April 23, 2002

Mr. Jack Skolds
President and CNO
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
5th Floor
Warrenville, IL 60555

SUBJECT: OYSTER CREEK GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 50-219/02-02

Dear Mr. Skolds:

On March 31, 2002, 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 April 19, 2002, with Mr. Ernie Harkness and other members of your staff.

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

No findings of significance were identified.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). On February 25, 2002, the NRC issued an Order to all nuclear power plant licensees, requiring them to take certain additional interim compensatory measures to address the generalized high-level threat environment. With the issuance of the Order, we will evaluate Amergen's compliance with these interim requirements.

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/reading-rm/adams.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 No. 50-219
License No. DPR-16

Enclosure: Inspection Report 50-219/02-02
Attachment: Supplemental Information

cc w/encl: Amergen Energy Company - Correspondence Control Deck
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REGION I

Docket No. License No.	50-219 DPR-16
Report No.	50-219/02-02
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Dates:	February 10, 2002 - March 31, 2002
Inspectors:	Laura A. Dudes, Senior Resident Inspector Steve Dennis, Resident Inspector Frank Arner, Reactor Inspector, Feb. 11- 25, 2002, March 25 through April 3, 2002 Stephen Barr, Project Engineer, Feb. 19 - Feb. 22, 2002 Frank Jacobs, Inspector, Spent Fuel Program Office, NMSS Gregory Smith, Sr. Physical Security Inspector, February 20, 2002, March 25-29, 2002 John McFadden, Health Physicist, March 18-21, 2002 John Caruso, Senior Operations Engineer, March 18-22, 2002 George Morris, Reactor Inspector, March 18-22, 2002 Tracy Walker, Senior Reactor Inspector, March 18-22, 2002
Approved By:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000219-02-02, on 02/10-03/31/02, AmerGen, Oyster Creek Generating Station, resident inspector report.

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

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Violations

No findings of significance were identified.

Report Details

Summary of Plant Status

Oyster Creek began the inspection period at full power. A planned power reduction to 69 percent occurred on February 23, 2001, in order to perform a control rod pattern adjustment for reactivity flux shaping. Reactor power was returned to 100 percent after approximately 30 hours. Oyster Creek remained at full power for the remainder of the inspection period.

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

1R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The inspectors reviewed safety evaluations associated with mitigating systems, initiating events, and barrier integrity cornerstones to verify that changes to the facility or procedures as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59. Safety evaluations were selected based upon the safety significance of the changes and the risk to structures, systems, and components.

The inspectors also reviewed applicability reviews (10 CFR 50.59 safety screens) for changes, tests, and experiments for which the licensee determined that a safety evaluation was not required. This review was performed to verify that the licensee's threshold for performing safety evaluations was consistent with 10 CFR 50.59.

Finally, the inspectors reviewed a sample of corrective action process (CAP) items documenting problems identified by the licensee related to safety evaluations. This review was performed to verify that the licensee was identifying issues and entering them into the CAP.

A listing of the 10 CFR 50.59 safety evaluations, safety screens, and CAP items reviewed is provided in Attachment 1, b.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

Partial walkdown inspections were performed on the systems listed below. A random sampling of valve positions in the field were verified to be properly aligned in accordance with system operating procedures and Oyster Creek procedure no.108.9, "Equipment Alignment and Verification." Control room indications and controls were verified to be appropriate for the standby or operating status of the system and system maintenance action requests were reviewed to assure no degraded conditions existed to adversely affect operability.

- Standby Liquid Control, procedure no.304, "Standby Liquid Control Operation."
- Emergency Diesel Generator (EDG) #2, procedure no.341, "EDG Operation."

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 in the following areas due to the potential to impact mitigating systems:

- DG-FA-17, No. 2 Emergency Diesel Generator Room
- DG-FA-15, No. 1 Emergency Diesel Generator Room
- FS-FA-16, Emergency Diesel Generator Fuel Storage Area
- TB-FZ-11C, Switchgear Room, West End of Turbine Building on Mezzanine Level
- TB-FZ-11F, Feedwater Pumps

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed licensed operator requalification training (LORT) on March 6 and March 20. The training/testing exercise was reviewed against criteria listed in NRC Inspection Procedure 7111111. The inspector reviewed the critical tasks associated with the simulated control room exercise, observed the operators performance during the exercise and observed the post exercise critique. The inspector also reviewed procedure 2611-PGD-2612, "Oyster Creek Licensed Operator Requalification Training Program."

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 in accordance with (IAW) Oyster Creek (OC) procedure 2000-ADM-1220.01, "Implementation of the Maintenance Rule."

- 480 V System (CAP 2002-0331, 2002-104, 2002-157)
- Containment Spray/Emergency Service Water System #2 (CAP 2002-0393)
- #1 Fire Pump Diesel (WO C200224417)
- Primary Containment Vacuum Breakers (WO R201309101)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

.1 Isolation Condenser Automatic Actuation Relay Replacement

a. Inspection Scope

On February 16, 2002, the licensee performed an emergent work activity replacing three isolation condenser automatic initiation relays. In order to perform this work the automatic functions of both the A and B isolation condensers was defeated for a period of time. The inspector reviewed the risk assessment performed by the licensee IAW procedure no. ER-AA-600-1042, "On-Line Risk Management," and verified that the compensatory operator actions associated with both isolation condensers being out of service were established.

b. Findings

No findings of significance were identified.

.2 Containment Spray/Emergency Service Water System II Outage

a. Inspection Scope

On March 6, 2002, the licensee commenced a containment spray/emergency service water system II outage. The inspector reviewed the operations turnover document and verified that both the mechanical and electrical components necessary to support system I were designated as protected equipment IAW OC procedure no. WC-AA-104, "Review and Screening For Production Risk." Also, the inspector verified that special processes implemented due to hydrolyzing did not adversely impact operating equipment and verified physical barriers for protected equipment in system I.

b. Findings

No findings of significance were identified.

.3 Emergency Diesel Generator (EDG) #1 - 24 Month Outage

a. Inspection Scope

On March 18, 2002, a 24 month planned maintenance commenced on the #1 EDG. The inspectors verified the tagging clearance associated with the work (Clearance # 02500500), barriers in place for protected systems IAW OC procedure no. WC-AA-104, “Review and Screening For Production Risk,” online risk assessment during the four day outage IAW OC procedure no. ER-AA-600-1042, “On-Line Risk Management,” and technical specification adherence.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations, performed IAW OC procedure no. LS-AA-105, “Operability Determinations,” in order to determine that proper operability justifications were performed for the following items. In addition, where a component was determined to be inoperable, the inspectors verified the technical specifications (TS) limiting condition for operation implications were properly addressed.

- Following the performance of Fire Pump Operability Test 645.4.036 on March 19, 2002, a licensee review found that engine speed was higher than allowed per the procedure for Fire Diesel #2. The inspectors reviewed the operability determination prepared by OC engineering, the associated pump curve, previous operability tests, and engineering calculations used to support operability.

- Two Agastat E7000 series relays contained in the isolation condenser automatic initiation logic circuit experienced age related failures. Replacement of similar relays will be performed during the next refueling outage. An operability determination for the existing relays in the circuit was performed in order to support continued operation until the next refueling outage. The inspector reviewed the basis for operability including previous component replacement data and credible common cause failure scenarios.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the operator work-around database and associated corrective action items to identify conditions that could adversely effect the functionality of mitigating systems or impact human reliability in responding to initiating events. The inspector also reviewed open control room deficiencies and corrective action items to determine if there were any degraded or non-conforming conditions that should have been identified and evaluated as operator work-arounds IAW OC procedure no. OP-AA-108-105, "Equipment Deficiency Identification and Documentation." The inspector reviewed the operations database containing operator concerns and performed walkdowns of a sample of the individual concerns to verify that none of these concerns met the operational definition for a work-around or impacted the functionality of safety related equipment.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors selected and reviewed a sample of permanent modifications at Oyster Creek. The modifications were selected from the population of design changes completed since early 2000 based on risk insights from the probabilistic risk assessment and the potential for impacting reactor safety cornerstones. The modifications involved safety-related piping and components, electrical power systems, and instrumentation.

The inspectors reviewed selected portions of the modification packages, including the design calculations, setpoint changes, work packages, and results of post-modification testing. Where appropriate, the inspectors discussed the scope and extent of the modifications, technical aspects of the changes, and implementation of the changes with the responsible engineering personnel. The purpose of these reviews was to verify that the design bases and performance capabilities of the systems, structures, and components had not been degraded through modification.

The inspectors reviewed a sample of CAP items documenting problems identified by the licensee related to plant modifications. This review was performed to verify that the licensee was identifying issues and entering them into the CAP. In addition, the inspectors reviewed three new CAPs generated by the design engineering staff (i.e., 2002-458, 460, 469) in response to minor documentation issues identified by the NRC during the week of the inspection. The inspectors also reviewed self-assessments in the form of monthly Quality Review Team Supervisory Briefs.

A listing of the modifications and CAPs reviewed by the inspectors is provided in Attachment 1, b.

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 (PMT) 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.

- Control Rod Drive Pump "B" rotating element and bearing replacement (WO C0526979). Performed procedure 617.4.001, "CRD Pump "B" Operability Test" as the PMT.
- Isolation Condenser Automatic Actuation Relay Replacement (WO C2002363). Performed procedure 609.3.113 as the PMT.
- Hydraulic Control Unit 2-31, Valve 126 and 127 Operator replacements (WO C0537867). PMT included initial general hydrostatic bench test of operator prior to installation, in-service leak test of installed component and performance of procedure 617.4.003, "Control Rod scram Insertion Time Test and Valve IST."

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector observed pre-test briefings and portions of the surveillance test (ST) performance for procedural adherence, and verified that the resulting data associated with the test met the requirements of Technical Specifications (TSs). The inspector also reviewed the results of past performances of the ST to verify that degraded or non-conforming conditions were identified and corrected. The following STs were observed:

- Procedure 636.4.003, "Diesel Generator #1 Load Test"
- Procedure 609.3.113, "Isolation Condenser Automatic Actuation Bistable Calibration and Test"
- Procedure 619.3.001, "Turbine Load Rejection"

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed temporary modification 2001-034, Compensatory Measures for Degraded Fire Barriers on Two Columns in the "C" and "D" 4160V Switchgear Vaults. The inspector verified the temporary modification was performed in accordance with plant procedures and the engineering evaluation was performed as required in GL 86-10 (Implementation of Fire Protection Requirements). Additionally, the inspectors reviewed the associated 10 CFR 50.59 applicability screening, Updated Final Safety Analysis Report (UFSAR) section 9.5.1 (Fire Protection Program), and technical specifications to verify regulatory compliance.

b. Findings

No findings of significance were identified.

Emergency Preparedness (EP)

1EP6 Drill Evaluation

a. Inspection Scope

On March 14th and 19th, 2002, the inspectors observed the Emergency Preparedness drills in the Technical Support Center, Control Room Simulator, and the Emergency Operating Facility. The inspector reviewed the selected scenario, checklists, forms used for classification notification, and Protective Action Recommendation development. The inspector also reviewed the following procedures to verify the emergency response team actions were performed as required.

- EPIP-OC-.01, “Classification of Emergency Conditions”
- EPIP-OC-.03, “Emergency Notification”
- EPIP-OC-.25, “The Emergency Operations Facility”
- EPIP-OC-.26, “The Technical Support Center”
- EPIP-OC-.41, “Emergency Duty Roster Activation”

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

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

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

- RWP OC-1-02-008, Low pressure turbine rotor refurbishment on turbine building operating floor during cycle 18, Rev. 00
- RWP OC-1-02-013, Condenser bay work at 40% power (locate and plug sources of salt water leakage), Rev. 00
- RWP OC-1-02-056, I & C instrument maintenance and fire protection, Rev. 00
- RWP OC-1-02-058, Observation and inspection, Rev. 00
- RWP OC-1-02-059, Station services (work support) including decontamination, Rev. 00
- RWP OC-1-02-062, "B" reactor water clean-up pump cable pull in reactor building/turbine building, Rev. 01

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

- Procedure 6630-ADM-4110.04, Radiological work process, Rev. 9
- RP-AA-376, Radiological postings, labeling, and markings, Rev. 0
- RP-AA-460, Controls for high and very high radiation areas, Rev. 2

- RP-AA-1001, Radiation protection fundamentals, Rev. 0
- Weekly high radiation area boundary checklist
- Locked high radiation areas and high radiation areas as of March 20, 2002
- 2002 Exposure summary reports for the weeks of March 11 and March 18, 2002
- Selected portions of the 10 CFR 50.75(g) file, records of information important to the safe and effective decommissioning of the facility
- SA-2001-5058, Self-assessment plan and report, personnel contamination performance, completed on July 31, 2001
- SA-2001-5060, Self-assessment plan and report, radioactive material control, in progress
- SA-2001-5062, Self-assessment plan and report, 10 CFR 20 assessment, completed on December 26, 2001
- Focus area self-assessment report, radioactive materials control practices and policies, performed on December 19 and 20, 2001
- NOA-OC-01-4Q, Nuclear oversight continuous assessment report for October - December 2001

The inspection included a review of the following twelve corrective action process (CAP) items for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action: CAP Nos. O2001-1155, O2001-1626, O2001-1724, O2001-1915, O2002-0046, O2002-0112, O2002-0121, O2002-0155, O2002-0186, O2002-0280, O2002-0330, and O2002-0448.

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

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector toured the facilities and inspected procedural implementation, and reviewed records and other program documents to determine the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control.

Discussions with radiological protection personnel and review of ALARA documentation showed that the cumulative person-rem for the site for 2001 was approximately 45.5 which represented the second lowest total in the plant's history. The approved ALARA goals for 2002 were 40 and 250 person-rem, for the operational period and for the scheduled refueling outage (R19), respectively.

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

- Procedure 6630-ADM-4110.04, Radiological work process, Rev. 9
- RP-AA-401, Operational ALARA planning and controls, Rev. 2
- RP-AA-4002, Radiation protection refuel outage readiness, Rev. 0
- Station ALARA committee meeting minutes for September 19, 200, and January 2, 2002

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

- RER 2002-002F, Low pressure turbine rotor refurbishment on turbine building operating floor during cycle 18, Rev. 00
- RER 2002-009B, Condenser bay work at 40% power (locate and plug sources of salt water leakage), Rev. 00
- RER 2002-011B, "B" reactor water clean-up pump cable pull in reactor building/turbine building, Rev. 01

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

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector reviewed the program for health physics instrumentation to determine the accuracy and operability of the instrumentation.

During plant tours, the inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity and radiation levels, including portable field survey instruments, hand-held contamination frisking instruments, and continuous air monitors. The inspector conducted a review of the instruments observed in the toured areas, specifically verification of current calibration, of appropriate source checks, and of proper function.

The review was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, site Technical Specifications, and site procedures.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS
Physical Protection (PP)

3PP1 Access Authorization

a. Inspection Scope

The following activities were conducted to determine the effectiveness of the licensee's behavior observation portion of the personnel screening and fitness-for-duty programs as measured against the requirements of 10 CFR 26.22 and the Licensee's Fitness for Duty Program documents.

Two (2) Access Authorization/Fitness-for-Duty self-assessments, an audit, and event reports and loggable events for the four previous quarters were reviewed during March 26-28, 2002. Behavior observation training procedures and records were reviewed on March 26, 2002.

b. Findings

No findings of significance were identified.

3PP2 Access Control

a. Inspection Scope

The following activities were conducted during the inspection period to verify that the licensee has effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area as measured against 10 CFR 73.55(d) and the Physical Security Plan and Procedures.

Site access control activities were observed, including personnel and package processing through the search equipment during peak ingress periods on March 26 and 27, 2002. Two vehicle searches were observed on March 27, 2002. On March 27, 2002, testing of all access control equipment, including metal detectors, explosive material detectors, and X-ray examination equipment, was observed. The access control event log, an audit, and a self-assessment was also reviewed.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

.1 Emergency Diesel Generator Unavailability

a. Inspection Scope

The inspectors reviewed the Performance indicator (PI) data from January 2001 through December 2001 for *Emergency Diesel Generator Unavailability* to verify its accuracy. The inspectors also reviewed the licensees process for identifying and documenting the data described in OC procedure LS-AA-2040 Rev. 1, “Monthly PI Data Elements for Safety System Unavailability,” and compared the data against criteria contained in NEI 99-02 Rev. 1, to verify it was properly dispositioned in the PI reports.

b. Findings

No findings of significance were identified.

.2 Scrams with Loss of Heat Removal

a. Inspection Scope

The inspectors reviewed the Performance indicator (PI) data from January 2001 through December 2001 for *Scrams with Loss of Normal Heat Removal* to verify its accuracy. The inspectors also reviewed the licensees process for identifying and documenting the data described in OC procedure LS-AA-2020 Rev. 1, “Monthly PI Data Elements for

Scrams with a Loss of Normal Heat Removal," and compared the data against criteria contained in NEI 99-02 Rev. 1, to verify it was properly dispositioned in the PI reports.

b. Findings

No findings of significance were identified.

4OA5 Preoperational Testing of An Independent Spent Fuel Storage Installation

a. Inspection Scope

The inspectors evaluated whether the licensee was adequately prepared to use the independent spent fuel storage installation (ISFSI). Plans, engineering analyses, work packages, work practices and procedures were reviewed to ensure they met and were consistent with the terms and conditions of the Certificate of Compliance (CoC) for the ISFSI project. The inspectors observed samples of the preoperational testing of ISFSI operations. This included direct observation of critical activities documented in the CoC Attachment 1, Section 1.1.6, "Pre-Operational Testing and Training Exercises." These activities included, but were not limited to, loading a mock-up fuel assembly into the dry shielded canister (DSC), sealing a mock-up DSC, transfer cask handling, DSC insertion into the horizontal storage module (HSM), and DSC recovery operations.

Selected operational procedures relative to dry cask storage system (DCSS) loading, unloading, and transferring activities were reviewed. The procedures were reviewed to determine if they provided clear instructions to users, established limitations and action levels consistent with CoC requirements and directed workers on what to do if unsafe conditions arose. The acceptance criteria established in procedures were reviewed against requirements and commitments specified in the Safety Analysis Report (SAR), Safety Evaluation Report (SER), Certificate of Compliance (CoC) (Certificate No. 1004), Standardized Nuclear Horizontal Modular Storage (NUHOMS) System and 10 Code of Federal Regulations (CFR) Part 72. These selected operational procedures were verified to have been prepared, reviewed, and initially approved in accordance with the licensee's administrative programs.

The inspectors reviewed information relative to the licensee's methods for verifying and documenting the parameters and characteristics of spent fuel placed in the dry shielded canister. The review was against criteria contained in the CoC No. 1004, Amendment No. 4, the CoC Technical Specifications and NRC Interim Staff Guidance - 1 (Damaged Fuel). The licensee's 10 CFR 50.59 and 72.48 processes for changes, tests and experiments were reviewed to confirm that a documented and acceptable program was in place for performing design changes or evaluating nonconforming conditions that could affect ISFSI activities. This review included a sample of 10 CFR 72.48 screening/evaluations performed by both the licensee and the CoC holder, Transnuclear West, Incorporated. The inspectors reviewed selected portions of the licensee's 10 CFR 72.212 evaluation, which documented the reviews conducted in accordance with Title 10 Code of Federal Regulations, Part 72, Subpart K, Paragraph 72.212 for the utilization of the NUHOMS 61 BT Dry Spent Fuel Storage System. An evaluation of the ISFSI basemat and approach slab was also reviewed.
The inspectors selectively reviewed radiation protection planning and preparation, radiation work permits, pre-job health physics briefing packages, dose calculations, and the specific radiological hazards identified and the controls to be implemented for the

dry cask storage system loading, unloading, and transferring activities. The review was against criteria contained in the CoC No. 1004, Amendment No. 4, the CoC Technical Specifications, 10 CFR 20 and written evaluations required by 10 CFR 72.212.

The welding and cutting procedures were reviewed and welding operations performed on the dry shielded canister were observed. The inspectors reviewed the Emergency Plan to ensure that Emergency Action Levels had been developed to address the ISFSI activities. With regard to training, the inspectors selectively reviewed the ISFSI training program and materials. The review verified the CoC requirement that training should include an overview, fuel loading, transfer cask handling, canister transfer procedures and off-normal procedures.

During the preoperational testing, lifting of heavy loads was performed. The inspectors reviewed selected documentation for the crane being single-failure proof, including safety evaluations and calculations of maximum loading conditions, applicable controlling procedures, and equipment certifications. This review was performed to ensure the adequacy of rigging, control of heavy loads and crane operations.

The effectiveness of security controls during the preoperational testing was reviewed. The NRC approved Security Plan and implementing procedures were reviewed to identify the security requirements for the ISFSI and ensure they were being effectively implemented. Testing of the intrusion detection and assessment systems was observed and self audits of ISFSI security implementation were reviewed. Security Management and officials were also interviewed to assure their understanding of the ISFSI security requirements.

Additionally, corrective action program issues associated with the ISFSI project were reviewed to ensure that identified deficiencies were properly prioritized for resolution before receipt of fuel at the Independent Spent Fuel Storage Installation.

b. <u>Findings</u>

No findings of significance were identified.

4OA6 <u>Meetings, including Exit</u>

<u>Exit Meeting Summary</u>

On April 19, 2002, the resident inspectors presented the inspection results to Mr. Ernie Harkness and other members of licensee management. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Licensee (in alphabetical order)
A. Agarwal, Branch Manager Electrical/ I&C
V. Aggarwal, Director, Engineering
G. Busch, Licensing
R. DeGregorio, Vice President
C. Desai, Electrical Engineer
W. Emberger, Maintenance Supervisor
R. Ewart, Manager, Nuclear Security
M. Filippone, Electrical Engineer
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b. Documents Reviewed

Plant Modifications

ECR OC-01-00475	Fabrication, Reroute, and Installation of ESW Bypass, Rev. 0 (Mitigating)
ECR OC-01-00481	Final Connection of ESW Bypass, Rev. 0
ECR OC-01-00492	Revision of the FSAR, System SDBD, and the Oyster Creek Line List for the ESW Bypass, Rev. 0
ECR OC-01-00497	Backfill Under the Tranwa Trench for ESW Bypass, Rev. 0
ECR OC-01-00498	Backfill Instructions for Large Excavations at ESW Bypass Tie-In, Rev. 0
ECR OC-01-00530,	Revise ECRs 01-00481 and 01-00475 With Respect to Vehicle Protection and Freeze Protection Configuration of the New ESW Line Above the Intake Tunnel, Rev. 0
MD-B746-001	Restoration of LPRMs 28-09 and 36-09, Rev. 0 (Initiating Events)
MD-H617-001	Removal of Relief Valves V-20-24&25, Rev. 0 (Barrier Integrity)
ECR OC 01-00552	RPS 1A Alternate Trip Signal - MSIV closure signal causing half-SCRAMs (Initiating Events)
ECR OC 01-00851	Return RPS to Original Configuration for MSIVs - change not implemented yet (Initiating Events)
MD-F697-001	Rotary Inverter Control Logic & SQUG Relay Replacement (Initiating Events)
MD-H246-001	MG Set Charger Reverse Power Trip Delay
MD-G075-007	4160V “1C & 1D” protective relay replacement (Mitigating System)
MD-H369-001	Install new valves V-14-36&37- GL 89-10 (Mitigating System)
MD-G690-001	Generic Letter 96-06 Mods (Barrier Integrity)
MD-H346-001	Isolation Condenser A Tube Bundle Replacement (Mitigating)

Safety Evaluations

SE-000731-011	4160V ”1C & 1D” Protective Relay Replacement
SE-000735-022	MG Set Trip Delay
SE-000733-043	Rotary Inverter Logic Change and SQUG Relay Replacement
SE-OC-2001-E-0003	Replacement of Condenser Fan Motor
SE-400035-001	Isolation Condenser Tube Bundle Replacement
SE-000541-039	Generic Letter 96-06 Mods
SE-000211-036	Installation of new V-14-36 & 37 to comply with GL 89-10, Rev. 2
SE 000212-067	Removal of Relief Valves V-20-24 & V020-25, Rev. 0
SE 000621-018	Restoration of LPRMs 28-09 to APRMs 4 and 8, Rev. 0
SE OC-2001-E-0008	Replacement of Portions of Emergency Service Water System 2 Piping, Rev. 1

10CFR50.59 Safety Screens

OC-2001-S-0477	2000-ABN-3200-18, Rev. 14
OC-2001-S-0481	2000-ABN-3200-17, Rev. 8
OC-2001-S-0478	2000-ABN-3200-21, Rev. 5
OC-2002-S-0053	2000-ABN-3200-38, Rev. 3
OC-2001-S-0476	2000-ABN-3200-16, Rev. 9
OC-2001-S-0038	ECR 01-00828, Rev. 0
OC-2002-S-0053	2000-ABN-3200-078, Rev. 15

OC-2002-S-0019	2000-ABN-3200-45, Rev. 0
OC-2002-S-0021	2000-ABN-3200-47, Rev. 0
OC-2002-S-0023	2000-ABN-3200-49, Rev. 0
OC-2002-S-0025	2000-ABN-3200-51, Rev. 0
OC-2001-S-0473	311, Fuel Pool Cooling, Rev. 67
OC-2001-S-0512	ECR 01-00552
OC-2001-S-0681	Mod OC-01-00851
OC-2002-S-0071	Calc C-1302-862-E310-007
OC-2002-S-0147	ECR OC 01-00989
OC-2001-S-0003	Turbine Generator Trip Abnormal Operating Procedure Revision (2000-ABN-3200.10, Rev. 18), Rev. 0
OC-2001-S-0142	Core Spray Pump NZ01D (P-20-001D) Vibration Absorber, Rev. 0
OC-2001-S-0438	Abnormal Control Rod Motion (2000-ABN-3200.06 Rev. 11), Rev. 0
OC-2001-S-0598	Temporary Pipe Clamps on SW Cross Tie to ESW1, Rev. 1
OC-2001-S-0626	Increase in Main Steam Line/Off-Gas Activity (2000-ABN-3200.26, Rev. 13), Rev. 0
OC-2001-S-0627	Electric Pressure Regulator Malfunction (2000-ABN-3200.09 Rev. 9), Rev. 0

Corrective Action Program (CAP) Reports

01999-1331, 02000-0679, 02001-0266, 02001-0449, 02001-0631, 02001-0881, 02001-0882, 02001-0476, 02001-0882, 02001-1233, 02002-0440, 02002-0438, 02002-0445, 02002-0460, 02002-0458, 02002-0469.

Procedures

Exelon 50.59	Resource Manual, LS-AA-104-1000, Rev. 2
Exelon, S-AA-104	Exelon 50.59 Review Process, Rev. 2
EP-045	(5000-ADM-7313.03), Engineering UFSAR Update Reviews and Control of System Design Basis Documents, Rev. 4
Standing Order 21	Allowable Bypass Configuration for APRM/LPRM System, Rev. 24
CC-AA-202-1001	Quality Review Team (QRT), June 12, 2001
CC-MA-102,	Design Inputs and Impact Screening, Rev. 0
CC-MA-102-1001	Design Inputs and Impact Screening - Implementation, February 19, 2001
CC-MA-103	Configuration Changes, Rev. 0
CC-MA-103-1001	Implementation of Configuration Changes, February 19, 2001
2400-SMM-3900.08	General Hydrostatic Test, IS Leak Test, Rev. 7
2400-SMM-3900.04	System Pressure Test (ASME XI), Rev. 7
MA-AA-716-012	Post Maintenance Testing, Rev. 0
2000-ADM-7320	Updated FSAR Revisions and Update Control, Rev. 0
5000-ADM-7313.03	Engineering UFSAR Update Reviews and Control of DBD Documents, Rev. 4
EMP-002	(5000-ADM-7350.06), Modifications, Rev. 7 (Historical?)
EP-016	(5000-ADM-1291.01), Nuclear Safety/Environmental Determination and Evaluation, Rev. 9 (Historical)
2400-SME-3411.06	MSIV Limit Switch Adjustment, Rev. 3
2400-SMI-3602.01	SAR Alarm Point Revision/ Verification, Rev. 18
602.4.004	MSIV 10% Closure Test, Rev. 4
635.2.001	4160 Switchgear Bus Protective Relay Surveillance, Rev. 34

202.1	Power Operation, Rev. 59 and 61, Core Daily Checks completed November 17, 2000 and November 18, 2000
308	Emergency Core Cooling System Operation, Rev. 68
403.1	Placing the LPRM-APRM System in Operation, Rev. 11
403.2	Operation of the LPRM-APRM System During Startup and Power Operation, Rev. 6
403.3	Bypass of the LPRM-APRM System, Rev. 12
610.3.105	Core Spray System 1 Instrument Channel Calibration Test and System Operability, Rev. 47
610.3.215	Core Spray System 2 Instrument Channel and Level Bistable Calibration and Test and System Operability, Rev. 27
610.4.002	Core Spray Pump Operability Test, Rev. 41
607.4.008	Containment Spray and Emergency Service Water System 2 Pump Operability Test, Rev. 11, completed August 9, 2001
620.3.001	LPRM Test and Calibration (Front Panel Test), Rev. 29 and Revs. 25 and 26 performed October 16, 2000 and November 13, 2000
620.3.003	APRM Surveillance Test and Calibration, Rev. 46 and Rev. 45 completed November 30, 2000
620.3.009	LPRM Detector Calibration, Rev. 3
664.3.008	Plant Computer Burr-Brown Isolators Calibration, Rev. 4
1001.22	Core Monitoring and Operation, Rev. 31
1001.39	LPRM Calibration Using PSMS, Rev. 15
A100-GMM-3900.52	Inspection and Assembly of Flanged Connection, Rev. 2

<u>Drawings</u>

BR 2006	Reactor Building Closed Cooling Water System, Sheet 3, Rev. 56
GE 148F262	Emergency Condenser Flow Diagram, Sheet 1, Rev. 50
GE 148F711	Reactor Shutdown Cooling System Flow Diagram, Sheet 1, Rev. 40
BRM001	Post Accident Sampling Flow Diagram, Rev. 40
3C-733-11-005	CIP-# Panel Schedule, Rev. 3
BR 3022, Sh 6	EBOP Elementary Diagram, Rev. 2
A2012198	RPS Channel 1A MSIV Input Modification, Rev. 0
F697-002	Rotary Inverter Control Logic Change, Rev. 0
G075-2A	Switchgear 1C & 1D Relay Replacement, Rev. 0
H246-001 & 002	Main Battery Charger “A” MG Time Delay Modification, Rev. 0
GE 237E566, Sh 2	RPS Electrical Elementary Diagram, Rev. 39
GE 237E566, Sh 12	RPS Electrical Elementary Diagram, Rev. 4
GE 237E566, Sh 14	RPS Electrical Elementary Diagram, Rev. 5
DWG-G690-001	Reactor Building Flow Diagram, GL 96-06 Modifications, Rev. 1
DWG-G690-002	Reactor Building Piping Isometrics, GL 96-06 Mods, Rev. 2
DWG-G690-003	Reactor Building Tubing Isometrics, GL 96-06 Mods, Rev. 2
DWG-H369-221	Replace Valves V-14-36&37, Rev. 0
DWG-B746-001	Neutron Monitoring Sys. Electrical Connection Diagram, Restoration of LPRMs 28-09 A & C and 36-09 A & C, Rev. 0
DWG-B746-002	Neutron Monitoring Sys. Electrical Connection Diagram, Restoration of LPRMs 28-09 B & D and 36-09 B & D, Rev. 0
DWG GE 706E812	Neutron Monitoring System Electrical Elementary Diagram Sheet 1, “Detector Tabulation,” Rev. 20 Sheet 18, “APRM Channel 4,” Rev. 5

	Sheet 25, "APRM Channel 8," Rev. 6
	Sheet 51, "APRM Power Range Monitor Indicators Div. 2," Rev. 3
DWG GE 885D781	Core Spray System Flow Diagram, Rev. 71
DWG-H617-221	Core Spray Relief System, Removal of Relief Valves V-20-24 & V-20-25, Reactor Bldg. Elev. 23'-6" & Elev. 51'-3", Rev. 0

Calculations

C-1302-733-E320-017	PP P3-3 Voltage Drop, Rev. 0
C-1302-735-E320-044	OC 125 VDC Voltage Drop, Rev. 1
C-1302-862-E310-007	DG Fuel Transfer Pump Inlet Pressure, Rev. 0
C-1302-212-E540-107	Oyster Creek NSR Pipe Support Evaluations - North Side Core Spray Systems from Pumps NZ01A and NZ01C to Drywell Penetration X12-B, Rev. 1

Setpoint Change Request

M-G Set A	Control Logic Time Delay

Work Orders/Job Orders

00539815	SAR Point 531 Change
00539816	SAR Point 534 Change
C0537859	Pull, Test and Terminate 4160 Volt Cable 14-24 for P-16-001B
00537877	Replace Both Bundles in the "A" Isol. Condenser
00537861	Post Accident Sampling
00540481	V-14-0037
00538526	V-14-0036
0539542	Restore Control Room Wiring for LPRMs 28-09 and 36-09 IAW PIMS B746
00541018	Remove Relief Valve V-20-24 During 18R IAW PIMS #H617
00541904	Remove Relief Valve V-20-25 During 18R IAW PIMS #H617
00543644	Perform LPRM Megger and TDR Traces to Verify Operability After Outage Work is Complete
C2001184	Investigate, Evaluate and Repair Emergency Service Water Leak

Action Requests

A0780065	Replace Cable 14-24 for the "B" RWCU Pump
A201146	Investigate and Repair Abandoned ESW II Chlorine Line

Vendor Information

GEI-30990B	Direct Current Directional Relays Instruction Booklet
Agastat 7000	Catalog Data
VM-OC-0299	Automatic Transfer Switch Service Manual

Quality Review Team Meeting Supervisory Briefs

Monthly Supervisory Briefs for months January 2001-February 2002

Other Documents

Control Room Log	October 30, 2000
ECD C313559	Update GMS2 and Drawings to Reflect Changes Done During OC-MD-H353-001, November 18, 1999
ECD C313606	Removal of Relief Valves V-20-24 & V-20-25, dated September 30, 2000
ECD C401119	LPRMs 28-09A & C and 36-09 A & C Restoration, January 15, 2001
NDE Data Report	2001-017-039, “ESW Piping Replacement Project, dated August 9, 2001
NDE Data Report	2001-017-047, “ESW Piping Replacement Project,” dated August 8, 2001
REP-0457-001206	Restoration of LPRMs 28-09 and 36-09, December 2000
SDBD-OC-212A	Design Basis Doc. for Low Press Core Spray, Rev. 3 (Proprietary)
SDBD-OC-532	System Design Basis Document for Emergency Service Water System, Rev. 3
Topical Report 133	Reload Information and Safety Analysis Report for Oyster Creek Cycle 18 Reload, Rev. 1

List of Documents Reviewed (ISFSI Project)

Calculations and Engineering Analyses

2820-01-006	Documentation of doses onsite and to members of the public as a result of ISFSI, Rev. 1
C-1302-226-5411-337	Isotopic inventory for a dry storage cask, Rev. 0
11199-01	Oyster Creek offsite dose calculation, Rev. 2
11199-02	Oyster Creek dose evaluation, Rev. 0
NUH61B.0501	NUHOMS-61BT surface dose rates and occupational exposures, Rev. 0
1098-1	NUHOMS 61B on-hook weight calculation for Oyster Creek, Rev. 2
101-018-02-BL	Max load on hook- OS 197 & DSC in fuel pool, Rev. 1
1093-72	NUHOMS-61B transport package maximum weight calculation, Rev. 0
1093-64	NUHOMS-61B TS canister lifting device structural analysis, Rev. 0
File No. 2118.0200	Oyster Creek NUHOMS ISFSI Basemat and Approach Slab Analysis and Design
TNY01-0102	Oyster Creek ISFSI Foundation Evaluation, Rev. 1
TNY01-0200	Oyster Creek NUHOMS ISFSI Basemat and Approach Slab Evaluation for Consideration of 61BT DSC
C-1302-153-E310-113	Tipping Analyses of NUHOMS-61B Canister at El. 119'
205.10	Fuel Assembly removal/insertion in fuel pool/dry shielded canister, Rev. 14
205.13	Dry fuel storage monitoring, Rev. 3
312.3	Operation of Reactor Building Railroad Air Lock Doors, R5

330	Standby Gas Treatment System, Rev. 39
2000-ADM-4532.04	Oyster Creek Offsite Dose Calculation Manual, Rev. 13, 4.4 Total dose-compliance with 40 CFR 190.10a
6630-ADM-4110.04	Radiological work process, Rev. 8
1000-ADM-3890.01	Lifting and Rigging, Rev. 0
2400-SMM-3891.04	Operation of the reactor building overhead crane, Rev. 7
LS-OC-104	Oyster Creek 72.48 Review Process, Rev. 0
EPIP-OC-01	Matrix of Emergency Action Levels for Emergency Classification, Appendix 1, Rev. 12
NF-OC-621	Independent Spent Fuel Storage Process Control Program, Rev. 0
NF-OC-624	Independent Spent Fuel Storage Activities During a Plant Emergency, Rev. 0
NF-OC-626	Fuel Loading/Unloading of a DSC, Rev. 0
NF-OC-628	Pre-Operational Testing and Training Exercise for The ISFSI Campaign, Rev. 2
NF-OC-629	Transport and Preparation of Transfer Cask and 61BT Dry Shielded Canister for Loading Fuel, Rev. 1
NF-OC-630	Transport and Loading of Transport Cask and DSC, R1
NF-OC-631	Transport of Loaded Transfer Cask and 61BT Dry Shielded Canister to Transfer Trailer, Rev. 0
NF-OC-632	Dry Shielded Canister (61BT) Vacuum Drying and Helium Fill, Rev. 0
NF-OC-633	Loaded Dry Shielded Canister Welding, Rev. 0
NF-OC-634	Transportation, Alignment and Insertion of the 61BT Dry Shielded Canister into the Horizontal Storage Module, R0
NF-OC-637	Dry Shielded Canister Recovery, Rev. 1
NF-OC-638	Fuel Bundle Selection Process for Loading NUHOMS 61 BT DSC, Rev. 1
OSEC-IMP-150.08	Security Responsibilities for the ISFSI, Revision 3, dated January 25, 2002

Corrective Action Program

CAP 02002-0273	CAP 02002-0226	CAP 02002-0237
CAP 02002-0274	CAP 02002-0178	CAP 02002-0137
CAP 02002-0224	CAP 02002-0198	CAP 02002-0113

Miscellaneous

- The NRC approved Physical Security Plan Revision 4c, dated August 29, 2000
- Final Safety Analysis Report (FSAR) for Standardized NUHOMS System, Model
- No. NUHOMS-61BT for Boiling Water Reactor Fuel, Certificate 1004, Amendment No. 4
- NRC Safety Evaluation Report for FSAR for Certificate 1004
- Oyster Creek ISFSI Project Canister Loading Protocol
- U.S. Department of Energy Nuclear Fuel Data Survey Form RW-859 for Oyster Creek for report period 1996 to 1998 covering cycles 01A,01B, and 02 through 16
- EPRI Cask Loader Training Manual for Exelon Nuclear by Northeast Technology Corporation, September 6, 2001

- Letter from Northeast Technology Corporation to EPRI, dated January 2, 2002; Subject: Caskloader v1.1 Verification and Validation: Task 3 of EPRI Research and Development Agreement No. EP-P5625/C2807
- General Electric Services Information Letter No. 636, Rev. 1, Additional terms included in reactor decay heat calculations, June 6, 2001
- Oyster Creek Generating Station Annual Radioactive Effluent Release Report-2000, Table 1, Annual Offsite Doses Due to Radionuclides in Effluents, January 1, 2000 through December 31, 2000
- Oyster Creek Generating Station Radiological Environmental Monitoring Report for 1999, Direct Radiation Monitoring
- Radiation Work Permit No. OC-1-02-00023, Rev. 01, ISFSI Project (All Areas)
- Radiological Engineering Review No. 2001-16F for 2002 Dry Fuel Storage - 4 casks
- Oyster Creek Nuclear Generating Station Amendment No. 223 to Facility Operating License No. DPR-16 (Changes to the Technical Specifications regarding handling of heavy loads over irradiated fuel stored in the spent fuel pool), January 23, 2002
- NRC Safety Evaluation Report related to Amendment No. 223 to Facility Operating License No. DPR-16
- Safety Evaluation No. 000882-004, Single failure proof modification of the reactor building crane/OC load lift management procedure/operation of the reactor building overhead crane/PFU, Rev. 1, July 25, 2000
- Safety Evaluation No. 400007-001, Technical Specification Change Request No. 281, Removal of heavy load restriction over the spent fuel storage pool, Rev. 0, March 20, 2001
- Evaluation of OS197-1 transfer skid for 105 ton payload, Transnuclear West Memorandum TNY01-02-01, January 8, 2002
- Training Course NUHOMS Dry Fuel Storage, “Certificate of Compliance”
- 50.59 Reviews for NF-OC-634,631,624,628,629,632
- Licensed Operator Requal. Simulator Training 885, #2611-PGD-2612, Cask Drop Walk Through
- Safety Review Screening 72-1662, Rev. 1, Down ending of the loaded cask
- Safety Review Screening 02-008, Spacers

c. List of Acronyms

ADAMS	Agency wide Documents Access and Management System
ALARA	As Low As Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
APRM	Average Power Range Monitor
BNE	Bureau of Nuclear Engineering
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CRD	Control Rod Drive
DBT	Design Basis Threat
DCSS	Dry Cask Storage System
DSC	Dry Shielded Canister
ECD	Engineering Change Document
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
EP	Emergency Preparedness
GL	Generic Letters
HRA	High Radiation Area
HSM	Horizontal Storage Module
I&C	Instrumentation and Control
IAW	In Accordance With
ISFSI	Independent Spent Fuel Storage Installation
JO	Job Order
LHRA	Locked High Radiation Area
LORT	Licensed Operator Requalification Training
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NJDEP	New Jersey Department of Environmental Protection
NMD	Nuclear Maintenance Department
NRC	Nuclear Regulatory Commission
NUHOMS	Nuclear Horizontal Modular Storage
OC	Oyster Creek
OS	Occupational Radiation Safety
PI	Performance Indicator
PMT	Post Maintenance Testing
QRT	Quality Review Team
RCA	Radiologically Controlled Area
RER	Radiological Engineering Review
RPS	Reactor Protective System
RWP	Radiation Work Permit
SAR	Safety Analysis Report
SDP	Significance Determination Process
SER	Safety Evaluation Report
SSCs	Structures, Systems and Components
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report