. An NRC identified non-cited violation of Technical Specification 6.8.1.a. was identified associated with failure to implement provisions of the radioactive effluent control program specified therein. Specifically, AmerGen did not determine cumulative or projected dose contributions for the current calendar quarter and current calendar year (2004), at least once per 31 calendar days, as required and did not determine, and adjust, the alarm setpoints for the stack and augmented off-gas building radioactive gaseous effluent monitoring instrumentation in accordance with specified methodology and parameters. Further, in April 2004, AmerGen did not take remedial actions to resolve an out-of-specification radioactivity analysis result from its radio-chemistry cross-check analysis laboratory. Lastly, no specific program was identified to ensure use of the gaseous waste treatment system when the projected annual dose could exceed 2 percent of the guidelines of Appendix I to 10 CFR 50.

The failure to implement Technical Specification effluent control requirements is a performance deficiency in that various requirements were not met by AmerGen which were reasonably within its ability to foresee and correct, and which should have been prevented. This finding is greater than minor because failure to implement Technical Specification radioactive effluents controls program requirements affected the cornerstone objective to ensure adequate protection of public health and safety. This finding was evaluated against criteria in NRC Manual Chapter 0609, Appendix D, and determined to be of very low safety significance (Green), in that: 1) it was not a radioactive material control issue, 2) it did involve the effluent release program, 3) there was no impaired ability to assess dose, and 4) public radiation doses did not exceed 10 CFR 50, Appendix I values. This finding also had a cross-cutting aspect of Problem Identification and Resolution in that AmerGen failed to identify this problem during routine self-assessments and audits of the effluent program. (Section 2PS1)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Oyster Creek began the integrated inspection period at full power. A short duration decrease in power to 60 percent rated thermal power (RTP) occurred on March 18, 2005, to adjust the control rod pattern, to conduct temporary leak repair to the main flash tank man-way flange, and to repair the "C" feedwater pump and feedwater control valve. Repairs were completed and the unit was returned to RTP on March 20, 2005, and operated at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events/Mitigating Systems/Barrier Integrity

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

a. Inspection Scope

This activity represented one inspection sample. Due to frequent high winds conditions and winter weather conditions, the inspectors reviewed the licensee's actions in order to verify that safety-related equipment would remain functional. The inspectors reviewed actions taken by the licensee regarding reactor building paneling that dislodged due to high winds on March 8, 2005. In addition, the inspectors conducted walk downs to verify that the safety-related equipment would remain functional during adverse winter weather conditions. The inspectors evaluated the condition of the Emergency Service Water System including heat tracing and support system components, the Service Water System, Electrical Switchyard, emergency diesel generators, station blackout transformers prior and during the onset of winter weather conditions and walkdowns of general grounds for loose debris which could become missiles during high wind conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (IP 71111.04Q and 71111.04S - 5 Samples Total)

a. Inspection Scope

1. Partial System Walkdown (71111.04Q - 4 Samples)

a. Inspection Scope

This activity represented four inspection samples. The inspectors performed four partial system walkdowns during this inspection period. To evaluate the operability of the selected systems, the inspectors checked for a correct valve lineup by comparing positions of valves with system drawings, as well as examining overall system material condition. The results of inspections as well as minor deficient equipment conditions identified by the inspector, were discussed with the appropriate system engineers.

This inspection reviewed the following four systems:

• Containment Spray System #2 during System #1 scheduled maintenance during the week of January 19, 2005

C Control Rod Drive (CRD) System Train “A” during Train “B” scheduled maintenance during the week of January 25, 2005

C Emergency Diesel Generator (EDG) #1 during EDG #2 scheduled maintenance during the week of February 14, 2005

C Core Spray System #2 during Core Spray System #1 scheduled maintenance during the week of February 21, 2005

b. Findings

No findings of significance were identified.

2. Complete System Walkdown (71111.04S - 1 Sample)

a. Inspection Scope

This activity represented one inspection sample. The inspectors conducted a detailed review of the alignment and conditions of the Station Blackout System. The inspectors used the licensee procedures and other documents listed in the Attachment (see listings under 1R04) to verify proper system alignment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (IP 71111.05Q - 11 Samples)

a. Inspection Scope

This activity represented eleven inspection samples. The inspectors walked down accessible portions of the 11 fire zones noted in the Attachment (see listings under 1R05) due to the potential to impact mitigating systems. Plant walkdowns included observations of combustible material control, fire detection and suppression equipment availability, and compensatory measures. As a part of the inspection, the inspectors had discussions with fire protection personnel, and reviewed procedure 333, “Plant Fire Protection System,” OP-AA-201-009, “Control of Transient Combustible Material,” and the Oyster Creek Fire Hazards Analysis Report to verify that the fire program was implemented in accordance with all conditions stated in the facility license.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (IP71111.07 - 1 Sample)

a. Inspection Scope

This activity represented one inspection sample. The inspectors reviewed performance testing results to ensure that the Containment Spray System #1 heat exchangers could perform their design functions as intended. The inspectors also reviewed the licensee's inspection, cleaning and performance monitoring records of the Containment Spray System #1 heat exchangers which are normally in a standby alignment and are cooled by the emergency service water system. The inspectors reviewed associated system corrective action and preventive maintenance records as listed in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

This activity represented one inspection sample. This inspection assessed the Licensed Operator Requalification Training (LORT) provided to the SROs and the ROs and the evaluation conducted on the simulator on January 25, 2005. The inspectors assessed the proficiency of the operating crew and verified that the evaluations of the crew identified and addressed operator performance issues. The inspection activities were performed using NUREG-1021, Rev. 8, "Operator Licensing Examination Standards for Power Reactors," and Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program."

The training included one scenario involving an ATWS with Main Steam Isolation Valve (MSIV) closure, and about three hours of testing/evaluation. The inspectors assessed the simulator crew's performance during the scenario. The inspectors also assessed the evaluator's assessment of the crew, to verify that operator performance issues were identified and appropriate remediation was conducted to address identified weaknesses. The inspector also reviewed CAP Report O2005-0328 that was initiated during this LORT activity for management participation in the crew evaluation.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (IP 71111.12Q and IP 71111.12B - 8 Samples Total)

1. Quarterly Maintenance Rule Implementation (7111112Q - 2 Samples)

a. Inspection Scope

This activity represented two inspection samples. The inspectors reviewed AmerGen's implementation of the maintenance rule as described in Oyster Creek procedure ER-AA-310, "Implementation of the Maintenance Rule." The inspectors verified that the selected Systems, Structures and/or Components (SSCs) were properly classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed Action Requests (ARs), Corrective Action Program reports (CAPs), (a)(1) corrective action plans and routine preventive maintenance activities. The inspectors also discussed the current system performance, associated issues and concerns, and planned activities to improve performance with the system engineers. Plant Health Committee presentations were reviewed. The licensee's Focus Area Self Assessment Report, AR A2104190, was reviewed for information pertaining to the two selected samples. In addition, unavailability data was compared with control room log entries to verify accuracy of data and compliance with (a)(1) goals. AmerGen trending data was also reviewed. The inspectors also reviewed the documents listed in the Attachment. The two SSCs reviewed during the inspection period were:

- SBO Transformer
- 4160 Volt Vital and Non-vital AC System

b. Findings

No findings of significance were identified.

2. Biennial Maintenance Rule Implementation (7111112B - 6 Samples)

a. Inspection Scope

This activity represented six inspection samples. The inspector reviewed the periodic evaluations required by 10 CFR 50.65 (a)(3) to verify adequate consideration was provided for the balancing of reliability and unavailability for structures, systems and components (SSCs) contained within the scope of the maintenance rule. The inspector reviewed the licensee's most recent maintenance rule program periodic evaluation report which covered the period from January 1, 2002 through December 31, 2003.

The inspector reviewed the safety significant systems that were in (a)(1) status to verify that: (1) goals and performance criteria were appropriate, (2) industry operating experience was considered, (3) corrective action plans were effective, and (4) performance was being effectively monitored. As of December 2003, there were eight systems in (a)(1) status; however, seven of the (a)(1) systems were returned to (a)(2) status prior to this inspection. The remaining system in (a)1) status was in monitoring status while corrective actions were developed and/or implemented. The following systems and components were reviewed:

C Core Monitoring System
C Heater Vents, Drains, and Pressure Relief System
C 4160 Volt AC System
C 480 Volt AC System
C Main Steam System
C Isolation Condenser

The inspector reviewed the following (a)(2) systems to confirm that the performance met the applicable maintenance rule performance criteria:

C Containment Isolation System
C Spent Fuel Pool Cooling System

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (IP 71111.13 - 6 Samples)

a. Inspection Scope

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

C "A" Recirculation pump return to service during the week of February 2, 2005

C Offgas sample flow decrease and unplanned entry into 72 hr LCO during the week of February 28, 2005

C "D" recirculation pump step flow increase during return to service testing on February 8, 2005

- C Emergency Diesel Generator #1 inoperability due to the trip of #86 lockout relay during troubleshooting on an area lighting circuit on February 15, 2005

- C Reactor Triple Low Water Level test and calibration surveillance conducted on March 8, 2005

- C Grid undervoltage functional test conducted on March 9, 2005

b. Findings

Introduction. A Green self-revealing finding was identified for an event on February 15, 2005, involving an inadvertent trip of the #1 Emergency Diesel Generator Lockout Relay and temporary loss of the #1 EDG train during troubleshooting repairs to the area lighting system. The finding constituted an NCV for failure to take appropriate corrective actions for prior similar events as prescribed in 10 CFR 50 Appendix B Criterion XVI.

Description. On February 14, 2005, plant operators requested that maintenance technicians investigate the cause of a failure of area lighting in the #1 EDG room. The technicians suspected a blown fuse in the EDG breaker cubicle. After discussing the planned troubleshooting with the shift manager, the technicians proceeded to open the breaker cubicle door to assess the fuse status. When the technicians opened the cubicle door, it struck a fire detector instrument in the overhead and the resultant vibration caused a trip of the EDG differential relays and an actuation of the EDG lockout relay. This made the EDG inoperable at 12:42 am, on February 15. At the time of the event, the No. 2 Standby Gas Treatment System was administratively inoperable due to ongoing post maintenance testing. As a result of the #1 EDG being inoperable and a safety system powered by the alternate train (EDG #2) also being inoperable, the licensee entered a 30 hour shutdown action statement in accordance with Technical Specification 3.7.C.2. The #1 EDG lockout relay was reset at 12:49 am, restoring the #1 EDG to an available condition and allowing plant operators to exit the shutdown action. #1 EDG operability was subsequently verified by surveillance test demonstration at 3:01 am on February 15, 2005.

Based on a review of corrective action program reports, the inspectors found that similar events involving inadvertent tripping of the EDG lockout relay occurred in CY 2001 and CY 2003. Both events identified a problem with relay actuation associated with opening the EDG breaker compartment doors, as well as other 4160 VAC breaker cabinet doors, due to the sensitivity of installed differential relays. Corrective actions were limited to replacing the affected relay on the second occasion since it would not reset properly. However, no corrective actions were developed to ensure that workers would not cause the same event in the future when working in this area.

The work was authorized by the shift manager although no formal pre-job briefing was conducted. This is consistent with Exelon practice on minor maintenance for lighting problems. However, since the workers had to access fuses inside the breaker cubicle to troubleshoot the lighting system problems, the maintenance activity should have been treated as potentially affecting the performance of a safety system component. Had the prior two events identified appropriate corrective actions to ensure that any work conducted inside the EDG breaker cubicle would not make the safety system inoperable, then the work would have only proceeded after properly evaluating the activity from a Maintenance Rule risk perspective. The failure to initiate corrective actions for the prior events for a condition adverse to quality is a performance deficiency and a violation of 10 CFR 50 Appendix B and the Oyster Creek Quality Assurance Plan.

Analysis. In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Disposition Screening," the inspectors determined that the finding was more than minor because it affected the mitigating system cornerstone objective to ensure the availability, reliability, and capability of systems (emergency AC power) that respond to initiating events to prevent undesirable consequences and the related attributes of equipment performance, human performance and procedure quality. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a significance determination process (SDP) Phase 1 screening and determined that this finding screened to Green based upon the loss of a safety function for less than its Technical Specification allowed outage time (AOT). The inspector noted that plant operators responded appropriately to the resultant alarm conditions and restored the EDG to an available condition within 10 minutes. This finding was determined to be of very low safety significance because the redundant mitigating systems equipment remained available, and the condition was promptly identified and corrected.

This finding also has a cross-cutting aspect of PI&R in that corrective actions for similar prior events were not effective at preventing a repeat condition.

Enforcement. 10 CFR 50 Appendix B Criterion XVI requires in part that corrective actions be taken to prevent the recurrence of a significant condition adverse to quality. On two prior occurrences, in 2001 (CAP O2001-0555) and 2003 (CAP O2003-1243), EDG lockout relay actuation resulted from workers opening or closing the EDG breaker cubicle doors, causing the EDG to be inoperable, a significant condition adverse to quality. Neither CAP resulted in actions to prevent recurrence. Contrary to the above, the licensee's failure to take appropriate corrective actions to prevent the recurrence of a significant condition adverse to quality resulted in the February 15, 2005, event when workers inadvertently tripped and made the #1 EDG inoperable upon opening the associated breaker cubicle door. Because this failure to maintain the #1 EDG operable for a brief period is of very low safety significance and the issue was entered into the licensee's corrective action program (CAP O2005-0696), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600. **(NCV 05000219/2005002-01)**

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

a. Inspection Scope

This activity represented one inspection sample. For the non-routine event described below, the inspectors reviewed operator logs, plant computer data, and system procedures to determine how the operators responded, and to determine if the response was in accordance with plant procedures.

On March 11, 2005, the inspectors observed the return to service of the "D" recirculation pump following a four week system overhaul. The return to service of the "D" recirculation pump while at power is an infrequently performed evolution requiring special oversight by operations due to the potential effect on reactor power and reactor water chemistry. The recirculation pump was successfully returned to service with no significant operational affects.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (IP 71111.15 - 5 Samples)

a. Inspection Scope

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

- CAP O2005-0054, Emergency Service Water System #1 flange leakage downstream of containment spray heat exchanger outlet valve, V-3-88.
- CAP O2005-0511, Air extraction offgas radiation monitor sample flow decrease.
- CAP O2005-0132 "A" recirculation pump flow oscillations.
- CAP O2005-0753 "A" CRD pump failure to start during local shutdown panel testing.
- CAP O2005-2060 Structural I beam found corroded in reactor building underground vault.

b. Findings

Introduction. A Green self-revealing finding was identified for failure to adequately restore the "A" CRD pump to standby readiness after testing and maintenance, as prescribed by plant Technical Specification 3.4.D., "Control Rod Drive Hydraulic System."

Description. On March 16, 2005, the "A" CRD pump failed to start during a surveillance test. Subsequent troubleshooting and investigation by the licensee determined that the cause was related to an inadequate evaluation of a previous problem with the TR-2 relay on February 17, 2005. This led to the inoperability of the "A" CRD pump for greater than its TS allowed outage time (7 days).

On February 17, 2005, the "A" CRD pump failed to start on several occasions during a scheduled surveillance and functional test of the 1A2 Local Shutdown Panel. The system was returned to service following troubleshooting maintenance and a subsequent successful start of the pump. The pump was placed in a standby readiness condition while the "B" CRD pump provided normal support for continued power operation. Several weeks later, on March 16, 2005, the "A" CRD pump again failed to start during a regular monthly surveillance test. Troubleshooting revealed that the breaker charging coil (spring) was discharged, resulting in the breaker failure to close on a demand signal.

A review of the time line of activities involving the breaker revealed no causes for the closing coil to have discharged since the pump was placed in standby readiness on February 17. Normally, the closing coil is charged by a drive motor as part of the normal opening cycle of the breaker. The last time that the breaker was cycled open was at the end of the Local Shutdown Panel test on February 17. The conclusion at that time was that dirty contacts in the associated starting circuit had likely cleaned itself resulting in no apparent problem. No additional actions were taken.

During the investigation into the March 16 failure to start, the licensee determined that the intermittent circuit failure that likely caused the failure to start on February 17, also caused the drive motor for the charging coil to malfunction when the breaker was opened. Therefore, the breaker would not have closed at any time after securing the system subsequent to the February surveillance. The inspectors noted that the February 17 troubleshooting action plan did not include a review of the complete circuit to identify other possible causes, but rather focused on the starting circuit relays and contacts. Also, the inspectors noted that the licensee's restoration to service portion of the surveillance procedure did not include a verification that the closing coil spring was charged. Failure to identify the problem with the closing circuit and the lack of procedural requirements to verify the breaker was properly restored to a standby readiness condition resulted in the subsequent March 16 event. The licensee found a loose connection in the circuit and replaced an associated relay. Subsequent testing demonstrated operability of the CRD pump controls.

<u>Analysis</u>. In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Disposition Screening," the inspectors determined that this finding was more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The specific attributes of equipment performance, human performance, and procedure quality were adversely impacted for the CRD system which functions as a high pressure injection makeup source for decay heat removal for transient event sequences. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a Significance Determination Process (SDP) Phase 1 screening and determined that this finding required a Phase 2 approximation based upon the loss of a safety function of a single train for greater than its Technical Specification allowed outage time (AOT). The inspectors conducted a Phase 2 SDP analysis and made the following assumptions:

- based upon the inoperability of the A CRD pump from between February 17 and March 16, the exposure time for determining the Initiating Event Likelihood was between 3 and 30 days; and,
- a revised mitigation credit of one (typically two) was assigned to the control rod pumps (CRD) based upon the B CRD pump remaining available for the duration of the degraded condition of the A CRD pump.

Solving the affected Phase 2 SDP worksheets, the inspector determined that this finding was of very low risk significance (Green). The most dominant core damage sequences involved the transients without power conversion system (TPCS) and the failure of make-up to the isolation condensers and either failure of the low pressure injection system or failure to depressurize the reactor vessel. This finding was of very low risk significance because of the availability of the redundant CRD pump and the relatively short period of time the A CRD pump was inoperable.

This issue also involved the cross-cutting aspect of PI&R, in that troubleshooting actions were not sufficient to identify the problem that caused the "A" CRD pump to fail to start on several occasions during testing on February 17.

This issue also revealed a lack of sufficient detail in plant procedures, and insufficient questioning on the part of maintenance and surveillance personnel, to ensure that the breaker closing spring was charged as expected at the end of the surveillance and maintenance activity in February. This procedure inadequacy led to a cross-cutting aspect of human performance since the equipment was not restored to a standby state of readiness.

<u>Enforcement</u>. Technical Specification 3.4.D. states in part, that if one CRD pump becomes inoperable when the reactor water temperature is above 212 degrees F, the reactor may remain in operation for a period to not exceed 7 days provided the second CRD pump is operating and checked at least once every eight hours, or the reactor water temperature shall be reduced to less than 212 degrees F. Contrary to the above, the licensee's failure to identify a problem with the "A" CRD control circuit on

February 17 led to the breaker closing spring not charging properly upon completion of maintenance. This resulted in the CRD pump being inoperable for the period February 17 through March 16, in excess of the 7 day allowed outage time. Because this failure to maintain the CRD pump operable is of very low safety significance and has been entered into the licensee's corrective action program (CAP O2005-0753 and O2005-1174), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600. **(NCV 05000219/2005002-02)**

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

a. Inspection Scope

This activity represented one inspection sample. The inspectors reviewed the operator work-around database and to identify conditions that could adversely affect the operability of mitigating systems, or impact human reliability in responding to initiating events. The inspector reviewed the licensee's implementation of procedure OP-AA-102-103, "Operator Work-Around Program." The licensee's database revealed no significant operator work-arounds or challenges. Based on an aggregate review of the database, the inspectors determined that the licensee has taken appropriate corrective and compensatory measures to minimize the effect of these conditions.

The inspector also discussed several important changes that the licensee was planning for this program. A benchmark study of this program from another Exelon plant was completed, resulting in planned changes to the implementing procedure. Based on discussions with Operations management, it is anticipated that the procedure changes will result in improvements to the methods employed at Oyster creek to identify, monitor and correct operator work-arounds.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (IP 71111.19 - 7 Samples)

a. Inspection Scope

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

C 607.4.016, “Containment Spray and Emergency Service Water System 1 pump Operability and Quarterly Inservice Test, following system maintenance during week of January 21, 2005

C 617.4.001, “B” CRD pump post maintenance test after system maintenance and replacement of the pump gearbox, during the week of January 23, 2005

C “D” Recirculation pump post maintenance testing after system planned maintenance and refurbishment of motor generator set during the week of March 11, 2005

C 645.4.036, #1-2 Fire Diesel pump post maintenance testing following scheduled maintenance during the week of February 10, 2005

C 610.3.115, Core Spray System #1 Instrument Channel and Level Bistable Test following scheduled maintenance on February 25, 2005

C 24 V DC battery charger refurbishment post maintenance test in accordance with work order R205655105 conducted on March 19, 2005

C “A” CRD pump post maintenance test following replacement of TR-2 relay conducted on March 17, 2005

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (IP 71111.22 - 7 Samples)

a. Inspection Scope

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

• 609.4.001, “Isolation Condenser Valve Operability and In-Service Test,” conducted during week of January 10, 2005

C 609.3.113, “Isolation Condenser Auto-actuation Bistable and Calibration,” during the week of January 10, 2005

- C 604.4.016, “Torus to Drywell Vacuum Breaker Operability and In-Service Test,” during the week of January 31, 2005

- C 636.4.003, “Emergency Diesel Generator #1 load test,” during the week of February 7, 2005

- C 636.4.013, “Emergency Diesel Generator No. 2 Load Test,” conducted on February 14, 2005

- C 619.3.006, “Reactor Triple Low Water Level Test and Calibration,” conducted on March 8, 2005

- C 632.2.002, “Grid undervoltage channel functional test,” conducted on March 9, 2005

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

This activity represented one inspection sample. The inspectors reviewed a Temporary Modification (TM) associated with the station air system piping on the reactor building 23' elevation. The inspectors reviewed the associated implementing documents to verify the plant design basis and the system or component operability was maintained, which included CC-AA-112, “Temporary Configuration Changes.” The TM allowed for continued operability of the scram discharge instrument volume. The inspectors verified that the temporary modification was in accordance with station procedures.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP4 Emergency Action Level (EAL) and Emergency Plan (E-Plan) Changes

a. Inspection Scope (IP 7111404)

During the period of January 11 through March 31, 2005, the NRC received and acknowledged the changes made to Oyster Creek’s E-Plan in accordance with 10 CFR 50.54(q), which Exelon Nuclear had determined resulted in no decrease in effectiveness to the Plan and continue to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR 50. The inspector conducted a sampling review of the Plan changes which could potentially result in a decrease in effectiveness. This review does not constitute an approval of the changes and, as such, the changes are subject to future NRC inspection. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (IP 71114.06 - 1 Sample)

a. Inspection Scope

This activity represented one inspection sample. The inspectors observed an emergency preparedness (EP) drill from the control room simulator and the technical support center on January 19, 2005. The inspectors evaluated the conduct of the drill and AmerGen's performance related to emergency action level classifications, notifications, and protective action recommendations. The drill contained four opportunities that contributed to the NRC Drill/Exercise Performance (DEP) performance indicator.

The inspectors reviewed the following documents:

- Oyster Creek EP Drill January 19, 2005, Scenario 58, Station Blackout Event, Rev. 0
- CAP O2005-0330, EP areas for improvement identified during the drill

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety [PS]

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

a. Inspection Scope

Inspection Planning and In-office Inspection (1 sample)

This activity represented one inspection sample. The inspector reviewed the 2004 Radiological Effluent Release Report, dated March 1, 2005, to verify that the program was implemented as described in the Radiological Effluents Technical Specifications

(RETS) and the Offsite Dose Calculation Manual (ODCM). The inspector reviewed the report for significant changes to the ODCM and to radioactive waste system design and operation. The inspector determined whether changes to the ODCM were technically justified and documented, as appropriate. The technical justifications, as appropriate, were reviewed during the onsite inspection.

The inspector determined whether modifications made to radioactive waste system design and operation, as applicable, changed the dose consequence to the public. The inspector verified, as appropriate, that technical and/or 10 CFR 50.59 reviews were performed. The inspector determined whether radioactive liquid and gaseous effluent radiation monitor setpoint calculation methodology changed since completion of the modifications and that AmerGen had set and adjusted its radioactive effluent alarm setpoints in accordance with the methodology and parameters specified within the current ODCM.

The inspector determined if anomalous results, reported in the current Radiological Effluent Release Report, were adequately resolved. The inspector also reviewed AmerGen’s actions to revolve out-of-specification interlaboratory cross-check analysis data for the effluent monitoring program and to determine if remedial action had been taken for the out-of-specification data.

The inspector reviewed the RETS/ODCM to identify the effluent radiation monitoring systems and applicable flow measurement devices. The inspector reviewed any effluent radiological occurrence performance indicator incidents for onsite follow-up and reviewed licensee self-assessments, audits, and event reports that involved unanticipated offsite releases of radioactive material, as appropriate. (See also Section 4OA2)

The inspector reviewed the Updated Final Safety Analysis Report (UFSAR) description of all radioactive effluent monitoring and waste systems, as appropriate.

<u>Onsite Inspection (8 samples)</u>

This activity represented eight inspection samples. The inspector walked-down selected components of the gaseous release systems (e.g., radiation and flow monitors) to observe current system configuration with respect to the description in the Final Safety Analysis Report (FSAR), ongoing activities, and equipment material condition. (This sample activity is not complete.)

The inspector confirmed that radioactive liquid waste was not released from Oyster Creek as described within the 2004 RETS/ODCM report. The inspector observed routine sample collections from the stack and the augmented off-gas building particulate and iodine samplers and conformance with procedures for these activities. The inspector reviewed use of radioactive gaseous effluent treatment equipment in accordance with RETS/ODCM requirements, as applicable, and reviewed use of system per ODCM guidance.

The inspector reviewed, as appropriate records of releases made with out-of-service effluent radiation monitors, and AmerGen's actions for these releases, to ensure an adequate defense-in-depth was maintained against an unmonitored, unanticipated release of radioactive material to the environment. The inspector determined, where appropriate, compensatory sampling and radiological analyses conducted, at the RETS/ODCM required frequency, when effluent monitors were declared out-of-service. The inspector also determined if AmerGen placed information on leaks or spills into its 10 CFR 50.75(g) decommissioning file, as appropriate.

The inspector reviewed changes made to the ODCM as well as to the liquid or gaseous radioactive waste system design, procedures, or operation since the last inspection. For each system modification and each ODCM revision that impacted effluent monitoring or release controls, the inspector reviewed AmerGen's technical justification to determine whether the changes affected AmerGen's ability to maintain effluents ALARA and whether changes made to monitoring instrumentation resulted in a non-representative monitoring of effluents. For significant changes to dose values reported in the Radiological Effluent Release Report from the previous report (2003 versus 2004), the inspector evaluated, where applicable, the factors which may have resulted in the change. If the change was not influenced by an operational issue (e.g., fuel integrity, extended outage, or major decontamination efforts), the inspector independently assessed selected licensee's offsite dose calculations by using the NRC PC-DOSE computer code or by reviewing the verification and validation records for the licensee's dose calculation.

The inspector reviewed a selection of 2004 monthly, quarterly, and annual dose calculations to ensure that AmerGen properly calculated the offsite dose (both cumulative and projected) from radiological effluent releases and to determine if any annual TS/ODCM (i.e., Appendix I to 10 CFR Part 50 values) were exceeded and, if appropriate, issued a PI report if any quarterly values were exceeded.

The inspector reviewed air cleaning system surveillance test results (standby gas treatment system, new radwaste building, offgas building) to ensure that the system was operating within the applicable acceptance criteria. The inspector reviewed surveillance test results and/or the methodology AmerGen used to determine the stack and vent flow rates. The inspector verified that the flow rates are consistent with RETS/ODCM or FSAR values.

The inspector reviewed records of instrument calibrations performed since the last inspection for each point of discharge effluent radiation monitor and flow measurement device; reviewed any completed system modifications; and reviewed the current effluent radiation monitor alarm setpoint value for agreement with RETS/ODCM requirements.

The inspector reviewed calibration records of radiation measurement (i.e., counting room) instrumentation associated with effluent monitoring and release activities. The inspector reviewed quality control records for the radiation measurement instruments and looked for indications of degraded instrument performance and the corrective actions taken.

The inspector reviewed the results of the interlaboratory comparison program to verify the quality of radioactive effluent sample analyses performed by AmerGen. The inspector reviewed AmerGen's quality control evaluation of the interlaboratory comparison test and associated corrective actions for any deficiencies identified. The inspector also, as applicable, reviewed AmerGen's assessment of any identified bias in the sample analysis results and the overall effect on calculated projected doses to members of the public.

The inspector reviewed the results from AmerGen's QA audits to determine whether it met the requirements of the RETS/ODCM.

b. Findings

Introduction. A Green non-cited violation of Technical Specification 6.8.4.a. was identified by the NRC associated with failure to meet provisions of the radioactive effluent control program specified therein. Specifically, AmerGen did not determine cumulative or projected dose contributions for the current calendar quarter and current calendar year (2004), at least once per 31 calendar days, as required. In addition, AmerGen did not determine, and adjust, the alarm setpoints for the stack and augmented off-gas building radioactive gaseous effluent monitoring instrumentation, in accordance with ODCM specified methodology and parameters. Further, AmerGen did not take, in April 2004, remedial actions to resolve out-of-specification radioactivity analysis results from its radiochemistry cross-check analysis laboratory. Lastly, no specific program was identified to ensure use of the gaseous waste treatment system when the projected annual dose could exceed 2 percent of Appendix I to 10 CFR 50.

Description. Technical Specification, Section 6.8.4.a, describes the program to provide for the control of radioactive effluent and for maintaining the dose to members of the public from radioactive effluent as low as reasonably achievable. The program is to be contained in the Offsite Dose Calculation Manual (ODCM), is to be implemented by operating procedures, and is to include remedial actions whenever program limits are exceeded. In particular, the Technical Specification provides that gaseous radioactive monitor alarm setpoint determination be made in accordance with the methodology of the ODCM; cumulative and projected dose contributions from radioactive gaseous effluents be determined, at least every 31 days; gaseous waste treatment systems are to be used if the projected dose would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR50.

On March 11, 2005, the inspector identified that the current alarm setpoints for both the stack and the augmented off-gas building exhaust gaseous effluent monitoring instrumentation were not determined in accordance with the methodology and parameters in the current ODCM. The inspector also identified on March 11, 2005, that AmerGen was not determining, at least once every 31 days, both the cumulative and projected dose contributions from radioactive effluents, for the current calendar quarter and the current year (2004), as required by the ODCM. The inspector also identified on March 11, 2005, that AmerGen had supplied a cross-check sample to its vendor laboratory for second quarter 2003 to cross-check that laboratory's capability to quantify a Sr-89 sample. AmerGen subsequently received the sample results, dated April 30, 2004, which indicated a disagreement in analysis results which was outside acceptable limitations. AmerGen did not take remedial actions to evaluate the out-of-specification sample. The inspector also identified on March 11, 2005, that no specific program procedure was identified to ensure use of the gaseous waste treatment system when projected doses, in the 31 day period, would exceed 2% of the annual dose or dose commitment conforming to Appendix I of 10 CFR 50.

Analysis. Failure to implement Technical Specifications requirements for effluent monitoring and control, as specified in the ODCM, is a performance deficiency in that procedure requirements for control of radioactive effluents were not met which were reasonably within AmerGen's ability to foresee and correct, and which should have been prevented.

The finding was greater than minor because failure to implement the Technical Specification described effluent control program affected the cornerstone objective to ensure adequate protection of public health and safety. Specifically, multiple program elements specified in Technical Specification 6.8.4 a, including conduct of specified dose assessments and alarm setpoint determinations, were not met.

Using NRC Manual Chapter 609, Appendix D, this finding was determined to be of very low safety significance (Green), in that: 1) it was not a radioactive material control issue, 2) it did involve the effluent release program, 3) there was no impaired ability to assess dose, and 4) public radiation doses did not exceed 10 CFR 50, Appendix I values. In addition, the inspector determined that, although the alarm setpoints for the stack and augmented off-gas building were not determined per the ODCM, the alarm setpoints were found to be conservatively set and below the calculated values determined by use of the current ODCM methodology. The inspector also determined that although AmerGen was not performing cumulative quarterly and annual dose contributions at least every 31 days, AmerGen was determining cumulative dose contributions for gaseous effluents on a quarterly basis and was determining annual dose contributions semi-annually and at the end of the year. The results of these dose assessments identified public dose values well below ODCM guidelines. Further, AmerGen calculated cumulative quarterly and annual dose assessments for each 31 day period for 2004 and did not identify any dose discrepancies. The inspector also determined that AmerGen routinely used its waste gas treatment system to maintain doses to the public as low as reasonably achievable. In addition, AmerGen verified secondary laboratory cross-check data (for the periods in question) indicated acceptable results for cross-checks. Also, AmerGen indicated conservatively increasing effluent releases would not result in any significant dose increases.

The above examples of failure to implement the Technical Specification effluent control provisions also reflect a cross-cutting issue in the area of problem identification and resolution. Specifically, AmerGen's focused area self-assessment of conformance with the ODCM, conducted in February 2005, did not identify any of the above issues. AmerGen placed this issue into its corrective action program (CAP-2005-1113).

Enforcement. Technical Specification 6.8.4.a describes the program to provide for the control of radioactive effluent and for maintaining the dose to members of the public from radioactive effluent as low as reasonably achievable. The program is to be contained in the Offsite Dose Calculation Manual (ODCM), is to be implemented by operating procedures, and is to include remedial actions whenever program limits are exceeded. In particular, the Technical Specification provides that gaseous radioactive monitor alarm setpoint determination be made in accordance with the methodology of the ODCM; cumulative and projected dose contributions from radioactive gaseous effluents be determined at least every 31 days; and the gaseous waste system be used if the projected dose to the public would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR50.

Contrary to Technical Specification 6.8.4.a, AmerGen did not determine cumulative and projected dose contributions, for the current calendar quarter and current calendar year (2004), at least once per 31 calendar days as required. In addition, AmerGen did not determine and adjust the alarm setpoints for the stack and augmented off-gas building radioactive gaseous effluent monitoring instrumentation, in accordance with current ODCM specified methodology and parameters. Further, AmerGen did not take, in April 2004, remedial actions to resolve an out-of-specification radioactivity analysis result from its radiochemistry cross-check analysis laboratory that was outside acceptance limits. Lastly, no specific program procedure was identified to ensure use of gaseous waste treatment system when the projected annual dose could exceed 2 percent of Appendix I to 10 CFR50.

This is a violation of TS 6.8.4. a. Because this finding was of very low safety significance (Green), and Exelon entered this finding into its corrective action program (AR 271404), this violation is being treated as a Non-Cited Violation (NCV) consistent with the NRC Enforcement Policy. **(NCV 05000219/2005002-03)**

In response to the above finding, AmerGen placed these issues into its corrective action program by issuance of several corrective action documents. (CAPS 2005-1109, 1113, 1118, 1269)

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution (IP 71152)

1. Routine Resident Review of CAP Documents

a. Inspection Scope

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

Enclosure

b. Findings

No findings of significance were identified.

2. Annual Sample Review (IP 71152 - 1 Sample)

a. Inspection Scope

This activity represented one inspection sample. The inspectors selected one CAP for detailed review. CAP O2004-3640 is associated with reactor building closed cooling water piping observed to be rusty during a drywell walkdown by NRC inspectors during the refueling outage in November 2004. The inspectors selected this sample due to observation of conditions and recent experience at the Nine Mile Point Nuclear Station. The inspectors evaluated the effectiveness of CAP O2004-3640 and work order A2101106 generated by the licensee. In addition, the inspectors evaluated the licensee's extent of condition review, classification and prioritization of the resolution of the problem commensurate with its safety significance and identification of corrective actions.

b. Finding and Observations

No findings of significance were identified. The inspectors identified through a review of CAP O2004-3640 that the licensee's extent of condition review was not thorough. The information in the CAP only addressed one section of piping on the RBCCW system. The corrective actions taken included polishing that section of pipe and performing a UT to verify piping wall thickness. The UT results indicated that the wall thickness was adequate; therefore no further action was required. A second UT is presently scheduled for the next refueling outage in order to determine a rate of corrosion on this section of piping. However, the inspector identified that, although the wall thickness of this section of piping was adequate, the operating experience indicated that other sections of the piping are of more concern. These sections of piping include threaded connections that exist on recirculation pump seal cooling, drywell equipment drain tank heat exchanger connections, and drywell cooling fan connections. The CAP did not require an extent of condition review to verify the wall thickness in these other areas of the piping. The inspectors identified that this represented a missed opportunity to gather initial baseline data on RBCCW piping inside the drywell during the fall outage in order to support present engineering assumptions that corrosion in other areas is limited by the corrosion inhibitor used at Oyster Creek.

The inspectors did identify that the operational experience database at Oyster Creek was difficult to review. The inconsistent documentation of the Oyster Creek responses to industry issues could potentially result in problems not being addressed adequately when system responsibilities are taken over by new engineering staff or staff not familiar with the particular history and background of the system.

3. Identification and Resolution of Problems - Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (IP 71122.01)

a. Inspection Scope

The inspector reviewed AmerGen's self assessments, audits, Licensee Event Reports, and Special Reports, as applicable, related to the radioactive effluent treatment and monitoring program since the last inspection to determine if identified problems are entered into the corrective action program for resolution. The inspector interviewed staff and reviewed documents to determine if follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety. For repetitive deficiencies or significant individual deficiencies in problem identification and resolution identified, the inspector determined if AmerGen's self-assessment activities were also identifying and addressing these deficiencies.

The inspector reviewed a selection of corrective action documents since the previous inspection:

- CAPs - 2003-2201, 2003-2203, 2003-2204; 2004-0147, 2004-0673, 2004-0702, 2004-0958, 2004-1127, 2004-1277, 2004-1625, 2004-1908, 2004-1917, 2004-2401, and 2004-2689

- NOS Audit OSA-OYS-03-08, NOS REMP, Non-Radiological Effluent Monitoring, NPDES Audit Report, dated October 29, 2003

- Focused Area Self-Assessment- Radiological Control Program and Offsite Dose Calculation Manual, dated February 17, 2005

b. Findings

No findings of significance were identified.

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

Section 1R13 of the report describes a finding for inadequate corrective actions in that the licensee had identified a problem with actuation of the EDG differential and generator lockout relays during access to the EDG breaker cubicle, but had not taken corrective actions to prevent recurrence or provide appropriate measures to control access to ensure that the EDG remained operable. Consequently, the EDG differential and lockout relay problems were not corrected and the #1 EDG tripped on February 14, 2005, during an area lighting troubleshooting maintenance activity.

Section 1R15 of the report describes a finding where trouble shooting actions were not sufficient to appropriately identify the problem that caused the “A” CRD pump to fail to start on several occasions during testing on February 17, 2005. Consequently, the “A” CRD pump failed to start again on March 16, 2005, because the initial problem with the

TR-2 relay was not identified or corrected by maintenance and surveillance personnel involved in the activity.

Section 2PS1 of the report describes a finding involving failure to adhere to Technical Specification 6.8.4.a, and the Offsite Dose Calculation Manual. Since the licensee failed to identify these deficiencies during self-assessments and periodic audits, the finding is indicative of a potential deficiency in the licensee's corrective action program regarding the Offsite Dose Calculation Manual program.

Section 4OA5 of the report describes a finding involving an inadequate evaluation of Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves." Consequently, pressure locking of the isolation condenser condensate return isolation valve from an at power condition was not fully evaluated and corrected.

4OA3 Event Follow-up (IP 71153)

a. Inspection Scope

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

1. (Closed) LER 05000219/2004-005, Rev.0. Main Steam Isolation Valve (MSIV) Failed to Close During Partial Valve Closure Surveillance Due to Mechanical Binding

On September 11, 2004, the outboard MSIV, valve NS04, failed the 10% closure test. Power was reduced to 40% rated thermal power and a full closure test was conducted in which the valve also failed to close. The failure resulted in a plant shutdown on September 14, 2004, to troubleshoot and repair the MSIV. Inspection of the valve internals revealed that the failure of the main disk (poppet) to close was caused by excessive rib and poppet guide wear resulting from poppet vibration induced by steam flow. The inspectors reviewed this event in Section 1R15 of Inspection Report 05000219/2004-04. At that time, a green finding was issued for a failure to timely implement corrective actions for a known deficient condition associated with the valve poppet and its susceptibility to vibration induced wear. The inspectors reviewed this LER and determined that it adequately described the event and associated corrective actions. The licensee documented the failed equipment in CAP O2004-2499. This LER is closed.

2. (Closed) Special Report 05000219/04-01, Rev.0. Reactor Start-up with the Rod Worth Minimizer Inoperable

   This special report was submitted in accordance with Technical Specification 3.2.B.2.(b). On November 21, 2004, the Oyster Creek reactor was made critical as part of the start-up sequence following completion of Refueling Outage 1R20. At the time of the start-up, the Rod Worth Minimizer was inoperable due to a failure in registering proper rod position indication for two control rods (30-19 and 34-23). Technical Specification 3.2.B.2.b was entered which allowed the start-up to continue provided that a second licensed operator verified that the reactor operator was following the rod program and that a reactor engineer also verified that the rod program was being followed. The inspector was in the control room at the time of the start-up and verified that the licensee adhered to the technical specification requirements. The licensee documented the failed equipment in CAP O2004-3977 and Work Order C2009175. The Special Report was reviewed by the inspectors and no findings of significance were identified. This Special Report is closed.

3. (Closed) LER 05000219/2004-006, Rev. 0. Local Leak Rate Test Results in Excess of Technical Specifications

   On November 5, 2004, the as-found local leak rate test of Main Steam Isolation Valve (MSIV) NS04A failed to meet the Technical Specification 4.5.D.2 acceptance criteria of 11.9 SCFH. The actual measured leak rate was 24.3 SCFH. The valve was successfully repaired and tested during the 1R20 refueling outage. The cause of the failure was minor seat mating surface irregularities that resulted from a decision to not lap the seating surfaces during an overhaul of the valve in September 2004. The work in September was deemed acceptable at the time because the valve met all the technical specification performance criteria, including local leak rate test and valve stroke acceptance criteria. Based on the subsequent failure to meet local leak rate acceptance criteria, the licensee committed to revising the MSIV overhaul procedure to include a management review prior to eliminating seat lapping steps. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the failed equipment in CAP O2004-3442. This LER is closed.

4. (Closed) Special Report 05000219/04-02, Rev. 0. Electromatic Relief Valve (EMRV) Lift during NSSS Leak Test

   This special report was submitted in accordance with Technical Specification 6.9.3.I. On November 19, 2004, the “C” EMRV lifted during the performance of the NSSS leak test. At the time control rod drive water system was maintaining reactor pressure in a band of 1055 to 1065 psig. The “C” EMRV lift setpoint was found to be at 1081 psig. As a result, there was very little margin between the test pressure and the setpoint for the “C” EMRV, which caused the valve to lift. The licensee determined that there was no damage to the valve or its associated piping; the pressure change did not result in a brittle fracture prevention limit curve violation; and, adequate core cooling was maintained throughout the event. The licensee determined that the procedure pressure test band was unnecessarily restrictive. As a result, the procedure was changed to restore a more appropriate margin between the test conditions and the setpoint for the EMRV. The special report was reviewed by the inspectors and no findings of significance were identified. This special report is closed.

5. (Closed) LER 05000219/2004-007, Rev.0. Automatic Containment Isolation Bypassed During Reactor Start-up Due to an Inadequate Procedure

On November 22, 2004, during a reactor start-up following completion of refueling outage 1R20, a reactor operator bypassed the automatic containment ventilation and purge isolation function due to an incorrect step in the plant start-up procedure. The inspectors reviewed this event in Section 4OA3 of Inspection Report 05000219/2004-05. At that time, a green finding was issued for a failure to establish and maintain appropriate procedure controls. The inspectors reviewed this LER and determined that it adequately described the event and associated corrective actions. The licensee documented the failed equipment in CAP O2004-4012. This LER is closed.

6. Lower Cable Spreading Room Smoke Alarm due to Failed Lighting Ballast

a. Inspection Scope

The inspectors observed the control room response to a station fire alarm declared on March 1, 2005. The inspectors arrived in the control room shortly after the station sounded the fire alarm and announcement of a fire in the lower cable spreading room. The inspectors observed the licensed operator actions and compared the response to abnormal operating procedures requirements. The following documents were reviewed and used as criteria for evaluating the operator's response to this event.

- ABN-29, "Plant Fires"
- CAP O2005-0944 smoke detected in lower cable spreading room
- Prompt investigation of lower cable spreading room smoke

b. Findings

No findings of significance were identified.

7. Reactor Building 51' Elevation Deluge System Actuation

a. Inspection Scope

The inspectors observed operator and station actions in response to an unexpected fire protection system deluge actuation on the 51' elevation. The inspectors observed operator response in comparison to abnormal procedural requirements. As part of the followup on this event, the inspectors reviewed prompt investigations, corrective actions taken, and station evaluation of potential collateral damage review of safety-related equipment inside the reactor building in order to evaluate station response. The documents reviewed are listed below.

- CAP O2005-0951, "Actuation of reactor building 51' north side deluge system"
- Oyster Creek station prompt investigation report on 51' north side deluge system actuation
- ABN-29, "Plant Fires"
- Apparent cause evaluation for reactor building 51' deluge system actuation

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings Other than PI&R Documented Elsewhere

Section 1R15 of the report describes a finding involving the "A" CRD pump to fail to start on March 16, 2005. This finding had a cross-cutting aspect of human performance in that surveillance and maintenance procedures did not appropriately include steps to ensure that breaker closing springs were charged when restoring the component to a standby readiness configuration and there was insufficient questioning on the part of maintenance and surveillance personnel involved in the activity.

4OA5 Other

(Closed) URI 05000219/2004005-01: Pressure Locking of Isolation Condenser Valve, V-14-35

Introduction. A self-revealing Green NCV was identified for failure to comply with 10 CFR 50, Appendix B, Criterion XVI, related to the evaluation of Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves."

Description. A finding for an event involving a failure to open of the "B" Train Isolation Condenser condensate return isolation valve (V-14-35) during maintenance testing on October 8 and October 12, and again on October 14, 2004, during accelerated surveillance testing as part of an operability evaluation, resulted in identifying a finding and an apparent violation. The finding involved the failure to properly evaluate Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves," and implement appropriate corrective actions to ensure that safety related power operated gate valves susceptible to pressure locking are capable of performing their safety functions within the current licensing bases of the facility as prescribed by 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions.

Additional descriptive information is provided in NRC Inspection Report 05000219/2004-005. In that report, the inspectors noted that the pressure locking condition was previously evaluated as part of an evaluation per NRC Generic Letter 95-07. That evaluation failed to include a pressure lock condition for the Isolation Condenser condensate return isolation valves that resulted from thermally induced pressure lock by heating of the valve bonnet entrapped water. However, the inspectors could not complete a review at that time without additional information from the licensee to determine: (1) the possible fault exposure time for the inoperable valve; (2) the use of manual operation of the valve handwheel and operator recovery credit for the IC function; (3) the possible affect on the redundant component, valve V-14-34; and (4) an understanding of the extent of condition for the inadequate evaluation per Generic Letter 95-07.

The licensee provided additional information to the inspectors to address these concerns. As a result, it was determined that: (1) the fault exposure time and total system unavailability was limited to about 6 days during October 2004; (2) the use of the valve manual handwheel was employed by the technicians when they encountered the motor stall condition, and further, that appropriate procedure guidance and access to the valve was available to plant operators if this condition had occurred under normal or emergency conditions; (3) that while the design of the redundant valve (V-14-34) was identical to the pressure locked valve, sufficient operational leakage was present on that valve to prevent a pressure lock condition from initiating; and, (4) the pressure lock condition would not likely correct itself once initiated. This latter determination helped reveal that the extent of condition for this issue was limited to the period in October 2004 when initially identified. Since the pressure lock would not self relieve, and since no prior testing had identified a pressure locked condition on either IC system train, the actual pressure lock condition was limited to the period in October 2004 after a packing adjustment had been made to valve V-14-35.

Analysis. In accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Disposition Screening," the inspectors determined that this finding was more than minor because it affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The specific attributes of design control and equipment performance were adversely impacted for the isolation condenser system which functions to remove post-shutdown decay heat. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors conducted a significance determination process (SDP) Phase 1 screening and determined that this finding screened to Green based upon the loss of a safety function for less than its Technical Specification allowed outage time (AOT). Inspectors examined the identified pressure locking condition and the licensee's valve stroke test results, and concluded that the longest single period of valve inoperability due to pressure locking was 66.9 hours, with a total train unavailability of 138.6 hours (~5.7 days) due to the pressure locking condition. The Technical Specification AOT for a single isolation condenser out of service is seven days. This finding was determined to be of very low safety significance because the redundant mitigating systems equipment was fully available and the condition was identified and corrected in a timely manner.

The Region I Senior Reactor Analyst (SRA) performed some additional risk assessments to ensure a more thorough understanding of the risk consequence of the inspection finding and conferred with the licensee's risk assessment staff to further validate the risk significance of the finding. The SRA's confirmatory risk evaluation concluded that the finding was of very low risk significance (Green), but that some additional risk consequence was attributable to external events (fire), consistent with the licensee's Individual Plant Examination of External Events (IPEEE).

Enforcement: Generic Letter 95-07 required an evaluation in order to implement appropriate corrective actions to ensure that safety related power operated gate valves susceptible to pressure locking are capable of performing their safety functions within the current licensing bases of the facility as prescribed by 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions. Contrary to this requirement, AmerGen did not conduct a reasonable assessment of thermally induced pressure locking for Isolation Condenser System condensate return isolation valves and implement corrective actions to prevent pressure locking from an at power condition. This is a violation of 19 CFR 50 Appendix B, Criterion XVI. Because this finding was of very low safety significance and AmerGen entered the finding into its corrective action program (CAP O2004-2947, O2004-2986), this violation is being treated as a Non-Cited Violation (NCV) consistent with the NRC Enforcement Policy. **(NCV 05000219/200502-04)**

This issue also had a cross-cutting aspect of PI&R, in that the evaluation of the Generic Letter with respect to the IC system condensate return valves was weak and led to ineffective measures being taken to prevent pressure locking during at power conditions.

4OA6 Meetings, including Exit

Exit Meeting Summary

On April 8, 2005, the resident inspectors presented the inspection results to Mr. C. N. Swenson and other members of licensee management. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- TS 6.8.1 requires written procedures to be followed for surveillance and test activities of equipment that affects nuclear safety. The fire pumps affect nuclear safety in that they are relied upon as an alternate makeup source of water for the ECCS systems as listed in TS 3.4.F. Contrary to this, on February 10, 2005, the #2 Fire Pump governor overspeed trip reset button was erroneously tripped during the conduct of maintenance and surveillance activities and not corrected during train restoration to service. This was identified in the licensee's corrective action program as CAP O2005-0581. This finding is of very low safety significance because the condition was identified during post-maintenance operability testing later that same day and was immediately corrected, and the redundant fire pumps and core spray loops remained fully available.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

R. Artz, Chemistry Supervisor
B. Barbieri, System Engineering
P. Bloss, BOP Systems Manager
J. Booty, System Engineering
M. Button, Director, Maintenance
C. Connelly, Radiation Protection/Chemistry Manager
J. Derby, Radiological Engineer
R. Detwiler, Director, Operations
R. Ewart, Security Manager
D. Fawcett, Licensing Engineer
M. Filippone, System Engineering
J. Freeman, Shift Operations, Superintendent
M. Godknecht, Maintenance Rule Coordinator
S. Hutchins, Electrical Systems Manager
E. Johnson, System Engineer
A. Judson, Radiological Engineer
J. Kandasamy, Manager, Regulatory Assurance
R. Larzo, Engineering
J. Magee, Director, Engineering
D. McMillan, Director, Training
B. Mussel, System Engineering
L. Newton, Chemistry Manager
J. O'Rourke, Assistant Engineering Director
T. Powell, Engineering Programs Manager
J. Randich, Plant Manager
J. Renda, Radiation Protection Manager
G. Seals, Radiological Engineer
H. Shoap, Normandeau Associates
C. Swenson, Site Vice President
D. Weible, Environmental Chemist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000219/2005002-01	NCV	Ineffective corrective actions leading to the #1 EDG being inoperable on February 15, 2005. (Section 1R13)
05000219/2005002-02	NCV	Ineffective corrective actions leading to the “A” CRD pump being inoperable on February 17, 2005. (Section 1R15)
05000219/2005002-03	NCV	Failure to implement ODCM requirements for radioactive gaseous and liquid effluent monitoring. (Section 2PS1)
05000219/2005002-04	NCV	Ineffective corrective actions leading to the “B” IC system being inoperable due to pressure locking in October 2004. (Section 4OA5)

Closed

05000219/2004005-00	LER	(Closed) LER 05000219/2004-005, Rev.0. Main Steam Isolation Valve (MSIV) Failed to Close During Partial Valve Closure Surveillance Due to Mechanical Binding. (Section 4OA3)
05000219/2004006-00	LER	(Closed) LER 05000219/2004-006, Rev. 0. Local Leak Rate Test Results in Excess of Technical Specifications. (Section 4OA3)
05000219/2004007-00	LER	(Closed) LER 05000219/2004-007, Rev.0. Automatic Containment Isolation Bypassed During Reactor Start-up Due to Inadequate Procedure. (Section 4OA3)
05000219/2004001-00	SR	(Closed) Special Report 05000219/04-01, Rev.0. Reactor Start-up with the Rod Worth Minimizer Inoperable. (Section 4OA3)
05000219/2004002-00	SR	(Closed) Special Report 05000219/04-02, Rev. 0. Electromatic Relief Valve (EMRV) Lift during NSSS Leak Test. (Section 4OA3)
05000219/2004005-01	URI	(Closed) URI 05000219/2004005-01: Pressure Locking of Isolation Condenser Valve, V-14-35. (Section 4OA5)

LIST OF DOCUMENTS REVIEWED
(not previously referenced)

Section 1R04:

Procedure 335, 34.5 KV and 13.8 KV Electrical System, Attachment 335-2, Station Blackout Electrical Lineup, Rev. 17, dated December 2004
Procedure 337, 4160 Volt Electrical System, Rev. 58, December 2004
Abnormal Operating Procedure, ABN-37, Station Blackout, Rev. 1, dated February 2005
Drawing, BR 3001, Electrical Main One Line
Drawing, 3E-743-11-001, Emergency Power System One Line Diagram (SBO)
Drawing, 3E-743-11-013, Sh 1,2, Emergency Power system One Line Diagram (Forked River Combustion Turbines C1, C2)
Drawing, 3E-743-18-006, Connection Diagram SBO Control Panel, Sh 1
Procedure, 678.4.005, Station Blackout Functional Test, Rev. 13, dated April 2003

Section 1R05:

OB-FZ-6A, “A” 480V Switchgear Room
OB-FZ-6B, “B” 480V Switchgear Room
TB-FA-26, Battery Room South of 4160V Switchgear
TB-FA-3B, Turbine-Mezzanine 23' Elevation
OB-FZ-8C, A&B Battery Room, Tunnel & Electric Tray Room, 35'-0" Elevation
TB-FA-3A, 4160V Emergency Switchgear (1C & 1D) Vaults
TB-FZ-11A, Turbine Operating Floor 46' Elevation
TB-FZ-11C, Switchgear Room, West end of Turbine Building on Mezzanine Level, Elev. 23'-6"
TB-FZ-11D, Basement Floor South End, Elev. 3'-6"
OB-FZ-4, Cable Spread Room 36' Elevation
Station Blackout transformer

Section 1R12.1:

ER-AA-310-1003, “Maintenance Rule - Performance Criteria Selection,” Rev. 2
ER-AA-310-1004, “Maintenance Rule - Performance Monitoring,” Rev. 1
ER-AA-310-1005, “Maintenance Rule - Dispositioning between (a)(1) AND (a)(2),” Rev. 1
ER-AA-310, “Implementation of the Maintenance Rule,” Rev. 3
Plant Health Committee Presentation - Station Blackout & Support Systems, December 2004
Maintenance Rule Performance Report - 4160V System
4160V System Walkdown Report, dated February 2004
Condition Assessment of Cable Circuits at Oyster Creek Nuclear Power Plant, DTE Energy Report # 2004-39

CAP Reports: O2005-1314, O2005-0959, O2005-0844, O2005-0845, O2005-0850, O2005-0832, O2005-0701, O2005-0279, O2004-2436, O2004-2313, O2004-1831, O2004-1548, O2004-1206, O2004-0530, O2004-0057

Section 1R12.2:

Oyster Creek Generating Station, Maintenance Rule Periodic (a)(3) Assessment for the period January, 2002 through December, 2003.
ER-AA-310-1004; Exelon Maintenance Rule-Performance Monitoring Procedure
ER-AA-310-1005; Exelon Maintenance Rule- Dispositioning Between a(1) and a(2).
ER-AA-310-1007; Exelon Maintenance Rule- A(3) Report
System Health Reports for selected systems
Maintenance Rule a(1) Evaluations and Action Plans for selected systems
OC-7 Maintenance Rule System Functionality Definitions
Electronic Task Tracking System (ETTS) Task No 3515: GE SIL 606
Unavailability Data for SFPC, ASFPC, and TBCCW Pumps for 1/02-12/04
Oyster Creek FSAR.

Action Requests: A2086259, A2096260, A2100479, A2100486, A2102086, A2102090, A2102093, A2102094, A2102096, A2102099

CAP Reports: O2000-1448, O2001-0977, O2001-1075, O2002-0496, O2002-0667, O2002-1557, O2003-0151, O2003-0222, O2003-1419, O2003-1912, O2003-2567, O2004-1846, O2004-1943, O2004-1953, O2004-2019, O2004-2029, O2004-2034, O2004-2499, O2004-2581, O2004-2970, O2004-2977, O2004-3109, O2004-3128, O2004-3406, O2004-3442, O2004-3826, O2004-4034, O2005-0157, O2005-0482

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
AR	Action Request
ASFPC	Augmented Spent Fuel Pool Cooling
ATWS	Anticipated Transient Without Scram
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
EDG	Emergency Diesel Generator
FSAR	Final Safety Analysis Report
IMC	Inspection Manual Chapter
LER	Licensee Event Report
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PI	Performance Indicator
PI&R	Problem Identification & Resolution
PMT	Post Maintenance Test

RBCCW	Reactor Building Closed Cooling Water
RO	Reactor Operator
RPS	Reactor Protection System
SDP	Significance Determination Process
SFPC	Spent Fuel Pool Cooling
SSC	Systems, Structures and Components
ST	Surveillance Test
TBCCW	Turbine Building Closed Loop Cooling Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	Work Order