UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

May 27, 2015

Mr. Bryan Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: OYSTER CREEK GENERATING STATION - NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT NO. 05000219/2015008

Dear Mr. Hanson:

On April 24, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Oyster Creek Generating Station. The enclosed inspection report documents the inspection results, which were discussed on March 19, 2015, with members of your staff.

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

This report documents two NRC-identified findings. One Severity Level IV violation and one finding of very low safety significance (Green). Both of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCV) consistent with Section 2.3.2.a of the NRC's Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspectors at Oyster Creek Generating Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Senior Resident Inspector at Oyster Creek Generating Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Docket Room or from the Publicly Available Records component of the NRC's document 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/

Paul G. Krohn, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure:
Inspection Report No. 05000219/2015008
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-219
License No.	DPR-16
Report No.	05000219/2015008
Licensee:	Exelon Generating Company, LLC
Facility:	Oyster Creek Generating Station
Location:	Forked River, New Jersey
Inspection Period:	April 6, 2015 through April 24, 2015
Inspectors:	J. Brand, Reactor Inspector, Division of Reactor Safety (DRS), Team Leader D. Kern, Senior Reactor Inspector, DRS J. Schoppy, Senior Reactor Inspector, DRS
Approved By:	Paul G. Krohn, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY

IR 05000219/2015008 4/6/2015 - 4/24/2015; Oyster Creek Generating Station; Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications.

This report covers a two week on-site inspection of the evaluations of changes, tests, and experiments and permanent plant modifications. The inspection was conducted by three region-based engineering inspectors. One Severity Level IV violation and one finding of very low safety significance (Green) were identified, both of which were also considered to be non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," dated June 2, 2011. 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 July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: **Barrier Integrity**

- Severity Level IV. The NRC identified a Severity Level IV non-cited violation (NCV) of Technical Specification (TS) 6.9.1.f.2 in that Exelon did not obtain NRC approval prior to using a specific analytical method to determine the core operating limits. Specifically, Exelon used an analytical method (TRACG04P) to determine the core operating limits (the average power range monitor protection settings which were identified in the Core Operating Limits Report (COLR)); however, that particular analytical method was not previously reviewed and approved by the NRC prior to Exelon's use. Exelon submitted a corrective action issue report (IR) to evaluate the condition (IR2482042).

  The team determined that Exelon did not comply with TS 6.9.1.f.2 requirements in that Exelon used an analytical method to determine the core operating limits without prior NRC approval. The team determined that this was a performance deficiency that was within Exelon's ability to foresee and correct. Because the issue had the potential to affect the NRC's ability to perform its regulatory function, the team evaluated this performance deficiency in accordance with the traditional enforcement process. Using the Enforcement Manual, the team characterized the violation as Severity Level IV because the underlying analytical method required NRC approval prior to use. Because this violation involves the traditional enforcement process and does not have an underlying technical violation that would be considered more-than-minor within the Reactor Oversight Process (ROP), the team did not assign a cross-cutting aspect to this violation in accordance with IMC 0612, "Power Reactor Inspection Reports," Section 07.03.c (Section 1R17.1).

- Green. The NRC identified an NCV of Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion XVI, "Corrective Action," for failure to promptly correct a condition adverse to quality. Specifically, corrective actions to restore design conformance of scram discharge volume (SDV) vent and drain valve pressure regulator valves V-6-961 and V-6-962 were not taken at the first opportunity of sufficient duration which was refueling outage 25 (1R25). Additionally, justification of the basis for deferral of corrective actions beyond the restart from 1R25 on October 2014, was not documented, reviewed, or approved by site management and/or oversight organizations as required by station procedure OP-AA-108-115, Section 4.5.5. Consequently, two non-conforming pressure regulator valves which perform a safety-related function remained installed following plant startup from 1R25, without appropriate evaluation and approval. Immediate corrective action included licensee determination that V-6-961 and 962 and the associated SDV vent and drain valves (V-15-119 and 121) remained operable, but non-conforming. Exelon entered the issue into their corrective action program as IR 2482851.

The finding was more than minor because it was associated with the design control and barrier performance attributes of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of ensuring the operational capability of the containment barrier to protect the public from radionuclide releases caused by accidents or events. Additionally, the finding was similar to example 5.c in Appendix E of Inspection Manual Chapter (IMC) 0612, because the control rod drive system was returned to service following 1R25 with two non-conforming (non-safety-related) pressure regulator valves installed in a safety-related application. The team determined the finding was of very low safety significance because it did not affect the reactor coolant system (RCS) boundary; did not affect the radiological barrier function of the control room, auxiliary building, or spent fuel pool systems or boundaries; and did not represent an actual open pathway in containment or involve a reduction in the function of hydrogen igniters. The team assigned a cross-cutting aspect in the area of Human Performance, Consistent Process (aspect H.13) because the organization did not use a consistent systematic approach to evaluate component operability after Exelon upgraded the classification of three pressure regulator valves from a non-safety to a safety-related status. (Section 1R17.2.2)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications (IP 71111.17T)

.1 Evaluations of Changes, Tests, and Experiments (26 samples)

a. Inspection Scope

The team reviewed five safety evaluations to evaluate whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59 requirements. In addition, the team evaluated whether Exelon had been required to obtain U.S. Nuclear Regulatory Commission (NRC) approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, TS, and plant drawings to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of twenty one 10 CFR 50.59 screenings for which Exelon had concluded that a safety evaluation was not required. These reviews were performed to assess whether Exelon's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, and procedure changes.

The team reviewed all the safety evaluations and a sampling of the screenings that Exelon had performed and approved during the time period covered by this inspection (i.e., since the last modifications team inspection) that had not previously reviewed by NRC inspectors. The samples selected were based on the safety significance, risk significance, and complexity of the change to the facility.

In addition, the team compared Exelon's administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to evaluate whether the procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the Attachment.

b. Findings

Introduction. The NRC identified a Severity Level IV NCV of TS 6.9.1.f.2 in that Exelon did not obtain NRC approval prior to using a specific analytical method to determine the core operating limits. Specifically, Exelon used an analytical method (TRACG04P) to determine the core operating limits; however, that particular analytical method was not previously reviewed and approved by the NRC prior to Exelon's use.

Description. Under certain conditions, boiling water reactors (BWRs) may be susceptible to coupled neutron/thermal-hydraulic instabilities. The NRC approved several stability long-term solutions (options) that provide protection against potential violations of the fuel cladding integrity safety limit for anticipated oscillations. Oyster Creek implemented stability long-term solution Option II. The Option II solution provides fuel cladding integrity safety limit protection by automatically detecting and suppressing reactor instability. The Option II solution implements modified flow biased average power range monitor (APRM) scram and rod block setpoints, as well as an operational exclusion region, in the low flow region (below approximately 45% core flow) of the power operation curve. In 2002, the NRC reviewed and approved thermal hydraulic transient code TRACG02A for use in determining APRM protection settings. Exelon reactor engineering, with vendor support, evaluates the APRM stability protection settings each operating cycle as part of the standard reload licensing process and includes the APRM stability settings in the COLR. The APRM stability protection methodology is described in the Oyster Creek UFSAR (Section 4.3.2.7.2), incorporated by reference into NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (GESTAR II) which is cited in Oyster Creek TS 6.9.1.f.2.

In August 2012, Exelon performed a 50.59 safety evaluation (OC-2012-E-0001) in conjunction with an engineering change request (ECR 11-00491) to evaluate an upgrade of the thermal hydraulic transient code from TRACG02A to TRACG04P. The team identified that Exelon should not have used the 50.59 evaluation process to approve this upgrade because Oyster Creek TS 6.9.1.f.2 requires that analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. Specifically, Exelon used the TRACG04P analytical method to determine the APRM protection settings which were specified in the COLR; however, that particular analytical method had not been previously reviewed and approved by the NRC prior to Exelon's use at Oyster Creek. The team also noted that Exelon had previously approved similar 50.59 safety evaluations for TRACG02 upgrades in 2006, (OC-2012-E-0001) and in 2010, (OC-2012-0001) without prior NRC review and approval. The team verified that the NRC subsequently reviewed and approved TRACG04P on October 24, 2014, and, as such, the team concluded that there were no concerns regarding continued operation of the unit. Exelon initiated corrective action IR 2482042 to evaluate the condition.

Analysis. The team determined that Exelon did not comply with TS 6.9.1.f.2 requirements in that Exelon used an analytical method to determine the core operating limits without prior NRC approval. The team determined that this was a performance deficiency that was within Exelon's ability to foresee and correct. Because the issue had the potential to affect the NRC's ability to perform its regulatory function, the team evaluated this performance deficiency in accordance with the traditional enforcement process. In accordance with the NRC Enforcement Manual, Part II, 2.1.3, "Enforcement of 10 CFR 50.59 and Related FSAR," the team characterized the violation as Severity Level IV because the underlying analytical method required NRC approval prior to use. Because this violation involved the traditional enforcement process and did not have an underlying technical violation that would be considered more-than-minor within the Reactor Oversight Process (ROP), the team did not assign a cross-cutting aspect to this violation in accordance with IMC 0612, "Power Reactor Inspection Reports," Section 07.03.c.

Enforcement. Technical Specification 6.9.1.f.2 requires that analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. Contrary to this requirement, from October 6, 2006, through October 24, 2014, Exelon used an analytical method (TRACG04P) to determine the core operating limits (the APRM protection settings which were identified in the COLR); however, that particular analytical method was not previously reviewed and approved by the NRC prior to use. Exelon submitted IR 2482042 to evaluate the condition. Since this violation was of minor significance, was not repetitive or willful, and was entered into the Exelon's corrective action program as IR 2482042, this violation is being treated as a non-cited violation, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000219/2015008-01, Use of an Analytical Method to Determine the Core Operating Limits without Prior NRC Approval)**.

.2 Permanent Plant Modifications (13 samples)

.2.1 Replacement of Pressure Regulator on Scram Discharge Volume Valves

a. Inspection Scope

The team reviewed a modification (ECR 13-00358) performed to identify and install replacement control air pressure regulator valves (V-6-960, 961, and 962) for three scram discharge volume (SDV) vent and drain valves (V-15-119, 120, and 121). During a scram, the SDVs partially fill with reactor water displaced from above the control rod hydraulic control unit (HCU). While scrammed, reactor water continues to flow from the HCU to the SDVs until SDV pressure equalizes with reactor vessel pressure. Hence, the SDV becomes an extension of the reactor coolant system pressure boundary. The safety function of the SDV vent and drain valves is to close following a scram to limit the loss of and contain reactor coolant from all 137 control rod HCUs. The safety function of the control air pressure regulator valves is to bleed air pressure off of the SDV vent and drain valves, enabling them to fail closed following a scram.

The team reviewed the modification to verify that design bases, licensing bases, and performance capability of the SDV vent and drain valves and the associated control air pressure regulator valves had not been degraded by the modification. Specifically, the team verified that the seismic qualification and evaluation, valve material, air filtering and flow characteristics, operating temperature and pressure ratings, valve actuation time, and associated preventive maintenance were equivalent or improved. The team interviewed design engineers and reviewed evaluations, vendor specifications, post-modification testing results, and associated maintenance work orders to verify that the air regulator valve replacement was appropriately implemented. Finally, the team walked down the SDV vent and drain valves and associated control air pressure regulator valves to verify the maintenance activities were properly completed.

The documents reviewed are listed in the Attachment.

b. Findings

Introduction. The NRC identified a Green NCV of CFR, Part 50, Appendix B, Criterion XVI, "Corrective Action," for failure to promptly correct a condition adverse to quality. Specifically, corrective actions to restore design conformance of SDV vent and drain valve pressure regulator valves V-6-961 and V-6-962 were not taken at the first opportunity of sufficient duration which was refueling outage 25 (1R25). Additionally, justification of the basis for deferral of corrective action beyond the restart from 1R25 on October 2014, was not documented, reviewed, or approved by site management and/or oversight organizations as required by station procedure OP-AA-108-115, Section 4.5.5. Consequently, two non-conforming pressure regulator valves which perform a safety-related function remained installed following 1R25, without appropriate evaluation and approval.

Description. During refueling outage 1R24 testing (October 22, 2012), SDV vent valve V-5-119 failed to close within 30 seconds as required by TS 4.2.H. Exelon's subsequent causal assessment determined the associated control air regulator valve (V-6-962) did not bleed air pressure from the V-5-119 actuator quickly enough to support prompt SDV vent valve closure (IR1430051, 1438798).

Based on the unexpected test failure, engineers questioned whether air pressure regulator valves V-15-960, 961, and 962 were properly classified as "non-safety-related." ECR 13-00358 was initiated to identify a replacement pressure regulator valve that would properly bleed air pressure following a scram and to determine whether non-safety-related pressure regulator valves V-15-960, 961, and 962 should be reclassified as safety-related. The ECR identified a replacement valve (Fisher Model 67CFA-224/SB) which has a controlled bleed option to ensure air is properly vented following a scram signal. Additionally, engineers determined the pressure regulator valves should be reclassified (upgraded) to a safety-related status. Actions were initiated to update the component quality classification list, procure three safety-related replacement pressure regulators, and to replace the pressure regulators.

Exelon installed and successfully tested the modification (safety-related replacement pressure regulator) for V-6-960 during 1R25 (October 2014). This was the first outage of sufficient duration to upgrade the pressure regulators. The team noted that the other two pressure regulators (V-6-961 and 962) remained non-safety-related, and had not been upgraded. The team determined V-6-961 and 962 were non-conforming to their design and questioned their operability. The basis for operability was documented in ECR 13-00358 and remained valid. However, the timeliness to restore design qualification conformance was not addressed in the ECR. Specifically, procedure OP-AA-108-115, "Operability Determinations," Section 4.5.5 requires that corrective action to establish or re-establish the design basis/qualification of a structure, system, and component (SSC) must be taken at the first opportunity, as determined by safety significance. It is expected that corrective action timeframes longer than the next refueling outage are to be explicitly justified as part of the Operability Evaluation or deficiency tracking documentation used to perform the corrective action. Furthermore, the justification should address the basis for why the repair or replacement activities will not be accomplished prior to restart after a planned outage and the review and approval of the schedule by site management and/or oversight organizations obtained. The team determined that no justification to defer corrective action past 1R25 was developed or reviewed. The team discussed this concern with engineers (IR 2482851). Exelon's initial assessment was that plant modification processes, used by engineers, did not provide clear guidance to implement OP-AA-108-115 when resolving a non-conforming condition.

Analysis. The team determined the failure to restore design qualification of SDV vent and drain valve pressure regulator valves V-6-961 and V-6-962 or properly evaluate and approve the basis for deferral of corrective actions beyond the restart from 1R25, as required by procedure OP-AA-108-115, was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control and barrier performance attributes of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of ensuring the operational capability of the containment barrier to protect the public from radionuclide releases caused by accidents or events. Additionally, the finding was similar to example 5.c in Appendix E of IMC 0612, because the control rod drive system (including the SDV vent and drain isolation valve) was returned to service following plant restart from 1R25 with non-conforming (non-safety-related) pressure regulator valves installed in a safety-related application.

The team evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Screening and Characterization of Findings," dated June 19, 2012, and IMC 0609, Appendix A, "The SDP for Findings At-Power," Exhibit 3, "Barrier Integrity Screening Questions," dated June 19, 2012. The team determined the finding was of very low safety significance because it: did not affect the reactor coolant system (RCS) boundary; did not affect the radiological barrier function of the control room, auxiliary building, or spent fuel pool systems or boundaries; and did not represent an actual open pathway in containment or involve a reduction in the function of hydrogen igniters. The team determined the issue had a cross-cutting aspect in the area of Human Performance, Consistent Process, because the organization did not use a consistent systematic approach to evaluate component operability after Exelon upgraded the classification of three pressure regulator valves from a non-safety to a safety-related status. Specifically, the ECR and Quality Classification processes did not guide station personnel to implement corrective actions to restore design conformance in a timely manner as described in procedure OP-AA-108-115 (IMC 0310, aspect H.13).

Enforcement. Title 10 of the CFR, Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality, such as non-conformances are promptly identified and corrected. In addition, Exelon procedure OP-AA-108-115, Section 4.5.5 requires that corrective action to establish or re-establish the design basis/qualification of a SSC must be taken at the first opportunity, as determined by safety significance. It is expected that corrective action timeframes longer than the next refueling outage are to be explicitly justified as part of the operability evaluation or deficiency tracking documentation used to perform the corrective action. The justification should address the basis for why the repair or replacement activities will not be accomplished prior to restart after a planned outage and the review and approval of the schedule by site management and/or oversight organizations obtained.

Contrary to the above, following component reclassification on May 19, 2014, corrective actions to restore design qualification of SDV vent and drain valve pressure regulator valves V-6-961 and V-6-962 was not taken at the first opportunity of sufficient duration which was 1R25. Additionally, justification of the basis for deferral of corrective actions beyond plant restart from 1R25 was not documented, reviewed, and approved by site management and/or oversight organizations. As a result, two non-conforming pressure regulator valves which perform a safety-related function remained installed following completion of 1R25 on October 11, 2014, without appropriate justification or approval. Immediate corrective actions included licensee determination that V-6-961 and 962 and the associated SDV vent and drain valves (V-15-119 and 121) remained operable, but non-conforming. This violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Exelon entered the issue into their corrective action program as IR 2482851. **(NCV 05000219/2015008-02, Untimely Corrective Action to Restore Design Conformance of Two Scram Discharge Volume Vent and Drain Valve Pressure Regulator Valves)**

.2.2 Emergency Service Water Pump Cable Replacement

a. Inspection Scope

The team reviewed modification 12-00095 that replaced the 4KV power supply cable to the 'C' emergency service water (ESW) pump (P-3-3C) motor. The previously installed supply cable to the 'C' ESW pump motor consisted of two spliced cable sections; original plant installation cable (GE Vulkene) running underground from the 4KV switchgear to an intake area manhole (MH-731-1) and an Anaconda ethylene propylene rubber (EPR) cable running above ground from the manhole to the ESW pump motor.

Based on Oyster Creek operating experience (previous Anaconda cable failures) and cable monitoring trend data, Exelon decided to proactively replace the entire length of cable, from the 4KV switchgear to the ESW pump motor, to reduce the probability of a water intrusion related cable failure.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the ESW and 4KV distribution system had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine if the C ESW pump would function in accordance with design requirements. The team reviewed the associated maintenance work orders to determine if Exelon appropriately implemented the modification. The team performed several walkdowns of the accessible portions of the 4KV power supply cable routing from the 4KV switchgear to the ‘C’ ESW pump motor (including a walkdown during a ‘C’ ESW pump surveillance on April 22) to independently, assess Exelon’s configuration control, motor running amperage, cable and conduit supports, and the material condition of the associated structures, systems, and components (SSCs).

The team reviewed the associated post-maintenance test (PMT) results, system health reports, and recent surveillance test results to verify that the ESW and 4KV distribution systems functioned as designed following the modification. The team also reviewed corrective action IRs to determine if there were reliability or performance issues that may have resulted from the modification.

The documents reviewed are listed in the Attachment.

b. <u>Findings</u>

No findings were identified.

.2.3 <u>Emergency Service Water Pipe Elbow Weld Overlay</u>

a. <u>Inspection Scope</u>

During a raw water piping integrity program inspection in October 2012, Exelon identified wall thinning degradation on an ESW discharge pipe elbow (ES0170). Exelon initiated IR 1430432, evaluated the condition in technical evaluation 1430432-02, and implemented modification 12-00503 to install a weld overlay to repair the elbow. The team reviewed the modification to verify that the design bases, licensing bases, and structural integrity of the ESW piping and supports had not been degraded by the modification. The team interviewed design engineers, and reviewed evaluations, pipe stress calculations, surveillance and non-destructive examination (NDE) results, and associated maintenance work orders to verify that Exelon appropriately implemented the ESW piping weld overlay repair and maintained the ESW piping in accordance with design assumptions. The team also performed several walkdowns of the accessible portions of the ESW system to ensure that the system configuration was in accordance with design instructions and that ESW piping integrity was maintained.

The team also reviewed corrective action IRs and the ESW system health report to determine if there were reliability or performance issues that may have resulted from the modification.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.4 Battery Room Ventilation Exhaust Damper Modification

a. Inspection Scope

The team reviewed modification 12-00197 that permanently gagged open 'C' 125 Vdc battery room ventilation exhaust damper DM-59-9. The damper was originally designed and installed to automatically close when the battery room halon system discharged. However, a halon system was never installed in the 'C' battery room obviating the damper's original intended function. Exelon gagged the damper open to improve battery ventilation system reliability by precluding additional exhaust fan trips due to damper actuator failures. The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the 'C' 125 Vdc battery and its support systems had not been degraded by the modification.

Specifically, the team reviewed calculations, technical evaluations, alarm response procedures, and operating procedures to verify that the permanently gagged open ventilation damper would not adversely impact the function of important-to-safety or safety-related SSCs during normal operation or under design basis conditions. The team reviewed the associated work order instructions and documentation to verify that Exelon had implemented the modification as designed. The team performed several walkdowns of the 'C' 125 Vdc battery room and ventilation system to ensure that Exelon installed the modification in accordance with design instructions and to independently assess Exelon's configuration control and the material condition of SSCs in the area. The team also reviewed corrective action IRs and system health reports to determine if there were reliability or performance issues that may have resulted from the modification. Additionally, the team reviewed the 10 CFR 50.59 screen and engineering evaluation associated with this modification.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.5 Spent Fuel Pool Cooling Heat Exchanger Tube Bundle Replacement

a. Inspection Scope

The team reviewed modification 11-00027 that replaced the tube bundle in the H-18-1A and H-18-1B spent fuel pool cooling (SFPC) heat exchangers (HXs).

Exelon implemented the modification to upgrade the SFPC HX tubes from carbon steel to stainless steel to improve overall SFPC HX performance and system integrity. The pre-existing carbon steel tube bundles had been installed since 1987, and corrosion and erosion resulted in scale buildup and tube wastage necessitating more frequent tube plugging and operational challenges. The team reviewed the modification to verify that the design bases, licensing bases, and structural integrity of the SFPC system and supporting SSCs had not been degraded by the modification. The team interviewed design engineers and reviewed calculations, evaluations, NDE and PMT results, and associated maintenance work orders to verify that the tube bundle replacement activities were appropriately implemented. The team also performed several walkdowns of the accessible portions of the SFPC system to ensure that the system configuration was in accordance with design instructions, the SFPC system integrity was maintained, and that important-to-safety or safety-related SSCs in the vicinity were not adversely impacted. The team also reviewed corrective action IRs and system health reports to determine if there were reliability or performance issues that may have resulted from the modification.

The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.6 Static Battery Charger CB-2 Alternating Current Breaker Replacement

a. Inspection Scope

The team reviewed an equivalent change evaluation (ECR 13-00246) performed to identify and install a replacement breaker for the safety-related solid state static battery charger. The static battery charger may be connected to either the non-safety-related ‘A’ or safety-related ‘B’ direct current (DC) distribution system. The battery charger supplies normal 125 volt DC system loads with the station batteries acting as a standby source of DC power upon the failure of the battery chargers or during high demand transients. One charger is maintained in-service and connected to each DC system to maintain the battery in a fully charged condition. The static charger has two alternating current (AC) input breakers and one DC output breaker. On May 26, 2013, the overload protection device on one of the two static charger AC input breakers (CB-2) failed its periodic surveillance test. The existing breaker design was obsolete and replacement parts for the overload device were not available.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the static charger and station 125 volt DC batteries had not been degraded by the modification. Specifically, the team reviewed attributes such as minimum and maximum operating voltages, breaker response timing, seismic qualification, environmental considerations, breaker failure modes, and protective tripping to verify the new breaker was equivalent or improved when compared to the previous AC input breaker. The team interviewed design engineers and reviewed evaluations, electrical drawings, TSs, purchase specifications, vendor documentation, seismic qualifications, post-modification test results, and associated maintenance work orders to determine whether the breaker replacements were appropriately implemented. Finally, the team walked down portions of the ‘A’ and ‘B’ DC distribution system to verify the breaker replacement was properly completed.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.7 Replacement ‘F’ Switch for Containment Spray Pump P-21-1B Breaker Charging Motor

a. Inspection Scope

The team reviewed an item equivalency evaluation (ECR 14-00508) performed to identify and install a replacement ‘F’ switch to repair a containment spray pump breaker charging motor and also to be used for similar safety-related GE model AK-50 and AK-75 breakers installed at Oyster Creek. On October 28, 2014, during preventive maintenance on the containment spray pump P-21-1B breaker, the ‘F’ switch malfunctioned (WO R2151513). The ‘F’ switch functions as part of the control circuit to start the motor that charges the breaker closing springs. If the ‘F’ switch malfunctions, the closing coil does not charge and manual action would be necessary to charge the closure springs to enable closing the breaker. A qualified replacement switch was not available.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the containment spray pump P-21-1B breaker had not been degraded by the modification. Specifically, the team reviewed attributes such as minimum and maximum operating voltages, breaker response timing, seismic qualification, environmental considerations, breaker failure modes, and protective tripping to verify the replacement ‘F’ switch was equivalent or improved when compared to the previous switch. The team interviewed procurement engineers and reviewed evaluations, electrical drawings, TSs, purchase specifications, vendor documentation, seismic qualifications, post-modification test results, and associated maintenance work orders to verify the safety-related breaker was properly repaired and returned to service. Finally, the team walked down warehouse material storage locations to verify that material packaging, labeling, and storage conditions for the replacement ‘F’ switches were appropriate.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.8 Replacement Emergency Diesel Generator Low Oil Pressure Switch

a. Inspection Scope

The team reviewed an item equivalency evaluation (ECR 08-00388) performed to evaluate and approve a calibrated spare low oil pressure switch for the emergency diesel generator (EDG) to be readily available if an installed switch failed and required replacement. The function of the low oil pressure switch is to provide an engine fault indication and initiate an EDG shutdown on low oil pressure sensed at the main bearing. The low oil pressure switch is bypassed during a fast start condition. The installed oil pressure switch in the 'A' and 'B' EDGs is obsolete and an exact replacement is not available. This ECR was performed to ensure a replacement pressure switch was available as a contingency for an emergent repair. The replacement switch was procured as safety-related.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the EDGs would not be degraded if the modification was installed. Specifically, the team reviewed attributes including physical dimensions, temperature and pressure ratings, materials, switch accuracy, repeatability, and drift, contact ratings, insulation resistance, seismic qualification, and environmental considerations to ensure the replacement was equivalent or improved when compared to the previous oil pressure switch. The team interviewed procurement engineers and reviewed evaluations, electrical drawings, technical specifications, purchase specifications, vendor documentation, and seismic qualifications to verify the safety-related replacement switch met design requirements. Finally, the team walked down the EDG to verify switch configuration and walked down warehouse material storage locations to verify material packaging, labeling, and storage conditions for the replacement low oil pressure switch was appropriate.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.9 Revision to 4160 Volt Class 1E Protective Device Relay Set Points

a. Inspection Scope

The team reviewed calculation C-1302-700-5350-003, "4160 Volt Class 1E Protective Device Relay Setpoints," Revision 6. Exelon implemented this revision to change the 50G ground relay tap settings, incorporate actual emergency service water pump and core spray main pump motor acceleration data, and technically justify using motor nameplate full load amperage instead of the highest measured motor current for this calculation of protective set points.

The team reviewed the calculation to verify that the design bases, licensing bases, and performance capability of the emergency service water pumps and the core spray main pumps had not been degraded by the modification. The team interviewed design engineers and reviewed associated calculations, evaluations, vendor manuals, industry standards, and issue reports to verify that the modification adequately evaluated and established protective device settings.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.10 Emergency Diesel Generator EDG-2, Speed Sensing Switch Replacement

a. Inspection Scope

The team reviewed modification ECR 11-00399 that replaced the EDG-2 speed sensing switch. The replacement was implemented because the old switch, a Synchro-Start Model ESSB-4T speed switch had become obsolete and replacement parts were no longer available. The replacement switch is a Dynalco Controls Model SST-2400A-8 which requires a 24 Vdc power supply, whereas the old switch was powered directly from 125 Vdc provided by the battery for EDG-2. Therefore, this modification also installed a 24 VDC power supply (ESSB-2\PS). The team noted that a similar modification (ECR OC 11-00228) had previously been implemented to replace the EDG-1 speed switch.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the EDG had not been degraded by the modification. The team interviewed design engineers and the EDG system manager and reviewed design drawings, vendor documentation, seismic qualifications, post-modification test results, and associated maintenance work orders to verify Exelon appropriately implemented the EDG-2 speed sensing switch replacement in accordance with the specified design.

Finally, the team walked down accessible portions of the EDG-2 electrical cabinet to verify the switch replacement was properly completed. The team also reviewed corrective action IRs associated with the EDGs to determine if there were reliability or performance issues that may have resulted from the modification.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.11 Isolation Condenser System Relay Replacement

a. Inspection Scope

The team reviewed ECR OC 12-00438 that replaced the isolation condenser system initiation logic relays 6K9, 6K10, 6K11, and 6K12. These relays are used to initiate both the 'A' and 'B' isolation condensers in the event of high reactor vessel pressure (RPV) or low-low RPV water level. The initiation circuit is redundant such that only one relay in each actuation channel is required to initiate both isolation condensers. Therefore, a failure of any one relay will not prevent or cause an initiation of the isolation condenser system. Exelon performed the modification to address a failure of Channel I, relay 6K10 on February 23, 2011, during the performance of a scheduled Core Spray system surveillance testing per procedure 610.3.115, Core Spray System 1, Instrument Channel and Level Bistable Test and Calibrations.

The team reviewed the modification to verify the design basis, licensing basis, and performance capability of the isolation condenser system had not been degraded by the modification. The team interviewed design engineers, the system manager and operators, and reviewed design drawings and calculations to determine if the new relays met design and licensing requirements. Additionally, the team reviewed associated maintenance work orders and post-maintenance test results to determine if Exelon appropriately implemented the modification. The team performed walkdowns of the accessible portions of the isolation condenser and the associated system initiation valves to evaluate the material condition. The team also reviewed corrective action IRs to determine if there were reliability or performance issues that may have resulted from the modification.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.12 Emergency Diesel Generator EDG-1, Cracked Terminal Block Replacement

a. Inspection Scope

The team reviewed commercial grade dedication package CGD-OC-12-005, performed to identify and install a replacement for an EDG-1 pinion failure switch terminal block. The replacement terminal block was dedicated as an exact replacement as verified by applicable critical characteristics and seismic equivalent of the original terminal block qualifications. On May 14, 2012, during scheduled inspection of EDG-1 control cabinet, technicians found the molded plastic separator cracked between two of the terminal board screws (IR 01366327).

The team reviewed the modification to determine if the design basis, licensing basis, and performance capability of the EDG's pinion failure switch had been degraded by the modification. Specifically, the team reviewed attributes such as minimum and maximum operating voltages, seismic qualification, environmental considerations, current and megger test results, and bench testing results. The team interviewed design engineers, the EDG system manager and plant operators, reviewed design drawings and reviewed post-maintenance test results and associated maintenance work orders to determine if Exelon appropriately implemented the modification. The team performed walkdowns of the accessible portions of the EDG control cabinet and inspected the replacement terminal block and associated components. The team also reviewed electrical and continuity tests performed on the replacement board, and reviewed corrective action IRs associated with the EDG to determine if there were reliability or performance issues that may have resulted from the modification. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.13 Revision to Emergency Diesel Generators Fuel Requirements, Usable Volume and Pump Net Positive Suction Head

a. Inspection Scope

The team reviewed calculation C-1302-862-5360-002, "Diesel Generator Fuel Requirements, Usable Volume and Pump Net Positive Suction Head (NPSH)," Revision 6. This calculation provides the basis for minimum inventory in the EDG fuel oil tank T-39-002, determines the amount of usable volume in the tank, and includes a new table that correlates tank level to gallons. In addition, Exelon implemented this revision to change the EDG consumption based on the maximum load derived from calculation C-1302-741-5350-001, "Loading of Emergency Diesel Generators, Unit Substation, and 4.16 KV Buses 1A and 1B," Revision 11, to include new scenarios developed to provide additional fuel oil margin.

The team reviewed the calculation to verify that the design bases, licensing bases, and performance capability of the emergency diesel generators had not been degraded by the modification. The team interviewed design engineers, the EDG system manager, and reviewed associated calculations, evaluations, vendor manuals, industry standards, and issue reports to verify that the modification adequately evaluated and established adequate design margins.

The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of IRs associated with 10 CFR 50.59 and plant modification issues to evaluate whether Exelon was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. In addition, the team reviewed IRs written on issues identified during the inspection to verify Exelon adequately described the problem and incorporated the issue into their corrective action system.

The IRs reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. Jeffrey Dostal, Plant Manager, and other members of the Oyster Creek Generating Station staff at an exit meeting on April 24, 2015. The team returned the proprietary information reviewed during the inspection and verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Exelon (Licensee) Personnel
A. Agarwal, Electrical Design Engineer
T. Cappuccino, Senior Regulatory Specialist
R. Csillag, Manager Design Mechanical Engineering
L. Dormann, System Manager (4160V AC Distribution)
M. Heck, Engineering Response Team, Engineer
T. Hutain, System Manager
R. Larzo, Procurement Engineer
G. Malone, Director, Engineering
M. McKenna, Regulatory Assurance Manager
K. Kalenak, Procurement Engineer
J. Makar, Site Supply Chain Manager
T. Nickerson, Senior Mechanical Design Engineer
P. Procacci, Senior Electrical Design Engineer
H. Ray, Senior Manager, Design Engineering
K. Rossi, Mechanical Design Engineer
T. Ruggiero, Senior Mechanical Staff Engineer
S. Schwartz, System Manager (ESW)
H. Tritt, Electrical Engineering Manager
D. Yatko, Engineering Response Team, Engineer

NRC Personnel
A. Patel, NRC, Senior Resident Inspector
E. Andrews, NRC, Resident Inspector

State of New Jersey
R. Penney

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened and Closed

NCV	05000219/2015008-1	Use of an Analytical Method to Determine the Core Operating Limits Without Prior NRC Approval (Section 1R17.1)
NCV	05000219/2015008-2	Untimely Corrective Actions to Restore Design Conformance of Two SDV Vent & Drain Valves Pressure Regulator Valves (Section 1R17.2.1)

LIST OF DOCUMENTS REVIEWED

Assessments and Nuclear Oversight Reports

AR 01659465-01, Permanent Configuration Changes and 10 CFR 50.59s for Oyster Creek Focused Area Self-Assessment, dated 3/2/15

10 CFR 50.59 Evaluations

OC-2012-E-0001, Application of TRACG04P Version 4.2.69.0 for APRM Stability Protection Settings Determination at OCNGS, Revision 0
OC-2012-E-0002, Shutdown Cooling System Operation/Loss of Shutdown Cooling, Revision 0
OC-2013-E-0001, Install Jumper Across 3 of 4 Trunnion Room Temperature Main Steam Line Isolation Switches (TCCP/ECR 13-00378 (Revision 0))
OC-2013-E-0002, Install Jumper Across Trunnion Room Temperature Main Steam Line Isolation Switches TS-IB0010E and TS-IB-0010F (TCCP/ECR OC 13-00378 (Revision 2)
OC-2014-E-0002, Gag for Damper DM-838-11, Revision 0

10 CFR 50.59 Screenings and/or Applicability Determinations

12-00504, 204 00229 - IEC for V-21, 13, 17, 18 - RF1 and RF2, EQ Upgrade, Revision 0
ECR 13-00495, 204 02155 V-21-3 Molded Case Circuit Breaker Replacement, Revision 0
ECR 15-00064, Abandon Old Cable and Pull New Cable for PSL-A, Revision 0
OC-2012-S-0015, ECR 10-00497, C-1302-741-5350-001, Loading of EDG, Unit Substations, and 4.16 KV Buses 1A and 1B, Revision 0
OC-2012-S-0016, Revise ABN-2 to Comply with Technical Specification 3.3.F.2, dated 3/12/12
OC-2012-S-0031, ECR 12-00135, ESW/SW Temp. Cross Connect in Support of Pipe Replacement, Revision 0
OC-2012-S-0037, Permanently Gag Open DM-59-9 - C Battery Room Damper, dated 4/19/12
OC-2012-S-0080, 311 Fuel Pool Cooling, Revision 107
OC-2012-S-0081, New RB Exhaust Isolation Damper ECR-OC-12-00304 (Dampers to Replace Valves V-28-21 and V-28-22), Revision 0
OC-2012-S-0092, Manual Operation of CH-AOV-42A, to Support ‘A’ Evaporator Operation, Revision 0
OC-2012-S-0124, CRD Return Line with Tooling Marks and Gouges/Tech. Eval 01441954-02, Revision 0
OC-2013-S-0004, Standby Liquid Control Pump and Valve Operability and In-Service Test, dated 1/22/13

OC-2013-S-0014, EDG 1 OV/GND and EDG 2 OV/GND, dated 5/30/13
OC-2013-S-0033, Abnormal Intake Level, dated 6/28/13
OC-2013-S-0036, Procedures ABN-36, Revision 24, Loss of Offsite Power & Station Blackout (Plant Control), Revision 0
OC-2013-S-0041, Procedure RAP-C8H, Drywell Temperature, Revision 0
OC-2013-S-0048, CDBI GL 89-10 MOV Voltage Calculation Revision, Revision 0
OC-2013-S-0064, Classification Downgrade for 64X Relays in Emergency Diesel Generator System, Revision 0
OC-2013-S-0067, Leak Injection Repair V-1-110, Revision 0
OC-2013-S-0069, Emergency Diesel Generator Operation, dated 3/26/15
OC-2014-S-0024, Emergency Core Cooling System Operation, dated 3/20/14
OC-2015-S-0003, Compensatory Measures for Operability Evaluation OC-2014-OE-0005, dated 1/29/15

Modification Packages
ECP 12-00241, EDG-1 Cracked Terminal Board, Revision 0
ECR 08-00388, Contingency Replacement Emergency Diesel Generator Low Oil Pressure Switch, Revision 0
ECR 11-00027, Spent Fuel Pool Cooling Heat Exchanger Tube Bundle Replacement, Revision 2
ECR 11-00399, Replacement of EDG #2 Speed Switch, Revision 0
ECR 12-00095, Emergency Service Water Pump Cable Replacement, Revision 2
ECR 12-00197, Battery Room Ventilation Exhaust Damper Modification, Revision 0
ECR 12-00438, Isolation Condenser System Relay Replacement, Revision 4
ECR 12-00503, Emergency Service Water Pipe Elbow Weld Overlay, Revision 0
ECR 13-00246, Static Battery Charger CB-2 Alternating Current Breaker Replacement, Revision 0
ECR 13-00358, Replacement of Pressure Regulator on SCRAM Discharge Volume Valves, Revision 0
ECR 14-00340, Gag Open DM-838-11, Revision 1
ECR 14-00508, Replacement ‘F’ Switch for Containment Spray Pump P-21-1B Breaker Charging Motor, Revision 0
IEC 12-00062, Item Equivalency Evaluation, Molded Case Circuit Breaker, Revision 0

Calculations
C-1302-214-E310-047, Shutdown Cooling Pump Net Positive Suction Head (NPSH), Revision 1
C-1302-214-E610-034, Shutdown Cooling System Performance, Revision 0
C-1302-251-5320-007, Oyster Creek Spent Fuel Pool Cooling System, dated 4/24/84
C-1302-532-E540-045, OC NSR Piping Analysis Emergency Service Water System Containment Spray HX 1, Revision 2
C-1302-642-E610-009, OC Motor Operated Valves Cycle Frequency, Revision 0
C-1302-700-5350-003, OC – 4160V Class 1E Protective Device Relay Set Points, Revisions 5, 5A, and 6
C-1302-735-E320-050, Oyster Creek Batteries A, B, and C Hydrogen Calculation, Revision 0
C-1302-862-5360-002, Diesel Generator Fuel Requirements, Usable Volume and Pump NPSH, Revision 0 and Revision 6
NES-MS-03.1, Piping Minimum Wall Thickness Calculation, Revision 5

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0632065
0868860
0871081
0879452
0917125
0920109
0932736
1178900
1258788
1263099
1351176
1354567
1366327
1367240
1415454
1430051
1430432
1438798
1443672
1511332
1516599
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2450232
2479114
2479367
2480745*
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2481108
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2481624
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2485790*
2485794*
2486313*
2487170
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2488548*
2489162*
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(*denotes NRC identified during this inspection)

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ASA B31.1, Code for Pressure Piping, 1955
Core Operating Limits Report for Oyster Creek 1 Cycle 25, Revision 10
ME7104, Final Safety Evaluation for General Electric Hitachi Nuclear Energy Americas, LLC Topical Report NEDO-32465, Supplement 1, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications,” dated 10/24/14

NRC Regulatory Guide 1.147, Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1, Revision 17
RA-11-005, Oyster Creek Nuclear Generating Station Letter to NRC, Biennial 10 CFR 50.59 and 10 CFR 72.48 Change Summary Reports – January 1, 2009 through December 31, 2010, dated 6/24/11
RA-13-052, Oyster Creek Nuclear Generating Station Letter to NRC, Biennial 10 CFR 50.59 and 10 CFR 72.48 Change Summary Reports – January 1, 2011 through December 31, 2012, dated 5/24/13
SDBD-OC-532, Design Basis Document for Emergency Service Water System, Revision 4
TDR 829, Pipe Integrity Inspection Program, Revision 11
Topical Report 140, Emergency Service Water and Service Water System Piping Plan, Revision 0

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082-1, 15000 Gallon Diesel Oil Storage Tank, Revision 6
082-2, 15000 Gallon Diesel Oil Storage Tank, Revision 5
3E-532-A2-1000 Sh. 1, ISI Configuration Drawing Emergency Service Water, Revision 5
3E-532-A3-1000 Sh. 1, Pipe Integrity Inspection Program Emergency Service Water Piping, Revision 1
532-WM-069, ESW Isometric Weld Map, Revision 0
3E-611-17-005, Sht. 1, Electrical Elementary Diagram, Control Panel 1F/2F-Annun. C, Revision 19
3E-839-17-1000, C Battery Room H&V Panel Electrical Elementary Diagram, Revision 2
3E-839-18-1000, C Battery Room H&V Panel Electrical Connection Diagram, Revision 3
BR 2002, Sht. 1, Main Steam System, Revision 62
BR 2005 Sh. 4, Emergency Service Water System Flow Diagram, Revision 86
BR 2009 Sh. 1, Turbine Building H & V Flow Diagram, Revision 49
BR 2013, Sht. 6, Instrument (Control) Air System, Revision 86
BR 2150, Containment Spray & Emergency Service Water Piping Plans - Reactor BLDG, Revision 11
BR 2150 Sh. 1, Containment Spray & Emergency Service Water Piping Plans, Section and Details - Reactor BLDG, Revision 8
2179 Sh. 1, 125 Volt DC SYS Modification Battery C Room Heating & Vent, Revision 2
BR 3001C, 4160V Emergency Switchgear Bus 1C & 1D, Revision 1
BR 3004, Sht. 3, Reactor Building 460 Volt MCC 1B21 and 1B21A, Revision 21
BR 3028, Sht. 1, 125 Volt DC Distribution Center A&B MCC DC-1, Revision 21
BR 3029, Sht. 1, Emergency Condenser System Electrical Elementary Diagram, Revision 24
BR 3029, Sht. 2A, Emergency Condenser System Electrical Elementary Diagram, Revision 25
BR 3104 Sh. 1, Turbine Building Conduit & Tray Plan Basement, Revision 22
BR 3179 Sh. 1, Miscellaneous Outdoor Facilities, Revision 9
BR M0012 Sh. 1, Post Accident Sampling Flow Diagram, Revision 42
BR E1102, Emergency Condenser System Elementary Diagram, Revision 15
GE 148F262, Sht. 1, Emergency Condenser Flow Diagram, Revision 54
DJP 3E-532-A3-1000 Sh. 1, Pipe Integrity Inspection Program Emergency Service Water Piping, Revision 1

EM 8393039, Sht. 3, Emergency Diesel Gen #1 Panel Arrangement, Revision 13
EM 8393039, Sht. 4, Emergency Diesel Generator Electrical Wiring Diagram, Revision 8
EM 8393039, Sht. 5, Emergency Diesel Generator Electrical Wiring Diagram, Revision 3
EM 8393039, Sht. 8, Emergency Diesel Gen #1 Panel Arrangement, Revision 4
GE 237E566, Sht. 2, Reactor Protection System Channel 1, Revision 46
GE 237E566, Sht. 3, Reactor Protection System Channel 1, Revision 15
GE 237E566, Sht. 6, Reactor Protection System Channel 2, Revision 40
GE 237E566, Sht. 7, Reactor Protection System Channel 2, Revision 12
GE 197E871, Scram Discharge Volume System and Control Rod Drive Hydraulic System, Revision 30
GE 237E756, Spent Fuel Pool Cooling Flow Diagram, Revision 57
JC 19481, North C.S.H.X. Loop Seal Piping Revisions, Revision 2
JH 8060-0088 Sh. 2, Battery Room C Control Diagram, Revision 2
N06893, Spent Fuel Pool Cooling Replacement Bundle, Revision 3

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A2088162-02, Take Hydramotor Project to PHC and DSC Evaluation, dated 4/30/04
A2188860-01, Change PM from Outage to Condition Directed for HX-16-2A Technical Evaluation, dated 10/2/09
A2246019-11, Evaluate ESW RTS with Megger Reading below Procedure Required Technical Evaluation, dated 10/3/12
A2255281-04, Perform Acceptance of Test Results for ECR 11-00027, dated 9/2/14
A2255281-15, Perform Owners Acceptance of API Basco Calculation Technical Evaluation, dated 3/18/11
AR 01660513-02, OC-2014-S-0003 Inadequate 50.59 Screening Work Group Evaluation, dated 5/27/14
BJR-10-6018, Effect of Stainless Steel Tubes on Heat Transfer of Spent Fuel Pool Cooling Heat Exchanger (API Basco Analysis), Revision 4
BRC Study No. 3731-013, Study for Compliance of Battery Rooms' Ventilation Systems with Appendix R requirements, Revision 3
EC 399975, Oyster Creek 2014 Racklife Update Engineering Evaluation, dated 12/4/14
EC 400120, Oyster Creek Boraflex Degradation Limits, dated 12/5/14
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OC-2006-E-0003, Application of TRACG04 for Stability Analysis, Revision 0
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A2190293	C2027733	R2174408
A2330917	C2028627	R2190529
C2026647	C2032052	R2201046
C2028359	R2104645	R2231612
C2028171	R2113877	R2242383
C2030393	R2119364	R2226047
C2030502	R2119384	R2239495
C2033863	R2132097	R2243297
C2016803	R2151513	R2248774
C2016804	R2092799	

<u>Hydrostatic Tests, In-Service Tests, and Non-Destruction Examinations</u>
1R25 Buried Pipe Raw Water Program Inspection/Replacement Scope and Results Summary
BOP-UT-2014-015, UT Raw Water Examination (ES0170), performed 9/16/14
ES0170, Ultrasonic Thickness Measurement, performed 10/22/12
OC-BOP-158, Magnetic Particle Examination, performed 11/11/12
OC-BOP-159, Ultrasonic Thickness Measurement, performed 11/11/12

<u>Miscellaneous</u>
ASME OMb Code-2006, Addenda to ASME OM Code-2004, Code for Operation and Maintenance of Nuclear Power Plants
ASME Code Case N-661-2, Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service Section XI, Division 1, dated 3/22/07
CGD-OC-12-005, Commercial Grade Dedication, Part of EDG Pinion Failure Switch, Revision 0
CGD-OC-90-0075, Commercial Grade Dedication, Revision 9
CGIPS01, EPRI JUTG Commercial Item Evaluation for Pressure/Differential Pressure Switch, Revision 0
CGISW04, EPRI JUTG Commercial Item Evaluation for Control and Transfer Switches, Revision 0
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Drywell Temperature Adverse Condition Monitoring Plan, Revision 0
EACE 00932736, Apparent Cause Report-Failure Analysis of Agastat Relay 6K12, Revision 4
ES-017, Engineering Standard Identification of Oyster Creek Plant Systems, Revision 29
ESOMS Reactor Building Rounds, dated 2/21/15 – 4/21/15
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VM-OC-2948, EDG Speed Switch and Power Supply, Revision 0
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LIST OF ACRONYMS

ADAMS	Agency Wide Documents Access and Management System
AC	Alternating Current
AOV	Air Operated Valve
APRM	Average Power Range Monitor
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
DC	Direct Current
DRS	Division of Reactor Safety
ECR	Engineering Change Request
EDG	Emergency Diesel Generator
EPR	Ethylene Propylene Rubber
ESW	Emergency Service Water
Exelon	Exelon Nuclear
GESTAR	General Electric Standard Application for Reactor Fuel
HCU	Hydraulic Control Unit
HVAC	Heating, Ventilation and Air Conditioning
HX	Heat Exchanger
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
OE	Operating Experience
OEM	Original Equipment Manufacturer
OP	Operating Procedure
OC	Oyster Creek Nuclear Generating Station
PMT	Post Maintenance Test
RCS	Reactor Coolant System
RPV	Reactor Vessel Pressure
1R25	Refueling Outage 25
SDP	Significance Determination Process
SDV	Scram Discharge Volume
SFPC	Spent Fuel Pool Cooling
SSC	Structure, System and Component
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Vdc	Volts direct current