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June 4, 1999
1940-99-20267

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
10 CFR 50.59(b) Report - June 1997 - December 1998

In accordance with 10 CFR 50.59(b), enclosed are the summaries of the changes to the Oyster Creek systems and procedures described in the Safety Analysis Report (SAR) for the period June 1997 to December 1998.

If any additional information is required, please contact George Busch at (609) 971-4643.

Very truly yours,

Michael B Roche

Michael B. Roche
Vice President and Director
Oyster Creek

Attachment

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

IE47

**Oyster Creek Nuclear Generating Station**

**50.59 Report**

**June 1997 - December 1998**

Document: **SE-000010-001, R0**, Revision to GE Drawing 846D640

FCN C-093012 documents revisions required to GE Dwg. 846D640, Rev. 0, "Heat Balance (3 Heater)," dated August 1964. Specifically, the feedwater flow, temperature and enthalpy parameters were changed to reflect their value at a licensed reactor power level of 1930 $MW_{th}$. Also, the schematic representation of the reactor water clean-up system has been revised by the subject FCN to correctly show that it is connected to the recirculation system.

The affected drawing was originally prepared to identify the gross and net heat rates at a reactor power level necessary to attain the full output of the turbine-generator. The basis for the 1950 $MW_{th}$ value is described in the historical Oyster Creek project file records. The temperature, pressure, enthalpy, and flow heat balance parameters applicable to the main steam, condensate, feedwater, and reheat systems represent conditions at approximately 1930 $MW_{th}$.

GE Dwg. 846D640 is a historical record that is categorized as "Information Only." The plant's true heat balance is routinely calculated by the plant computer which analyzes data inputs from installed plant flow, temperature, and pressure instrumentation. The changes implemented by this FCN do not constitute an Unreviewed Safety Question. This Safety Evaluation has been prepared because the affected drawing is referenced in Section 10.1 of the Oyster Creek FSAR.

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Modification: **SE-000104-009, R0**, Small Bore Piping Support

The purpose of this document was to evaluate the safety of modifying supports and tubing on small bore piping as detailed in CC-MD-H275-001.

The piping support modification brings the piping systems small bore piping into conformance with ASA B31.1.1955 code. In addition, the changes assure the integrity of the NSR pipe to which they are connected.

This evaluation demonstrates that this modification does not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10 CFR 50.59.

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Document: **SE-000106-004, R0**, As Found FCN C059307 - Emergency Stretcher Anchors

The change being evaluated is the mounting of anchor bolts to support new emergency stretchers in four locations in the plant: one in the 119' el. Reactor Building stairwell, one in the Warehouse, one in the New Radwaste Building, and one in the south extension of the Turbine Building. Only the New Radwaste Building and the Reactor Building locations are in a Class I structure, and the particular location (on the 119' elevation stairwell) in the Reactor Building is not an integral part of the Class I structure itself. Therefore there is no safety issue raised by the installation of these anchor bolts. The Reactor Building location is shown on drawing 3E-153-02DOIB (referred to in section 3.8 of the OC FSAR) and the Turbine Building location is shown in drawing 3E-188-02-001 (referred to in section 1.2.2.1 of the OC FSAR). This modification does not require a change to the Technical Specifications and does not involve an unreviewed safety question.

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Document: **SE-000120-001, R0**, FCN C053142

FCN C053142 requested that Site Plan drawings JC-19702 and GU-3E-120-01-001 be changed to show the domestic water facility, heli-port, hydrogen bulk storage facility and the Forked River combustion turbines. JC-19702 is referenced in Section 1.2.2.1 of the FSAR. The "activity" evaluated by this safety evaluation is the modification of JC-19702 to include the domestic water facility, heli-port, hydrogen bulk storage facility and the Forked River combustion turbines.

The intent of the Site Plan is to convey the approximate location of facilities on and in the general vicinity of the site. It does not depict plant systems or their manner of operation. Updating JC-19702 to include the domestic water facility, heli-port, hydrogen bulk storage facility and the Forked River combustion turbines does not change the design, function or manner of operation of any structures, systems or components.

This activity does not change any plant equipment, plant system, or plant technical specification. Existing margins of safety defined in the plant technical specifications or FSAR are not reduced as a result of this activity. Nor is the probability of occurrence or consequence of an accident previously evaluated in the FSAR affected by this activity. Therefore, there is no unreviewed safety questions resulting from this activity.

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Document: **SE-000120-002, R0**, FCN C073101

FCN 073101 requested that Site Plan drawing JC-19702 be changed to reflect the current configuration of the site's general layout. JC-19702 is referenced in Section 1.2.2.1 of the FSAR. The "activity" evaluated by this safety evaluation is the modification of JC-19702 to reflect the existing general site layout.

The intent of the site plan is to convey the approximate location of facilities on and in the general vicinity of the site. It does not depict plant systems or their manner of operation. Updating JC-19702 to reflect the site's existing layout does not change the design, function, or manner of operation of any structures, systems or components.

This activity does not change any plant equipment, plant system, or plant technical specification. Existing margins of safety defined in the plant technical specifications or FSAR are not reduced as a result of this activity. Nor is the probability of occurrence or consequence of an accident previously evaluated in the FSAR affected by this activity. Therefore, there is no unreviewed safety question resulting from this activity

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Document: **SE-000153-020, R0**, MNCR Expansion Joint Between Turbine and Reactor Buildings

The purpose of this safety evaluation was to assess the affects, if any, on nuclear safety of the nonconformances documented in MNCR 97-0005. The MNCR documented the presence of four conduits passing through the gap located in the south block wall of the Computer Room, and documented the size of the gap as 2 1/4 inches. Per Oyster Creek FSAR, these gaps must be free from interferences, and the design gap width is three inches minimum. Verification of the gap width between the Turbine and Reactor/Office buildings at other locations determined the minimum width to be two inches. This is contrary to the minimum design width of three inches, per Section 3.7.2.8 of the Updated OC FSAR.

Specifically, this document justifies the impact on nuclear safety of the two inch minimum as built expansion gap between the buildings and the presence of conduit in the gap.

Document: **SE-000153-020, R0 (Cont'd.)**

This existing condition alters the gap between the Reactor and Turbine buildings. The functions of the buildings are not affected. This safety evaluation has determined that this existing condition does not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) a malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the bases of any Technical Specification.

Since there is no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this existing condition is acceptable.

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Modification: **SE-000153-024, R0**, $CO_2$ Blast Decontamination Station

The purpose of this document was to evaluate the safety of installing a Mini Blast Contamination Station. This semi-mobile $C0_2$ decontamination booth station is located on the 75 feet elevation of the Reactor Building in the CRD rebuild room area. This unit will clean contaminated tools with a blast of $C0_2$ in the enclosed booth. Units similar to this are in use at nuclear facilities throughout the United States.

There will be no unmonitored radioactive release when this system is used. This evaluation demonstrates that this modification does not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10 CFR 50.59.

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Modification: **SE-000153-026, R1**, Condensate Transfer System Camera Installation

An 8 inch diameter core bore was drilled in the northwest passageway slab at EL 23 feet 6 inches, between the Turbine Building and the Reactor Building to allow for the insertion of a camera to monitor the Condensate Transfer Line.

GPUN Calculation C-1302-153-E310-099 demonstrates that core boring the slab has no affect on its design integrity. Sealing the opening after completion of the inspection in accordance with the requirements of the modification will effectively prevent the intrusion of ground water into the plant and the escape of any water that might accumulate on the slab to the environment. The area remains protected from natural phenomena.

As there is no impact on nuclear safety and/or safe plant operations, no creation of an unreviewed safety question and no environmental impact, this modification is deemed acceptable.

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Modification: **SE-000153-028, R0**, Shielding Over Surge Tank Cover

The purpose of this safety assessment was to document the structural capability of the east and west skimmer surge tank cover plates located on elevation 119 feet 3 inches of the reactor building to accommodate the weight of lead blanket shielding under normal plant and seismic conditions.

Modification: **SE-000153-028, R0**, Shielding Over Surge Tank Cover (Cont'd.)

The seismic classification of the reactor building is seismic class 1, the classification of the surge tank covers are also seismic class 1.

By demonstrating the structural capability of the surge tank covers, permanently placing lead blankets over the surge tanks does not degrade the performance of the plant safety systems during accident or normal plant conditions or, impose a safety concern on the plant. This safety evaluation determined that no unreviewed safety questions are created and that no environmental impact is involved. The addition of the lead blankets does not decrease the margin of safety as defined by the OC Technical Specification, violate any licensing requirements, cause a radiological safety concern, or affect the plant conditions.

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Modification: **SE-000154-002, R0,** High Radiation Door and Hatch in Old Radwaste Building

A new locked high radiation wire mesh door was installed in the Old Radwaste (ORW) Building mezzanine at the entrance to steps leading to the small pump room. A new locked high radiation grating over the floor plug entrance way to the tunnel located in the ORW Building large pump room was also installed. The Old Radwaste Building is classified as a seismic Class II structure and is no longer used for normal radioactive waste handling. It is only used for waste compaction and transfer during outages and for limited waste storage.

The installation of the looked door and grating allowed the removal of the high rad controls for the 3 door entrances into the ORW Mezzanine area. The addition of this door and grating plug requires a change to the FSAR due to their addition to drawings referenced in FSAR Section 1.2.2.

This safety evaluation has determined that this modification does not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Modification: **SE-000161-004, R0**, Notching of Fuel Pool Gate Gasket

The Maintenance Department found it difficult to install the fuel pool gate gaskets at the corners of the gate. The inner edge of the gasket tended to wrinkle. The aluminum retaining strips flattened the wrinkles on the inner edge but the outer edge would curl. In order to alleviate the problem the inner flat edge of the gasket was notched. The number of notches is limited to four (4) with the bottom of the notches to be rounded.

The modification does not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR because the modification has no adverse effect on the operation of the fuel pool gate gaskets. To the contrary, the modification allows the gaskets to operate more efficiently. The modification will not increase the probability of occurrence or consequence of a malfunction of equipment important to safety previously evaluated in the SAR because the modification has no adverse effect on the operation of the fuel pool gate gaskets.

The modification does not create a possibility for an accident or malfunction of a different type than any previously identified in the SAR because the modification has no adverse effect on the operation of the fuel pool gate gaskets. The modification does not affect any Technical Specification or licensing requirement. The modification does not involve a radiological safety concern because there is no change to plant function or operation.

Document: **SE-000161-006, R0**, Spent Fuel Pool Clean-Up Load Drops in the Spent Fuel Pool Procedure Change

The purpose of this safety evaluation was to evaluate the consequences of load drops inside the Spent Fuel Pool (SFP) that may occur during the Spent Fuel Pool Clean-Up project (SFPCP).

This safety evaluation only evaluates those load drops that may occur during the clean up of the SFP. This includes miscellaneous volume reduction equipment and disposal materials such as crushers/shearers, empty liners, test weights, baskets, liners with irradiated materials, control rod storage rack, local power range monitors (LPRM), source range monitors (SRM), intermediate range monitors (IRM), filters, miscellaneous small parts, containers, debris, etc.

This safety evaluation has determined that this modification does not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Document: **SE-000166-002, R0**, Storage of Non-Waste Radioactive Material

This safety evaluation addresses the storage of non-waste radioactive material at Oyster Creek. The locations involved are:

- The cleaned-out radwaste drum storage area in the Old Radwaste Building (ORW)
- The Scaffold Storage shed attached to the south wall of the New Radwaste Building (NRW),
- The Laundry Facility and Rad Material Trailer,
- The Rad Material Freight Containers east of ORW,
- The Rad Material Freight Container at the north wall of ORW,
- The Rad Material Freight Containers east of NRW,
- The Rad Material storage Freight Container Trailers north of the Low Level Radwaste Storage Facility (LLRWSF)

Storage of non-waste material for containers with less than 35 curies total activity at Oyster Creek is acceptable. The effect of the worst possible accident with the storage is within the plant operating guidelines of 10 CFR 20 and 40 CFR 190.10 (based on administrative controls), the activity will not affect any margin of safety, nuclear plant safety, plant operation, or any accident previously evaluated in the SAR. It will not effect important to safety equipment. It does not have the ability to create an accident not previously evaluated in the SAR. It has no interaction with any Technical Specification.

Document - **SE 000200-003, R0**, UFSAR Update (PFU 11-053)

On May 22, 1992, the USNRC issued GL 87-002, Supplement 1, which included the NRC Supplemental Safety Evaluation Report (SSER-2) on Revision 2 of the Seismic Qualification Utilities Group Generic Implementation Procedure (GIP). The GIP provides a detailed technical approach to verify the seismic adequacy of mechanical and electrical equipment installed in nuclear power plants to address concerns identified in Unresolved Safety Issue USI-A46. In SSER-2, the USNRC states that the GIP approach to resolution USI-A -46 provides an acceptable evaluation method for verifying the seismic adequacy of equipment in USI-A-46 plants including modifications and new and replacement equipment. The purpose of this Safety Evaluation was to assess a change to the OC FSAR to permit use of the SQUG methodology as presented in the GIP Revision 2 and supplemented by the USNRC SSER-2 as an alternate acceptable method to verify the seismic adequacy of existing, new, modified, and replacement equipment installed at Oyster Creek.

The USNRC has concluded in its Supplemental Safety Evaluation Report No. 2 that " the criteria and procedures described in the SQUG Generic Implementation Procedure and the NRC SSER No. 2, are determined to be an acceptable evaluation method for verifying the seismic adequacy of the equipment in USI-A-46 plants including future modifications and replacement equipment." The NRC also concludes that the criteria are acceptable for verifying the seismic adequacy of new equipment in USI-A-46 plants.

The incorporation of the SQUG methodology as an alternate method to demonstrate seismic adequacy does not increase the probability of consequences of an accident or malfunction of equipment, create the possibility of a malfunction of a different type than previously evaluated in the FSAR or decrease the margin as defined in the basis for the Technical Specification. The methodology as implemented at Oyster Creek is consistent with the existing FSAR requirements. Therefore, the change does not result in an unreviewed safety issue and the change can be implemented under 10 CFR 50.59.

Document: **SE-000200-005, R0**, EPG/SAM Revision 1 Changes to PSTG Sections RPV Control, Level Restoration and Level Power

This revision to the Oyster Creek Plant Specific Technical Guidelines (PSTGS) is made to:

1) Incorporate several changes that have been made to the BWROG Emergency Procedure Guidelines (EPGs) on which the PSTGs and Emergency Operating Procedures (EOPs) are based;

2) Make technical changes that are plant specific to Oyster Creek or are required to enhance the function/use of the Emergency Operating Procedures, or

3) Correct non-technical errors in the PSTGS.

These changes are being made to the Introduction, Cautions, RPV Control Level Restoration and Level/Power Control sections of the PSTGS.

The specific changes made include:

- In the Introduction section, a non-technical change is made to reflect the latest structure for the Engineering groups of GPUN.

- Caution 3 has been deleted since it provides authorization for exceeding normal operating limits rather than warning of a potential hazard. The affected steps have been reworded to direct that the prescribed actions be performed "irrespective of the resulting cooldown rate." The steps affected are:

  1) 2nd. Conditional Statement before Step RC/P-1
  2) 1st. and 2nd. Conditional Statements before Step RC/P-2

  This is consistent with EPG/SAG changes per EPG Issue 9324 of the BWROG Emergency Procedures Committee (EPC).

- New caution #4 has been added to emphasize that reducing Primary Containment pressure will reduce available NPSH for pumps drawing suction from the Torus. It is referenced in the steps defining action for using Containment Sprays in the Primary Containment Control Guideline. Since operating Containment Sprays will reduce Primary Containment pressure, Core Spray pump NPSH requirements should be considered when Containment Sprays are operated. This change is identified in EPG Issue 9705.

- New caution #5 has been added to stress the concern for Primary Containment integrity during ATWS conditions. The relief capability of the EMRVs is only 45% after which the Safety Valves will lift unless the ability to use the Main Condenser is maintained. Primary Containment heatup is significant even when only the EMRVs are being used to provide heat removal from the RPV, therefore it is important to maintain the Main Condenser as a heat sink as long as possible during an ATWS. This caution is being added per operator request to ensure that the importance of keeping the MSIVs open and the Main Condenser available is recognized when the Level/Power Control procedure is entered. This caution is referenced upon entry into the Level/Power Control Guideline. This is an Oyster Creek specific change being made to enhance the use of the EOPS.

This safety evaluation has determined that this change does not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Document: **SE-000200-006, R0**, Emergency Procedure Guidelines/SAM Revision 1 Changes to Plant Specific Technical Guidelines Section of Steam Cooling

This revision to the Oyster Creek Plant Specific Technical Guidelines (PSTGS) was made to:

1) Incorporate several changes that have been made to the BWROG Emergency Procedure Guidelines (EPGS) on which the PSTGs and Emergency Operating Procedures (EOPS) are based;

2) Make technical changes that are plant specific to Oyster Creek or are required to enhance the function/use of the Emergency Operating Procedures, or

3) Correct non-technical errors in the PSTGS.

These changes were made to the Steam Cooling section of the PSTGS:

A. The Conditional Statement after Step C3-1 has been reformatted into 3 separate Conditional Statements and moved to before Step C3-1. This change was made for clarity purposes and was moved because the Conditional Statements apply to Step C3-1 also. This change is identified in EPG Issue 9303.

B. The 3rd. Conditional Statement before Step C3-1 has been changed to allow the operator to attempt to restore and maintain water level above the Minimum Steam Cooling RPV Water Level of -30 in. when any injection source has been recovered. Rev. 4 of the EPGs directed an immediate Emergency Depressurization when an injection source was recovered. With the changes based on the latest revision to the EPG/SAGs, the operator is directed to attempt to restore and maintain RPV level before resorting to Emergency Depressurization. Studies by GE for the BWROG have demonstrated that adequate core cooling with injection can be established with RPV water levels at or above -30 in. TAF.

Step C3-3 requires Emergency Depressurization when RPV water level drops to the Minimum Zero-Injection RPV Water Level (MZIRWL) of -42 in. TAF. Thus, only a range of approximately one foot (-30 in. to -42 in.) exists where steam cooling of the core is not guaranteed to be adequate when water is being injected into the RPV. If an RPV injection source becomes available during this time, it is appropriate to attempt to restore level above -30 in. TAF. If RPV water level cannot be restored and maintained above -30 in- TAF, Emergency Depressurization is required for the following reasons:

1) The calculation for the MZIRWL assumes that there is no subcooling at the core inlet. Since any injection would invalidate this assumption, adequate core cooling cannot be assured if RPV water level is below the MSCRWL and water is being injected into the RPV.

2) Depressurization will reduce any break flow and maximize any injection.

3) Depressurization reverses the heatup of the upper core region by increasing the steam flow through the fuel bundles and quenches the fuel rods through the accompanying RPV water level swell.

This safety evaluation has determined that this change does not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

This revision to the Oyster Creek Plant Specific Technical Guidelines (PSTGS) was made to:

1.) Incorporate several changes that have been made to the BWROG Emergency Procedure Guidelines (EPGS) on which the PSTGs and Emergency Operating Procedures (EOPs) are based;

2) Make changes to incorporate commitments made in response to GL 96-06 with regard to isolation of the RBCCW System during LOCA conditions;

3) Make changes to incorporate the complete strategy for preventing the loss of Net Positive Suction Head (NPSH) associated with ECCS Suction Strainer Plugging;

4) Make technical changes that are plant specific to Oyster Creek or are required to enhance the function/use of the Emergency Operating Procedures; or

5) Correct non-technical errors in the PSTGS.

The specific changes made include:

- The Entry Condition for Torus Water Temperature has been changed to match the EPG entry point of "the most limiting Torus water temperature LCO" of 95 degrees F. This is an increase from the alarm setpoint of 90 degrees F. This change is made due to the fact that the Torus temperature, because of environmental conditions in the summer months, can be above 90 degrees F. This change is made to prevent unnecessary entry into the EOP for normal plant conditions. The temperature control point in the Torus Water Temperature Control section is also increased to the most limiting Torus water temperature LCO of 95 degrees F for the same reason. In the past, Oyster Creek had taken a deviation from the Revision 4 EPGs as a conservative measure. This change will put Oyster Creek in compliance with EPG Revision 4 and the EPG/SAG Revision 1 and with the limits of the Oyster Creek Technical Specifications.

- Changes were made to incorporate the complete strategy for Core Spray/Containment Spray Suction Strainer Plugging. These changes include statements in the Torus Water Temperature, Drywell Temperature and Primary Containment Pressure Control sections, which limit Containment Spray Pump operation to maintain adequate Core Spray NPSH. This was added to ensure that the priority for operating systems taking suction from the Torus is clearly understood. In addition, this change will ensure that the operators have the guidance necessary to ensure that the systems are operated within the suction strainer design limits. This statement is not being added to the operation of the Containment Spray System in the PC/H section of the guidelines because at this point the Primary Containment is in danger of failure from a hydrogen deflagration. Because protecting the Primary Containment takes precedence over core cooling, Containment Sprays are used to reduce pressure in the Hydrogen Control Guideline even if adequate Core Spray NPSH cannot be maintained.

- Oyster Creek has installed new suction strainers in response to the ECCS Suction Strainer Plugging issue. The new design requires the use of Primary Containment pressure in excess of one atmosphere to maintain adequate Net Positive Suction Head (NPSH) while the Core Spray System is used with Booster pumps in operation. The design assumes an overpressure of 1.25 psig while the booster pumps are in operation following the LOCA. When the Booster pumps are no longer in operation, the NPSH analysis assume 0.0 psig in the containment. The value used in the EOP to address the 1.25 psig requirement will be 2.0 psig, which ensures that the 1.25 psig value is not violated and is readable from the Control Room indications. The overpressure requirement has been evaluated separately in SE-315403-040. To allow the flexibility to minimize containment pressure below 2.0 psig later in the event, the procedure will allow the operators to trip the Drywell Sprays at 1.0 psig when booster pump operation has been terminated. Therefore, the guidance in the Drywell Temperature, Primary Containment Pressure and Hydrogen Control procedure is changed to direct the securing of Drywell Sprays when pressure drops to 2.0 psig if Core Spray Booster pumps are running or 1.0 psig if Booster pumps are not running. Within the design basis of the plant, core spray booster pumps can be secured after 30 minutes without endangering the Appendix K analysis results.

  This change in containment pressure value used to trip Containment Spray pumps is consistent with the strainer design analysis. In addition, since the operators are allowed to minimize containment pressures within 30 minutes of an accident, the change will not impact those analyses associated with containment post accident off site release.

- Changes were made to the Drywell Temperature Control, Primary Containment Pressure Control and Hydrogen Control Guidelines to incorporate the Oyster Creek response to NRC Generic Letter 96-06 regarding overpressurization of isolated piping systems and water hammer failure after the system is reinitiated. The guidance on RBCCW control in these sections has been changed to confirm isolation of RBCCW upon initiation of Drywell Sprays. The need to operate Drywell Sprays would indicate a LOCA type condition that could result in a failure of the piping in the Drywell. RBCCW is not needed at this point so no negative ramifications result from this isolation. In addition, the Drywell Temperature Control section is being changed to determine if RBCCW has been isolated by a LOCA or Main Steam Line Break before Drywell cooling is maximized. If an isolation has occurred, then Drywell cooling is left secured and other means of cooling the Drywell (i.e. Drywell Sprays) are used.

- The point at which the RPV is Emergency Depressurized due to bulk Drywell temperature has been changed to when temperature cannot be restored and maintained. This change was made to allow operators time to initiate Drywell Sprays in response to LOCA conditions in the Primary Containment. Under several postulated above core SBLOCA conditions, the temperature in the containment can be expected to rise quickly to a point approaching or exceeding 281 degrees F. Since the Containment Spray System is a manual system, the operator will take a finite period of time to initiate Drywell Sprays in accordance with the Drywell Temperature Control guidelines. The new wording will allow the operators an opportunity to restore Drywell Temperature below 281 degrees F before initiating an Emergency Depressurization of the RPV.

Failure to restore drywell temperature below the specified value may result in a challenge to components required for accident mitigation. The components of concern at elevated temperatures are the EMRVS. Testing has shown that these valves will be able to perform their intended function at or above 281 degrees F unless a significantly degraded DC voltage exists. The function provided by the EMRVs is also available via the Isolation Condenser, which can be used to quickly depressurize the vessel per the Emergency Depressurization Guideline. Once Drywell Sprays have been established, temperature in the Drywell will be quickly reduced to the point where EMRV availability due to temperature is not of concern. The short period of time (less than 1 minute) required for the operators to initiate sprays to restore Drywell temperature is not sufficient to represent a significant threat to the EMRV capability.

- Generically throughout the Primary Containment Control Guideline, reference to water level in the Torus or Drywell has been changed to Primary Containment when discussing the levels of the Primary Containment vent points to provide consistency throughout the guideline. This is a change from the latest revision of the EPG/SAGs.

- In the Torus Water Level Control section, reference to the Heat Capacity Level Limit has been removed. Since the Heat Capacity Temperature Limit is now defined as a function of Torus water as well as RPV pressure and Torus temperature, a separate Heat Capacity Level Limit need no longer be defined. This change is identified in EPG Issue 9307.

- Several definitions have been added to the action points in the Primary Containment Control Guidelines including the definition of 110 inches in the Torus as the elevation of the Drywell to Torus downcomer openings. These and other wording changes are made to clarify systems and action points in the guidelines.

Implementation of these changes did not reduce the margin of safety as defined in the basis for any Technical Specification. The changes to the PSTG and thus to the Emergency Operating Procedures discussed above act to clarify required actions or enhance the response to accident conditions. The guidelines use the installed safety equipment and non-safety equipment to mitigate accident conditions. These changes do not change the methods used to operate, or the limits of, any equipment already used the Oyster Creek Emergency Operating Procedures and included in the OC Technical Specifications.

This safety evaluation has determined that this change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this change is acceptable.

Document: **SE-000211-021, R0**, As Found FCN 101048 Emergency Condenser Drain Line

The specific change evaluated was a re-positioning of the depiction of the drain line and associated valves (V-14-0025/0038) on the return line from the tube side of Emergency Condenser NE 01-A as shown on GE 148F262, the system flow diagram shown in the OC FSAR. The line and associated valves provides drainage capability to the Reactor Building Equipment Drain Tank, the valves are normally closed.

The implementation of the evaluated drawing change does not adversely effect either nuclear safety or safe plant operation because the function of the system as described above has not been changed. The function of neither the drain line nor the rest of the system was effected by the change, nor did the change effect operating procedures.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report have not increased because the system whose drawing was modified performs no safety function in the areas modified (the return line drain) and the change has no impact on any other system which has a safety function.

The possibility for a malfunction or an accident of a different type than any evaluated previously in the Safety Analysis Report has not been created. The changes documented do not impact the design and operation of the system such that the safety functions of any component or system would be challenged.

The margin of safety as defined in the basis for any Technical Specification has not been reduced because the portion of the drawing modified addresses a portion of the system which has no safety function and is not included in the basis of any Technical Specification. Furthermore, the requisite drawing change does not impact the portion of any system defined in the basis of the Technical Specifications.

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Document: **SE-000212-038, R0**, FSAR Update, Chapter 6

The purpose of this document was to evaluate the effect on safety of the following changes to the FSAR:

- Revise the maximum opening time of the Core Spray System outside isolation valves as reported in FSAR Table 6.3-4. The value was listed in this Table as 22 seconds maximum. The revised value for the opening time is 24 seconds, maximum.
- Delete the stroke times for the main Core Spray pump suction valves, which were listed as 60 seconds.
- Revise the wording associated with the time delay starting of the back-up main and booster Core Spray Pumps. The FSAR should reflect nominal actuation value as each time delay setting is subject to tolerances.

This safety evaluation has determined that this change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Document: **SE-00212-039, R1**, FSAR Update, Change of EMRV Flow Rate (PFU 10-213)

The rated EMRV capacity documented in the Updated FSAR was inconsistent. Based on revised vendor documentation a recent Technical Specification Amendment No. 177 has established a standard capacity of 602,900 lb/hr at 1250 psig. Its impact on plant licensing basis was evaluated in C-1302-212-E610-104. As a result, all references of the EMRV capacity in the FSAR have been changed to be consistent with the correct information. This SE justifies the FSAR Update PFU 10-213.

With a changed EMRV flow rate to be consistent with the Amendment 177 submittal, documentation in the FSAR is hereby updated. Consistent rating capacity is provided in all sections in the FSAR. The shape and PCT for the analyzed small break curves is slightly different, but the over-all margin to safety and plant safe plant operations are not adversely affected. There were no unreviewed safety questions.

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Procedure Change: **SE-000212-046, R0**, Core Spray System Main Pump Operability Criteria

This document evaluated the effect on safety of changing the Core Spray Pump Operability Surveillance Procedure. The proposed change covered testing of the main Core Spray pumps without booster pumps in service and specifies the acceptance criteria for each main pump.

This evaluation has determined that the proposed change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety evaluated previously in the SAR, (3) create the possibility for an accident or malfunction of a different type then any previously identified in the SAR, or (4) decrease the margin of safety as defined in the basis of the Technical Specifications,

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Document: **SE-000212-047, R0**, Plant FSAR Update

The purpose of this change to the FSAR was to allow the discharge/test valve motor controllers for the Core Spray system to be racked in and locked off during plant operation. The controllers' description in the FSAR is racked out. The reason these valve controllers need to be disabled is to prevent a single failure from disabling the core spray system. This was found in 1975 when the core spray system single failure proof modifications were performed.

Based on the above discussion, this change to the FSAR did not pose an unreviewed safety question and does not adversely affect nuclear safety or safe plant operations. This change does not have an environmental or radiological concern. This change to the FSAR will change the description of the status of the core spray discharge/test valves from racked out to locked off.

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Maintenance: **SE-000212-048, R0**, FY-212-845 Replacement

This evaluation analyzes the condition of losing power to six current converters (i.e., LY-211-847, LY-211-848, FY-212-845, FY-212-846, FY-241-843, FY-241-844) used in level and flow loops for the Isolation Condenser, Core Spray and Containment Spray Systems. Power was interrupted during the maintenance activity to replace the current converter for Core Spray System 1 flow indication instrumentation.

Maintenance: **SE-000212-048, R0**, FY-212-845 Replacement (Cont'd.)

The current converter for Core Spray System 1 flow indication, FY-212-845, was drifty and requires replacement. The converter was hard wired into the system and supplied with 120 VAC power from Panel IP-4, breaker #15.

In order to safely replace this converter, technicians interrupted power to all six converters by lifting upstream leads. The power leads connected to FY-212-845 were then disconnected from the converter, line to line was terminated and insulated and common to common was terminated and insulated. Technicians then land the lifted upstream leads, providing power to the five remaining converters, until the deenergized, defective converter is replaced.

A second deenergization occurred for technicians to reinstall the lifted and insulated terminals to the new converter. Each period of power loss was estimated to be 10 minutes or less in duration.

The total period that the Core Spray System 1 converter was out of service was less than 2 hours.

By the foregoing evaluation it is demonstrated that this activity did not adversely affect nuclear safety or the environment. No unreviewed safety question was generated by this activity

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Document: **SE-000212-056, R0**, Design Basis for EMRV Solenoid Actuators

The purpose of this document was to provide a Design Basis for the Electromatic Relief Valve (EMRV) solenoid actuator based on the various modes of operation for the EMRVs. This document defines each mode of operation/use of the EMRVs, and is used to document the conditions where an EMRV is expected to be operable under each mode.

The bounding cases for environmental conditions, actuation sequences and valve cycling times were determined from analyses already in place at Oyster Creek. The single failure and loss of off-site power criteria were used as appropriate to ensure the most limiting case of 125 VDC System availability was included in the Design Basis.

This document enhances nuclear safety and safe plant operation because it provides a basis for ensuring that the EMRV solenoid actuators are installed and maintained such that they are operable for all required modes of operation.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Document: **SE-000213-009, R1**, Liquid Poison Design Temperature

Several document discrepancies were discovered and documented in a Deviation report. In the course of responding to the Deviation Report, an error was discovered in the FSAR. The discrepancies were resolved and a correction to the FSAR was made. The Flow Diagram, Line List and ISI Boundary drawing were revised to resolve the discrepancies. This safety evaluation has determined that the Liquid poison system and the storage tank are within their design basis. Based on the results of this evaluation, there was no unreviewed safety question.

Document: **SE-000213-010, R0,** SLCS Reactivity Requirements in FSAR

A review of the FSAR, performed as part of a GPUN commitment to the NRC, revealed two statements about the standby liquid control system (SLCS) that are no longer current. These statements were revised to reflect the current design basis of the plant. Both statements deal with the reactivity requirements for the SLCS.

Section 9.3.5 of the FSAR described the SLCS and indicated that a 4.0% shutdown margin was used for the SLCS system design. This is a value that was in the FDSAR and was later changed to 1.0% design margin for the SLCS system. The 1.0% design value was reestablished for cycle 12 with the submittal of GPUN methods for reload analysis. The FSAR was been updated to reflect the change.

The FSAR description for the SLCS was revised when the system was modified to use enriched B-10 instead of natural B-10 to address 10 CFR 50.62 (ATWS) requirements. Section 4.3.2.4 on control requirements contained a statement on the adequacy of the SLCS system to deal with the reactivity addition of xenon burn. The numbers in this section are not supported by any current analysis.

This SER was written to implement the revision to the FSAR.

The changes to the FSAR are consistent with previously approved revisions to the plant design basis. The FSAR statements identified were no longer current and the changes reflect the current plant design basis. Since the changes to the design basis were previously reviewed and approved by the NRC or under 10 CFR 50.59, the changes do not constitute an unreviewed safety question nor does it adversely impact nuclear safety or safe plant operation.

Document: **SE-000214-012, R0**, Evaluation of Downgrade of QCL Items

The items considered in this evaluation are classified either Regulatory Required (RR) or "Other"; therefore, they do not perform a safety-related function. However, items associated with regulatory commitments may interface with plant components such that safe plant operation may be impacted, although their failure to perform as designed will not affect nuclear safety. In order to ensure that these items perform as designed, the QCL classification imposes certain quality requirements from the standpoint of procurement; receipt inspection, installation, testing, maintenance and calibration. All these requirements are imposed on the item if it is classified as RR. For those items that were upgraded back to RR, safe plant operation (with regard to material quality and activities associated with material quality) is assured. An even more conservative approach was taken with some items; although not administratively required to be reclassified as RR, they were returned to their RR classification based on reliability considerations, complexity, uniqueness, etc. Those items that remained classified as "Other" are considered satisfactory based on their lack of regulatory commitments, impact on components associated with safe plant operation and degree of material control necessary to maintain equivalency.

This change did not require a change to the Technical Specifications and did not involve an unreviewed safety question.

Modification: **SE-000214-014, R0**, FCN C074434

FCN C074434 addressed implementation of a modification to the shutdown cooling pumps. The specific change was the removal of the PACLOC reservoir and its associated piping from shutdown cooling pumps P-17-001, -002, and -003. The inlet and outlet ports in the seal flanges have been plugged with SS pipe plugs.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Document: **SE-000215-009, R0**, Cleanup Demineralizer System (FCN C094302

The purpose of this evaluation was to review the impact of the as-found condition reported by FCN C094032. Reactor Water Cleanup System (RWCU) flow diagram GE 148F444, Sheet 2, Revision 46, showed the Electro-Chemical Potential Monitoring System (ECPMS) tie-in point to the RWCU downstream of relief valve V-16-76. The tie-in point as depicted on construction drawings and confirmed by walkdown, is actually upstream of V-16-76. At the time the FCN was issued, flow diagram GE 148F444 appeared in the FSAR. Revision of the flow diagram to correct this configuration oversight constitutes a change to the FSAR.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no environmental impact and no unreviewed safety issues, this change is considered acceptable.

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Modification: **SE-000215-017, R1**, Isolation Valve V-16-2 HELB Modification

This safety evaluation evaluated modification OC-CCD-000215-002 which allowed opening V-16-2 under high energy fluid system conditions with V-16-1 also open. The modification:

1) Bypassed the close torque switch. The existing switch setting was based on a much lower differential pressure. As a result of bypassing the torque switch, full operator capability is available which will ensure the valve isolates a RWCU HELB.

   Energization of the close starter (bypassing the close torque switch) was accomplished by tapping the open light indication, at contact 9, and placing it in series with either a spare contact off of 6K22 relay (Lo-Lo Level or Hi Drywell Pressure) or a bypass plug (control room cabinet 3F). The valve is controlled by the open light indication (Contact 9), which was set to open at 99.6% of electrical stroke.

2. Bypassed the TOL Heater. This optimized the available operator torque by reducing voltage drop to the motor.

   By accepting operation of the RWCU systems in the manner described (i.e., V16-2 open while V-16-1 is open under high energy fluid system conditions) results in a change in the valve design basis. The safety implications of this design are included in this safety evaluation.

Modification: **SE-000215-017, R1**, Isolation Valve V-16-2 HELB Modification (Cont'd.)

The modification of V-16-2 did not adversely impact Nuclear Safety. The MOV, as modified, is capable of isolating a HELB in the appropriate time frame, in accordance with its new GL 89-10 Design Basis. Jumpering of the subject valve's TOL heaters during plant operation ensures the proper available voltage to the motor and does not jeopardize the valve's safety function. Qualified personnel shall be stationed at the local starter during planned MOV strokes with instruction and capability of interrupting motor current. For the justification described herein, it is concluded the subject evaluation does not have any adverse effect on Nuclear Safety or Safe Plant Operations or the environment. This modification does not constitute an unreviewed safety question as determined by 10CFR50.59.

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Document: **SE-000215-019, R0**, Field Change Notice

FCN C073484 requested that Reactor Cleanup Demineralizer System flow diagram GE 148F444 be changed to indicate that V-16-008 is a one inch rather than a three-eighths inch valve. GE 148F444 is referenced in Section 5.4.8.2 of the FSAR.

V-16-0086 isolates the Filter Air System from the inlet to the cleanup filters and, as a consequence, is either fully open or fully closed. In this isolation capacity, the larger valve size will not adversely affect system performance. The temperature and pressure design ratings of V-16-0086 encompass normal system operating conditions and are consistent with the design rating of other components in the system. Installation of the larger valve will not impede the cleanup system's ability to remove impurities from the reactor water. Based on the above, replacing V- 1 6-0086 with a one inch valve did not change the function or manner of operation of any structures, systems or components. This replacement did not involve any unreviewed safety questions nor a change to the Technical Specifications.

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Alternate Replacement: **SE 000215-020, R0**, Alternate Replacement Parts for V-16-1, 2, 14, 61

This alternate replacement replaced valve internal components with improved, engineered functionally equivalent components (an alternate replacement).

As a result of this alternate replacement, the valves are now considered "predictable" because a friction value can now accurately be determined and thus a reduced valve force can be used to isolate a HELB. This enables the valves to be setup to close for design basis accidents.

This alternate replacement did not adversely affect nuclear safety or safe plant operation, did not create new accidents or malfunctions of equipment, did not increase any probability of an accident or consequences of an accident and does not decrease any margin of safety. This replacement did not involve any unreviewed safety question nor require any changes to the Technical Specifications.

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Document: **SE-000215-021, R0**, FCN C094523

This safety evaluation addressed the plant configuration described in Field Change Notice C094523. The subject change notice concerns the Reactor Cleanup Demineralizer System and requested that GE 112C2556 and GE 148F444 be revised to indicate the removal of local pressure indicator PI-16-239. The instrument is located downstream of non-regenerative heat exchanger outlet control valve PCV-ND11 and is isolated from the system by root valve V-16-1094. The revised drawings show PI-16-239 removed, V-16-1094 left intact in closed position and the line capped.

It is concluded that the subject change did not have any adverse effect on nuclear safety, safe plant operation or the environment. This modification did not involve any unreviewed safety questions as defined by 10 CFR 50.59. No change to the Technical Specifications were required.

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Modification: **SE-000215-025, R1**, RWCU HELB Detection and Isolation

This modification installed a safety grade break detection/isolation system that monitors RWCU pump room temperature and initiate RWCU System isolation when ambient temperature exceeds a preset limit. This modification also removed electrical bypass of Thermal Overload (TOL) contacts for valves V-16-1 and V-16-61 added by previous modifications, for valve V-16-14 added by modification and for valve V16-2 added by modification.

This modification introduced an additional safety grade signal to isolate the clean-up system when a HELB in the system occurs. It is concluded that the proposed modification did not constitute an unreviewed safety question as determined by 10 CFR 50.59, and does not have any adverse effect on nuclear safety, safe plant operation or the environment.

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Modification: **SE-000215-031, R0**, Replace Valve Body on ND-25B

The purpose of this document was to evaluate the safety of replacing the valve body to ND-25B in the Reactor Water Cleanup System. The existing cast steel valve body contained a gouge in the seat that was not repairable. A bronze valve body, with threaded ends was installed in its place. This is documented in OC-MD-H281 -001.

This modification did not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10 CFR 50.59.

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Modification: **SE 000215-033, R0**, V-16-1 Body Replacement

The purpose of this evaluation was to determine the acceptability of replacing the valve body on V-16-1 with a valve body that was purchased as a mockup for V-16-62.

The body replacement has no affect on the valve and the valve will function in the same manner as previously analyzed.

The replacement body did not adversely affect nuclear safety or safe plant operation, it does not create new accidents or malfunctions of equipment, it does not increase any probability of an accident or consequences of an accident and it does not decrease any margin of safety. The replacement body does not involve an unreviewed safety question and no changes to the Technical Specifications are required.

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Procedure: **SE-000221-006, R0**, Reactor Vessel Closure Head Stud Tensioning Procedure

The requirements for operational bolt-up stud elongation was .044 inches nominal with tolerances of ± .002 inches for each individual stud and ± .001 inches as the average for all 64 studs. These requirements are considered very difficult to achieve requiring potentially unnecessary work in a high radiation field. The stud elongation criteria was specified in Combustion Engineering Instruction Manual for Oyster Creek Reactor Vessel. The basis for the "tight" tolerance is unknown. The new requirements for operational bolt-up stud elongation is .044 inches nominal (no change) with tolerances of ± .003 inches for each individual stud and ± .002 inches as the average for all 64 studs.

This change had insignificant impact on flange membrane stresses and head stud fatigue usage factor. Therefore, the procedure change to relax the reactor closure head stud elongation tolerances is acceptable.

This safety evaluation has determined that the proposed change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Furthermore, this safety evaluation determined that no unreviewed safety questions have been created and no environmental impacts are involved. This change did not violate any licensing requirements, cause any radiological concerns and does not affect the plant's permit condition.

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Modification: **SE-000222-002, R0**, Core Plate Wedge Installation

The core plate wedges were installed to address potential degraded conditions in the core Plate assembly (as discussed in BWRVIP-25). Specifically, the wedges were installed to provide redundant lateral support for the core plate assembly to ensure lateral alignment of the core plate and insertion of the control rod drives (CRDs). Lateral support for the core plate is normally provided by 36 hold-down bolts (as well as by alignment cams and jacking screws). Although no damage has been found or reported on these bolts, installation of the wedges provides a fully redundant support mechanism for the core plate and eliminate GPUN's need to inspect the hold-down bolts (as required by BWRVIP-25).

As a result of the above, it has been demonstrated that the proposed core plate wedge installation:

1) Did not reduce the margin of safety as defined in the SAR or in the bases of any Technical Specification,

2) Did not increase the probability of occurrence or the consequences of:

   - An accident previously evaluated in the SAR,
   - A malfunction of equipment important to safety, or
   - An accident or malfunction not previously identified.

3) Did not violate the plant technical specifications or other licensing requirements or regulations, and,

4) Did not involve radiological safety concerns.

As a result, installation of the core plate wedges does not involve an unreviewed safety question and does not adversely affect nuclear safety or safe plant operations.

Modification: **SE-000223-014, R0**, Routing of Batch Tank Decant Hose

The Batch Tank decant hose was routed from the Batch Tank to the Ultra Filtration (UF) Holdup Tank (SLT-009). Temporary Modification 97-11 (TM) rerouted this hose to the floor drain in the NRW Fill Aisle when the UF skid was out of service, while waiting for replacement filters to be installed. During this time, Plant Chemistry kept track of the decanted liquid to see if it would have any impact on Radwaste Operations. It was determined there was no adverse impact on the chem waste filters and that there would be a positive impact on UF filter skid, i.e. the new UF filters will last longer, and on the Radwaste Shipping Department, i.e. would not have to process as much water through the UF skid. This TM was so successful it has been determined to make it permanent.

The modification did not: 1) adversely affect nuclear safety and safe plant operation, 2) increase the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, 3) create the possibility for an accident or malfunction of a different type than any previously identified in the SAR, or 4) reduce the margin of safety as defined in the basis of any Technical Specifications. Furthermore, no unreviewed safety questions have been created and no environmental impacts are involved. This modification does not violate any licensing conditions, cause any radiological concerns, and will not affect the plant's discharge permit condition.

Modification: **SE-000225-026, R0,** CRD Pumps Start Time Delay

The purpose of this safety evaluation was to document the basis for utilizing 60 second time delay in starting the CRD feed pumps on loss of offsite power (LOOP) condition. Presently, 60-second time delay is applied to these pump motors during Loss of Coolant Accident (LOCA) with a LOOP. The time delay permissive replaces the instantaneous start by removing the core spray system logic contacts.

This change utilized the existing 60-second time delay in starting the CRD pumps following a LOOP event. This is consistent with FSAR Table 8.3-1. CRD pumps are not needed in this 60-second time period to assure shutdown of the reactor or to maintain it in safe shutdown condition. The 60 second time delay in starting the CRD pump motor does not adversely affect the ability to prevent or mitigate the consequences of an accident to within 10 CFR 100 dose limits. Based on this safety evaluation, it is concluded that there are no unreviewed safety questions associated with this activity.

Temporary Modification: **SE-000225-032, R0**, Freeze Seals on HCU 101 & 102 Valves per 2400 GMM-3225.52

This evaluation involved isolating the Control Rod Drive (CRD) System Hydraulic Control Units (HCU) isolation valves 101 and 102 from the reactor by means of freeze seals while the 101 and 102 valves are opened up for overhaul. During refueling outages it may be necessary to perform overhauls on some HCU 101 and 102 isolation valves. These valves cannot be isolated from the CRD drives, which are in direct communication with the reactor vessel, without use of freeze seals. The freeze seals become the Reactor Coolant (RCS) System pressure boundary during this maintenance activity

This activity was implemented with the RPV depressurized vented to the atmosphere and maintained in a cold condition. Freeze seal(s) barrier served as reactor vessel pressure boundaries while up to six sets of Hydraulic Control Unit 101 and 102 valves were overhauled.

Temporary Modification: **SE-000225-032, R0**, Freeze Seals on HCU 101 & 102 Valves per 2400 GMM-3225.52 (Cont'd.)

This repair activity did not (1) adversely affect nuclear safety and safe plant operation, (2) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, (3) increase the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR, and (4) decrease the Margin of Safety as defined in the bases of any Plant Technical Specification. Furthermore, this Safety Evaluation determined that the repair activity did not involve any non-nuclear environmental concerns. This repair activity did not violate any licensing requirements, cause any radiological concerns and did not affect the conditions of the plant's discharge permit.

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Modification: **SE-000225-033, R0**, V-305-0104 Freeze Seal (JO 521866)
Modification: **SE-000225-034, R0**, V-305-106 Freeze Seal, (JO 521869)

The installation of a freeze seal on the HCU riser above the 104 and 106 valve did not adversely affect nuclear safety or safe plant operations. Freeze sealing of stainless steel piping is common practice and has been found not to damage the pipe. Physical measurements and Non-Destructive Examination (NDE) will be taken in accordance with the freeze seal and the HCU maintenance procedures to confirm that the pipe maintains its as-found physical properties.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification was acceptable.

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Configuration Change: **SE 000232-033, R0**, Chem Waste Sludge Transfer

In the past when sludge was removed from any of the Chem Waste Collection Tanks, an air driven pump was placed in the Chem Waste Tank Valve Operating Area on the 23' 6" elevation of the NRW Building. This was an infrequently performed operation that occurred maybe once or twice a year. With the installation of the Chem Waste Tank Side Suction Modification, OC-CCD-328373-001, this operation b performed more frequently.

This configuration change establishes the new location of the pump and it's connection to the existing piping and equipment. The change will utilize either the existing carbon steel line from the valve gallery to the Fill Aisle or use a hose in place of the carbon steel piping.

The safety concerns associated with the proposed change have been evaluated and it is determined that it will not adversely affect nuclear safety or safe plant operation. It does not involve any unreviewed safety questions or environmental concerns, and it does not require a change to the Plant Technical Specifications.

Modification: SE-**000232-055 Rev. 0**, Addition of Flanges in "A" Distillate Cooler Piping

This modification provides enhanced maintainability of the "A" Distillate Cooler without impacting system operating or design parameters. The modification leaves the plant in the same functional condition it is presently in.

This modification did not: 1) adversely affect nuclear safety and safe plant operations, 2) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, 3) create the possibility for an accident or malfunction of a different type than any previously identified in the SAR or 4) reduce the margin of safety as defined in the basis of any Technical Specifications. Furthermore, this Safety Evaluation has determined that no unreviewed safety questions have been created and that no environmental impacts are involved. This modification does not violate any licensing requirements, cause any radiological concerns and does not affect the plant's discharge permit condition.

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Modification: **SE-000232-057, R0**, Direct DWEDT/RBEDT to Sample Tanks

This modification allowed the contents of the High Purity Collection Tank to be routed directly to the hotwell without unnecessary processing. This reduces the time to process the contents of the High Purity Collection Tank in the High Purity Demineralizer and the associated radwaste.

This modification added operational flexibility to bypass the High Purity Demineralizers should the quality of the contents of the High Purity Waste Collection Tank be high enough to allow return directly to the hotwell. This modification does not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10 CFR 50.59.

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Document: **SE-000232-058, R0**, FSAR Update - PFU 11-110

This Safety Evaluation was written to address changes to FSAR tables 9.1-2, 11,2-11, and 11.2-23, and the description in section 9.1.3.2.1. Deviation Report 97-853 identified data regarding Fuel Pool and Radwaste demineralizers, which does not agree with current operating practices.

This modification did <u>not</u> (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there is no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification is acceptable.

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Modification: **SE-000232-059, R0**, Chem Waste Filter Outlet Flow Monitor

The change permanently installs an ultrasonic flow monitor to measure the outlet flow of each of the Chem Waste Filters. It is necessary to provide the information needed to adjust the flow through the system when only the demineralizers are in service and the evaporator is out of service.
This change did not adversely affect nuclear safety or safe plant operations; does not constitute any unreviewed safety question and did not require a revision to the Technical Specification or the FSAR.

Modification: **SE-000233-013, R0**, Addition of Solid Radwaste System Drain

This modification provided the ability to drain portions of the Solid Radwaste System, when required for maintenance or routine system evolutions, by installing permanent drain connections at the Rapid Dewatering System equipment skid. Installation of the drain valves diminishes the possibility of an inadvertent spill of potentially contaminated effluent and will leave the plant in the same functional condition it is presently in.

The proposed modification did not: 1) adversely affect nuclear safety and safe plant operation, 2) increase the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, 3) create the possibility for an accident or malfunction of a different type than any previously identified in the SAR, or 4) reduce the margin of safety as defined in the basis of any Technical Specification. Furthermore, this Safety Evaluation has determined that no unreviewed safety questions have been created and that no environmental impacts are involved. This modification did not violate any licensing conditions, cause any radiological concerns, and will not affect the plant's discharge permit condition.

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Modification: **SE-000233-015, R0**, Alternate Spent Resin Tank Drainage Path

This configuration change provides an alternate process path for SL-T-2A/B drain down. This path has been demonstrated to be effective for the removal of organic sulfates from spent resin tank water. The change primarily removes the internals from 1-inch check valve SL-CKV-155 and reroutes process water to the UF batch tank using qualified hoses and hose connections.

The safety concerns associated with the proposed change have been evaluated and it is determined that it does not adversely affect nuclear safety or safe plant operations. It does not involve any unreviewed safety questions or environmental concerns, and it does not require a change to the Plant Technical Specifications.

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Configuration Change: **SE-000241-017, R0**, Containment Spray/Emergency Service Water Thermowell Plugs

This activity removed the tape and existing plugs from the thermowells on CS System I and ESW System I and installed new threaded pipe plugs. The affected RTDs are high accuracy RTDs that we use to obtain accurate temperature information in order to assess CS heat exchanger performance. The reason for removing the RTDs and installing them only when CS heat exchanger performance data is to be obtained is to ensure that the RTDs and the thermowells are maintained in the best condition possible.

This activity did not adversely affect nuclear safety or safe plant operations, nor did this activity increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, nor does it create the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR, nor did this activity reduce the margin of safety as defined the basis of any Technical Specification. Therefore, this activity does not result in or introduce an unreviewed safety question as defined by 10 CFR50. 59.

MNCR Disposition: **SE-000241-018, R1**, Rigid Strut Clamp Rotated

The MNCR is for Rigid Strut 241-BP-433-R7-77. The pipe clamp is rotated 10 degrees to the south and the strut is rotated 2.5 degrees from the vertical. The design function for the support is to restrain vertical seismic load. As evaluated on November 21, 1997, the design function is not altered and the current configuration meets the intent of NRC Generic Letter 91-18, Enclosure 2, Paragraph 6.1.3. It is returned to the design alignment prior to the next scheduled system operation.

By the responses to the Safety Evaluation and 50.59 considerations, the restraint as configured does not have an adverse affect on nuclear safety or safe plant operations and there is no unreviewed safety question.

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Temporary Modification: **SE-000241-024, R1**, Implementation of Temporary Modifications Supporting the Emergency Core Cooling System Strainer Project

The implementation of (3) Temporary Modifications (TMs) will support the installation activities for the ECCS Strainer Project. The removal and restoration of the maintenance platform/handrail assembly that services CS System Heat Exchanger H-21-1C and CS System Support No. 241-BP-NQ-2-R5-44 provides additional clearance between these components and the south Hydraulic Control Units (HCU's). The installation and removal of the RISI Calibrated Test Gauge establishes the existing and new suction pressure values for the Containment Spray Pumps and the Core Spray Pumps.

The above TMs do not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there is no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification is acceptable. These TMs did not violate any licensing requirements, cause any radiological concerns and will not affect the plant's discharge permit condition.

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Document: **SE-000243-017, R0**, Drywell and Torus Minimum Temperature

The lowest expected Reactor Building temperature was revised from 50 degrees to 40 degrees. This reduction in Reactor Building temperature did not reduce the Drywell or Torus metal temperature to the threshold for brittle fracture.

This FSAR change does not involve an unreviewed safety question nor a change to Technical Specification.

Modification: **SE-000250-001, R0**, Fuel Pool Gasket Retainer Plate, In Support of FCN C094565

This modification utilized 1/2-inch x 1/4-inch retainer plate in lieu of the existing 1/2-inch x 1/8-inch retainer plate previously used to hold down the sealing gasket on the fuel pool gates.

Neither nuclear safety nor safe plant operations are adversely affected because there were no changes to the function or operation of the fuel pool gate gaskets. It is the gasket that provides the seal for the fuel pool gate. The modification did not increase the probability of occurrence or consequences of an accident previously evaluated in the SAR because the modification has no effect on the operation of the fuel pool gate gaskets. The gaskets perform the safety function of sealing the gate to the fuel pool. The modification did not increase the probability of occurrence or consequence of a malfunction of equipment important to safety previously evaluated in the SAR because the modification has no effect on the operation of the fuel pool gate gaskets. The gaskets perform the safety function of sealing the gate to the fuel pool.

The modification does not create a possibility for an accident or malfunction of a different type than any previously identified in the SAR because the modification has no effect on the operation of the fuel pool gate gaskets. The gaskets perform the safety function of sealing the gate to the fuel pool. The modification will not affect any Technical Specification or licensing requirement because that which is changed by this modification is not the basis for any Technical Specification or licensing requirement. The modification does not involve a radiological safety concern because there is no change to plant function or operation. The modification which changed the thickness of the fuel pool gate seal plate is safe and acceptable.

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Document: **SE-000250-002, R0**, Remove Restrictions on Loading of Intact Fuel in the Existing Spent Fuel Pool

The current procedural restrictions for the spent fuel racks requires that the fuel load in each quadrant of the racks be symmetric in nature. The restrictions for symmetric loading was contained in GPU Nuclear submittal dated September 2, 1983 entitled "Technical Specification Change Request No. 111 and Facility Description and Safety Analysis Report Amendment No. 79." However, the NRC Safety Evaluation supporting the above license change (Amendment No. 76) issued on September 17, 1984, and its appendix (Appendix A), Technical Evaluation Report, prepared by Franklin Research Center, did not specify or discuss the symmetrical loading. In the symmetrical loading, the centroid of the totality of the fuel in each rack must be aligned with the rack metal cross section centroid within a 10 percent tolerance. This restriction was placed at the time the racks were installed since Holtec and the industry did not possess sophisticated analysis tools and thereby proposed to the NRC that racks shall be balanced for all fuel evolutions.

This change results in less personnel dose during fuel racking evolutions, facilitate the planned Badger testing conducted in the fuel pool and allows more economical use of the fuel pool.

Removal of the current loading restrictions from fuel racks A, B, C, E, F, G, J and K and modifying the restrictions on fuel racks D and H will not: affect nuclear safety or safe plant operation; increase the probability of occurrence or consequences of an accident or malfunction previously evaluated; introduce the possibility of an accident or malfunction of a different type than any previously evaluated, nor require any Technical Specification change. The margin of safety defined in the Technical Specifications is not reduced.

Modification: **SE-000251-018, R0**, Addition of "Y" Lateral to Rx. Cavity Drain Line

This modification installed a permanent 2-inch wye lateral to the Reactor Cavity concrete trough drain line for purposes of cleaning and video inspection of the drain line. It is demonstrated that this modification did not adversely effect nuclear safety or the environment. No unreviewed safety question is involved by this modification and it was implemented under 10CFR50.59.

Modification: **SE-000251-020, R0,** Addition of Valves to Spent Fuel Pool Tell-Tale Drains

The purpose of this document was to evaluate the safety of modifying the spent fuel pool liner leakage drain piping to prevent loss of spent fuel pool inventory through the liner leakage drain piping in the event of a heavy load drop causing a leak in the pool liner. These new valves may also be operated at any time when significant flow is observed through these drain lines.

This modification added valves to the spent fuel pool liner leakage drain piping. The modification does not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can he implemented under 10 CFR 50.59.

Configuration Change: **SE-000252-006, R0**, Defeat of Refuel Bridge Backup Slack Cable Limit Switch

A spring was removed that has caused erratic load cell indication at low hoist loads. The purpose of the spring was to operate the backup slack cable limit switch. With the removal of the spring, the function of the backup slack cable limit switch is being defeated.

The defeat of the refueling bridge backup slack cable limit switch does not adversely affect plant safety and does not involve an unreviewed safety question. This change does not require a revision to any system/component description or procedural/operating description in the SAR.

Modification: **SE-000331-008, R1**, Discharge Pressure Gauge for Steam Packing Exhauster

There is no Steam Packing Exhausters discharge pressure indication in the Steam Jet Air Ejector and OffGas System. This indication is helpful to determine the performance of Steam Packing Exhausters. This modification installed a pressure gauge on the wall in Mechanical Vacuum Pump Room for the Steam Packing Exhauster to determine the performance of the gas booster pumps. The gauge, as well as the Steam Jet Air Ejector and OffGas System do not perform any safety function.

The modification did not constitute an unreviewed safety question and did not have any adverse effect on safety or the environment.

Modification: **SE-000411-015, R0**, MSIV Cover Modification - Poppet Backseat

The intent of this modification was to improve valve reliability by reducing internal wear. This was accomplished by installing a seat on the inside of the valve cover. The valve poppet will rest on this seat when in the open position. This reduces flow-induced vibration, and its effects, on the poppet.

This modification did not adversely effect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it was implemented under 10 CFR 50.59.

Alternate Replacement - **SE 000411-017, R0**, Alternate Replacement of Low Pressure Main Steam Line Switches

The existing Main Steam Low Pressure switches were replaced with hermetically sealed switches. the replacement serves the same form, fit and function as the original.

The safety of the plant is not affected by this replacement. The RE23 switches will operate during all modes of plant operation. This replacement did not adversely affect nuclear safety, did not reduce the margin of safety, did not involve an unreviewed safety question and was performed in accordance within existing electrical/instrument installation requirements.

Document: **SE-000411-023, R0**, Plant FSAR Update Review - Main Steam Isolation Valves

Plant FSAR Update 11-57 requested a change to section 4.5.1 in which the design bases of the employment of the Main Steam Isolation Valves is described. The specific change to be made was the deletion of the description of the range of MSIV opening times from a fixed range of "25 to 35 seconds..." to allow the use of a range to be specified "to coincide with the acceptable results of historical in-service testing."

This revision posed no unreviewed safety questions or environmental concerns.

Modification: **SE-000421-012, R0**, "A" Condenser Sight Glass Removal

The purpose of this modification was to remove and discard sight glass indicators LI-0001 and LI-0002 from the hotwell piping of condenser CD-2-IA.

The removal of the sight glass indicators LI-0001 and LI-0002 with associated instrument isolation valves V-2-1022, 1023, 1025, 1026 from the Hotwell piping on Condenser CD-2-1A does not adversely affect plant safety and does not involve, an unreviewed safety question.

Modification: **SE-000421-018, R0** - Install Pulsation Dampeners on Condensate Pump Discharge Pressure Gauges

The purpose of the modification was to install pressure dampeners on the Condensate Pump discharge pressure gauges. This was done to reduce the number of transient pressure surge induced failures of the pressure gauges. The purpose of the gauges is to indicate condensate pump discharge pressure to the plant operators locally in the feed pump room. The pressure gauges and the pressure dampeners are classified as "OTHER".

This configuration change to install pressure dampeners on the Condensate Pump discharge pressure gauges did not adversely affect nuclear safety or safe plant operations and did not involve an unreveiwed safety question.

Temporary Modification: **SE-000421-019, R0**, Condensate Pump Initial Startup Seal Water Line (98-90)

This modification involved the installation of a temporary modification to isolate a leak, which appears to be underground, in the initial startup seal water line from the CST for the Condensate Pumps. The temporary modification consisted of cutting the line between valves V-2-25 and V-2-23, threading the line, and install a 1" globe valve to effect positive isolation of the leaking portion of the line from the CST side. The temporary valve was installed in the Condensate Transfer house.

This safety evaluation determined that this temporary modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Modification: **SE-000422-008, R1**, Additional Valve to Pump 1A Suction Drain Line

This modification provided an additional valve to the feedwater Pump 1A suction drain line. The additional valve was used to add a zinc injection skid to the system at a later date while the plant is operating.

This modification did not adversely effect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it was implemented under 10 CFR 50.59.

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Temporary Modification: SE-**000422-025, R1**, Disconnect Failed Portion of Circuit 14-9 (RFP-1A)

This was a temporary modification to limit the current on the 'A' pump motor to less than 475 amps, and permit operation with only one 500 MCM cable per phase until the I8R refueling outage. A reduction in load on The "A" pump motor also increases the load on the "B" and "C" pump motors.

The temporary modification does not involve any unreviewed safety questions or environmental concerns. No changes to the Technical Specifications were required.

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Modification: **SE-000423-006, R0**, Addition of Vent and Drain Line

FCN c009467 requests that drawing 147434 be modified to reflect the "mod addressed by SROC 82-81- and provided an attached sketch to show the change. The specific change was the addition of common vent and drain piping for the full train of condensate demineralizers. The piping allows demineralizers to be vented and/or drained with the liquids directed to the floor drains instead of being spilled onto the floor. There is no way to determine when this modification was performed or to what standards as "SROC" does not appear in EDMS as a valid database. The modification is outside the pressure boundary of any operating Condensate Demineralizer.

Incorporation of this information on the flow diagram does not adversely effect nuclear safety or safe plant operations because the functions of the system have not changed. The modification allows the condensate demineralizers to be vented and/or drained without contaminating the floor. This improves the ALARA aspects of such an evolution.

Modification: **SE-000423-006, R0**, Addition of Vent and Drain Line (Cont'd.)

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report has not increased because the system whose drawing was modified performs no safety function and the change can only have a positive effect on contamination control.

The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report has not been created. The changes documented do not impact the design and operation of the system such that the safety function of any component or system would-be challenged.

The margin of safety as defined in the basis for any Technical Specification has not been reduced because the system has no safety function and is not included in the basis of any Technical Specification. Furthermore, this drawing change does not impact any system defined in the basis of the Technical Specifications. Therefore, no unreviewed safety question exists.

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Modification: **SE-000423-008, R0**, Condensate Demineralizer Underdrains Replacement

This modification replaced the Condensate System Demineralizer internal underdrains with a new underdrain system, which is capable of withstanding 310 psid pressure differential.

It was concluded the modification to the Condensate System Demineralizer Underdrains does not involve any unreviewed safety questions; nor requires a change to the Technical Specifications.

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Modification: **SE-000431-011, R0**, Replace Auxiliary Flush Tank Drain Support DR-13-H4

This modification upgraded a temporary support on the drain line from the Auxiliary Drain Tank. The support replaces the original spring can support (DR-13-H4).

This modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there is no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification is acceptable.

Modification: **SE-000431-014. R0**, Target Plate for Flash Tank Monitor

The header that is the inlet to Flash Tank 1-2 (T-4-002) had a leak directly downstream of the outlet from V-4-99. The leak was caused by impingement on the wall of the header. This modification installed a connection and target plate in this area of the header. A pipe was welded to the header at the area where flow impinges on the header. A weld neck flange along with a target plate was added to the pipe.

This modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact and will not affect the plant's discharge permit condition.

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Drawing Change: **SE-000521-003, R0**, Field Change Notice

FCN C090202 requested that Well and Domestic Water System drawing GU-3E-871-21-1000 sheet 1 be changed to show valves V-10-30 and V-10-409 as normally closed. GU-3E-871-21-1000 is referenced in Section 9.2.4.1.2 of the FS AR.

V-10-409 is located in a 3 inch line running between Clearwell Tank T-10-14 and the pretreatment trailer. V-10-30 is located in a one-half inch capped line branching off the discharge of Filtered Water Pump P-10-001A. Per FSAR Section 9.2.4.1.3, "The Well and Domestic Water System is not required for the safe shutdown of the reactor nor to mitigate the consequence of postulated accidents'

Indicating the normal valve position on the drawing does not change the design function or manner of operation of any structures, systems or components. Valve operation is controlled by approved plant procedures.

This activity did not change any plant equipment, plant system, or plant technical specification. Existing margins of safety defined in the plant technical specifications or FSAR were not reduced as a result of this activity. Nor is the probability of occurrence or consequence of an accident previously evaluated in the FSAR affected by this activity. Therefore, there is no unreviewed safety questions resulting from this activity.

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Corrective Change: **SE-000523-018, R0**, Valve Change

FCN c063231 documented a "Corrective Change" performed via 125-1 Form Number 149-92 to a system that does not perform a safety related function. The specific change was to valve WD-HV-0140 wherein 3/4-inch Woodhall globe valve Model "W30" was replaced with 3/4-inch Whitey ball valve Model "65TSW12P". The substitution was evaluated as acceptable in the "Equivalent Evaluation Table."

Incorporation of this information on the flow diagram did not adversely effect nuclear safety or safe plant operations because the functions of the system have not changed.

Corrective Change: **SE-000523-018, R0**, Valve Change (Cont'd.)

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report was not increased because the system whose drawing was modified performs no safety function and the change has no impact on any system which has a safety function.

The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report was not created. The changes documented do not impact the design and operation of the system such that the safety function of any component or system would be challenged.

The margin of safety as defined in the basis for any Technical Specification was not reduced because the system has no safety function and is not included in the basis of any Technical Specification. Furthermore, this drawing change did not impact any system defined in the basis of the Technical Specifications.

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Modification: **SE-000523-019**, R0, Flow Diagram Revision, BR 2004

The modification identified by FCN C-060068 is: Valve V-12-186 has been changed from 3/4-in. gate valve to 1/2 in. ball valve. Valve V-12-186 is the block valve for a demineralized water hose station. The ball valve will pass water with an essentially equivalent pressure drop than the earlier gate valve. Therefore, the capability of this hose station to provide demineralized water as needed has not been diminished. Per the Oyster Creek Line List, the piping in which the subject ball valve is installed has a design rating of 175 psig and 100°F. A review of the vendor's technical literature applicable to this ball valve identifies that its service pressure/temperature rating exceeds the design ratings of the piping in which it is installed.

Per section 9.2.3.2.3 of the Oyster Creek Updated FSAR, "The WD System is not required for the safe shutdown of the reactor nor to mitigate the consequences of postulated accidents. The system is not safety related."

This change did not require any change to the Technical Specification and does not involve an unreviewed safety question.

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Modification: **SE-000523-020, R0**, RBCCW Flow Integrator Replacement

The purpose of this document was to evaluate the safety of installing new flow instrumentation in the Demineralized Water Transfer System in the RBCCW tank makeup line in place of the existing Controllatron flow instrument.

This modification did not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10 CFR 50.59.

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Modification: **SE-000523-021, R0**, Abandonment of the Make Up Sample Sink

The purpose of this safety evaluation was to add a valve that was already installed in the system but never given an Instrument ID number. The valve was installed at the old Makeup Sample Sink that was recently removed from the Turbine Building basement. The valve was revised as part of the removal modification to make the valve a locking handle to prevent the inadvertent operation of the valve. However, upon completion of the task, the valve was never entered into the GMS 2 database nor was the valve shown on any procedure for valve line up. This activity is being done because the valve is being added to a control room drawing that is part of the FSAR and, therefore, a safety evaluation is needed. This safety evaluation proves that the addition of this valve V-12-0369 is not a safety concern and that an unreviewed safety concern does not exist as a result of this activity.

Configuration Change: **SE-000531-021, R1**, Replacement of ESW Keep Full Throttle Valves

The purpose of this document was to evaluate the safety of replacing the ESW keep full throttle valves with valves that have better characteristics for maintaining fluid velocity within the capability of the check valve. The existing 2 inch valves V-3-940 and V-3-941 were replaced with 1 1/2 inch globe valves. The new valves are specifically sized for this application.

This modification did not adversely effect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it can be implemented under 10 CFR50.59.

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Modification: **SE-000531-025, R0**, Service Water Booster Pump Expansion Joint

This modification replaced a portion of the 3" CPVC piping with an expansion joint. This 3" line had developed a leak and required repair. The root cause of the leak is due to piping misalignment which may have been caused by a minor settling of the south wall of the Condensate Transfer Building. The expansion joint is considered prudent should the wall continue to settle.

This modification does not adversely effect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10CFR50.59.

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Modification: **SE-000531-026, R0**, Relocation of TAP for PT-6

This modification rerouted the pressure tap from the Service Water Pumps discharge line to pressure transmitter PT-6. The reason for the relocation was to be able to provide freeze protection for the line. During the winter months, the 3 inch piping routed under the deck and not heat traced can freeze resulting in loss of pressure indication. With the transmitter sensing line above the deck, the line can be easily heat traced.

This modification did not adversely affect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it can be implemented under 10 CFR 50.59.

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Configuration Change: **SE-000532-018, R2**

The purpose of this document was to evaluate the safety of installing a vent connection and root valve in ESW System I just downstream of orifice RO 21A. The connection allows the installation of a hose that will vent the system outside of the Reactor Building. This permits verification that venting of the pipe downstream of the orifice will reduce cavitation and flow induced vibration that was present. This safety evaluation also addressed the Temporary Installation of a hose to the Reactor Building Service Air penetration to allow the ESW System to be vented to the outdoors. This SE also addressed the method of installing the connection to the existing pipe so that secondary containment remains intact.

This modification does not adversely affect nuclear safety or the environment. No unreviewed safety question is generated by this modification and it can be implemented under 10 CFR 50.59.

Modification: **SE-000532-019, R0**, Conduit Support Replacement on Intake Structure North End

On the Intake Structure operating deck at Elevation 6' 6", there were electrical cable conduits providing power to various pumps, MCC, electrical panels, heat trace and pressure gauges, etc. Among them was the power supply to ESW pumps classified as "NSR." Due to the harsh environment in the Intake Structure operating deck, some conduit supports were corroded and required replacement. Several walkdowns were performed to identify the deteriorated "NSR" conduit supports. As a result, a total of five supports were determined to need replacement at this time. The new supports have the same structural configuration including the material (painted A36 steel plate and unistrut material) except, instead of 4 anchor bolts, only 2 anchor bolts were installed due to interference. The locations of the new supports will be within one foot of the existing ones.

There was no impact to nuclear safety or safe plant operation. The probability of occurrence of the consequence of an accident or equipment malfunction was not increased. The possibility for an accident or malfunction of a different type was not created. The margin of safety for the new conduit support was not reduced as a result of this activity. Therefore, this activity does not involve any unreviewed safety questions. No changes to the Technical Specifications were required.

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Modification: **SE-000532-023, R1**, ESW System 1 Pipe Vent

The purpose of this document was to evaluate the safety of extending the piping from the vent connection and root valve in ESW System 1, downstream of orifice RO 21A, through the North wall of the Reactor Building. This allows venting of the ESW system. It was confirmed that a vacuum existed downstream of RO-21A. The vacuum caused the pipe to vibrate. A monitoring program verified that the vibration was significantly attenuated by venting the pipe (breaking the vacuum). The safety of the vent itself and the Temporary Modification for the monitoring program was previously evaluated via SE 000532-018. The modification replaced the temporary modification.

This modification completed the venting ESW System I downstream of orifice RO 21A. A vent pipe routed to the outside of the Reactor Building breaks the vacuum in the ESW pipe to significantly reduce flow-induced vibration.

This safety evaluation determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification was acceptable.

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Modification: **SE-000533-007, R0**, Intake Low Level Alarm Addition

This modification added a local alarm in ER 42 annunciator window 11B and retransmitted this alarm to control room panel 5F/6F annunciator k, window k-5-f. This control room alarm provides control room operators sufficient time to evaluate intake degradation. The modification did not constitute an unreviewed safety question and will not have any adverse effect on safety or the environment.

Modification: **SE-000535-016, R0**, Circulating Water Pumps Seal Water Line Isolation Valve

This modification installed a new isolation valve in the circulating water pumps seal water line. The valve tag number is V-3-1104.

This modification did not adversely affect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it was implemented under 10 CFR 50.59.

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Modification: **SE-000536-005, R0**, Dilution Pump House Modifications: Pump Seal/Lubricating Water Supply, Pump Lube Oil System, Pump Lube Oil Cooler Water Supply

The Fire Protection Water (FPW) System provides cooling water to the lube oil coolers, seal water to the mechanical seals and lubricating water to the rubber bearings of the three Dilution Pumps (P-43-002A, P-43-002B and P-43-002C). The existing water supply piping configuration did not provide sufficient flowrate of cooling/seal/lubricating (CSL) water to support the simultaneous operation of (3) Dilution Pumps as stated in GPUN OC Station Procedure No. 324, Section 4.12. When a Dilution Pump trips Off-line, an operator must be dispatched to the Dilution Plant and must manually realign the CSL water to the standby Dilution Pump, prior to starting the standby pump.

For the standby Dilution Pump, the lube oil was constantly being cooled by Lube Oil Cooler (C-43-001, C-43-002 or C-43-003) and reheated by Lube Oil Heater (C-43-004, C-43-005 or C-43-006). This cyclic mode of operation was due to the standby status of the Dilution Pump. Cooling water was provided to the standby lube oil cooler in order to prevent fouling of the cooling water supply line to the cooler. Also, the lube oil and reduction gearbox were maintained at operating temperature so that the standby Dilution Pump could be started under normal startup conditions or within the (15) minute duration as required by our NJSDEP environmental discharge permit.

This modification to the piping configuration of the existing Cooling/Seal/Lubricating (CSL) water supply to the diluton pumps provides a sufficient flowrate of water to support the simultaneous operation of three Dilution Pumps as stated in GPUN OC Station Procedure No. 324, Section 4.12. Regulating Valve V-43-152 was removed from service and the cooling water filtered to minimize fouling of the cooling water supply line to the lube oil coolers. Flush connections are provided in the cooling water supply line and the seal/lubricating water supply line. The CSL water supply lines are heat traced for freeze protection. A lube oil bypass line was provided around each of the lube oil coolers. The bypass line eliminates the cyclic operation of the lube oil coolers and lube oil heaters for the standby Dilution Pump.

This configuration change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this configuration change is acceptable.

Document: **SE 000541-022, R0**, Addition of Block Valve V-5-775 (FCN C088933)

The modification, as described in FCN C088933, is the addition of a block valve (V-5-775) downstream of Reactor Building Equipment Drain Tank temperature control valve (V-5-128). This valve permits isolation of V-5-128 for maintenance.

This safety evaluation determined that this change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Modification: **SE-000551-012, R1**, Sampling System Data Acquisition System (DAS)

The purpose of this modification WAs to display conductivity and oxygen data for Reactor Water and Final Feedwater on an office building computer. This was accomplished by monitoring the existing signals within the Reactor Water Sampling System (RWSS) and Final Feedwater Facility (FFF) sampling panels and transmitting the data over a RS485 bus for display remote from the radiation area of the RWSS and FFF panels.

This modification did not adversely affect plant safety or involve an unreviewed safety question. No changes to the Technical Specifications were required.

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Modification: SE-**000551-013, R0**, Abandonment of ECPMS Equipment Skid

Due to performance issues, the ECPM system had been abandoned in place per Configuration Change OC-CCD-000551-008. The abandoned equipment skid was connected to the Post-Accident Sampling and the Clean-Up Demineralizer systems by 1/2" tubing and 1" piping connections. Leak-by of isolation valves was creating radiological concerns. This proposal provided direction for severing the interconnecting piping/tubing and providing a means of positive isolation.

The modification did not: 1) adversely affect nuclear safety and safe plant operation, 2) increase the probability of occurrence, or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, 3) create the possibility for an accident or malfunction of a different type than any previously identified in the SAR, or 4) reduce the margin of safety as defined in the basis of any Technical Specifications. No unreviewed safety questions have been created and no environmental impacts were involved. This modification did not violate any licensing conditions, cause any radiological concerns, and did not affect the plant's discharge permit condition.

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Document: **SE-000556-002, R2**, Downgrading of System 556 from NSR to RR

The purpose of this Safety Evaluation was to document the basis for the downgrade of the "Hydrogen Detection/Sampling System", System A 556, from Nuclear Safety Related (NSR) to Regulatory Required (RR). Most of the area hydrogen monitors on site are Regulatory Required, however, there are some that are classified as "Other" and these will remain as such.

This downgrade was done because there appeared to be some contradiction as to the correct QCL rating of the system. This document also describes the system and the system functions as required by regulation or commitments and provided the justification for the QA classification for the system and its components.

Document: **SE-000556-002, R2**, Downgrading of System 556 from NSR to RR(Cont'd.)

The conclusion of this Safety Evaluation was that the Area Hydrogen monitors located throughout the plant are generally required to be "Regulatory Required" specifically where the areas involved are associated with the piping of the Offgas System. This Requirement comes from the IE Bulletin 78-03. Other Area Hydrogen monitors can be classified as "Other" if part of another system and the safety evaluation used to install the Monitoring system provides the justification for that QCL Classification.

Therefore, there were no unreviewed safety concerns associated with this activity.

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Document: **SE-000567-003, R3**, Evaluation of Raising the Maximum Hydrogen Injection Flow Rate

The purpose of this evaluation was to provide evidence that there were no adverse affects on the operation of the Reactor with hydrogen injection rates anywhere in the design control range of 0 to 15 SCFM and up as high as 40 SCFM during a transient. The highest known flowrate is based on the limitations of the control loop which is 15 SCFM max. In addition, the purpose was to allow the High Flow trip setting to be placed anywhere in the design band of 0 to 15 SCFM, including no trip at all. The only limitation is that the MSLR monitors do not exceed 480 MR/HR and the AOG throughput does not exceed 40 SCFM.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there were no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification was acceptable.

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Document: **SE-000571-014, R1**, Covered Floor Drains

Three floor drains in the Turbine Building have plastic covers over them to prevent a radiological concern. Either contaminated water can splash out or airborne material from the offgas system could potentially leak out. This safety evaluation addressed the fact that flow into these floor drains is restricted due to being covered.

There were no unreviewed safety concerns resulting from the fact that three floor drains in the Turbine Building are covered.

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Modification: **SE-000571-017, R0**, As-Found FCNs

This modification replaced the Turbine Building Floor and Equipment Drain Sump level control and sump pump mechanical alternator with a bubbler type level control and electrical alternator. As discussed in this Safety Evaluation, this change did not adversely affect nuclear safety or safe plant operations. This modification did not involve an unreviewed safety question as determined by 10CFR50.59.

Modification/Document: **SE-000572-004, R0**, Reactor Building Equipment Drain Tank Support & Platform

The calculation and the modification resolve the USI A-46 concerns with the RBEDT support and platform. The modification only involved limited structural strengthening of bolted connections and the addition of stiffeners. There was no effect on system function or performance, accidents or malfunctions. All seismic requirements will be met so there is no affect on any margin of safety or the SAR. No unreviewed safety concern is created by this activity.

Document: **SE-000575-001, R0**, Engineering Change Document

This modification corrected drawing GE 148F437, sheet 2, to properly reflect that level switch LS-687 is now a blue ball float and not the bubbler system that the drawing shows. The new float system works as well as the old bubbler system for providing activation of Hi Level alarm WC-LAH-099. The ECD was submitted to have the drawing reflect the as found condition in the Laboratory Drain Tank T-22-003.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this modification was acceptable.

Modification: **SE-000600-001, R0**, Process Noise Reduction in Instrument Impulse Lines Using Snubbers, Pulsation Dampeners and Gauge Savers

The purpose of this safety evaluation was to justify the use of snubbers and gauge savers in the impulse lines to plant instruments. As a result of this review, it is concluded that the use of snubbers and/or gauge savers in the applications is acceptable. The use of snubbers and/or gauge savers does not create an unreviewed safety question, and nuclear safety and safe plant operations are not adversely impacted.

Modification: **SE-000621-012, R0**, APRM Flow Bias Trip Slope Change

This modification replaced the fixed resistors in the flow biased scram and rod block circuits as the means for changing the slope of the trip functions to the desired value. This change was limited to the activity (FCN C049481) that replaced the resistors in 1986. The system requirements and related parameters such as setpoints may have changed between then and now and such changes are not addressed here.

Replacing the fixed resistors in the flow biased scram and rod block circuits as the means for changing the slope of the trip functions to the desired value are acceptable. It was concluded that an unreviewed safety question is not created and nuclear safety and safe plant operations were not adversely affected.

Configuration Change: **SE-000628-001, R0**, Scram Accumulator Rod Block Time Delay

The purpose of the configuration change was to prevent inadvertent control rod motion during rod withdrawal due to the "break before make" characteristic of the scram accumulator alarm relays, 4K5's. There had been repeated instances of rod movement in the wrong direction due to this phenomenon. Troubleshooting determined that momentary rod block signals from a scram accumulator during the rod withdrawal sequence was causing this problem. The configuration change installed a time delay relay such that momentary accumulator alarm signals would not cause a rod block.

The addition of a time delay relay in the scram accumulator rod block circuitry did not adversely affect plant safety and did not involve an unreviewed safety question. No changes to Technical Specifications were required.

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Modification: **SE-000642-008, R2**, EMRV Cable Addition

The purpose of this modification was to reduce circuit resistance from DC battery to the EMRV solenoids. The following describes the scope of the modification:

a) Installed a larger size cable between DC power panel DC-D and relay panel ER-18A. This new cable was connected in parallel with the existing cable 62-69. New 1 inch conduits between panel DC-D and the cable tray and the Chem. Lab. closet were installed to facilitate cable routing.

b) Spare conductors were connected in parallel for the following circuits: 63-541, 542, 543, 544, 596; 82-274, 276, 278, 325, 538; 63 GC 072, 0723, 0732; 63 RC 0721, 0724, 0731.

This modification added additional cable and conduit in a portion of the circuit and parallel spare conductors to reduce voltage drop resulting in higher available voltage at the solenoid terminals. This modification did not introduce any new accident or malfunction not previously evaluated nor does the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because this modification reduces voltage drop in a portion of the circuit and thus increasing available voltage at the solenoid terminals. This safety evaluation concluded that there was no unreviewed safety question per 10 CFR 50.59 and no changes to the Technical Specifications and UFSAR were required.

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Modification: **SE-000660-002, R0**, Install Reactor Building Barometer

This modification installed an electronic barometer and a DC power supply on Instrument Rack RK01 to provide Reactor Bldg. barometric pressure indication in plant computer. This loop does not perform safety related function. The barometer as well as DC power supply are seismically mounted to meet RG 1.29 Anti-fall down criteria. It is concluded this proposed modification does not constitute an unreviewed safety question and does not have any adverse effect on safety or the environment. This modification does not require any changes to the Technical Specifications.

Modification: **SE-000661-032, R0**, Stack Flow Transmitter Replacement

The purpose of this modification was to replace the pneumatic type transmitter (FT-50-0424) with an electronic SMART transmitter. The existing configuration was susceptible to a signal bias that shifts the zero reference one direction with morning sun exposure and the opposite direction with afternoon sun exposure on the sensing lines. Also, the existing configuration had a signal oscillation that continuously cycles the radiation monitoring valve (V-50-3 7).

This modification did not adversely affect plant safety, involve an unreviewed safety question. This modification did not affect safe plant operation or nuclear safety. No changes to the Technical Specifications were required.

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Engineering Evaluation: **SE-000666-004, R0**, Drywell $H_20_2$ Sample Pump Replacement Evaluation

Engineering Change Document (ECD) 211909 provided the documentation to change the governing document EQ file OC-347 to increase the replacement cycle for the $H_2/0_2$ sample pumps IT-001A/B (diaphragm) from 1000 hours to 1344 hours or 4 years (which ever comes first) of normal operation (required for surveillance purposes). All other components within the $H_2/0_2$ drywell sample system will maintain their qualified life at 40 years.

After repeated replacement of the sample pumps, the observed condition of the pump indicated that there was no excessive wear. This prompted the system engineer to request if it was possible to extend the qualified life of the pumps. Deviation Report 97-112 was issued to look into changing the replacement cycle.

This configuration change did not introduce any new accident or malfunction not previously evaluated, nor did the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR/SAR. This configuration change did not decrease the margin of safety as described in the Technical Specification because the configuration change is within the environmental qualification testing performed by COMSIP meeting the requirements of 10 CFR 50.49. The additional thermal life did not influence any system monitoring functions. This evaluation concluded there was no unreviewed safety question per 10 CFR 50.59 and no changes to the OC Technical Specifications were required.

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Modification: **SE-000666-006, R0**, Elimination of $H_2O_2$ Analyzer $H_2$ Gas Storage Requirement (<400 SCF)

The proposed change to the maximum hydrogen bottle pressure did not increase the consequences of an accident or the malfunction of equipment important to safety. The only change made was an increase in the volume of hydrogen gas stored for use in the event of a LOCA.

This safety evaluation has determined that this proposed change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Since there was no adverse affect on nuclear safety and/or safe plant operations, no creation of an unreviewed nuclear safety question and no environmental impact, this change was acceptable.

Document: **SE-000713-004, R0**, Revision to GE Drawing 234R166 (FCN C094501)

FCN C09451 documented an as-found condition of the stator winding cooling system. Specifically, instrument root valve V-Y-0112 and local pressure indicators PI-713-006/-007 were found not to be identified on GE Drawing 234R166, Rev. 9, "Stator Winding Cooling Water System."

A review of historical drawing revisions indicate that the subject pressure indicators are not original plant system components. These pressure indicators were most likely installed for collecting differential pressure data to determine system flow in support of the system's flow instrumentation upgrade performed under BA 408773 in early 1990.

The margin of safety as defined in Licensing Basis Documents and Technical Specifications has not been reduced because the modified system performs no safety function and the change has no impact on any system which has a safety function. No unreviewed safety question exists.

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Modification: **SE-000731-009, R0**, "D" Reactor Recirculation MG Set Motor Power Conduit Reroute

The purpose of this modification is to prepare for the potential failure of 4160 Volt Reactor Recirculation Pump 'D' to MG Set Motor power feeder cable that has degrading insulation.

This scope of this modification was to bypass a section of the existing conduit routing in the turbine building by the installation of new conduit following a different path from the Feedwater Pump Room to the Reactor Recirculation MG Set Motor. This eliminates the anticipated problems with using the existing conduit routing passing through the Feedwater Pump Room to the Reactor Building. This work was performed during 17R outage to avoid work in the Heater Bay during power operation when radiation dosage is higher. Cable will not be installed until needed. Conduit routing follows a path similar to that previously installed for Reactor Recirculation Pump 'E' (CCD-000731-002). The path is from the Office Building MG Set Room through the wall into the Heater Bay of the Turbine Building, through the floor slab into the Feedwater Pump Room and then over to an area above the conduit pit; therefore, the cable pull tension should be acceptable if and when the cable is required.

This modification did not introduce any new accident or malfunction not previously evaluated, nor did the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because the modification does not impact any system safety functions. This evaluation concludes there was no unreviewed safety question and no changes to the OC Technical Specifications were required.

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Modification: **SE-000731-010, R0**, 4160 480 Volt Transformer USS 1B2 Cover

The compartment cover for transformer USS 1B2 was cut into sections to allow for ease of access to the connections. This safety evaluation will assess the effects (if any) on Nuclear Safety of this modification.

The purpose of this modification was to provide ease of access to the three electrical connections at the top of the USS 1B2 transformer to allow for cable testing and inspection. Covering the "cuts" with an additional cover plate prevents the ingress of dust and allows the compartment cover to perform it's intended design function.

As there is no impact on nuclear safety and/or safe plant operations, no creation of an unreviewed safety question and no environmental impact, this modification is deemed acceptable.

Modification: **SE-000732-019, R0**, SF-1-25 MCC Modification

The purpose of this modification (OC-CCD-000732-002) was to:

- Replace the non-ambient compensated thermal overload (TOL) relays and heaters with newly resized heaters and ambient compensated relays.
- Install a fuse block with 1A type KTK fuses in the power feed to the UV coil.

The changes under these modifications had no impact on the Nuclear Safety or safe plant operations, and no unreviewed safety concerns exist.

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Modification: **SE-000735-009, R0**, "A" 125 VDC MG Set Rheostat Replacement

The scope of the modification was to permanently install a new rheostat for the existing rheostat that has been determined to be faulty. The new rheostat is functionally a one for one replacement. The new rheostat was mounted on a new rack anchored to the wall near the existing rheostat rack. The existing rheostat was removed and a terminal block installed in a new terminal box. Wiring was extended from the terminal block through flex conduit to the new rheostat rack.

This modification did not introduce any new accident or malfunction not previously evaluated, nor did the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because the modification did not impact any system safety functions. There was no unreviewed safety question and no changes to the OC Technical Specifications were required.

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Modification: **SE-000735-010, R0**, Alternate DC Control Power for Switchgear 1D

This activity provided jumpers to extend DC control power to alternate position of knifeswitch for switchgear 1D. Plant procedures are in place to ensure that the knifeswitch is aligned to the Battery B. There were no unreviewed safety questions or environmental questions due to this change. No changes to Technical Specifications were required.

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Document: **SE-000735-013, R0**, FSAR Update - Section 8.3.2 DC Power Systems (PFUs 11-098 thru 11-102

The purpose of this safety evaluation was to make administrative and technical changes to Section 8.3.2, DC Power System, of the OC FSAR.

It was concluded that no impact to nuclear safety or unreviewed safety question existed regarding the evaluated changes to Section 8.3.2 of the FSAR.

Modification: **SE-000735-015, R0**, Relocation of Term. Blocks in MCC DC-2 Units A01 and A02

The scope of this modification was the re-location of terminal blocks TB-2 and TB-3 approximately 2 inches to the left of its present position in MCC DC-2 Unit A0l and MCC DC-2 Unit A02.

In addition, the Plexiglas shield partially covering the pass-through hole between the bucket and the cable trough was modified to ease the re-connection of terminal block wires.

The MCC DC-2 Unit A01 and Unit A02 modification does not adversely affect plant safety and does not involve an unreviewed safety question This modification was implemented without NRC approval in accordance with 10 CFR 50.59.

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Modification: **SE-000735-016, R0**, Battery Charge C1 and C2 Alarm Modification

This modification modified the alarm circuitry of the C1 and C2 Battery Chargers by bypassing their respective Charger Failure Alarm (CFA) cards. The CFA card provides an alarm at a predetermined set point of low output current from the chargers. The existing configuration had the CFA in series with the charger low DC volt, high DC volt, system ground and AC low volt alarms to a common control room alarm.

This modification changed the existing control room alarms for Annunciator U, Alarm windows U-4-f and U-5-f by eliminating the existing low current portion of the charger trouble alarm.

This modification did not introduce any new accident or malfunction not previously evaluated, nor did the modification increase the likelihood of 1) occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because the modification still provides the control room operator information that the Chargers are functioning and providing output current and rated voltage to the 125 VDC Distribution system C. Therefore, this modification did not result in or introduce an unreviewed safety question as defined by 10 CFR 50.59.

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Modification: **SE-000735-017, R1**, Replacement Breaker for DC-C

The scope of this modification was replacement of all breakers in Distribution Center DC-C 125V with new devices that have test data showing trip clearing loads. Since the manufacturer is different, the fit required that new front panels also be installed. Distribution Center DC-C 125V is located within the Turbine Building 4160V Switchgear Room, on floor elevation 23'6". Distribution Center DC-C supplies the DC voltage to Division A safety related components that require DC voltage.

The Distribution Center DC-C modification did not adversely affect plant safety and did not involve an unreviewed safety question.

Modification: **SE-000741-009, R1**, EDG Switches Replacement

The purpose of this change was to ensure the EDG remained stopped at all conditions when the EDG mode selector switch (43MS) is positioned to STOP. The purpose of this change was to replace the existing level switch mechanisms with new level switch mechanisms that had its switches independently wired. For EDG application, this requires only two switch mechanisms. The newly created spare stillwell will be occupied with a level transmitter providing an additional parameter for the EDG Data Acquisition System (DAS). The new switches are field adjustable allowing the HI/LO level alarm to actuate just prior to the related automatic system protective action.

The modification improved EDG control, assured EDG fuel inventory, increased operator awareness of any EDG automatic fuel transfer protective actions, and provided an additional parameter for system monitoring. The modification meets these objectives without constituting an Unreviewed Safety Question as defined by 10 CFR 50.59. Additionally, no changes to the Technical Specifications were required.

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Modification: **SE-000741-013, R0**, Local Shutdown Panel DG-2 Transfer Switch Wiring

The scope of this modification was to re-terminate the wiring on fuse F18 pin 2 to ensure current is not drawn through the fuse during normal operation. After the transfer from the Local Shutdown Panel (LSP-DG2), power was available through fuse F18 in the event that main fuse F18A is blown.

The scope of this modification also included the re-termination of the wiring on main return fuse F19 pin 2 such that the power return path is completed when the transfer is made and main return fuse F19 is blown.

This modification did not introduce any new accident or malfunction not previously evaluated, nor did the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because the modification did not change function of the Local Shutdown Panel (LSP-DG2) or the Emergency Diesel Generator Panel EDG-2. There were no unreviewed safety question and no changes to the OC Technical Specifications were required.

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Modification: **SE-000741-014, R0**, EDG-1 & 2 Start Contactor & Relay ST Surge Suppressor

The purpose of this modification was to reduce the arcing across the STZ relay CD contacts caused by the ST relay that did not have a surge suppressor on its coil.

This scope of this modification was to add a surge suppressor across the coil of ST relay and wire STZ contacts EF in parallel with the existing contacts CD to increase the current carrying capacity. All work was internal to Emergency Diesel Generator Panels for EDG-1 & 2.

This modification did not introduce any new accident or malfunction not previously evaluated, nor did the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because the modification did not change function of Emergency Diesel Generator Panels EDG-1 & 2 or EDG 1 & 2. This evaluation concludes there was no unreviewed safety question and no changes to the OC Technical Specifications were required.

Temporary Modification: **SE-000811-037, R0**, PS-811-0022 Setpoint Change to 85 psig

The purpose of this modification was to change the setpoint of PS-811-0022, which monitors fire header pressure and provides a start signal to the 1-2 Fire Diesel/Pump. The setpoint was changed from 95 psig to 85 psig. A procedure change to 645.6.012, "Fire Pump Functional Test" was also made.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Document: **SE-000822-003, R2**, Trunnion Room Access

The purpose of this activity was to evaluate, with respect to Secondary Containment, passage through the Trunnion Room entrance at times when secondary containment is required. This activity specifically evaluated the impact of momentarily opening and closing the Trunnion Room door and the impact of such upon maintaining secondary containment integrity since the Trunnion Room is not provided with an air lock. Momentary opening and closing the Trunnion Room door is defined as the time necessary to allow passage through the door opening by personnel with closure of the door following their passage.

A secondary purpose of this activity was to clarify that the Trunnion Room entrance and door does not constitute an access opening to the reactor building for the purpose of evaluating Technical Specification 1.14 A, Secondary Containment Integrity. Access openings by original plant design include openings for passage of personnel and equipment which have been purposely designed to assure reactor building leakage rates are maintained within the designed maximum leakage of 100% of the building free volume per day at 0.25 inches of water differential pressure. To accomplish this design objective personnel and equipment access openings have been designed with interlocked double doors, airlocks, and weather strip type seals to minimize leakage. The Trunnion Room door has not been designed with these features since it is not utilized as a designed entrance or access to the reactor building for personnel or equipment.

Technical Specification 1.14A is not applicable to this door. The administrative controls in place are adequate to assure the above remains true. Based upon this evaluation this activity had no adverse impact on nuclear safety, is not an unreviewed safety question and is concluded to be acceptable.

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Setpoint Change: **SE-000822-031, R0**, RE-822-1148/1149 & R0014B9/C9 Setpoint Change

This modification involved changing the setpoint of radiation monitors as follows:

| Instrument | Existing Setpoint | New Setpoint | Tech. Spec Limit |
|---|---|---|---|
| RE-822-1148 & 1149 | 13 ± 1 mr/hr | 9 ± 1 mr/hr | ≤ 17 mr/hr |
| R0014B9 & C9 | 70 ± 10 mr/hr | 50 ± 10 mr/hr | ≤ 100 mr/hr |

This modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Engineering Evaluation: **SE-000822-048, R0,** Secondary Containment Integrity During Replacement of Valve

The replacement of the valve V-22-0189 created a temporary breach in the secondary containment boundary. However, this does not adversely affect nuclear safety and safe plant operation because no system or component is being affected beyond its design function. During the replacement of the valve, secondary containment was maintained. The opening from the 2-inch diameter pipe is smaller than the one identified in calculation C-1302-822-E150-066. The calculation results showed that an opening of 2.5 inches diameter would only add a small amount of in-leakage relative to the total building leakage. This additional leakage would not result in a loss of building negative pressure that was below the allowable Technical Specification limit. Additional controls were in place before commencing work, to ensure secondary containment integrity. They were:

- Both Railroad airlock doors were closed.
- Secondary containment had not degraded significantly since the last test, which was evidenced by a current successful surveillance (9/12/98, 665.5.002), and no known deficiencies.
- Personnel in direct communications with the Control Room were stationed at the opening ready to plug the opening upon operators' request if plant conditions should change.
- Work was performed during a refueling outage, which minimizes any issues on plant safety regarding accident conditions.

There were no adverse safety concerns by allowing the pipe to be open during the valve replacement activity as long as the precautions described above were followed. Thus, no unreviewed safety concerns or environmental issues were created by this activity.

Document: **SE-000823-008, R0**, Hot Exhaust Windscreen (FCN C059356)

The HVAC unit is located in the Multiplexer (MUX) Room. The air cooled condenser is located outdoors. The propeller fan air cooled condenser purchased under Specification SP-1302-17-001 is installed on support angles and supports posts and mounted on top of the corridor roof at El. 55' 8". The location of the condenser is in the path of the hot exhaust from the feed pump ventilation system. This causes reduced condenser performance. A permanent 4' x 4' aluminum sheet barrier was installed on the condenser frame as an air deflector and this action resulted in improved system performance. The HVAC Unit along with its associated air cooled condenser is used to provide condition air to the MUX room to maintain the room at design temperature and humidity level.

This modification did not interface with any safety related system or components and did not involve a change to plant operating procedures. No unreviewed safety questions existed.

Modification: **SE-000823-009, R0**, A&B Battery Room Outlet Lo Flow SW Modification

The scope of this modification was to:

1. Replace the 0.25 inch tubing from the A&B Battery Room low flow switch to the duct with a pitot tube.
2. Install 0.25 inch tubing from the pitot tube to the high/low port of the existing low flow switch FSL-823-1032.

The function of the existing low flow switch was not altered. The modification did not constitute an unreviewed safety question and does not have any adverse effect on safety or the environment. No change to the Technical Specifications were required.

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Modification: SE-**000823-010, R0**, 480 Volt "A" Switchgear Room Fan Control

The scope of this modification was installation of a 2 position, maintain contact, auxiliary transfer switch (PNL-823-11RCS6) located in the Control Room Panel (11R). The first position is OFF for normal operation. The second position is HALON TRIP BYPASSED that actuates a new time delay relay to be located in the 480 Volt 'A' Switchgear room's Local Shutdown Panel (LSP-1A2). The relay immediately isolates the faulty circuit and, after 2 seconds, closes contacts that will re-establish the power through a new alternate fuse and re-establish the function of the seal-in circuit around the Halon contact. The operator can then restart the fans from Control Room Panel (11R). An additional contact of the auxiliary transfer switch was wired into the alarm circuit of the existing HVAC trouble window U-7-a to indicate when the switch is actuated and that the Halon trip contact has been bypassed.

In addition to the new control room switch, new fan starter fuse, and the new time delay relay, minor rewiring of the 'A' Switchgear room, Local Shutdown Panel (LSP-1A2) was also completed.

This modification did not adversely affect plant safety or involve an unreviewed safety question. This modification was implemented as written, since it did not affect safe plant operation or nuclear safety.

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Modification: **SE-000834-006, R0**, Pinning Open Vortex Dampers EF-1-16/17

This modification assures maximum negative pressure in the Old Rad Waste Building by pinning the Vortex dampers in the full open position.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Modification: **SE-000837-003, R0**, Modification AOG Vent Particulate & Iodine Radiation Monitors

The purpose of this modification was to remove the on-line monitoring capability for the AOG Ventilation Radiation Monitoring System for the Particulate and Iodine channels and replace them with a "Grab Sample" type of collection method similar to the RAGEMS system. The "Grab Sample" type of collecting a sample for Particulate and Iodine is much simpler and assures a more accurate sample then the current method of using traveling paper tape and a non-standard charcoal collection method.

In addition, two other radiation monitoring channels (OG-RI-0075 and OG-RI-0066) that existed in the AOG system, but provided no real function, were deleted. OG-RI-0066 was the radiation monitor to the inlet of the charcoal adsorbers and OG-RI-0075 was the outlet of the charcoal adsorbers.

This modification did not increase the consequences of any accident or malfunction, or the possibility of an accident or malfunction different from that already addressed in the FSAR. There were no unreviewed safety questions resulting from this modifcation.

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Modification: **SE-000852-023, R0,** Replacement of In Line Filter (FCN C102301)

A replacement in-line filter was installed in the control air supply line to the 1-6 sump isolation valves. The function of the regulator is to reduce the pressure and maintain (regulate) an actuator pressure of approximately 30 psig. Since the filter is installed close to the 4" instrument air header (which has a normal pressure of 80 psig and minimum pressure of 60 psig), the pressure after the installed filter (inlet to the pressure regulator) will always be greater than the output pressure of 30 psig even with a filter dP of 10 psig (assumed conservatively). When the inlet pressure of the regulator is greater than the output pressure, proper operation of the regulator is assured. Thus, the function of the actuator and safe operation of the plant is maintained.

Since the replacement filter is installed at the end user (not in the instrument air main header), the overall performance of the IAS is not affected by the replacement filter. Therefore, safe operation of the plant is maintained.

This activity did not change any plant equipment, plant system, plant technical specification, or the FSAR. Existing margins of safety defined in the plant technical specifications or FSAR were not reduced as a result of this activity. Nor was the probability of occurrence or consequence of an accident previously evaluated in the SAR affected by this activity. Therefore, no unreviewed safety questions existed from this activity.

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Document: **SE-000852-025, R0** - FCN C088979

The purpose of the safety evaluation was to assess the impact of the removal of regulator valve V-6-2280 from the flow diagram.

This as-found FCN demonstrated that the stator winding liquid cooling unit had been operating for quite a while without V-6-2280. V-6-2280 is a regulating valve. As indicated in the previous revision of the flow diagram, V-6-2280, was installed in series with another regulating valve (V-6-2278). The function of V-6-2280 is to ensure that the required output pressure is maintained. In other words, V-6-2280 is the redundant regulator in that when the upstream regulator fails (fail open), the downstream valve will maintain the required output pressure. Therefore, removal of the downstream regulator does not alter the function or output pressure of the upstream valve. Proper operation of the liquid cooling unit is maintained and safe operation is assurred.

Document: **SE-000852-043, R0**, Field Change Notice

FCN C101341 requested the Service and Instrument Air System flow diagram BR 2013 sheet 1 be changed to show V-6-3491 downstream of exiting valve V-6-3187. BR 2013 is referenced in Section 9.3.1.2 of the FSAR.

V-6-3491 is part of a one and a half inch hose connection line coming off the air supply line between the post-filters and the compressor unloading system. Existing valve V-6-3187 is part of the same hose connection line and is upstream of V-6-3491.

The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report was not created. Adding an additional isolation valve to the existing hose connection line did not change the function or margin of operation of any structures, systems or components and, therefore, the possibility for an accident or malfunction of a different type than previously evaluated in the Safety Analysis Report could not be created. The Service and Instrument Air System is not referenced in any Technical specification basis.

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Document: **SE-000852-046, R0,** Drawing Change FCN C076125

FCN C076125 addressed changes to drawings of a system that performs a safety related function. The specific change is the accurate depiction of air inlet control valves, and air operator control valves on drawings listed on the subject FCN.

It is concluded that the subject change did not have any adverse effect on nuclear safety, safe plant operation or the environment. This modification did not constitute an unreviewed safety question as determined by 10 CFR 50.59.

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Configuration Change: **SE-000856-008, R**: Batch Tank Pump Air Supply

Supply air valves (SL-SOV-0414 and SL-HV-0413) to the Batch Tank Pump (SL-P-008) were located in the NRW Fill Aisle. This is a locked high radiation and high contamination area. Furthermore, the air hose going to this pump was only one-half inch. The change deletes these valves and installs a compressed air distribution manifold so that air will be controlled from a standard air regulatory/dryer/oiler located in the Truck Bay Control Area.

This evaluation has determined that the change did not have an adverse affect on nuclear safety or safe plant operations. Furthermore, it did not involve an unreviewed safety question or require a change to the Technical Specifications.

Modification: **SE-000862-007, R0**, EDG System Fuel Transfer Pipe Modification

The purpose of this document was to evaluate the safety of modifying the pipe supports and their locations in the EMERGENCY DIESEL GENERATOR (EDG) FUEL TRANSFER piping around valve V-39-24 located at the south end of the EDG building 2.

Calculation C-1302-862-E310-006 was generated to justify that the support modification satisfies the design criteria for the power plant.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification. This modification did not adversely effect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it was implemented under 10 CFR 50.59.

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Modification: **SE-000871-021, R0,** Dp Instrument Bypass/Demineralizer

The scope of this modification was to replace the leaking 3-way manifold valve (V-10-0422) of system 871 that is across the Differential Pressure Indicating Switch (PDIS-521-0007). The Differential Pressure Indicating Switch provides indication of pressure difference across the carbon filter and the Demineralizer trailer.

The modification does not adversely affect plant safety and does not involve an unreviewed safety question. This modification was implemented as written since it did not affect safe plant operation or nuclear safety. No changes to Technical Specifications were required.

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Document: **SE-000882-005 R0**, Top Guide Sample Shipment Using GE Series 100 Cask

Three samples of highly irradiated top guide beams were transported from the Oyster Creek Spent Fuel Pool (SFP) to a laboratory off site. These samples were transported in a General Electric Model Series 100 cask.

The movements of this cask were accomplished in a safe manner and in accordance with all licensing requirements. Lifting and transferring of the G. E. Series 100 cask from the RB floor elevation 23', to the operating floor, next to the CDPS and back to the RB floor elevation 23' had been evaluated. NUREG-0612 Phase 1 requirements have been used as a guideline for review of heavy load control measures.

This Safety Evaluation determined that this activity did not 1) adversely affect nuclear safety and safe plant operations, 2) increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR, 3) create the possibility for an accident or malfunction of a different type than any previously identified in the SAR or 4) reduce the margin of safety as defined in the basis of any Technical Specifications. Furthermore, this Safety Evaluation determined that no unreviewed safety questions were created and no environmental impacts were involved. The proposed activity did not violate any licensing requirements, cause any unreviewed radiological concern, and did not affect the plant's discharge permit condition.

Document: **SE-000900-007, R0**, GL 89-10 MOV Stem Factor and Rate of Loading

This evaluation documented selection and basis for the design stem friction coefficient, Rate of Loading adjustment and stem factor degradation adjustment. In addition, the EPRI Performance Prediction Program reports which dealt with stem friction were reviewed to ensure information which EPRI uncovered through their extensive test programs are properly considered within the OC MOV Program. This evaluation directly applies to the OC GL 89-10 MOV Program and may be applied to other MOVs as well.

This safety evaluation has determined that this change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Document: **SE-000900-008, R0**, Engineering Evaluation 125.1

This engineering evaluation documented the selection and basis for the design valve factor for certain OC gate valves. This evaluation applies only to those Oyster Creek Generic Letter 89-10 MOV Program MOVs listed in 125.1 Engineering Evaluation 0235-97.

This safety evaluation has determined that this modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Procedure: **SE-311004-001, R1**, Rated Power Operation at Reduced Feedwater Temperature

The Procedure was changed to allow Oyster Creek operation at rated thermal power with reduced feedwater temperature. Feedwater temperature is reduced by isolating the extraction steam to the high pressure feedwater heaters. The reduced feedwater temperature increases core inlet subcooling adding positive reactivity. With the increased subcooling, more heat must be used to raise the coolant entering the core to saturation. Therefore, overall efficiency will decrease.

Operation of Oyster Creek at rated thermal power with the extraction steam to the high pressure and intermediate pressure feedwater heaters removed is acceptable for Cycle 17. This activity has not adversely affected the reactor vessel integrity and the performance of reactor vessel internal components. Analyzed accidents and transients were reevaluated in this condition and determined to be bounded by previous analyses for Cycle 17. Therefore, this activity does not affect nuclear safety or safe plant operation and did not involve an unreviewed safety question.

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Document: **SE-315101-014, R0**, Valve Positions (FCN C101044)

FCN C101044 requested that flow diagrams GE 237E726 (Sheets 1 and 2), GU3#-243-21-1000, JC 147434 (Sheets 2 and 3), GU3E-570-A1-001 (Sheets 1 and 2), GU 3E-243-A1-001, and the Oyster Creek Valve List depict the valve open/shut configuration for valves V-6S-0137, V-6S-0138, V-22-752, V-22-753, V-22-792, V-22-794, V-26-24, V-27-6, V-27-8, and V-28-70, that reflects the normal operating condition of the system as defined in the system operating procedure. In this case, that was "locked closed." It also requested corrections in equipment and flow diagram size numbers and the addition of flow reference arrows for continued lines.

Document: **SE 315101-014, R0**, Valve Positions (FCN C101044) (Cont'd.)

Incorporation of this information on the flow diagram does not adversely effect nuclear safety or safe plant operations because the functions of the system have not changed. Plant operation is controlled by operating procedures. No unreviewed safety questions exist.

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Modification: **SE-315403-038, R2**, Provide Additional Cleanup System Break Detection

This modification included an additional break detection system for the isolation of cleanup system line breaks. The cleanup system had break detection using the reactor Lo-Lo level signal to initiate system isolation. However, when feedwater remains available the trip setpoint may not be reached. To ensure that these conditions do not prevent the automatic isolation of the cleanup system following a line failure, an alternate means of break detection was installed.

This modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Procedure: **SE-315600-001, R0**, Computer Program Control

As a result of QDR - 971012, it was determined that all Engineering procedures and subsequent revisions require a safety review. The procedure Computer Program Control, EP-007, implements the OQA program requirement for the control of computer software. It had not previously received a safety review. The procedure is currently in its fifth revision and this safety evaluation applies to revisions zero through five.

EP-007 provides a method for controlling software used for numerical analyses. The safety review looked at the requirements for software control and how EP-007 interacts with the calculation, verification and safety review process. It was concluded that EP-007 did not allow for any changes to the plant without going through the appropriate review processes and therefore the procedure did not involve an unreviewed safety question.

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Modification: **SE-320006-001, R0**, Reactor Cavity Trough Access Plant Modification

This modification included the following changes:

- removing from service and blind flanging the 2 inch reactor cavity concrete trough drain
- replacing the existing 7 inches x 27 inches flat trough access plate.

The removal from service of the 2 inch steel trough drain and blind flanging it and replacing the existing 7 inch x 27 inch plate of same dimensions without a drain connection was considered to be safe and justified and did not adversely affect nuclear safety, and plant operations and did not create a new postulated accident.

Modification: **SE-320006-002, R0**, Reactor Cavity Trough Access Plate Temporary Modification

The purpose of this Safety Evaluation was to show that no safety impact was caused by the temporary removal of the 7 inch x 27 inch plate at the bottom of the Reactor Cavity Trough and the temporary installation of a 7 inch x 27inch plate with two 3 inch temporary flange connections, as well as associated hoses and valves from the temporary plate to elevation 119 feet.

The reason for the temporary installation was to allow for the concrete trough (below the cavity) inspection and possible minor maintenance while the cavity was flooded.

The temporary modification of the temporary plate installation and its associated valves and hoses was performed safely and helps in identifying reactor cavity leakage such that drywell shell corrosion margin is maintained.

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Procedure: **SE-320006-003, R1**, Application of Strippable Coating on Reactor Cavity Liner

The coating work does not reduce the performance of the affected systems, affect the safety functions of these systems, increase the probability of occurrence or consequence of an accident, create a possibility for an accident, decrease the margin of safety, (as defined in the bases of Oyster Creek Technical Specifications), violate any licensing requirements, cause a radiological concern and does not affect the environmental permits. Therefore, it is not an unreviewed safety question.

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Modification: **SE-328218-004, R1**, Reactor Building Penetrations

The purpose of this modification was to provide two (2) 6-inch diameter piping penetrations through the north wall of the Reactor Building. The modification also included provisions for the potential installation of a third 6-inch diameter piping penetration through this same wall. These penetrations were currently to be used for temporary piping/hoses for the torus $H_2O_2$ connections. In addition, this modification provided one (1) 8-inch diameter penetration through the Reactor Building south wall. This penetration is currently used to support the feedwater nozzle inspection project.

Modification: **SE-328218-004, R1**, Reactor Building Penetrations (cont'd.)

Revision 1 of this safety evaluation addressed the additional 6" diameter penetration in the North wall 4" and 8" diameter penetrations in the South wall and 10" penetration on the East wall of the Reactor Building installed earlier for torus modifications in accordance with Burns and Roe drawing no. S712, Rev. 3, GPUN Installation Specification OC-TS-402023-001, Revision 1, and FCN C120103.

The penetrations installed by these modifications were sealed. Since these penetrations are intended for general use, the safety evaluation for each modification using the penetrations addresses their task specific functions.

These modifications provide pipe sleeve penetrations through the Reactor Building walls. The gaps between the core bores and the pipe sleeves were sealed by grouting and caulking and the sleeves were capped at both ends with gasketed blind flanges or other sealing methods to maintain the air seal condition. In addition, a temporary enclosure was attached to the inside face and sealed all around to minimize the amount of potential air leakage into the Reactor Building during installation. As a result, the existing function of "Secondary Containment" is not affected.

It was concluded that this modification had no adverse affect on safety or environmental impact.

Modification: **SE-328333-003, R1,** Emergency Service Water Orifice Relocation

The purpose of this document was to evaluate the safety of removal of orifices RO 21A through D from their present location and replacing them with a single orifice in the common line downstream of V-3-88 in system 1, and V-3-87 In system 2.

The existing orifices were mounted directly on the Containment Spray Heat Exchanger nozzles at the inlet of an internally Coated carbon steel elbow. In this location the orifices had caused failure of the elbow coating and subsequent damage to the elbow. Further, the existing orifices were oversized for the present method in which the system is operated. This required that the heat exchanger outlet butterfly valves be throttled severely. The position of the valve disc causes premature failure of the coating directly downstream of the valve with subsequent degradation of the pipe. The replacement orifice is sized to permit the valve to be opened wider and therefore reduce or eliminate this failure mechanism.

This modification removed orifices RO 21A, -B, -C, and -D and inserted spacers in their place. A new orifice was installed downstream of valves V-3-87 and V-3-88. The pipe directly downstream of the new orifices (the recovery zone) was replaced with stainless steel pipe. This was to obviate the need for internal coating in an area where high velocity flow is expected. In the elbow directly upstream of the new orifices, an inspection and cleaning port was provided to provide relatively easy access to the new orifices without the need to remove it from the pipeline.

This modification relocated orifices within the ESW system. It was demonstrated that this modification does not adversely effect nuclear safety or the environment. No unreviewed safety question was generated by this modification and it was implemented under 10 CFR 50.59.

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Test or Experiment: **SE-328404-001, R0**, Zinc Addition to the Spent Fuel Pool

The Oyster Creek Nuclear Generating Station spent fuel pool uses Boraflex sheets within the spent fuel pool racks. Boraflex is a rubber-like material containing boron used to provide adequate neutron absorption to maintain the spent fuel pool neutron multiplication factor ($K_{eff}$)below 0.95. When Boraflex is subjected to gamma radiation and water in the spent fuel pool it can degrade through dissolution. Test data has indicated that exposing Boraflex to 1 ppm solution of zinc for periods up to 60 days reduces the dissolution rate of irradiated Boraflex.

Zinc acetate was added to the fuel pool water using NETCO Procedure, to achieve a zinc concentration of approximately 1 ppm. The zinc acetate remained in the fuel pool for approximately 60 days. It was then removed using the fuel pool filter and/or demineralizer. The zinc was depleted in zinc-64 (<1% Zn-64) to minimize the potential for forming radioactive zinc-65.

This change did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Modification: **SE-400001-001, R1**, GL 89-10, 17R Cable Conduit

The purpose of this modification was to provide greater available valve actuator motor torque for Isolation Condenser System valves V-14-30 and V-14-36 by increasing the voltage available at the motor terminals.

This modification did not introduce any new accident or malfunction not previously evaluated, nor does the modification increase the likelihood of occurrence or consequences of any accident as analyzed in the UFSAR. This modification did not decrease the margin of safety as described in the Technical Specification because the modification does not impact any system safety functions. This evaluation concluded there was no unreviewed safety question per 10 CFR 50.59 and no changes to the OC Technical Specifications were required.

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Configuration Change: **SE-400011-001, R2,** Replacement of the 1-2 Fire Pump Diesel

This modification replaced the 1-2 Fire Pump Diesel and the associated Controller in the fire protection pumphouse. The configuration change also includes the addition of a 60 second time delay on the "Loss of Power" diesel start signal

The intended replacement of the 1-2 fire pump diesel did not involve any unreviewed safety questions. This modification did not result in a change to the Oyster Creek Technical Specifications and their basis.

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Modification: **SE-400025-001, R2,** Separation of Blower #1 from the Heating Steam System

The configuration change isolated several connections to the existing #1 boiler so that a new boiler could be installed with the #2 Boiler operating. The changes associated with the aforementioned CCD are as follows:

- The change to the #1 boiler steam line replaced and reoriented 8-inch valve V-13-62 to allow connection of the steam line from the new package boiler. This provided positive isolation from the steam line. In addition, the new heating steam boiler capacity did not provide the minimum required velocity for operation of existing 8-inch swing check valve SH-CKV-182. This configuration change (per ECD C306510) removed the internals of the check valve to ensure proper operation of the Heating Steam System after the new boiler was installed.
- Isolation from the Dearator was accomplished by removing piping between the Boiler Feed Pump Discharge Header and valve V-13-346. The pipe between valves V-13-346 and V-13-347 was cut. A cap was placed on the pipe end, west of the cut (V-13-346 side). The 1 1/2 inch pipe header from Boiler Feed Pumps CH-P-4A & 4B (downstream of valves CH-HV-163A & 163B) was cut. The 1 1/2 inch piping from the cut to and including valve V-13-347 was removed. This provides isolation from the Dearator. It maintains the connection from the Heater Boiler Condensate Storage Tank (T-13-2) to the Dearator via pumps CH-P-5A/5B.
- The boiler exhaust stack flow was blanked off at damper SH-DM-2 to isolate the old #l boiler from the common stack so that the #2 boiler can operate with #1 abandoned in place. Damper SH-DM-2 was removed and a steel plate installed in its place. The dampers associated limit switch was removed to allow clearance for the removal of the damper.

This modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

Modification: **SE-400025-003, R1**, New Package Boiler SH-B-001 Installation and Tie-In

This document evaluated the nuclear safety aspects of the installation and tie-in of the new unit 1 boiler (SH-B-00 1) and enclosure that was located southeast of the boiler house. The existing unit 1 boiler has been isolated and abandoned in place per OC-CCD-400025-001 and SE-400025-001. The existing unit 2 boiler remains operational.

This modification did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.

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Modification: **SE-400025-006, R1**, Heating Steam System #1 Boiler Replacement

The purpose of this modification was the installation of the electrical and control modification required for the new package boiler, SH-B-001, associated with the replacement of Unit #1 Heating Boiler, B-13-001. The Unit #1 Heating Boiler, B-13-001, was abandoned in place. The Unit #2 Heating Boiler, SH-13-002, remains operational and was not changed by this modification.

The electrical and control changes were installed by this modification. The piping and boiler installation was completed under separate CCDs and/or MDs. The isolation of the existing Unit #1 boiler was performed under OC-CCD-400025-001.

The replacement of the Unit #1 Boiler with the new packaged boiler did not adversely affect plant safety or safe plant operation and did not involve an unreviewed safety question as determined by 10 CFR 50.59.

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Modification: **SE-400035-001, R2**, Isolation Condenser Tube Bundle Replacement

This modification replaced the original Isolation Condenser tube bundle with a new bundle manufactured to be equivalent or better in the form and function utilizing upgraded materials to be more resistant to intergranular stress corrosion cracking (IGSCC).

There was no unreviewed safety question or adverse effect to nuclear safety or safe plant operations associated with the subject design change to replace the Isolation Condenser Tube Bundles. The pressure integrity of the bundles is ensured and the specified heat removal capacity of the bundles is maintained. The Isolation Condenser System continues to be capable to perform its safety function to remove decay heat for all analyzed conditions for which the design basis requires it to perform and no additional mechanism which could create the possibility for a new accident or malfunction or increase the probability of occurrence or consequences of an accident or malfunction previously evaluated is introduced. There is no reduction in the margin of safety as defined in the UFSAR or Technical Specification Bases.

Modification: **SE-403042-001, R3**, Thermolag Cable Raceway Fire Barrier Upgrades/Modifications

The purpose of this safety evaluation was to address modifications and/or upgrades to existing cable raceway fire barriers at Oyster Creek and TMI-1 as manufactured by Thermal Science, Inc. (Thermolag). The fire rating of these barriers was considered invalid. These barriers have been evaluated with actual fire ratings established. The actual fire ratings are documented in GPUN Topical Report No.'s 094 "TMI-1 Evaluation of Thermolag Fire Barriers" and 102 "Oyster Creek Evaluation of Thermolag Fire Barriers." The modifications and/or upgrades restored the original 1 hour or 3 hour ratings to selected barriers as identified in the aforementioned Topical Reports, Table 1 of Configuration Change Document OC-CCD-403042-001, Rev. 0, and Table 1 of Configuration Change Document, TI-CCD-417109-002, Rev. 0. The requirement for 1 hour or 3 hour ratings on specific cable raceways is described in the Oyster Creek and TMI-1 Fire Hazards Analysis Reports (FHAR).

This evaluation demonstrated that the configurations designed to upgrade or modify (replace) existing Thermolag cable raceway fire barriers will restore required fire barrier ratings to said raceways. Proper design consideration has been given to the increased load on raceway supports due to the additional fire barrier material and the increased derating effect on protected circuits. Therefore, this modification is acceptable as-is and does not result in an unreviewed safety question. No changes to Technical Specifications were required.

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Evaluation: **SE-403042-002, R0**, Evaluation of Ductwork Fire Barriers

This change eliminated the requirement for one hour fire barrier protection for supply and return ductwork for the "A" 480 Volt switchgear room (herein referred to as Fire Zone OB-FZ-6A) ductwork. This ductwork was provided with "one hour" fire barriers manufactured by Thermal Science, Inc. (Thermolag) where it passes through the "B" 480 Volt switchgear room and adjacent corridor (herein referred to as Fire Zone OB-FZ-6B). The purpose of the protection is to preserve ventilation for Fire Zone OB-FZ-6A as a support system for equipment operability there in the event of a fire in Fire Zone OB-FZ-6B.

The construction of the OB-FZ-6A ventilation system ductwork in Fire Zone OB-FZ-6B is sufficient to insure safe shutdown in the event of a fire in either Fire Zone. The existing arrangement is therefore acceptable and would also be acceptable if the fire barrier material is removed. Therefore, the existing ductwork fire barrier material is acceptable as is and does not constitute an unreviewed safety question. No changes to the Technical Specifications were required.

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Modification: **SE-408847-002, R1,** Emergency Diesel Generator Data Acquisition

The purpose of this modification was to install a data acquisition and performance monitoring system for each of the Emergency Diesel Generators (EDGs).

This modification added a new data acquisition and performance monitoring system for each of the Emergency Diesel Generators, installed a fuel tank room heater and provided electrical power for these systems. It was concluded the modification did not constitute an unreviewed safety question as determined by 10 CFR 50.59 and does not have any adverse effect on safety or the environment.

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Document - Procedure: **SE-945100-010, R3**, Calculations

Revisions 3, 4, and 5 of EP-006 which provide requirements for methods and responsibilities for documentation and control of calculations do not involve an unreviewed safety question nor a technical specification change.

Procedure: **SE-945100-099, R0**, Methodology for Preparing the Quality Classification List

The procedure, EP-011, and the changes in revision 5 provide the programmatic controls to implement the requirements in the GPUN OQA Plan and the PDMS QA Plan for Quality Classification of items. These classifications provide the basis for application o the applicable QA requirements which will provide reasonable assurance that the items will perform their intended safety function. Based on the results of this review, this document enhances the programmatic controls which affect nuclear safety and safe plant operations, and there is no unreviewed safety question associated with this procedure or the revision. Additionally, no changes to the Technical Specifications were required.

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Document: **SE-945100-115, R0,** Use of SQUG Methodology

On May 22, 1992, the USNRC issued Generic Letter 87-02, Supplement 1, which included the NRC Supplemental Safety Evaluation Report (SSER-2) on Revision 2 of the Seismic Qualification Utilities Group (SQUG) Generic Implementation Procedure (GIP). The GIP provides a detailed technical approach to verify the seismic adequacy of mechanical and electrical equipment installed in nuclear power plants to address concerns identified in Unresolved Safety Issue USI A-46. In SSER-2, the USNRC states that the GIP approach to resolution for USI A-46 provides an acceptable evaluation method for verifying the seismic adequacy of equipment in USI A-46 plants including modifications and new and replacement equipment. SE 000200-003, R0, assessed a change to Oyster Creek's FSAR to permit the use of the SQUG methodology as presented in GIP Revision 2 and supplemented by the USNRC SSER-2 as an acceptable method to verify the seismic adequacy of existing, new, modified and replacement equipment installed at Oyster Creek.

EP-050 is a new Engineering Procedure, which discusses the requirements, method, controls and responsibilities for utilizing the SQUG methodology at GPU Nuclear in a manner consistent with GPU Nuclear's Operational QA Plan and the FSAR of Oyster Creek

EP-050 provides the necessary administrative controls for implementing the SQUG Methodology at Oyster Creek and the generation of SQUG documentation. These controls and the implementation of this procedure do not involve an unreviewed safety question nor require any changes to the Technical Specifications.

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Document: **SE-945100-164, R0,** Realignment of STA Function to Plant Operations

On or about January 1, 1998, a change in the reporting relationship of the STAs will occur. The STAs currently report to the Manager, Shift Engineering. This change transfers the STAs to Plant Operations. The change will affect both figures in Chapter 13 of the Oyster Creek UFSAR. The safety evaluation considers the transfer of responsibilities.

This safety evaluation has concluded that this activity does not involve a significant increase in the probability of occurrence or consequences of an accident previously considered; and does not increase the probability or consequences of a malfunction of equipment important to safety; does not create the possibility of an accident or malfunction of a different type than previously evaluated; and does not decrease the margin of safety as defined in any Technical Specification. Therefore, an unreviewed safety question did not exist and implementation of the change and its description in the FSAR are acceptable.

EP-101 is a new procedure written to provide a formal process for the conduct of fuel reload activities. Since the process will govern the control of plant configuration changes (i.e. reload fuel core loading patterns), the procedure has the potential to impact nuclear safety and safe plant operation. The procedure is intended to ensure that plant configuration changes resulting from fuel and core consumable replacement are performed in a prescribed manner that is consistent with OQA plan requirements for plant modifications. The procedure incorporates requirements from existing Fuel Standards and recommendations from INPO SOER 96-02. It was determined that Fuel Standards are not appropriate documents for controlling processes that could impact plant configuration. The conduct of fuel reload activities includes core loading configuration as well as loading new fuel designs and other core consumables (control rods, channels, etc.).

It was concluded that EP-101 did not allow for any changes to the plant without going through the appropriate review processes and therefore the procedure does not involve an unreviewed safety question nor does it adversely impact nuclear safety and safe plant operation.

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Safety Evaluations - UFSAR Update 11

Numerous changes to the FSAR were incorporated in UFSAR Update 11 previously submitted on April 26, 1999. All changes were evaluated with regard to 50.59 as a part of the update process.

In all cases these changes did not (1) adversely affect nuclear safety and/or safe plant operations, (2) increase the probability of occurrence or the consequences of (a) an accident or (b) malfunction of equipment important to safety previously evaluated in the SAR, (3) create the possibility for (a) an accident or (b) a malfunction of a different type than any previously identified in the SAR or (4) decrease the margin of safety as defined in the basis of any Technical Specification.