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June 22, 1992
C321-92-2164

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
10CFR50.59(b) Reporting Requirements

In accordance with the requirements of 10 CFR 50.59, enclosed are summaries of the changes to Oyster Creek systems and procedures, for the period January to December 1991, as described in the Safety Analysis Report (SAR). Attachment 1 of this report addresses those activities which directly affected systems/components described in the SAR. Attachment 2 of this report addresses those activities for which a GPU Nuclear safety evaluation was performed, due to the potential for the activity to adversely affect nuclear safety or safe plant operations, but which do not directly impact SAR systems/components.

If you have any questions, please contact Mr. M. W. Laggart, Manager Corporate Licensing, at (201) 316-7968.

Very truly yours,

J. J. Barton
Vice President and Director
Oyster Creek

cc: Administrator, Region I
NRC Resident Inspector
Oyster Creek NRC Project Manager

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Activities Directly Affecting Systems/Components
Described in the Safety Analysis Report

I. Procedure/Document Changes

Procedure: Process Control Plan for Transfer & Solidification of Solid Wet Waste Via CNSI Cement Solidification System

Description of Change: The purpose of this procedure change is to add operational instructions and guidance for the new Rapid Dewatering System (RDS-1000) unit. This new system will allow us another method of processing filter media for land disposal. This system has been approved for use by the NRC under Topical Report No. RDS-25506-01-P/NP, Rev. 1.

Safety Evaluation Summary: This procedure change allowing the Radwaste Shipping Department to use the RDS-1000 Dewatering System will in no way change the final waste form acceptance criteria as described in 10 CFR 61.56, FSAR, Tech Specs and in the P.C.P. Procedure 351.36. What this procedure change does is to allow us another method of processing filter media for land disposal instead of our only current method which is cement solidification. This new method of dewatering has b[illegible]tudied and approved by the NRC as evidenced by their acceptance of CNSI Topical Report [illegible]506-01-P-A. All of the requirements of this Topical Report have been included into [illegible]nment VII of Procedure 351.36, thereby insuring that the final waste form will meet the requirements of 10 CFR 61.56 (b) (2) and all of the existing licensed burial site criteria as is required by 10 CFR 61, the Tech Specs and FSAR.

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Procedure: Augmented Offgas System Operation

Description of Change: This is a one time change to Procedure 350.1 to allow exceeding the prerequisite to maintain charcoal temperature below or at 50 degrees F. This temperature limit was imposed in order to obtain the delay times of 20 days from xenon and 26 hours for krypton. Increasing the temperature allowance will probably decrease these holdup times. However, it is better to operate the AOG in this decreased efficiency mode than to shut it down and have no holdup time. This activity is safe because:

1) The charcoal beds are designed for -20 degrees F temperature.

2) Vendor manual VM-OC-0193 page 4041 item 17 states if the vault coolers both fail causing high temperature in any adsorber bed then operate the system unless high radiation in the effluent stream is detected.

Safety Evaluation Summary: Since there has been no increase seen in activity in the stack gas monitors and no increase seen in AOG radiation monitors, this safety evaluation is good for 60 days.

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<u>Procedure</u>: Alternate Water Supply to AOG Heat Exchanger

<u>Description of Change</u>: This mini-mod will install a new connection upstream of the AOG Closed Cooling Water Heat Exchanger (CC-H-2A). The exact location will be between the inlet isolation valve (S[illegible]V-105A) and the heat exchanger. This new connection will permit the use of an alternate cooling water supply when NRW Service Water is unavailable. It will permit connection without taking AOG out of service.

<u>Safety Evaluation Summary</u>: The safety concerns associated with performing this work have been evaluated and it is determined that this work will not adversely affect nuclear safety or safe plant operation, does not involve an unreviewed safety question, and does not require a change to the Tech. Specs.

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II. Modification: Replacement of Valves V-3-87 and V-3-88
SE 408773-010

Description of Modification: This modification replaced nuclear safety related motor-driven butterfly valves V-3-87 and V-3-88 with hand-operated dedicated commercial grade butterfly valves.

Safety Evaluation Summary: The new valves are locked in positions hydraulically equivalent to the existing valves. The modification does not generate an unreviewed safety question. It will not generate an environmental impact different from that which currently exists. The modification has no detrimental affect on plant or personnel safety or function. This modification does not require a Tech. Spec. change.

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Modification: Feedwater Drain and Vent Addition
S.E. 408773-007

Description of Modification: These modifications improve the vent and drain capability for the Feedwater System. Additionally, this modification includes a connection to permit draining the feedwater system back to the hotwell when the plant is off-line. The modification installs a one inch vent line from each FW string between the HP Heater and its discharge isolation valve and add 2" drain connections on the suction and discharge sides of the FW pumps. Additionally, a two inch connection has been added to the condensate pump suction header to permit draining to the hotwell. The existing 3/4 inch penetration to the hi-lo conductivity room has been expanded to two inches. Draining to the hotwell, floor drains or lo conductivity tank is via a temporary hose. The vent line can be used as an injection point for the on-line calibration of the combined FW flow element.

Safety Evaluation Summary: This modification will not adversely affect nuclear safety or safe plant operations because operation of the FW and condensate system is not changed. This mod is only intended to facilitate drainage for maintenance and system refilling. These modifications will not change the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety. These lines are all isolated by hand operated valves. The valves are only open when the system is isolated for maintenance or, in the case of the vent connections, when an on-line calibration of the combined FW flow element is made. The systems in which these connections has been installed are not nuclear safety related. The connections added by this modification are intended for maintenance or test. The operational modes of the affected systems are not changed. Therefore, no accident or malfunction of a different type than any previously evaluated in the SAR is introduced by these modifications. The margin of safety as defined in the Tech. Spec. is not changed by these modifications since none of these systems are discussed in the Tech. Spec.

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Modification: 1-2 Sump Upgrade
SE 408773-005

Description of Modification: The purpose of this modification is to replace the existing 1-2 Sump Pumps and Level Control System which have had a history of unreliability. Their failures have led to unnecessary maintenance and personnel radiation exposure. In addition, the controls for the pumps will be relocated from a "High Radiation Area," the Condenser Bay Flash Tank Pit, to a "Low Radiation Area," the Feedpump Room North Wall.

Safety Evaluation Summary: This modification will have no environmental impact. Modification upgrades 1-2 sump to a more reliable status. This modification will have no adverse impact on nuclear safety or safe plant operations.

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Modification: A/B Air Dryer Replacement
SE 402956-001

Description of Modification: The purpose of this modification is to continue providing dry, filtered air for instrumentation and controls at Oyster Creek Nuclear Generating facility. This has been obtained by the installation of the new A/B desiccant air dryer. The previous A/B dryer had had internal failures due to components such as the desiccant bed screens. Higher moisture conditions and desiccant break-throughs have resulted. Therefore, there is a potential to create loss of air events and as a result a plant trip. This safety evaluation pertains to the installation of the new Pall Pneumatics internally heated desiccant air dryer. The dryer has been installed in the Turbine Building basement.

Safety Evaluation Summary: The activity replaces an existing dryer with a new one. The air dryer shall be used as the primary supplier of dry, filtered air for instrumentation and controls at Oyster Creek. The dryer modification will not cause an unfavorable environmental impact. The Instrument Air System is neither required for safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. Therefore, the activity will neither adversely affect nuclear safety or safe plant operations nor reduce the margin of safety as defined in the UFSAR and basis of the Plant Technical Specifications. Hence, there is no unreviewed safety question in accordance with 10 CFR 50.59.

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Modification: RPV Head Vent Valve Addition
SE 402953-010

Description of Modification: The purpose of this modification was to install a 2-inch manual isolation valve (V-22-767) downstream of the reactor head vent valves V-25-21 and 22. This valve provides an isolation boundary for reactor coolant in case of leakage through the head vent valves in order to reduce identified leakage to the drywell equipment drain tank (DEDT).

Safety Evaluation Summary: This modification adds a manual valve in the reactor head vent piping. This modification does not generate an unreviewed safety question. It will not generate an environmental impact different from that which currently exists. The modification has no detrimental affect on plant or personnel safety or function.

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Modification: Service Water Venturi Inst. Tubing Cleaning Tap Modification
SE 402953-008

Description of Modification: The service water flow venturi Y-3-001 on occasion experiences plugging of the high and low pressure taps at the venturi. This may affect the element's accuracy. Accurate flow measurement is required for IST on the SW Pumps. This modification installs tubing and valves to the venturi pressure sensory lines which allows access to the venturi taps and allows for manual cleaning of the taps.

Safety Evaluation Summary: This modification does not affect plant environment interfaces nor violate plant environmental Technical Specification. This modification installs tubing and valves which allow for cleaning of the pressure sensing taps on the Service Water Flow Elements. The Service Water System is not required for safe shutdown of the plant nor to mitigate the consequences of a postulated accident. Therefore, modification to this system does not affect Nuclear Safety.

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Modification: Appendix J Replacement Options
SE 402946-001

Description of Modification: This modification includes modifications for several primary containment penetrations so that their respective isolation valves could be properly tested in accordance with the requirements of 10CFR50, Appendix J. Only one of the penetrations was installed during cycle 13R, the remaining work will be completed in Cycle 14R.

The modification which was installed providod a test barrier in the form of a spectacle flange in the 20" diameter torus to Reactor Building Vacuum Breaker Line. Also included in the modification was a 3/4" test connection and a pipe support. The spectacle flange itself was designed so as to be testable when rotated to its normal operational position.

This modification provided the ability to do a type "C" test on several isolation valves in the proper direction (i.e., from the containment side). Affected isolation valves included: V-26-16, V-26-18, V-23-16, and V-23-20.

Safety Evaluation Summary: This modification will not create or impact existing plant effluents. Therefore, this modification will not have an adverse effect on plant effluents nor will it create an environmental concern not evaluated in current environmental requirements documents. The proposed modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse affect on nuclear safety or the environment.
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Modification: Total Feedwater Flow Element Addition
SE 402945-001

Description of Modification: The purpose of this modification was to install one new flow element in the common feedwater header located in the Heater Bay of the Turbine Building. This location is downstream of the HP feedwater heaters and upstream of the Reactor Vessel inlets. This flow element provides input to an electronic transmitter mounted in the Feedwater Pump Room. As a result of the installation of one flow element in the feedwater header, the following associated modifications are required:

- Relocate temperature elements, TE-0046 and TE-0047, downstream of new flow element because of interference and detrimental impact on the upstream flow metering run.

- Final Feedwater (FW) sample isolation valve, V-2-365, must be relocated since its present location interferes with the new flow element installation.

- The new flow element will also require relocation of a high point vent (isolation valves V-2-238 and V-2-239) on the 24 inch common header. The design will minimize welding to the existing process line by combining the functions of the high point vent and the final Feedwater sample takeoff into one connection.

- The new FW flow element will be calibrated "in-situ" using a chemical trace injection method. A separate Safety Evaluation (SE 402-945-002) has been prepared by GPUN for this test.

- A section of the Emergency Service Water (ESW) 14 inch line will be rerouted due to interference from the new feedwater flow element flanges.

Safety Evaluation Summary: The proposed modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse effect on nuclear safety or the environment. No changes to the Plant Technical Specifications are required as a result of this modification.

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Modification: Diesel Fuel Oil Storage Tank Replacement
SE 408815-001

Description of Modification: The Oyster Creek Tech. Spec. require that an active source of power and a standby source of power are aligned to plant essential loads while the plant is shutdown. During shutdown, the startup transformers are available as the active source of electrical power and the Emergency Diesel Generators (EDG's) are available as the standby source. No other sources of power are readily available. The Tech Specs state that both EDG's must be declared inoperable if the 15,000 gallon fuel oil storage tank (T-39-002) contains less than 14,000 gallons. Because of these restrictions, the diesel fuel oil storage tank cannot be partially or totally drained, which hinders tank maintenance and inspection, and the tank cannot be taken out of service for tank replacement.

Safety Evaluation SE-000862-003 requested a change to the Tech Specs to allow the storage and use of 14,000 gallons of fuel oil stored in temporary tanker trucks. These tanker trucks would be connected to the EDG's when the plant is shutdown for refueling. This would allow partial or full drainage of the diesel fuel oil storage tank and would allow the tank to be taken out of service and replaced.

Safety Evaluation SE-000862-003 addresses the use of the temporary tanker trucks, their connection to the EDG's, and the drainage or taking out of service of the Diesel Fuel Oil Storage Tank (T-39-002).

This modification was installed after obtaining the required change to the Tech Specs requested in SE-000862-003.

Safety Evaluation Summary: This modification will not create nor impact existing plant effluents because it is not changing existing plant conditions. Temporary tanks and hoses have been designed to collect oil leaks and prevent them from entering soil or water. The new oil flow meter allows buried oil piping to be monitored for leakage. Therefore, it will not have an adverse effect on plant effluents or create an environmental concern not evaluated in current environmental requirement documents.

The modification replaced the diesel fuel oil tank with a larger capacity tank and revised existing oil supply piping by adding a fuel oil flow meter and hose connections. These modifications enhance system maintenance and allows the system to be monitored for oil leakage. The overall function of the diesel fuel oil storage and transfer system was not changed by this modification. The modification will not have any adverse effect on nuclear safety or the environment, and does not result in an unreviewed safety question as determined by the 10CFR50.59 evaluation.

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Modification: Tornado Missile Shield
SE 409745-001

Description of Modification: The previous hurricane shelter entrance to the Radiation Control Area (RCA) will be reopened during 13R. This 7'-4"x9'-0" opening is now covered by steel plates. These steel plates provide protection from potential airborne missiles to the 480V switchgear located inside this opening. A Monitor and Control (MAC) facility will be provided to increase ease of access into and out of the RCA. As part of this modification, the existing steel plates will be removed. Therefore, a tornado missile shield must be added to shield the opening so that any postulated airborne missile strike during a tornado event will not damage/disable the 480V switchgear equipment which is located right inside of the opening. In order to protect the 480V Switchgear Room from an exposure fire, the missile shield and a fire door shall be three-hour fire rated.

Safety Evaluation Summary: The tornado missile shield is designed to withstand applicable design loads including dead load, live load, safe shutdown earthquake loads, tornado wind and missile loads. It is structurally separate from the Reactor/Office Building so that any potential settlement will not impact the NSR structures. No impact on any safety-related equipment or structures will occur and no change to any plant procedure or operation will required. Therefore, it is concluded that the modification will not have any adverse effect on nuclear safety or the environment.

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Modification: Temporary Demineralizer System
SE 312100-003

Description of Modification: A temporary demineralizer system was installed at Oyster Creek to process water collected in the Chem Waste/Floor Drain Collection Tanks, WC-T-1A/B/C, in lieu of using the Radwaste Concentrators, WC-E-1A/B. The initial Demineralizer System is expected to be in operation for 2-5 years. The Safety Evaluation is to demonstrate the safety of the installation and design of the system as described in Installation Specification OC-IS-312100-002.

Safety Evaluation Summary: The plant margin of safety is not reduced by use of this system. Nuclear safety and safe plant operation are not adversely affected. There is not an increase in the probability of consequences of an accident previously evaluated in the SAR. Important to safety equipment is not adversely affected. The system does not create an accident or malfunction of a type previously identified in the SAR. No Tech Specs or other license based document is violated. No radiological safety concern exists. Use of this system complies with the intent of Reg. Guide 1.143 to "...provide reasonable assurance that...radioactive waste management...systems are designed, constructed, installed and tested on a level commensurate with the needs to protect the health and safety of the public and plant operating personnel. The system is acceptable to install and use.

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Modification: TBCCW, SFP and SDC Heat Exchanger DP Gauges
SE 312400-010

Description of Modification: The purpose of this modification was to install differential pressure indicators across the Reactor Building Closed Cooling Water (RBCCW) side of the Spent Fuel Pool Heat Exchangers and the Shut Down Cooling (SDC) Heat Exchanger (C-17-003), also across the Service Water/Circulating Water side and the Turbine Building Closed Cooling Water (TBCCW) side of the TBCCW Heat Exchangers. These indicators are used to monitor flow through the heat exchangers to support flow balancing the RBCCW and TBCCW systems and monitor tube fouling in the TBCCW Heat Exchangers.

Safety Evaluation Summary: This modification has no impact on the Spent Fuel Pool Cooling System since the barrier between this system which is safety related and RBCCW remains the tubes and tube sheets of the heat exchangers. This modification also adds local differential pressure indication across the tube side (Circulating or Service Water) and shell side (TBCCW) of the TBCCW heat exchangers. This provides some indication of the tube fouling and allow the plant to better determine required cleaning frequency. This also enables the plant to better control flow through the heat exchangers to minimize velocity induced tube vibrations. This modification has no adverse effect on TBCCW, Service Water or Circulating Water System availability or performance. The modification also adds local differential pressure indication across the shell side (RBCCW) of the third Shutdown Cooling Heat Exchanger (Note, other two already have dp indication). This will not impact the Shutdown Cooling System since the barrier between this system and RBCCW remains the tubes and tube sheets of the heat exchanger.

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Modification: Connection to Condensate Transfer for Alternative Supply
SE 323368-001

Description of Modification: In the event of unavailability of the normal source of condensate transfer water due to maintenance or repair it is still necessary to provide a source of appropriate quality water to the Condensate Transfer System. During plant cold shutdown, the Condensate Transfer System is required to provide seal water to various Radwaste pumps and for other services. This modification enables High Purity Sample Tank, HP-T-2B, to be used as the source of the water. It is not meant to provide an alternate source of Tech Spec mandated water.

Safety Evaluation Summary: The modification will not increase the probability of occurrence or the consequences of an accident previously evaluated in the SAR because loss of Condensate Transfer System flow is already evaluated and spills of fluids with higher radionuclide concentrations have also been evaluated. This modification will not create the possibility for an accident or malfunction of a different type than any previously identified in the SAR since the only two credible accidents/malfunctions are the loss of system flow or system leakage. This modification does not generate an unreviewed safety question.

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Modification: Installation of Test Plugs for Control Rod Scram Insertion Time Test and Valve IST test
SE 323560-003

Description of Modification: This modification eliminates lifting leads and installation of jumpers used for connecting the recorder terminals with each control rod for recording control rod scram insertion time test. This modification allows the Plant Operators and the technicians to connect and/or disconnect the recorder located in the cable spreading without disturbing the original designed circuit terminations in Control Panel 6XR.

Safety Evaluation Summary: This modification provides a capability to perform control rod scram insertion time test and valve IST test surveillance in Panel 6XR without installing jumpers and/or lifting leads. This modification will not alter designed functions of the control rod system. This modification minimizes the possibility of human error when performing the related surveillance. This modification does not constitute an unreviewed safety question. There is no environmental impact due to this modification and there are no changes required to the Tech Specs.

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Modification: Installation of Test Plugs for Intermediate Range Monitors Test & Calibration
SE 323560-004

Description of Modification: This modification eliminates lifting leads and installation of jumpers used for simulating the conditions to perform Intermediate Range Monitors Test and calibration surveillance on startup or shutdown of the plant. This modification allows the Plant Operators and the technicians to connect and/or disconnect the instruments to be used for surveillance without disturbing the original designed circuit termination.

Safety Evaluation Summary: This modification provides a capability to perform IRM logic test and calibration surveillance in Panel 4F without installing jumpers and/or lifting leads. This modification will not alter designed functions of the Intermediate Range Monitors. This modification minimizes the possibility of human error when performing the related surveillance. This modification does not constitute an unreviewed safety question. There is no environmental impact due to this modification and there are no changes required to the Tech Specs.

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Modification: 125V DC System Knife Switch Modification
SE 323560-008

Description of Modification: On September 16, 1989, prior to plant startup, the 125V DC control power to Unit Substation 1B2 (USS1B2) was found misaligned to Station Battery A instead of Station Battery B. The former is not qualified to supply safety related loads.

In response to this incident an Independent Review Group was convened to review the circumstances related to the misaligned manual throwover switch. The review group issued IOSRG Root Cause Analysis Report No. 8916 which identified the root cause, conclusions and recommendations. The recommendations included procedure revisions, verification of all plant components based on drawings, verification of line-ups and improve the labeling on the manual throwover switches.

This modification ensures that the incident will not recur. The manual throwover switches for USS 1B2, USS 1B3, 4160V SWGR 1B and 4160V SWGR 1D have been modified as followings:

a. A two point terminal block was installed adjacent to the manual throwover switches.

b. The existing alternate power supply from 125V Distribution Center A has been disconnected from the manual throwover switch and connected to the two point terminal block.

c. Identification labels and instructions have been installed next to the two point terminal blocks.

Alternate power will still be available but is controlled using Tempora Variations (TV). In the event that alternate power is required temporary jumpers will be connected from the two point terminal block to the manual throwover switch.

Safety Evaluation Summary: In conclusion, it can be stated that this modification:

o Disconnect the alternate 125V DC control power from the manual throwover switch and reconnect it to the new two point terminal blocks adjacent to the switches. All installation and wiring requirements will be done within t switchgear compartment. This modification will prevent recurrence of the incident on September 16, 1989.

o Does not change the function of the interfacing systems.

o Has no effects on plant effluents.

It is concluded that the modification will not have any adverse affect on safety or safe plant operations, existence of an unreviewed safety question or a need for a Technical Specification change.

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Modification: Safety Valve Reduction
SE 402915-001

Description of Modification: The purpose of this modification was to eliminate seven (7) main steam safety valves (MSSV) at Oyster Creek. Four safety valves from one main steam line and three from the other main steam line inside the drywell were deleted, and the nine valves with the lowest setpoints were maintained. Appropriate safety analyses have been performed to demonstrate the acceptability of the reduction in the number of valves. A reduction in safety valves at OC results in significant cost savings in maintenance and surveillance testing. In addition, man-rem exposure is also considerably reduced.

The reactor pressure vessel (RPV) and the pressure relief system for OC were designed in accordance with Section I, 1962 edition of the American Society of Mechanical Engineers (ASME) "Boiler and Pressure Vessel Code". Under the provision of Section I, code qualified safety valves must limit the rise in the RPV pressure to less than the ASME code limit. Previous analyses performed to demonstrate compliance with the code requirements did not take credit for reactor scram, EMRVs, turbine bypass valves and isolation condensers. To satisfy this requirement OC previously employed 16 steam safety valves. The current version of the ASME code, Section I, allows credit for independent sensing devices that stop the flow of fuel to the boiler. Since the code is for fossil boilers, the analogy for a nuclear plant, such as Oyster Creek, is that credit for an independent or diverse shutdown system such as flux scram, would perform the same function of fuel stoppage, i.e. boiler shutdown. Thus, credit could be taken for their functioning in the overpressure protection analysis consistent with the current interpretation of the ASME code. NRC approval of high flux scram with no RPT as the new licensing basis for Oyster Creek allowed for the removal of seven main steam safety valves.

Safety Evaluation Summary: In summary, the purpose of this modification was to eliminate seven steam safety valves at Oyster Creek. This required a change to the plant licensing basis to take credit for high flux scram with no RPT in the design basis accident analyses. This safety evaluation demonstrates that no environmental impact is involved. A technical evaluation was performed to insure that there is an acceptable safety margin between the event acceptance limits and consequences of postulated accidents. This safety evaluation concludes that, with the changes to the FSAR and Technical Specifications, the change will not impact on the safety of the plant. Since the deletion of safety valves and credit for high flux scram in the design basis accident analyses involved a Technical Specification change, we have received NRC approval prior to implementation.

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Modification: Core Spray Alarm Modification
SE 402916-001

Description of Modification: The purpose of this modification was to improve operational effectiveness by providing new and upgraded Control Room alarms for the Core Spray System.

The first modification alerts the operator to a loss of fill water in the Core Spray lines, thus minimizing potential water hammer damage.

The second part of this modification involved a change to the existing "System Pressure Sw. Off Normal" alarms. The previous system design was a one-out-of-three-logic and activated the alarm on initiation of any one of the following conditions:

a. Booster pump differential pressure greater than 50 psig.
b. Core spray pump discharge pressure greater than 140 psig.
c. Reactor pressure below 300 psig.

The modification changes the alarm logic to a three-out-of-three configuration thus insuring more effective and meaningful alarming. In addition, wording for alarm windows B-2-e and B-2-f has been changed to provide more meaningful alarm information to the operators.

Safety Evaluation Summary: The modifications provide the plant operator with more useful alarm information, thereby improving operator effectiveness. In addition, the modification does not contain an unreviewed safety question as defined by 50.59.

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Modification: RPS Half Iso. Signal Monitoring
SE 402918-003

Description of Modification: This modification provides an input to the annunciator and Sequence of Alarm Recorder from the RPS isolation relays to make the operator aware that a half isolation signal is present (separate annunciator window and SAR input are provided for each channel). The annunciator/SAR input is made up by series connecting a spare contact on each of the RPS isolation relays 1(2)K71, 1(2)K72, 1(2)K73, 1(2)K74, 1(2)K75 and 1(2)K76. Providing the input from these relays provides indication when half an isolation signal has been generated as a result of a single erroneous system input, test switch in the test position, or inadvertent failure of a circuit component.

Safety Evaluation Summary: There is no adverse affect on nuclear safety or safe plant operation since the interface is at the relay contacts and does not interconnect with the relay coil or respective control power. No unreviewed safety questions exist as a result of this modification.

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Modification: RWCU Drywell Penetration Pipe Replacement
SE 402938-001

Description of Modification: The modification workscope consists of removing the portion of the RWCU system process piping located within the two drywell penetrations and replacing it with nuclear grade stainless steel material which is resistant to intergranular stress corrosion cracking (IGSCC). The replacement pipe has been designed to eliminate pressure retaining welds in the portion of the pipe located within the penetration, since in the current arrangement these welds are inaccessible for inspections.

Safety Evaluation Summary: This safety evaluation addressed the replacement of penetration piping in the reactor water cleanup system. As indicated in Section 3.0, this evaluation shows that an unreviewed safety question does not exist as defined in 10CFR50.59. In summary, the modifications eliminate uninspectable pressure retaining welds inside the RWCU system drywell penetrations.

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Modification: R.B. - RWCU Valve Nest Hatchway Cover
SE 402938-002

Description of Modification: In order to permit the installation of a replacement Reactor Water Cleanup (RWCU) process piping during the 13R outage and in order to provide future access for personnel into the RWCU valve nest, a 6 foot by 4.5 foot access opening was cut in the 4 foot thick concrete floor slab at elevation 75'-3" of the OCNGS Reactor Building. A permanent hatch cover with lead shielding was installed in this opening in order to reduce the radiation exposure on the elevation 75'-3" floor to levels similar to those existing prior to the implementation of this modification. In order to maintain existing floor traffic patterns, a checkered plate cover has been installed over the shielding, at floor level. This plate is designed to support loads equal to those previously specified for the existing concrete floor.

This Safety Evaluation addresses operation of the plant after this hatchway is cut and its cover installed. Cutting of the opening and installation of the cover and shielding is addressed in the construction package.

Safety Evaluation Summary: The purpose of this modification was to provide a hatchway through the elevation 75'-3" concrete slab of the Reactor Building, into the RWCU valve nest below, to be used for installation of modifications and personnel access. The concrete slab with the opening, as well as the hatchway cover will be able to withstand all applicable design loads and will provide adequate shielding. No impact on any NSR/RR equipment or structures will occur and no change to any plant procedure or operation will be required. Therefore it is concluded that the modification will not have any adverse effect on nuclear safety or the environment.

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Modification: Drywell Chiller Unit Piping Modification
SE 402939-001

Description of Modification: During outages, temporary piping must be installed inside and outside of the Drywell so that chilled water can be supplied to the Drywell Cooling Units. However, this installation has been dependent upon opening both the inner and outer doors of the Drywell Airlock, causing a delay in supplying chilled water at the outset of an outage. The Drywell Chiller Unit provides chilled water, via Reactor Building Closed Cooling Water (RBCCW) System lines, to the Drywell Cooling Units, which cool the Drywell to a more comfortable temperature for outage activities.

The purpose of this modification was tc install permanent piping from the Drywell Chiller to new RBCCW System tie-in poin's external to the Drywell. This allows the Chiller to be placed in service early in an outage.

Safety Evaluation Summary: This modification installs permanent piping from the Drywell Chiller to new RBCCW tie-in points outside the Drywell. This allows the Chiller to be placed in service early in an outage. This modification will not have any adverse effect on nuclear safety or the environment, and does not result in an unreviewed safety question as determined by the 10CFR50.59 evaluation.

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Modification: Torus Oxygen Sample Line Isolation Valves Modification
SE 402872-001

Description of Modification: The purpose of the modification was to accomplish the following activities:

- Modify the control circuit for the TOSL isolation valves V-38-22 and V-38-23 to add independent reactor protection contacts for the initiation of each containment isolation valve.

- Replace existing two position (OFF-ON), maintained control switch 12XR-18 with a new three position (OFF-AUTO-ON), spring return to AUTO from ON, control switch to prevent automatic reopening of valves when the CIA signal is reset.

- Modify wiring and utilize an existing spare relay in Panel 12XR in the control circuit for V-38-22 and V-38-23 to comply with single failure criteria and to prevent automatic reopening of valves when CIA signal is reset.

- Modify the control wiring of TOSL isolation Valve V-38-23 to utilize the spare conductors routed under BA 402815 to prevent a common cable failure from disabling both TOSL isolation valves.

Safety Evaluation Summary: This modification will not create or impact existing plant effluents. Therefore, this modification will not have an adverse effect on plant effluents nor does it create an environmental concern not evaluated in current environmental requirements document. It is concluded the proposed modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse effect on nuclear safety or the environment.

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Modification: RPS/ESG Instrument Upgrade - RE23A/B/C/D Pressure Switch Replacement
SE 402879-002

Description of Modification: The purpose of the RPS/ESG Instrument Upgrade Modification was to install replacement pressure switches for PS-RE0023A, B, C, and D because of component obsolescence.

These switches initiate Main Steam Isolation Valve and Main Steam Line Drain Valve closure when activated by low main steam line pressure.

Safety Evaluation Summary: This modification makes no change in any existing function performed by these switches. This modification introduces no new single failure which would prevent initiation of required safety function.

The probability of occurrence, consequences or type of an accident or malfunction other than previously described in the FSAR has not been [illegible].

There is no environmental impact resulting from this modification.

Based on the above evaluation, it is concluded that this modification will not have an adverse effect on nuclear safety or an adverse environmental impact. This modification does not create an unreviewed safety question as described in 10CFR50.59.

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Modification: Isolation Condenser Pipe Replacement
SE 402900-001

Description of Modification: The modification workscope includes replacement of all isolation condenser piping outside the drywell, and the replacement of the piping within the four drywell penetrations. These changes eliminate IGSCC susceptible piping material and reduce the number of welds outside the drywell, and eliminate uninspectable pipe pressure boundary welds within the penetrations. Piping inside the drywell is not part of this workscope except for the piping within the penetrations and the first elbow closest to each penetration. In addition, the six isolation valves on the system piping outside the drywell have been replaced with valves of an upgraded design. These valves are also equipped with connections to facilitate performing leak tests in accordance with Appendix J of 10CFR50.

Safety Evaluation Summary: This safety evaluation addressed the replacement of penetration piping and all the isolation condenser system piping outside the drywell and within the penetrations. This evaluation shows that an unreviewed safety question does not exist as defined in 10CFR50.59. In summary, the modification resolves several existing high energy line break licensing issues and result in a plant configuration more closely in conformance with present day licensing requirements.

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Modification: Turbine Building Closed Cooling Water Corrosion Monitoring
SE 323636-001

Description of Modification: The purpose of installing corrosion coupons in the Turbine Building Closed Cooling Water System (TBCCWS) was to monitor the corrosion rates of the major component materials of the system. Corrosion coupons and a by-pass piping assembly have been added to the 3/4" line (CC-2 and CF) on the chemical addition tank. The installation may be done while the system is inservice since there are two existing isolation valves. Valves were added to isolate the chemical addition tank and corrosion monitoring assembly, and to drain the assembly. The specific materials that will be monitored are aluminum brass, aluminum bronze, carbon steel and 70/30 copper nickel. The system was previously using a molybdate-based corrosion inhibitor. The coupons shall be removed and/or replaced to determine the corrosion rates and if pitting has occurred. This shall verify the effectiveness of the sodium molybdate-based inhibitor. This modification is in response to an INPO finding CY.3-2 (1989).

Safety Evaluation Summary: The corrosion monitoring will not cause an unfavorable environmental impact. The TBCCWS is neither required for safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. Therefore, the activity will neither adversely effect nuclear safety or safe plant operations nor reduce the margin of safety as defined in the UFSAR and Technical Specifications.

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Modification: Reactor Building Closed Cooling Water Corrosion Monitoring
SE 323636-002

Description of Modification: The purpose of installing corrosion coupons in the Reactor Building Closed Cooling Water System (RBCCWS) was to monitor the corrosion rates of the major component materials of the system. Corrosion coupons and a by-pass piping assembly has been added to the 3/4" line (CC-4 and CF) on the chemical addition tank. The installation was done while the system is inservice since there are two existing isolation valves. Valves were added to isolate the chemical addition tank and corrosion monitoring assembly, and to drain the assembly. The specific materials that will be monitored are stainless steel, titanium, carbon steel and 70/30 copper nickel. The system was previously using a molybdate-based corrosion inhibitor. The coupons shall be removed and/or replaced to determine the corrosion rates and if pitting has occurred. This shall verify the effectiveness of the sodium molybdate-based inhibitor. This modification is in response to an INPO finding CY.3-2 (1989).

Safety Evaluation Summary: The corrosion monitoring will not cause an unfavorable environmental impact. The RBCCWS is neither required for safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. Therefore, the activity will neither adversely effect nuclear safety or safe plant operations nor reduce the margin of safety as defined in the UFSAR and Technical Specifications.

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Modification: Augmented Offgas Closed Cooling Water Corrosion Monitoring
SE 323636-003

Description of Modification: The purpose of installing corrosion coupons in the Augmented Offgas Closed Cooling Water System (AUGCCWS) was to monitor the corrosion rates of the major component materials of the system. Corrosion coupons and a by-pass piping assembly have been added to the 1" line (CC-191 and CC-189) on the chemical addition tank. The installation may be done while the system is in-service since there are two existing isolation valves. Valves shall be added to isolate the chemical addition tank and corrosion monitoring assembly, and to drain the assembly. The specific materials that will be monitored are 90/10 copper nickel, stainless steel, carbon steel and 70/30 copper nickel. The system was previously using a molybdate-based corrosion inhibitor. The coupons shall be removed and/or replaced to determine the corrosion rates and if pitting has occurred. This shall verify the effectiveness of the sodium molybdate-based inhibitor. This modification is in response to an INPO finding CY.3-2 (1989).

Safety Evaluation Summary: The corrosion monitoring will not cause an unfavorable environmental impact. The AOGCCWS is neither required for safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. Therefore, the activity will neither adversely effect nuclear safety or safe plant operations nor reduce the margin of safety as defined in the UFSAR and Technical Specifications.

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Modification: Core Spray/ADS Logic Modification
SE 328279-001

Description of Modification: This modification modifies the automatic start logics of the ADS and Core Spray System. The ADS logic is modified such that the Core Spray booster pump differential pressure on either Core Spray System will enable both ADS Divisions provided coincident reactor vessel triple low water level and drywell high pressure signals are present.

Currently, if a Core Spray System becomes inoperable, the ADS is affected such that two logic subchannels in the division become inoperable. As a result, the plant must be brought to cold shutdown within 30 hours of the event as required by Tech. Spec. 3.1. The ADS logic change modifies the CS interface such that all ADS logic subchannels remain operable should a Core Spray System become inoperable. This allows the plant to remain in operation for a period not to exceed 7 days. (Tech. Spec. 3.4) Basically, the change involved a realignment of the 16K114 relay contacts only, with no other changes being made to the ADS system.

The Core Spray System logic is modified to lock out the backup booster pump if the primary booster pump on the same electrical division is running. Currently, should a LOCA occur coincident with a loss of offsite power and loss of a diesel generator, all core spray pumps associated with the remaining electrical division (diesel) will come on line. This is a total of four pumps (2 mains and 2 boosters). This modification locks out the backup booster pump such that only 3 pumps will be loaded onto the diesel (2 mains and 1 booster).

Safety Evaluation Summary: The modification to the Core Spray System and ADS logic will not increase the probability of the occurrence of any accident. Further, it will not reduce the margins of safety as defined in the FSAR and the bases for Technical Specifications. Therefore, no unreviewed safety questions as defined by 10CFR50.59 are involved.

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Modification: Reactor Manual Control System Modification
SE 328298-001

Description of Modification: The purpose of this modification was to remedy a condition in the Reactor Manual Control System which enabled the selection and possibly simultaneous withdrawal of two (2) control rods as a result of a malfunction in the Reactor Manual Control System. A situation of this type occurred at Oyster Creek and is documented in a Post Trip Review Report, PTRG-89-133, Abnormal Rod Motion. Potential safety concerns are identified in PSC 90-003 and evaluated by Safety Evaluations SE-644-001- and SE-644-002.

Two problem areas were identified as a misoperation of the rod select push buttons. The modification corrects the situation by rewiring the rod control selector pushbuttons in such a manner as to preclude selecting two rods for movement. All wiring modifications were done internally in control room panel 4F.

Safety Evaluation Summary: The modification will greatly reduce the probability of simultaneous control rod withdrawal. In addition, the modification does not contain an unreviewed safety question as defined by 50.59.

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Modification: Containment Airborne Particulate and Gaseous Radiation Monitor System (CAPGRMS) - Phase II
SE 402815-002

Description of Modification: This modification, identified as CAPGRMS-Phase II augments the prerequisite activities performed in CPM-Phase 1. Phase II removes the existing CPM and completes the installation and startup of the replacement CAPGRMS.

This modification includes the reinforcing of the existing unistrut wall in the 480V Switchgear Room "A" so that the power supply to CAPGRMS will be available following a seismic event.

Safety Evaluation Summary: It is concluded the modification does not create an unreviewed safety question and will not have any adverse effect on nuclear safety or the environment.

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Modification: SWRM Upgrade Phase II
SE 402857-003

Description of Modification: The purpose of the SWRM Upgrade Phase II was to make permanent installation of the temporary changes made in Phase I and upgrade the system based on considerations of service conditions. The Phase I configuration is maintained with the exception of the flow indicator in the bypass line. The flow indicator in the bypass line was replaced with an in-line magnetically coupled flow indicator and a differential pressure gauge across the duplex strainer. The differential pressure gauge was installed across the duplex strainer to preclude the need for filter replacement on a routine basis. Phase II also replaces SWRM system piping and components with materials resistant to corrosion in a seawater environment. Phase II revises the existing Hewlett-Packard (HP) Processor program for the "Environment Water Monitor" alarm located in Control Room Panel 5F/6F to include the condition of the loss of canal dilution.

Safety Evaluation Summary: The purpose of this modification to the Service Water Radiation Monitoring System was to upgrade the system by eliminating electrical noise and reducing potential for material degradation. Since this modification does not affect safety related equipment or systems, this modification will have no adverse effect on nuclear safety or the environment and will not result in an Unreviewed Safety Question.

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Modification: ESW Flow Indication
SE 402859-001

Description of Modification: The purpose of the Emergency Service Water (ESW) flow indication modification was to add flow indication in the control room via the plant computer system. Two displays of Emergency Service Water flow, one to each of the containment spray systems are provided. This parameter was not displayed in the control room. This display is required to comply with USNRC Regulatory Guide 1.97. Additionally, the local flow indicator for each containment spray system was relocated from their current locations on the emergency service water piping to local instrument supports.

Safety Evaluation Summary: It is concluded the modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse effect on nuclear safety or the environment.

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Modification: Heater Bay Roof Trailers, Fire Protection
SE 402860-003

Description of Modification: This modification provided sprinkler fire protection for five (5) trailers located on the heater bay roof. The system is a dry-pipe type utilizing nitrogen to pressurize the piping to keep the control valve closed. Two sprinkler heads are provided in each trailer.

Safety Evaluation Summary: The implementation of this modification does not affect nuclear safety nor safe plant operation, nor does it increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety evaluated in the Safety Analysis Report and does not constitute an Unreviewed Safety Question.

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Modification: Circulating Water Pump Pit Conduit Replacement
SE 402869-001

Description of Modification: This modification replaced the corroded conduits with a new raceway system consisting of a combination of corrosion resistant (PVC coated) conduits within the pit area and a fiberglass tray system above the pit area. This approach is based on BRC Report 3731-050 and the results of a subsequent review meeting held between BRC and GPUN. This approach provides flexibility of performing major construction work during normal plant operation, allows sequencing of work activities over a number of operating or outage periods and has minimal schedule impact on the outage duration.

Safety Evaluation Summary: It is concluded the modification does not constitute an Unreviewed Safety Question and will not have any adverse effect on nuclear safety or the environment.

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Modification: Sewer Water Radiation Monitor Upgrade
SE 328293-004

Description of Modification: This corrective change replaces various components of the sewer water Radiation Monitoring System. The replacement components are functionally identical to the previous ones. The change is to increase the reliability of the system. The system has a scintillation detector located outside of the Nuclear plant facility with the remaining monitoring equipment located in the Ragems Building. The system is not considered or identified as a radiological release point. There are no connections to plant systems where radioactive materials may be postulated to be released during normal operations or accident conditions. Failure of this system would in no way affect the operation of the plant or the capability to monitor release of radioactive material. It is classified as Regulatory Required.

Safety Evaluation Summary: Based on the above, this modification does not have the potential to adversely effect nuclear safety or safe plant operations. This evaluation precludes the occurrences of an unreviewed safety question.

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Modification: Main Feedwater Line Block Valve Addition
SE 402901-002

Description of Modification: The purpose of this modification was to add tight shutoff valves downstream of the two Main Feedwater Regulating Valves (MFRV) that are in parallel with the two low flow feedwater regulating valves. This will prevent leakage via the MFRVs during plant startup and shutdown, allowing the flow from either feedwater pump used for plant startup or shutdown to be positively controlled by the low flow feedwater regulating valves. The lockup valves on each MFRV have been replaced with one new valve that provides positive lockup of the MFRVs. The pneumatic tubing and associated components have also been modified.

Safety Evaluation Summary: The MFRV block valve addition modification does not constitute an unreviewed safety question and will not have any adverse effect on nuclear safety, safe plant operations, or the environment.

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Modification: Repaired Transformer M1A Installation and Reconnection
SE 992100-002

Description of Modification: The purpose of this modification was to reconnect the repaired main transformer M1A to the 23KV isolated phase bus and various plant auxiliary systems. This modification will also be utilized to rearrange wiring and cabling within the junction box (TB# 37-4) located at the south-end of the 1C Condenser Bay in order to terminate auxiliary transformer differential CT cable, main transformer neutral CT cable, main generator CT cable, and 230KV bus differential CT cable in a new junction box located near the M1A transformer to facilitate testing by JCP&L.

Safety Evaluation Summary: This modification installed the repaired main transformer M1A. In this mode of operation, the plant on-site electrical distribution system remains powered from the main generator via the Unit Auxiliary Transformer in the same manner and extent as the original plant design. There is no change to the operation of plant safety systems, Technical Specification requirements and limits or adverse impact on the plant environment. No experiments or tests are performed, which would adversely affect the plants safety. Hence, this modification which installed the repaired main transformer M1A does not affect the margin of safety or create an unreviewed safety question as described under 10CFR50.59.

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Modification: Temporary Torus Fans TM-SF-1-2 and TM-EF-1-1 and Filter TM-EF-1
SE 000820-002

Description of Modification: A temporary torus supply fan (TM-SF-1-2) was installed in the Reactor Building at elevation 23'-6" and a temporary exhaust fan (TM-EF-1-1) and filter (TM-EF-1) were installed in the Reactor Building at elevation 75'-3" to support torus modification during the 1983-84 outage. The fans, filter, and support bases are non-seismic, however it was desirable to allow them to remain permanently in place for use in future outages to support torus and torus room work.

Safety Evaluation Summary: Permanent storage of fans TM-SF-1-2, TM-EF-1-1, and filter TM-EF-1 in the Reactor Building do not involve an unreviewed safety question or require a technical specification or license amendment, nor does it affect nuclear safety or safe plant operation.

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Modification: Transformer M1B Permanent Use
SE 000723-002

Description of Modification: The purpose of this evaluation is to document the permanent connection of the UP&L GSU transformer to the 23KV isolated phase bus and other plant auxiliary systems and to operate it in parallel with the repaired M1A transformer.

Safety Evaluation Summary: The permanent use of the repaired M1A and UP&L GSU transformers in parallel provides the same configuration as the original plant design. The plant on-site power system remains powered from the main generator via the Unit Auxiliary Transformer in the same manner and extent as the original plant design. The 230KV system fault current contribution to the 4160V safety buses is slightly greater than before, but is within the equipment ratings. There is no change to the operation of plant safety systems, Technical Specifications requirements and limits, no interface with any radiological system or material, and no impact on the plant environment. No experiments or tests are performed which would adversely affect the plant's safety. Therefore, permanent use of the UP&L GSU in parallel with the repaired M1A transformer does not affect the margin of safety or create an unreviewed safety concern as described under 10CFR50.59.

****

Modification: Monitor Sump for Conduit Pit in Feedwater Pump Room
SE 408843-001

Description of Modification: This modification provided a new monitor sump into the conduit pit for the purpose of drainage and removal of water from the underslab area in the feedwater pump room. The water is piped to the 1-3 sump which is in the feedwater pump room. The 1-3 sump is a controlled sump to radwaste where the water is treated and released.

Safety Evaluation Summary: The implementation of this modification does not affect nuclear safety nor safe plant operation, nor does it increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety evaluated in the Safety Analysis Report.

****

Modification: Chemical Decontamination Support Reactor Cavity Alternate Draindown Mod
SE 328265-003

Description of Modification: The purpose of this modification was to install connections to provide an alternate reactor vessel draindown path while the Reactor Water Cleanup System (RWCU) is out of service. This path connects the Shutdown Cooling (SDC) system pump "C" minimum flow recirculation line to the RWCU draindown station piping. The SDC and RWCU piping tie-ins are permanent connections. They were temporarily joined during the 13R outage using rubber hose. Upon completion of reactor cavity operations, the hose will be removed, the SDC tie-in capped and the RWCU tie-in blind flanged.

Safety Evaluation Summary: This modification added piping tie-in connections to the RWCU and SDC piping. These modifications will not affect the normal operation of the RWCU or SDC system, nor impact the safe shutdown of the reactor, nor will they be utilized during normal power operations. As a result, it has been concluded that the modifications will not have any adverse effect on nuclear safety or the environment.

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Modification: Seismic Modification of Accumulator Piping
SE 402933-002

Description of Modification: In response to Generic Letter 88-14 issued by the Nuclear Regulatory Commission, a seismic review of the air supply lines to the Reactor Building Heating and Ventilating System isolation valves was conducted. This evaluation concluded that the air piping to the ten valves listed below lacked sufficient flexibility to permit seismic induced movement of the ventilation duct or air supply header. During an earthquake, the forces created by this movement could result in a breach in the pressure boundary of the air piping. Such a failure could render the isolation valves inoperable.

To remedy this situation, a modification was initiated which installed a flexible metal hose in the air supply piping of the following valves.

V-28-5	V-28-13	V-28-42
V-28-6	V-28-14	V-28-43
V-28-7	V-28-21	
V-28-8	V-28-22	

Installation of the flex hose permits the anticipated seismic movements without jeopardizing the pressure boundary integrity of the air piping.

Safety Evaluation Summary: The installation of the flexible hose in the piping which supplies air to operate the RBHV System isolation valves is required to ensure these valves remain functional both during and after a design basis earthquake. The ability of the system to perform its safety function of isolating the reactor building to minimize the risk of a ground level release of airborne radioactive material, is not impaired. It is determined that no unreviewed safety question or environmental impact will result from implementing this modification.

****

Modification: Remove Core Spray Min. Flow Check Valves
SE 000212-018

Description of Modification: The check valves in the Core Spray System minimum flow line perform the following functions.

1.1 During minimum flow operation of the Core Spray System, these valves must open to permit sufficient recirculation flow to prevent possible damage to the pumps.

1.2 They close to prevent backflow from the operating booster pump to the idle pump. This ensures proper minimum flow for pump protection and required flow to the reactor core can be obtained.

The modification involves removing the internals of the Core Spray System minimum flow check valves. These valves require testing to show operational readiness. No methods are currently available to perform such tests. It has been calculated that these valves are not required to prevent backflow through an idle booster pump. Adequate flows will be obtained to provide protection to the running pumps. The ability of the Core Spray System to perform its intended safety function, which is to deliver low pressure, rated flow to the reactor core following any loss of coolant accident, will not be impaired by this modification.

Safety Evaluation Summary: It is determined no unreviewed safety question or environmental impact will result from implementation of the subject modification.

****

Modification: Plugs in Drain Lines from WC-T-003A & 3B
SE 000232-004

Description of Modification: Temporary variations 88-27 and 90-03 installed a rubber pipe plug in the Chem Waste Distillate Sample Tanks (WC-T-3A and WC-T-3B) drain line (2"-NV-21) to stop the tank from leaking. The rubber pipe plug has a stainless steel wire attached which runs to the top of the tank to allow removal of the plug. Since repair of the line would be difficult and costly (requires removal of tank bottom) and the line is used only when complete tank draining is required it has been decided to leave the plug and attached removal wire in place. FCN C094535 documents this change as a permanent chance annotating the existence of the drain plug on the system flow diagram which is included in the OCNGS FSAR (Figure 11.2-2C and 11.2-3).

Safety Evaluation Summary: This change does not adversely affect nuclear safety or safe plant operations because the function of the Chem Waste/Floor Drain Collection/Processing System has not changed.

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Modification: Off Gas Delay Pipe Drain Line
SE 000331-004

Description of Modification: This modification provides a means of using demineralized water to clear the gas delay line drain that leads to the 1-12 sump. Temporary Variation (Mech) 90-16 will be converted to a permanent configuration change.

Safety Evaluation Summary: The modification to use demineralized water to clear the gas delay line drain produces no detrimental effects on safety or the environment because the resultant drainline is equivalent to the present drainline except for the clearing function.

<u>Modification</u>: Temperature Monitoring Systems' Replacement, C.R. Panels 12XR and 10R
SE 408773-004

<u>Description of Modification</u>: The purpose for this modification was to upgrade the temperature monitoring systems, located in the Control Room Panels 12XR and 10R. The original equipment by Rochester Instrument Systems Inc., Model TM-400, is discontinued, obsolete and aged to the point that there are no available spare parts. The scope of this modification was limited to replacement of the electronic TM-400 modules and associated panel mounted meters only. The selected replacement system is by the same manufacturer, Rochester Instrument Systems, Inc., Model TM-2480. It offers greater accuracy, reliability and maintainability along with functions, similar to TM-400, but improved by modern technology.

<u>Safety Evaluation Summary</u>: The replacement of the obsolete electronic temperature monitors with functionally similar but otherwise updated modern systems improves the accuracy, maintainability and reliability of the affected Temperature Monitoring Systems. This modification does not affect any important to safety equipment. Therefore the probability of occurrence or the consequences of an accident or malfunction of such equipment is not increased by this modification. The SAR evaluated accidents or malfunctions do not include the Temperature Monitoring Systems, affected by this modification. Additionally, the greater accuracy and reliability of the new systems shall decrease the possibility for associated malfunctions and accidents. The Technical Specification defined safety margin is not affected by this modification. As determine above, there is no adverse impact on nuclear safety or safe plant operation by this modification.

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<u>Modification</u>: AOG Blower Motor AMP Recoater Upgrade
SE 408773-009

<u>Description of Modification</u>: This modification permanently installs the recorder in the AOG control room panel EE-OB-1. The connections to each of the motor control centers shall be permanently wired to the new recorder. For convenience of the AOG operators, the motor control center for train B (MCC-1E14) will use the spare pen on recorder case #3. This will put the recorder and digital indication of current draw on the same end of the control panel, in close proximation to the instrumentation needed to put the train in service, rather than put both pens together in the same recorder and make it difficult to take readings from the far end of the panel.

<u>Safety Evaluation Summary</u>: The consequences of previously evaluated accidents or consequences of equipment malfunctions will not be affected by this modification because this modification does not change the function of the existing chart recorder. This modification does not create an unreviewed safety question, as determined by 50.59. This modification will not increase the likelihood of an accident or malfunction beyond those

already postulated for the Augmented Offgas System. The function of the recorder replaced by this modification will be unchanged. This modification does not reduce the margin of safety of any Technical Specification basis or safety system design associated with the Augment Offgas System. This modification only addresses a strip chart recorder, which is not required for Technical Specification compliance.

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Modification: Installation of Hydraulic Snubbers on RWCU Pressure Switches
SE 402953-003

Description of Modification: The installation of hydraulic snubbers upstream of pressure switches PS-IJ04A, PS-551-0082 and PS-215-1044 does not have the potential to adversely affect nuclear safety or safe plant operation. This modification does not require a revision to the system, component, procedural and/or operation description in the FSAR or any other part of the SAR. The RWCU and ECPM Systems have been experiencing spurious isolation trips. It is believed that some of these spurious trips may be caused by pressure spikes in the regions of pressure switches PS-1J04A, PS 551-0082 and PS-215-1044 which provide input signals to RWCU and ECPM isolation interlocks. Installation of hydraulic snubbers in sensing lines to these switches would dampen pressure spikes and thus eliminate some or all of the spurious trips.

Safety Evaluation Summary: The installation of hydraulic snubbers in sensing lines to switches PS-IJ04A, PS-551-0082, and PS-215-1044 are intended to dampen pressure spikes and thus eliminate RWCU and ECPM spurious isolation trips. This modification will in no way affect Nuclear Safety or Safe Plant Operation; not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety; not increase the possibility for an accident or malfunction of a different type than any evaluated previously; and not change the margin of safety as defined in the basis for any Technical Specification.

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Modification: Installation of Control Air Trip Valves on V-26-16 and V-26-18
SE 402953-012

Description of Modification: The V-26-16 and V-26-18 control air trip valves have become obsolete. Replacement parts are not available.
This mini mod replaces the existing obsolete trip valves with new trip valves and modify control air tubing and valves.

Safety Evaluation Summary: This modification replaces existing V-26-16 and 18 control air trip valves which have become obsolete. This replacement requires modification of the control air system. Modification will be mounted seismically and will be thoroughly tested to ensure that the requirements of V-26-16 and 18 are met. Therefore this modification does not effect nuclear safety or safe plant operation or the margin of safety.

****

Modification: Installation of 3/8" Tubing in Reactor Bldg.
SE 402953-006

Description of Modification: The purpose of this activity was to install a 3/8" tubing through the 4" penetration sleeve on the south wall of the Reactor Building at elev. 29'10". This installation is required to support Hydro Nuclear Services in hydrolazing activities.

Safety Evaluation Summary: This modification does not have the potential to introduce an accident or malfunction of a different type than any previously evaluated. The function of this penetration sleeve is to maintain secondary containment while allowing access to the Reactor Building for hydrolazing services. Loss of containment has been evaluated in the FSAR. No other credible malfunctions or accidents can result from this installation. Since secondary containment will be maintained at all times when required, this modification does not violate environmental requirements or Technical Specifications. The installation of 3/8" tubing in the subject penetration sleeve is within the design basis and safety bounds of the SAR.

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Modification: Reactor Building Construction Power Upgrade
SE 402949-001

Description of Modification: The purpose of this modification was to provide additional construction grade power for the Reactor Building and for the Drywell. The drywell power will only be operable when the drywell is open.

Safety Evaluation Summary: This modification provides increased construction power availability on Reactor Building elevation 23'-6" and provides construction power distribution equipment inside the drywell. The modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse effect on safety or the environment.

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Modification: Installation of Test Plugs for Rx. Bldg. Ventilation Supply Valves Position Indication Check Surveillance
SE 323560-002

Description of Modification: This modification eliminates installation of contact blocks on the relay contacts of DX1, DX2, DX3 and DX4 relays in the control panel 11R in the control panel, used to simulate deenergization of Rx. Bldg. ventilation supply valves when performing surveillance on the Rx. Bldg. ventilation supply valves position indication check. This modification will allow the Plant Electricians and technicians to block the HFA relay contacts circuit without disturbing the original designed circuit terminations and the relay contacts.

Safety Evaluation Summary: This modification provides a capability to perform surveillance testing on Rx. Bldg. Ventilation Supply Valves position indication check without installing relay contact blocks in DX1, DX2, DX3 and DX4 HFA type relays installed in control panel 11R. This modification will minimize the possibility of human error when performing the related surveillance. This modification does not constitute an unreviewed safety question. There is no environmental impact due to this modification and there are no changes required to the Technical Specification.

****

Modification: Drywell Power Improvement
SE 402939-004

Description of Modification: The purpose of this modification was to provide non-diesel backed, construction grade power for welding and convenience receptacles in the drywell.

Safety Evaluation Summary: This modification provides 480V and 120V receptacle inside the drywell for maintenance activities during outages. The modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse effect on safety or the environment.

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Modification: Drywell Improvement - Audio & Visual Mod
SE 402940-002

Description of Modification: The Drywell Improvement Modification requires working conditions in the Oyster Creek drywell be enhanced with the primary objective to reduce worker exposure while increasing productivity. One area being addressed by the modification is Drywell Communications. Both audio and video improvements are addressed. The video portion of the modification provides/installs the closed circuit video system with all necessary electronic equipment, including the cameras, lenses, pan/tilt units, video recorders and monitors. The audio portion of the modification provides communication capability in high noise areas and/or while wearing full-face respirators. This system is configured with headsets, throat microphone, beltpak and base station to permit the user to talk to nearby co-workers while wearing a full-face respirator and protective clothing. In addition, this system also permits communication from outside to inside as well as inside to outside drywell for coordination of various tasks. This Nuclear Safety/Environmental Input Evaluation addresses the sealing of a 6" secondary containment penetration sleeve and cables.

Safety Evaluation Summary: This modification is for observation and communication purposes only. The sealing of cables within the sleeve provides the transition from secondary containment to outside containment (Drywell Processing Center). The modification does not constitute an Unreviewed Safety Question as determined by 10CFR50.59 and will not have any adverse effect on safety or the environment.

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Modification: IC Steam Line Vent P ing Modification
SE 402900-002

Description of Modification: The steam supply lines entering each Isolation Condenser are provided with 3/4 inch vent piping connected to the high points in the supply lines. The vent piping is responsible for removing noncondensable gases from the reactor steam which would otherwise collect at the high points and inhibit operation. The steam supply lines undergo relatively large displacements due to thermal expansion. These movements produce cyclic stresses in the vent piping.

The purpose of this modification is to alleviate the cyclic stresses in the Isolation Condenser System (ICS) steam line vent piping. The stresses will be reduced to acceptable levels consistent with the expected remaining life of the plant. Both thermal and Safe Shutdown Earthquake (SSE) displacements will be accommodated.

Safety Evaluation Summary: This modification reroutes and resupports the ICS steam line vent piping in order to reduce the cyclic stresses they experience. The modification will not have any adverse effect on nuclear safety or the environment.

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Modification: MOV Limit Switch Modification
SE 323616-001

Description of Modification: Institute of Nuclear Power Operations Significant Operating Experience Report (INPO SOER) No. 86-92 identified reliability concerns on inaccurate closed indication on Motor Operated Valves (MOVs). Motor Operated Valves use limit switches for control and position indication purposes. A limit switch assembly contains multiple electrical contacts grouped on rotors. The rotors are gear driven by the motor operator and are set so the contacts will operate at a desired setpoint in the valve's cycle. Rotor settings are adjustable and all the contacts on a rotor will either open or close at the established setting. Therefore, the options available for settings at various points in the valve's cycle are dependent upon the number of rotors in the limit switch assembly. Remote valve position indication normally consists of open and closed indicating lights. Only the open indication light will be ON when the valve is fully open. As the valve begins to close, contacts on the first limit switch rotor closes (switch #3, rotor #1) to turn closed indicating light ON. Both lights then remain ON until contacts on the second light switch rotor (switch #7, rotor #2) opens to turn off the open indicating light. The open indicating light going OFF is used to indicate the valve closed position. Contacts on the same rotor that turns the open indicating light OFF near the end of the closing cycle may also be used to provide torque switch bypass at the beginning of the opening cycle. This bypass prevents the torque switch from tripping the motor operator due to the high torque that may be necessary to unseat the valve and overcome system differential pressure. As the result of the Davis-Besse incident, the close-to-open bypass switch at Oyster Creek was set to within 20-25 percent of valve travel. Original setpoint was between 10-15 percent. This new setpoint provided additional assurance for the valve to overcome system differential pressure but also caused inaccurate closed indication. The valve may indicate closed while it is still partially open. Because control room operators may assume the valve to be fully closed, systems operational problems may occur.

Safety Evaluation Summary: In conclusion, it can be stated that this modification:

- Relocates the "open" indication switch to another switch within the operator compartment and readjust it to provide an accurate valve indication in the control room.
- Does not change the functions of the interfacing systems.
- Has no effects on plant effluents.

For the justifications described herein, it is concluded that the modification will not have any adverse affect on nuclear safety or safe plant operations, existence of an unreviewed safety question or a need for a Tech. Spec. change.

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Modification: SGTS Local Disconnects
SE 328293-001

Description of Modification: The Standby Gas Treatment System Orifice Purge Inlet Valves V-28-0024 and V-28-0028, and the Cross Tie valve V-28-0048 previously had pigtails on the associated solenoid valves and limit switches to provide electrical interface. These pigtails were butt spliced to the field cable in a conduit fitting at each of the six components. The butt splice must be cut out of the cable to remove the component for maintenance and/or replacement which results in shortening the cable each time this activity is performed.

This modification provides a means to locally disconnect and reconnect the component from the field cable without reducing the field cable length thereby enabling maintenance to be performed in a more timely manner. Once installed, the termination heads shall eliminate the need of requiring additional splices to extend the field cable back to an adequate length for the electrical interface to be accomplished in the local electrical fitting.

Safety Evaluation Summary: This modification is considered a corrective maintenance activity providing for ease of removal and installation of the associated components when required. This evaluation precludes the occurrence of an Unreviewed Safety Question or a Tech Spec change.

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Modification: Chlorination System Upgrade
SE 402792-001

Description of Modification: This safety evaluation provides and evaluates the safety impacts resulting from a modification to the Oyster Creek chlorination system. The modification provides a sodium hypochlorite storage and feed system to replace the existing Fischer & Porter liquid chlorine gas system. The primary purpose of the chlorine gas system is to control biofouling on the heat transfer surfaces of equipment cooled by the circulating, service, and emergency service water systems.

Safety Evaluation Summary: The modification will not have any adverse effect on the nuclear safety or the environment. This modification does not create an unreviewed safety question as described in 10CFR50.59.

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Modification: Emergency Diesel Generator Bldg. - Roof Panel Anchor Replacement
SE 323476-002

Description of Modification: The purpose of this modification was to replace the roof panel hold down bolts on the Emergency Diesel Generator (EDG) Building. The bolts have been significantly corroded due to water standing in the pockets around the bolts.

The anchor bolt hold down capacity has been evaluated and it has been determined that the number of anchor bolts on each panel can be reduced.

Safety Evaluation Summary: The new roof panel hold down bolt design is in compliance with all safety requirements, codes and regulations and will not affect the safety function or the environment. The replacement of panel anchor bolts was performed during plant shut down. Therefore, it is concluded that the proposed modification will not have any adverse effect on nuclear safety or the environment.

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Modification: Modification for Torus-to-Drywell Vacuum Breakers
SE 328294-001

Description of Modification: The purpose and scope of this modification was to modify the valve shaft/disc arm so that they move as one integral assembly and provide accurate disc position indication during monthly and refueling outage surveillance tests. The physical modification to implement the design modification is shown on MPR Dwg. 1083-139-01. It consists of two tapered pins with threaded ends to be inserted through the middle section of the shaft/disc arm assembly. Each pin will be secured with retained plate over it, attached to the disc arm by two cap screws with lock wires.

Safety Evaluation Summary: It is concluded that this modification will not affect the ability of the vacuum breaker valves to perform their safety function or their operational requirements. The valves operability is exercised monthly to assure that system performance does not degrade between refueling inspections. In addition, this modification will not affect the basic system function or requirement.

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Modification: FHAR Amendment for CRD Hydraulic System Flow Element
SE 402933-005

Description of Modification: The purpose of this safety evaluation is to support a change to the OCNGS Fire Hazards Analysis Report as well as a change to Technical Specification 3.12, for use of CRD Flow Indication FI-225-002 in lieu of FI-225-998. The installation of FI-225-002 is covered separately by Safety Evaluation No. 402933-006.

Safety Evaluation Summary: The purpose of this activity is to evaluate the use of FI-225-002 in lieu of FI-225-998 for fires which may damage Control Room Indication FI-RD36. Both local indicators fulfill the commitment to 10CFR50 Appendix R contained in the FHAR (Doc. 990-1746) to provide CRD flow rate instrumentation to monitor system performance when reactor water level is controlled using valve V-15-30 in the CRD bypass for make-up capability. Use of FI-225-002 requires less operator action.

Since local flow indication is utilized for Alternate Shutdown in the event of a fire in the Control Room Complex, a Technical Specification Change Request must be approved by the NRC prior to changing the T.S. Surveillance for CRD flow per Table 3.12-6 and Table 4.12-1.

The installation of FI-225-002 was evaluated in Safety Evaluation 402933-006. The use of this local indication to achieve and maintain safe shutdown in the event of a fire does not adversely affect that evaluation.

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Modification: Installation of Test Modules in the SDIV Level Circuit
SE 402953-002

Description of Modification: The purpose of this modification was to install test modules in the (SDIV) Level Circuit. This modification will permit testing to be accomplished, electrically, from the Control Room, helping to reduce radiation exposure during the required testing and surveillance. SDIV testing and surveillance result in significant radiation exposure to the instrumentation and controls technicians. The ALARA saving have been documented in the Technical Specification Change Request No. 195.

Safety Evaluation Summary: This modification to provide Foxboro Test Modules in the SDIV Level Circuit and reconfigure the level transmitters does not decrease the margin of safety as defined in the SAR or in the Technical Specification; does not increase the probability of occurrence or consequence of an accident or malfunction of equipment important to safety; does not create a possibility for an accident; and does not involve any radiological or environmental effluent. Therefore, this modification does not create any unreviewed safety question as determined by 10CFR50.59, and does not involve a potential environmental impact.

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Modification: Chemical Decontamination Support Modifications
SE 328265-004

Description of Modification: This modification supports the chemical decontamination effort for 13R with the performance of three tasks: the DWEDT tie-in, the RBEDT tie-in, and the CRD Flush/Vessel level modification upgrade.

Safety Evaluation Summary: It is concluded that the modification will not have any adverse effect on nuclear safety or the environment, and does not result in an unreviewed safety question as determined by the 10CFR50.59 evaluation.

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