July 6, 2007

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

SUBJECT: OYSTER CREEK GENERATING STATION - NRC COMPONENT DESIGN BASIS INSPECTION REPORT 05000219/2007006

Dear Mr. Crane:

On May 24, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Oyster Creek Generating Station. The enclosed inspection report documents the results of the inspection, which were discussed on May 24, 2007, with Mr. T. Rausch, Site Vice President, 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. In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, initiating events, and design basis accidents. The inspection also reviewed AmerGen's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two NRC-identified findings which were of very low safety significance (Green). One of these findings was determined to involve a violation of an NRC requirement. However, because of the very low safety significance of the finding and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the 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 U.S. 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, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at Oyster Creek.

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

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report 05000219/2007006

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REGION I

Docket Nos.:	50-219
License Nos.:	DPR-16
Report Nos.	05000219/2007006
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	April 17 to May 24, 2007
Inspectors:	F. Arner, Senior Reactor Inspector (Team Leader) S. Pindale, Senior Reactor Inspector J. Lilliendahl, Reactor Inspector A. Patel, Reactor Inspector A. Ayegbusi, Reactor Inspector (Trainee) F. Baxter, NRC Electrical Contractor W. Sherbin, NRC Mechanical Contractor
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000219/2007006; 04/17/2007 - 05/24/2007; AmerGen Energy Company, LLC, Oyster Creek Generating Station; Component Design Bases Inspection.

The report covers the Component Design Bases Inspection conducted by a team of four NRC inspectors and two NRC contractors. Two findings of very low safety significance (Green) were identified, one of which was considered to be a non-cited violation. 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 may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance (Green), in that, the licensee did not correctly translate the design of the containment hardened vent valve accumulators into test procedures. Specifically, the acceptance criteria for allowable accumulator pressure drop within the periodic test procedure was not consistent with the original design criteria and did not ensure the assumed design capability of the valves during loss of instrument air events. The valves provide a method of permitting a controlled depressurization of primary containment during severe accident sequences that involve loss of decay heat removal. AmerGen entered this issue into their corrective action program to revise the test criteria to be consistent with the original design of the valve accumulators.

  The finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and conservatively determined a more detailed Phase 2 SDP evaluation was required to assess the safety significance because the finding affected the mitigation system containment vent function. The finding was determined to be of very low safety significance (Green) based upon the Phase 2 SDP evaluation. There was no violation of NRC requirements because the performance deficiency was associated with postulated beyond design basis events. (Section 1R21.2.2.2)

- Green. The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” in that, AmerGen did not incorporate the requirements and acceptance limits contained in applicable design documents into the EDG battery service test procedures.

Specifically, the design requirement of the EDG batteries to supply adequate voltage to the EDG output breakers was not incorporated into the service test load profile for the EDG batteries. This prevented verification within the test of the capability of the batteries to close the output breakers which is a design requirement during events with a postulated loss-of-offsite power. AmerGen entered the issue into their corrective action program to revise the EDG battery sizing calculation and evaluate the appropriate incorporation of the design requirements into the service test procedure.

The finding is more than minor because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding has very low safety significance, based on a Phase 1 review of the SDP, documented in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it did not represent the loss of safety function of the EDG batteries. (Section 1R21.2.1.15)

B. Licensee-identified Violations.

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the Oyster Creek Probabilistic Risk Assessment (PRA) and the U.S. Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model. Additionally, the Oyster Creek Significance Determination Process (SDP) Phase 2 Notebook, Revision 2, was referenced in the selection of potential components and actions for review. In general, the selection process focused on components and operator actions that had a risk achievement worth (RAW) factor greater than 2.0 or a Risk Reduction Worth (RRW) factor greater than 1.005. The components selected were located within both safety-related and non-safety related systems, and included a variety of components such as pumps, valves, generators, transformers and batteries. There were 10 mechanical and 6 electrical components selected for review.

The team initially compiled a list of a nominal 50 components and 10 operator actions based on the risk factors previously mentioned. The team performed a margin assessment to narrow the focus of the inspection to 16 components and 4 operator actions. The team's evaluation of possible low design margin included consideration of original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. The assessment included items such as failed performance test results, corrective action history, repeated maintenance, maintenance rule (a)1 status, operability reviews for degraded conditions, NRC resident inspector input of equipment problems, plant personnel input of equipment issues, and industry operating experience. Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. The margin review of operator actions included complexity of the action, time to complete action and extent of training on the action.

This inspection effort included walk-downs of selected components, including a review of selected simulator scenarios. It also included interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet both design bases and risk informed beyond design basis functions. A summary of the reviews performed for each component, operator action, operating experience sample, and the specific inspection findings identified are discussed in the following sections of the report. Documents reviewed for this inspection are listed in the attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component Design Reviews (16 Samples)

.2.1.1 Isolation Condenser (IC), NE-01-B

a. Inspection Scope

The isolation condenser, NE-01-B, was reviewed to verify its capability to meet design basis heat removal requirements in response to transient and accident events. The isolation condenser system (ICS) is a standby heat exchanger which is utilized to remove fission product decay heat when the reactor vessel is isolated from the main condenser. The heat exchanger prevents overheating of the reactor fuel, controls the reactor pressure rise, and limits the loss of reactor coolant through the electromagnetic relief valves. The team reviewed the maintenance history including historical tube replacements, issue reports, drawings, and surveillance testing to verify consistency with the assumed design capability of the heat exchanger. The team verified that the heat removal calculations satisfied the original design specifications. The team also performed walkdowns of accessible areas to assess the current condition of the isolation condenser.

b. Findings

No findings of significance were identified.

.2.1.2 Isolation Condenser Condensate Return Valve, V-14-35

a. Inspection Scope

The team reviewed the Isolation Condenser condensate return valve, V-14-35, to verify that it was capable of meeting its design basis requirement of opening during transient conditions to assist in the removal of decay heat. The review included system calculations and motor operated valve (MOV) analyses to verify that thrust and torque limits and actuator settings were appropriately evaluated. Surveillance testing results were reviewed to verify that the stroke time acceptance criteria were in accordance with the design bases and accident analysis assumptions. Additionally, issue reports related to the valve were reviewed to ensure conditions did not exist which would invalidate previous assumptions for the capability of the valve. The team verified that the system operating conditions and terminal voltage values used in the valve analyses were appropriate. The team performed walkdowns of accessible areas to assess the current material condition of the valve. The team also verified that the current valve configuration was not susceptible to thermal binding or pressure locking conditions.

b. Findings

No findings of significance were identified.

.2.1.3 Emergency Diesel Generator (EDG) Fuel Oil Transfer Pumps, P-39-15 and P-39-16

a. Inspection Scope

The EDG-2 fuel oil pumps, P-39-15 and P-39-16, were reviewed to verify their ability to adequately support the EDG-2 in response to transient and accident conditions. These fuel oil pumps are required to operate automatically to transfer diesel fuel from the 15,000 gallon tank to the EDG day tank. The equipment consists of two motor driven fuel transfer pumps per EDG with a suction strainer and discharge filter for each pump. The team reviewed the pump net positive suction head (NPSH) calculation, and the usable tank volume calculation related to fuel consumption of the proposed low sulphur diesel fuel to ensure adequate NPSH existed. The team also verified that a vortex condition at the suction inlet in the tank was considered in tank usable volume. The team verified seismic adequacy of the fuel oil pumps and suction tank and verified the equipment elevations were high enough to preclude external flooding damage. The team reviewed EDG surveillances to ensure adequate fuel oil was delivered to the day tank by the pumps. The team reviewed maintenance history to ensure the suction strainers and discharge filters were periodically replaced at appropriate intervals. Lastly, the team performed a walkdown of the fuel oil transfer pumps and 15,000 gallon tank areas to assess the general material condition of the equipment.

b. Findings

No findings of significance were identified.

.2.1.4 EDG-2 Radiator Heat Exchanger, M-39-2

a. Inspection Scope

The EDG-2 radiator heat exchanger was reviewed to verify its ability to support EDG operation in response to loss-of-offsite power conditions. The EDG coolant rejects heat to an air cooled radiator. Heated water from the discharge manifold exits the engine and flows through the water outlet to the radiator. Water from the radiator is piped to the lube oil cooler and from there to the engine water pumps. A 135,000 cubic feet per minute shaft driven fan is mounted at the front of the EDG skid to provide airflow in sufficient quantity to remove heat from the engine water radiator. The team reviewed worst case temperature conditions from testing performed during July and August at full EDG load to ensure the radiator can remove the heat rejected by the engine. The team reviewed maintenance history and issue reports for the radiator and associated air louvers to verify that no conditions existed which may challenge the ability to remove the heat under the most challenging conditions of load and temperature. The team also performed walkdowns of accessible areas to assess the current material condition of the radiator and louvers.

b. Findings

No findings of significance were identified.

.2.1.5 Core Spray Pump, P-20-1C

a. Inspection Scope

The ‘C’ main core spray pump was reviewed to verify its ability to meet its design basis head and flowrate requirements in response to transient and accident events. There are two core spray systems each of which contain two main and two booster pumps. The team verified that design inputs were properly translated into system procedures and tests, and reviewed completed surveillance tests associated with the demonstration of pump operability. The team reviewed the maintenance history, design changes, issue reports, design calculations, design specifications, drawings, and selected tests. Loss-of-coolant accident analysis reports were reviewed to ensure appropriate design criteria for the core spray pump were used. Lastly, the design and operation of the core spray keep-fill system were reviewed with respect to supporting operability of the main core spray pumps.

b. Findings

No findings of significance were identified.

.2.1.6 Emergency Service Water (ESW) Pump, P-3-3A

a. Inspection Scope

ESW pump P-3-3A was reviewed as a representative sample of ESW pumps to verify its ability to meet design basis requirements in response to transient and accident events. The team selected the ESW pump to determine whether there was the potential for a common cause failure mechanism of the pumps. The team reviewed design documents, including drawings, calculations, procedures, design bases documents, tests and modifications. This review was performed to ensure the pumps were capable of meeting their design flowrates, with consideration of allowable pump degradation, net positive suction head requirements, and potential for water hammer induced piping failure due to the potential loss of the keep-fill system. The team interviewed engineers to assess the current condition of the pumps and reviewed system health and issue reports. The team performed walkdowns of the ESW pump area to assess the general condition of the pumps. Additionally, the team reviewed the condition monitoring and replacement criteria of the ESW piping.

b. Findings

No findings of significance were identified.

.2.1.7 Containment Spray Heat Exchanger, H-21-1A

a. Inspection Scope

Containment spray heat exchanger, H-21-1A, was reviewed to verify its ability to meet its design basis heat removal requirement in response to transient and accident events. There are four containment spray heat exchangers, consisting of two 50% heat exchangers in parallel in each of the two containment spray trains. The heat exchangers are cooled by the ESW system. The team verified that design inputs were properly translated into system procedures and tests, and reviewed completed thermal performance tests intended to demonstrate heat exchanger operabiliy. The team reviewed the maintenance history, design changes, issue reports, calculations, design specifications, drawings and surveillance testing to ensure that the heat exchanger heat removal capability was consistent with accident analyses assumptions. The team also performed walkdowns of accessible areas to assess the current material condition of the heat exchanger.

b. Findings

No findings of significance were identified.

.2.1.8 Containment Hardened Vent Valve, V-23-358

a. Inspection Scope

The containment hardened vent valve was reviewed to verify its ability to operate if called upon in the emergency operating procedures. The vent valve is manually opened to allow operators to vent primary containment during severe accidents which involve the loss of decay heat removal. The team reviewed the maintenance history, design changes, issue reports, drawings, and associated surveillance testing to ensure the valve was capable of performing its function. The team also performed a walkdown of accessible areas to assess the current material condition of the valve.

b. Findings

No findings of significance were identified.

.2.1.9 Electromagnetic Relief Valve (EMRV), V-1-173

a. Inspection Scope

The ‘A’ electromagnetic relief valve was reviewed as a representative sample of the EMRVs to verify its ability to meet design basis requirements in response to transient and accident events, including automatic reactor depressurization conditions. The EMRVs are pilot-operated to automatically open at a specified reactor pressure concurrent with core spray pump running logic being satisfied. The valves also can be manually opened from a remote switch. The team verified that instrument setpoints were properly translated into system procedures and tests and reviewed completed tests intended to demonstrate component operability. The team reviewed the maintenance history, design changes, issue reports, calculations, design specifications, drawings, and surveillance testing to ensure the valve operation was consistent with accident analyses assumptions. The team also reviewed the voltage drop calculations to verify that adequate voltage was provided to the valve under worst-case conditions.

b. Findings

No findings of significance were identified.

.2.1.10 Drywell To Torus Vacuum Breaker, V-26-1

a. Inspection Scope

The drywell-to-torus vacuum breaker, V-26-1, was reviewed to verify its ability to meet the design basis requirement to prevent drywell pressure from dropping below the pressure in the suppression pool space. The team verified that design inputs were properly translated into system procedures and surveillance tests, and reviewed completed tests intended to demonstrate component operability.

b. Findings

No findings of significance were identified.

.2.1.11 4.16 kV Buses, 1C and 1D

a. Inspection Scope

The team reviewed the design of the 4kV buses to verify their ability to supply electrical power to reactor plant auxiliary equipment during normal plant operating conditions and to emergency core cooling system (ECCS) components under accident conditions. The team reviewed the logic and controls for the fast transfer scheme of the buses from the unit auxiliary transformer to the start-up transformers. Bus loading and the use of bus ties between the buses were also reviewed. The team reviewed the incoming source circuit breaker's protective devices to ensure adequate coordination existed with both upstream and load breakers. The short circuit and voltage drop calculations were reviewed to determine if the circuit breakers were adequately rated, and if adequate voltage was available to the associated buses. The team reviewed the voltage setting of the grid undervoltage relays to determine if the relays would cause bus separation during accident loading. The protection guidelines for 4kV motors and their relay settings were reviewed, and setpoints were evaluated to determine if adequate coordination existed. These setpoints were then verified by walkdown to check for consistency of as-built with design values. The history of cable failures was revieweed along with the criteria for cable replacement and testing programs. The switchgear were walked down to determine their material condition and the cable spreading room was walked down to verify the absence of high pressure piping.

b. Findings

No findings of significance were identified.

.2.1.12 Emergency Diesel Generator (EDG) 2, M-39-2

a. Inspection Scope

The team reviewed the ratings of the EDG to determine if the rating used in the loading calculations was consistent with the manufacturer's rating and the licensing bases. The loading calculations were also reviewed to determine if all loads that could be connected to the EDG had been evaluated. The tolerance and effects of higher frequencies on EDG loading was reviewed with respect to load margins. The voltage and frequency response of the EDG following step loading was reviewed to determine compliance with the licensing bases and to verify if load applications could overlap due to timer inaccuracies. The team also reviewed the EDG protection design scheme and protective features and trips. EDG surveillance results were reviewed to determine if the tests were conducted in accordance with the technical specifications, and if the results met the acceptance criteria. A walkdown of the EDG was performed to determine the material condition of the unit and the EDG auxiliaries.

b. Findings

No findings of significance were identified.

.2.1.13 480 Volt Alternating Current (Vac) Unit Substation, 1B2

a. Inspection Scope

The team reviewed short circuit and voltage drop studies to verify the short circuit capability of the circuit breakers and the adequacy of voltage for the 480 Vac Unit Substation, 1B2, under worst case conditions. The 4160-480 Vac transformer tap position was field verified to determine consistency with the tap position assumed in the calculations of record. The team reviewed circuit breaker selective tripping and coordination between incoming and feeder breakers, and also for the breakers associated with the transfer switches between redundant safety buses to ensure that a transfer switch failure would not cause the loss of both redundant buses. Additionally, the team performed a walkdown to determine the material condition of the unit substation.

b. Findings

No findings of significance were identified.

.2.1.14 Station Blackout Transformer, 1B0 (Bank 3)

a. Inspection Scope

The team reviewed calculations, drawings, maintenance procedures and vendor data to ensure that the station black-out (SBO) transformer function to supply power from the combustion turbines to an engineered safety bus was adequately designed and maintained. Specifically, the team reviewed load flow calculations and the SBO evaluation to determine if the loading of the transformer was within its rating. The team also reviewed the design of protective relaying for the associated bus and transformer to determine whether the equipment was properly protected, and not subject to spurious tripping under expected transient and steady state loading conditions. The team reviewed vendor manual and surveillance test results to verify that applicable test acceptance criteria and test frequency requirements were met.

b. Findings

No findings of significance were identified.

.2.1.15 Emergency Diesel Generator (EDG) 2 Battery

a. Inspection Scope

The team reviewed the EDG-2 station battery design calculations to verify that the battery sizing would satisfy the requirements of the safety related emergency diesel generator No. 2, and that the minimum possible voltage was taken into account. In particular, the evaluation focused on verifying that the battery was adequately sized to start the EDG under design conditions, and that adequate voltage would remain available for the individual loads required to operate. Plant drawings were reviewed to ensure that all loads were considered. Additionally, a walkdown was performed to evaluate the material condition of the battery and battery charger. The team reviewed battery test procedures and results to determine whether test acceptance criteria and frequency requirements specified in technical specifications and appropriate standards were satisfied. Engineers were interviewed regarding design aspects and operating history for the battery, and a sample of condition reports were selected to verify that design and testing issues related to the EDG-2 battery were adequately addressed.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” in that AmerGen did not incorporate the requirements and acceptance limits contained in applicable design documents into the EDG battery service test procedures. Specifically, the design requirement of the EDG batteries to supply adequate voltage to the EDG output breakers was not incorporated into the service test load profile for the EDG batteries. This prevented verification within the test of the capability of the batteries to close the output breakers which is a design requirement during events with a postulated loss-of-offsite power.

Description: The EDGs at Oyster Creek each have a dedicated battery to provide starting power. Each EDG battery powers two starter motors for starting up the associated EDG, a few auxiliary loads, and the control power to close the associated EDG output breaker. The output breaker is required to close to provide power to the safety related 4kV AC buses and the associated EDG battery charger. The starter motors require a large amount of current with a low minimum voltage, while the output breakers require a small amount of current with a high minimum voltage (nominal 90 Vdc). Therefore, the limiting design conditions are to provide sufficient current to energize the starter motors and sufficient voltage to close the output breaker. The EDG battery sizing calculation conservatively addressed the ability to supply sufficient current to the starter motors, but did not address the requirement to supply sufficient voltage to close the output breaker.

The team noted that technical specification (TS) section 4.7.B.4, requires that, "at least once per 12 months, the diesel generator battery capacity shall be demonstrated to be able to supply the design duty loads (diesel start) during a battery service test." IEEE 450-1995, "Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," states that a service test is "a test, in the as found condition, of the battery's capability to satisfy the battery duty cycle." IEEE 450-1995 also states that, "trending battery voltage during the critical periods of the load duty cycle will provide the user with a means of predicting when the battery will no longer meet design requirements." The TS required EDG battery service test is performed by monitoring the battery during a biweekly EDG load test. However, the team noted that during these load tests the battery charger is restored to the battery prior to closure of the output breaker. Consequently, the team determined that the service test does not fully envelope the output breaker closure which is the most limiting voltage requirement for completing the EDG function of supplying the associated safety bus. The team determined that there was no acceptance criteria or trending for the most critical voltage in the EDG battery load profile.

In response to the team's concerns, AmerGen performed a preliminary evaluation during the inspection which determined that the EDG batteries remained operable. The team agreed with this conclusion based on the age of the batteries, extensive testing done by the licensee in 1993, which showed sufficient capacity for numerous starts of the starter motors, and acceptable results in the other EDG battery surveillances. Corrective action program issue report (IR) 630434 was written to revise the sizing calculation and IR 630284 was written to incorporate the design requirements into the service test procedure.

Analysis: The performance deficiency associated with this finding was that AmerGen did not incorporate the requirements and acceptance limits contained in applicable design documents into the EDG battery service test procedures. This resulted in the service tests being performed without adequate acceptance criteria for verifying the design capability of the EDG batteries. The finding was greater than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstones objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The team assessed this finding in accordance with NRC Manual Chapter 0609, Appendix A, "Significance Determination Process (SDP) for Reactor Inspection Findings for At-Power Situations," Phase 1, and determined that it was of very low safety significance (Green) since it did not result in a loss of the safety function of the EDF batteries.

Enforcement: 10 CFR 50 Appendix B, Criterion XI, "Test Control," requires, in part, that written test procedures incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, as of May 15, 2007, AmerGen did not incorporate the requirement of the EDG batteries to provide sufficient voltage to close the EDG output breakers into the EDG battery service test procedures. Because the finding was of very low safety significance and has been entered into AmerGen's corrective action program (IR 630284), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000219/2007006-01, Inadequate Acceptance Criteria In Emergency Diesel Generator Battery Service Test Procedures)**

.2.1.16 125 VDC B & C Station Batteries

a. Inspection Scope

The team reviewed the B and C station battery calculations to verify that the sizing of the batteries would satisfy the requirements of the safety related and risk significant direct current (DC) loads, and that the minimum possible voltage was taken into account. In particular, the evaluation focused on verifying that the batteries were adequately sized to supply the design duty cycle of the 125 Vdc systems for the loss-of-coolant accident/loss-of-offsite power and station blackout loading scenarios. The team reviewed the scenarios, battery sizing calculations, and voltage drop calculations to ensure that adequate voltage would remain available for the individual loads required to operate during the scenario durations. Plant drawings were reviewed to ensure that all loads were considered. The B and C station battery charger sizing calculations were reviewed to verify consistency with the design and licensing bases. The team reviewed the DC protective coordination study to verify that adequate protection exists for postulated faults in the DC system. Additionally, a walkdown was performed to evaluate the material condition of the battery and battery charger. The team reviewed battery surveillance test procedures and results to determine whether test acceptance criteria and frequency requirements specified in technical specifications and appropriate standards were satisfied. Engineers were interviewed regarding design aspects and operating history of the batteries, and a sample of condition reports was selected to verify that design and testing issues related to the batteries were adequately addressed.

b. Findings

No findings of significance were identified.

.2.2 Detailed Operator Action Reviews (4 Samples)

The team assessed manual operator actions and selected a sample of four operator actions for detailed review based upon risk significance, time urgency, and factors affecting the likelihood of human error. The operator actions were selected from a PRA ranking of operator action importance based on RAW and RRW values. The non-PRA considerations in the selection process included the following factors:

- Margin between the time needed to complete the actions and the time available prior to adverse reactor consequences;
- Complexity of the actions;
- Reliability and/or redundancy of components associated with the actions;
- Extent of actions to be performed outside of the control room;
- Procedural guidance; and
- Training.

.2.2.1 Alternating Current (AC) Power Recovery

a. Inspection Scope

The team selected operator actions to recover AC power to the safety related buses via various sources, which included the alternate AC (AAC) power source (combustion turbines 1 and 2) and the emergency diesel generators (EDG). The potential consequences of failure of the associated operator actions include core damage. The incorporation of these actions into station procedures, classroom training, and simulator training was reviewed to assess operator knowledge and expertise for the actions. The team also walked down portions of the switchyard, the AAC power source, the EDGs, and the electrical buses/switchgear with operations and engineering representatives to verify that AmerGen could restore AC power within the analyzed time frames. The team observed an operating crew respond to and implement procedures for a loss of AC scenario in the simulator. Finally, a sample of incident reports was selected to verify that related issues were adequately addressed.

b. Findings

No findings of significance were identified.

.2.2.2 Containment Venting

a. Inspection Scope

The team selected the operator action to locally align valves associated with the containment hardened vent path, and the related actions to remotely operate specific containment vent valves. The Oyster Creek Nuclear Power Plant Probabilistic Safety Assessment, Appendix H, Human Reliability Analysis Notebook, considered the operator actions as moderate to high stress actions. While the use of the containment hardened vent would not be required until many hours following an initiating event (about 19 hours depending on equipment failure scenarios), the required time to diagnose and perform the actions is about 35 minutes. The team reviewed the incorporation of the operator actions into emergency and abnormal operating procedures, job performance measures, and training. The team observed an operations representative locate the required valves and controls and walk through the actions to locally operate the containment vent valves. The team also reviewed related design basis documents to verify whether required parameters were appropriately translated into specifications and procedures. Finally, a sample of incident reports was selected to verify that related issues were adequately addressed.

b. Findings

Introduction: The team identified a finding of very low safety significance (Green), in that the licensee did not correctly translate the design of the containment hardened vent valve accumulators into test procedures. Specifically, the acceptance criteria within procedure, PM 24201I, for allowable accumulator pressure drop was not consistent with the original design criteria and did not ensure the assumed design capability of the valves during loss of instrument air events. The valves provide a method of permitting a controlled depressurization of primary containment during severe accident sequences that involve loss of decay heat removal.

Description: The team noted that as part of a comprehensive plan for closing out severe accident issues, the Nuclear Regulatory Commission had issued Generic Letter 89-16, “Installation of Hardened Wetwell Vent,” which requested the installation of a hardened vent system to allow operators a method to avoid exceeding the primary containment pressure limit for postulated events where it may be required. In response to this, the licensee had modified the containment vent and purge subsystem to provide a hardened vent path designed to permit a controlled depressurization of containment during severe accident sequences which involve the loss of decay heat removal capability. The modification, in part, installed air accumulators for both the drywell (V-23-13&14) and torus (V-23-15&16) vent valves to provide the capability of operating the valves during scenarios which included loss of plant instrument air events. These valves were originally designed as containment isolation valves and this safety function was not affected by the installation of the accumulators. The team noted that the nuclear plant operator lesson plan for primary containment, module 2611-PGD-2621, reflected the capability to vent the primary containment on loss of instrument air for a maximum of 6 venting operations in 24 hours.

The team determined that the accumulators were sized in accordance with calculation, C-1302-242-5360-012, “accumulator sizing for V-23-13,14,15&16,” to cycle the hardened vent valves 6 times within a 24 hour period. The intent of the accumulator modification as delineated in modification package MDD-OC-822-A, Rev. 2, “Hardened Vent System,” was to ensure an adequate air supply for a 24 hour period. The team determined that the original modification package tested the modification and had appropriate and conservative acceptance criteria to ensure the valves would be capable of stroking six times over a 24 hour period without instrument air. The original pressure drop criteria was 0.8 psig in 30 minutes which corresponded to a nominal 40 psi drop in 24 hours, thereby ensuring continued functionality of the valves. The team noted that subsequent to the modification, the licensee created a procedure, PM24201I, to periodically test the accumulators/check valves for leakage.

The team identified that the testing procedure established an acceptance criteria of a 1.5 psig drop in 10 minutes which is well above the original modification design leakage limit of 0.8 psig in 30 minutes. Therefore, the team concluded the periodic testing did not verify the original design capability of the valves to perform their intended function. In fact, given this allowable leak rate, pressure would decay much more rapidly than the original intent of the design. The team reviewed past test data from October 2004 to April 2007, and noted that 3 of the 4 valves failed their current non-conservative as found allowable leakage criteria which indicated that the actual timeframe the valves would be able to function would be less than that assumed in the original design. AmerGen has entered this into their corrective action process (IR 00631071), to revise the test criteria to be consistent with the original design of the valve accumulators.

Analysis: The team determined this issue was a performance deficiency because the licensee did not correctly translate the design of the containment hardened vent valve accumulators into test procedures. Specifically, the acceptance criteria of the periodic test was non-conservative with respect to the original design and capability of the valves.

This finding is more than minor because it is associated with the procedure quality attribute of the mitigating systems cornerstone, and affected the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the team conducted a Phase 1 SDP screening and conservatively determined a more detailed Phase 2 SDP evaluation was required to assess the safety significance because the finding resulted in a loss of the containment vent function under certain scenarios. The team considered this to be a conservative assumption because the existing test criteria would still allow for the capability to stroke the valves for some amount of time before loss of air pressure. The team noted that the applicable transient worksheets were the loss of offsite power and loss of instrument air because those were the only two scenarios where the accumulator function was considered for containment venting in the SDP worksheets. Using the Oyster Creek Nuclear Generating Station Risk Informed Inspection Notebook, revision 2, this finding was determined to be of very low safety significance (Green). The dominant core damage sequences involved loss of instrument air initiating events with a coincident stuck open electromatic relief valve and loss of containment heat removal.

Additionally, no credit for instrument air recovery was reflected to further increase the conservatism of the risk assessment.

Enforcement: This finding was not a violation of NRC requirements, in that the performance deficiency involved beyond design basis event scenarios. AmerGen entered this problem into their corrective action program as IR 631071.
(FIN 05000219/2007006-02, Inadequate Containment Hardened Vent Valve Accumulator Test Criteria)

.2.2.3 Manually Provide Isolation Condenser Makeup

a. Inspection Scope

The team selected the operator action to manually provide water inventory makeup to the isolation condensers (IC), given an IC initiation signal and inventory depletion. This action is time critical as operation of the ICs depletes the initial stored volume of water inventory within 45 minutes of initiation, and manual IC inventory makeup is required for continued IC operation. The team reviewed emergency, abnormal and annunciator response procedures to verify that the instructions to perform the necessary manual actions were appropriate. The team also reviewed licensed operator training plans to verify that operators would understand the various options for inventory makeup, including any restrictions or limitations associated with any of the options. The team observed IC operation at the Oyster Creek simulator to assess inventory makeup guidance and options available to the operators. The team also walked down the operator actions with an operations representative to assess the capability of operators to perform the required actions and to verify the availability of equipment and tools necessary to successfully complete the actions. Finally, a sample of incident reports was selected to verify that related issues were adequately addressed.

b. Findings

No findings of significance were identified.

.2.2.4 Start/Align Combustion Turbines Following a Station Blackout

a. Inspection Scope

The team selected the operator action to manually start and align the combustion turbines in the event of a station blackout (SBO) event. The associated actions are time critical (one hour), and the Oyster Creek Nuclear Power Plant Probabilistic Safety Assessment, Appendix H, Human Reliability Analysis Notebook, considered the operator actions as extremely high stress actions. The team observed an operations representative walk through the actions to align alternate AC electrical power to the station via the SBO transformer. The incorporation of this action into site procedures, classroom training, and job performance measures were also reviewed. The team also interviewed operations and engineering personnel and reviewed system health reports to assess whether there were any adverse maintenance or performance trends with the related equipment. Finally, a sample of incident reports was selected to verify that related issues were adequately addressed.

b. Findings

No findings of significance were identified.

.3 Review of Industry Operating Experience (OE) and Generic Issues (5 Samples)

a. Inspection Scope

The team reviewed selected OE issues for applicability at the Oyster Creek generating station. The team performed a review of the OE issues listed below to verify that the licensee had appropriately assessed potential applicability to site equipment and implemented corrective actions as required.

NRC Information Notice (IN) 2006-21: Entrainment of Air into Emergency Core Cooling and Containment Spray Systems

The team assessed AmerGen's review and disposition of NRC IN 2006-21. The basis of the information notice was a concern for circumstances that could result in air entrainment in pump suction lines, potentially affecting the operability of emergency core cooling system pumps. The team reviewed AmerGen's evaluation for potential vortexing in the ECCS pump suction lines from the torus. This review included verifying that the inputs and assumptions used in the original evaluation remained valid.

NRC Information Notice (IN) 1995-37: Inadequate Offsite Power System Voltages During Design Basis Events

The team reviewed the applicability and disposition of NRC IN 95-37. The basis of IN 95-37 was a concern for circumstances that could result in inadequate offsite power system voltages during design basis events. The failure to periodically update the original voltage analyses as the result of changing offsite grid or plant conditions could result in unintentional operation outside regulatory requirements. The Information Notice had identified the potential that the setpoints of undervoltage protection relays may require changes to ensure adequate voltages at the terminals of all safety related equipment. The team reviewed the licensee's identification and disposition of the concerns identified in the information notice.

Service Information Letter (SIL) 299: High Drywell Temperature Effect on Reactor Vessel Water Level Instrumentation

The team assessed the licensee's review and disposition of SIL 299, including supplements 1 and 2. SIL 299 was issued by the vendor related to a concern of potential inaccuracy on reactor vessel water level instrumentation which would occur under the unusual condition of very high drywell temperature characterized by accident conditions. The inaccuracy is a result of heating up instrument reference legs in the drywell resulting in density differences in the water column in the instrument taps. The team reviewed the licensee's evaluation of level trip setpoints and instrument scale readings. This review included verifying level instrument inaccuracies were noted in control room level indication, and verifying that plant operators were aware of potential level inaccuracies under high drywell temperature conditions.

NRC Generic Letter (GL) 1996-05: Periodic Verification of Design-Basis Capability of Safety Related Motor Operated Valves

The team reviewed the applicability and disposition of GL 96-05 with respect to a sample of valves. The NRC issued this letter to request that licensee's perform or confirm that they previously performed: (1) establishment of a program, or ensure the effectiveness of its current program; and (2) verification on a periodic basis that safety related MOVs continue to be capable of performing their safety functions within the current licensing bases of the facility. The team sampled the isolation condenser return valve V-14-34, and core spray valve V-20-15 with respect to the methodology implemented by the licensee for the periodic verification program. This review included verifying that the testing periodicity remained valid based on a review of the latest valve test data.

NRC Bulletin 1988-04: Potential Safety Related Pump Loss

The team reviewed the applicability and disposition of Bulletin 88-04. This bulletin described conditions where potential design deficiencies in the minimum flow lines and interactions between pumps running in parallel in a system, may lead to permanent pump damage due to prolonged operation at or near shutoff head conditions. The review included verifying that the pump vendor evaluated the minimum flowrate requirements for the main and booster core spray pumps, and that the potential for pump-to-pump interaction for parallel pumps was addressed. The review also verified that procedural cautions were in place to make operators aware of time limits when pumps are operated under minimum flowrate conditions.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

a. Inspection Scope

The team reviewed a sample of problems that were identified by the licensee and entered into the corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design or qualification issues. In addition, Issue Reports (IRs) written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4AO6 Meetings, Including Exit

Exit Meeting Summary

On May 24, 2007, the team presented the inspection results to Mr. T. Rausch, Site Vice President, and other members of the Oyster Creek Generating Station staff. The team verified that no proprietary information is documented in the report.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

A. Agarwal, Electrical Design Engineer
D. Barnes, Manager, Electrical Design
R. Brown, Procedure Writer Supervisor
T. Carroll, MOV Engineer
M. Godknecht, Programs Engineer
M. Heck, System Engineer
J. Kandasamy, Manager, Regulatory Assurance
D. Kettering, Director, Engineering
R. Pruthi, Electrical Design Engineer
T. Rausch, Site Vice President
S. Schwartz, System Engineer
P. Tamburro, Design Engineer
D. Yatko, Electrical Design Engineer

NRC Personnel

W. Cook, Region I Senior Risk Analyst
M. Ferdas, Senior Resident Inspector
R. Treadway, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000219/2007006-01	NCV	Inadequate Acceptance Criteria In Emergency Diesel Generator Battery Service Test Procedures (Section 1R21.2.1.15)
05000219/2007006-02	FIN	Inadequate Containment Hardened Vent Valve Accumulator Test Criteria (Section 1R21.2.2.2)

LIST OF DOCUMENTS REVIEWED

Calculations

125-1-303-93, Diesel Hot Engine Restart at Minimum Battery Voltage, 3/25/93
C-1302-211-E540-104, PPM Thrust Calculation for Isolation Condenser Valves, Rev. 0
C-1302-212-5310-091, Low Pressure Core Spray Operability Criteria, Rev. 1
C-1302-212-5450-027, Vortex Limit for Core Spray, Rev. 3
C-1302-241-E610-074, Core Spray NPSH, Rev. 2
C-1302-241-E610-080, Calculation of Torus Pool Temperature as NPSH Input, Rev. 3
C-1302-241-E120-085, Containment Spray System Heat Exchanger Performance Evaluation, Rev. 1
C-1302-241-E120-078, Containment Spray Heat Exchanger Performance Evaluation, Rev. 0
C-1302-241-5450-073, Acceptable Containment Spray Heat Exchanger Fouling Resistance, Rev. 0
C-1302-241-E540-096, OCNGS Containment Spray/Emergency Service Water System Pressure Profile, Rev. 1
C-1302-242-5360-012, Accumulator Sizing for V-23-13, 14, 15 & 16, Rev. 0
C-1302-243-5310-047, OCNGS 18 inch Torus-to-Drywell Vacuum Breaker- Verification of Test Procedure 604.4.016, Rev. 0
C-1302-243-5450-062, Containment Vent Valve Cycles at Beyond Design Basis Accidents, Rev. 0
C-1302-532-5450-019, OC ESW Pump and Piping Model With Seismic Related Breaks, Rev. 1
C-1302-700-5350-004, OC Electrical Model, Rev. 3
C-1302-700-5350-012, OC Short Circuit Study, Rev. 2
C-1302-730-5350-008, GL 89-10 MOVs Voltage Drop Calculation for DC MOVs, Rev. 4
C-1302-732-E510-015, OC Degraded Grid Undervoltage Relay Setpoint Evaluation, Rev. 4
C-1302-735-5350-008, Battery Bus Coordination, Rev. 4
C-1302-735-E320-037, 125 VDC Operated EMRV/ADS Scenario, Rev. 2
C-1302-735-E320-038, 125 VDC Operated EMRV High Pressure Relief, Rev.3
C-1302-735-E320-040, Station Battery A, B&C Capacity Calculation, Rev. 2
C-1302-735-E320-044, 125 VDC Voltage Drop, Rev. 2A
C-1302-735-E320-047, Station Battery B and C Capacity for SBO, Rev. 1
C-1302-735-E320-048, Mechanical Timeline Input for DC Battery Calculation, Rev. 0
C-1302-735-E320-049, B and C Station Battery Sizing Calculation, Rev. 0
C-1302-741-5350-009, EDG Battery Sizing Calculation, Rev. 1
C-1302-741-5350-001, Appendix A1, Diesel Generator Ratings, Rev. 10
C-1302-743-5350-002, Sizing of SBO Transformer to Start Reactor Feedpump, Rev. 1
C-1302-822-5320-037, Hardened Vent Isolation Valves (V-23-13/16), Rev. 0
C-1302-822-5450-050, Reactor Building Corner Room Response to Electrical Heat Loads During LOCA, Rev. 1
C-1302-862-5360-002, Diesel Generator Fuel Requirements, Rev. 4
C-1302-862-E310-007, Diesel Generator Fuel Transfer Pump Inlet Pressure, Rev. 0
C-1302-900-E540-013, MOV Delta P and Basis, GL 89-10, Rev. 2
C-1302-5350-013, ADS Fuse/Breaker Coordination, 12/30/93
C-3731-71-11-002, Appendix R-Condenser Shell Makeup Valve Accumulator Volume, Rev. 1

CC-AA-309-1001, Solid State Trip Device Settings for 480V Unit Substation Circuit Breakers, Rev. 2
EXOC005-CALC-001, Suction Strainer Effects on Core Spray Pump NPSH, Rev. 0
EXOC005-CALC-002, System Acceptance Criteria for Containment Spray and ESW Flowrates, Rev. 0
EXOC011-CALC-001, Isolation Condenser Heat Capacity, Rev. 0
V-14-0034, DC Motor Operated Gate Valve Calculation Midacalc Results, Rev. 1
V-14-0035, DC Motor Operated Gate Valve Calculation Midacalc Results, Rev. 1

Surveillance Test Procedures (completed)

602.3.005, ADS Actuation Circuit Test and Calibration (10/31/06)
602.4.001, NSSS Leak Test (11/4/06)
604.1.005, Torus to Drywell Vacuum Breaker, Mechanical Surveillance and Limit Switch Calibration (10/24/06)
604.4.016, Torus to Drywell Vacuum Breaker Operability and In-Service Test (11/7/06 and 3/12/07)
607.4.016, Containment Spray and ESW System Pump Operability and Quarterly IST (10/5/06 and 5/16/07)
607.4.021, Core Spray System 1 Pump Operability and Quarterly IST (2/22/06, 6/2/06, and 9/6/06)
609.1.005, Isolation Condenser Isolation Valve Inspection (11/7/06)
609.4.001, Isolation Condenser Valve Operability and IST (11/4/06, 1/4/07, and 4/3/07)
610.4.012, Core Spray System 1 Comprehensive/Preservice Pump IST (12/01/06)
610.4.013, Core Spray System 2 Comprehensive/Preservice Pump IST (2/15/07)
634.2.002, Main Station Battery Weekly (2/6/07, 2/13/07, 2/20/07, and 2/27/07)
634.2.003, Main Station Battery Quarterly (6/13/06, 8/15/06, 9/12/06, 11/14/06, 12/12/06, and 2/13/07)
634.2.011, Main Station Battery Monthly (10/10/06, 11/14/06, 12/12/06, 1/9/07, and 2/6/07)
634.2.201, Main Station B Battery Discharge (10/31/00, 10/13/02, and 10/28/06)
634.2.207, Main Station B Battery Service Test (10/26/00, 10/15/02, and 2/16/05)
634.2.301, Main Station C Battery Discharge (10/17/06)
636.2.004, Diesel Generator Battery Discharge (2/13/03 and 5/19/05)
636.2.005, Diesel Generator #1 Weekly (2/4/07, 2/11/07, and 2/18/07)
636.2.007, Diesel Generator #2 Weekly (2/4/07, 2/11/07, and 2/18/07)
636.2.008, Diesel Generator #2 Quarterly (8/27/06, 11/27/06, and 2/25/07)
636.2.012, Diesel Generator #1 Battery Service (2/5/02)
636.2.013, Diesel Generator #2 Battery Service (8/1/05 and 8/14/06)
636.4.013, Diesel Generator Number 2 Load Test (7/3/06, 7/17/06, 7/31/06, and 8/28/06)
678.4.001, Primary Containment Isolation Valve Operability and IST (1/10/07)
678.4.005, Station Blackout Functional Test, Rev. 13, (11/17/04)
678.4.005, Station Blackout Functional Test, Rev. 15, (11/06/06)
680.4.009, Remote Shutdown Panel Functional Test for Control Power Transfer and Isolation Condenser Valves (1/5/07)
681.4.004, Technical Specification Log Sheet (3/8/07)
2400-SMM-3228.05, Torus to Drywell Vacuum Breaker Inspection/Repair (11/4/06)

Completed Work Orders

R0805533, R0803679, R2015592, R2025648, R2063762, R2060815, R2061060, R2074750, R2097152, R2095697, R2094121

Issue Reports

373040	544326	618891*	624797*	631824*
386929	544328	619239*	624950*	631967*
389285	545301	620718*	625029*	632222*
391449	545677	621500*	625071*	632265*
427640	549254	622261*	625120*	632276*
435168	552152	622816*	625138*	632339*
441492	554396	623709*	625325*	632340*
441519	554400	623711*	626806*	632365*
467848	554402	623714*	627399*	632374*
468426	566545	623716*	630284*	632449*
494633	572902	623717*	630288*	632485*
498447	573692	623718*	630307*	632515*
498484	581847	624063*	630434*	632719*
510096	589739	624174*	631025*	
540059	608121	624177*	631071*	
541029	618853*	624493*	631090*	

* Issue Report written as a result of inspection effort

Design Basis Documents

SDBD-OC-532, Emergency Service Water, Rev. 4
SDBD-OC-241, Containment Spray System, Rev. 5
SDBD-OC-243, Design Basis Document for Containment Spray, Rev. 0
SDBD-OC-740, Emergency Power System, Rev. 1

Drawings

BR 2004, Condensate Transfer System Flow Diagram, Rev. 84
BR 2005, Sheet 4, ESW Flow Diagram, Rev. 77
BR 3000, Electric Power System Key One Line Diagram, Rev. 10
BR 3001, Main One Line Diagram Aux, Startup, and Main Generator, Rev. 9
BR 3001, Sht 2, Emergency Power System, Rev. 4
BR 3001A, One Line Diagram, 4160 V Bus 1A, Rev. 10
BR 3001B, One Line Diagram, 4160 V Bus 1A, Rev. 16
BR 3001C, 4160 V System One Line Diagram, Rev. 0
BR 3002, 480 V System One Line Diagram 480 V Unit Substation 1A1 & 1B1, Rev. 12
BR 3002, 480 V System One Line Diagram 480 V Unit Substation 1A2 & 1B2, Rev. 11
BR 3002, 480 V System One Line Diagram, 480 V Unit Substation 1A3 & 1B3, Rev. 8
BR 3028, 125 VDC Distribution Centers A&B, Rev. 15

BR E1102, Emergency Condenser Electrical Elementary Diagram, Rev. 15
EB D-3033, 125 VDC Distribution Center C, Rev. 31
EM 8393039, EDG #1 Elementary Wiring Diagram DG AC &DC Auxiliaries, Sh. 7, Rev. 4
EM 8393039, EDG #1 Elementary Wiring Diagram DG DC Control Ckts. Sh. 3, Rev. 11
EM 8393039, EDG #1 DC Switchgear DC Control Ckt. Sh. 4, Rev. 4
EM 8397907, Emergency Diesel Generator #2 Electrical Elementary Diagram, Rev. 11
GE 116B8328, Core Spray System, Core Spray Booster Pump NZ03-C, P-20-002C, Rev. 19
GE 148F262, Emergency Condenser Flow Diagram, Rev. 51
GE 148F740, Containment Spray System Flow Diagram, Rev. 43
GE 223R0173, Core Spray System, Core Spray Pump NZ01-A, P-20-001B, Sh. 18, Rev. 24
GE 223R0173, Core Spray System, Core Spray Pump NZ01-B, P-20-001B, Sh. 24, Rev. 24
GE 223R0173, Core Spray System, Core Spray Pump NZ01-C, P-20-001C, Sh. 25, Rev. 22
GE 223R0173, Core Spray System, Core Spray Pump NZ01-D, P-20-001D, Sh. 26, Rev. 20
GE 729E182, Auto Depressurization System Electrical Elementary Diagram, Rev. 33
GE 885D781, Core Spray System Flow Diagram, Rev. 71
GU 3C-735-11-002, 125 VDC Power Panel DC-E, Rev. 5
GU 3C-735-11-001, 125 VDC Power Panel DC-D, Rev. 3
P-6-50-00, Oyster Creek Substation, 230 kV Single Line Diagram, Rev. 11
P7-50-00, Forked River Substation Single Line Diagram, Rev. 1
3D-711-17-001, Main Generator Protection Relays, Generator LO Relay, Rev. 4
3D-822-22-001, Hardened Vent System Modification, Rev. 2
3E-243-21-1000, Drywell and Torus Vacuum Relief System, Rev. 28
3E-532-A2-1000, ISI Configuration Drawing, Emergency Service Water, Rev. 5
3E-532-A3-1000, Pipe Integrity Inspection Program, Emergency Service Water, Rev. 1
3E-743-11-001, One Line Diagram, Station Blackout Transformer, XMR-743-043, Rev. 2
3E-862-21-1000, EDG Diesel Fuel Oil and Transfer System, Rev. 23
3E-861-21-1001, EDG Water Cooling System Flow Diagram, Rev. 10
237E726, Drywell and Suppression System, Rev. 74
9023253, 10"-900 Weld Ends Stainless Steel Double Disc Gate Valve, Limitorque SMB-2-40 Actuator with 2.86:1 SGA, and Drain Pipes, Rev. C
JC 19642, Automatic Transfer Switch IT-3, Rev. 1

Miscellaneous

ANSI/IEEE Std 242-1986, IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems
Combustion Turbine Train 1 & 2 Out of Service Performance Data (October 2003- April 2007)
ER-AA-302-1003, MOV Margin Analysis and Periodic Verification Test Intervals, Rev. 5
ER-OC-300-1001, OC MOV GL 89-10 Scoping Document, Rev. 1
Flowserve Pump Curve T-42584-1, ESW Pump
Functional Test Procedure TP 257/1, V-11-34, V-11-36 Loss of Air Cycle Test, 7/9/86
General Electric Company Report number GE-NE-0000-0001-7486-01P, Oyster Creek Generating Station Loss-of-Coolant Accident Evaluation for GE11, dated July 2002
General Electric Company Report number NEDC-31462P, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTER-LOCA Loss-Of-Coolant Accident Analysis, 1987
General Electric Company Report number GE-NE-B13-02080-00-15, Core Spray Sparger and Piping Flow Evaluation Plan Oyster Creek Nuclear Generating Station, 2000

General Electric Company Report number GE-NE-0000-0006-3699-01P-R1, ECCS-LOCA Evaluation for Oyster Creek With Improved GE9 LHGR Limits, 2002
General Electric Company Report number NEDE-24802, Mark 1 Wetwell-to-Drywell Vacuum Breaker Functional Requirements, Task 9.4.3, April 1980
General Electric Company Design Specification for Containment Spray System, 22A1018, Rev. 2
General Electric Company Design Specification for Emergency Condenser, 21A1607, Rev. 1
GE Spec 21A5404, Tag number NZ-01/A/B/C/D Core Spray Pump Curves, 7/2/65
GL 96-05, Periodic Verification of Design Basis Capability of SR MOV
IEEE Std. 741-1997, IEEE Standard for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations
Job Performance Measure 207.01, Add makeup to B-IC Using Condensate Transfer
Job Performance Measure 223.01, Bypass Isolation Interlock for Torus Vent Valves
Job Performance Measure 223.03, Line Up to Vent the Torus Through the Hardened Vent
Job Performance Measure 286.03, Lineup Firewater to A Isolation Condenser
Letter: GPUN Response to NRC Bulletin 88-04, Potential Safety Related Pump Loss, 7/11/88
Letter: Ingersoll Rand Pumps to GPUN, Subject: Operation of Main Core Spray System Pumps at Minimum Recirculation Flowrate, 7/8/91
Letter from Flowserve to Palas, Subject: Minimum Flow Operation of Flowserve (I-R) Models 8x21AL Core Spray Pumps While Operating in Series With Flowserve (I-R) Model 10X17A Core Spray Booster Pumps-Clarification of Historical Documents, 5/18/07
Licensed Operator Lesson Plan - Emergency Diesel Generators, Rev. 10
Licensed Operator Lesson Plan - Primary Containment, Rev. 8
MA-AA-723-300, Diagnostic Testing of Motor Operated Valves, Rev. 3
MA-AA-723-300-1004, Quiklook Diagnostic Test Equipment/Sensor Guideline, Rev. 1
NEDC-32264-A, Application of Probabilistic Safety Assessment to GL 89-10 Implementation, Rev. 2
NEDC-32719, BWR Owners Group Program on Motor Operated Valve Periodic Verification, Rev. 2
NRC Bulletin 88-04, Potential Safety-Related Pump Loss
OCNGS Station Blackout Agreement, 05/10/00
Operability Evaluation OC-2003-OE-0013, ESW System Leaks, Coating Failure and Wall Failure, Rev. 1
Report for Oyster Creek Reload 21 Cycle 21, September 2006
Safety Evaluation 000212-036, Core Spray System Operability Criteria, Rev. 1
SE-328312-003, GL 89-10 MOV Modification, Rev. 1
Summary of Oyster Creek ESW Refill Analytical Evaluation Study Prelim (5/21/07)
TDR 1099, Station Blackout Evaluation Report, Rev. 3
Technical Data Report (TDR) number 829, Pipe Integrity Inspection Program, Rev. 4
Topical Report 116, Oyster Creek Underground Piping Program Description and Status, Rev. 1
Topical Report 140, ESW and Service Water System Piping Plan, Rev. 2
Review of EMRV Test Results from Wyle Labs, 4/2/98
Vendor Manual VM-OC-0100, EDG Woodward Governing Components
Vendor Manual VM-OC-0095, Emergency Diesel Generators
VM-OC-0096, EDG Starting Motor Description and Details
50.59 Screening OC-2005-S-0382, Procedure 2000-OPS-3024.11 - Emergency Diesel Generator- Diagnostic and Restoration Activities, Rev. 0

6730-96-2342, GPU Letter: GPUN Response to NRC GL 96-05, 11/21/96
2130-03-20058, AmerGen letter: License Amendment Request 306 - DC Electrical Power Sources Based on TSTF-360, 4/21/03

Design Change Packages

MDD-OC-822A, GPUN Modification Design Description for Hardened Vent, Rev. 2
OC-04-00895, V-14-34/35 Modification to Prevent Bonnet Over-Pressurization, 10/29/04

Operations Procedures

307, Isolation Condenser System, Rev. 91
308, ECCS Operation, Rev. 78
316.1, Condensate Transfer System, Rev. 44
338, 480 Volt Electrical System, Rev. 44
609.3.002, Isolation Condenser Test and Calibration- A1/B1, Rev. 54
619.3.004, Primary Containment Isolation Functional Test, Rev. 50
678.4.001, Primary Containment Isolation Valve Operability and IST, Rev. 34
680.4.009, RSP Functional Test for Control Power Transfer and IC Valves, Rev. 7
ABN-35, Loss of Instrument Air, Rev. 1
ABN-36, Loss of Offsite Power, Rev. 4
ABN-37, Station Blackout, Rev. 4
EMG-3200.01A (11), Alternate Pressure Control Systems, Rev. 12
EMG-3200.01A, RPV Control - No ATWS, Rev. 6
EMG-3200.01A (4), Operation of the Core Spray System, Rev. 12
EMG-3200.01B, RPV Control - With ATWS, Rev. 7
EMG-3200.02 (29), Initiation of the Containment Spray System for Drywell Sprays, Rev. 17
EMG-3200.02, Primary Containment Control, Rev. 5
OP-AA-103-105, Limitorque MOV Operations, Rev. 1
RAP-C6a, Shell A Level Hi/Lo, Rev. 1
RAP-C6b, Shell B Level Hi/Lo, Rev. 1
RAP-T1b, Lockout Relay Trip (EDG 1), Rev. 1
RAP-T1f, Lockout Relay Trip (EDG 2), Rev. 1
RAP-T4b, EDG 1 Disabled, Rev. 1
RAP-T5b, EDG 1 Not in Auto, Rev. 1
RAP-T7b, EDG 1 Day Tank Level Hi/Lo/Lube Oil Fail, Rev. 1

System Health Reports

Containment Spray System, 03/07
Core Spray/ADS, 03/07
EDG and EDG Fuel Oil, 03/07
Emergency Isolation Condensers, 03/07
Emergency Service Water System, 03/07
LPI/Decay Heat Removal System, 12/06

Vendor Manuals

VM-OC-0095, Emergency Diesel Generator Starter, 3/20/01
VM-OC-0227, Main Station Battery 125V C Battery, Rev. 0
VM-OC-2256, Station Blackout Transformer Instructions, Rev. 1
VM-OC-2328, Round Cell Batteries, Rev. 5

LIST OF ACRONYMS

AC	Alternating Current
AOV	Air Operated Valve
App	Appendix
DBD	Design Basis Documents
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
GL	Generic Letter
IN	Information Notice
IR	Issue Report
LOCA	Loss of Coolant Accident
MOV	Motor-Operated Valve
NCV	Non-cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
OE	Operating Experience
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
RPV	Reactor Pressure Vessel
RRW	Risk Reduction Worth
SBO	Station Black Out
SDP	Significance Determination Process
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Vac	Volts Alternating Current
Vdc	Volts Direct Current