UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD.
KING OF PRUSSIA, PA 19406-2713

January 25, 2017

EA-16-241

Mr. Bryan Hanson
Senior Vice President, Exelon Generation Co., LLC
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Rd.
Warrenville, IL 60555

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION – INTEGRATED INSPECTION REPORT 05000219/2016004 WITH PRELIMINARY WHITE FINDING

Dear Mr. Hanson:

On December 31, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Oyster Creek Nuclear Generating Station. On January 4, 2017, the NRC inspectors discussed the results of this inspection with Mr. Garey Stathes, Site Vice President, and other members of your staff. The results of this inspection are documented in the enclosed report.

The enclosed inspection report discusses a finding that the NRC has preliminary determined to be White, a finding of low to moderate safety significance. As described in Section 4OA2 of the enclosed report, the finding is associated with an apparent violation of Technical Specification 6.8.1, “Procedures and Programs,” because Exelon failed to follow the electromatic relief valve (EMRV) reassembly instructions, which caused the ‘E’ EMRV to be incorrectly reassembled. This caused excessive friction between the solenoid frame and cut-out switch lever, which impacted the ability of the ‘E’ EMRV to perform its safety function. As a consequence, Exelon also violated Technical Specification 3.4.B, since the ‘E’ EMRV was determined to be inoperable for greater than the technical specification allowed outage time. The finding was assessed based on the best available information, using Inspection Manual Chapter (IMC) 0609.04, “Initial Characterization of Findings,” and Exhibit 2 of IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012. The basis for the NRC’s preliminary significance determination is described in the enclosed report.

As an apparent violation of NRC requirements, this finding is being considered for escalated enforcement action in accordance with the Enforcement Policy, which appears on the NRC’s Website at https://www.nrc.gov/about-nrc/regulatory/enforcement/enforce-pol.html. The NRC will inform you, in writing, when the final significance has been determined. We intend to complete and issue our final safety significance determination within 90 days from the date of this letter. The NRC’s SDP is designed to encourage an open dialogue between your staff and the NRC, however, the dialogue should not affect the timeliness of our final determination.

We believe that we have sufficient information to make the final significance determination. However, before we make a final decision, we are providing you an opportunity to provide your perspective on this matter, including the significance, causes, and corrective actions, as well as any other information that you believe the NRC should take into consideration. Accordingly, you may notify us of your decision within 10 days to: (1) request a regulatory conference to meet with the NRC and provide your views in person, (2) submit your position on the finding in writing, or (3) accept the finding as characterized in the enclosed inspection report.

If you choose to request a regulatory conference, the meeting should be held in the NRC Region I office within 40 days of the date of this letter, and will be open for public observation. The NRC will issue a public meeting notice and a press release to announce the date and time of the conference. We encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. If you choose to provide a written response, it should be sent to the NRC within 40 days of the date of this letter. You should clearly mark the response as "Response to Preliminary White Finding in Inspection Report No. 05000219/2016004; EA-16-241," and send it to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Senior Resident Inspector at Oyster Creek Nuclear Generating Station.

You may also elect to accept the finding as characterized in this letter and the inspection report, in which case the NRC will proceed with its regulatory decision. However, if you choose not to request a regulatory conference or to submit a written response, you will not be allowed to appeal the NRC's final significance determination.

Please contact Silas Kennedy at (610) 337-5046 within 10 days from the issue date of this letter to notify the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision.

Because the NRC has not made a final determination in this matter, no notice of violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation may change based on further NRC review. The final resolution of this matter will be conveyed in separate correspondence.

In addition, if you disagree with a cross-cutting aspect assignment, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I, and the NRC Senior Resident Inspector at Oyster Creek Nuclear Generating Station.

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

Sincerely,

/RA/

Michael L. Scott, Director
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000219/2016004
w/Attachment: Supplementary Information

cc w/encl: Distribution via ListServ

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION – INTEGRATED INSPECTION REPORT 05000219/2016004 WITH PRELIMINARY WHITE FINDING DATED JANUARY 25, 2017.

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-219
License No.	DPR-16
Report No.	05000219/2016004
Licensee:	Exelon Nuclear
Facility:	Oyster Creek
Location:	Forked River, New Jersey
Dates:	October 1, 2016 – December 31, 2016
Inspectors:	A. Patel, Senior Resident Inspector E. Andrews, Resident Inspector W. Cook, Senior Reactor Analyst J. DeBoer, Emergency Preparedness Inspector J. Lilliendahl, Senior Emergency Response Coordinator J. Richmond, Senior Reactor Inspector
Approved By:	Silas R. Kennedy, Chief Reactor Projects Branch 6 Division of Reactor Projects

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SUMMARY

IR 05000219/2016004; 10/01/2016 – 12/31/2016; Exelon Energy Company, LLC, Oyster Creek Generating Station; Problem Identification and Resolution Annual Sample.

This report covered a three-month period of inspection by resident inspectors and announced baseline inspections performed by regional inspectors. The inspectors identified one apparent violation of preliminary low to moderate safety significance (White). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6.

Cornerstone: Mitigating Systems

- Preliminary White. The NRC identified a preliminary White finding and associated apparent violation of Technical Specification 6.8.1, "Procedures and Programs," and Technical Specification 3.4.B, "Automatic Depressurization System," because Exelon failed to implement a procedure related to the maintenance of safety related equipment. Specifically, Exelon personnel did not follow electromatic relief valve (EMRV) reassembly instructions that required personnel to reinstall previously removed lock washers from the 'E' EMRV cut-out switch lever. The incorrect reassembly caused excessive friction between the solenoid frame and the cut-out switch lever, which led to the 'E' EMRV's failure to perform its safety function. This resulted in one inoperable EMRV for greater than the Technical Specification allowed outage time. The issue was entered into the corrective action program as issue report 2722109, and Exelon's immediate corrective actions include installing new cut-out switch lever plates with increased clearances, replacing star lock washers with split ring lock washers for additional clearance, and verifying the five EMRV solenoid actuators being installed into the drywell following the most recent refueling outage were correctly assembled.

  The finding is more than minor because it adversely affects the human performance quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the missing lock washers due to the incorrect EMRV lever plate reassembly caused excessive friction between the solenoid frame and the cut-out switch lever, causing the cut-out switch lever to become bound in the energized position. This led to the 'E' EMRV's failure to perform its safety function. The inspectors screened this issue for safety significance in accordance with Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," and determined a detailed risk evaluation was required because the 'E' EMRV had potentially failed or was unreliable for greater than the Technical Specification allowed outage time.

A detailed risk evaluation concluded that the increase in core damage frequency (CDF) related to the failure of the 'E' EMRV is 5.4E-6/year; therefore, this finding was preliminary determined to have a low to moderate safety significance (White). Due to the nature of the failure, no recovery credit was assigned. The dominant core damage sequences involve loss of main feedwater events with operator errors resulting in failure to make-up to the isolation condensers or otherwise maintain reactor vessel level and the loss of reactor pressure vessel depressurization capability (due to common cause failure of the remaining four EMRVs).

The finding has a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because Exelon personnel did not follow station processes. Specifically, Exelon did not follow written instructions when reassembling the 'E' EMRV. The missing lock washers resulted in excessive friction between the solenoid frame and cut-out switch lever, causing the cut-out switch lever to become bound in the energized position, which led to the 'E' EMRV's failure to perform its safety function. [H.8] (Section 4OA2)

REPORT DETAILS

Summary of Plant Status

Oyster Creek began the inspection period with the reactor shut down for the 1R26 refueling outage. Operators commenced a startup of the reactor on October 12. The unit reached full power on October 15. On November 19, operators lowered power to 85 percent for a rod pattern adjustment. On November 20, an automatic scram occurred at approximately 90 percent due to a fault in the turbine controls system. Following repairs, Oyster Creek operators commenced startup on November 23. The unit remained at 85 percent power until December 12. On December 12, operators commenced a reactor shutdown and entered a planned maintenance outage (1M40). Following repairs, Oyster Creek operators commenced startup on December 18. The unit reached full power on December 19. On December 20, operators lowered power to 85 percent for a rod pattern adjustment and returned the unit to full power later that day. Oyster Creek remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 1 sample)

Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors reviewed Exelon's readiness for the onset of seasonal cold temperatures. The review focused on the intake structure and the emergency diesel generators. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), technical specifications, control room logs, and the corrective action program to determine what temperatures or other seasonal weather could challenge these systems, and to ensure Exelon personnel had adequately prepared for these challenges. The inspectors reviewed station procedures, including Exelon's seasonal weather preparation procedure and applicable operating procedures. The inspectors performed walkdowns of the selected systems to ensure station personnel identified issues that could challenge the operability of the systems during cold weather conditions. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment

Partial System Walkdowns (71111.04 – 2 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- ‘B’ emergency service water pump on October 17, 2016
- ‘A’ isolation condenser when the ‘B’ isolation condenser was out of service on October 19, 2016

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the UFSAR, technical specifications, work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted the system’s performance of its intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether Exelon staff had properly identified equipment issues and entered them into the corrective action program for resolution with the appropriate significance characterization.

b. Findings

No findings were identified.

1R05 Fire Protection

Resident Inspector Quarterly Walkdowns (71111.05Q – 3 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that Exelon controlled combustible materials and ignition sources in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment was available for use as specified in the area pre-fire plan, and passive fire barriers were maintained in good material condition. The inspectors also verified that station personnel implemented compensatory measures for out of service, degraded, or inoperable fire protection equipment, as applicable, in accordance with procedures.

- Motor generator set room on November 30, 2016
- New cable spreading room on November 30, 2016
- ‘C’ battery room on November 30, 2016

b. Findings

No findings were identified.

1R07 Heat Sink Performance (711111.07A – 1 sample)

a. Inspection Scope

The inspectors reviewed the 'A', 'B', and 'C' shutdown cooling heat exchanger readiness and availability to perform its safety functions. The inspectors reviewed the design basis for the component and verified Exelon's commitments to NRC Generic Letter 89-13, "Service Water System Requirements Affecting Safety-Related Equipment." The inspectors discussed the results of the most recent inspection with engineering staff and reviewed pictures of the as-found and as-left conditions. The inspectors verified that Exelon initiated appropriate corrective actions for identified deficiencies. The inspectors also verified that the number of tubes plugged within the heat exchanger did not exceed the maximum amount allowed.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11Q – 1 sample)

Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on October 26, 2016, which included an anticipated transient without a scram and a loss of coolant accident. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the technical specification action statements entered by the shift senior reactor operator. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 3 samples)

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the maintenance and emergent work activities listed below to verify that Exelon performed the appropriate risk assessments prior to removing equipment for work. The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that Exelon personnel performed risk assessments as required by Title 10 of the *Code of Federal Regulations* (CFR) 50.65(a)(4) and that the assessments were accurate and complete.

When Exelon performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work and discussed the results of the assessment with the station's probabilistic risk analyst to verify plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- Containment spray system II and emergency service water system II out of service for planned maintenance on October 17, 2016
- 'B' isolation condenser out of service for emergent maintenance on October 19, 2016
- Core spray system I out of service for planned maintenance on November 10, 2016

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 3 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions based on the risk significance of the associated components and systems:

- EMRVs following inspection to rule out common cause failure on October 1, 2016
- Control rod control switch following switch issues during control rod withdrawal on October 11, 2016
- Exhaust fan 1-9 high temperature on October 17, 2016

The inspectors evaluated the technical adequacy of the operability determinations to assess whether technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and UFSAR to Exelon's evaluations to determine whether the components or systems were operable. The inspectors confirmed, where appropriate, compliance with bounding limitations associated with the evaluations. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled by Exelon.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 1 sample)

Permanent Modifications

a. Inspection Scope

The inspectors evaluated a modification to the design of the EMRV solenoid actuators. The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the upgrade and design change, including installation of new cut-out switch lever plates with increased clearances and replacement of star lock washers with split-ring lock washers. The inspectors also reviewed revisions to the maintenance refurbishment procedure and interviewed engineering and maintenance personnel to ensure the procedure could be reasonably performed.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 7 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests for the maintenance activities listed below to verify that procedures and test activities adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure were consistent with the information in the applicable licensing basis and/or design basis documents, and that the test results were properly reviewed and accepted and problems were appropriately documented. The inspectors also walked down the affected job site, observed the pre-job brief and post-job critique where possible, confirmed work site cleanliness was maintained, and witnessed the test or reviewed test data to verify quality control hold point were performed and checked, and that results adequately demonstrated restoration of the affected safety functions.

- No. 2 emergency diesel generator following relay replacement on October 1, 2016
- Isolation condenser high point vent valve after valve replacement on October 3, 2016
- ‘B’ control rod drive pump following coupling inspection on October 3, 2016
- ‘A’ core spray main pump motor after relay replacement on October 4, 2016
- Drywell bulk temperature element after replacement on October 5, 2016
- Main steam isolation valve accumulator check valve after replacement on October 6, 2016
- EMRVs following design modification on October 7, 2016

b. Findings

No findings were identified.

1R20 Refueling and Other Outage Activities (71111.20 – 3 samples)

a. Inspection Scope

The inspectors reviewed the station's work schedule and outage risk plan for the Oyster Creek's maintenance and refueling outage (1R26), conducted September 19 through October 12, 2016, forced outage (1F39), and maintenance outage (1M40). The inspectors reviewed Exelon development and implementation of outage plans and schedules to verify that risk, industry experience, previous site-specific problems, and defense-in-depth were considered. During the outage, the inspectors observed portions of the shutdown and cooldown processes and monitored controls associated with the following outage activities:

- Configuration management, including maintenance of defense-in-depth, commensurate with the outage plan for the key safety functions and compliance with the applicable technical specifications when taking equipment out of service
- Implementation of clearance activities and confirmation that tags were properly hung and that equipment was appropriately configured to safely support the associated work or testing
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and instrument error accounting
- Status and configuration of electrical systems and switchyard activities to ensure that technical specifications were met
- Monitoring of decay heat removal operations
- Impact of outage work on the ability of the operators to operate the spent fuel pool cooling system
- Reactor water inventory controls, including flow paths, configurations, alternative means for inventory additions, and controls to prevent inventory loss
- Activities that could affect reactivity
- Maintenance of secondary containment as required by technical specifications
- Refueling activities, including fuel handling and fuel receipt inspections
- Fatigue management
- Tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block the emergency core cooling system suction strainers, and startup and ascension to full power operation
- Identification and resolution of problems related to refueling outage activities

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 3 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant structures, systems, and components to assess whether test results satisfied technical specifications, the UFSAR, and Exelon procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied.

Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

- No. 2 emergency diesel generator automatic actuation test on October 1, 2016
- Automatic depressurization system actuation circuit test and calibration on October 3, 2016
- Standby liquid control functional test on October 3, 2016

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (IP 71114.04 – 1 sample)

a. Inspection Scope

Exelon implemented various changes to the Oyster Creek Emergency Action Levels, Emergency Plan, and Implementing Procedures. Exelon had determined that, in accordance with 10 CFR 50.54(q)(3), any change made to the Emergency Action Levels, Emergency Plan, and its lower-tier implementing procedures, had not resulted in any reduction in effectiveness of the Plan, and that the revised Plan continued to meet the standards in 50.47(b) and the requirements of 10 CFR 50 Appendix E.

The inspectors performed an in-office review of all Emergency Action Level and Emergency Plan changes submitted by Exelon as required by 10 CFR 50.54(q)(5), including the changes to lower-tier emergency plan implementing procedures, to evaluate for any potential reductions in effectiveness of the Emergency Plan. This review by the inspectors was not documented in an NRC Safety Evaluation Report and does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety. The requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06 – 1 sample)

Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine Exelon emergency drill on October 26, 2016, to identify any weaknesses and deficiencies in the classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator, technical support center, and emergency operations facility to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures.

The inspectors also attended the station drill critique to compare inspector observations with those identified by Exelon staff in order to evaluate Exelon's critique and to verify whether the Exelon staff was properly identifying weaknesses and entering them into the corrective action program.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Mitigating Systems Performance Index (5 samples)

a. Inspection Scope

The inspectors reviewed Exelon's submittal of the Mitigating Systems Performance Index for the following systems for the period of October 1, 2015, through September 30, 2016:

- Emergency Alternating Current Power System
- High Pressure Injection System – Core Spray
- Heat Removal – Isolation Condensers
- Residual Heat Removal – Containment Spray
- Cooling Water System

To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7. The inspectors also reviewed Exelon's operator narrative logs, condition reports, mitigating systems performance index derivation reports, event reports, and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 2 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, "Problem Identification and Resolution," the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify Exelon entered issues into the corrective action program at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the corrective action program and periodically attended condition report screening meetings.

The inspectors also confirmed, on a sampling basis, that, as applicable, for identified defects and non-conformances, Exelon performed an evaluation in accordance with 10 CFR Part 21.

b. Findings

No findings were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a semi-annual review of site issues to identify trends that might indicate the existence of more significant safety concerns. As part of this review, the inspectors included repetitive or closely-related issues that may have been documented by Exelon in trend reports, site performance indicators, major equipment problem lists, system health reports, and maintenance rule assessments, and maintenance or corrective action program backlogs. The inspectors also reviewed Exelon's corrective action program database for third and fourth quarters of 2016 to assess condition reports written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRC's daily condition report review.

b. Findings

No findings were identified.

The inspectors evaluated a sample of corrective maintenance backlogs, control room deficiency tags, open operability evaluations, and operator workarounds. The inspectors verified that these issues were addressed within the scope of the corrective action program.

.3 Annual Sample: 'E' EMRV Failure to Stroke

a. Inspection Scope

The inspectors conducted an in-depth review of Exelon's apparent cause and corrective actions associated with the 'E' EMRV failure to stroke (issue report 2722109). Specifically, on September 19, 2016, during the 1R26 refueling outage with the reactor in cold shutdown, as-found testing was performed on all five EMRVs. The 'E' EMRV failed to open from the main control room.

The inspectors assessed Exelon's problem identification threshold, cause analysis, extent of condition review, and the prioritization and timeliness of Exelon's corrective actions to determine whether Exelon was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the completed corrective actions were appropriate. The inspectors compared the actions taken to Exelon's corrective action program and the requirements of 10 CFR 50, Appendix B. In addition, the inspectors performed field walkdowns and interviewed engineering personnel to assess the effectiveness of the implemented corrective actions.

b. Findings

Introduction. The inspectors identified a preliminary White finding and associated apparent violation of Technical Specification 6.8.1, "Procedures and Programs," and Technical Specification 3.4.B, "Automatic Depressurization System," because Exelon failed to implement a procedure related to the maintenance of safety related equipment. Specifically, Exelon personnel did not follow EMRV reassembly instructions that required personnel to reinstall previously removed lock washers from the 'E' EMRV cut-out switch lever. The incorrect reassembly caused excessive friction between the solenoid frame and the cut-out switch lever, which led to the 'E' EMRV's failure to perform its safety function.

Description. The Oyster Creek EMRVs are six-inch electrically actuated pressure relief valves originally manufactured by Dresser Industries. There are five EMRVs on the main steam lines between the reactor pressure vessel and the main steam line isolation valves within the drywell. The EMRVs consist of a main valve assembly, pilot valve assembly, and a solenoid actuator. When the solenoid actuator is energized, the solenoid coil pulls the plunger down, which pushes the pilot valve operating lever down, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented, resulting in an unbalanced steam pressure across the main disc and moving the main disc downward from its seat, opening the main valve.

The safety function of the EMRV is to depressurize the reactor during a small break loss-of-coolant accident to permit the low pressure core spray system to inject water into the reactor core. Oyster Creek Technical Specifications section 3.4.B, "Automatic Depressurization System," requires, in part, that five EMRVs shall be operable and if at any time there are only four operable EMRVs, the reactor may remain in operation for a period not to exceed three days. If one EMRV is inoperable for more than three days, then reactor pressure shall be reduced to 110 psig or less, within 24 hours.

The EMRV solenoid actuator has a pickup coil and a hold coil in series. When the EMRV is closed (i.e. de-energized), the cut-out switch on the side of the solenoid actuator has a normally closed contact that bypasses the hold coil. When the EMRV is energized to open, 125 volts direct current is applied to the pilot valve solenoid, and the current goes through the pickup coil and the cut-out switch. When the pilot valve changes position, the plunger is pulled down, which strikes the cut-out switch lever. This opens the cut-out switch and places the hold coil in series with the pickup coil. This is done to prevent burning out the pickup coil.

Exelon conducted as found testing per work order R2246997 on September 19, 2016, after a planned shutdown for refueling outage 1R26. The testing included stroking the five EMRVs in-situ from the control room to verify past operability. The control room operators successfully stroked A-D EMRVs, but the 'E' EMRV failed to open. Operators once again attempted to open the 'E' EMRV from the control room, and electricians in the field confirmed that the 'E' EMRV failed to open.

While troubleshooting the 'E' EMRV, Exelon determined that the lock washers, located on the cut-out switch hinge pin bolts, were missing, and the cut-out switch lever was bent. Exelon subsequently determined that the bent cut-out switch lever caused excessive mechanical binding, which prevented the cut-out switch lever from changing positions. Because the cut-out switch remained open, the hold coil remained in series with the pickup coil and prevented the actuator from developing sufficient magnetic force to pull the plunger down and engage the pilot valve operating lever.

The lock washers are installed to prevent the hinge pin bolts from becoming loose due to vibration. A consequence of missing lock washers was that the hinge pins went in too far and bottomed out. The licensee concluded this condition caused the cut-out switch lever to bend. This bending resulted in excessive friction between the solenoid frame and cut-out switch lever, causing the cut-out switch lever to become bound in the energized position. This binding kept the switch open and the hold coil in the circuit.

Exelon completed an equipment apparent cause evaluation, issue report 2717363. The evaluation stated that on September 5, 2014, Oyster Creek refurbished their EMRVs per 2400-SME-3918.03, "EMRV Solenoid Operator Removal, Refurbishment, and Installation," supplemented with modification instructions from Engineering Change Request 14-00371. "Unmodified Actuator Disassembly" section, Step 9 of the written instructions required technicians, in part, "remove the two hex head screw pins (i.e. the hinge pin bolts) from the cut-out switch lever arm… retain the lock washers for re-use." "Modification Instructions" section, Step 2 in the written instructions states, in part, "install the two new screw pins with the previously removed lock washers." Exelon determined that maintenance technicians incorrectly reassembled the 'E' EMRV when they failed to install the washers on the cut-out switch hinge pins. The incorrect reassembly led to the 'E' EMRV's failure to perform its safety function.

Exelon entered the issue into their corrective action program, issue report 2722109. Exelon's corrective actions include installing new cut-out switch lever plates with increased clearances, replacing star lock washers with split ring lock washers for additional clearance, and verifying the five EMRV solenoid actuators being installed in the plant following the refueling outage were correctly assembled.

<u>Analysis</u>. The inspectors determined that Exelon's failure to accomplish the EMRV refurbishment per the written instructions was a performance deficiency. The inspectors reviewed IMC 0612, Appendix B, "Issue Screening," and determined that the finding is more than minor because it adversely affected the human performance quality attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, the missing lock washers due to the incorrect EMRV lever plate reassembly caused excessive friction between the solenoid frame and the cut-out switch lever, causing the cut-out switch lever to become bound in the energized position. This led to the inability of the 'E' EMRV to perform its safety function if demanded.

The inspectors screened this issue for safety significance in accordance with IMC 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," and determined a detailed risk evaluation was required because the 'E' EMRV had potentially failed or was unreliable for greater than the Technical Specification allowed outage time. A Region I Senior Reactor Analyst (SRA) used the Standardized Plant Analysis Risk (SPAR) model for Oyster Creek, Version 8.22, dated 11/18/2009, and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 8.1.4, to complete the detailed risk evaluation.

The SRA made the following assumptions and changes to the SPAR model to best represent the failed condition of the ‘E’ EMRV and current plant design and operation:

- Exposure time for the failed ‘E’ EMRV was established at one year.
- The failure probability for the SPAR model basic event for the ‘E’ EMRV, ADS-SRV-CC-NR108E, was changed to TRUE, to represent the failure and to account for the increased potential for common cause failure of the remaining EMRVs in the five valve grouping. An increased common cause failure probability is used based upon the technicians who performed the EMRV refurbishment in September 2014, potentially having introduced the same failure mechanism in all five EMRVs. This assumption is consistent with guidance in Risk Assessment Standardization Project Volume 1, Section 5.
- With the exception of anticipated transient without scram events, the EMRV success criterion is 2 of 5 valves opening for successful reactor pressure vessel depressurization. No change was made to the anticipated transient without scram depressurization success criteria because these events represent a small fraction of the postulated core damage sequences. This revised success criterion was achieved by changing the Manual Reactor Depress (DEP) Fault Tree, DEP-1 N of M OR Gate from 3 of 5 to 4 of 5.
- Based upon more realistic data, the initiating event likelihood for IE-LOFC was changed from 1.69E-1/year to 7.54E-2/year, and the failure probability for ISO-ICWATERHMR was changed from 0.5 to 0.01.
- Compound event ADS-SRV-CF-VALVS was changed from staggered testing (PLUGCCFSTAG) to non-staggered testing (PLUGCCFALPHA) to reflect station practice of post-maintenance testing all valves at approximately the same time. This compound event failure count was also revised from 3 to 4.
- Based upon ‘E’ EMRV failure being identified during an outage, no unavailability time due to maintenance was used.
- No recovery credit was applied.
- Truncation remained at 1E-11.

Based upon the above assumptions and modeling changes, the estimated conditional core damage frequency was calculated at 9.47E-6/year. Subtracting the baseline risk of 4.76E-6/year, resulted in the estimated increase in internal events risk of approximately 4.71E-6/year for the failure of the ‘E’ EMRV. The dominant core damage sequences involve loss of main feedwater events with operator errors resulting in failure to make-up to the isolation condensers or otherwise maintain reactor vessel level and the loss of reactor pressure vessel depressurization capability (due to common cause failure of the remaining four EMRVs).

The Oyster Creek SPAR model does not include external events. Accordingly, the SRA reviewed the Oyster Creek Individual Plant Examination of External Events (IPEEE), the Fire Hazards Analysis and Exelon staff provided Oyster Creek fire probabilistic risk assessment (PRA) results and insights. From this review, the SRA determined that the importance of the EMRVs to fire mitigation is significantly higher than seismic and other external event hazards. Consequently the external events contribution to CDF was predominately due to postulated fire events.

Exelon estimated the increase in core damage frequency (due to postulated fires and the failure of the 'E' EMRV) at 7E-7/year. The dominant cutsets involve fires initiating in the Turbine and Office Buildings that not only compromise the EMRVs, but also result in failure of the control rod drive system (high pressure make-up) and/or make-up sources to the isolation condensers.

Based upon the Oyster Creek SPAR model internal events estimated increase in CDF = 4.71E-6/year and an Exelon fire PRA approximated external events increase in CDF = 7E-7/year, the total increase in CDF as a result of the failure of the E EMRV for one year is in the mid E-6/year range, or low to moderate safety significance (White).

For issues resulting in an increase in CDF > 1E-7, IMC 0609 requires an evaluation of Large Early Release Frequency (LERF) using the guidance of NUREG-1765, "Basis Document for LERF Significance Determination Process," and IMC 0609, Appendix H, "Containment Integrity SDP." The failure of the 'E' EMRV would be considered a Type A finding and, as such, the calculated increase in CDF value is used in conjunction with an appropriate LERF factor (multiplier) to determine the estimated increase in LERF associated with the issue. Per Appendix H, Table 5.2, LERF factors of 1.0 or 0.6 are used for high pressure core damage accident sequences with the drywell dry or flooded, respectively. These Appendix H LERF factors are considered conservative bounding values. More recent insights from an NRC Office of Research sponsored study by Energy Research, Inc. (ERI/NRC-03-04, November 2003) and the State of the Art Reactor Consequence Analysis Project at Peach Bottom Nuclear Power Station (NUREG/CR-7110) have identified that improved modeling and analysis of anticipated types and sizes of reactor coolant system ruptures, projected containment heating and fuel-coolant interactions, and operator actions taken in accordance with emergency operating procedures significantly reduce the calculated potential for containment breach and the likelihood of a large early release. Exelon's internal events model estimates LERF values directly and identified a delta LERF value of 9.12E-9/year. Exelon's external (fire) delta LERF estimate was qualitatively evaluated and used an upper bound LERF multiplier of 0.1. The external events increase in LERF contribution was estimated at approximately 8E-8/year. Exelon's total estimated increase (internal and external) in LERF attributed to the failure of the 'E' EMRV was calculated to be < 1E-7/year or Green. The SRA considered the 0.1 LERF multiplier a reasonable and appropriate value for estimating the increase in LERF for this issue. Using the SPAR model estimated internal events delta CDF (4.71E-6/year) and Exelon's calculated external (fire) events contribution to delta CDF (7E-7/year), the total delta CDF value is approximately 5.4E-6/year or low to moderate safety significance (White). Using the conservative LERF multiplier of 0.1, the increase in LERF is estimated at 5.4E-7/year, or low to moderate safety significance (White).

In accordance with IMC 0609, the higher of the two risk metric values is used to assign a preliminary safety significance. In this instance, both the delta CDF (5.4E-6/year) and delta LERF (5.4E-7/year) values equate to a low to moderate safety significant (White) finding.

The finding has a cross-cutting aspect in the area of Human Performance, Procedure Adherence, because Exelon personnel did not follow station processes. Specifically, Exelon did not follow written instructions when reassembling the 'E' EMRV. The missing lock washers resulted in excessive friction between the solenoid frame and cut-out switch lever, causing the cut-out switch lever to become bound in the energized position, which led to the 'E' EMRV's failure to perform its safety function. (H.8)

Enforcement. Technical Specification 6.8.1, "Procedures and Programs," requires, in part, that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, as referenced in NO-AA-10, "Quality Assurance Topical Report." Regulatory Guide 1.33, Appendix A, Section 9 requires, in part, that procedures for maintenance that can affect the performance of safety-related equipment be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. ECR 14-00371, "Oyster Creek EMRV Conversion Instructions," Revision 1, "Unmodified Actuator Disassembly" section, Step 9 requires, in part, removal of the two hex head screw pins from the cutout switch lever arm and retaining the lock washers for re-use during disassembly. "Modification Instructions" section, Step 2 requires, in part, that two new screw pins be installed with the previously removed lockwashers. Technical Specification 3.4.B states, in part, that five electromatic relief valves shall be operable and if at any time there are only four operable EMRVs, the reactor may remain in operation for a period not to exceed three days. If one EMRV is inoperable for more than three days, then reactor pressure shall be reduced to 110 psig or less, within 24 hours.

Contrary to the above, on September 5, 2014, Exelon failed to properly implement a procedure related to the maintenance of safety-related equipment. Specifically, Exelon personnel did not follow the EMRV reassembly instructions for the 'E' EMRV that directed reinstallation of the lever plate with the previously removed lock washers. The incorrect reassembly caused excessive friction between the solenoid frame and the cut-out switch lever, causing the cut-out switch lever to become bound in the energized position, impacting the ability of the 'E' EMRV to perform its safety function. Between September 5, 2014 and September 19, 2016, this incorrect reassembly resulted in the 'E' EMRV being inoperable for greater than the Technical Specification allowed outage time.

Exelon entered the issue into the corrective action program as issue report 2722109. Exelon's immediate corrective actions included installing new cut-out switch lever plates with increased clearances, replacing star lock washers with split ring lock washers for additional clearance, and verifying the five EMRV solenoid actuators being installed in the plant following the refueling outage were correctly assembled. This issue is being characterized as an apparent violation (AV) in accordance with the NRC's Enforcement Policy, and its final significance will be dispositioned in separate future correspondence. (**AV 05000219/2016004-01, 'E' EMRV Failure to Stroke Due to Incorrect Reassembly**)

c. Observations

The inspectors determined Exelon's overall response to the issue was commensurate with the safety significance, was timely, and included appropriate compensatory actions. Specifically, once the issue was identified, Exelon promptly established a cross-discipline team to investigate the cause of the issue through a thorough extent of condition and technical cause determination. The inspectors concluded that actions completed to install new cut-out switch lever plates with increased clearances, replace the star lock washers with the split-ring lock washers, and verify the five EMRV solenoid actuators being installed in the plant were correctly assembled were reasonable to correct the problem and prevent reoccurrence.

The inspectors noted inconsistencies between Exelon's technical training, maintenance procedure 2400-SME-3918.03, "EMRV Solenoid Operator Removal, Refurbishment, and Installation," and modification work order C2032680. The technical training emphasized step by step performance of 2400-SME-3918.03, which did not include steps to remove and re-install the cut-out switch lever plate. The modification work order provided written instructions to remove and re-install the lever plate, with specific steps to also remove and re-install the lock washers on the plate hinge pin bolts.

When interviewed, an EMRV lead technician stated that the plate was always removed during the refurbishment process.

While troubleshooting the 'E' EMRV failure to stroke, the inspectors identified that the completed work orders for the 2014 modification had not been retained by Exelon and were listed as "lost in the field." Exelon documented this issue in issue reports 2736322 and 3959349.

The inspectors independently evaluated this issue for significance in accordance with IMC 0612, Appendix B, "Issue Screening," and IMC 0612, Appendix E, "Examples of Minor Issues." Although the completed work orders were lost in the field, the action was not a precursor to a significant event, did not have the potential to lead to a more significant safety concern, would not have caused a performance indicator to exceed a threshold, and did not negatively affect any of the cornerstone objectives, Therefore, the inspectors determined this issue was minor, and as a result, was not subject to enforcement action in accordance with the NRC's Enforcement Policy.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 3 samples)

.1 Plant Events

a. Inspection Scope

On November 20, 2016, the inspectors reviewed and/or observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems regarding an automatic reactor scram caused by a fault in the turbine controls system. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that Exelon made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR Parts 50.72 and 50.73. The inspectors reviewed Exelon's follow-up actions related to the events to assure that Exelon implemented appropriate corrective actions commensurate with their safety significance.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report (LER) 05000219/2016-004-00: Technical Specification Violation Due to Main Steam Safety Valve Setpoint Discovered Out of Tolerance

On September 29, 2016, during the 1R26 refueling outage with the reactor in cold shutdown, Exelon discovered through routine laboratory as-found testing that a safety valve, V-1-164, did not meet the acceptance criteria of Technical Specification 2.3.F, "Reactor High Pressure, Safety Valve Initiation." Specifically, the as-found setpoint value for the safety valve was 1181psig, which exceed the Technical Specification acceptance criteria 1221psig +/- 36psi (+/- 3%).

The safety significance of the issue is considered minimal because the safety valve opened lower than the American Society of Mechanical Engineers code setpoint, which would allow the valve to maintain reactor pressure vessel pressure below the 1375 psig limit. Additionally, this valve is no longer installed in the plant.

Exelon determined the cause of the failure was attributed to setpoint drift. Per the American Society of Mechanical Engineers code, upon discovering the safety valve setpoint was out of tolerance, four additional safety valves were tested. The setpoints for these valves met the acceptance criteria of Technical Specification 2.3.F.

The inspectors determined that the reported violation was minor, as it was not a precursor to a significant event, did not have the potential to lead to a more significant safety concern, did not relate to a performance indicator that would have exceeded a threshold, and did not adversely affect any of the cornerstone objectives. The inspectors did not identify any new issues during the review of the LER. This LER is closed.

.3 (Closed) LER 05000219/2016-005-00: Technical Specification Prohibited Condition Caused by One Electromatic Relief Valve Inoperable for Greater than Allowed Outage Time

On September 19, 2016, during the 1R26 refueling outage with the reactor in cold shutdown, as-found testing was performed on all five EMRVs. The ‘E’ EMRV failed to open from the main control room. Exelon determined the cause of the failure was due to missing lock washers not installed on the cut-out switch assembly. Based on this information, Exelon concluded that one of the five EMRVs were inoperable longer than the Technical Specification allowed outage time of 24 hours.

The inspectors did not identify any new issues during the review of the LER. One violation was identified and discussed in Section 4OA2.3 of this report. This LER is closed.

4OA6 Meetings, Including Exit

On January 4, 2017, the inspectors presented the inspection results to Mr. Stathes, Site Vice President, and other members of the Oyster Creek staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

G. Stathes, Site Vice President
M. Gillin, Plant Manager
A. Bready, Risk Analyst
M. Capone, System Engineer
M. Chanda, Emergency Preparedness Manager
D. Chernesky, Director, Maintenance
J. Clark, Manager, Environmental/Chemistry
P. Deckman, Manager, Maintenance and Technical Training
R. Dutes, Regulatory Assurance Specialist
J. Jimenez, Senior Regulatory Assurance Specialist
T. Keenan, Manager, Site Security
A. Krukowski, Shift Operations Superintendent
M. McKenna, Manager, Regulatory Assurance
H. Ray, Senior Manager, Design Engineering
J. Renda, Director, Work Management
M. Rossi, Licensed Operator Requalification Training Lead
S. Schwartz, Engineering, System Manager
J. Sharkey, Operations Training Manager
J. Stanley, Director, Engineering
C. Symonds, Director, Training
J. Weissinger, Director, Operations
K. Wolf, Radiation Protection Manager

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened

05000219/2016004-01	AV	'E' EMRV Failure to Stroke Due to Incorrect Reassembly (Section 4OA2)

Closed

05000219/2016-004-00	LER	Technical Specification Violation Due to Main Steam Safety Valve Setpoint Discovered Out of Tolerance (Section 4OA3.2)
05000219/2016-005-00	LER	Technical Specification Prohibited Condition Caused by One Electromatic Relief Valve Inoperable for Greater than Allowed Outage Time (Section 4OA3.3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
322, Service Water System, Revision 90
323.3, Normal Operation of the Main Condenser Circulating Water System, Revision 17
341, Emergency Diesel Generator Operation, Revision 112
OP-OC-108-109-1001, Severe Weather Preparation T&RM for Oyster Creek, Revision 35
OP-OC-108-109-1002, Cold Weather Freeze Inspections, Revision 5
SY-AA-101-146, Severe Weather Preparation and Response, Revision 2
WC-AA-107, Seasonal Readiness, Revision 17

Condition Reports
2640632 3952299 3952300 2647874 3951380

Miscellaneous
Oyster Creek Certification of 2016-2017 Winter Readiness, dated November 25, 2016

Section 1R04: Equipment Alignment

Procedures
307, Isolation Condenser System, Revision 125
310, Containment Spray System, Revision 11

Condition Reports
2728981

Section 1R05: Fire Protection

Procedures
ER-AA-600-1069, High Risk Area Identification, Revision 1
OP-OC-201-008-1014, MG Set Room/Mechanical Equipment Room (MOB2), Revision 0
OP-OC-201-008-1019, New Cable Spreading Room (NCSR), Revision 1
OP-OC-201-008-1024, 4160V Switchgear Room, "C" Battery Room, Revision 2

Section 1R07: Heat Sink Performance

Procedures
305, Shutdown Cooling System Operation, Revision 123
ER-OC-340-1001, Oyster Creek Generic Letter 89-13 Program Basis Document, Revision 4
ABN-3, Loss of Shutdown Cooling, Revision 4

Calculations
C-3560-214-005, Oyster Creek Shutdown Cooling System Calculation of Shutdown Heat Loads, Revision 0
C-5360-201-006, Shutdown Cooling System Heat Exchangers Performance Evaluation, Revision 0

Drawings
GE 148F711, Reactor Shutdown Cooling System Flow Diagram, Sheet 1, Revision 45

Miscellaneous

Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.5, Containment, Amendment 247

Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 5.4, Component and Subsystem Design, Revision 18

Section 1R11: Licensed Operator Requalification Program

Procedures

HU-AA-101, Human Performance Tools and Verification Practices, Revision 9

TQ-AA-150, Operator Training Programs, Revision 11

TQ-AA-155, Conduct of Simulator Training and Evaluation, Revision 5

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

WC-AA-101, Online Work Control Process, Revision 26

WC-AA-101-1002, Online Scheduling Process, Revision 17

WC-AA-104, Integrated Risk Management, Revision 23

OP-AA-108-117, Protected Equipment Program, Revision 4

OP-MA-109-101, Clearance and Tagging, Revision 20

WC-OC-101-1001, Online Risk Management and Assessment, Revision 19

607.4.017, Containment Spray and Emergency Service Water Pump System 2 Operability and Quarterly Inservice Test, Revision 43

RAP-C3b, Cond B Flow Hi Possible Rupture, Revision 5

610.3.115, Core Spray System I Instrument Channel and Level Bistable Calibration and Test, Revision 51

Condition Reports

2730096

2730024

Maintenance Orders/Work Orders

M2414471

Miscellaneous

Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247

Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.8, Isolation Condenser, Amendment 241

Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18

Section 1R15: Operability Determinations and Functionality Assessments

Procedures

2400-SME-3918.03, EMRV Solenoid Operator Removal, Refurbishment, and Installation, Revision 23

Calculations
C-1302-822-5360-020, Condensation Through the SGTS Fan Drains, Revision 0
C-1302-732-5350-016, 120VAC Control Power Voltage Drop for Fan EF-1-9 (OCNGS), Revision 0

Condition Reports
2596285 2717363 2717451 2725367 2722109

Drawings
BR 2011, Reactor Building Ventilation Flow Diagram, Sheet 2, Revision 64
GU-3E-822-21-1000, Standby Gas Treatment Flow Diagram, Sheet 1, Revision 11
GE 729E182, Auto Depressurization System Electrical Elementary Diagram, Sheet 5, Revision 4

Maintenance Orders/Work Orders
R2275158 R2271230 R2267197 R2262905 R2260148 C2036953
M2411983

Miscellaneous
SDBD-OC-822, Design Basis Document for Standby Gas Treatment System/ Secondary Containment, Revision 1
OC-14-00371, EMRV Actuator Spring Modification, Revision 1
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.5, Fission Product Removal and Control Systems, Revision 17
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18

Section 1R18: Plant Modifications

Procedures
2400-SME-3918.03, EMRV Solenoid Operator Removal, Refurbishment, and Installation, Revision 23

Condition Reports
2723639

Drawings
GE 729E182, Auto Depressurization System Electrical Elementary Diagram, Sheet 5, Revision 4

Maintenance Orders/Work Orders
C2036953 R2208895 R2208891 R2246997 C2032680

Miscellaneous
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18

Section 1R19: Post-Maintenance Testing

Procedures
636.4.013, Diesel Generator #2 Load Test, Revision 49
341, Emergency Diesel Generator Operation, Revision 112
302.1, Control Rod Drive System, Revision 116
617.4.001, CRD Pump Operability Test, Revision 48
624.4.001, Main Steam Valve Position Indication and IST Test, Revision 18
635.2.001, 4160 Switchgear Buses (A, B, C, D) and Circulating Water Pump Protective Relay Surveillance, Revision 73

Condition Reports

2724777	2722892	2723227	2723124	2722519	2722730
2723058	2723303	2723298	2723298	2723290	2723289
2723220	2723931	2724156	2719504		

Drawings
GE 729E182, Auto Depressurization System Electrical Elementary Diagram, Sheet 5, Revision 4

Maintenance Orders/Work Orders

R2280833	R2175512	C2037012	A2412862	C2036953

Miscellaneous
SBBD-OC-225, Design Basis Document for Control Rod Drive System, Revision 1
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.7, Auxiliary Electrical Power, Amendment 278
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 8.3, Onsite Power Systems, Revision 18
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.2, Reactivity Control, Amendment 178
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 3.9, Mechanical Systems and Components, Revision 17
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18

Section 1R20: Refueling and Other Outage Activities

Procedures
201, Plant Startup, Revision 102
203, Plant Shutdown, Revision 88
619.4.022, Scram Discharge Volume Vent And Drain Valve Functional Test, Revision 24

Condition Reports

2726221	2726039	2725842	2725917	2725837	2723932
2724153	2724121	2724123	2724149	2724165	2724213
2724242	2720990	2720993	2721333	2721379	2721491
2721510	2717363	2717206	2411955	2717206	2719288
2719508	2719510	2719522	2719737	2719777	2719797
2719842	2719849	2719850	2719856	2719867	2719937
2720186	2720380	2720416	2720465	2720832	2719734
2719106					

Section 1R22: Surveillance Testing

Procedures
636.4.001, Diesel Generator No. 1 Automatic Actuation Test, Revision 21
636.4.002, Diesel Generator No. 1 Automatic Actuation Test, Revision 18
602.3.005, ADS Actuation Circuit Test and Calibration, Revision 35

Condition Reports

2715743	2723051	2723056	2723060	2723061	2723033
2723046	2723041	2723037	2723040	2723044	2723042
2708304	2723675	2723681	2723684	2723676	2723678
2723682	2723679	2723091	2723087	2724153	2724345
2724523	2724226				

Maintenance Orders/Work Orders
R2246715 R2247951 R2246713

Miscellaneous
Technical Evaluation 1145336-07, Molded Case Circuit Breaker with Undervoltage Fitted Device Electrical Loading, Revision 0
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.7, Auxiliary Electrical Power, Amendment 278
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 8.3, Onsite Power Systems, Revision 18
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18
Oyster Creek Crosby Liquid Poison Valve Analysis, 11/2/2016
Wyle Laboratories Certificate of Conformance for Crosby Liquid Relief Valves, 9/10/2013 and 9/25/13

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Procedures
EP-AA-1010, Addendum 3 Emergency Action Levels for Oyster Creek Station, Revision 1
EP-AA-112-300, Operations Support Center Activation and Operation, Revision 9
EP-AA-120, Emergency Plan Administration, Revision 18
EP-AA-120-1001, 10 CFR 50.54(q) Change Evaluation, Revision 8

Section 1EP6: Drill Evaluation

Procedures
EP-MA-114-100-F-04, PAR Notification/Update Form, Revision G

Section 4OA1: Performance Indicator Verification

Procedures
ER-AA-600-1047, Mitigating Systems Performance Index Basis Document, Revision 11
LS-AA-2001, Collecting and Reporting of NRC Performance Indicator Data, Revision 14
LS-AA-2200, Mitigating System Performance Index Data Acquisition and Reporting, Revision 5

Miscellaneous
Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7
MSPI Margin Monthly Reports – 4Q2015-3Q2016
Oyster Creek Unit 1 4Q2015-3Q2016 MSPI Data

Section 4OA2: Problem Identification and Resolution

Procedures
2400-SME-3918.03, EMRV Solenoid Operator Removal, Refurbishment, and Installation, Revision 23
MA-AA-716-010, Maintenance Planning, Revision 23
MA-OC-716-1018, Oyster Creek Maintenance Standards and Expectations Criteria, Revision 2
MA-AA-716-003, Tool Pouch/Minor Maintenance, Revision 9
OP-OC-108-1001, Craft Capability, Revision 1
TQ-AA-161-J010, Maintenance Initial Training Matrix Job Aid, Revision 0
HU-AA-104-101, Procedure Use and Adherence, Revision 5

Condition Reports

2717363	2736322	2717451	2722121	2725367	2723639
2722109	2719504	2723645	2723655	1292853	1298593
1305459	4190727	1296568	1302712	4185904	

Drawings
GE 729E182, Auto Depressurization System Electrical Elementary Diagram, Sheet 5, Revision 4

Maintenance Orders/Work Orders

C2032680	C2036953	R2208895	R2208891	R2208890	R2208896
R2209077	R2246997	M2411983			

Miscellaneous
OC-14-00371, EMRV Actuator Spring Modification, Revision 1
M&T Common 100 Series Course, dated August 5, 2009
392, Instructions for Installation and Maintenance Consolidated Electromatic Relief Valve (ERV), Revision 1
Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247
Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18

Section 4OA3: Follow-up of Events and Notices of Enforcement Discretion

Condition Reports

2722356	3948641	3947774	3947740	3947440	3945051
3945375	3945123	3945099	3944611	3944618	3944804
2945068	3953195	3953128	3953207	2952961	2953107
3953113	3953215	3943104	3943113	3943114	

Miscellaneous

Oyster Creek Nuclear Generating Station Technical Specifications, Section 2.3, Limiting Safety System Settings, Amendment 261

Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 5.2, Integrity of Reactor Coolant Pressure Boundary, Revision 17

Oyster Creek Nuclear Generating Station Technical Specifications, Section 3.4, Emergency Cooling, Amendment 247

Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 6.3, Emergency Core Cooling System, Revision 18

Oyster Creek Nuclear Generating Station Updated Final Safety Analysis Report, Section 10.2, Turbine Generator, Revision 18

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AV	apparent violation
CDF	core damage frequency
CFR	*Code of Federal Regulations*
EMRV	electromatic relief valve
IMC	Inspection Manual Chapter
LER	licensee event report
LERF	large early release frequency
NRC	Nuclear Regulatory Commission
PRA	probabilistic risk assessment
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SRA	senior reactor analyst
UFSAR	Updated Final Safety Analysis Report