July 27, 2001

EA-01-197

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

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 50-219/01-06

Dear Mr. DeGregorio:

On June 30, 2001, the NRC completed an integrated inspection at your Oyster Creek reactor facility. The enclosed report documents the inspection findings which were discussed on July 20, 2001 with you and other members of your staff.

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

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). This finding was determined to be a violation of NRC requirements. However, because of the very low safety significance and because the issue has been entered into your corrective action program, the NRC is treating this issue as a Non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Oyster Creek facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index/html (the Public Electronic Reading Room).

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 No. 7
Division of Reactor Projects

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

Enclosure: Inspection Report 50-219/01-06
Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-219
License No.	DPR-16
Report No.	50-219/01-06
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	May 13, 2001 - June 30, 2001
Inspectors:	Laura A. Dudes, Senior Resident Inspector Thomas R. Hipschman, Resident Inspector Julian H. Williams, Senior Operations Engineer, June 4-8, 2001
Approved By:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000219-01-06, on 05/13 - 06/30/2001, AmerGen, Oyster Creek Generating Station. Emergent work.

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

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- GREEN. Operators failed to adequately assess the risk prior to closing both of the isolation condenser motor operated valves inside containment to line the system up for a maintenance activity. This condition would have rendered the isolation condensers unavailable under station black out conditions and resulted in an entry to an unacceptable risk level according to the licensee's procedure. This violation of 10CFR 50.65 (a)(4) is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy. This issue was entered into the corrective action program as CAP 2001-1024 **(NCV 05000219/2001-006-001).**

  The finding was of very low safety significance because the isolation condenser valves were closed for a short duration.

B. Licensee Identified Violations

- No violations were identified.

Report Details

Summary of Plant Status:

Oyster Creek began the inspection period at full power and remained there for the duration of the inspection period except for a twenty percent power reduction on June 25, 2001, associated with the loss of feedwater pump room ventilation.

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector reviewed the licensee's hot weather readiness plan to verify that safety related equipment would be functional under summer temperature conditions. The inspector reviewed the following maintenance job orders (JOs) to verify that the preventive or corrective maintenance would facilitate the operations of the critical warm weather equipment throughout the summer months:

- JO 00547677, "Feedpump Motor Cooling Exhaust Fan Motor,"
- JO 00545510, "Inspect, clean and replace anodes in turbine building closed cooling water heat exchanger,"
- JO 00548948, "Replace leaking diaphragm on air operator (Chlorination system)
- JO 00551006, "Service Water Pump 1-1 oiler repair

The inspector also reviewed operator logs and corrective action documents associated with the supply and exhaust fans for rooms containing safety related equipment to verify that recent ventilation failures were captured and resolved by the licensee's corrective action program.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Reactor Protection System M-G sets

a. Inspection Scope

The inspector performed a partial walkdown of accessible areas of the reactor protection motor-generator (M-G) sets. The inspector used procedure 339, "Vital Power System," to verify the MG sets and support systems were operating properly. The inspector also used portions of procedure 339 to perform a partial walkdown of the electrical line-up that provides vital alternating current (AC) power to the reactor protection system.

b. Findings

No findings of significance were identified.

.2 Reactor Vessel Level Instrumentation

a. Inspection Scope

The inspector performed a partial walkdown of accessible portions of the reactor vessel level instrumentation including cable pathways, differential pressure transmitters and control room indicators. The inspector used the Oyster Creek Updated Final Safety Analysis Report, chapter 7.6 and system design description SDD-II-OC-622D, Revision 2, for acceptance critera. In addition, the inspector used line diagram 148F712, “Reactor Vessel Level/Pressure/Temperature Instruments,” to verify level instrumentation penetration locations in the plant. The inspector reviewed the current design documents against the requirements for primary and alternate level instrumentation in the technical specifications (TSs) to verify that instrumentation necessary for normal and emergency operations was available and capable of performing its safety function.

b. Findings

No findings of significance were 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 to verify that the fire program was implemented in accordance with all conditions stated in the facility license. Plant walkdowns included observations of combustible material control, fire detection and suppression equipment availability, and compensatory measures. The inspectors conducted fire protection inspections for the following areas and equipment due to the potential to impact mitigating systems:

- Reactor Building Water Deluge Systems
- Fire Brigade Equipment
- 4160 Electrical Switchgear Room CO2
- control room fire suppression equipment
- 480 Electrical Switchgear Room Halon System
- “A/B” Battery Room

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator simulator training on June 6, 2001, to verify that the Oyster Creek operator requalification program adequately evaluated how well operators have mastered the training objectives, including training on high-risk operator actions. In addition the inspectors observed the training critique to assess the licensee's effectiveness in evaluating and correcting any observed deficiencies. A review was conducted of recent operating history documentation found in inspection reports, licensee event reports, the licensee's Corrective Action Process (CAPs), and the NRC plant issues matrix (PIM) from 1998 and 2001. The inspectors selected specific events from the CAPs which indicated possible training deficiencies to verify that they had been appropriately addressed.

The following inspection activities were performed using NUREG 1021, Rev. 8, "Operator Licensing Examination Standards for Power Reactors," Inspection Procedure Attachment 71111.11, "Licensed Operator Requalification Program, " Appendix A "Checklist for Evaluating Facility Testing Material" and Appendix B "Suggested Interview Topics" and NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)," as acceptance criteria.

- The year 2001 biennial written exams for the first three weeks and the operating test for the week of June 4, 2001, were reviewed for quality and performance.
- The results of the year 1999 biennial written exams and annual operating tests for years 1999 and 2000 were reviewed for performance and grading.
- Observations were made of the dynamic simulator exams and job performance measures (JPMs) being administered. These observations included facility evaluations of crew and individual performance on the dynamic simulator exam.
- The remediation plan for a crew failure in the simulator was reviewed. Observations of both remediation training in the classroom and the simulator were conducted.
- Operators were interviewed to determine their impressions of the training program.
- Observations of operator performance in the control room were made.

Simulator fidelity was reviewed against the guidance in ANSI/ANS 3.5-1993/1998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination."

A sample of records for requalification training attendance, program feedback, reporting, and medical examinations was reviewed for compliance with license conditions, including NRC regulations.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

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

- 120/208V Reactor Protection M-G Sets: (a)(1)
- Main Control Room Panels (reactor vessel instrumentation): (a)(2)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Control

.1 Emergent Relay Replacement for the Isolation Condenser Actuation Logic

a. Inspection Scope

The inspector reviewed the licensee's failure to perform an adequate risk assessment after a tagging order was revised to replace an isolation condenser actuation relay. The inspector reviewed JO 00551754, procedure 2000-ADM-3022.01, "Work Management and On-line Risk Management & Assessment," and procedure 2000-PLN-3022.01, "Oyster Creek On-Line Risk Management Plan," in order to determine the appropriate guidance for performing on-line maintenance risk assessments. This review was against criteria contained in the above procedures and verified against the requirements contained in 10 CFR 50.65 (a)(4).

b. Findings

The inspectors determined that operators did not perform an adequate risk assessment in accordance with 10 CFR 50.65 (a)(4) because the impact to plant safety was not appropriately evaluated prior to closing the isolation condenser motor operated condensate return valves located inside primary containment. The safety significance of this finding was very low (Green) because the time the valves were actually closed was small (40 minutes). However, failure to assess the increase in risk prior to removing the system from service constitutes a violation of 10 CFR 50.65 (a)(4).

On June 20, 2001, the licensee developed an emergent work package to replace a defective relay located in the isolation condenser automatic initiation logic. In preparation for this work, the licensee noted that the power would have to be removed from the condensate return valves for each isolation condenser. The licensee developed a work plan that removed power from the two direct current (DC) motor operated condensate return valves located outside primary containment and established a compensatory measure that included stationing operators at the valve circuit breakers in order to return them to service if necessary. The operators noted that the risk assessment yielded an increased core damage frequency (CDF) (7.31E-6) such that they met the criteria for a yellow risk management category (two times the base CDF of 3.346E-6).

Subsequent to implementing the above work package, the operators changed the configuration of the valve tagout, such that the two AC motor operated condensate return valves in the containment would be closed, while the DC condensate return valves were tagged open. At this time, a verbal discussion of plant risk between the shift manager and the engineer erroneously indicated that the risk assessment would remain the same or may improve to a Green risk category (less than 2 times the base CDF). The licensee proceeded with the new work plan and closed the AC powered valves, which are located inside primary containment. The valves were closed for approximately 40 minutes. The licensee completed the valve lineup and prepared to perform the relay replacement. Engineering performed a quantitative assessment of the new valve lineup after the configuration had been implemented and determined that this configuration had a substantially higher increase in CDF than the original tagout. The engineer determined that closing the AC powered valves resulted in a CDF of 1.2E-4, approximately thirty four times the base CDF, a risk management color of RED. The engineer notified the control room of this increase in CDF and the valves were restored to the open position.

This finding is more than minor because closing the isolation condenser valves that are powered by alternating current significantly increased the CDF under station blackout conditions. Specifically, due to the large contribution to CDF from the loss of offsite power and the importance of isolation condensers for mitigation under this condition the closing of these valves for 40 minutes represented a credible impact on safety. The original configuration had less impact on station risk because the valves were powered by DC power, which is available through station batteries and if necessary could be opened manually because they are located outside primary containment. The two AC powered valves inside primary containment would render both isolation condensers unavailable during a loss of station AC power and are not easily accessible to be opened manually. However, the safety significance of this finding was very low (Green) because the time the valves were actually closed was small (40 minutes). The inspectors determined that operators did not perform an adequate risk assessment in accordance with 10 CFR 50.65 (a)(4) because the impact to plant safety was not appropriately evaluated prior to closing the isolation condenser motor operated condensate return valves located inside primary containment.

10 CFR 50.65 (a)(4) requires that before performing maintenance activities, licensee's shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to this requirement, operators failed to adequately assess the risk prior to closing both of the isolation condenser motor operated valves inside containment. This condition would have rendered the isolation condensers unavailable under station blackout conditions and resulted in an entry to an unacceptable risk level according to the licensee's procedure. This violation of 10CFR 50.65 (a)(4) is being treated as a non-cited violation, consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). This issue was entered

into the corrective action program as CAP 2001-1024**(EA-01-197) (NCV 05000219/2001-006-001).**

.2 Continued Operation with a Degraded Reactor Water Cleanup Pump

a. Inspection Scope

On June 5, 2001, the "A" reactor water cleanup pump exhibited high vibrations due to a degraded pump outboard bearing. The licensee planned to removed the pump from service late that evening but chose to continue running the pump through the peak load time. The inspector reviewed the online maintenance risk assessment and verified that no other equipment out of service would challenge the operators if this pump were to fail prior to being removed from service. The inspectors verified that the operators were appropriately briefed on the situation and were cognizant of the procedures that would be necessary if the pump failed. In addition, the inspector verified that other potential plant transient risks were minimized during the time this pump was to be operated in a degraded condition.

b. Findings

No findings of significance were identified.

.3 Continued Operation with Loss of Normal and Alternate Power Supplies to 1E1

a. Inspection Scope

On June 26, 2001, a 900 amp disconnect on a 34.5 kilovolt (KV) line failed which resulted in a trip of the Z-52 line. This resulted in a loss on normal and alternate power supplies to 1E1, which provides power to support equipment. The licensee restored power to 1E1 and re-energized lost loads. Offsite power from Q-121 remained available, therefore maintaining TS requirements for offsite power. The inspector reviewed the weekly on-line maintenance schedule to verify that no changes in risk management were made due to the temporary loss of power.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

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

- Local Power Range Monitor input to Average Power Range Monitor Channel #4 failed upscale causing a half-scram on RPS 1. (CAP 2001-0992)
- Full core Power Shape Monitoring System showed higher Maximum Average Planar Linear Heat Generation Rate values than quarter core cases and exceeded 100% (CAP 2001-0874)

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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

- JO 00551754, "Isolation Condenser Automatic Initiation Relay Replacement"
- JO 00551595, " 'A' Reactor Water Cleanup Pump Bearing Replacement"
- JO 00867956, "CAPGRMS Equipment Calibration"
- JO 00809058, "Fuel Zone Temperature Calibration"

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspector observed preparations for plant shutdown and startup for maintenance outage 18U1, May 15 - May 17, 2001. The licensee implemented a short duration outage to perform a recirculation pump seal replacement. The inspector reviewed plant shutdown (procedure 203) and startup (procedure 201) procedures to verify that the plant was being operated in accordance with approved procedures and that all licensee requirements were being met.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Containment Spray/Emergency Service Water (ESW) Pump Operability and Inservice Test (IST)

a. Inspection Scope

On June 27, 2001, the inspector reviewed surveillance procedure 607.4.004, “Containment Spray/ESW Pump Operability and IST.” The inspector verified that the performance and resulting data associated with the surveillance test met the requirements of TSs. The inspector also reviewed CAP 2001-1059 to verify that test discrepancies were identified and resolved in accordance with appropriate procedures.

b. Findings

No findings of significance were identified.

.2 Core Spray System 2 Instrument Channel Calibration, Test and System Operability

a. Inspection Scope

On June 7, 2001, the inspector reviewed surveillance procedure 610.3.205, “Core Spray System 2 Instrument Channel Calibration, Test and System Operability.” The inspector verified that the performance and resulting data associated with the surveillance test met the requirements of technical specifications. The inspector also reviewed CAP 2001-973 to verify that test discrepancies were identified and resolved in accordance with appropriate procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

.1 Scrams With Loss of Normal Heat Removal

a. Inspection Scope

The inspectors reviewed performance indicator (PI) data from the 2nd quarter of 2000, through the 1st quarter of 2001, for *Scrams With Loss of Normal Heat Removal* to verify its accuracy. The inspectors used Nuclear Energy Institute (NEI) 99-02, Revision 0, “Regulatory Assessment Performance Indicator Guideline,” as guidance.

b. Findings

No findings of significance were identified.

.2 Emergency Diesel Generator Unavailability

a. Inspection Scope

The inspectors reviewed PI data from the 2nd quarter of 2000, through the 1st quarter of 2001, for *Emergency Diesel Generator Unavailability* to verify its accuracy. The inspectors used NEI 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline," as guidance.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 00-009: Missed Technical Specification Required Surveillance Tests Due to Personnel Error. The licensee failed to perform a technical specification surveillance on the intermediate and source range neutron monitoring systems during an August 2000 plant shutdown. Upon identification of the procedural error the licensee performed the surveillance and both instruments satisfied the technical specification acceptance criteria. This issue is considered minor as it had no credible impact on safety. The licensee has entered the issue into their corrective action program as CAP 2000-1059. The inspector performed an in office review of this LER and concluded that there were no issues which required additional review. This LER is closed.

.2 (Closed) Licensee Event Report (LER) 00-010 and LER 00-010, Rev. 1: Local Leak Rate Test Results in Excess of Technical Specification Limits due to Component Wear. A main steam isolation valve did not meet its individual leak rate criteria during local leak rate testing and therefore did not meet the acceptance criteria as designated in technical specification 4.5.2.D. This issue is considered minor due to a second primary containment valve in the same header that which would have prevented the containment from exceeding its overall leakage limits. The inspector performed an in office review of this LER. This LER is closed.

.3 (Closed) Licensee Event Report (LER) 00-011: Reactor scram due to low reactor water level resulting from personnel error. The inspector performed an in office review of this LER and concluded that no new issues were raised requiring an additional review. This event was dispositioned in NRC Inspection Report 05000219/2000-008. This LER is closed.

.4 (Closed) Licensee Event Report (LER) 00-006, Rev 1: Skin Dose Associated with Control Room System Heating Ventilation and Air Conditioning (HVAC) System B Exceed Limits After Re-evaluation. This Ler was reviewed previously in NRC Inspection Report 05000219/2000-007, no additional issues were identified during this review. The inspector performed an in office review of this revision. No violations of NRC requirements were identified. This LER is closed.

.5 (Closed) Licensee Event Report (LER) 98-11, Rev 2: Three Small Bore Pipe Lines did not Meet Design Bases for Seismic and/or Thermal Allowables. The inspector performed an in office review of this LER. Revision one of this LER was reviewed in NRC Inspection Report 05000219/1999-003, section E8. Revision two revised the dates for corrective actions to be completed due to site work prioritization issues. No additional information that would alter the previous disposition of this LER was identified. No violations of NRC requirements were identified. This LER is closed.

.6 (Closed) Licensee Event Report (LER) 00-002, Supplement 1: Unanalyzed Condition with Backup Pressure Regulator Inoperable between 25% and 90% Power. This LER was reviewed in NRC Inspection Report 05000219/2000-002, section E8. The supplement confirms thermal margins are adequate for operation above ninety percent power with one of two main steam pressure regulators out of service. Information provided in the supplement does not change the conclusion in the referenced inspection report. The inspector performed an in office review of this LER. No violations of NRC requirements were identified. This LER is closed.

4OA6 Meetings, including Exit

Exit Meeting Summary

On July 20, 2001, the resident inspectors presented the inspection results to Mr. Ron DeGregorio 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.

Annual Assessment Letter Public Meeting

On June 20, 2001, the NRC met with the AmerGen staff to present the conclusions associated with the NRC's Annual Assessment of Oyster Creek issued in a letter dated May 31, 2001. The meeting was open for public observation. NRC presentation slides were placed into ADAMS under ML011780304 and were made available for public access.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Licensee

V. Aggarwal, Director, Engineering
R. DeGregorio, Vice President
E. Harkness, Plant Manager
R. Hillman, Manager, Chemistry & Radwaste
J. Magee, Director, Maintenance
M. Massaro, Director, Work Management
D. McMillan, Director, Training
D. Slear, Senior Manager, Design
B. Stewart, Manager, Regulatory Affairs
J. Vaccaro, Operations Training Manager
C. Wilson, Senior Manager, Operations
G. Young, Supervisor, Operator Training

b. List of Items Opened, Closed, and Discussed

Opened and Closed

50-219/2001-006-001	NCV	Failure to perform an adequate risk assessment prior to re-configuring the isolation condenser valves for a maintenance activity. (Section R13.1)

Closed

50-219/1998-011-02	LER	Three Small Bore Pipe Lines Did Not Meet Design Bases for Seismic and/or Thermal Allowables. (Section4OA3.5)
50-219/2000-002-01 Sup 1	LER	Unanalyzed Condition with Backup Pressure Regulator Inoperable Between 25% and 90% Power. (Section 4OA3.6)
50-219/2000-006-01 Rev 1	LER	Skin Dose Associated with Control Room System HVAC System B Exceed Limits After Re-evaluation. (Section 4OA3.4)
50-219/2000-009-00	LER	Missed Technical Specification Required Surveillance Tests Due to Personnel Error. (Section4OA3.1)
50-219/2000-010-00	LER	Local Leak Rate Test Results in Excess of Technical Specification Limits Due to Component Wear. (Section 4OA3.2)

50-219/2000-010-01 Rev. 1	LER	Local Leak Rate Test Results in Excess of Technical Specification Limits Due to Component Wear. (Section 4OA3.2)
50-219/2000-011	LER	Reactor Scram Due to Low Reactor Water Level Resulting From Personnel Error. (Section 4OA3.3)

c. List of Acronyms

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ALARA	As Low As Is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
CAP	Corrective Action Process
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DC	Direct Current
ESW	Emergency Service Water
HVAC	Heating, Ventilation and Air Conditioning
IST	Inservice Test
JO	Job Order
JPMs	Job Performance Measures
KV	kilovolt
LER	Licensee Event Report
M-G	Motor Generated
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
PIM	Plant Issues Matrix
RPS	Reactor Protection System
SDD	System Design Description
SDP	Significance Determination Process
SSCs	Structures, Systems and Components
TS	Technical Specification