September 6, 2001

Mr. Ron J. DeGregorio
Vice President - Oyster Creek
AmerGen Energy Company, LLC
P.O. Box 388
Forked River, New Jersey 08731

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 50-219/01-07

Dear Mr. DeGregorio:

On August 11, 2001, the NRC completed an integrated inspection at your Oyster Creek reactor facility. The enclosed report documents the inspection findings which were discussed on August 31, 2001, with Mr. Ernie Harkness and other members of your staff.

This 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, the inspectors identified one issue of very low safety significance (Green). This finding was determined to be a violation of NRC requirements. However, because of the very low safety significance and because the issue has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, 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 the Oyster Creek facility.

In accordance with 10 CFR 2.790 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 Web site at http://www.nrc.gov/NRC/ADAMS/index/html (the Public Electronic Reading Room).

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch No. 7
Division of Reactor Projects

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

Enclosure: Inspection Report 50-219/01-07
Attachment 1: Supplemental Information

cc w/encl: Amergen Energy Company - Correspondence Control Deck
J. A. Benjamin, Licensing - Vice President, Exelon Corporation
M. Gallagher, Director-Licensing
B. Stewart, Acting Regulatory Affairs Manager
R. Shadis, New England Coalition Staff
State of New Jersey

<u>Distribution w/encl</u>: Region I Docket Room (with concurrences)
L. Dudes, DRP - NRC Resident Inspector
H. Miller, RA
J. Wiggins, DRA
J. Rogge, DRP
N. Perry, DRP
T. Haverkamp, DRP
D. Loveless, OEDO
E. Adensam, NRR
H. Pastis, PM, NRR
T. Colburn, Backup PM, NRR

DOCUMENT NAME: C:\Program Files\Adobe\Acrobat 4.0\PDF Output\OC0107.wpd
After declaring this document "An Official Agency Record" it **<u>will</u>** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI:DRP		RI:DRP							
NAME	Dudes/JFR f/		Rogge/JFR							
DATE	09/05/01		09/05/01							

OFFICIAL RECORD COPY

U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No.	50-219/01-07
Docket No.	50-219
License No.	DPR-16
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	July 1, 2001- August 11, 2001
Inspectors:	Laura A. Dudes, Senior Resident Inspector Thomas R. Hipschman, Resident Inspector Frank Arner, Reactor Inspector, July 10-13, 2001 Neil Perry, Senior Project Engineer, July 23-27, 2001 George Morris, Reactor Inspector, July 30-August 3, 2001 Joseph G. Schoppy, Jr., Senior Resident Inspector, Hope Creek, July 30-August 3, 2001 John R. McFadden, Health Physicist, July 23-27, 2001
Approved By:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000219-01-07, on 07/01-08/11/01, AmerGen, Oyster Creek Generating Station. Fire Protection.

The inspection was conducted by resident and region based inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation for failure to maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report (UFSAR) as required by Oyster Creek Facility Operating License Condition 2.C.3. For approximately 15 days, AmerGen personnel failed to take appropriate compensatory measure for an impaired fire barrier in the reactor building.

  This finding was determined to have very low safety significance due to the low combustible loading, fire detection capability, and fire suppression system availability in the area of concern. (Section 1R05)

B. Licensee-Identified Findings

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspector. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status:

Oyster Creek began the inspection period at full power and remained there for the duration of the inspection except for two occasions where power was reduced in order to reduce condenser discharge temperature to meet environmental temperature limits.

1. REACTOR SAFETY
Initiating Events, Mitigating Systems, Barrier Integrity (REACTOR-R)

1R04 Equipment Alignment

.1 Core Spray System Full Equipment Alignment

a. Inspection Scope

The inspectors performed a complete equipment alignment check on the core spray system to verify that the system was properly configured and to identify any discrepancies that might impact the function of the system. The alignment check included a review of documents to determine the correct system lineup and performance of a field walkdown to identify any discrepancies between the existing lineup and the prescribed lineup. Specifically the following documents and procedures were reviewed:

- Procedure 308, *Emergency Core Cooling System Operation*
- Procedure 308, Attachment 308-1, *Valve Checkoff List*
- Procedure 308, Attachment 308-2, *Electrical Checkoff List*
- *Core Spray System Flow Diagram* (GE 885D781)
- *Updated Final Safety Analysis Report*, Section 9.5.4
- *Core Spray/Auto Depressurization System Health Overview Report for 2nd Quarter 2001*
- Maintenance Rule Unavailability Tracking Chart for the Core Spray Train 1 and Train 2
- Technical Specifications
- Maintenance Rule Database
- Workaround Database
- Defeated Alarm Log
- Control Room Deficiency Tags
- Temporary Modification Log
- 610.4.002, *Core Spray Pump Operability Test*, dated 5/19/01
- 610.4.003, *Core Spray Valve Operability and In-Service Test*, dated 6/4/01
- 610.4.007, *Core Spray System Firewater Valve Test*, dated 6/9/01
- 610.4.008, *Core Spray Testable Check Valve Operability Test*, dated 11/12/00
- 610.4.011, *Core Spray System Testable Check Valve Leakage and In-Service Test*, dated 11/15/00
- 610.4.012, *Core Spray System 1 Pump In-Service Test*, dated 6/8/01
- 610.4.013, *Core Spray System 2 Pump In-Service Test*, dated 6/6/01
- Engineering Evaluation 0162-99; *Operability Determination for Core Spray, Core Spray Booster, and Containment Spray Pump Motors when Motor Heaters are Found Inoperable*

In addition, the inspectors reviewed several corrective action process (CAP) reports associated with the core spray system (CAP Nos. 2000-0953, 2000-1970, 2000-2041, 2001-003, 2001-0338, 2001-0496. and 2001-1161) to verify that system degradations were being identified and corrected in a timely manner.

b. Findings

No findings of significance were identified.

.2 Emergency Diesel Generator No. 1

a. Inspection Scope

The inspector performed a partial walkdown of accessible areas of the emergency diesel generator (EDG) No. 1 during the period that the No. 2 EDG was unavailable due to maintenance. The inspector used procedure 341, *Emergency Diesel Generator Operation*, to verify the EDG was aligned per the operational procedure and ready to perform its safety function.

b. Findings

No findings of significance were identified.

.3 Combustion Turbines

a. Inspection Scope

The Forked River Combustion Turbines serve as the power source for the Oyster Creek plant in the event of a station blackout. The inspectors performed a partial walkdown of the electrical power breakers and circuit switches to confirm the system was lined-up to support abnormal procedure 2000-ABN-3200.37, Rev. 10, *Station Blackout*. The inspectors also walked down the combustion turbine control cubical to compare the operator response guidelines found in the ABN to the plant lineup.

b. Findings

No findings of significance were identified.

.4 Emergency Service Water System Alignment Check

a. Inspection Scope

On August 2, 2001, AmerGen identified an underground leak from emergency service water (ESW) system 2 that potentially impacted the continued operability of startup transformer bank 6 (CAP No. 2001-1233). The inspectors verified by plant walkdowns and main control room tours that the emergent ESW system 2 issue did not adversely affect the redundant ESW system 1 components and safe shutdown electrical components. The inspector verified the normal system component alignment of ESW system 1 using procedure 310, *Containment Spray System Operation*, attachments 310-

1 and 310-2. The inspectors performed walkdowns of the following ESW/Containment Spray support systems and areas:

- Emergency diesel generators 1 and 2
- Startup transformer banks 5 and 6
- Station blackout (SBO) transformer and control panel
- 4160V vital buses C and D
- 4180V vital switchgear room A and B
- Control room instrumentation and equipment control panels

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Fire Protection Walkdowns

a. Inspection Scope

The inspectors conducted fire protection inspection activities consisting of plant walkdowns, discussions with fire protection personnel, and reviews of the Oyster Creek Fire Hazards Analysis Report (FHAR) and special procedure 97-003, *Oyster Creek Pre-Fire Plans*, to verify that the fire protection program was implemented in accordance with all conditions stated in the facility license. Plant walkdowns included observations of combustible material control, fire detection and suppression equipment availability, and passive fire protection features (i.e., electrical raceway fire barriers, penetration seals, fire doors, and fire dampers). The inspectors assessed the fire protection systems and features for material condition and operational effectiveness. The following areas and equipment were inspected due to the potential impact on mitigating systems:

- Reactor Building 119' Elevation (general walkdown)
- Reactor Building 95' Elevation (general walkdown)
- Reactor Building 75' Elevation (detailed walkdown)
- Reactor Building 51' Elevation (detailed walkdown)
- Reactor Building 23' Elevation (general walkdown)
- CRD Pump Room/Core Spray System I Pump Room (detailed walkdown)
- Reactor Building Fire Hose Stations

b. Findings

The inspectors identified a Non-Cited Violation for failure to maintain in effect all provisions of the approved fire protection program as AmerGen personnel failed to take appropriate compensatory measure for an impaired fire barrier in the reactor building. This finding was determined to have very low safety significance due to the low combustible loading, fire detection capability, and fire suppression system availability in the area of concern.

On July 30, 2001, the inspectors noted that an access hatch in the northwest corner of reactor building elevation 23' (penetration PH46) had its fire stop (floor plug) removed and that a small radiation monitor coax cable impaired the steel plate access door. The control room supervisor stated that the PH46 fire barrier was considered operable with the steel plate in place. On July 31, the inspectors discussed the PH46 fire barrier operability with the fire protection coordinator. Prompt follow-up by fire protection engineers determined that the FHAR (Volume II, Revision 11, Section 1, page 92) states that the steel plate provides adequate separation between RB-FZ-1E and RB-FZ-1F5 (PH46) based on six assumptions. They noted, however, that one of the assumptions stated “there are no combustibles passing through the steel plate access door.” The coax cable passing through PH46 invalidated this assumption and impaired the fire barrier. AmerGen Procedure 101.2, *Fire Protection Program*, requires AmerGen to take appropriate compensatory measures (e.g., fire watch) for degraded fire barriers. Contrary to this requirement, AmerGen did not establish appropriate compensatory measures. On July 31, fire protection engineers initiated corrective actions via CAP No. 2001-1214 and coordinated with radiation protection to remove the coax cable.

The finding involved impairment (without adequate compensation) of a fire barrier and if left uncorrected could become a more significant safety concern dependent upon combustible loading and the availability of detection and suppression systems. The inspectors assessed the risk significance of this issue using the NRC fire protection SDP (NRC Manual Chapter 0609, Appendix F). For this phase 2 SDP evaluation, the inspectors conservatively considered a fire scenario originating in fire area RB-FZ-1F5 (reactor building elevation -19') and spreading to fire area RB-FZ-1E (reactor building elevation 23') via the combustible coax cable passing through PH46. This is a conservative assumption based on the low combustible loading in the northwest corner of the reactor building on elevation 23' and -19' and the limited combustible loading of the cable (one instrument cable). Based on the Oyster Creek Individual Plant Examination for External Events (IPEEE), a fire ignition frequency (IF) of 3E-2 was used for RB-FZ-1E. A medium level degradation was assumed for the 1-hour fire barrier (FB = -0.5). The inspectors witnessed two fire brigade drills in the past twelve months. One of these drills did not meet AmerGen’s expectations due to inadequate response by members of the fire brigade. However, the inadequate response was the result of a recent staffing change that had not been communicated clearly and was immediately corrected by the licensee. Based on this performance, the inspectors considered the manual fire fighting capability outside the control room as normal operating state, no degradation (MS = -1.0). AmerGen took no compensatory measures for the degraded barrier, such as a fire watch, resulting in no additional credit in this area. Automatic fire detection alarms locally in RB-FZ-1E and in the control room. In addition, there are two automatic open-head water spray deluge systems installed in RB-FZ-1E. Based on the availability of these systems, full credit was assigned for automatic suppression and detection (AS = -1.25). An adjustment was added to account for the common water delivery and supply system for both the automatic deluge system and the normal hose stations (CC = +0.25). Based on these factors, the fire mitigating frequency (FMF) was calculated to be -4.5. Using Table 5.4 of the SDP, the approximate frequency was 1 per 10,000 to 100,000 years. Using Table 5.7 of the SDP, the estimated likelihood rating was in the "F" category based on an exposure time for the degraded condition of 3-30 days.

To complete the SDP assessment, the inspectors evaluated the equipment available to place the plant in a hot shutdown condition. Based on the FHAR (Volume II, Revision 11, Section 1, pages 89-90), RB-FZ-1E contains electrical circuits for hot shutdown paths (HPs) 1,2,3,4 and 5. For a fire in RB-FZ-1E, hot shutdown is achieved using HP1. The FHAR describes the reasons why the HP1 circuits on this elevation are not required for a fire in this zone. The inspectors considered HP1 equipment as one train. Using Table 5.8 of the SDP, the issue has a very low safety significance and results in a Green finding.

Oyster Creek Facility Operating License Condition 2.C.3 requires AmerGen to maintain in effect all provisions of the approved fire protection program as described in the UFSAR. The UFSAR references the Oyster Creek FHAR as part of the fire protection program. AmerGen's failure to maintain the PH46 fire barrier in accordance with the FHAR without taking appropriate compensatory measures is a violation of the fire protection program. However, because this violation is of very low significance and the deficiency was entered into the corrective action system, this finding is being treated as a Non-Cited Violation consistent with Section V1.A of the NRC Enforcement Policy issued on May 1, 2000 (65FR25368). **(NCV 50-219/01-07-01)**

.2 Annual Fire Drill Observation

a. Inspection Scope

On August 2, 2001, the inspector observed a fire drill conducted to verify the licensee's readiness to successfully mitigate a fire at the Oyster Creek facility. The drill was conducted at the reactor recirculation pump motor generator room. The inspector verified that the fire brigade response was in accordance with the licensee's procedures. The conduct of the drill was verified against the criteria in licensee procedure 101.2, *Fire Protection Program*, Attachment 101.2-2. In addition, the critique of the fire drill was reviewed to verify that any problems identified as a result of the drill were being captured in the licensee's corrective action program.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector verified that the licensee's maintenance, testing, inspection and evaluation of results were adequate to ensure proper heat transfer for the containment spray system 1 and 2 heat exchangers.

The inspector examined design calculations to ensure that acceptance criteria contained within surveillance tests were consistent with assumptions found within the plant Updated Final Safety Analysis Report (UFSAR) and engineering evaluations of the Design Basis Accident (DBA) containment performance. The methodology and results of the containment spray heat exchanger calculations were reviewed to ensure consistency with accepted industry practices.

The ESW chemical treatment program was reviewed and discussed with the system engineer to verify potential biofouling mechanisms had been identified and corrective measures implemented when necessary. Corrective actions related to selected findings from a 1996 service water self assessment were reviewed to verify adequate resolution. Additionally, the inspector examined the service water intake low level procedure and ESW operational procedures to ensure that required flow rates through the containment spray heat exchangers would be maintained in order to achieve the assumed heat transfer capability.

The inspector performed a walkdown of selected portions of the intake structure and the four containment spray heat exchangers to assess the material condition of the components.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors selected the following safety significant systems in (a)(1) and (a)(2) status to verify that: (1) failed structures, systems and components (SSCs) were properly characterized, (2) goals and performance criteria were appropriate, (3) corrective action plans were appropriate, and (4) performance was being effectively monitored:

- Reactor Building Ventilation: (a)(1)
- Chlorination System: (a)(2)
- Combustion Turbines: (a)(2)
- M-G set/"B" Battery Room Ventilation: (a)(2)

b. Findings

No findings of significance were identified.

1R13 Maintenance Rule Risk Assessment and Emergent Work Control

.1 Emergency Service Water System II Out of Service Risk Management

a. Inspection Scope

The inspectors evaluated on-line risk management for the following emergent corrective maintenance issues associated with an ESW system 2 leak that potentially impacted the continued operability of startup transformer bank 6 (CAP No. 2001-1233). The inspectors reviewed maintenance risk evaluations, work schedules, recent corrective action reports, 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 ESW system and the potentially degraded startup transformer. The inspectors also discussed AmerGen's on-line risk assessment monitor (ORAM Sentinel) evaluations with on-shift senior reactor operators and AmerGen management.

To assess AmerGen's risk management, the inspectors reviewed the following documents:

- NRC Regulatory Guide 1.182, *Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants*

- Section 11, *Assessment of Risk Resulting from Performance of Maintenance Activities,* dated February 11, 2000, of NUMARC 93-01, *Industry Guideline For Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*

b. Findings

No findings of significance were identified.

.2 Risk Management Evaluation Due to Grid Maximum Generation Alert

a. Inspection Scope

On July 24 and during the week of August 6-10, 2001, the inspector reviewed the licensee's risk management strategy regarding the maximum generation alert issued by Pennsylvania, New Jersey, Maryland (PJM) grid operators due to extreme elevated temperatures. The inspector verified that the licensee reviewed potential risk conditions associated with a short duration loss of offsite power due to possible grid instabilities and verified that routine station work planned during the extreme heat would not increase the potential for a reactor shutdown or a loss of offsite power.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determinations associated with the following plant equipment deficiencies to verify that all equipment was capable of performing its design basis function and in order to determine that operability justifications were performed in accordance with procedures OC-2, *Operability Review and Analysis*, and 2000-ADM-7216.01, *Corrective Action Process*. In addition, where a component was determined to be inoperable, the inspectors verified the technical specification (TS) limiting condition for operation implications were properly addressed.

- Emergency Diesel Generator No. 2 Trip during load test (CAP 2001-1121)
- Containment Electrical Penetrations perform part of the function of the containment barrier. These penetrations are normally pressurized with a dry nitrogen blanket for long term electrical insulation integrity. The inspectors reviewed CAP O2001-1206, Containment Electrical Penetration number 8, Nitrogen blanket less than required pressure to assess its associated operability evaluation. The inspectors reviewed the licensee’s response to a related NRC Bulletin 77-06, Potential Problems with Containment Electrical Penetration Assemblies, submitted to the NRC on December 2, 1977, and compared the CAP with the requirements of station procedure 665.3.021, Rev. 7, Containment Electrical Penetration Nitrogen Blanket Surveillance. The inspectors interviewed the responsible system engineer to discuss any potential concerns.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the operator work-around database and associated corrective action items to identify conditions that could adversely effect the functionality of mitigating systems or impact human reliability in responding to initiating events. The inspector also reviewed open control room deficiencies and corrective action items to determine if there were any degraded or non-conforming conditions that should have been identified and evaluated as operator work-arounds.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspector reviewed the following ESW pipe replacement modification documents to verify that the design inputs were equivalent to the current underground piping scheme for the ESW piping.

- ECR OC 01-00475, “Investigate and Repair/ Replace Emergency Service Water Piping System II, due to leak in portion of underground piping.
- ECR OC 01-00481, ESW piping replacement Design Inputs and Impact Review.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed and observed portions of the post maintenance testing associated with the following 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.

- Action Request (AR) A2009427, EDG No. 2 Governor Solenoid Replacement
- Work Order No. C2001184, “Replacement of Emergency Service Water System II Piping.” Post maintenance test, “Containment Spray and Emergency Service Water System 2 Pump Operability Test.”

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Main Steam Line Low Pressure Functional Test and Calibration

a. Inspection Scope

On August 8, 2001, the inspector observed the performance of surveillance procedure 619.3.008, *Low Pressure Main Steam Line Functional and Calibration Test While Operating*. The inspector also observed the removal of a temporary modification during performance of the surveillance test. The inspector verified that the performance and resulting data associated with the surveillance test met the requirements of the technical specifications. The inspector also reviewed the results of past performances of the surveillance test and discussed instrument performance with the system engineer to verify that degraded or non-conforming conditions were identified and corrected.

b. Findings

No findings of significance were identified.

.2 Emergency Diesel Generator #2 Load Test

a. Inspection Scope

On August 9, 2001, the inspector observed the performance of surveillance procedure 636.4.013, *Diesel Generator #2 Load Test*. The inspector also reviewed the completed surveillance document against the requirements in Technical Specification (TS) sections 3.7.C and 4.7.A, to verify that all criteria in the TS were met.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

.1 Alternate Power Supply for Drywell Sump Integrators

a. Inspection Scope

Off-site car-pole accidents in the past have resulted in the loss of the 1E1 35.5 kV line. That power source feeds the control circuit for the drywell sump pumps which could provide indication of leaks in the drywell. (The 480 Volt power supply for the sump pump motors comes from an unaffected source). The inspectors reviewed a contingency temporary modification, dated 4/30/2001 and revised 7/7/2001, prepared by engineering and submitted to operations for their use. This modification, unnumbered because it had not been implemented, could be used to provide temporary power the drywell sumps from a power receptacle in panel IM175. This panel is powered from a diesel backed bus. The inspectors also interviewed the responsible engineer.

b. Findings

No findings of significance were identified.

.2 Emergency Service Water Piping Supports

a. Inspection Scope

The inspector reviewed temporary modification document 2000-067, *Temporary supports for Emergency Service Water Piping downstream of Pumps P-3-3A, P-3-3C, and P-3-3D*. The inspector verified that the temporary supports provided adequate structural support to maintain the systems seismic design margins. In addition, the inspector verified that the supports did not interfere with other design features, such as heat tracing as a result of their installation.

b. Findings

No findings of significance were identified.

Emergency Preparedness (EP)

EP6 Drill Evaluation

a. Inspection Scope

On July 11, 2001, the inspector observed the Emergency Preparedness drill in the Technical Support Center. The inspector reviewed checklists and forms used for classification notification and Protective Action Recommendation development. The inspector reviewed EPIP-OC-.01, *Classification of Emergency Conditions*, EPIP-OC-.03, *Emergency Notification*, and EPIP-0C-.26, *The Technical Support Center* to verify the emergency response team actions were in accordance with licensee approved procedures.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

The inspector toured the facilities and inspected procedure implementation, records, and reviewed program documents to evaluate the effectiveness of the licensee's access controls to radiologically significant areas. The inspector toured the reactor building, turbine building, augmented off-gas building, the new and old radioactive waste buildings, the survey meter calibration facility, and outside yard areas within the radiologically controlled area (RCA) boundary. Within these areas, the inspector observed activities to verify conformance with applicable requirements for RCA entry and exit, use of personnel dosimetry (primary and secondary), and setpoints used for dose and dose rate alarms. Also reviewed were posting, labeling, barricading, and level of access control for locked high radiation areas (LHRAs), high radiation areas (HRAs), radiation and contamination areas, and radioactive material areas. Independent radiation level measurements were performed during portions of the tours to verify conformance with applicable requirements.

On July 23, the inspector observed a Radiation Performance Committee meeting in order to assess the plans and methods being discussed to practice ALARA (As Low As Is Reasonably Achievable). On July 26, the inspector observed the pre-job brief for an entry into the steam-affected areas of the turbine building in order to evaluate the adequacy of the instructions to the workers and the radiological controls to be implemented.

The following radiation work permits (RWPs) and surveys were reviewed for the adequacy of radiological survey data, required radiological controls and personal protective equipment, and instructions to radiation workers.

- RWP OC-1-01-00001 Turbine building, steam-affected areas
- RWP OC-1-01-00002 RCA clean-up project/radwaste removal and shipping
- RWP OC-1-01-00003 Sludge/resin transfers
- RWP OC-1-01-00056 I & C instrument maintenance
- RWP OC-1-01-00058 Observation and inspection
- Radiological survey of old radwaste west tanks dated June 01, 2001

Selected sections of the following procedures and documents were also reviewed to evaluate their adequacy and compliance with applicable regulations.

- Procedure 6630-ADM-4000.11, Revision 3, *Rules of Conduct of Radiological Work*
- Procedure 6630-ADM-4110.04, Revision 8, *Radiological work process*
- Procedure 6630-ADM-4200.01, Rev. 6, *Radiological Surveys*
- Exposure summary report for May 2001 maintenance outage
- Weekly exposure summary reports for June 25 and July 2, 9, and 16
- Clean sweep 2001 project - Radioactive material survey log
- Radiation Performance Committee charter dated July 23, 2001
- Self-assessment on challenge radiological surveys dated May 13, 2001
- Nuclear Oversight continuous-assessment observations (approximately twenty) from May 9 to July 24, 2001

The inspection included a review of the following CAP items for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action: CAP Nos. O2001-0307, O2001-0804, O2001-0810, O2001-0906, O2001-1007, O2001-1055, O2001-1065, O2001-1069, O2001-1155, and O2001-1198.

The review was against criteria contained in 10 CFR 19.12, 10 CFR 20 (Subparts D, F, G, H, I, and J), site TSs, and site procedures.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Control (71121.02)

a. Inspection Scope

The inspector toured the facilities and inspected procedural implementation, and reviewed records and other program documents to determine the effectiveness of ALARA planning and control. On July 23, the inspector attended and observed a Radiation Performance Committee meeting.

The following procedures and program documents were reviewed.

- Procedure 6630-ADM-4010.02, Revision 10, *Conduct of radiological engineering*
- Procedure ES-007, Rev. 4, *ALARA guidelines for configuration changes*
- Current status of actual cumulative personnel exposure versus the projected annual estimate
- Exposure report for the May 2001 maintenance outage
- Agenda for Radiation Performance Committee meeting on July 23, 2001
- Draft five year 2001-2005 exposure reduction plan dated June 2001
- Radiological Engineering Calculation 2820-01-003, Rev. 0, February 12, 2001, 18R airborne alpha activity in the refueling cavity
- Radiological Engineering Calculation 2820-01-005, Rev. 0, May 08, 2001, assessment of gross beta and gross alpha derived air concentrations (DACs) based on the year 2000, 10 CFR 61 analytical review of waste streams.

The following pre-job ALARA Reviews, associated with Radiological Engineering Reviews (RERs), were reviewed for the adequacy of scope and of documentation.

- RER 2001-6A 18U drywell/replacement of recirculation pump's mechanical seal
- RER 2001-8B Detailed plant clean-up
- RER 2001-9B Rebuild "A" clean-up pump

The review was against criteria contained in 10 CFR 20.1101, 10 CFR 20.1702, site TSs, and site procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data from the 2nd quarter of 2000, through the 2nd quarter of 2001, for *Safety System Functional Failures* to verify its accuracy. The inspectors used Nuclear Energy Institute (NEI) 99-02, Revision 0, *Regulatory Assessment Performance Indicator Guideline*, as guidance.

b. Findings

No findings of significance were identified.

.2 Residual Heat Removal System Unavailability

a. Inspection Scope

The inspectors reviewed PI data from the 2nd quarter of 2000, through the 2nd quarter of 2001, for *Residual Heat Removal System Unavailability* to verify its accuracy. The inspectors used NEI 99-02, Revision 0, “Regulatory Assessment Performance Indicator Guideline,” as guidance.

b. Findings

No findings of significance were identified.

.3 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector selectively examined records used by the licensee to identify occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the period from April 2001 to the time of this inspection against the applicable criteria specified in NEI 99-02, *Regulatory Assessment Performance Indicator Guideline*, Revision 1, to verify that all conditions that met the NEI criteria were recognized and identified as PIs. The reviewed records/activities included corrective action program records and reviews of daily individual and RWP exposures.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the following CAP items for the adequacy of event categorization, immediate and corrective action to prevent recurrence, and timeliness of corrective action: CAP Nos. O2001-0307, O2001-0804, O2001-0810, O2001-0906, O2001-1007, O2001-1055, O2001-1065, O2001-1069, O2001-1155, and O2001-1198.

b. Findings

The licensee has not established effective problem resolution relative to recurring problems involving personnel failing to exit radiological controlled areas upon alarm of electronic self-reading dosimetry (ESRD) equipment in accordance with licensee site procedure 6630-ADM-4000.11, *Rules of Conduct of Radiological Work*. Corrective Action Program reports CAP 02001-0307 and CAP 02001-1155, dated February 28 and July 18, 2001, respectively, were developed to address licensee-identified non-conformance with this procedure, and are referenced in Section 4OA7 of this report.

Notwithstanding, in-field verification of corrective action effectiveness during this inspection revealed that some radiation protection personnel were still incorrectly instructing workers that they need not exit the area upon a dose-rate alarm from their ESRD, but that it was acceptable to retreat from the specific radiation field until the alarm was silent and continue to work, contrary to the procedural requirements. The licensee has established CAP 02001-1198 to address this latest issue.

On July 24, 2001, the inspector determined, through interviews with a radiation protection technician, that the technician was not familiar with the requirement to exit the area upon occurrence of a dose-rate alarm of an ESRD. Further, on July 26, 2001, the inspector observed a radiation protection supervisor provide a radiological pre-work briefing for entry into a LHRA, and incorrectly state to the workers that they need not exit the area and report to radiation protection upon a dose-rate alarm of their ESRDs. These inspector observations indicate that the implementation of resolution for the two previous occurrences may not have been effectively communicated to responsible personnel, and may not be implemented in accordance with the established procedures and expectations. Notwithstanding, no new violations were identified, and no actual unintended or excess personnel exposures are known to have occurred. This issue was placed in the licensee's corrective action system as CAP 2001-1198.

4OA6 Event Follow Up

On August 10, 2001, during severe lightening storms, a large electrical transient occurred on the electrical system 230 kilovolt and the 34.5 kilovolt offsite power lines. All power lines were restored within two hours and the plant remained stable throughout the transient. The inspector risk assessed the impact of this event on the plant and reviewed CAP 2001-1267 which documented the event and provided immediate and long term corrective actions. No safety issues were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On August 31, 2001, the resident inspectors presented the inspection results to Mr. Ernie Harkness 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.

4AO7 Licensee-Identified Violations

The following finding of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations (NCVs).

NCV Tracking Number	Requirement Licensee Failed to Meet
50-219/01-07-02	Technical Specification 6.11, the Oyster Creek Safety and Health Guide, and Site procedure 6630-ADM-4000.11, Rev. 3 require that personnel are to immediately exit the area upon an alarm of their electronic self-reading dosimetry (ESRD) and notify Radiation Protection. Contrary to this requirement, on February 28, 2001 (CAP O2001-0307) and on July 18, 2001 (CAP O2001-1155), personnel experienced ESRD dose-rate alarms and did not exit the area and report to radiation protection. These repetitive events were more than minor in that worker safety could be impacted if they failed to properly respond to alarming dosimeters in situations with the potential for unplanned radiation dose. However, the issues were determined to be of very low significance (GREEN) because the issues did not result in an over exposure, did not create a substantial potential for an over exposure and did not compromise the licensee’s ability to assess dose to workers.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

R. Adams, NOS Manager
V. Aggarwal, Director, Engineering
F. Buckley, Manager NSSS/RE Engineering
R. DeGregorio, Vice President
J. Frank, System Engineer
R. Gayley, Engineering Programs
M. Godknecht, Engineering
E. Harkness, Plant Manager
R. Heffner, Radiological Engineer
R. Hillman, Manager, Chemistry & Radwaste
E. Hosterman, MAROG Program Manager
A. Judson, Radiological Engineer
J. Magee, Director, Maintenance
R. Maldonado, Engineering Manager
M. Massaro, Director, Work Management
D. McMillan, Director, Training
M. Moore, Radiation Protection Manager
J. Renda, Radiation Protection Supervisor
J. Rogers, Regulatory Assurance
P. Sawyer, Radiological Engineering Manager
G. Seals, Radiological Engineer
D. Slear, Senior Manager, Design Engineering
W. Stewart, (Acting) Regulatory Assurance Manager
C. Wilson, Senior Manager, Operations

b. List of Items Opened, Closed, and Discussed

Opened and Closed

50-219/01-07-01	Failure to maintain the a fire barrier in accordance with the Fire Hazards Analysis Report without taking appropriate compensatory measures (Section 1R05)
50-219/01-07-02	Personnel failed to exit a radiation area after receiving an ESRD dose-rate alarm. Technical Specification 6.11, the Oyster Creek Safety and Health Guide, and Site procedure 6630-ADM-4000.11, Rev. 3 require that personnel are to immediately exit the area upon an alarm of their electronic self-reading dosimetry (ESRD) and notify Radiation Protection.

c. List of Acronyms

ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
AR	Action Request
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CT	Combustion Turbine
DAC	Derived Air Concentration
DBA	Design Basis Accident
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
ESRD	Electronic Self Reading Dosimetry
FHAR	Fire Hazards Analysis Report
FMF	Fire Mitigating Frequency
HP	Hot Shutdown Path
HRA	High Radiation Area
I&C	Instrumentation and Control
IPEEE	Individual Plant Examination for External Events
JO	Job Order
LHRA	Locked High Radiation Area
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ORAM	On-line Risk Assessment Monitor
OS	Occupational Safety
PI	Performance Indicator
PJM	Pennsylvania, New Jersey, Maryland (grid operator)
RCA	Radiologically Controlled Area
RER	Radiological Engineering Review
RWP	Radiation Work Permit
SBO	Station Blackout
SDP	Significance Determination Process
SJAE	Steam Jet Air Ejector
SSCs	Structures, Systems and Components
TDR	Technical Data Report
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report