December 20, 2000

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

SUBJECT: NRC's OYSTER CREEK GENERATING STATION INTEGRATED INSPECTION REPORT 05000219/2000-008

Dear Mr. DeGregorio:

On November 18, 2000, the NRC completed an integrated inspection at your Oyster Creek reactor facility. The enclosed report presents the results of that inspection. The results of this inspection were discussed on December 8, 2000, with Mr. Kevin Mulligan and other members of your staff.

This inspection was an examination of 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC identified five findings that were evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). These findings have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report.

In addition, the NRC has determined that four of these findings are also Severity Level IV violations of NRC requirements. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VI.A.1 of the Enforcement Policy. These NCVs regard the failure to implement a critical procedural step during a reactor start up, the failure to properly control and implement procedures directing welding activities on safety related equipment, the failure to control combustible materials in the turbine building and a violation of technical specification 3.3.C.1, "reactor vessel cool down rate," and are described in this inspection report. If you contest the violations or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region 1; 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).

We appreciate your cooperation. Please contact me at 610 337-5146 if you have any questions regarding this letter.

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch 7
Division of Reactor Projects

Docket/License Nos.: 05000219/DPR-16

Enclosure: NRC Inspection Report No. 05000219/2000-008

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REGION I

Report No.	05000219/2000-008
Docket No.	05000219
License No.	DPR-16
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	September 30, 2000 - November 18, 2000
Inspectors:	Laura A. Dudes, Senior Resident Inspector Thomas R. Hipschman, Resident Inspector Joseph T. Furia, Senior Health Physicist, October 30 - November 3, 2000 Thomas F. Burns, Reactor Engineer, November 13 - November 17, 2000
Approved By:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

Oyster Creek Generating Station
NRC Inspection Report 05000219/2000-008

IR 05000219-00-008; 9/30-11/18/00; Oyster Creek; Fire Protection, In-Service Inspection, Personnel Performance during Non Routine Events, Refueling Outage and Event Follow Up.

The inspection was conducted by resident and region based inspectors. The inspection identified five green issues, four of these issues were Non-Cited violations. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) in Inspection Manual 0609 (see Attachment 1).

Cornerstone: Initiating Events

GREEN. The inspectors determined that the amount of combustible materials stored in the turbine building was not evaluated per procedure 120.5, “Control of Combustibles.” This procedures requires that the fire protection coordinator approve of transient combustible materials that exceed administrative fire load limits to determine if the temporary storage is acceptable and if any additional measures need to be taken. Contrary to the requirements of procedure 120.5, the licensee did not perform an evaluation to permit the amount of combustible materials stored on the turbine building mezzanine. The inspector reviewed this issue in accordance with NRC manual chapter 609 and determined that amount of transient combustibles loaded in the turbine building and the smoking area located near the piles of debris could have contributed to a fire in the area. The inspector evaluated the finding per Appendix F, “Fire Protection Significant Determination Process,” of manual chapter 609. This issue was considered to have very low safety significance (Green). The failure to follow procedure 120.5, “Control of Combustibles,” is a violation of Technical Specification 6.8.1, “Procedures and Programs,” and 10 CFR 50 Appendix B, Criterion V, ”Instructions, Procedures, and Drawings.” However, this violation is being treated as a non-cited violation, consistent with Section VI.A.1 of the Enforcement Policy. The licensee documented this issue in CAP 2000-1920. **(NCV 05000219/2000-008-01)**(Section 1R05)

GREEN. The failure to implement a critical step in procedure 315.1, “Main Turbine Operation,” because of operator knowledge deficiencies and inadequate control room oversight led to an automatic reactor scram. This issue was considered to have very low safety significance (Green) using the Significance Determination Process (SDP) phase 1 evaluation for initiating events because all mitigating systems were available. However, the issue is considered to be substantive with respect to the crosscutting issue of human performance. This is a violation of Technical Specification Section 6.8.1, “Procedures and Programs,” and 10 CFR 50 Appendix B, Criterion V, ”Instructions, Procedures, and Drawings.” Therefore, in accordance with the Section VI.A.1 of the NRC Enforcement Policy and the NRC Significance Determination Process, this issue is considered to be a Non-Cited Violation (Green). This issue has been entered into the licensee’s corrective action program as CAP 2000-1919. **(NCV 05000219/2000-008-04)**(Section 4OA3)

Cornerstone: Mitigating Systems

GREEN. The inspectors identified that the drywell to torus downcomer foreign material exclusion (FME) covers were installed in a manner that was ineffective in preventing foreign material from entering the torus ring header. During a routine walkdown, the inspectors observed foreign material (i.e. hard hat) lodged in the downcomer region. This issue was considered to have very low safety significance (Green) using the Significance Determination Process (SDP) phase 1 evaluation for mitigating systems because there was not an actual loss of safety function, the debris was removed from the drywell and torus areas, and an inspection of these areas was performed prior to reactor start-up. There was no violation of NRC requirements because the licensee complied with the Technical Specifications limiting conditions for operations. (Section 1R20)

Cornerstone: Barrier Integrity

GREEN. The licensee had not adhered to procedure requirements governing the control of special processes (welding) for work performed on the core spray and isolation condenser systems during the 18R refueling outage. Replacement and repair activities were conducted using alternate weld filler metals not specified in the weld procedures. The inspector determined that the risk significance of this issue was very low (Green) because the activities were subsequently determined to be technically acceptable and that boundary integrity was maintained. The licensee entered these issues into their corrective action program. This procedural adherence issue is a violation of Technical Specification, Section 6.8.1, “Procedures and Programs,” and 10 CFR 50, Appendix B, Criterion V, “Instruction, Procedures, and Drawings.” These issues are being treated as a non-cited violation in accordance with the Section VI.A.1 of the NRC Enforcement Policy and the NRC Significance Determination Process. **(NCV 05000219/2000-008-02)** (Section 1R08)

GREEN. The failure to maintain the reactor coolant system cooldown rate within the technical specifications limit of 100 degrees per hour, is a violation of NRC requirements. However, the technical specification bases considers 10 cooldowns exceeding 300° F/hr to be allowable during the lifetime of the facility, and the licensee has not exceeded this. In addition, because this was a depressurization event, the pressure within the reactor vessel followed the saturation curve and the licensee stayed within the pressure/temperature limitations of the reactor vessel. Lastly, the inspector concluded that this issue had a negligible effect on the fatigue usage factors for the reactor vessel components. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the Enforcement Policy. This issue was determined to be of very low safety significance, which resulted in a Green finding. **(NCV 05000219/2000-008-03)** (Section 1R14)

Cross-cutting Issues: Human Performance

NO COLOR Human performance errors were identified in the initiating event and barrier integrity cornerstone areas. Operations personnel exhibited a lack of system knowledge, poor self checking and inadequate shift oversight while performing a reactor start up (Section 1R14 and 4OA3). This led to an automatic reactor scram and subsequent excessive reactor vessel cooldown rate. Also, the inspectors noted poor procedural adherence and self checking issues while implementing the licensee's welding and fire protection procedures (Sections 1R05 and 1R08). The safety significance of these individual events was very low.

Report Details

Summary of Plant Status:

Oyster Creek began the inspection period at approximately 84 percent power. On October 13, 2000, the licensee commenced a planned reactor shutdown and began refueling outage 18. A reactor start up began on November 14, 2000, and the reactor was critical on November 15, 2000. A reactor scram occurred on November 15, 2000, while operators were attempting to warm the main turbine (Section R14 and 4OA3). A second reactor start up began on November 16, 2000. At the end of the inspection period the reactor was at approximately 40 percent power in ascension to full power.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Shutdown Cooling

a. Inspection Scope

The inspector performed a full walkdown of the shutdown cooling system, including the piping and valves located inside the primary containment. The inspection consisted of a full valve verification per licensee procedure 305, "Shutdown Cooling System Operation," and its associated valve and instrument line up attachments. In addition, the inspector reviewed corrective action documents dating back to 1998 to verify that there were no outstanding equipment problems with the system.

b. Issues and Findings

There were no findings identified.

.2 Electromatic and Safety Relief Valves

a. Inspection Scope

The inspectors performed a walkdown of electromatic and safety relief valves located inside the drywell to determine if the valves and associated instrumentation were correctly configured. The inspectors reviewed applicable documents, outstanding work requests and corrective actions relative to electromatic and safety relief valves to verify that there were no outstanding equipment problems with the system.

b. Issues and Findings

There were no findings identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted fire protection inspection activities consisting of plant walkdowns, discussions with fire protection personnel, and reviews of procedure 333, “Plant Fire Protection System,” and the Oyster Creek Fire Hazards Analysis Report. Plant walkdowns included observations of combustible material control, fire detection and suppression equipment availability, and compensatory measures. The inspectors conducted fire protection inspections in the following areas:

- Fire pond and diesel driven fire pumps
- Drywell Elevations 13', 23' and 46' for control of combustible material
- Control room halon system
- 480V switchgear room halon system
- Turbine Building combustible material control

b. Issues and Findings

On November 15, 2000, while performing a walkdown of the turbine building mezzanine, the inspectors noticed a large accumulation of combustible material. The inspectors determined that the amount of transient combustibles stored in the turbine building was not evaluated per procedure 120.5, “Control of Combustibles.” This procedure requires that the fire protection coordinator approve transient combustible materials that exceed administrative fire load limits to determine if the temporary storage is acceptable and if any additional measures need to be taken. Contrary to the requirements of procedure 120.5, the licensee did not perform an evaluation to permit the amount of combustible materials stored on the turbine building mezzanine.

The inspector reviewed this issue in accordance with NRC manual chapter 609 and determined that amount of transient combustibles loaded in the turbine building and the smoking area located near the piles of debris could have contributed to a fire in the area. The inspector evaluated the finding per Appendix F, “Fire Protection Significant Determination Process,” of manual chapter 609. The following assumptions were made in the evaluation:

- A fire ignition frequency (IF) of 4.73 x 10^{-3} per year was taken from the licensee’s IPEEE/PRA data for the mezzanine common areas of the turbine building (Fire Area TB-FZ-11G). Log IF = -2.39794.

- A fire brigade drill was observed and documented as satisfactory in NRC inspection report 05000219-00-007, therefore, manual suppression (MS) was considered to be in its normal operating state (MS = -1.0).

- The fire area was protected by area smoke detection and a partial area sprinkler system. No significant obstructions to the sprinklers were observed. Therefore, no degradation was assigned for automatic suppression (AS) term (AS = -1.25).

- A fire mitigation frequency (FMF) was calculated to be "1 per 10^4 to 10^5 " per year using the formula, FMF = log IF + FB + AS + MS + CC.

- This FMF corresponded to an initiating event likelihood rating of G since the condition existed for less than 3 days. (Table 5.7, Appendix F, MC 609)

The analysis concluded that this issue was of very low safety significance (GREEN). The failure to follow procedure 120.5, "Control of Combustibles," is a violation of Technical Specification 6.8.1, "Procedures and Programs," and 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." However, because this condition was corrected in a timely manner and has been entered into the corrective action program (CAP 2000-1920) this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the Enforcement Policy. **(NCV 05000219/2000-008-01)**

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspector selected samples of nondestructive examination (NDE) and American Society of Mechanical Engineers (ASME) Section XI code repair/replacement activities for evaluation based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in an increase in risk of core damage. Also, the inspector evaluated the effectiveness in the resolution and corrective action of problems identified during Inservice Inspection (ISI) activities. The inspector reviewed a sample of corrective action reports initiated as a result of problems identified during the conduct of ISI examinations.

The inspector reviewed three types of NDE activities including volumetric, surface and visual examinations to verify the effectiveness in monitoring degradation of risk significant systems, structures and components. This review included evaluating the disposition of non-conforming conditions identified in the inspection sample and verifying analyses were performed for acceptance and continued operation without repair. The inspector reviewed the ultrasonic test (UT) reports for reactor pressure vessel (RPV) welds NR-02-N8 and NR-02-2-567A and magnetic particle test (MT) reports for shutdown cooling welds NU-1-004 and NQ-2-0026 for compliance with the requirements of ASME Section XI of the boiler and pressure vessel code. In addition, the inspector reviewed the results of the liquid penetrant testing (PT) and UT inspections performed on the weld overlay repair made on the intergranular stress corrosion crack (IGSCC) in the core spray pipe to pipe weld heat affected zone of weld NZ-3-68. The inspector reviewed the increase in ISI inspection scope, sample selection and test results of UT performed on the four additional welds (NZ-3-26, 3-28, 3-75 and 3-76) in the core spray system.

The inspector reviewed a sample of video recordings of the remote in-vessel visual inspection (IVVI) of core spray piping, various welds in the core shroud and the reactor pressure vessel corrosion resistant cladding. The review included the verification that previously identified indications were being examined and evaluated. The inspector reviewed a sample of visual examination reports and corrective action reports initiated as a result of the visual inspection performed during this outage (18R) of the containment liner (coating failure, corrosion and other damage) for compliance with the requirements of ASME Section XI, IWE (requirements for class MC and metallic liners of class CC components).

The inspector reviewed welding activities associated with the repair and replacement of selected components to verify the activities were performed in accordance with the requirements of ASME Section XI and IX. The inspector reviewed weld procedures GE WPS 8.8.65-(F)-OC Rev.1 and GE WPS 8.8.32-OC, and procedure and personnel qualification records for compliance with the requirements of ASME Section IX. The inspector interviewed the licensee's personnel responsible for interpretation of the radiographs evaluated. Radiographs of welding activities were reviewed for welds W211-201, 202 root and final and 202R1 (final repair) for the replacement of valve V-14-36 in the isolation condenser system to ensure proper identification, sizing and evaluation.

b. Issues and Findings

The inspector identified several instances in which licensee personnel did not adhere to procedures for the repair/replacement of safety related components. These instances involved welding activities associated with the core spray and isolation condenser systems. During these activities, the inspector identified several instances where weld craftsman withdrew and used weld filler metal that was not specified in the welding procedure specification. Regarding the isolation condenser system, welders used weld filler metal of classification 316L instead of the classification 308L/309L, as specified by the welding procedure on welds 201, 202, 203, 205, 205R1, 206, 207, 208, 214, 216, 240X and 241X. Although the use of the 316L filler metal was an allowed substitution by ASME Section IX and was stated by the materials engineer as approved, the welding procedure did not specify the use of 316L filler metal. Regarding the core spray system, a welder used weld filler metal of classification 308L instead of classification 309L for repair of the area adjacent to weld NZ-3-68. However, the welding procedure specified, by reference to the design and structural analysis of the engineered weld overlay, the use of 309L weld filler. The licensee subsequently contacted the originator of the structural analysis and obtained concurrence that the alternate filler metal was an acceptable substitution and the original analysis was unaffected. Further, the use of 308L filler metal was allowed by ASME Section IX.

These failures to properly control and implement procedures is a violation of Technical Specification Section 6.8.1, “Procedures and Programs,” and 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The licensee documented this issue in CAP 2000-1932. The issue was determined to be of very low safety significance, because the technical adequacy of the welding activities were subsequently determined to be acceptable and that boundary integrity was maintained. However, the issue is considered to be substantive with respect to the crosscutting issue of human performance. Therefore, in accordance with the Section VI.A.1 of the NRC Enforcement Policy and the NRC Significance Determination Process, this issue is considered to be a Non-Cited Violation (Green). **(NCV 05000219/2000-008-02)**

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the periodic evaluations required by 10 CFR 50.65 (a)(3) for Oyster Creek Generating Station to verify that structures, systems and components (SSC) within the scope of the maintenance rule were properly evaluated and dispositioned.

The inspectors selected the following safety significant system in (a)(1) status to verify that: (1) goals and performance criteria were appropriate, (2) industry operating experience was considered, (3) corrective action plans were effective, and (4) performance was being effectively monitored:

- Fuel Handling Equipment

In addition, the inspectors reviewed the following safety significant systems in (a)(2) status to verify that system performance compared to the licensee's performance criteria was acceptable.

- Control Rod Drive Mechanisms

b. Issues and Findings

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

.1 Repair of Reactor Water Cleanup Manual Isolation Valve (V-16-63)

a. Inspection Scope

The inspector reviewed this activity due to the risk significance associated with the isolation of the valve for repair work and the potential impact on reactor safety. Specifically, the repair activities associated with this valve had potential to create an unisolable leak path from the reactor vessel to the containment. Special contingency plans and procedures were developed to perform the activity. The inspector reviewed the installation plan for the strong back device used to retain the valve disk in its seat during repair and associated contingency work orders including the use of a freeze seal. In addition, the inspector observed the repair activity including the placement of the freeze seal contingency equipment and the installation of the stem restraining device.

b. Issues and Findings

There were no findings identified.

.2 Service Water Seal Well Diversion Procedure and Maintenance of Fuel Pool Temperatures

a. Inspection Scope

The inspector reviewed special procedure, 00-002, “Service Water Seal Well Diversion System,” and verified the prerequisites for the implementation of the procedure. Specifically, the inspector verified that operators were aware of contingency actions associated with the loss of service water and subsequently fuel pool cooling. The inspector also reviewed the outage risk analysis and work activity contingency plans to assure that concurrent work would not negatively impact the overall safety of the facility.

b. Issues and Findings

There were no findings identified.

.3 Emergent Control Rod Drive Stub Tube Roll Repair

a. Inspection Scope

The inspector reviewed the availability of water inventory during the control rod drive stub tube roll repair. In particular, a reactor building closed loop cooling water surveillance procedure required the core spray pumps to be placed in “pull to lock” for a period of time when the roll repair activities were ongoing. The inspector reviewed the potential increase in loss of inventory risk due to operator action instead of automatic action becoming necessary for core spray initiation.

b. Issues and Findings

There were no findings identified.

.4 Control Rod Drive Mechanism O-Ring Replacement

a. Inspection Scope

The inspector reviewed this activity due to the expanded scope of the repair work and the potential impact on reactor safety because the repair had the potential to create an unisolable leak path from the reactor vessel. In addition, the inspector reviewed the daily outage work plans to assure that the expanded scope was accounted for and other activities affecting inventory control were appropriately managed.

b. Issues and Findings

There were no findings identified.

1R14 Personnel Performance During Non-routine Plant Evolutions and Events

a. Inspection Scope

On November 15, 2000, a reactor water low level scram occurred during turbine heat up. The inspector reviewed the personnel performance that led to a missed critical step in the turbine warming procedure and subsequently resulted in a reactor pressure transient that caused a low level automatic scram and an excessive cooldown of the reactor coolant system.

b. Issues and Findings

A review of the operator actions prior to and after the reactor scram indicated weaknesses in integrated plant knowledge, inadequate mitigation actions and control room oversight.

The initiating cause of the transient was the failure to implement step 3.3.15 of procedure 315.1, “Main Turbine Operation.” This failure caused two steam bypass valves to open and resulted in a decrease in reactor pressure and an increase in reactor vessel water level.

Immediately after the bypass valves were opened, operators responded to the reactor water level increase by increasing the reactor water cleanup letdown flow. Once the level began to decrease, operators then focused on reducing letdown flow and attempted to increase feedwater flow. Because of the low power levels, the main feedwater regulating valve (MFRV) block valves were closed limiting the feedwater injection to the low flow valves in the system. The inspector noted that the operators primarily responded to controlling water level rather than addressing pressure control during the transient.

An apparent lack of plant knowledge in conjunction with poor command and control in the control room contributed to the operators failing to close the bypass valves as a means of pressure and level control during the transient. A review of the control room recorder strip charts indicated that the bypass valves remained open for approximately 17 minutes. In addition, recovery actions were not effective due to a lack of awareness of plant parameters and conditions. The sustained pressure decrease and loss of inventory via the open bypass valves caused the average reactor coolant cooldown rate to exceed the Technical Specification limit of 100 °F per hour.

The excessive cooldown rate was entered into the licensee’s corrective action program as CAP 2000-1921. The inspector reviewed the engineering evaluation of the cooldown rate, which was determined to be 111°F within a one hour period. This exceeded the technical specification limits as stated in TS 3.3.C.1, “The average rate of reactor coolant temperature change during normal heat up and cool down shall not exceed 100°F in any one hour period.” The failure to maintain the reactor coolant system average rate within the technical specifications limit of 100 °F per hour, is a violation of NRC requirements. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the Enforcement Policy. However, the technical specification bases considers 10 cool downs exceeding 300° F/hr to be acceptable during the lifetime of the facility, and the licensee has not exceeded this. In addition, because this was a depressurization event, the pressure within the reactor vessel followed the saturation curve and stayed within the pressure/temperature limitations of the reactor vessel. Lastly, the inspector concluded that this issue had a negligible effect on the fatigue usage factors for the reactor vessel components. This issue was determined to be of very low safety significance, which resulted in a Green finding. **(NCV 05000219/2000-008-03)**

1R15 Operability Evaluations

.1 Isolation Condenser Non-condensible Gas Collection

a. Inspection Scope

On October 15, 2000, while cutting an isolation condenser steam inlet pipe to perform a tube bundle replacement modification, a hydrogen burn occurred within the ‘A’ isolation condenser. The inspector reviewed the engineering evaluation of the event.

The engineering evaluation determined that hydrogen gases accumulated within the isolation condenser piping as a result of operating the plant with the steam line vent valves closed. Normally, these vent valves remain open to prevent non-condensible from being trapped within the condenser tubing. However, due to packing leaks these valves had been closed for several months. The inspector reviewed a previous engineering evaluation for isolation condenser operability with the vent valves closed. The evaluation determined that the build up of non-condensible gases within the condenser tubes would not impact the ability of the condenser to perform its function. The evaluation recommended that the gases be vented every 44 days to assure operability of the condenser. A preventive maintenance work order was established to direct operators to open these vents valves monthly in order to remove any non-condensible gases. The inspector noted that the isolation condenser was scheduled to be vented prior to the beginning of the refueling outage; however, this preventive maintenance was skipped based on the fact that the outage work would be performed within the monthly requirement. Although engineering personnel were aware of the impact the non-condensible gases would have on the heat exchanger operability during power operation, the potential for hydrogen gas pressure during maintenance was overlooked. There were no injuries or equipment damage as a result of the hydrogen burn. This issue was entered into the licensee’s corrective action program as CAP 2000-1454.

b. Issues and Findings

There were no findings identified.

.2 Containment Spray Torus Nozzle Air Blockage

a. <u>Inspection Scope</u>

On November 5, 2000, during the performance of procedure 607.4.01, “Containment Spray Nozzle Air Test,” the licensee identified that two of the 10 containment spray nozzles in the torus exhibited signs of blockage. The inspector reviewed safety evaluation SE-315403-018, which determined that a partial spray flow blockage to the torus air space does not cause a reduction of the containment spray function during accident mitigation.

b. <u>Issues and Findings</u>

There were no findings identified.

1R17 Permanent Plant Modifications

.1 Permanent Repair of Reactor Water Cleanup Valve (V-16-63)

a. <u>Inspection Scope</u>

The inspector reviewed the following engineering modification documents developed to permanently repair a manual reactor water cleanup valve pressure seal that had leaked during previous operating cycles:

- Modification Document No. OC-MD-H298-001,
- Safety Evaluation SE-000215-043,

In addition, the inspector reviewed the engineering change documents pertaining to the the installation of an additional bonnet retaining ring. The inspector also reviewed CAPs 2000-1784, 2000-1783 and 2000-1811.

b. <u>Issues and Findings</u>

There were no findings identified

.2 Isolation Condenser Condensate Return Valve Replacement

a. <u>Inspection Scope</u>

The replacement of isolation condenser valves V-14-36 and V-14-37 was performed in order to meet the licensee’s Generic Letter (GL) 89-10, ”Motor Operated Valves,” commitments. The inspector reviewed the following modification documents associated with the replacement activity:

- OC-MD-H369-001, “Installation of New V-14-36 and V-14-37 to comply with GL 89-10,”
- TR00,035, “Calculation: Design, Seismic and Weak Link analysis, 10 inch class 900 Stainless Steel Flex Wedge Gate Valve with SMB-2-40, Limitorque Motor Actuator,” and
- MPR 2112, “Evaluation of Stem Thrust Requirements for V-14-36 and V-14-37.

b. Issues and Findings

The were no findings identified.

.3 Containment Spray Heat Exchanger Anchorage Modification

a. Inspection Scope

The inspector reviewed modification document OC-MD-H690-001, which detailed the requirements of a containment spray heat exchanger anchorage modification. The containment spray heat exchanger anchors were modified to improve seismic event qualification due to corrosion of the existing anchors.

b. Issues and Findings

There we no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector reviewed and observed portions of the following post maintenance testing:

- Replacement of Isolation Condenser Valves, V-14-36 and V-14-37: Limitorque Monitoring and Stroke Testing (JO 538526)
- Nuclear Steam Supply System Leak Test , 602.4.001
- Core Spray Pump and Valve Operability (610.4.002, 610.4.003)
- Control Rod Drive Pump Operability

b. Issues and Findings

There were no findings identified.

1R20 Refueling and Outage Activities

.1 Refueling Outage Inspection

a. Inspection Scope

The inspectors reviewed and/or observed various risk significant activities associated with the refueling outage. These inspections included:

- Reviewing the overall outage schedule and risk assessment of the licensee’s planned work.
- Observing portions of the reactor shutdown and cool down.
- Reviewing the availability and technical adequacy of reactor water level and temperature indicating instrumentation.

- Reviewing the availability of protected equipment specified by AmerGen's shutdown risk assessment.
- Reviewing the adequacy of contingency plans as specified by the shutdown risk assessment.
- Verifying tagged out equipment was in the correct position as described by the associated tag.
- Touring spaces normally inaccessible during power operation.
- Observing portions of refueling activities including: reactor disassembly, core fuel movement and reactor vessel pressure testing. Regarding refueling activities, the inspectors reviewed the licensee's corrective actions associated with a fuel mispositioning error during the refuel process (CAP 2000-1679).
- Observing portions of the reactor startup.

b. Issues and Findings

There were no findings identified.

.2 Torus Foreign Material Exclusion

a. Inspection Scope

On October 24, 2000, the inspectors performed a walkdown of the drywell to review outage activities.

b. Issues and Findings

The inspectors noted that the drywell to torus downcomer foreign material exclusion (FME) covers were installed in a manner that was ineffective in preventing foreign material from entering the torus ring header. The inspectors noticed a hard hat, a plastic sign, and plastic tie-wraps in the ring header. The licensee performed a clean-up of the torus (JO 543883) and identified additional debris. Due to poor housekeeping practices, and ineffective FME control, workers received additional occupational exposure during the clean up of the debris. The concern for debris in the torus is due to the potential for fouling emergency core cooling system (ECCS) suction strainers if left uncorrected following the outage. NRC Information Notice 96-59 discussed the potential degradation of emergency safety systems due to debris in the suppression pool.

This issue was considered to have very low safety significance (Green) using the Significance Determination Process (SDP) phase 1 evaluation for mitigating systems because there was not an actual loss of safety function, debris was removed from the drywell and torus areas, and an inspection of these areas was performed prior to reactor start-up. There was no violation of NRC requirements because the licensee complied with the Technical Specifications limiting conditions for operations.

1R22 Surveillance Testing

.1 Primary Containment Integrated Leak Rate Testing

a. Inspection Scope

The inspector reviewed procedure 666.5.007, “Primary Containment Integrated Leak Rate Test.” The inspector reviewed the stabilization criteria and method for calculating the primary leakage the licensee used to accurately reflect the primary containment leakage rate. The inspector also reviewed corrective actions associated with identified leakage and the final acceptance of the test results.

b. Issues and Findings

There were no findings identified.

.2 Station Blackout Functional Test

a. Inspection Scope

The inspector reviewed surveillance procedure 678.4.005, “Station Blackout Functional Test,” and verified that the combustion turbines and associated circuitry used to energize 4160 volt vital bus “B” during a loss of all station alternating current power were capable of performing their function. In addition, the inspector reviewed Regulatory Guide 1.155, “Station Blackout,” and Final Safety Analysis Chapter 8.3.4, to verify that the refueling outage testing was performed in accordance with these documents.

b. Issues and Findings

There were no findings identified.

.3 Emergency Diesel Generator Automatic Actuation Test

a. Inspection Scope

The inspector reviewed surveillance procedure 636.2.001, “Diesel Generator Automatic Actuation Test,” to verify that the emergency diesel generators and associated circuitry used to energize 4160 volt vital busses during a loss of station electrical power were capable of performing their safety function as required by the Final Safety Analysis Report and Technical Specifications. The inspector observed the No. 1 emergency diesel generator automatic actuation test and also reviewed the final acceptance of the test results.

b. Issues and Findings

There were no findings identified.

.4 Containment Spray Nozzle Air Test, 607.4.010

a. Inspection Scope

The inspector reviewed the containment spray nozzle air surveillance tests and the test data to verify that the test performance met technical specification and procedure requirements. The inspector sampled the licensee's corrective action program for problems identified during past performance of this surveillance to determine the licensee's threshold for identifying and resolving problems.

b. Issues and Findings

There were no findings identified.

1R23 Temporary Plant Modifications

.1 Temporary Alternate Power to Vital Alternating Current (ac) Panel No. 1

a. Inspection Scope

The inspector reviewed the temporary modification evaluation that provided alternate power to vital ac power panel 1 during a circuit breaker replacement activity. The inspector reviewed the supporting technical evaluation, alternate feed diagrams, and safety evaluation. In addition, the inspector reviewed the outage risk assessment and contingency planning for this evolution.

b. Issues and Findings

There were no findings identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector reviewed the access control program by examining the controls established for exposure significant areas including postings, markings, control of access, dosimetry, surveys and alarm setpoints. Areas selected were located throughout the radiologically controlled area (RCA) and included: condenser bay, drywell, turbine deck, and refueling floor. Controls reviewed included: key control for locked high radiation areas, use of radiation work permits to control access to radiologically significant areas, and pre-job radiological briefings.

The inspector conducted job performance observations to evaluate radiation worker performance with respect to stated radiation protection work requirements. This also included verification of radiological controls, such as adequacy of surveys and radiation protection technician coverage. The inspector reviewed the radiation work permit utilized for these entries, attended the pre-job briefings presented to the workers by the radiation protection staff, observed controls present for access to these posted high radiation areas, and reviewed alarm setpoints. Specific jobs observed and reviewed during 18R were: reactor reassembly, inboard and outboard main steam isolation valve (MSIV) inspection and repair, seal and o-ring replacement under vessel, welding on the feedwater heaters, safety relief valve (SRV) inspection and replacement, repairs to valve V-16-63 in the drywell, and ISI work in the drywell. The inspector also reviewed licensee-identified issues related to access control to radiologically significant areas as documented in their corrective action process (CAP). CAP 02000-1334 (High Radiation Area in the Base of the Plant Stack) was reviewed during this inspection.

b. Issues and Findings

There were no findings identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed work performance during the current refueling outage (18R). Areas reviewed included an evaluation of the use of engineering controls to achieve dose reductions, review of the use of low dose waiting areas, review of on-job supervision provided to workers, and a review of individual exposures from selected work groups. An evaluation of engineering controls utilized to achieve dose reductions, and analysis of licensee source term reduction plans was also conducted.
The inspector conducted observations of radiation worker and radiation protection technician performance during high dose rate and/or high exposure jobs to determine if the training/skill level was sufficient with respect to the radiological hazards. The inspector identified the five highest cumulative dose jobs in the outage schedule, and reviewed the calculations, assumptions and work control plans being established for these areas. The jobs identified were: intra granular stress corrosion cracking (IGSCC)/in-service inspection (ISI), recirculation pump motor cooler replacement, emergency condenser isolation valve installation, main steam isolation valve inspection/repair, reactor disassembly, fuel shuffle, and reactor reassembly. The inspector also reviewed the installation of permanent lead shielding in the drywell that was a licensee plant modification made to improve the As Low as is Reasonably Achievable (ALARA) program.

An outage exposure goal of 400 person-rem was established as part of the station's 525 person-rem annual goal. This annual goal was a revised goal that was established late in 1999, and included exposures from three unplanned forced shutdowns that have taken place in 2000. The inspector also reviewed licensee-identified issues related to ALARA control as documented in their corrective action process. CAP 02000-1601 (Incorrect Part Installation During 18R Leads to Unnecessary Radiation Exposure) was reviewed during this inspection.

b. Issues and Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspector reviewed performance indicator (PI) data from the 4th quarter of 1999, through the 3rd quarter of 2000, for *Unplanned Scrams per 7000 Critical Hours* to verify its accuracy. The inspectors used Nuclear Energy Institute (NEI) 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline," as guidance and interviewed licensee personnel responsible for compiling the information.

b. Issues and Findings

The inspector noted that the number of hours of critical operation for the third quarter was incorrectly calculated by the licensee and included approximately 103 hours when the reactor was not critical during August, 2000. This error slightly increased the value of the PI, but it did not affect the color of the performance indicator. The licensee entered this issue into the corrective action program as CAP 2000-2099.

.2 Emergency Diesel Generator Unavailability

a. Inspection Scope

The inspector reviewed performance indicator (PI) data from the 4th quarter of 1999, through the 3rd quarter of 2000, for *Emergency Diesel Generator Unavailability* to verify its accuracy. The inspectors used Nuclear Energy Institute (NEI) 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline," as guidance and interviewed licensee personnel responsible for compiling the information.

b. Issues and Findings

The inspector noted an error in the data reporting associated with the number of required available hours for the emergency diesel reported in the 4th quarter of 1999. In this case, the licensee reported more hours required than possible and therefore the error was conservative and did not affect the color of the performance indicator. The licensee entered this issue into the corrective action program as CAP 2000-2045.

4OA3 Event Follow-up

Automatic Reactor Scram, November 15, 2000.

a. Inspection Scope

On November 15, 2000, during plant start up after refueling outage 18, an automatic reactor scram occurred. The inspector reviewed the post scram plant parameters, the equipment performance during the transient and the operator performance prior to and after the transient. In addition the inspector reviewed the results of the licensee's Post Transient Review Group (PTRG) report.

b. Issues and Findings

The causes of the automatic reactor scram on low reactor water level were the failure to implement step 3.3.15 of procedure 315.1, "Main Turbine Operation," and inadequate response to the reactor pressure transient that occurred after the procedural step was missed. Operators did not recognize that the bypass opening jack (BOJ) was controlling system pressure during the turbine warming stage of the plant start up. Because of this, they failed to close the BOJ (step 3.3.15) prior to closing the main turbine control valves. The plant responded to this pressure change by opening two bypass valves. This initiated a pressure decrease and subsequent reactor water level increase. Operators responded to this transient focusing on level control rather than pressure control. As a result, pressure was not stabilized in a timely manner and the reactor water level decreased to the low level scram setpoint (139 inches).

A primary contributor to this operator error was a poorly worded procedure that contained several critical action steps embedded within a single "IF/THEN" statement. In addition, shift management appeared to be engaged in the start up activities and did not remain objective prior to and during the transient. Specifically, the shift manger became involved in the turbine heat up evolution and provided peer checking for the reactor operator performing the procedure. Once the transient began, because of his involvement in the procedure, he did not retain a big picture of the reactor control and therefore did not readily identify the pressure control issues with the bypass valves.

The failure to follow procedure 315.1, "Main Turbine Operation," led to an automatic reactor scram and is a violation of Technical Specification Section 6.8.1, "Procedures and Programs," and 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings." This issue was considered to have very low safety significance (Green) using the Significance Determination Process (SDP) phase 1 evaluation for initiating events because all mitigating systems were available at the time. However, the issue is considered to be substantive with respect to the crosscutting issue of human performance. Therefore, in accordance with the NRC Enforcement Policy, NUREG 1600, and the NRC Significance Determination Process, these issues are considered to be a Non-Cited Violation (Green). **(NCV 05000219/2000-008-04)**

Additional issues associated with this reactor scram are discussed in sections 1R14 and 4OA4. This issue has been entered into the licensee's corrective action program as CAP 2000-1919.

4OA4 Cross-cutting Issues: Human Performance

Human performance errors were identified in the initiating event and barrier integrity cornerstone areas. Operations personnel exhibited a lack of system knowledge, poor self checking and inadequate shift oversight while performing a reactor start up (Section 1R14 and 4OA3). This led to an automatic reactor scram and subsequent excessive reactor vessel cooldown rate. Also, the inspectors noted poor procedural adherence and self checking issues while implementing the licensee’s welding and fire protection procedures (Sections 1R05 and 1R08). The safety significance of these individual events was very low.

4OA6 Meetings

Exit Meeting Summary

On December 8, the resident inspectors presented the inspection results to Mr. Kevin Mulligan 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.

PARTIAL LIST OF PERSONS CONTACTED

Licensee (in alphabetical order)
V. Aggarwal, Director, Engineering
R. Brown, Manager, Experience Assessment
R. DeGregorio, Vice President
B. DeMerchant, Licensing Engineer
J. Magee, Director, Maintenance
D. McMillan, Senior Manager, Systems
K. Mulligan, Plant Manager
D. Slear, Senior Manager, Design
R. Tilton, Manager, Assessment
W. Truax, Director, Work Management
C. Wilson, Senior Manager, Operations
K. Wolf, Manager, Radiological Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000219/2000-008-01	NCV	Contrary to the requirements of procedure 120.5, the licensee did not perform an evaluation to permit the amount of combustible materials stored on the turbine building mezzanine. (Section 1R05)
05000219/2000-008-02	NCV	Several examples of failing to follow station welding procedures. (Section 1R08)
05000219/2000-008-03	NCV	Failure to maintain the reactor coolant system cooldown rate within the technical specifications limit of 100 degrees per hour. (Section 1R14)
05000219/2000-008-04	NCV	Failure to follow procedure 315.1, “Main Turbine Operation,” which led to an automatic reactor scram is a violation of Technical Specification Section 6.8.1, “Procedures and Programs,” and 10 CFR 50 Appendix B, Criterion V, ”Instructions, Procedures, and Drawings.” (Section 4OA3)

LIST OF ACRONYMS USED

ac	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
AS	Automatic Suppression
ASME	American Society of Mechanical Engineers
BOJ	Bypass Opening Jack
CAP	Corrective Action Process
CFR	Code of Federal Regulations
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
FME	Foreign Material Exclusion
FMF	Fire Mitigation Frequency
FPC	Fuel Pool Cooling
GL	Generic Letter
IF	Ignition Frequency
IGSCC	InterGranular Stress Corrosion Crack
ISI	Inservice Inspection
IST	Inservice Test
IVVI	In-Vessel Visual Inspection
JO	Job Order
MD	Modification Document
MFRV	Main Feedwater Regulating Valve
MNCR	Material Noncomformance Report
MS	Manual Suppression
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Test
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSIC	Nuclear Safety Information Center
PI	Performance Indicator
psig	Pounds Per Square Inch Gauge
PT	Liquid Penetrant Testing
PTRG	Post Transient Review Group
RCA	Radiologically Controlled Area
RPV	Reactor Pressure Vessel
rem	Roentgen Equivalent Man
SDP	Significance Determination Process
SRV	Safety Relief Valve
SSC	Structures, Systems and Components
TS	Technical Specification
UT	Ultrasonic Test

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety
- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety
- Occupational
- Public

Safeguards
- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.