AmerGen[SM]

AmerGen Energy Company, LLC
Oyster Creek
US Route 9 South
P.O. Box 388
Forked River, NJ 08731-0388

An Exelon/British Energy Company

10 CFR 50.59

July 11, 2003
2130-03-20188

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Oyster Creek Generating Station
Facility Operating License No. DPR-16
NRC Docket No. 50-219
10 CFR 50.59 Summary Report

Pursuant to the provisions of 10CFR50.59(d)(2), enclosed is a brief description of changes made to the Oyster Creek Nuclear Generating Station as authorized under 10 CFR 50.59(c)(1) for the period July 2000 through December 2002.

In each case, the corresponding safety evaluation concluded that changes did not involve an unreviewed safety question.

If any additional information is needed, please contact Mr. Richard Milos, of my staff, at 609.971.4973.

Very truly yours,

Ernest J Harkness P.E., Vice President
Oyster Creek Generating Station

EJH/RAM
Enclosure

cc: Regional Administrator, USNRC Region I
USNRC Senior Project Manager, Oyster Creek
USNRC Senior Resident Inspector, Oyster Creek
File No. 03070

IE47

Enclosure
10 CFR 50.59 Summary Report
July 1, 2000 – December 31, 2002

SE-000020-003, Rev. 0
Revision to Oyster Creek EQ Master List

This evaluation technically justified the prior addition of pressure transmitter, PT-IP0012, to the Environmental Qualification Master List (EQML), and re-evaluated the operability requirement of pressure transmitter, PT-IP0012, and various Containment Isolation Valve (CIV) limit switches identified in revision 11 of the EQML. The operability requirement of these components was evaluated against the Reg. Guide 1.97 licensing basis commitments, as presented in the UFSAR. The conclusions of the safety evaluation did not result in a required change to any design basis analysis that credits the function of torus pressure transmitter, PT-IP0012, and the CIV limit switches in the harsh environmental conditions during a design basis LOCA event. There was no effect on nuclear safety or safe plant operation.

This activity did not identify an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000020-004, Rev. 0
Historical EQ Master List Deletions

This activity re-evaluated and documented the justification for the historical deletion of various electrical components from the scope of the Oyster Creek Environmental Qualification (EQ) Program. The historical deletions of components from the Oyster Creek EQ Master List (EQML) were previously technically evaluated but the evaluations were not documented as safety evaluations. This safety evaluation concluded that the historic exemption of the evaluated electrical components from the requirements of 10CFR50.49 was both appropriate and justifiable at the time they were deleted from the Oyster Creek EQML. There was no effect on nuclear safety or safe plant operation.

This activity did not identify an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000153-033, Rev. 0
Replacement of the Hydrolaze Connection in the Reactor Building Penetration

This modification replaced the old hydrolaze piping and fittings with components rated at higher pressure. The ½-inch, schedule 80 pipe inside the 4-inch penetration in the south Reactor Building wall at the 29-ft, 10-inch elevation was replaced with ½-inch schedule 160 pipe and associated fittings capable of 10,000 psi pressure. Secondary containment was maintained during installation of the modification. There was no effect on nuclear safety or safe plant operation.

This activity did not identify an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000211-039, Rev. 0
18R "A" Isolation Condenser Hydrogen Burn Assessment

An evaluation was made of the apparent hydrogen detonation in the "A" Isolation Condenser piping in the Reactor Building on 10/15/2000, during the 18R tube bundle replacement activities. Nuclear safety or safe plant operation was not affected because the piping is adequate to withstand the dynamic pressure pulses as shown by analysis. Furthermore, subsequent walkdowns of the piping showed no deformation or distress of the piping and associated supports.

This evaluation did not identify an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000212-064, Rev. 0
Technical Specification Change Request 275

Technical Specification Change Request (TSCR) No. 275 proposed changes to sections 3.4.A.7.c and 3.4.A.8.c of Oyster Creek's Technical Specifications. The proposed change involved replacement of the requirement to demonstrate operability of core spray pumps and system components by repetitive testing during refueling/shutdown modes of operation. Periodic verifications will be used in place of the operability testing. These verifications will be completed by performing an administrative check/review of the appropriate plant records.

This evaluation determined that the proposed changes were acceptable without creating an unreviewed safety question or an adverse environmental impact. NRC approval was required prior to implementation. This TSCR was approved by Amendment 231 to the Operating License.

SE-000212-069, Rev. 0
Contingency to Seal Weld Pump Casing Vents on Core Spray Pumps

Seal welding of pump casing vents on Core Spray pumps was evaluated as a contingency if the vent joint exhibited leakage. This evaluation indicated that seal welding the vent connection on the Core Spray and Core Spray Booster Pumps has no effect on the performance of any component or system. There are no changes made to or required of system operation or its function. There are no changes required to design basis documents nor are there any changes to system performance or seismic capability. There was no effect on nuclear safety or safe plant operation.

This evaluation did not identify an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000212-070, Rev. 1
Removal of Core Spray Gage Savers

Core Spray Main Pump Suction Pressure Gages were equipped with gage savers for overpressure protection. The setpoint repeatability of the gage savers was very poor, causing the corresponding pressure gage readings to be unreliable. This modification restored pressure gage reliability by removing the gage savers. These pressure gages are normally isolated and used only during quarterly pump In-Service Testing. Removal of the gage savers was approved by a previous revision. This revision (Rev. 1) to SE-000212-070 added ½-inch Whitey valves and removed the original test tap isolation valves. . There was no effect on nuclear safety or safe plant operation.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000212-071, Rev. 0
Change in Position of PI-25 Root Isolation Valves

Core Spray Main Pump Suction Pressure Gages (PI-25A, B, C, & D) were isolated by root isolation valves located near the pump housings. The valves were difficult to reach and to shut tight. The gages also had local instrument isolation valves, which were normally open. This evaluation determined that it was acceptable to leave the root isolation valves open and to close the local instrument isolation valves in order to isolate the suction gages. The Core Spray System function was not affected by the swap of these valve positions. There was no effect on nuclear safety or safe plant operation.

This evaluation did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000215-043, Rev. 1
Reactor Water Cleanup Valve, V-16-63, Modifications and Repairs

This 18R modification/repair of Reactor Water CleanUp (RWCU) System Valve, V-16-63 included several items:
(1) Removal of the four seal injection valves installed in 15R and repair of the injection ports with pipe plugs,
(2) Removal of the valve bonnet leak containment enclosure that was also installed in 15R,
(3) Replacement of the metallic pressure seal gasket with a new graphitic pressure seal gasket,
(4) Installation of a valve disc/stem retention fixture to facilitate pressure seal repairs. This fixture provided a positive retention force on the valve disc against the seat to maintain the valve disc in the closed position during repair activities,
(5) Installation of a supplemental spacer ring above the graphitic pressure seal gasket,
(6) Installation of a bonnet spacer ring to ensure that the segmented ring is secured in the groove of the valve body, and
(7) Installation of ½-inch high strength stainless steel threaded rod, nuts and washers in lieu of cap screws.

The last three items above were addressed in Rev. 1 of the evaluation, SE-000215-043. The previous revision reviewed the first four items as well as the overall modification/repair to V-16-63. The repair/modification restored the pressure boundary of the RWCU System valve, V-16-63. This modification did not alter the performance, operation, function, interfaces, and design basis of either V-16-63 or the RWCU System. There was no effect on nuclear safety or safe plant operation.

This repair/modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000223-013, Rev. 0
Specification for Recirc Pump Cover-to-Casing Seal Weld

This evaluation applied to generically seal welding any recirculation pump cover-to-casing joint per general Specification, SP-1302-42-013, Rev. 1, as revised by ECD C-314623. The specification also includes changing the casing bolts to studs and nuts to improve joint pre-load and enhance maintenance. These changes will not affect the normal operation of the recirculation system nor impact the safe shutdown of the reactor. There was no effect on nuclear safety or safe plant operation.

This specification/modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000223-014, Rev. 0
Temporary Modification to Support 18R Recirc Pump Maintenance Activities

This temporary modification supported 18R maintenance activities on Recirc Pumps NG01-A & C. The temp mod installed a level column attached to a temporary flange that mated to the 2-inch Recirc Pump suction Chem Decon flange. The purpose of the temp mod was to assure that the pump bowls remain filled with water for shielding purposes during repair activities. The Recirc pump is isolated from its respective loop during installation of the temp mod and subsequent maintenance activities. The temp mod has no impact on nuclear safety or safe plant operations.

This temp mod did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000225-034, Rev. 2
Freeze Seal on Valve, V-305-0106

This evaluation applied to the installation of a freeze seal on the Hydraulic Control Unit (HCU) riser above the 106 valve. The only function of the freeze seal barrier was to provide reliable isolation of the Reactor Coolant System pressure boundary while maintenance activities were in progress. No active safety function was affected. These freeze seal barriers would not prevent the fulfillment on any safety-related function. This type of freeze seal application was initially approved by previous revisions of the safety evaluation. This revision (Rev. 2) of SE-000225-034 extends use of the freeze seal technique to any of the 137 HCU's. There was no effect on nuclear safety or safe plant operation.

This maintenance activity does not involve an unreviewed safety question and no changes to the Technical Specifications are required.

SE-000225-044, Rev. 0
Removal of a Stuck Control Rod Drive

This evaluation reviewed the maintenance activity of using a Control Rod Guide Tool Seal while cutting out a Control Rod Drive that cannot be withdrawn. This maintenance evolution was essentially the same as a normal Control Rod Drive exchange, except that a different sealing method was used in the guide tube. This seal was designed and built for this purpose by General Electric. The maintenance activity did not affect nuclear safety or safe plant operation.

This maintenance activity did not involve an unreviewed safety question and no changes to the Technical Specifications are required.

SE-000225-048, Rev. 0
Defeat HCU Accumulator Trouble Alarm H-8-c for 4S1 Maintenance

Maintenance was performed to replace the rod control switch, 4S1. The activity required removal of control power fuses, which also defeated the HCU accumulator trouble alarm, H-8-c. The alarm monitors accumulator high level and low pressure. Justification was provided for defeating the alarm for a 4-hour time period based on reactor conditions and monitoring local gages. The control rod scram function was not made inoperable by this maintenance activity. Defeating the alarm did not affect nuclear safety or safe plant operations.

This maintenance activity did not involve an unreviewed safety question and no changes to the Technical Specifications are required.

SE-000231-029, Rev. 1
Operation of the AOG System with Charcoal Vault Greater than 50 F

This evaluation supported a temporary procedure change allowing the Augmented Off Gas (AOG) system to return to service with the Charcoal Vault coolers not able to maintain vault temperature less that 50 F. The required justification was provided by the previous revision of the safety evaluation. This revision (Rev. 1) to SE-000231-029 corrected an Environmental Qualification (EQ) statement to say that an EQIS form was not required because AOG equipment was not listed in the EQ Master List. There was no effect on nuclear safety or safe plant operation.

This document correction did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000232-061, Rev. 1
HP-P-1B Drain Valves As-Found ECD

This evaluation applied to a correction of the P&ID for pumps used in processing high purity liquid waste. The pump casing drain valves were shown incorrectly. The affected system has no nuclear safety-related function. The Engineering Change Document (ECD) was evaluated by the previous revision to SE-000232-061. This revision (Rev. 1) added the Section Manager's signature to the safety evaluation. There was no effect on nuclear safety or safe plant operation.

This document correction did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000232-064, Rev. 0
Updated Evaluation of EMC Water Eater Evaporation Usage

Type EMC Water Eater Evaporators have been previously approved for use in order to reduce the inventory of low activity liquid radwaste. The drum evaporators perform no nuclear safety function. This safety evaluation reviewed the FSAR change to add information on the drum evaporators to reflect their use in the plant. Nuclear safety or safe plant operations were not affected.

This FSAR change did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000232-070, Rev. 1
Precoat System Valving

Engineering Change Document C312182 identified several valves and instruments associated with the Precoat Tank in the Old Radwaste Bldg. that were not shown on the respective flow diagram. The omissions have been corrected and reviewed under a previous revision of this evaluation. This revision (Rev. 1) to SE-000232-070 added the Section Manager's signature to the safety evaluation. There was no effect on nuclear safety or safe plant operation.

This document correction did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000233-015, Rev. 2
Alternate Spent Resin Tank Drainage Path

Resin fines from the drain down of spent resin tanks, SL-T-2A & B, entered the liquid waste processing system. These fines contained organic sulfur compounds that were not effectively removed by the normal processing system. This modification established an alternate process path, using an Ultrafiltration skid, for removal of the organic sulfates prior to treatment by the conventional processing system. The modification was evaluated by the previous revision to SE-000233-015 and there were no effects on nuclear safety or safe plant operations. This revision (Rev. 2) added the Section Manager's signature to the safety evaluation.

This document revision did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000243-023, Rev. 0
Drywell Permanent Scaffolding Elevation 23'-6"

This modification provided three permanent scaffold frames in the drywell at elevation 23'-6" thereby reducing radiation doses during outages. The scaffold frames have been designed to withstand dead, temperature, and seismic loading conditions and are positively anchored. They do not interfere with equipment important to plant safety. There is no impact on nuclear safety or safe plant operation.

This document revision did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000243-025, Rev. 0
Torus Water Storage Tank Heaters

This was a temporary modification to power only three of the six heaters for the Torus Water Storage Tank (TWST) because one of the two parallel feeder cables from the 1E1-480V Unit Substation had failed on ground fault. The TWST was only 5% full and the three operable heaters were more than adequate for freeze protection. There are no safety-related loads on 1E1. There was no impact on nuclear safety or safe plant operation.

This temporary modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000254-003, Rev. 2
Criticality Analysis for the Oyster Creek Fuel Racks with Boraflex Degradation

The previous revisions to this evaluation addressed Oyster Creek's re-evaluation of the criticality analysis for the high density fuel racks in response to Generic Letter 96-04 concluded that there was no adverse impact on nuclear safety with the Boraflex degradation, provided specified limits, monitored by surveillances, were maintained. Limits were documented in SE-000254-003. The latter also determined that the Boraflex degradation was an Unreviewed Safety Question and a submittal for License Amendment was made to the NRC. The purpose of Rev. 2 to safety evaluation, SE-000254-003, was to correct certain documentation errors in previous versions of the evaluation. The errors had no effect on the previous conclusions reached. There was no impact on nuclear safety or safe plant operation.

This document revision did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000262-002, Rev. 1
Include Multiple-LPRM Handling Strongback in Heavy Load Proc. 2400-SMM-3891.04

The Multiple-LPRM Handling Strongback has been evaluated as an acceptable heavy load that can be safely rigged and transported over fuel in the RPV. This heavy load was added to Exhibit 1 of Procedure 2400-SMM-3891.04. The use of the LPRM Handling Strongback as part of the Spent Fuel Pool handling system on the refuel floor has been determined to be a safe evolution due to the extensive safety factors designed into the device. Use of this device does not affect nuclear safety or safe plant operation.

This activity does not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000301-003, Rev. 0
Refurbishment of LP Steam Turbine Rotors

This evaluation applied to the refurbishment work on the spare Low Pressure (LP) turbine rotor and replacement of the old coupling bolts with hydraulic coupling bolts. The installation of new turbine blades and improvements to the wheel dovetail design improved the reliability of the turbine rotor by reducing wheel stresses. The hydraulic coupling bolts were not a missile concern due to their size and location. There were no safety-related systems or equipment altered by this activity. This activity has no affect on nuclear safety or safe plant operation

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000411-030, Rev. 0
Minimum Equivalent Engagement on MSIV Studs

The Main Steam Isolation Valve (MSIV) cover-to-body studs were removed to support internal valve maintenance. Some of the threaded holes were damaged during removal of studs. This evaluation determined that it was acceptable to reduce the effective thread engagement of the cover-to-body studs on the MSIV's. This change will not reduce the structural integrity, reliability, or sealing ability of this flanged joint. Nuclear safety or safe plant operation was not affected.

This determination did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000411-031, Rev. 0
Tech Spec Change Request 286

The proposed Tech Spec Change Request (TSCR) No. 286 will change the frequency of the full stroke testing of the Main Steam Isolation Valves, performed to meet ASME Section XI requirements and the IST program requirements, from quarterly to a cold shutdown basis. Part-stroke testing will be performed on a quarterly basis for protective system testing and will help assure valve operability. Elimination of the downpower required to perform a full stroke test will reduce the transients and challenges to the operators. Nuclear safety or safe plant operation was not affected.

This evaluation determined that the proposed changes were acceptable without creating an unreviewed safety question or an adverse environmental impact. NRC approval was required prior to implementation. This TSCR was approved by Amendment 221 to the Operating License.

SE-000411-032, Rev. 1
Replacement of MS Header Drain Trap Y-1-38

The previous revision of this evaluation applied to the replacement of the steam trap, Y-1-38, on the 30-inch Main Steam Header Drain. The old steam trap was replaced because it was oversized (sized for startup loads and not normal operation) and had become a high maintenance item. This revision (Rev. 1) to safety evaluation, SE-000411-032, corrected an omission on the SEDR Form, by answering question 8 "No." The conclusions of the safety evaluation were not affected. Nuclear safety or safe plant operation was not affected.

This document correction did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000421-024, Rev. 0
Hotwell Conductivity Monitoring Reliability Improvement

This modification replaced cables for hotwell conductivity monitors, CE/CM-0001 through 0006, in order to improve their reliability. These cables had become frayed and brittle. The conductivity monitors are required to detect condenser tube leaks since Oyster Creek uses salt water for cooling. The hotwell conductivity monitors do not perform safety related functions nor do they interfere with safety related systems. Nuclear safety or safe plant operation was not affected.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000422-028, Rev. 0
Removing "B" Feedpump Min. Recirc Breakdown Orifice RO-2-463 from Service

This modification removed the "B" Feedpump Minimum Recirculation Line Breakdown Orifice, downstream of V-2-231, located in the Condenser Bay. The "A" and "C" Feedpump Minimum Recirculation Line Breakdown Orifices were previously removed by a different modification. The orifices were no longer required since previously upgraded minimum flow control valves now perform the pressure breakdown function of the orifices. The feedpump minimum recirc lines are not required to safety shutdown the plant or to maintain the plant in the shutdown condition. Nuclear safety or safe plant operation was not affected.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000423-013, Rev. 0
Condensate Demineralizer Tank Over Pressure Issue

The Condensate Demineralizer Tanks have a nameplate rating of 310 psig @ 650 F, built to ASME Section VIII Division I 1965. There were occasions when the system was operated up to 370 psig, but the system temperature was never allowed to exceed 140 F in order to protect the resins. This evaluation provided an interpretation of ASME Section VIII code that allowed the Condensate Demineralizer System to occasionally operate above the nameplate pressure rating. Nuclear safety or safe plant operation was not affected.

This evaluation did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000424-023, Rev. 1
WC-P-4A/B Seal Water Check Valve Installation

The previous revision to this evaluation was for a modification, which added a spring loaded check valve to the seal water flush of the chevron packing for the Chemical Waste/ Floor Drain System body feed pumps. The modification reduced the seal line pressure to the pump manufacturer's recommendations. This revision (Rev. 1) to SE-000424-023 only added the Section Manager's signature to the safety evaluation. Nuclear safety or safe plant operation was not affected.

This document correction did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000431-017, Rev. 1
Replacement of SJAE Intercondenser and Aftercondenser Drain Traps

The previous revision of this evaluation reviewed the modification which replaced the three Steam Jet Air Ejector (SJAE) Intercondenser Traps and the three SJAE Aftercondenser Traps. The old traps were inverted bucket type and were oversized for this application. The new traps are float and thermostatic traps and more suitable for this service. This revision (Rev. 1) to evaluation, SE-000431-017, corrected an omission on the SEDR Form, by answering question 8 "No." The conclusions of the previous revision of the safety evaluation were not affected. Nuclear safety or safe plant operation was not affected.

This document correction did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000445-003, Rev. 0
Replacement of Heating Steam Valve, V-13-98

This modification replaced valve, V-13-98, that supplies steam to the heating coil for the Reactor Building. The original valve had a broken yoke and an exact replacement was not available, so a reasonable compromise valve was selected as the replacement. The new valve will provide the required steam to the heating coil as required by the system design. Nuclear safety and safe plant operation is not affected.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000531-036, Rev. 0
Secondary Service Water Operability with Non Code Leak Repair

A temporary sealing device was installed to stop the leak on the discharge of Service Water Pump 1-1. This temporary non-code repair was evaluated in accordance with Generic Letter 90-05. The leak repair was in effect until the 19R outage when a permanent repair was made. The temporary repair did not affect nuclear safety or safe plant operations.

This temporary repair did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000531-038, Rev. 0
Temporary Modification to Seal Service Water Leak Inside Reactor Building

This temporary modification sealed a small leak in the Service Water System piping, upstream of the Reactor Bldg. Closed Cooling Water Heat Exchanger in the Reactor Bldg.

The purpose of the temporary seal was to eliminate or reduce salt water intrusion into the Reactor Bldg. A permanent repair was achieved during the 19R outage. The activity did not affect nuclear safety or safe plant operation.

This temporary modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000531-039, Rev. 0
Engineering Change Document C307477

Service Water Pump casing vent valves, V-3-299 & 300, were made locked-open valves by this change. This will improve the reliability of the pumps because an inadvertent closure of these valves could prevent the associated pump from starting when required. The Service Water System does not perform a safety-related function. The change does not affect nuclear safety or safe plant operation.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000532-029, Rev. 0
Continued Operation with one ESW Pump Out-of-Service

The purpose of this contingency evaluation was to determine if continued operation of Oyster Creek with one Emergency Service Water pump out-of-service for an additional 14 days longer than Technical Specification maximum specified 15 days could be justified. The evaluation determined that the extension would not affect nuclear safety or safe plant operation. If the extension were actually needed, however, a request to the NRC to approve a Notice of Enforcement Discretion for the additional 14 days would be required. Ultimately, the 14-day extension was not required and no Notice of Enforcement Discretion was requested.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000532-030, Rev. 3
Continued Operation with Old Style ESW Pumps re the Failure of Pumps 52C

Previous revisions of this safety evaluation determined that both Emergency Service Water (ESW) subsystems are fully operable with a failed ESW pump. Only one pump per each of two subsystems was required to ensure full functionality of the ESW system. Nuclear safety or safe plant operation were not affected. This revision (Rev. 3) of evaluation, SE-000532-030, specifically addressed ESW pump 52C.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000532-031, Rev. 1
Replace Leaking Emergency Service Water Keep-Fill Piping

This modification replaced the cross fitting in the Emergency Service Water Keep-Fill piping with two tees. The cross was not replaced-in-kind because of availability and time constraints. All design criteria and functions were the same but the drawing referenced in the FSAR had to be revised. Nuclear safety or safe plant operations was not affected. Subsequent revision to the evaluation, SE-000532-031, made minor corrections to the evaluations forms.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000542-023, Rev. 0
Lock Closed V-5-777 in TBCCW System

Turbine Bldg. Closed Cooling Water (TBCCW) drain valve, V-5-777, was made a "locked-closed" valve by this change. This will improve the reliability of the TBCCW system by preventing an inadvertent opening of the drain valve. The TBCCW System does not perform a safety-related function. The change does not affect nuclear safety or safe plant operation.

This change did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000600-002, Rev. 0
Specification for I & C Equipment Mounting and Tubing Installation

This evaluation reviewed SP-9000-44-001, Rev. 2, Specification for Instrument and Control (I & C)Equipment Mounting and Tubing Installation, as amended by ECD C400267 regarding its applicability to Oyster Creek. This Spec was a generic installation specification that provided the design guidelines and criteria for installation of tubing and instruments to be used at Oyster Creek. This activity did not modify or change the plant in any way. There was no effect on plant safety or safe plant operations.

This change did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000624-006, Rev. 0
Installation of Jacking Device on Vacuum Bellows

This temporary modification installed a jacking or blocking device to override a defective vacuum bellows in the Turbine Control System #1. There was sufficient redundancy so that one of the six installed bellows can be mechanically bypassed. Plant safety is not jeopardized by this temporary modification. The modification remained in place for a short time during maintenance/troubleshooting activities. There was no effect on nuclear safety or safe plant operation.

This temporary modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000641-025, Rev. 0
UFSAR Changes Resulting from a One-Time UFSAR Review of RPS

This evaluation reviewed six changes to the Reactor Protection System (RPS) description in the UFSAR. Three of the changes were editorial in nature. One change revised Table 7.2-1 to correct nomenclature of two Main Steam Isolation Valve closure relays to be consistent with drawing GE 237E566. One change revised Table 7.3-1 to correct nomenclature of Main Steam Line high flow sensors to be consistent with drawing GE 237E566. The sixth change revised the calculated impact pressure caused by water steam slugs impacting the Main Steam System flow restrictors from 1510 psig to 1510 psia.

The current design bases and operating requirements for the RPS have not been changed by the FSAR update. The changes to the tables were revised to reflect current design and operating characteristics of the RPS and Main Steam isolation. The change to the impact pressure units was consistent with the higher predicted impact pressure contained in FDSAR Amendment 33. Nuclear safety or safe plant operations were not affected.

These UFSAR changes did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000641-120, Rev. 0
Bypass of RSCS-011 & 012 at Low Power for Troubleshooting Turbine Controls

This temporary modification electrically jumpered out two of the four Condenser Low Vacuum Scram Switches, RSCS-011 & 012. These two switches are both part of the Vacuum Trip System #1 on the Turbine Hydraulic Controls System. The jumper remained in effect until maintenance/troubleshooting was completed and then the jumper was removed. Vacuum Trip System #2 remained operable as required by Tech Specs. The FSAR states that reduced availability is acceptable because the condenser low vacuum trip is not required for safety or for reactor protection. Plant safety or safe plant operations were not affected by this temporary modification.

This temporary modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000664-011, Rev. 0
TSCR #279 – Request for Safety/Relief Valve Monitoring System

This Tech Spec Change Request (TSCR) No. 279 deletes the Technical Specification requirements for minimum number of operable channels associated with the Acoustic Monitoring System used for indicating the position of the Electromatic Relief Valves (EMRVs) in Tech Spec 3.13.A.3 and Table 3.13.1. The TSCR also changes the requirement for the total number of channels required for each EMRV to one of either the primary or the backup indications (T.S. 3.13.A.2). These proposed changes did not affect nuclear safety or safe plant operation.

This evaluation determined that the proposed changes were acceptable without creating an unreviewed safety question or an adverse environmental impact. NRC approval was required prior to implementation. This TSCR was approved by Amendment 214 to the Operating License.

SE-000665-015, Rev. 1
Surveillance Test Frequency for Fire Protection Detection Alarms

The Fire Protection Program was revised to change the frequency of the fire protection system testing from once per 6 months to once per 12 months. This change was consistent with the requirements of the National Fire Code and Nuclear Electric Insurance Manual. The deviation from NFPA 72D and 72E regarding the surveillance frequency of fire detectors, supervised circuits, water flow switches and pressure switches has been justified in the evaluations in accordance with NRC GL 86-10. Nuclear safety or safe plant operations were not affected.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000731-013, Rev. 1
Recirc. Pump MG Set Motor "D" Cable Replacement

The previous revision of this modification addressed replacement of the power cable to the "D" Recirc. Pump Motor Generator (MG) Set Motor. The cable ran from the 4160V Switchgear Room to the Recirc. Pump MG Set Room. This revision (Rev. 1) to evaluation, SE-000731-013, covered changes in the cable routing. The old cable/conduit, which ran below ground from the Feed Pump Room Pit to the 4160V Switchgear Room, was abandoned and replaced by new conduit and cable above ground. Nuclear safety or safe plant operations were not affected.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000731-014, Rev. 2
Offsite Power Clarifications

Revision 1 of this evaluation covered the changes to the status of some of the offsite power connections to the plant. The 34.5 kV line (Z52) changed from a normally active source of power, to a normally open circuit that may be closed under certain conditions. The 69 kV Sands Point line is being credited as a normally active power source. Technical Specifications are met in that three active power sources were available with the addition of the Sands Point line. Revision 2 of evaluation, SE-000731-014, clarified what constitutes active offsite power sources. Nuclear safety or safe plant operation was not affected by these clarifications.

These clarifications did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000731-015, Rev. 0
Feedwater Pump Motor 1A Cable Replacement

This modification replaced the power feed cable 14-9A from the 4160V Switchgear to Feed Water Pump 1A motor. The old cable had failed during operating cycle 17. New conduits were also installed between the 4160 V Switchgear Room and the Feed Pump Room. This modification did not affect nuclear safety or safe plant operation.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000735-024, Rev. 1
Removal of Auto Transfer Switch (ST-C)

The previous revision of the evaluation addressed removal of the 480 VAC Auto Transfer Switch (ST-C) formerly being used as a disconnect switch. The transfer switch had only one power feed. The device was not being used as a transfer switch and could have failed, affecting the reliability of the 125 VDC System. This revision (Rev. 1) of evaluation (SE-000735-024) added a seismic evaluation in section 3.3.3. This additional evaluation did not affect nuclear safety or safe plant operation.

This addition did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000814-014, Rev. 0
Listing of Technical Requirements – Fire Barrier Doors

Table 6 of the Fire Protection Program (Procedure 101.2) was revised to provide more clarity regarding fire doors. Those doors (whether fire rated or not) are listed which are required to be closed in order to insure safe reactor shutdown in the event of a fire. These doors are designated as Category 1. Category 1 doors will require compensatory measures, when determined to be inoperable. The evaluation revision also identifies those doors, which are fire rated for reasons other than safe shutdown. The latter are designated as Category 2. These clarifications did not affect nuclear safety or safe plant operation.

These clarifications did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000826-016, Rev. 0
Changing X/Qs for the Control Room Intake on Licensing Basis Calculated Doses

This safety evaluation reviewed the use of a new computer code (ARCON96) for evaluating the dose consequences and hence habitability of the Oyster Creek control room. The USNRC accepted ARCON96 as a valid method of calculating X/Qs. Although ARCON96 is an approved code, the Murphy Campe method was specified as the method used in the original submittals for Oyster Creek. It was concluded that changing the code used to calculate X/Qs for control room habitability would not be consistent with Oyster Creek's licensing basis. A License Amendment would be required if the ARCON96 code were to be used at Oyster Creek in the future.

Subsequently, License Amendment 225 was approved which permitted the use of ARCON96 for Oyster Creek.

SE-000851-008, Rev. 1
Service Air System Flow Diagram

ECD C307472 was issued to document a discrepancy between the plant drawing and the as-found condition in the Service Air/Instrument Air Systems in Old Radwaste. Several valves and instruments on instrument rack, RK-09, were shown on the flow diagram as Service Air for purposes of tank mixing in the Filter Sludge Storage Tank, NV-09. The applicable documents were corrected. The previous revision evaluated the discrepancy. This revision (Rev. 1) to evaluation SE-000851-008) added the Section Manager's signature to the evaluation documents. This activity did not affect nuclear safety or safe plant operation.

This clarification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000856-012, Rev. 1
Service Air System As-Found ECD

ECD C312183 was issued to document a discrepancy between the plant drawing and the as-found condition in the radwaste service air source hose connection for WC-P-10, Chem Waste Sludge Transfer Pump. The applicable documents were corrected. The previous revision of this evaluation reviewed the discrepancy. This revision (Rev. 1) to evaluation (SE-000856-012) added the Section Manager's signature to the evaluation documents. This activity did not affect nuclear safety or safe plant operation.

This clarification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000856-013, Rev. 0
Temp Mod – Alternate SA-P-001 Power Feed

This Temporary Modification supplied an alternate power source to the New Radwaste Air Compressor, SA-P-001. The normal power cable from 1E1-480V USS had failed on ground fault. There are no safety-related functions for the New Radwaste Service Air System. Nuclear safety or safe plant operations were not affected. Subsequently, the Temporary Modification was replaced with a permanent electrical feed from MCC-1E12, which was installed during Operating Cycle 18.

This temporary modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000871-029, Rev. 1
Removal of North Yard Domestic Water Hypochlorite Mixer (X-10-006)

The previous revision evaluated the removal of the North Yard Domestic Water Hypochlorite Mixer (X-10-006). A UFSAR referenced drawing had to be updated. This revision (Rev. 1) to SE-000871-029 adds the Section Manager's signature to the evaluation documents. Nuclear safety or safe plant operations are not affected.

This clarification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000882-011, Rev. 1
Removal of the Reactor Building Crane Fixed Link Support System

Oyster Creek's 100-ton Reactor Bldg. Crane was upgraded to a single failure proof system defined by NUREG-0612 and NUREG-0554. The previous revision of the evaluation reviewed the removal of the Fixed Link Support System (FLSS) that existed on the old crane to be used during 100-ton cask handling. The FLSS was no longer necessary and had to be removed since it interfered with installation of the new trolley. This revision (Rev. 1) to SE-000882-011 made minor changes to the evaluation. Nuclear safety or safe plant operations were not affected.

This clarification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000882-012, Rev. 0
Electrical Work Associated with Reactor Bldg. SFP Crane Upgrade

This modification made electrical changes on the Reactor Building Crane in preparation for the new trolley and subsequent upgrade of the crane to a Single Failure Proof (SFP) design. The electrical loading on Vital MCC 1B2 remained bounded by the previous load calculation. The electrical changes made by this modification increased crane reliability. The changes did not affect nuclear safety or safe plant operations.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000882-014, Rev. 1
Reactor Building Crane Seismic Switch

The previous revision to the evaluation reviewed a modification which installed seismic instrumentation to the Reactor Bldg. Crane as part of the upgrade to a single failure proof design. The modification will remove crane power during a seismic event and cause all crane brakes to be set, holding any loads in a safe condition. This revision (Rev. 1) to SE-000882-014 covered a minor wiring change and changed the setpoint of the seismic vertical axis sensor. These changes did not affect nuclear safety or safe plant operation.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000882-015, Rev. 1
Installation, Use and Removal of Jacking Tower during Reactor Bldg. Crane Upgrade

The previous revision to the evaluation reviewed the use of the Jacking Tower, which was used to lift the new trolley to the Reactor Bldg. Crane bridge girders and remove the old trolley from the bridge girders to the refueling floor. Installation/removal of the jacking tower was controlled by an ACECP work plan (REP-18410-004) and activities to replace the trolley were controlled by an ACECP procedure (REP-18410-006). This revision (Rev. 1) to SE-000882-015 revised rigging and assembly drawings. These changes did not affect nuclear safety or safe plant operation.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000882-016, Rev. 0
Reactor Bldg. Crane Weld Sample Extraction and Test Plan – NW Middle Flange Weld

This evaluation addressed the removal of a small weld sample from the Reactor Bldg. Crane Bridge Girder. The structural function or load carrying ability of the bridge girder was not affected by the small sample removal. The sample was removed to analyze an UT indication. This activity did not affect nuclear safety or safe plant operation.

This weld sample removal did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000911-004, Rev. 0
Fire Hazards Analysis Change

Administrative changes were made to the Fire Hazards Analysis Report (FHAR) and Procedure 101.2 that did not change regulatory commitments or system configuration or operating modes. The changes eliminated duplication of information between the FHAR and related documents, provided additional or updated references, and made clarifications to the FHAR. The changes did not affect nuclear safety or safe plant operation.

These document revisions did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000911-005, Rev. 0
Fire Hazards Analysis Change – Fire Doors

Changes were made to the Fire Hazards Analysis Report (FHAR) to eliminate discrepancies between selected fire doors and NEPA 80-1983, "Standard for Fire Doors and Windows." The changes did not impact the existing functional evaluations documented for these doors in the FHAR. The FHAR changes did not affect nuclear safety or safe plant operation.

These document changes did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000911-006, Rev. 0
Upper Cable Spreading Room Vestibule Evaluation

This evaluation addressed the existing configuration of a vestibule wall located on the South boundary of Fire Zone OB-FZ-22A (Upper Cable Spreading Room). The evaluation concluded that the wall meets the requirements of a two-hour barrier as defined by the Oyster Creek Fire Hazards Analysis Report (FHAR). The wall was adequate to prevent the spread of flames, smoke and hot gases across it. The FHAR will be updated to say that the wall is considered acceptable as is. Nuclear safety or safe plant operation was not affected.

This evaluation did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000911-007, Rev. 0
Evaluation of C Battery Room Ceiling/Roof Barrier Rating – FHAR Revision

This evaluation addressed the acceptability of the existing configuration of the ceiling/roof barrier in the C Battery Room. Based on an inspection of the C Battery Room and the 4160VAC Switchgear A & B Area, the minimal amount of combustibles in the C Battery Room, the limited access to the room, and the control of transient combustible materials, the boundary between the two areas can be reasonably expected to provide a fire barrier rating equivalent to the hazards in the areas in order to meet the intent of Appendix A to Branch Technical Position APCSB 9.5-1 guidelines. Nuclear safety and safe plant operation has not been affected.

This evaluation did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-000911-008, Rev. 1
Evaluation of the 4160V "C" & "D" Vaults – FHAR Revision

This evaluation addressed the acceptability of the existing configuration for the areas associated with the C and D 4160V Switchgear Rooms. Specific items evaluated included the ceiling/roof barrier, the beams within the C and D room dividing wall, and other specified walls and interior beams. The evaluation concluded that this configuration has been determined to be acceptable for Appendix A to BTP APCSB 9.5-1 and for Appendix R in accordance with GL 86-10 criteria. Nuclear safety and safe plant operation has not been affected.

This evaluation did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-335400-040, Rev. 0
Oyster Creek Core Operating Limits Report – Cycle 18 Core Loading Configuration

This evaluation addressed the revision to the Core Operating Limits Report (COLR) which incorporates the Cycle 18 reload. The Cycle 18 reference design did not reduce the margin of safety defined in the UFSAR. The reload safety analysis has shown that a Minimum Critical Power Ratio (MCPR) operating limit of 1.56 will ensure that the new fuel cladding safety limit will not be violated. All other safety limits and fuel operating limits will not be challenged with the core loading. The new MCPR limit was added to the Cycle 18 COLR Report. This activity did not affect nuclear safety or safe plant operation.

This evaluation and COLR revision did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-400007-001, Rev. 0
TSCR 281 – Removal of Heavy Load Restriction over the Spent Fuel Storage Pool

The Reactor Bldg. Crane has been upgraded to single-failure-proof design as defined in NUREG-0554. Utilizing the single-failure-proof crane as a method of controlling heavy loads in, out, and over the spent fuel storage pool requires an amendment to the plant Technical Specifications. This evaluation addressed Tech Spec Change Request (TSCR) 281, which documented the justification for revising Tech Spec sections 5.3.1B and 5.3.1C to remove restrictions for moving heavy loads over the Spent Fuel Storage Pool. Nuclear safety or safe plant operation was not affected.

This evaluation determined that the proposed changes were acceptable without creating an unreviewed safety question or an adverse environmental impact. NRC approval was required prior to implementation. This TSCR was approved by Amendment 223 to the Operating License.

SE-402925-002, Rev. 2
Main Turbine Generator Protection Relay

The previous revisions of this evaluation addressed a modification which provided additional protective features for the main turbine generator. This revision (Rev. 2) to SE-402925-002 specifically removed statements that incorrectly indicated that the emergency diesel generator timing in response to LOOP/LOCA is dependent on turbine steam valve cutoff trip to the 1A and 1B breakers. Nuclear safety or safe plant operation was not affected by this revision.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-403024-001, Rev. 2
Miscellaneous SQUG Support and Modifications

Previous revisions of this evaluation addressed the original SQUG program and modifications required to close identified issues developed through implementation of the Seismic Qualification Utility Group (SQUG) program. This revision (Rev. 2) to SE-403024-001 specifically addressed the Emergency Diesel Generator Bldg. Roof modification with respect to high winds, tornado and tornado generated missile loads. The roof modification was found to be adequate for these loads, which were documented in Calculation C-1302-157-E540-009. Nuclear safety or safe plant operation was not affected by this revision.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-403092-002, Rev. 0
Anchorage Upgrade of MCC-1A12, MCC-1B12, and ER-661-100

This modification upgraded the equipment support anchorages for electrical equipment, MCC-1A12, MCC-1B12, and ER-661-100 to assure seismic adequacy in conformance to Plant design bases. These anchorages were previously identified as outliers in the Seismic Qualification Utility Group (SQUG) program, which addressed Unresolved Safety Issue No. A-46. Equipment was not moved, added, or replaced and pressure boundaries were not impacted. Anchor bolts were installed in accordance with site procedures to assure that reinforcing was not cut and that building integrity was not affected. Nuclear safety or safe plant operation was not affected by this revision.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-945100-357, Rev. 0
Revision to the Oyster Creek Emergency Plan

Changes to the Emergency Plan were necessary to provide flexibility, if needed. The change allows the Communications Coordinator role in the OSEO to be assumed by appropriately trained, qualified personnel other than a CRO licensed individual. This allows more efficient utilization of personnel to meet the demands of any emergency situation. Additionally, the position of OSC Coordinator allows assignment of appropriately qualified individuals other than an Operations or Maintenance Supervisor. These changes did not decrease the effectiveness of the Emergency Plan and Implementing Procedures.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-945100-359, Rev. 2
UFSAR Section 13.1 Management and Technical Support Organizational Changes

Previous revisions of this evaluation addressed changes to Section 13.1, "Organizational Structure' in Oyster Creek's UFSAR. Changes resulted from the merger and restructuring of Exelon Corp. to the extent that AmerGen Energy Co. was affected. This revision (Rev. 2) to SE-945100-359 specifically addressed additional changes that reflect further restructuring and realignment of management and technical support organizational structure for the Exelon nuclear fleet. The organizational realignment was made to take advantage of various department synergies, and to improve and optimize support of the entire nuclear fleet. The changes did not affect nuclear safety or safe plant operation.

This UFSAR revision did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

SE-966200-001, Rev. 1
Permanent Drywell Shielding

Previous revision of this evaluation addressed the modification which installed permanent drywell shielding that has been designed to remain safely in place during plant operation and postulated seismic conditions. This revision (Rev. 1) to SE-966200-001 specifically addressed the number of lead wool blankets and attachment methods. This revision also added references to the evaluation. There was no effect on nuclear safety or safe plant operation.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0002, Rev. 0
Replacement of Service / Instrument Air Compressors P-6-1 & P-6-2

Oyster Creek has three full capacity Service / Instrument Air Compressors. This modification replaced the existing Service / Instrument Air Compressors 1-1 & 1-2 with two package units on individual skids. The third Air Compressor 1-3 was not be affected by this modification. The UFSAR was updated to reflect this modification and to remove extraneous detail about the air compressors.

The Service & Instrument Air Compressors have been high maintenance items for quite some time. This modification replaced two of the air compressors with more reliable units. The replacement air compressors are functionally the same with regard to the operation and design basis of the UFSAR described SSC's. There are no adverse effects on system functions. Additionally, the air compressors are not required for the safe shutdown of the reactor nor to mitigate the consequences of postulated accidents.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0003, Rev. 0
Alternate Replacement of Condenser Fan Motors @M-826-001B (2B1 through 2B6)

This activity was an Alternate Replacement of Condenser Fan Motors for Control Room HVAC "B." The replacement motors were equivalent in performance to the original motors. This activity was classified as an Alternate Replacement. There was no design change to the Control Room HVAC "B" as described in the UFSAR. This replacement activity was expected to contribute to the increased reliability on the Control Room HVAC.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0004, Rev. 0
Procedure Revisions to Account for 2 DWEDT Pumps Being Inoperable

These procedure changes provided an alternate means of calculating identified leakage if both Dry Well Equipment Drain Tank (DWEDT) pumps fail to operate. The revised procedures included 351.2 (Rev. 44), 351.1 (Rev. 76), 312.9 (Rev. 23), and 106 (Rev. 124). These procedure changes established a fixed value for identified leakage, and subtracts this value from the 1-8 sump integrator readings to obtain a conservative value for the unidentified leak rate when no DWEDT pumps are operating. The 5-gpm Technical Specification limit remains the same.

These procedure changes did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0005, Rev. 0
Temporary Breach of Secondary Containment for Cleanup Pump Cable Replacement

This activity removed the old cables and installed new cables in 2.5-inch diameter electrical conduits for Cleanup System pumps, P-16-001A and P-16-001B. The conduit passed through the Reactor Building wall. It provided a path between the Stack Tunnel and the Torus Room during the activity. The cable replacement required a temporary breach in the Secondary Containment boundary. The opening could have been immediately closed by personnel stationed at the job during the work activity, if required. At the end of the activity, a permanent air seal was installed.

This activity did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0007, Rev. 0
UFSAR Chapter 3.5.1.3, Turbine Missiles, Clarification Change

Oyster Creek committed to the NRC in response to SEP Topic III.4.B to follow General Electric (GE) recommendations for managing annual turbine missile generation probability. GE developed a PRA-type evaluation methodology that determined the annual turbine missile generation probability through the analysis of key inspection and operational parameters. The NRC approved the GE methodology for use by licensees in establishing operational and maintenance practices in NUREG 1048, Appendix U.

This change to the UFSAR clearly identified the parameters that make up the annual turbine missile generation probability. It also identified Oyster Creek's commitment to the NRC to maintain the annual probability of generating a turbine missile below 1E-5 through proper implementation of GE recommendations.

This UFSAR change did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0008, Rev. 3
Replacement of Portions of Emergency Service Water System 2 Piping

This modification installed a piping bypass (approximately 150 feet long) around a portion of Emergency Service Water System 2 (ESW-2) piping that was leaking. The piping to be bypassed is in the buried section between the Intake Area and the Turbine Building. The reason for this activity was that a leak developed in the buried ESW-2 piping near the Chlorination House.

The replacement piping was functionally identical to the existing piping except that it has not been coated as indicated in the UFSAR. The lack of coating has been satisfactorily addressed in the 50.59 Evaluation. The UFSAR, Section 6.2.2.2 will be revised to discuss the uncoated pipe section. The leaking section of ESW-2 pipe, which was bypassed, was abandoned in place.

This Modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2001-E-0011, Rev. 1
Addition of Maintenance Isolation Valves in the CRD System

This modification installed hand wheel operated isolation valves in the branch headers of the Control Rod Drive (CRD) Drive Water and Cooling Water lines. The installation of isolation valves in the drive and cooling headers alleviates the need to tag out as many as 274 individual isolation valves in order to perform certain outage maintenance on the CRD hydraulic system. The net effect of the modification is to reduce dose, manpower, time, and processing the drained hydraulic control unit water inventory.

This modification did not alter the design basis of the UFSAR described SSC's. There were no adverse effects on any function of the CRD hydraulic system. There was no change to the text of the UFSAR but the flow diagram that is referenced by the UFSAR will be revised to include the new valves.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2002-E-0001, Rev. 0
MNCR O2001-1839 Disposition/Operability Assessment

During the review of the pump nozzle loads of Containment Spray pumps 1-1 and 1-2, the pump manufacturer determined that the cast iron suction nozzles were overstressed based on their design code, ASME Section VIII, 1965 Edition. Due to the late identification of the pump nozzle issue and the fact that the evaluation and resolution are complex, the work could not be completed in time to support the next refueling outage, 1R19. Therefore, to comply with NRC Generic Letter 91-18, Rev. 1, a 50.59 evaluation was completed to extend the operability assessment of these pumps until 1R20. Restoration of these pumps to design compliance in 1R20 is being tracked by MNCR O2001-1839.

This operability assessment did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2002-E-0003, Rev. 0
Oyster Creek Increased Core Flow Implementation

This activity provided an evaluation and justification for the installation of new digital APRM Flow Control Trip Reference (FCTR) cards and for operation in the analyzed Increased Core Flow domain at Oyster Creek. The results of the evaluation demonstrated that installation of the new APRM-FCTR cards was acceptable and that the proposed increase in maximum flow was acceptable – no fuel thermal limits or other licensing basis acceptance criteria were violated. This activity has no effect on UFSAR-described design functions, or methods of performing or controlling design functions and all existing safety analyses continued to be bounding.

This activity was implemented near the end of operating Cycle 18. The reload safety analyses for operating Cycle 18 has been performed utilizing the revised maximum core flow value. This analysis has established the required operating Cycle 18 fuel thermal limits such that all fuel related acceptance criteria were satisfied. The required Cycle 18 fuel thermal limits have been documented in a revision to the Core Operating Limits Report.

This evaluation did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2002-E-0004, Rev. 1
Evaluation of the Six-hour Service Water System Outage During 1R19

A section of the 20-inch Service Water (SW) piping upstream of the SW flow element was replaced with a tee and set of isolation valves. The purpose of this modification was to provide a tie-in point for a new 12-inch line, which cross connects Emergency Service Water (ESW) System 1 to the SW System. The new 12-inch line will allow ESW System 1 flow to the RBCCW heat exchangers so that maintenance can be performed, during cold shutdown conditions, on portions of the SW System upstream of the new tee and valves.

This activity does not alter the design basis of the UFSAR described SSC's. The Safety Related function of the ESW remains the same and there are no adverse effects on any system function.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2002-E-0005, Rev. 0
New Procedure (205.94) to Flush Core Spray System Piping

A new procedure (205.94 – RPV Floodup Using Core Spray) was developed to flush Core Spray piping from the Condensate Storage Tank and to flood up the reactor cavity during a refueling outage. These activities were initially performed during the 1R19 refueling outage. This method of flooding up will save time during refueling outages by increasing the flow rate at which floodup can occur.

System flushing will be done while reactor temperature is above 212F. Core Spray System 1 will be out of service and will be aligned to the Condensate Storage Tank. Core Spray System 2 will be fully operational. There will be no adverse effect on operation of SSC's due to this new procedure.

This new procedure did not involve an unreviewed safety question and no changes to the Technical Specifications were required.

OC-2002-E-0006, Rev. 0
Oyster Creek GE11 New Fuel Introduction

GE11 fuel was introduced into Oyster Creek beginning with operating Cycle 19. The GE11 fuel design has improved operational characteristics that result in improved operational flexibility and improved core efficiency. Plant specific analyses and evaluations have been performed to demonstrate that the GE11 fuel can be safely used at Oyster Creek. The evaluations demonstrated that no fuel thermal limits or licensing basis acceptance criteria were violated. Therefore, this activity will have no effect on UFSAR-described design functions, or methods of performing or controlling design functions, and all existing safety analyses continued to be bounding. Procedures were revised, as required, to reflect operation with GE11 fuel. The Core Operating Limits Report was also revised to reflect updated operating conditions associated with Cycle 19 operations.

This modification did not involve an unreviewed safety question and no changes to the Technical Specifications were required.
