

Enclosure to  
NMP2L 1359

**NINE MILE POINT - UNIT 2**

**SAFETY EVALUATION SUMMARY REPORT**

**1992**

Docket No. 50-410  
License No. NPF-69

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Safety Evaluation No.: 87-124  
Implementation Document No.: Mod. PN2Y87MX078  
USAR Affected Pages: Table 11.4-4  
System: Solid Radwaste  
Title of Change: Replace Waste Concentrate and  
Waste Sludge Pumps

**Description of Change:**

The waste concentrate and waste sludge metering pumps are part of the radioactive waste solidification system equipment. The original ROPER pumps did not perform their metering function accurately and required frequent and lengthy maintenance in a high radiation area. This modification replaced the ROPER pumps with NETZSCH pumps for the waste concentrate metering pumps and NORTHERN pumps for the waste sludge metering pumps.

**Safety Evaluation Summary:**

The solid radwaste (WSS) system does not perform a safety-related function nor is it required to effect or support safe shutdown of the plant. This modification only replaced originally installed pumps with new pumps and did not change system function or design.

The new pumps have fewer maintenance requirements and will result in a significant saving in radiation exposures over the life of the plant.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 88-025, Rev. 1  
Implementation Document No.: Mod. PN2Y89MX024  
USAR Affected Pages: Figures 10.1-5c, 10.4-10  
Sht. 23  
System: Condensate  
Title of Change: Remove Seal-In Circuit for  
2CNM-MOV32A, B, C

**Description of Change:**

Motor-operated valves 2CNM-MOV32A, B, and C are low-pressure (LP) heater string outlet isolation valves. These valves automatically close on a high-high water level in the low-pressure heater strings. This modification permanently removed the seal-in circuits of these valves to prevent full closure of the valves and isolation of the LP heater strings on a spurious high-high water level signal. This change replaced a previously-installed temporary modification, which was reported in letter NMP2L 1239, dated June 11, 1990.

**Safety Evaluation Summary:**

The condensate and feedwater heating system is nonsafety related and is not required for safe operation or shutdown of the plant. The removal of the seal-in feature does not inhibit the complete closure of the LP heater string inlet and outlet isolation valves in the event of a sustained high level condition. The design function of the high level trip is still maintained.

This modification reduces the challenges to ECCS by maintaining feedwater to the reactor vessel when spurious signals from feedwater heater level switches are received.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 88-084  
Implementation Document No.: Temp. Mod. 88-231  
USAR Affected Pages: N/A  
System: Plant Hot Water Heating (HVH)  
Title of Change: Temporary Change to Hot Water Heating Makeup Control

**Description of Change:**

The makeup level control for the plant hot water heating system expansion tank (2HVH-TK1) has a range of  $\pm 6$ " at the centerline of the tank. This is controlled by limit switch 2HVH-LS114. As identified in Problem Report 08101, the pressure increase caused by the compressed nitrogen (when level is increased to normal level) causes a release of nitrogen when the pressure exceeds the relief valve setpoint of approximately 300 psig. This results in excessive nitrogen usage for the 12-inch band. This temporary change reduced the control band to  $\pm 7/16$  inch. This was designed to maintain the desired water level in the expansion tank and greatly reduce the nitrogen consumed in level control.

A functional test of the level control will be performed and a permanent change will be implemented through the modification program based on the test results.

**Safety Evaluation Summary:**

This temporary change revised the operating range of the nonsafety-related hot water heating makeup system. Implementation of this change was intended to verify acceptability of the revised level control range. This will not affect the operation of any safety systems or affect the safe operation or shutdown of the plant.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 89-044, Rev. 1  
Implementation Document No.: Mod. PN2Y89MX100  
USAR Affected Pages: Figures 1.2-21 Sht. 1, 9A.3-7,  
12.3-17, 12.3-50  
System: N/A  
Title of Change: Stair Installation in Lieu of  
Ladder, El. 320'-3 1/4"  
Screenwell Building

**Description of Change:**

In order to provide a safer and more convenient means of accessing the screenwell building HVAC equipment room for grabbing samples, the ladder for the el. 320'-3 1/4" platform was removed and replaced with a set of stairs with landing. The stairs were prefabricated and meet or exceed OSHA federal standards.

**Safety Evaluation Summary:**

The platform and ladder are nonsafety related and are located in the screenwell building at el. 320'-3 1/4". Installation of stairs in lieu of a ladder will not block or hinder means of egress from the area or access to plant equipment, and does not affect the structural integrity of the platform. This change will not adversely affect plant operation or safe shutdown of the plant.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 89-057, Rev. 1  
Implementation Document No.: Procedure N2-RTP-123  
USAR Affected Pages: 11.5-7; Figure 11.5-6  
System: Digital Radiation Monitoring  
(DRMS)  
Title of Change: Removable Charcoal Filter  
Cartridges in CAMs

**Description of Change:**

Continuous airborne radiation monitors (CAMs) 2HVW-CAB195, 2HVW-CAB196, 2HVW-CAB197, 2HVW-CAB199, 2HVR-CAB229, 2HVR-CAB237, and 2HVR-CAB238, and 2HVT-CAB206 are used to monitor process ventilation systems and local area ventilation activity conditions. Licensee Event Reports (LERs) 86-03-01 and 86-11-1 documented two instances of problems involving degradation of the CAMs' removable charcoal filter cartridges. This modification addressed operation of the subject CAMs without the filter cartridges installed. Occasional re-installation of the filter cartridges to obtain grab samples was also assessed.

**Safety Evaluation Summary:**

This modification improves the availability and reliability of the CAMs by preventing the situation (degraded charcoal filter cartridges) which has previously resulted in erroneous low flow alarms and increased maintenance.

The operation of the CAMs without the charcoal filter cartridges does not affect component operation or compliance with applicable regulations. The capability to demonstrate compliance with 10CFR20.103 and 10CFR20.201 is provided by low volume air sampling and grab samples. The CAMs are monitors required for plant operation. They have no automatic control or accident mitigating functions and are not required for plant safety.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 89-075, Rev. 6  
Implementation Document No.: Mod. PN2Y87MX038  
USAR Affected Pages: Figure 9.5-7  
System: Communications  
Title of Change: Addition of Communication Equipment

**Description of Change:**

This modification added Gaitronic/communication capabilities in various plant areas by adding phone jacks, speakers, speaker volume controls, handsets, strobe lights, associated wiring and conduit, and administrative controls as required. The modification satisfied the commitments addressed in LER 87-025, and incorporated improvements identified from system verification testing, site operating experience, and an NRC Emergency Preparedness Exercise Inspection (10/29/86). Revision 5 of this safety evaluation was previously reported in letter NMP2L 1324, dated October 30, 1991. Revision 6 of this safety evaluation was prepared to address the conditions under which this modification can be made permanent. These conditions are: (1) replacement of 2VBB-UPS1C (to be accomplished under a separate modification), and/or (2) removal of sufficient load from 2VBB-UPS1C to accommodate the new communications equipment loads. In addition, the installation of new Gaitronic handset stations in the main control room area has been completed.

**Safety Evaluation Summary:**

This modification enhances communication capabilities for the performance of surveillance testing, enables personnel to respond to alarms in areas with inherently high noise levels, and adds communication equipment in areas that have been identified as needing communication capabilities. These changes do not diminish the capability of the plant communication systems to provide effective and reliable communications capability necessary for plant personnel during times of: (1) plant accidents and transients combined with total loss of offsite power, and (2) use of the remote shutdown panel for a plant shutdown.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-026  
Implementation Document No.: Mod. PN2Y88MX047  
USAR Affected Pages: Figures 1.2-21, Sht. 1, 9.5-29,  
9.5-32, 12.3-17, 12.3-50  
System: Turbine Building  
Title of Change: Turbine Building Tool Room at  
Elevation 306'

**Description of Change:**

This modification consisted of the design and construction of a tool room, including a mezzanine, in the equipment laydown area at el. 306' in the turbine building. The tool room will be utilized when the main turbine requires teardown for scheduled maintenance.

This tool room was formed from a prefabricated mezzanine (or platform) with cabinets placed under and on the platform. Lighting, Gaitrronics and 120-V outlets were provided at the new el. 306' tool room and the existing tool cage at el. 247'.

**Safety Evaluation Summary:**

The tool room and mezzanine are located in an open, nonsafety-related area in the turbine building at el. 306'. Addition of the tool room, lighting, and communications equipment does not impact any safety analyses, safe shutdown, or plant operation. The tool room and mezzanine are located in an area where radiation levels are less than 5 m/hr.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-035, Rev. 1

Implementation Document No.: Drawings EV-4K-3, EV-4L-3,  
EV-4M-1, EV-198AQ-0, EM-002G,  
EM-021M; Procedures  
N2-MMP-GEN-930, 931

USAR Affected Pages: N/A

System: Refueling Equipment

Title of Change: Reactor Internals Storage Pool  
Gate

**Description of Change:**

This change involves the purchase and installation of the reactor internals storage pool gate to support refueling outage activities. This safety evaluation was previously reported in letter NMP2L 1258, dated October 31, 1990. Revision 1 of the safety evaluation was prepared to address the consequences of using the reactor internals storage pool gate without final surface finishing (i.e., mechanical polishing) during the first refueling outage.

**Safety Evaluation Summary:**

Installation of the gate without final surface finishing creates a potential ALARA concern, as the estimated exposure resulting from decontaminating the unpolished gate could increase. However, plant safety and offsite doses will not be affected.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-053, Rev. 1  
Implementation Document No.: Mod. PN2Y89MX004  
USAR Affected Pages: 9A.3-27, 9A.3-54; Tables  
6.1-3, 9.5-3 Sht. 9,  
9A.3-18 Sht. 2; Figures  
9A.3-4, 9A.3-5, 9A.3-6  
System: Fire Protection (FPM)  
Title of Change: Smoke Detectors in the Primary  
Containment

**Description of Change:**

Originally, there were no smoke/fire detectors in the primary containment. This modification installed permanent wiring and detector mounts, during the first refueling outage, so that environmentally-sensitive detectors may be quickly installed during refueling and major maintenance periods and removed when work is complete. These "portable" detectors, when installed, will communicate with fire panel 2FPM-PNL105 for local and control room alarm and annunciation. The method of transmitting the instrument circuit from primary containment to secondary containment was through an existing spare electrical penetration.

**Safety Evaluation Summary:**

This modification satisfied a commitment made in USAR Section 9A.3.7.1.1.3 which stated that "General area smoke detectors will be provided in the primary containment only during refueling and major maintenance periods." This commitment was reviewed and accepted by the NRC based upon the fact that it would be completed by the first refueling outage, as documented in NUREG-1047, Supplement 4, Section 9.5.1.6(1).

The use of the spare electrical penetration for the fire detector instrumentation circuit is acceptable because for all instrumentation circuits, the penetrations can carry continuously the maximum short circuit current available without exceeding their thermal limit (see USAR Section 8.3.1.1.5).

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-069  
Implementation Document No.: Mod. PN2Y89MX047  
USAR Affected Pages: Figure 9.5-52c  
System: Auxiliary Boiler  
Title of Change: Installation of Mechanical  
Seals for 2ABM-PlA & B

**Description of Change:**

The auxiliary boiler recirculation pumps were originally equipped with garlock packing and lantern ring with high-pressure cooling water supply to prevent leakage. This packing/stuffing box arrangement, with the systems intermittent usage, was found to be ineffective. This modification replaced the packing/stuffing box arrangement with John Crane mechanical seals.

Unlike the existing packing, the new mechanical seals do not require the seal water to be discharged externally. The seal cooling water is discharged into the auxiliary boiler. Therefore, the external cooling water discharge piping was removed or abandoned in-place where possible. All necessary instruments (flow switch, temperature indicator, etc.) were relocated on the cooling water inlet piping, and the existing flow switch was replaced with a new high-pressure, high-capacity switch.

**Safety Evaluation Summary:**

This modification is an improvement to the system which will increase reliability and availability of the auxiliary boiler system by installing mechanical seals on the pumps and by minimizing recirculation pump maintenance. This change will not adversely impact the operation of the auxiliary boiler feed or steam portion of the system, and will not adversely affect the ability of any safety-related systems or components to perform their safety function.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-080, Rev. 1  
Implementation Document No.: Test Procedure N2-88-18,  
Rev. 3  
USAR Affected Pages: N/A  
System: Main Turbine - Generator  
Title of Change: Main Turbine - Generator Rotor  
System Torsional Screening  
Test - NMP2 (N2-88-18, Rev. 3)

**Description of Change:**

The turbine-generator torsional screening test conducted in September 1989 measured the torsional natural frequencies of the rotor train. The precise location of the mode identified closest to 120 Hz was at 119.9 Hz. Therefore, as recommended by GE, actions were taken for detuning the unit by installing an inertia ring on the turbine-generator "D" coupling which would lower the 119.9 Hz mode by 0.8 Hz. Telemetry torque collars on the turbine rotor shaft at bearing #4, #6 and #8 for strain gauge signals were also made permanent. Following the installation of the inertia ring, the torsional test was repeated to measure the torsional natural frequencies of the rotor train and verify if natural frequency had moved further away from a mode of 119.9 Hz to 119.1 Hz or better.

**Safety Evaluation Summary:**

General Electric required conducting the torsional screening test after the installation of the detuning ring to assure the functional performance, as required for continued safe operation, and to provide maximum protection to the equipment and operating personnel. The performance of the torsional screening test will not affect the safe operation or shutdown of the plant. Turbine missiles were not a concern since rotational speed was within design limits. The controlled acceleration maintains power spikes within the response capabilities of the steam bypass valves. Differential protection relays and generator ground relays will be in service to protect the unit in an unlikely event of an electrical fault. The turbine electronic backup overspeed trip will be operable to provide protection from overspeeding the unit.

Based on the evaluation performed, it is concluded that this change and test do not involve an unreviewed safety question.

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Safety Evaluation No.: 90-083, Rev. 1  
Implementation Document No.: Temp. Mod. 91-116  
USAR Affected Pages: N/A  
System: Breathing Air (AAS)  
Title of Change: Temporary Air Compressor for Breathing Air

**Description of Change:**

This temporary modification provided a temporary air compressor to supply the source for breathing air for the second refueling outage. The temporary air compressor was powered from a 480V construction power source. Installation of the temporary oil-free air compressor necessitated de-energizing the breathing air compressor, 2AAS-C1, and its related auxiliaries. In addition, the piping spool piece between the compressor and aftercooler was removed in order that the air hose from the temporary compressor could be connected to the aftercooler inlet.

**Safety Evaluation Summary:**

The breathing air system and the temporary compressor are nonsafety related and are not required for the safe operation or safe shutdown of the plant. The temporary compressor will provide the required air pressure, capacity, and air quality for use in the applicable areas of the plant. In addition, the operating conditions of the temporary compressor are compatible with the installed breathing air system design and operating parameters. The system operating pressure of 100 psig will be maintained.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-089  
Implementation Document No.: EDC 2F00140  
USAR Affected Pages: Figure 9.1-5d  
System: Fuel Pool Cooling & Cleanup  
(SFC)  
Title of Change: Spent Fuel Pool Effluent  
Conductivity Element  
Replacement 2SFC-CE46A & B

**Description of Change:**

Conductivity elements 2SFC-CE46A & B, Bechman Model No. CEL-II(SS)XI-001-N, were not functioning and replacement elements were required.

The replacement conductivity elements, Bechman Model No. CEL-I(SS)-001-N, are longer and supplied with a 1 1/4" Powell gate valve. The gate valve provides isolation so that the cell element may be removed for inspection or servicing without allowing liquid to escape.

There was a minor change in location for 2SFC-CE46A. Due to the increased length of the conductivity elements, the pipe connection was lowered three inches to avoid interferences. This new location continues to provide a good sampling point.

**Safety Evaluation Summary:**

The replacement and installation of the new conductivity elements, complete with gate valves providing isolation, enhance the system for both operation and maintenance. In addition, this replacement meets all of the original design requirements, thus maintaining system integrity. This replacement is within the cleanup portion of the SFC system. It is used primarily during the refuel mode only, and is not credited to fail or function during shutdown.

Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question.

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Safety Evaluation No.: 90-110  
Implementation Document No.: Temp. Mod. 90-073  
USAR Affected Pages: N/A  
System: DFM  
Title of Change: Temporary Replacement of  
Auxiliary Boiler Building Sump  
Pump

**Description of Change:**

Auxiliary boiler building sump #3 collects runoff from the auxiliary boiler building floor and equipment drains. With the auxiliary boilers operating during plant shutdown, sump pumps 2DFM-P3A and 2DFM-P3B are required to keep the sump from overflowing. Pump 2DFM-P3B became inoperable due to a motor failure, and pump 2DFM-P3A was operating at less than design capacity.

A temporary air-operated sump pump is currently being used to process the effluent since spare pumps or parts for the existing pumps were not available. This effluent is discharging to the radwaste building for proper processing through the existing piping, in accordance with the system design, by connecting the temporary pump's discharge line to valve 2DFM-V40. The suction line of the temporary sump pump was installed no lower in the tank than the existing pump suction elevation.

This temporary modification allows the DFM system to perform its intended function until the pump can be repaired or replaced.

The auxiliary boiler building sump pump is nonsafety related and the requirements of 10CFR50 Appendix B do not apply.

**Safety Evaluation Summary:**

The change implemented by this temporary modification does not change the function of the existing floor drain system or impact the safe operation or shutdown of the plant. This change supports the continued operation of the auxiliary boilers, as required, during plant shutdown.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-118, Rev. 1  
Implementation Document No.: Mod. PN2Y89MX132  
USAR Affected Pages: Figures 9.2-1c, 9.2-1e,  
9.2-1f, 9.2-1g, 9.2-1L,  
9.2-1m, 9.2-1p  
System: Service Water (SWP) and  
Reactor Building Ventilation  
(HVR)  
Title of Change: Unit Cooler Cleaning - Related  
to Secondary Containment  
Drawdown

**Description of Change:**

There are 33 unit coolers related to secondary containment drawdown. The capability existed to flush 6 of these units. This modification added breakout spools or connections in the service water supply and return lines and/or modified the supports of the remaining 27 unit coolers to provide the capability for flushing.

Flushing of the unit coolers will help in maintaining and improving their performance by removing some of the silt, fouling, and corrosion deposits.

**Safety Evaluation Summary:**

The original design basis of the piping and supports was maintained. Piping installation and material is ASME III Class 3 under the jurisdiction of ASME XI. The modification will not adversely affect the safety function of any of the structures, systems or components or the capability to safely shut down the plant and maintain it in a safe shutdown condition.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-120  
Implementation Document No.: Mod. PN2Y88MX028  
USAR Affected Pages: Figures 10.1-3h, 10.1-3k  
System: Main Steam  
Title of Change: Modify 2MSS-MOV199 Control  
Circuit to Delete Interlock  
from 2MSS-MOV19A,B

**Description of Change:**

Temporary Modification 2292 was implemented in October 1987 (associated Safety Evaluation 87-147) to change the control circuit for reheat steam header drain valve 2MSS-MOV199 such that the interlock with the reheater blanketing steam valves 2MSS-MOV19A,B was deleted. This enabled the condensed steam in the reheat steam header to be drained more effectively, and independent of the position of valves 2MSS-MOV19A,B.

This modification makes this a permanent change. The 2MSS-MOV19A,B valve permissives in circuit 2MSSN06 (2MSS-MOV199) are disabled by manipulating two wires within the local junction box 2-JB7370 (at turbine building el. 277').

Safety Evaluation 87-147 was previously reported in letter NMP2L 1239, dated June 11, 1990.

**Safety Evaluation Summary:**

The moisture separator reheater produces large quantities of condensate. A number of drains are provided to control and remove condensed steam. Valve 2MSS-MOV199 is used to remove condensed steam in the reheat steam header prior to placing the moisture separator reheater in service.

This modification facilitates the removal of condensed steam from the reheat steam header prior to placing the moisture separator reheater in service. It does not affect any safety-related system, structure, or component, and will not impact safe operation or shutdown of the plant.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 90-126  
Implementation Document No.: Temp. Mod. 5316 (Unit 1)  
USAR Affected Pages: N/A  
System: Technical Support Center  
Heating, Ventilation, and Air  
Conditioning  
Title of Change: Fix Damper EPN 212-41 for TSC  
Emergency HVAC

**Description of Change:**

The onsite Technical Support Center (TSC) is located in the basement of the Unit 1 administration building. The facility is common to both Unit 1 and Unit 2. A description of the change and a summary of Unit 1 Safety Evaluation 90-054 was reported in the Unit 1 1992 Safety Evaluation Summary Report.

Damper EPN 212-41 (MD-3) was hunting when the TSC HVAC system was in the emergency mode. This hunting in system flow caused inadequate flow, at times, to meet the test requirements of Surveillance Procedure N1-ST-Q9.

This temporary modification de-energized the motor actuator and mechanically restrained damper 212-41 (MD-3) at a position to provide a flow rate of 3000 cfm  $\pm$  10%. This change maintains the operability of the TSC emergency ventilation system until the design and installation of a permanent modification can be completed.

**Safety Evaluation Summary:**

The emergency ventilation system provides filtered air and maintains a positive pressure under accident conditions. The temporary modification does not adversely affect the design function of the TSC HVAC system.

The function of damper 212-41 (MD-3) is to isolate the emergency filter train from the normal TSC HVAC during normal operation and is not required for the emergency mode of operation.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-001, Rev. 1 and 2  
Implementation Document No.: Test Procedure N2-STP-9,  
Mod. PN2Y89MX134  
USAR Affected Pages: N/A  
System: Main Turbine - Generator  
Title of Change: Main Turbine - Generator Rotor  
System Torsional Screening  
Test - NMP2 (N2-STP-9)

Description of Change:

The Main Turbine-Generator Rotor System Torsional Screening Test Procedure N2-88-18 has been rewritten as a special test procedure. Special Test Procedure N2-STP-9 supersedes procedure N2-88-18, which was previously evaluated under Safety Evaluation 90-080, Revision 1.

The turbine-generator torsional screening test conducted in September 1989 measured the torsional natural frequencies of the rotor train. The precise location of the mode identified closest to 120 Hz was at 119.9 Hz. Therefore, as recommended by GE, actions were taken for detuning the unit by installing an inertia ring on the turbine-generator "D" coupling which would lower the 119.9 Hz mode by 0.8 Hz. Telemetry torque collars on the turbine rotor shaft at bearing #4, #6 and #8 for strain gauge signals were also made permanent.

Following the installation of the inertia ring, the torsional test was repeated to measure the torsional natural frequencies of the rotor train and verify if natural frequency had moved further away from a mode of 119.9 Hz to 119.1 Hz or better.

Safety Evaluation Summary:

General Electric required conducting the torsional screening test after the installation of the detuning ring to assure the functional performance, as required for continued safe operation, and to provide maximum protection to the equipment and operating personnel. The performance of the torsional screening test did not affect the safe operation or shutdown of the plant. Turbine missiles were not a concern since rotational speed was within design limits. The controlled acceleration maintained power spikes within the response capabilities of the steam bypass

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Safety Evaluation No.: 91-001, Rev. 1 and 2

Safety Evaluation Summary: (Cont'd)

valves. Differential protective relays and generator ground relays will be in service to protect the unit in an unlikely event of an electrical fault. The turbine electronic backup overspeed trip was operable to provide protection from overspeeding the unit.

Based on the evaluation performed, it is concluded that this change and test do not involve an unreviewed safety question.

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Safety Evaluation No.: 91-003, Rev. 3, 4, and 5  
Implementation Document No.: Calculation ES-258-1  
USAR Affected Pages: 6.2-57, 6.2-57c, 6.2-57d;  
Table 6.2-54; Figure 6.2-77  
System: Secondary Containment  
Title of Change: Secondary Containment Drawdown  
Analysis

**Description of Change:**

Safety Evaluation 91-003, Revision 0, addressed the delta-T (difference between secondary containment and service water temperature), from the secondary containment drawdown standpoint, for the cooler months through June 1, 1991, when no secondary containment heating was anticipated to meet these requirements. The secondary containment temperature was not anticipated to exceed 85°F.

Revisions 1 and 2 of the safety evaluation addressed the delta-T requirements for secondary containment temperature between 80°F and 100°F (summer conditions), took into consideration reduced spent fuel pool heat loads (due to the elapsed time since the first refueling) and unit cooler performance, as exhibited by tests conducted since April 1991, and clarified temperature requirements for certain secondary containment equipment rooms.

Revision 3 of the safety evaluation addressed the delta-T requirements for a revised range of secondary containment temperatures consistent with those expected during the winter months, and took into consideration a more conservative assumption regarding unit cooler performance degradation. These changes were as follows:

1. The secondary containment temperature range varies between 70°F and 85°F.
2. Performance degradation for unit coolers, other than 2HVR\*UC413A & B, was assumed to be 30%, versus the 25% degradation value assumed in Revision 1 of the safety evaluation, in order to account for any additional degradation that may have occurred since the last tests.

Safety Evaluation No.: 91-003, Rev. 3, 4, and 5

Description of Change: (Cont'd)

Revision 4 of the safety evaluation added historical explanations concerning various USAR figures that were deleted in Revision 0 of this safety evaluation and corrected an error in USAR Table 6.2-54, but did not, in any way, alter the analysis or conclusions presented in Revision 3 of the safety evaluation.

Revision 5 of the safety evaluation specified additional delta-T requirements if specific unit coolers or a combination of those unit coolers are out of service.

#### Safety Evaluation Summary:

The methodology, design basis accident, realistic accident conditions, most conservative heat load scenarios and various assumptions and preconditions for defining the delta-T requirements for Unit 2 operation have already been established and approved under previous safety evaluations.

The new analysis takes into consideration revised assumptions concerning initial secondary containment temperature range and unit cooler performance without violating any other criteria already established and approved.

Since no other changes (besides the ones already discussed) are made to the analytical bases, the results of LOCA analyses in USAR Sections 6.2.1, 6.2.3 and 15.6.5, as revised by Safety Evaluation 87-110, remain unaffected.

The analysis meets the applicable criteria of GDC 19, 10CFR100 and SRP 6.4. The analysis also meets the guidelines of SRP 6.2.3 and is conservative with respect to the analysis in USAR Section 6.2.3.3.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-003, Rev. 6  
Implementation Document No.: Calc. ES-263-1  
USAR Affected Pages: 6.2-57, 6.2-57a, 6.2-57c,  
6.2-57d; Table 6.2-54; Figure  
6.2-77  
System: Secondary Containment  
Title of Change: Secondary Containment Drawdown  
Analysis

**Description of Change:**

Revision 6 of Safety Evaluation 91-003 defined the required differential temperature between secondary containment air temperature and service water temperature ( $\Delta T$ ) for the entire third operating cycle. The following design criteria were utilized to determine the differential temperature requirements:

1. Secondary containment inleakage at -0.25 in. W.G. and 40°F outside temperature of 2000 cfm, based on latest surveillance test results.
2. Increased spent fuel pool heat load to account for spent fuel removed during the second refueling. The heat load was calculated at 75 days after shutdown.
3. Secondary containment temperature ranging from 70°F to 105°F.
4. Performance degradation for unit coolers, other than 2HVR\*UC413A & B, of 30% (conservative as demonstrated by tests on 12 unit coolers).
5. Service water temperature of 45°F for outside temperatures of -20°F and 0°F, and service water temperature of 82°F for all other outside temperatures.

**Safety Evaluation Summary:**

The methodology, design basis accident, realistic accident conditions, most conservative heat load scenarios and various assumptions and preconditions for defining the  $\Delta T$  requirements for Unit 2 operation have already been established

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Safety Evaluation Summary: (Cont'd)

and approved under previous safety evaluations. The new delta-T requirements are consistent with the design bases established in these previous safety evaluations and are valid from May 15, 1992, to the end of the third fuel cycle.

Since no other changes (besides the ones already discussed) are made to the analytical bases, the results of LOCA analyses in USAR Sections 6.2.1, 6.2.3 and 15.6.5, as revised by Safety Evaluation 87-110, remain unaffected.

The analysis meets the applicable criteria of GDC 19, 10CFR100 and SRP 6.4. The analysis also meets the guidelines of SRP 6.2.3 and is conservative with respect to the analysis in USAR Section 6.2.3.3..

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-006, Rev. 1  
Implementation Document No.: Mod. PN2Y87MX111  
USAR Affected Pages: Figure 1.2-15 Sht. 2  
System: Process Computer System (PMS)  
Title of Change: Process Computer Terminet  
Remote Interface Module (TRIM)

**Description of Change:**

The Process Computer Terminet Remote Interface Module (TRIM) consists of a personal computer connected to the plant process computers. TRIM provides remote access to the main plant process computer to allow for operation and maintenance of the PMS software by personnel offsite. The TRIM equipment was originally installed under the temporary modification program (Temporary Modification 87-1944), and was used to eliminate an inefficient data link in sending GE BUCLE data using the then-existing method via a plant I/O typer. TRIM is no longer used to transfer BUCLE data. Instead it will be used as a communications link to personnel outside the plant computer room and to allow for local PC file dumping from the process computer. Temporary Modification 90-013 enhanced TRIM and provided a communication link between the tellabs network and the on-line and off-line Honeywell 4500 processor to enable these processors to be available from a remote location.

This modification made Temporary Modification 90-013 into a permanent change and added two modems, two modem sharing devices and an A/B crossover switch.

**Safety Evaluation Summary:**

This modification provides easier access to the plant process computer from a remote location. Installation and operation of the described communications link does not affect plant safety or operation.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-024, Rev. 1  
Implementation Document No.: Simple Design Change  
SD2-0078-91  
USAR Affected Pages: N/A  
System: Reactor Water Cleanup (WCS)  
Title of Change: Substitute Model for  
2WCS-TIS1008

**Description of Change:**

Temperature indicating switch 2WCS-TIS1008 provides an isolation signal to close the outboard containment isolation valve (2WCS\*MOV112) if the water temperature entering the reactor water cleanup system filter/demineralizers reaches 140°F. This instrument was replaced with a qualified, updated model capable of performing the function of the original instrument. The instrument was also relocated on instrument rack 2CES\*RAK005.

To allow the instrument changeout, a jumper was temporarily installed in panel 2CEC\*PNL623 to defeat the isolation signal from 2WCS-TIS1008 during the time period the instrument changeout was performed.

**Safety Evaluation Summary:**

The substitution to a newer model of the instrument and the relocation of the instrument will have no impact on the system. The component and its function is not required for the safe operation or shutdown of the plant. The function of the instrument is to automatically isolate the system and protect the filter demineralizer resin beds in the event of high effluent temperatures. The temporary installation of a jumper to prevent this automatic system isolation allowed the system to be in operation during the period of time required to change the instrument. The system could still be monitored and remotely isolated from the control room in the event of high effluent temperatures.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-031

Implementation Document No.: Appendix B Determination  
91-010

USAR Affected Pages: 9.1-7, 9.1-12, 9.1-15a; Table  
3.9A-12 Sht. 14

System: Spent Fuel Pool Cooling

Title of Change: Change in Safety-Related  
Function of Check Valves  
2SFC\*V9 and V11

**Description of Change:**

Check valves 2SFC\*V9 and V11 are located in the spent fuel pool cooling return piping to the spargers located at the bottom of the spent fuel pool. The active safety-related function of these valves is indicated in the USAR as "prevent reverse flow." This change revised their safety-related function to "allowing flow in the forward direction."

**Safety Evaluation Summary:**

The passive anti-siphon breakers (1" open-ended pipe nipples) installed in the system will prevent the siphoning of the spent fuel pool in the unlikely event of a pipe break in the spent fuel pool cooling system piping. Also, when makeup water is supplied to the spent fuel pool by the service water system, pump discharge check valves 2SFC\*V20A and B will prevent the service water flow from bypassing the spent fuel pool. Therefore, failure of check valves 2SFC\*V9 and V11 to prevent reverse flow will not affect the safety-related function of the spent fuel pool cooling system or affect adequate service water system flow to the spent fuel pool.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-032, Rev. 1 and 2  
Implementation Document No.: LDCN U-1537, Rev. 2  
USAR Affected Pages: 7.2-6, 8.3-13, 8.3-14, 8.3-20,  
8.3-21, 9.2-43, 9.3-9, 9.5-14,  
9.5-55; Table 5.2-7 Sht. 1  
System: Various  
Title of Change: USAR Text Correction

Description of Change:

Various inconsistencies were identified in the text and a table of the USAR, as listed below. The inconsistencies were either editorial in nature or represented inaccurate information in minor details of system descriptions. These changes correct inconsistencies. No hardware or logic changes were involved.

1. Table 5.2-7 erroneously showed conductivity recorders for the individual condensate demineralizer outlets, and for main condenser hotwell and tube sheet samples. Recorders are not provided for these samples.
2. On page 7.2-6, turbine stop valve closure input to the RPS was described inaccurately. There are only four position switches installed - one for each valve, not eight as stated.
3. On page 8.3-14, it was incorrectly implied that a DBA condition automatically starts the diesel generators. Only a LOCA signal or a loss of offsite power will result in the automatic starting of the diesel generators.
4. Pages 8.3-20 and 8.3-21 - The standby diesel generator jacket water pressure low condition was incorrectly shown as shutting down the diesel generator under the test run condition, whereas it is actually only a locally annunciated emergency condition.
5. On page 9.2-43, auxiliary boiler deaerator level control was erroneously included in the condensate storage system. This is a part of the auxiliary boiler feedwater system.

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Description of Change: (Cont'd)

6. On page 9.3-9, the breathing air compressor control time delay was erroneously shown as 15 seconds. The correct setpoint is 10 seconds, consistent with that shown on Figure 9.3-4.
7. On page 9.5-14, power supply to the communication system was described inaccurately. There is no transfer switch in relay and control cabinets power supply. These cabinets are fed from 2VBB-UPS1D.
8. On page 9.5-55, the diesel generator lube oil system crankcase level alarm was erroneously called sump level alarm.

**Safety Evaluation Summary:**

The described USAR changes do not affect the performance of any system or equipment and have no impact on the safe operation or shutdown of the plant. They correct inconsistencies or inaccurate information and do not involve any hardware or logic changes.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-033  
Implementation Document No.: Mod. PN2Y90MX043  
USAR Affected Pages: 6.2-77a; Figure 6.5-1 Sht. 7  
System: Standby Gas Treatment  
Title of Change: Standby Gas Rad. Monitor  
Grounding

**Description of Change:**

This modification involved the addition of a time delay relay to the circuit for radiation monitor 2GTS-RU105 and the addition of a ground strap to the GEMS sample line. The time delay relay provides a 15-second delay of signal initiation for closure of the primary containment purge system isolation valves. The ground strap provides additional grounding protection at the lower end of the GEMS sample line. These changes were initiated as a result of three lightning strikes on the main stack. These lightning strikes caused power transients to radiation monitor 2GTS-RU105 which caused spurious actuation of the primary containment purge system isolation valves.

**Safety Evaluation Summary:**

The function of 2GTS-RU105 is to monitor exhaust from the standby gas treatment system as it enters the main stack. When high radiation levels are detected, the group 9 valves (containment purge system) isolate the primary containment. These valves are used primarily during outage situations. The addition of the 15-second time delay is intended to prevent automatic closure of the containment purge system isolation valves due to spurious power transients, thus eliminating an unnecessary challenge to an engineered safety feature. The radiological consequences of this change were evaluated for ALARA, control room, and offsite dose, and were found to be unaffected. Also, no credit is taken for this monitor in the fuel handling accident.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-035  
Implementation Document No.: Mod. PN2Y87MX200  
USAR Affected Pages: 9.3-23, 12.2-11  
System: Turbine Building Drains  
Title of Change: Venting of Sump 2DFT-SUMP2H

**Description of Change:**

This modification installed piping from existing low-point drain valves 2MSS-V229, V230 on line 2MSS-006-119-4 to floor sump 2DFT-SUMP2H on el. 239'-0" in the pipe tunnel. The purpose of this 1" drain line, 2MSS-001-412-4, is to drain line 2MSS-006-119-4 of condensate prior to allowing this line to be utilized for plant operations.

**Safety Evaluation Summary:**

Drain valves 2MSS-V229, V230 are used to drain condensate from line 2MSS-006-119-4 through a 1" drain line to SUMP 2H during startup and shutdown, off-normal procedures, and emergency operating procedures. At all other times, drain valves 2MSS-V229, V230 will be closed to isolate the 1" drain line to SUMP 2H.

The temperature of the condensate in line 2MSS-006-119-4 will be less than 200°F so that steam flashing will not occur through the 1" drain line and cause airborne radioactivity in the vicinity of SUMP 2H. Also, noble gases, which could become airborne even in the absence of flashing, are expected to be negligible. Based on this reasoning, there is no ALARA justification to ventilate SUMP 2H.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-037  
Implementation Document No.: EDC 2E10312  
USAR Affected Pages: 6.5-7; Figures 6.5-1 Sht. 3,  
9.4-8L  
System: Standby Gas Treatment (GTS)  
Title of Change: Standby Gas Treatment System  
Decay Heat Air Flow

**Description of Change:**

The standby gas treatment system (SGTS) as-built configuration allows the decay heat produced by the radioactive particles in the inactive charcoal filter train to be removed by passing equipment room air through the inactive filter train. The air is then exhausted to the main stack by the fan of the active filter train.

In order to prevent decay heat cooling air flow from bypassing the charcoal filters, the SGTS filter train air inlet pressure control valves 2GTS\*PV5A, 5B are interlocked with SGTS filter train fans 2GTS\*FN1A, 1B such that when the fan is not running the respective valve is isolated (closed).

This change revises design documentation and the associated USAR Figures 6.5-1 and 9.4-8L to agree with the as-built plant configuration as described above.

**Safety Evaluation Summary:**

Interlocking 2GTS\*PV5A, 5B with 2GTS\*FN1A, 1B ensures that decay heat cooling air does not bypass the filter train prior to exhausting through the main stack, thereby satisfying design basis requirements for the SGTS, as outlined in USAR Section 6.5.1. This change will have no impact on the safe operation or shutdown of the plant.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-039, Rev. 0 and 1  
Implementation Document No.: Temp. Mods. 91-020 and 91-024  
USAR Affected Pages: N/A  
System: Circulating Water; Copper-Trol  
CU-1 (WTC), Betz-Powerline  
3450 Dispersant (WTD) Water  
Treatment  
Title of Change: Copper-Trol and Betz-Powerline  
3450 Dispersant Storage Tank  
Installations

**Description of Change:**

The injection of Betz Copper-Trol CU-1 and Betz-Powerline 3450 Dispersant into the circulating water system was previously addressed in Safety Evaluation 89-069 as part of Temporary Modification 89-092. Temporary Modifications 91-020 and 91-024 provide for the continued use of these two chemicals, until the permanent chemical injection systems (WTC, WTD) are installed.

**Safety Evaluation Summary:**

The addition of a scale inhibitor (Betz-Powerline 3450 Dispersant) to the Unit 2 circulating water will have no negative impact on system and equipment operation or reliability and will help to extend service life by controlling the corrosive effects of scaling. The addition of a corrosion inhibitor (Copper-Trol CU-1) will reduce copper leaching from the condenser tubes and reduce the level of soluble copper in the circulating water. In the event of a condenser tube leak, the addition of these compounds will not have a negative impact on the chemistry of condensate water, feedwater, or reactor water. An increase in the level of chlorides in the Unit 2 circulating water is within the ability of the demineralizers to maintain the quality of feedwater/reactor water. Revision 1 to the safety evaluation addressed a revised informational sketch to reflect the as-built condition of the WTC system.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-040  
Implementation Document No.: COLR, Cycle 2, Revision 01  
USAR Affected Pages: N/A  
System: Fuel  
Title of Change: Core Operating Limits Report,  
Cycle 2 - Revision 01

**Description of Change:**

This safety evaluation addresses a revision to the rod block monitor setpoints in the Nine Mile Point Unit 2, Cycle 2, Core Operating Limits Report. The revision changed the trip setpoint and allowable value from "less than" the specified values to "less than or equal to" the specified values.

**Safety Evaluation Summary:**

The rod block monitor setpoints were originally expressed in the Technical Specifications as "less than or equal to." When the Core Operating Limits Report was first instituted they were inadvertently expressed as "less than." This change revised the Core Operating Limits Report to be consistent with the Technical Specification from which it was derived. No physical plant changes were made.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-044  
Implementation Document No.: LDCN U-1540  
USAR Affected Pages: Appendix 9C, Tables 3-5,  
4-1 Sht. 9  
System: Service Water  
Title of Change: Add Screenwell Building El.  
280'-0" Concrete Floor Plugs  
to USAR Tables 3-5 and 4-1 of  
Appendix 9C

**Description of Change:**

This change incorporated the concrete floor plugs on el. 280'-0" of the screenwell building into the heavy loads analysis (USAR Appendix 9C). These plugs are lifted using crane 2MHW-CRN1.

**Safety Evaluation Summary:**

The floor plugs are a required lift using the screenwell crane to access the SWP equipment below el. 280'-0". A review of the screenwell building safety-related equipment which is located below these concrete floor plugs indicated that in the event of a load drop of a single floor plug, only one of the two redundant trains of the service water system could be affected. There were no changes to the facility or to heavy load procedures.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-045  
Implementation Document No.: Mod. PN2Y86MX085  
USAR Affected Pages: Figures 9.2-1a, 9.2-1b,  
9.2-2 Sht. 5  
System: Service Water (SWP)  
Title of Change: Control Room Nuisance Alarm  
Windows 601101, 601201,  
852103, 852203

**Description of Change:**

This change replaced the SWP strainer backwash MOVs control switch to permit manual operation of the associated valve such that the associated valve can be maintained in the open position when the switch is placed in the OPEN position. Certain control room nuisance alarms were also eliminated. The existing control switch provided only for the automatic operation of these valves activated by high strainer differential pressure, and were spring returned to the auto position. To alleviate a problem with frequent cycling of the backwash MOVs, Operations had to hold the existing switch in the OPEN position and de-energize the control circuit to cause the valves to fail open. This achieved a continuous backwash mode of operation, but also resulted in activation of inoperability alarms.

**Safety Evaluation Summary:**

The new control switch design does not inhibit or compromise the service water system's ability to perform its safety function. There is no impact on the system design basis, including the hydraulic transient analysis. The strainers are rated for continuous backwash.

The new design does not compromise the design for the existing bypass/inoperable status indication and annunciation for these circuits, and does not affect the SWP system's ability to be tested during operation as well as during intervals when the system is shut down.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-047, Rev. 1  
Implementation Document No.: EDC 2E10402A  
USAR Affected Pages: Figure 7.7-2 Shts. 13, 24  
System: Reactor Manual Control  
Title of Change: Documentation Only Change of  
the Reactor Manual Control  
System to Reflect Proper  
Design Configuration

**Description of Change:**

This change reconciled the field as-built condition of the reactor manual control system to the system elementary diagram, which is the source document for USAR Figure 7.7-2.

Originally, each time a rod was withdrawn to a bank or notch limit, the rod sequence control system removed the rod withdraw permissive. Loss of the withdraw permissive generated inputs into the rod drive control, where the activity card generated both a rod block and an annunciation. A change was made (in 1985) to maintain pattern control while eliminating unnecessary rod block annunciation during normal operations. However, the logic and wiring changes were not shown on the system elementary diagram or the corresponding USAR figure.

**Safety Evaluation Summary:**

This change revises USAR Figure 7.7-2 and the corresponding elementary diagram for the reactor manual control system. This is a documentation-only change and does not affect the design or function of the system or components. The reactor manual control system is nonsafety related.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-048  
Implementation Document No.: Calculation 12177-ES-196-0 &  
LDCN U-1498  
USAR Affected Pages: 15.6-12; Table 15.6-13 Sht. 10  
System: Primary Containment  
Title of Change: TIP Leakage Rate Clarification

**Description of Change:**

A discrepancy was identified between Calculation No. 12177-ES-196-0 and the USAR concerning the volume that the traversing incore probe (TIP) leakage rate is based on. The calculation bases the leakage rate on drywell volume, whereas USAR Table 15.6-13 states the leakage rate in terms of primary containment volume. Primary containment volume can be construed to mean the drywell volume plus the free air volume of the wetwell. The TIP leakage rate should be based on drywell volume only, as correctly shown in the calculation. USAR Table 15.6-13 has therefore been revised accordingly.

**Safety Evaluation Summary:**

The TIP guide tubes run from underneath the reactor vessel to the TIP drive mechanism outside the primary containment on el. 250'-0". The TIP guide tubes are located totally within the drywell and do not interface with the wetwell. Therefore, in order to determine the TIP leakage rate, it is necessary to base it on drywell volume only as shown in Calculation No. 12177-ES-196-0. This is a clarification of the volume used for TIP leakage rate and does not affect other design documents.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-050  
Implementation Document No.: Mod. PN2Y86MX085  
USAR Affected Pages: 9.4-27; Figures 9.4-8j,  
9.4-9 Sht. 21  
System: Drywell Cooling (DRS)  
Title of Change: Control Room Nuisance Alarm  
Windows 873201, 873202

**Description of Change:**

The drywell unit coolers originally had flow switches that monitored unit cooler air flow and annunciated a system trouble alarm when low flow conditions were sensed. To eliminate nuisance alarms, the discharge flow switches and their associated pitot tubes were spared in place by this modification.

**Safety Evaluation Summary:**

The drywell cooling system operation and function is not affected by this change. Other indication (i.e., start/stop indicating lights, fan motor overload annunciation and temperature annunciation) is provided to the operators, confirming that the unit cooler's fans are running. Lack of adequate containment atmospheric cooling due to unit cooler flow blockage would be detected by the thermocouples located in the primary containment and alarm in the control room. No automatic function occurs directly as a result of drywell unit cooler flow blockage.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-053, Rev. 1  
Implementation Document No.: Temp. Mods. 91-034, 91-062,  
91-063, 91-064  
USAR Affected Pages: N/A  
System: Reactor Water Cleanup (WCS)  
Title of Change: Defeat F/D Tank Level Elements  
Description of Change:

The level elements (2WCS-LE62A/B/C/D) on the reactor water cleanup system filter demineralizer vessels are unreliable and have provided erroneous signals indicating the vessel(s) were full. This has caused the computer-controlled precoat logic to place the individual filter/demineralizer in service at a point when the vessel was not completely filled, which resulted in a high differential flow signal, an ESF actuation, and system isolation (reference LERs 91-08 and 91-13). This temporary modification physically defeated the automatic function of the level elements by lifting of four wires (one wire for each element) at the instrument racks. The function of the level elements was performed manually, as directed by temporary revisions to the system operating procedure. This temporary modification will remain in place until new, more reliable level elements are installed.

**Safety Evaluation Summary:**

This temporary change prevents the filter/demineralizers from being automatically returned to service before the vessel is properly filled, insures the proper manual operation of the filter/demineralizers (prevents the unintentional backwash or contamination of the reactor water with precoat resins), and prevents possible system isolations or ESF actuations due to faulty level elements. This change does not require the operator to manipulate any valves, and therefore will not cause the inadvertent opening of any high/low pressure interface isolation valves. The change will also not affect any of the automatic system isolation functions.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-056, Rev. 1 and 2  
Implementation Document No.: LDCN U-1672 Rev. 1  
USAR Affected Pages: 9A.3-2  
System: N/A  
Title of Change: Reduction in Fire Brigade  
Manning Through Partial  
Combination of the Unit Fire  
Brigades

**Description of Change:**

Niagara Mohawk will minimize fire brigade manpower costs while maintaining the required five-person response (and required unit-specific training) to fire alarms by the partial combination of the two unit fire brigades. The change involves the elimination of one of the shift positions from each brigade, leaving the manning at each unit at one brigade leader (fire chief) and three brigade members with unit-specific training. The five-man brigade response will be maintained by having one of the brigade members from the unaffected unit respond to a fire alarm at the affected unit.

The safety evaluation also addresses the situation where an additional brigade member may call in sick. The unit fire brigade will not be required to call in an off-duty member (or retain a member going off-duty), instead taking credit for two members of the unaffected unit responding to a fire at the affected unit. The use of two brigade members from the unaffected unit is intended only for emergency (unscheduled) absences and not for scheduled absences.

**Safety Evaluation Summary:**

The reduction in staffing levels will not result in a loss of the fire watch function or in a lesser response to a fire at either unit (either in number of personnel fighting the fire or in a significant increase in their response time). A five-member fire brigade will continue to respond to fires at either unit and initiation of fire fighting will occur with equal efficiency.

Safety Evaluation No.: 91-056, Rev. 1 and 2

Safety Evaluation Summary: (Cont'd)

Even with the reduced staffing levels of the fire brigades, the manual fire suppression capabilities at Nine Mile Point are further enhanced by:

1. Use of dedicated fire brigade personnel for whom brigade response duties are not just another assignment in addition to their normal duties; and
2. Brigade members who perform the routine surveillance of the fire protection equipment, reinforcing their training on the availability, operation, location, actuation mechanisms, coverage, etc., of this equipment.

Based on the evaluation performed, it is concluded that this change will not affect the effectiveness of fire fighting efforts and does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-057  
Implementation Document No.: Temp. Mod. 91-021  
USAR Affected Pages: N/A  
System: Circulating Water (CWS), Water Treatment Hypochlorite (WTH)  
Title of Change: Hypochlorite Storage Tank 2WTH-TK12 Level Indicator

**Description of Change:**

The permanent plant hypochlorite system lacked proper storage tank (2WTH-TK12) level indication. Since maintaining proper chlorine levels within the tank is essential to efficient system performance, a temporary modification was implemented to provide an accurate level indicator. The level indicator utilized was a transparent tube that extended from the bottom of the tank, at the existing drain valve connection (2WTH-V143), up vertically along the tank face to the top. The tube at the top was open to atmosphere and the drain valve remained open to assure correct displacement of chlorine within the tube. This temporary modification will remain in place until a permanent means of detecting tank level can be implemented.

**Safety Evaluation Summary:**

At Nine Mile Point Unit 2, biofouling of the circulating water system is controlled by the addition of sodium hypochlorite to the water. This reduces the reproduction of algae, fungi and bacteria. With implementation of the temporary modification, tank level indication and required filling can be accurately gauged and maintained, thus facilitating system operation.

The water treatment hypochlorite system is nonsafety related and is not required for safe operation or shutdown of the plant. This temporary method of level indication for the system storage tank provides an alternate means of performing a nonsafety-related function that will have no negative impact on the safe operation or shutdown of the plant.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-058  
Implementation Document No.: Operating Procedure N2-OP-14  
USAR Affected Pages: N/A  
System: Makeup Water Storage (MWS),  
Turbine Building Closed Loop  
Cooling (TBCLCW)  
Title of Change: Mechanical Jumper for Makeup  
Water to Tank 2CCS-TK1

Description of Change:

This change affects the surge and makeup tank, 2CCS-TK1, of the turbine building closed loop cooling water system (TBCLCW) and its associated makeup water line, 2MWS-002-33-4. In order to perform maintenance work on the positive displacement flow totalizer, 2MWS-FQIS122, the makeup water supply line to the tank must be closed. Since the system must be operating during this maintenance activity, a temporary makeup water supply arrangement was provided to assure the proper functioning of the TBCLCW. This temporary makeup water supply arrangement was accomplished via a mechanical jumper which bypassed the segment of the normal makeup water line that was under maintenance work. A hose connection from valves 2MWS-V95 to 2MWS-V106 was made to accomplish the temporary makeup water supply.

Safety Evaluation Summary:

The TBCLCW and makeup water storage systems are both nonsafety related and are not required to effect or support the safe shutdown of the plant. The temporary makeup water will be monitored to properly establish the required water level in the tank. In addition, no change to the level control instrumentation of the tank will be made after the initial fill. In the event of a failure of the temporary makeup water line, the water may flood the immediate area. The amount of water entering the immediate area, from the time of temporary component failure until the water flow is stopped by manually closing the valve, will not have any impact on any equipment important to safe shutdown of the plant and maintaining the plant in a safe shutdown condition.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-059  
Implementation Document No.: DER 2-91-Q-660  
USAR Affected Pages: N/A  
System: N/A  
Title of Change: Equipment Qualification  
Documentation

**Description of Change:**

This safety evaluation was written to document and evaluate a variance in the method that equipment qualification (EQ) documentation is maintained. Update of the equipment qualification environmental design criteria document (EQEDC) is being deferred pending development of an interrelational environmental design criteria database. The EQEDC consists of a summary of results of the design basis calculations which determine zonal environmental conditions for normal, abnormal and accident conditions. Until the interrelational EQ database is functional, the EQEDC is not being used for EQ parameter determination. Assurance of compliance with 10CFR50.49 qualification requirements is maintained in this interim period by review of the individual plant modifications and EQ program oversight, using the EQ basis design calculations and equipment qualification test reports to determine specific applicable EQ data (rather than the EQEDC which summarizes this basis data).

**Safety Evaluation Summary:**

The EQEDC is a document that was developed by the Unit 2 architect engineer (AE) to summarize the results of design calculations for plant ambient and accident environmental conditions. This was an arbitrary format chosen by the AE. The EQ rule does not specify an EQEDC, nor an EQEDC format, nor does it require that the included data be assembled in summary format. This change does not result in any modification to any system, and adequate processes are in place, via EQ group program oversight, to ensure that compliance with 10CFR50.49 EQ rule requirements is maintained.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-060  
Implementation Document No.: Simple Design Change  
SC2-0245-91  
USAR Affected Pages: Figure 9.4-9 Sht. 6  
System: Reactor Building Ventilation  
(HVR)  
Title of Change: Revision to 2HVR\*AOD6A/B  
Control Logic

**Description of Change:**

Per USAR Section 9.4.2.5.3, the suction dampers (2HVR\*AOD6A/B) for unit coolers 2HVR\*UC413A/B open or close whenever the associated unit cooler starts or stops, respectively. Contrary to this and per the original design, the suction dampers did not remain closed (energized) if the breakers for the unit coolers were "racked out", as shown in a recent situation where the 2HVR\*AOD6A suction damper failed open while 2HVR\*UC413A was de-energized (reference LER 91-05). This change keeps the 2HVR\*AOD6A/B damper energized when the breaker (2EJS\*US1(3)-4C) for unit cooler 2HVR\*UC413A/B is racked out. New/existing wires from the 2HVR\*AOD6A/B control switches (1A-2HVRA88 and 1A-2HVRB88) were installed/reworked to an available set of contacts in the unit cooler control switches (1-2HVRA03 and 1-2HVRB03) which close only when the unit cooler control switches are in the "pull to lock" position.

**Safety Evaluation Summary:**

The safe operation and safe shutdown of the reactor is not affected by this change. The change to the circuit will not impact the damper's requirement to fail open in the event of a loss of power, nor will it impact the damper's position indication lights in the control room. The dampers will continue to operate as intended to support the operation of the unit coolers, and will not affect the unit coolers' ability to respond to a low air flow signal from above or below the refueling floor, a high-high radiation signal from the refueling floor, or a LOCA signal.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-061  
Implementation Document No.: Temp. Mod. 91-059  
USAR Affected Pages: N/A  
System: Main Transformers Including Auxiliaries (MTX)  
Title of Change: Removing 2MTX-XM1B from Service and Placing 2MTX-XM1D into Service

**Description of Change:**

**Safety Evaluation Summary:**

On August 13, 1991, one of the main transformers (2MTX-XM1B) at Nine Mile Point Unit 2 failed. This is one of three 25-kV to 202-kV transformers (one transformer per phase) which steps up the Unit 2 main generator's voltage for interconnection with the NMPC 345-kV grid through the Scriba Substation.

This temporary change removed the damaged 2MTX-XM1B transformer from service and placed the identical and available spare transformer (2MTX-XM1D) into service. This involved the disconnection of the 25-kV and the 345-kV buses from the "B" transformer and the connection of these buses to the "D" transformer.

**Safety Evaluation Summary:**

The four main generator transformers are shell-type, oil-immersed, single-phase 25-kV to 202-kV step-up transformers. Three of the four units (2MTX-XM1A/B/C) are connected to form a three-phase 24.3-kV delta on the low voltage side and a three-phase 350-kV grounded wye on the high voltage side. The fourth unit (2MTX-XM1D) is a spare. This transformer is of the same make and model as the three normally used transformers and was specifically designed to be capable of performing the function of any one of the three normally used units. The 25-kV and 345-kV buses were also designed to be capable of easy disconnections and reconnections to accommodate any of the desired transformers. The main transformers do not provide onsite Class 1E power nor are they required for the assurance of

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offsite power onto the site. Therefore, they do not affect equipment important to safety.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-063, Rev. 0 and 1  
Implementation Document No.: Temp. Mod. 91-069  
USAR Affected Pages: N/A  
System: Reactor Water Cleanup (WCS)  
Title of Change: Leak Repair, 3/4" Double  
Isolation Drain Valve  
Connection 2WCS\*V325 and  
2WCS\*V326

**Description of Change:**

This temporary modification consisted of the repair of a leaking 3/4" drain connection off of line 2-WCS-008-250-1, the reactor water cleanup system processed water return line to the RPV (via the feedwater system). The drain connection includes double isolation valves (2WCS\*V325 and 2WCS\*V326) and a threaded cap. The leak existed between the 3/4" closure nipple and the threaded cap, and resulted from both valves failing to isolate properly. The temporary repair consisted of removing valve 2WCS\*V326 and welding a cap onto the remaining pipe nipple. An enclosure device injected with Furmanite was required to seal the leak. A freeze seal was then employed to isolate the drain line to complete the repair.

**Safety Evaluation Summary:**

This temporary modification eliminates a leak on the WCS system, thereby eliminating contamination and restoring system integrity. All work will be performed in accordance with approved site procedures and to the original design requirements. Thus, the reactor water cleanup system will continue to function as designed. A rupture of the repaired 3/4" line would be within the normal makeup capability of the reactor coolant system.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-064  
Implementation Document No.: EDC 2F00359  
USAR Affected Pages: Figure 10.1-6b  
System: Feedwater  
Title of Change: Revise 2FWS-V65A, 65B, 66A,  
and 66B to Normally Open  
Designation

**Description of Change:**

During normal plant operation, all of the feedwater flows through the two 24-inch feedwater lines to the reactor. The two feedwater lines are provided with flow elements FE1A, 1B. For measuring flow and pressure drop across the elements, flow transmitters FT1A, 1B and differential pressure indicators PDI1A, 1B are provided, each with a root isolation valve.

The system P&ID and associated USAR Figure 10.1-6b had shown these root isolation valves as normally closed. This change corrects the drawings to show these valves as normally open, which is required for normal functioning of the instruments and is consistent with system operating procedures.

**Safety Evaluation Summary:**

This is a documentation-only change to show the correct instrument line root isolation valve positions on the feedwater system P&ID. No physical change to the plant is involved. The feedwater system is not required to effect or support safe shutdown of the reactor.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-065, Rev. 2  
Implementation Document No.: Special Test Procedure  
N2-STP-27  
USAR Affected Pages: N/A  
System: Condensate  
Title of Change: Test N2-STP-27

**Description of Change:**

On August 31, 1991, feedwater isolation valves 2CNM-MOV84A and B failed to open against a differential pressure of approximately 560 psig. After evaluations, field observations and static field testing, Special Test Procedure N2-STP-27 was used to demonstrate whether valve 2CNM-MOV84B would open against the condensate booster pump discharge head without equalizing the pressure across the disc.

One condensate pump and one condensate booster pump were operated during the test. Two temporary pressure gauges were installed for accurate indication of condensate booster header pressure and feedwater pump suction pressure during this test. Temporary diagnostic equipment was installed on 2CNM-MOV84B and removed at the end of this test.

**Safety Evaluation Summary:**

The test will be performed per approved plant procedures with the plant shut down. During the test, reactor water level will be maintained by the control rod drive system and reactor water cleanup or residual heat removal reject. The reactor vessel will be isolated from the condensate system to prevent overflowing the reactor vessel.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-066, Rev. 1

Implementation Document No.: Site Emergency Plan (SEP),  
Revision 23; S-EAP-2, Revision  
14; S-EPP-25, Revision 5

USAR Affected Pages: N/A

System: Emergency Preparedness

Title of Change: Site Emergency Plan (SEP),  
Revision 23; S-EAP-2, Revision  
14; S-EPP-25, Revision 5

**Description of Change:**

Nine Mile Point Nuclear Station (NMPNS) Emergency Preparedness Procedures and the Site Emergency Plan (SEP) were revised as a result of procedure comments made during and after the Site Area Emergency at NMPNS on August 13, 1991. During this emergency it was noted that the termination criteria were too inflexible, and SORC/SRAB reviews and/or approvals in procedure S-EPP-25 were not appropriate. These comments led to Revision 5 to S-EPP-25, and subsequent Revision 23 to the SEP. In addition, a review of Site Area Emergency (SAE) Emergency Action Level (EAL) of loss of all control room alarms (annunciators) and a plant transient was found to be inappropriate. Therefore, this SAE EAL was changed in Revision 14 of procedure S-EAP-2.

**Safety Evaluation Summary:**

The changes to the SEP, S-EAP-2, and S-EPP-25 were specifically made to improve the response by Niagara Mohawk's emergency response organization. The changes allow for a more rapid entry into the recovery phase of operations by allowing the Site Emergency Director greater discretionary authority to terminate a Site Area Emergency or General Emergency prior to the reactor being in a cold shutdown. Furthermore, the Site Emergency Director and Corporate Emergency Director are given the discretion of implementing a recovery plan and proposed recovery organization without review by SORC and SRAB, and discretion to terminate a Site Area Emergency or General Emergency without concurrence by SORC. The changes are consistent with 10CFR50, Appendix E, §IV(b) and (c), and NUREG-0654, §D.1 and Appendix 1.

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Safety Evaluation Summary:

(Cont'd)

The Emergency Action Level for entering into a Site Area Emergency, based upon a "loss of indicators, annunciators, or alarms in the Control Room and loss of emergency assessment or communication capability," has been changed to require that not only control room annunciators but also specific indicators of system parameters be inoperable during a plant transient. This change is consistent with 10CFR50.47 and is considered necessary to avoid overclassification of the emergency.

Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question and do not reduce the effectiveness of the emergency plan.

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Safety Evaluation No.: 91-067  
Implementation Document No.: Simple Design Change  
SC2-0123-91  
USAR Affected Pages: Figure 7.7-2 Sht. 14  
System: Reactor Manual Control, Rod  
Worth Minimizer  
Title of Change: Eliminate Rod Worth Minimizer  
Block Annunciation

**Description of Change:**

The reactor manual control system (RMCS) provides the operator the ability to make changes in nuclear reactivity by the manipulation of control rods so that reactor power level and core power distribution can be controlled. Originally, each time a control rod was withdrawn to a bank or notch limit, the rod worth minimizer removed the rod withdraw permissive so that the activity card, located in panel 2CEC-PNL616, would re-establish the rod block. Every time the activity card generates a rod block, it also produces an annunciation at window 603442, even though no rod pattern violations have occurred. This change altered the function of the activity card to eliminate annunciation each time a rod block is applied. Annunciation for equipment failure or other abnormal conditions is unaffected by this change. The rod block function and application are not affected by the described annunciation change.

**Safety Evaluation Summary:**

This change allows the rod worth minimizer to maintain rod pattern control while eliminating unnecessary annunciation of normal rod blocks during startup. The rod worth minimizer operator display will continue to display rod blocks and identify what caused them to occur. Alarms will still be generated and recorded on the 3D-Monocore computer facility to provide a record of rod block activity. This change does not impact the rod worth minimizer or the reactor manual control system's method or ability to generate a rod block. The RMCS does not include any of the circuitry or devices used to automatically or manually scram the reactor.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-070, Rev. 1  
Implementation Document No.: Mod. PN2Y89MX021  
USAR Affected Pages: 11.3-5a; Figure 11.3-1b  
System: Offgas  
Title of Change: Offgas Freeze-Out Dryer High  
D/P Swapover

**Description of Change:**

When a high differential pressure occurs across a freeze-out dryer, there is a normal swapover to one that is in the standby mode. However, a problem in the control logic currently resets the timer to zero and causes a second swapover back to the faulty freeze-out dryer. When this happens, defrost circuits are disabled; there is no indication the event has occurred, and operators cannot be aware of this until it is seen during routine surveillance. The operator must then place the iced up freeze-out dryer in the manual defrost mode and pay close attention to it to ensure complete defrosting takes place to prevent the compressor from overheating.

This modification made Temporary Modification 88-149 permanent, whereby annunciation was provided on local panel 20FG-IPNL122 to indicate that a swap on high differential pressure occurred so an operator could manually change the freeze-out dryer mode. A manual defrost timer was added so that a freeze-out dryer can be placed in the manual defrost mode for a predetermined amount of time. Also, the faulty control circuitry involved was corrected so that the timer will not be reset to zero.

**Safety Evaluation Summary:**

The offgas system freeze-out dryers do not perform any safety function. This change enhances system operation by facilitating more efficient and timely defrosting, reduces the possibility of compressor damage due to overheating, and reduces operator residence time in a contaminated area. The design basis of the offgas system is not altered by this change.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-071  
Implementation Document No.: EDC 2E10461  
USAR Affected Pages: Figure 8.3-3 Shts. 1, 2  
System: N/A  
Title of Change: Plant Master One-Line Diagram,  
USAR Fig. 8.3-3 Correction

**Description of Change:**

This change corrected USAR Figure 8.3-3 to show the proper power supply for essential lighting distribution panels. Originally, the power supply for essential lighting panels 2LAX-PNLU01 and 2LAC-PNLU02 was shown on USAR Figure 8.3-3 as 2VBB-UPS1D and 2VBB-UPS1C, respectively. This was not correct. The as-built configuration of the power supply is 2VBB-UPS1D for panel 2LAC-PNLU02 and 2VBB-UPS1C for panel 2LAX-PNLU01.

**Safety Evaluation Summary:**

This is a documentation-only change to USAR Figure 8.3-3, and does not involve any physical changes to the plant.

The UPSs 2VBB-UPS1C and 2VBB-UPS1D are identical in their design, parameters, power supply, function and qualification. They are both designed to feed nonsafety-related instrumentation, control and lighting loads. These power supplies are fed from the onsite normal power distribution system and use the onsite nonsafety-related dc system as a backup power supply. Thus, the ability of the system to provide essential lighting to various areas is not affected.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-075  
Implementation Document No.: Simple Design Change  
SC2-0312-91  
USAR Affected Pages: Figure 1.2-32 Sht. 3  
System: Switchgear - Normal Power  
13.8 kV (NPS)  
Title of Change: Relocation of 13.8 kV Breaker  
Test Cabinet 2NPS-BTC3

**Description of Change:**

The existing 13.8-kV breaker test cabinet 2NPS-BTC3 was located in the southwest corner of the normal switchgear building el. 261' along the south wall. This area was not a convenient location for the test breaker area since it is not well lit and it is not easily accessible for the large 13.8-kV breakers due to other equipment (welding transformers 2WPS-X13, -X39, and -X49) in the immediate area. This simple design change relocated the test cabinet (2NPS-BTC3), along with a conduit and cables, onto the column in the center of the room.

**Safety Evaluation Summary:**

The new location for the test breaker cabinet is in the normal switchgear building where there are no safety-related components required for the safe operation or the safe shutdown of the reactor. The breaker test cabinet and the breakers it will service are not required for onsite Class 1E power distribution nor do they function in any capacity to assure reliable offsite power to the site.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-081, Rev. 1  
Implementation Document No.: Procedure N2-TTP-HVR-@001  
USAR Affected Pages: N/A  
System: Reactor Building HVAC (HVR)  
Title of Change: Performance Testing of  
Secondary Containment Unit  
Coolers

**Description of Change:**

This modification involved the installation of a temporary test loop to facilitate measurement of water flow rate through the secondary containment unit coolers. The test loop consisted of either flexible fire hose with a turbine flow meter, or a clean steel spool piece with an ultrasonic flow meter. The test loop was installed in place of spool pieces used for flushing of the unit coolers.

**Safety Evaluation Summary:**

The test program does not impact the function of the unit cooler. The test procedure controlling the conduct of the test assures that the unit cooler as well as the associated test equipment will not exceed their rated capacity, and directs members of the test team to be prepared to isolate the unit cooler upon leakage or spraying. Addition of the test loop is only for the duration of each test and no other changes (pressure, piping arrangements, positions) will take place. The system is returned to its original configuration upon completion of each test.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-082  
Implementation Document No.: EDC 2M10354  
USAR Affected Pages: Figure 9.4-22a  
System: Hot Water Glycol Heating (HVH)  
Title of Change: HVH System N<sub>2</sub> Cylinder Outlet  
Valve Configuration

**Description of Change:**

Drawings showed valves 2HVH-V290, V282, V283 and V284 as being isolation valves for four nitrogen (N<sub>2</sub>) cylinders that supply N<sub>2</sub> to the hot water glycol heating system (HVH) expansion tank 2HVH-TK1. These valves did not exist. Nitrogen cylinder outlet valves 2HVH-V286, V287 and V288 were shown as 3-way valves and valve 2HVH-V285 as an angle valve which was not correct. Valves 2HVH-V285, V286, V287 and V288 are actually 2-way gate valves with each valve isolating one cylinder and only one valve normally open with HVH in service.

This change revised the HVH system drawings to show the correct valve configuration, consistent with the as-built condition.

**Safety Evaluation Summary:**

Nitrogen is used to pressurize the plant hot water heating system to minimize oxidation of the piping system. Nitrogen is supplied to the expansion tank 2HVH-TK1 from four N<sub>2</sub> cylinders. Each N<sub>2</sub> cylinder has one isolation valve, with one valve normally open with HVH in service. Deletion of valves 2HVH-V290, V282, V283 and V284, and showing valves 2HVH-V286, V287 and V288 as N<sub>2</sub> cylinder isolation valves, are editorial changes only and conform to as-built conditions. The N<sub>2</sub> cylinders remain capable of being isolated. This system has no safety-related function.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-083  
Implementation Document No.: Temp. Mod. 91-071  
USAR Affected Pages: N/A  
System: Circulating Water  
Title of Change: Temporary Modification 91-071,  
Installation of Corrosion and  
Scaling Potential Monitoring  
Equipment

**Description of Change:**

This temporary modification consists of a corrosion and scaling monitor, with associated pump, pipe, hose and components, to analyze for corrosion and scaling potential degradation to the circulating water system equipment. The portable equipment is located downstream of the screens within the cooling tower flume house. The pump circulates approximately 7 gpm of the circulating water from the discharge flume through the monitor. After analysis, the water is returned back to the discharge flume. The monitor and associated equipment are supplied by Betz Laboratories and are powered by normal 110-V electricity. The sampling and monitoring equipment are not physically connected to the circulating water piping, but drain and return water at the discharge flume via flexible hose.

**Safety Evaluation Summary:**

The temporary addition of the corrosion and scaling equipment will have no adverse effect on the design characteristics of the circulating water system. Environmental parameters in relation to water chemistry blowdown requirements will continue to be strictly adhered to.

The water analysis equipment will not be used to perform any FSAR analysis or in lieu of any other plant equipment or system. The circulating water system is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-088  
Implementation Document No.: Temp. Mod. 91-089  
USAR Affected Pages: N/A  
System: Miscellaneous Floor Drains  
(DFM)  
Title of Change: Temporary Sump Pump  
Installation into 2DFM-SUMP1B

**Description of Change:**

A temporary sump pump was installed within 2DFM-SUMP1B to function in place of the two permanently installed pumps which were inoperable. 2DFM-SUMP1B is located in the screenwell building, el. 215'-0", north of the circulating water pumps, and collects water leaking from the circulating water pump seal piping. A flexible hose was connected to the pump and routed across the screenwell building floor, discharging into the service water system intake at the trash rake area.

**Safety Evaluation Summary:**

The use of the temporary sump pump and flexible hose does not adversely impact the miscellaneous drain system operation, or any other component or system required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features. Both the permanent and temporary sump pumps are powered by normal plant, nonsafety-related electrical distribution. The pump is of adequate size to displace any influent which may enter the sump. Flow to the sump is negligible under normal conditions. The screenwell building miscellaneous floor drain system has no potential for radioactive contamination; thus, discharge of the sump to the service water system intake will not contaminate service water.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-090  
Implementation Document No.: EDC 2E10346  
USAR Affected Pages: Figure 9.1-6 Sht. 6  
System: Spent Fuel Pool Cooling  
Title of Change: USAR Figure 9.1-6 Set Points  
Correction for the Spent Fuel  
Pool Cooling Pumps Timers

**Description of Change:**

Timers 62-2-2SFCA01/B01 delay alarm and trip of the running spent fuel pool cooling pump (P1A or P1B) in case of low-low discharge flow so that the standby pump can take over. The timers eliminate unnecessary alarms and trips in case of spurious short-time decrease of the flow and allow the flow build-up after start of the pump.

This change corrects the timer setpoints shown on USAR Figure 9.1-6 from 5.5 min. to 5 min. for pump trip, and from 3.7 min. to 3.33 min. for the alarm.

**Safety Evaluation Summary:**

This change corrects an error on USAR Figure 9.1-6 to agree with the as-built condition. No field work is required. The change does not alter the design, function, or method of performance of the system, and has no impact on the safe operation or shutdown of the plant.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-092  
Implementation Document No.: Operating Procedure N2-OP-3  
USAR Affected Pages: N/A  
System: Feedwater  
Title of Change: Throttle Capability for  
2FWS-MOV47s

**Description of Change:**

The normal method for shutting down a feedwater train is to reduce flow in the desired loop using the respective normal feedwater throttle valve (2FWS-LV10). When flow is reduced to a given level, the pump can be tripped. Should 2FWS-LV10 fail, feedwater pump flow rate should still be throttled prior to shutting down the pump. This temporary change allows throttling of feedwater flow using isolation valve 2FWS-MOV47, in the event that valve 2FWS-LV10 fails, by defeating the seal-in circuit (open/closed) for the main control room switch controlling feedwater isolation valve 2FWS-MOV47. Normal valve operation is unaffected except that the operator will be required to hold the switch until the desired valve position is obtained. The necessary actions to accomplish this temporary change are contained in an attachment to Operating Procedure N2-OP-3.

**Safety Evaluation Summary:**

This temporary modification is to be used only when the normal throttle valve (LV10) becomes inoperable and flow must be reduced to allow tripping the feedwater pump. No other system operation will be affected. Engineering analysis has demonstrated that the isolation valve (MOV47) and actuator are capable of such throttling.

The subject valves: (1) are neither part of nor contribute to the integrity of the reactor coolant pressure boundary; (2) are not required to shut down the reactor or maintain it in a safe shutdown condition; and (3) are not required to prevent or mitigate the consequences of accidents that could result in potential offsite exposures.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-093  
Implementation Document No.: Operating Procedure N2-OP-10A  
USAR Affected Pages: N/A  
System: Circulating Water (CWS)  
Title of Change: Defeat Low-Level Interlock for Pump Start

**Description of Change:**

A CWS pump cannot be started when the water box is not full. Sometimes the level switch sends a false signal that the water box is not full and the pump cannot be started. This change involves a temporary modification to Operating Procedure N2-OP-10A to permit the operators to bypass the water box full pump start permissive, after verifying that the water box is actually full prior to CWS pump start.

**Safety Evaluation Summary:**

The circulating water system is not required for safe operation or shutdown of the plant. Bypassing a faulty water box level switch will not cause the system to operate beyond its design configuration.

If the CWS pump is started when the water box is not full, a water slug could cause system damage. To assure that such damage does not occur, the operators must verify that the water box is actually full, in spite of level switch (2CWS-LS51) indication that the water box is not full. This is verified by either the "full" indicating light and computer point, or by using a Tygon tube to provide local visual indication (at the water box). Only when the operators are assured that the water box is actually full may they proceed to bypass the false (not full) signal.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-098, Rev. 1  
Implementation Document No.: Procedure N2-OP-29  
USAR Affected Pages: N/A  
System: Reactor Recirculation (RCS)  
Title of Change: Isolation of RCS Pump, Closure  
of 2RCS-SOV90A and B

**Description of Change:**

During recirculation pump operation, when the recirculation loop is not isolated, seal injection is provided from the control rod drive pumps since this supply of water has been filtered. When the recirculation pump is shut down and isolated, seal injection is not desired since continued injection would pressurize the isolated piping.

This change addresses the use of a jumper to defeat the open signal that the seal staging valve (2RCS-SOV90A or B) receives when either control rod drive pump is running, thereby allowing seal injection to be secured.

**Safety Evaluation Summary:**

The interlock associated with valve 2RCS-SOV90A or B is to ensure cooled filtered water to recirculation pump seals during operation. This change allows the interlock to be defeated only when the recirculation pumps are secured and isolated. Under the conditions described in the procedure, the shaft seals are not required to perform their design function. Worst-case failure would be gross failure of both stages of seals. The failure of both stages of the shaft seal has been analyzed. A pump shaft breakdown bushing provides redundancy in the event of gross failure of both seals.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-099  
Implementation Document No.: Procedure N2-OP-10A  
USAR Affected Pages: N/A  
System: Circulating Water (CWS)  
Title of Change: Defeat Temperature Interlock  
for Cooling Tower Bypass  
Valves

**Description of Change:**

When the cooling tower basin's temperature is low (<40°F), a temperature interlock prevents closure of the tower bypass gate valves (2CWS-MOG52A, B, C), since their function is to bypass the cooling tower and direct the water to the basin to prevent it from freezing. However, if the valves are open prior to starting the circulating water pumps, the pumps may run out due to very low system resistance. This change addresses the defeat of the tower bypass gate valves' temperature interlock (by removal of a fuse) to enable throttling of the valves, thereby creating sufficient downstream pressure to prevent pump runout.

**Safety Evaluation Summary:**

The circulating water system is not required for safe operation or shutdown of the plant. The function of gate valves 2CWS-MOG52A, B, C is to direct the circulating water to the basin and bypass the cooling tower when the CWS pumps are operating. However, this function is not required when pump flow is not yet established. These valves will be throttled only when the operators attempt to start the pump while the cooling tower basin's temperature is below 40°F. Once normal pump flow is established, the removed fuse will be replaced, and as long as the cooling tower basin's water temperature remains below 40°F, the throttled gate valve will return to its fully open position.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-100, Rev. 1  
Implementation Document No.: Procedure N2-OP-36A  
USAR Affected Pages: N/A  
System: Standby Liquid Control (SLC)  
Title of Change: Defeat of SLC Tank Low Level  
Pump Interlock from RRCS

**Description of Change:**

The redundant reactivity control system (RRCS) automatically initiates both loops of the standby liquid control system (SLCS) in the event of anticipated transient without scram. Manual initiation of the SLCS is possible from the main control room using two keylock switches.

If there is a loss of power to RRCS (taken out of service for maintenance or inoperable for any other reason), the standby liquid control (SLC) tank low level pump interlock (which is powered by RRCS) will open, preventing a manual SLCS injection.

This temporary modification addresses the use of a jumper to defeat the SLC tank low level pump interlock, thereby avoiding the loss of the standby liquid control system when the RRCS is inoperable.

**Safety Evaluation Summary:**

The SLC tank low level pump interlock protects the SLC pump from damage when the level of the liquid in the SLC tank is low. Therefore, before the operators defeat the interlock, they must first verify that the level of the liquid in the SLC tank is not at the low level. Defeating the interlock restores the operability of the SLCS under certain postulated failures of RRCS to maintain the assumptions in the USAR analyses. The design and operation of the SLC system are not altered.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-102  
Implementation Document No.: Operating Procedure N2-OP-8  
USAR Affected Pages: N/A  
System: Feedwater Heaters and  
Extraction Steam  
Title of Change: Throttle Capabilities for MOVs  
When Returning 6th Point  
Heater to Service

Description of Change:

When the 6th point heater is isolated from the feedwater system and reactor power is less than or equal to 90%, and the operators are ready to return the 6th point heater to service, the heat-up rate must not exceed 250°F/hr to avoid damaging the heater. In order to limit the heatup rate, the operators must throttle motor-operated valves 2FWS-MOV22A (or B, or C) and 2ESS-MOV3A (or B, or C). This change addresses defeating the open seal-in circuitry for the subject valves by lifting the appropriate leads in order to allow throttling of the valves. This temporary modification concludes when the 6th point heater reaches its normal operating parameters and the operators return the leads to their original design configuration.

Safety Evaluation Summary:

The throttling process proceeds slowly and is closely monitored to assure that the heat-up rate of the 6th point heater does not exceed 250°F/hr. The operators also assure that reactor power as a function of feedwater temperature is within the allowable band.

The transients analyzed in USAR Chapter 15 concerning the loss of feedwater heating are not affected and remain bounding. Technical Specification requirements regarding maintaining feedwater flow rates within design parameters as related to reactor pressure and power, and limiting the number of loss of feedwater heater transients, are also not affected.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-103  
Implementation Document No.: Temp. Mod. 91-117  
USAR Affected Pages: N/A  
System: Reactor Building Floor Drains  
Title of Change: Use of Temporary Sump Pump

**Description of Change:**

Reactor building floor drain sump pumps 2DFR-P2A and P2B, located in floor drain tank 2DFR-TK2A, were inoperable because hot water was discharged into the tank, rendering the discharge hose on each pump nonfunctional. This temporary modification installed an air-driven Wilden M8 pump to take suction from floor drain tank 2DFR-TK2A and discharge the contents of the tank through floor drain 2DNE-1401, which then routes the effluent to floor drain tank 2DFR-TK2E. Motive force for the air-driven pump was provided from a local service air system service connection.

**Safety Evaluation Summary:**

This temporary modification provides an alternate method of draining reactor building floor drain tank 2DFR-TK2A until the replacement pump discharge hoses can be installed.

This change does not impact the function of the reactor building floor drain system, does not affect the safe shutdown capability of the plant, and does not impact the plant accident analysis.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 91-110  
Implementation Document No.: Temp. Mod. 91-100  
USAR Affected Pages: N/A  
System: Reactor Manual Control System  
(RMCS)  
Title of Change: Defeat Refueling Interlocks

**Description of Change:**

Safety interlocks exist between the RMCS and the service and refuel platforms to protect personnel during refueling operations. These interlocks operate in conjunction with the MODE SWITCH setting and are intended for use during core alterations in the REFUEL and STARTUP modes while the reactor vessel head is detensioned or removed. With the refueling platform near or over the core, or the refueling grapple loaded, the RMCS prevents the withdrawal of a control rod through a rod block trip initiated by the rod block trip system (RBTS). With a control rod withdrawn, interlocks on the refuel platform prevent the platform from moving over the core and blocks movement of the hoists. This condition is also true for normal plant startups while in the STARTUP MODE because of circuitry design. As a result, the refuel platform is normally placed in its storage position during plant startups.

While modifications were being performed on the refuel platform, the refuel platform was located near or over the vessel to prevent performing the modification over the spent fuel pool. This temporary change defeated the interlocks to allow restart of the unit from an unscheduled outage without first completing the modifications to the refuel platform.

**Safety Evaluation Summary:**

Except during the performance of refueling operations with the core exposed, there is no need for the refueling interlocks on the refuel platform. Since the reactor vessel head is installed and tensioned, there is no potential for exposing personnel to unsafe radiological conditions or for damage to the fuel in the core due to positioning the refuel platform near or over the core with control rods withdrawn. This temporary change will be removed prior to detensioning the reactor vessel head and entering REFUEL MODE.

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Safety Evaluation Summary: (Cont'd)

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-001  
Implementation Document No.: Temp. Mod. 91-107  
USAR Affected Pages: N/A  
System: Service Water (SWP)  
Title of Change: Installation of Service Water  
Multiple Coupon Corrosion  
Racks and Associated  
Components

**Description of Change:**

This temporary modification added two multiple coupon corrosion racks, a biobox and associated pipe, isolation valves, hoses and components to the service water system for the purpose of collecting data on microbiologically-influenced corrosion, biofouling, and zebra mussels. This equipment required no electrical power and utilized existing service water pressure for flow through the racks from the nonsafety-related portion of the system. The equipment was connected to valve 2SWP-V24 in the screenwell building at el. 261'-0".

**Safety Evaluation Summary:**

The temporary corrosion coupon racks and associated components are minimal in size and weight and will all be contained within a small area within the screenwell building. The continual surveillance over the temporary equipment will be provided. This portion of the service water system, at the connection with valve 2SWP-V24, is not required to effect or support shutdown of the reactor, or to perform in the operation of reactor safety features.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-002  
Implementation Document No.: EDC 2F00111  
USAR Affected Pages: Figure 6.3-7a; Table 18.2-2  
Sht. 10  
System: Low Pressure Core Spray (CSL)  
Title of Change: Correction of Computer Point  
CSLEA100

**Description of Change:**

The discharge of low pressure core spray pump 2CSL\*P1 is monitored by flow element 2CSL\*FE107, which is associated with two Rosemount transmitters: differential transmitter 2CSL\*FT107 and flow transmitter 2CSL\*FT126. This change corrected the system P&ID, associated USAR Figure 6.3-7a, and USAR Table 18.2-2 to identify the source of computer point CSLEA100 as transmitter 2CSL\*FT126 rather than 2CSL\*FT107. The ERF computer software in the radwaste computer room was also corrected and verified per approved site procedures.

**Safety Evaluation Summary:**

This is primarily a documentation change to reflect the actual plant configuration and design. There are no physical panel or wiring changes required for this document correction. There is no impact on the CSL system since the ERF computer system is isolated from it. The CSL system will continue functioning as originally designed and the 2CSL\*FT126 transmitter will perform it's post-accident monitoring function as designed.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-004  
Implementation Document No.: Temp. Mod. 92-007  
USAR Affected Pages: N/A  
System: Reactor Water Cleanup (WCS)  
Title of Change: Temporary Removal of  
2WCS-RV143

**Description of Change:**

This change involved the temporary removal of relief valve 2WCS-RV143, which was not functioning properly. The valve was originally installed to provide thermal relief for the piping installed by Modification 88-059.

In order for this valve to be required to operate, the piping would have to be isolated while filled with water and exposed to an outside source of heat to cause the trapped volume of water to expand. The operating procedure (N2-OP-37) was revised with a caution to require the section of piping to be drained if it is isolated while the relief valve is removed. The procedure was also revised to fill and vent the line before the system is returned to service.

**Safety Evaluation Summary:**

The removal of 2WCS-RV143 is considered to be acceptable based on the use of administrative controls and the low probability of it actually being required. Modification 88-059 was installed to provide an outside source of cooling water to the WCS pump seals. The original system was not disabled and can be utilized if the new system is required to be isolated. The blind flange that was installed in place of the relief valve is capable of withstanding the system pressure and temperature.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-005, Rev. 1

Implementation Document No.: Procedures N2-FSP-FPM-A001-1,  
N2-FSP-FPM-A001-2,  
N2-FSP-FPM-A001-3,  
N2-FSP-FPM-A001-4,  
N2-FSP-FPM-A001-5

USAR Affected Pages: 9A.3-48a

System: Fire Detection

Title of Change: Fire Detector Surveillance  
Frequency Change

**Description of Change:**

Fire detectors in accessible locations during operations were originally required to be demonstrated operable at least once per 6 months. This change reduced the surveillance frequency so that fire detectors in accessible locations during operation will be required to be demonstrated operable at least annually. This change also clarified that supervised circuits associated with fire detector alarms are continuously monitored by their associated fire alarm panel.

**Safety Evaluation Summary:**

Surveillance records for fire detector testing at Unit 2 indicate that very few deficiencies have been revealed by fire detector surveillance testing conducted to date. The reduced surveillance frequency is in accordance with NFPA-72E, 1990, for smoke detectors, which presently requires a minimum one-year surveillance frequency, and is in accordance with NFPA-72E, 1990, paragraph 8-3.1, for the thermal detectors since the authority having jurisdiction may accept testing at less frequent intervals. The reduced frequency of fire detector testing still assures a high degree of fire detector reliability based on the number of failures experienced to date.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-008, Rev. 1  
Implementation Document No.: Temp. Mod. 92-003  
USAR Affected Pages: N/A  
System: Liquid Waste Management (LWS)  
Title of Change: Advanced Liquid Processing System (ALPS)

**Description of Change:**

This change consists of a temporary waste water processing system to process plant liquid waste while the radwaste evaporator is out of service. The system is known as the advanced liquid processing system (ALPS), supplied by Chem Nuclear Systems, Inc. This system is a temporary substitution for the functions of the flat bed filter and the evaporator.

Chem Nuclear Systems, Inc., is currently and successfully operating several ALPS units at various nuclear sites.

**Safety Evaluation Summary:**

An engineering review of ALPS information supplied by Chem Nuclear Systems and the ALPS compatibility with NMP2's systems has been performed. Aspects of the system reviewed included design, testing, ALARA considerations, floor loading, performance, fire protection, and plant services requirements.

This change will substitute one type of waste water processing system for another type without affecting the permanent plant waste monitoring and sampling systems. The processed water will still be monitored by the existing instrumentation to assure that NMP2's established effluent criteria and criteria for reuse are met prior to the release of the processed water.

In the event of a failure of the temporary waste water processing unit, such as rupture of a hose or pressure-retaining components, the unit can be isolated either at the unit or at the source. The probability of occurrence and the consequences of such an event are bounded by the analyses already contained within the USAR.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-010  
Implementation Document No.: Simple Design Change  
SC2-0046-92  
USAR Affected Pages: N/A  
System: Reactor Building Polar Crane  
Title of Change: Polar Crane Self-Adjusting  
Brake Mechanism

**Description of Change:**

This change converted the self-adjusting brake mechanism of the polar crane bridge to manually-adjusting brake mechanisms. The reason for the change is that the self-adjusting mechanism parts and linkages have been worn causing the self-adjusting feature to not function properly.

**Safety Evaluation Summary:**

An engineering review of the electrically-operated, self-adjusting brake mechanisms has been performed. Based on this review and consultation with the crane manufacturer, this change will not affect the braking capability of the polar crane bridge. The hydraulically-operated bridge brake system, the load-carrying capacity of the polar crane, and the crane's structural integrity are also not affected. The preventive maintenance program was revised to incorporate a periodic inspection and manual adjustment requirement for the brake mechanisms to assure that proper gap between the brake pads and the brake wheels is maintained.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-011  
Implementation Document No.: Temp. Mod. 92-13, EDC 2F00495  
USAR Affected Pages: N/A  
System: Service Water (SWP) and Fire Protection-Water (FPW)  
Title of Change: Alternate Cooling Water Supply to Auxiliary Boiler Circulating Pump Seal Coolers and Sample Coolers

**Description of Change:**

This temporary modification involved the installation of a temporary fire hose from fire hose reel 2FPW-FHR32 to service water system drain valve 2SWP-V1011 in order to supply cooling water to the seal and sample coolers of the auxiliary boiler circulating pumps while the normal supply of service water was isolated during Refueling Outage 2.

**Safety Evaluation Summary:**

The portions of service water and fire protection-water systems impacted by this temporary modification are not safety related and do not affect any safety-related functions of any other safety-related plant components, systems or structures. Interconnecting the nonsafety-related service water piping with the fire protection system will not degrade the function of either system or either system's functional integrity.

Fire protection capabilities will not be degraded at any time, because system, pressure and flow will be maintained in the fire protection system while this temporary modification is in effect.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-012  
Implementation Document No.: Simple Design Change  
SC2-0017-92, EDC 2F00485  
USAR Affected Pages: 7.7-37; Figure 1.2-15 Sht. 2  
System: 3-D Monicore  
Title of Change: Addition of Color Copier to  
the 3-D Monicore System

**Description of Change:**

This simple design change added a Seiko D Scan Color Copier to the existing 3-D Monicore Vaxstation (2CEC-DSPL881) located in the computer room at el. 288' of the control building. The color copier will provide hardcopies of logs and color displays available at the computer room Vaxstation without disruption of the operation of the Monicore System.

**Safety Evaluation Summary:**

Implementation of this change will enhance the operation of the 3-D Monicore System by enabling it to provide hardcopy data to the Reactor Engineering Department. Because the 3-D Monicore System is a monitoring system that performs technical evaluations and provides predictive information, but has no control or impact on the operation of the reactor, there will be no impact on the safe operation or shutdown of the plant.

In order to prevent the introduction of ground loops, power for the copier will be taken from receptacles whose ground is common with the receptacles that feed the remainder of the 3-D Monicore System. The power source for these receptacles is 2VBB-UPS1G, which will not be overloaded by the addition of the copier load.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-013  
Implementation Document No.: N/A  
USAR Affected Pages: N/A  
System: N/A  
Title of Change: Plant Manager Unit 1 Reporting  
to Executive Vice President  
Nuclear

**Description of Change:**

Currently, the Unit 1 Plant Manager reports to the Vice President Nuclear Generation. To put all of the Vice President Nuclear Generation's attention on Unit 2 during the Unit 2 refueling outage, the Plant Manager Unit 1 will report directly to the Executive Vice President Nuclear.

**Safety Evaluation Summary:**

This organizational change will provide the nuclear organization with resources to be both efficient and effective. All Technical Specification responsibilities of the Vice President Nuclear Generation will be delegated up to the Executive Vice President Nuclear. These organizational changes will not affect the safe operation or safe shutdown of the plant. This change does not constitute an unreviewed safety question and is in compliance with NRC Standards.

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Safety Evaluation No.: 92-016  
Implementation Document No.: Procedures NIP-ECA-01,  
NIP-SRE-01  
USAR Affected Pages: 1.10-33, 1.10-33a, 13.4-2,  
13.4-2a; Figure 13.4-1  
System: N/A  
Title of Change: Operations Experience  
Assessment

**Description of Change:**

This procedural change fulfills the operating experience assessment function by utilizing an alternative approach to that presently described in the USAR. The assessment function is no longer fulfilled primarily by the Operations Experience Assessment (OEA) group. The procedural changes require that the operational experience assessment function be accomplished by a responsible organization which is considered most cognizant over the operating information being evaluated. This assessment function is controlled by a Nuclear Division Interface Procedure, NIP-ECA-01, entitled "Deviation Event Report." The Deviation Event Report (DER) process, in conjunction with Nuclear Division Interface Procedure NIP-SRE-01, entitled "Operating Experience Assessment," meets the requirements of TMI Issue I.C.5.

In addition, the procedural changes eliminate the need for mandatory SORC participation every two months with the OEA group. The alternative approach allows the Plant Manager to request SORC involvement in the processing of DERs related to the operational experience assessment function on an as-needed basis.

**Safety Evaluation Summary:**

The use of the DER process to fulfill the operational experience assessment function, as mandated by TMI Issue I.C.5, is acceptable based upon the following:

1. The DER process is proceduralized,
2. Operational experience assessment for any given applicable DER is performed by the most qualified NMPC group since the selection criteria of the responsible organization for

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Safety Evaluation Summary: (Cont'd)

processing the DER is procedurally required to be the most-cognizant group for the subject matter of a given DER,

3. The DER disposition process provides a mechanism by which necessary plant actions, training, and retraining will be stipulated, and
4. SORC involvement as mandated by either Plant Manager on an as-needed basis ensures fulfillment of SORC's review function of advising the Plant Manager on matters related to nuclear safety.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-017  
Implementation Document No.: Special Test Procedure  
N2-STP-31  
USAR Affected Pages: N/A  
System: Reactor Recirculation Pump  
Title of Change: Recirculation Pump Test With  
Higher Viscosity Oil in Motor  
Bearings

**Description of Change:**

The recirculation pump/motors have experienced higher than normal vibration levels. The vibration levels at full power are acceptable; however, the frequency component at half the pump running speed (1/2 X) is high for both low speed and high speed low flow conditions. Bearing (oil) whirl is suspected as the cause of 1/2 X vibration. A frequently successful fix for oil whirl is to change the hydrodynamic characteristics of the oil. A test was performed for the recirculation pump (Loop B) utilizing a higher viscosity oil (Mobil SHC 629) in order to determine effect on vibration levels. The test was performed with the plant in the cold shutdown condition. Upon completion of the test, the high viscosity oil was replaced by the existing bearing oil.

**Safety Evaluation Summary:**

An evaluation for acceptability of the replacement bearing oil for the test was performed. The evaluation showed that the heavier oil would not affect the motor during the test from the standpoint of increased temperature and lubrication design. The special test to be performed has no impact on the safe operation or shutdown of the plant. The bearing temperature was monitored during the test, and any unusual pump running condition would have required the test to be aborted.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-019

Implementation Document No.: 23A7138, Rev. 0; Engineering Report NFD92-016; Design Report GE9B-P8CW320 - 9GZ1-100M-150-T, EDB No. 1956

USAR Affected Pages: Appendix A

System: Various

Title of Change: Operation of NMP2 Reload 2/Cycle 3

**Description of Change:**

This evaluation addressed the changes associated with the operation of Reload 2/Cycle 3. As in Cycle 2, GE9B fuel was loaded for Cycle 3. GE9B is an approved fuel design in GESTAR.

The Cycle 3 core loading inserted 248 fresh Reload 2 bundles and discharged all 136 of the 1.76% enriched initial core bundles as well as 112 of the 2.19% enriched initial core bundles to the spent fuel pool.

The Cycle 3 core operating limits, contained in the Core Operating Limits Report, Revision 0, Cycle 3, were implemented through a modification to the process computer/3-D Monicore data bank.

**Safety Evaluation Summary:**

Minimum critical power ratio (MCPR), linear heat generation rate (LHGR), and maximum average planar linear heat generation rate (MAPLHGR) limits were addressed to accommodate the Cycle 3 core composition. Expanded equipment-out-of-service operational capability, vessel pressurization, and shutdown margin evaluations were also addressed. These evaluations utilized approved analysis methodology.

Cycle 3 limits for average planar linear heat generation rate, average power range monitor setpoints, MCPR including  $K_f$ , rod block monitor setpoints and LHGR are contained in the Core Operating Limits Report.

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The Reload 2 fuel does not affect source term and spent fuel storage design bases.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-020, Rev. 1  
Implementation Document No.: Procedure S-EPP-17, Rev. 11  
USAR Affected Pages: SEP Pages 7-11, 7-35  
System: N/A  
Title of Change: Removal of Station  
Radio-Activated Pagers from  
the Site Emergency Plan (SEP)

**Description of Change:**

The radio-activated pager system consists of 36 voice message pagers that can be activated by the control room radio consoles. This pager system has been deleted. These pagers had a range of only 10 miles from the site. The limited number of these pagers, their short range, and the availability of the newer, more versatile commercial ("Bravo") type pagers with an expanded range, renders the radio-activated pager system obsolete and of no usefulness to the Emergency Preparedness programs.

**Safety Evaluation Summary:**

The deletion of the radio-activated pager system will not affect the safe operation or safe shutdown of either Unit 1 or Unit 2.

The radio-activated pager system has never been a primary system but was intended to be a backup to other emergency notification means, including Gaitronics, commercial emergency pagers, and commercial telephone systems.

Existing procedures allow for alerting, notifying, and mobilizing emergency response personnel without the use of the radio-activated pagers.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question and does not decrease the effectiveness of the Site Emergency Plan.

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Safety Evaluation No.: 92-022, Rev. 0 and 1  
Implementation Document No.: Operating Procedure N2-OP-61B  
USAR Affected Pages: N/A  
System: Standby Gas Treatment (SGTS)  
Title of Change: Use of GTS with Valves  
2GTS\*V51 and V52 Closed

**Description of Change:**

Due to operability problems with throttle valves 2GTS\*PV5A and 2GTS\*PV5B, a change to Operating Procedure N2-OP-61B was required to allow valves 2GTS\*V51 (V52) to be closed while maintaining SGTS trains A and B operable. Throttle valves 2GTS\*PV5A(B) are located on the discharge side (bypass) of SGTS fans 2GTS\*FN1A(B), and are used to modulate reactor building differential pressure at  $\leq -0.25$  in. WG when required. Valves 2GTS\*V51 and V52 are safety related and are located upstream of 2GTS\*PV5B and 2GTS\*PV5A, respectively. This change applied for the duration of the second refueling outage.

**Safety Evaluation Summary:**

The effect of securing valves 2GTS\*V51, V52 would be that valves 2GTS\*PV5A(B) would not be able to modulate flow. Thus, no air would recirculate back to the filter train inlet in the event the SGTS is operating.

With both SGTS trains operating, the required reactor building differential pressure of  $\leq -0.25$  in. WG relative to the outside atmosphere would be achieved in the same time with throttle valves 2GTS\*PV5A(B) not modulating flow since these valves, during startup of the SGTS, are in the closed position until differential pressure of  $\leq -0.25$  in. WG is achieved.

Preliminary estimates indicate that if both of the SGTS trains are left running with the present reactor building inleakage conditions and 2GTS\*V51 and V52 closed, a negative pressure of about 6 in. WG (0.216 psi) could develop in the secondary containment. Although this negative pressure is within the design capability of the building and roof, it could present

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Safety Evaluation Summary: (Cont'd)

problems for plant personnel entering, exiting, or working in the secondary containment. Therefore, when the desired negative pressure ( $\leq -0.25$  in. WG) is established, one of the GTS filter trains would be secured within 15 minutes or less for personnel ingress/egress purposes (ability to open doors at a reduced Delta P) and placed in standby mode per N2-OP-61B.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-023, Rev. 1  
Implementation Document No.: Procedure N2-PM-@18  
USAR Affected Pages: N/A  
System: Normal DC Power Distribution  
Title of Change: Temporary Change of the  
Battery Charger 2BYS-CHGR1B1  
Power Supply per Procedure  
N2-PM-@18, Step 8.12

**Description of Change:**

During preventive maintenance performance, per Procedure N2-PM-@18, stub bus load center 2NJS-US6, which is the power source to battery charger 2BYS-CHGR1B1, was de-energized for a long period of time--approximately 5 days. To avoid the discharge of batteries 2BYS-BAT1B1, 2, 3, 4, a temporary ac power supply was provided to battery charger 2BYS-CHGR1B1 from stub bus 2NJS-US5, which is also the ac power supply to battery charger 2BYS-CHGR1A1. This change was temporary and was applied only when the plant was in the shutdown condition. The original plant configuration was restored prior to exit from the procedure.

**Safety Evaluation Summary:**

The change applies only to the shutdown condition of the plant and will affect nonsafety-related normal 125-Vdc electric power systems, which are not required for safe operation or safe shutdown of the plant. An engineering evaluation determined that the temporary ac power supply to charger 2BYS-CHGR1B1 will provide sufficient power for continuous dc loads, floating charge of batteries 2BYS-BAT1B1, 2, 3, 4, and recharge of these batteries after a 2-hour emergency discharge cycle. The maximum load of charger 2BYS-CHGR1B1 is 100 kVA. The rated loads of transformers, switchgear and circuit breakers will not be exceeded, and the coordination and short circuit study will not be affected.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-024  
Implementation Document No.: QATR-1, Rev. 7  
USAR Affected Pages: 17.2-1  
System: N/A  
Title of Change: Revision 7 to the Quality Assurance Topical Report NMPC-QATR-1

**Description of Change:**

Revision 7 to the Quality Assurance Topical Report (QATR-1) has been issued. Revision 7 is a general update and clarification, including changes requested by Nuclear Division organizations since the issue of Revision 6. The changes clarify the scope of the QA program by limiting activities under 10CFR71 to procurement and use of packaging for shipment of radioactive materials, and clarification of the hierarchy of the most relevant NMPC procedures that implement the 10CFR50 Appendix B program. Editorial changes are also included as required.

**Safety Evaluation Summary:**

The QATR-1 Revision 7 was transmitted to the NRC in letter NMP1L 0654, dated March 25, 1992, to satisfy the requirement of 10CFR50.54 for an annual update to the QA Program. As noted in NMP1L 0654, the changes included in QATR-1 Revision 7 do not reduce the effectiveness of the QA Program. Thus, these changes will not adversely affect the safe operation or shutdown of the plant and do not involve an unreviewed safety question.

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Safety Evaluation No.: 92-025  
Implementation Document No.: Procedure N2-PM-@19  
USAR Affected Pages: N/A  
System: Normal DC Power Distribution  
Title of Change: Temporary Change of the  
Battery Charger 2BYS-CHGR1A1  
Power Supply per Procedure  
N2-PM-@19, Step 8.12

**Description of Change:**

During preventive maintenance performance per Procedure N2-PM-@19, stub bus load center 2NJS-US5, which is the power source to battery charger 2BYS-CHGR1A1, was de-energized for a long period of time--approximately 5 days. To avoid the discharge of batteries 2BYS-BAT1A1, 2, 3, 4, a temporary ac power supply was provided to battery charger 2BYS-CHGR1A1 from stub bus 2NJS-US6, which is also the ac power supply to battery charger 2BYS-CHGR1B1.

This change was temporary and was applied only when the plant was in the shutdown condition. The original plant configuration was restored prior to exit from the procedure.

**Safety Evaluation Summary:**

The change applies only to the shutdown condition of the plant and will affect nonsafety-related normal 125-Vdc electric power systems, which are not required for safe operation or safe shutdown of the plant.

An engineering evaluation determined that the temporary ac power supply to charger 2BYS-CHGR1A1 provided sufficient power for continuous dc loads, floating charge of batteries 2BYS-BAT1A1, 2, 3, 4, and recharge of these batteries after a 2-hour emergency discharge cycle. The maximum load of charger 2BYS-CHGR1A1 is 100 kVA. The rated loads of transformers, switchgear and circuit breakers are not exceeded, and the coordination and short circuit study is not affected.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-027  
Implementation Document No.: Procedure N2-PM-@12  
USAR Affected Pages: N/A  
System: Normal Lighting (LAR), Normal AC Power Distribution (NJS), Emergency AC Power Distribution (EJS)  
Title of Change: Procedure N2-PM-@12, Temporary Power Feed of 2LAR-PNL200 from 2NJS-US2

Description of Change:

During station refueling outages it is necessary to remove Division I buses from service to perform required maintenance. With Division I buses out of service, normal reactor building lighting is also de-energized.

This change provided temporary power to the reactor building lighting distribution panel 2LAR-PNL200 from the nondivisional switchgear 2NJS-US2 to allow continued work in the reactor building. This change was temporary and remained in effect only during Mode 5 of plant operation with all fuel off-loaded. After completion of the Division I power outage, 2LAR-PNL200 was once again returned to bus 2EJS\*US1.

Safety Evaluation Summary:

Power for the reactor building lighting distribution panel 2LAR-PNL200 is normally fed from the Division I, Class 1E, 600-V bus 2EJS\*US1. The breaker for reactor building lighting is tripped upon sustained undervoltage on the load center or by a LOCA signal to eliminate any adverse effects the lighting load might have on (1) diesel loading and starting, and (2) secondary containment drawdown time due to unnecessary heat loads. When the reactor is in cold shutdown (Mode 5) and secondary containment is not required, reactor building lighting may remain energized since there can be no adverse effects on secondary containment drawdown time.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-028

Implementation Document No.: E&DCR C47471; Specification E035A, Addendum 4; Calculations EC-81, Rev. 3, EC-138, Rev. 1

USAR Affected Pages: 8.3-12, 8.3-28

System: Uninterruptible Power Supply (UPS)

Title of Change: Addition of Vendor-Approved Setpoints to Existing Manual for UPS (2VBA\*UPS2A, 2VBA\*UPS2B, 2VBB-UPS3A, 2VBB-UPS3B)

**Description of Change:**

This change documented operating and alarm setpoints received from the vendor and included them in the vendor manual for UPS units 2VBA\*UPS2A, 2VBA\*UPS2B, 2VBB-UPS3A and 2VBB-UPS3B.

This change also included changes to USAR Sections 8.3.1.1.2 and 8.3.1.1.3 to make the USAR consistent with the as-built plant configuration and procurement specifications for the UPS system, as follows:

1. Output frequency stability was corrected from 60 Hz  $\pm$  0.3 Hz to 60 Hz  $\pm$  0.5 Hz.
2. For the reactor protection system UPS units (2VBB-UPS3A, 3B), the output voltage was revised from 120 V  $\pm$  2% to 124 V  $\pm$  2%.

**Safety Evaluation Summary:**

The existing vendor manuals for the UPS units do not provide all the setpoints required for the UPS units. Concurrence from the vendor will be obtained to ensure that the as-built setpoints are appropriate.

The USAR changes will not affect the performance of the UPS units. These changes were approved prior to the procurement of the UPS but were not incorporated in the USAR.

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Safety Evaluation No.: 92-028

Safety Evaluation Summary: (Cont'd)

The UPS units will continue to operate within the design parameters specified by the vendor and will perform the function described in the USAR.

Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question.

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Safety Evaluation No.: 92-029  
Implementation Document No.: Procedure N2-PM-@13  
USAR Affected Pages: N/A  
System: Service Water (SWP)  
Title of Change: Procedure No. N2-PM-@13

Description of Change:

The SWP pump discharge header cross-connect isolation valves (2SWP\*MOV50A/B) normally close automatically upon a total loss of offsite power (LOOP). They are designed to remain open on a partial LOOP. This design is to maintain the service water system within its hydraulic analysis.

During the Division II switchgear outage, the Division II SWP pumps were out of service prior to de-energizing and after re-energizing the Division II switchgear. During these time periods when these pumps were out of service with the switchgear still energized, a loss of Division I power and resultant loss of Division I SWP pumps would effectively result in the SWP system being configured as if a total LOOP had occurred. This temporary change required that a licensed operator be stationed at panel 601 to immediately close Division II powered cross-connect valve 2SWP\*MOV50B to manually accomplish header isolation, in lieu of the automatic function that would not occur due to the switchgear being still energized.

Safety Evaluation Summary:

The substitution of manual operator action for automatic actuation of the closure of the service water system cross-connect header will effectively implement the automatic design feature of the system and maintain the intent of the design. The operator would immediately be aware of the loss of Division I power (by being in the control room) and could quickly initiate the closure of the 2SWP\*MOV50B. Additionally, the likelihood of the loss of Division I power during this short time period is very small.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-030  
Implementation Document No.: Temp. Mod. 92-035  
USAR Affected Pages: N/A  
System: Turbine Building Drains (DFT)  
Title of Change: Draining of the Condenser Pit,  
2DFT-SUMP1

**Description of Change:**

This temporary change consisted of using a supplementary pump to assist the permanent condenser pit sump pump, 2DFT-P2, in draining the condenser pit into the turbine building floor drain sump, 2DFT-TK1C. The pump utilized was an air-driven pump using the plant service air system (SAS) as the air source. Adequately rated rubber hoses were used to transfer the water. The pump's suction hose was submersed into the sump from which the water was being transferred, and the pump's discharge was directed into the discharge sump without connecting to permanent plant valves.

**Safety Evaluation Summary:**

This change will transfer water from one sump to another within the same system. The overall operation will be simple and nondetrimental to the plant's systems or components. The implementation of this change will not have a significant impact on the service air system's capacity to meet other services demands of the plant. The turbine building drains and service air systems are not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-031, Rev. 0, 1 and 2  
Implementation Document No.: Procedure No. N2-PM-012 and  
N2-PM-013  
USAR Affected Pages: N/A  
System: Reactor Building Ventilation  
(HVR)  
Title of Change: Procedure No. N2-PM-012 and  
N2-PM-013

**Description of Change:**

During the refueling outage, the Division I/II power buses were removed from service, one at a time, to perform required maintenance or modifications. During this period it was desirable to maintain the HVR system functional. This temporary change provided an alternate nondivisional source of power to the divisional group of isolation dampers affected by the bus outage in order to maintain the HVR system functional. Although the isolation dampers were maintained energized by the alternate source discussed above, the "damper open" permissive was lost since it is achieved through a divisional optical isolator. This change altered the permissive wiring to allow the fans to run while the divisional optical isolators were de-energized. In addition, the drywell fan override switch was placed in the override position to keep the drywell unit coolers operational.

**Safety Evaluation Summary:**

Although use of nondivisional power for these safety-related dampers is not per design as described in the USAR, the operation and function of the dampers on the unaffected division and the HVR system will not be adversely affected by the temporary change. Isolation dampers on the unaffected division would operate as before since changes will be made to only one Division at a time. During the time period for which this temporary change will be in effect, secondary containment will not be required.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-032  
Implementation Document No.: Procedure No. N2-PM-012  
USAR Affected Pages: N/A  
System: Service Water (SWP)  
Title of Change: Procedure No. N2-PM-012

**Description of Change:**

During the process of accomplishing and restoring from the Division I switchgear outage, the Division I SWP pumps were out of service prior to de-energizing and after re-energizing the Division I switchgear. During these time periods when these pumps were out of service with the switchgear still energized, had there been a loss of Division II power and resultant loss of Division II SWP pumps, the SWP system would effectively be configured as if a total loss of offsite power (LOOP) had occurred. This temporary procedure change required a licensed operator to immediately close Division I powered cross-connect valve 2SWP\*MOV50A to manually accomplish header isolation, in lieu of the automatic function that would not occur due to the switchgear being still energized, in the event power was lost to the Division II electrical bus.

**Safety Evaluation Summary:**

The SWP pump discharge header cross connect isolation valves (2SWP\*MOV50A/B) normally close automatically upon a total loss of offsite power (LOOP). They are designed to remain open on a partial LOOP. This design is to maintain the service water system within its design bases. The manual operator action to close 2SWP\*MOV50A, if required, effectively implements the automatic design feature of the system and maintains the intent of the design.

The operator would immediately be aware of the loss of Division II power (by being in the control room) and could quickly initiate the closure of the 2SWP\*MOV50A. Additionally, the likelihood of the loss of Division II power during this short time period would be very small.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-033, Rev. 1  
Implementation Document No.: Procedure N2-OSP-RHS-R@009  
USAR Affected Pages: N/A  
System: Residual Heat Removal (RHS)  
Title of Change: Operations Procedure  
N2-OSP-RHS-R@009

**Description of Change:**

This safety evaluation covered procedure N2-OSP-RHS-R@009 which allowed the testing of the pressure isolation valves in the RHS system which isolate the RHS heat exchanger from the reactor core cooling injection system (ICS). The steps of the procedure delineated the methodology for testing the system in order to comply with Technical Specifications 4.0.5 and 4.4.3.2.2. Testing of the valves in the RHS system was conducted during Refuel Outage 2, with the fuel removed from the reactor vessel. The system test boundary was established, filled with demineralized water compatible with the system water quality requirements, vented and filled until free of air, and then pressurized to 1000 to 1040 psig by the control rod drive hydraulic system pumps to evaluate the measured leakage past pressure isolation valves 2RHS\*MOV22A, MOV22B, MOV80A, and MOV80B.

**Safety Evaluation Summary:**

This procedure and method of testing will have no impact on the safe operation or capability to keep the plant in the safe shutdown condition because the ICS and RHS system safety functions are not required in the plant condition with the fuel removed from the vessel. A safety valve will be installed in the test boundary to provide overpressure protection. During the test, the CRD pump will supply flow to the control rod drive mechanisms and any makeup capability to the test boundary due to leakage past the tested valves. The CRD pumps are designed to provide the required flow and pressure to perform both functions simultaneously.

Based on the evaluation performed, it is concluded that this test does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-048

Implementation Document No.: Temp. Mod. 91-107; Chemistry  
Technical Procedure  
N2-CTP-SWP-D612

USAR Affected Pages: N/A

System: Service Water (SWP)

Title of Change: Installation and Use of  
Service Water Biocide  
Injection and Monitoring  
Equipment

**Description of Change:**

The installation and use of this temporary modification is for the purpose of injecting biocide into the service water system to control microbiologically-induced corrosion (MIC) and reduce heat transfer surface fouling (reference NRC Information Notice 90-39 and Generic Letter 89-13).

NaBr and NaOCl (sodium bromide and sodium hypochlorite) are fed to the service water via a side-stream feed pump system. The equipment is installed in the screenwell building. Only the post-treatment rack is physically connected to service water piping at valve 2SWP-V24.

**Safety Evaluation Summary:**

Operation of the service water system will not be affected by this change and will continue to be within the original design bases. The affected portion of the service water system, at the connection at valve 2SWP-V24, is not required to effect or support shutdown of the reactor or to perform in the operation of reactor safety features. In addition, the biocide will not affect the safety of any components that are cooled by service water. The biocide injection and monitoring equipment and associated components are minimal in size and weight and will be contained within a small area within the screenwell building. Appropriate procedural controls for operation of the equipment will be implemented.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-053  
Implementation Document No.: Temp. Mod. 92-063  
USAR Affected Pages: N/A  
System: High Pressure Core Spray  
Title of Change: Clamping Device for  
2CSH\*MOV118

**Description of Change:**

This temporary change consisted of the installation of a clamp on the valve stem of 2CSH\*MOV118, the containment isolation valve which isolates the high pressure core spray (CSH) pump suction from the suppression pool. The clamp provided retentive force on the valve stem to preclude the stem from rising while the actuator was de-energized and disassembled for inspection and maintenance. Valve 2CSH\*MOV118 provides a containment isolation function by providing isolation and long-term leakage control from the suppression pool.

**Safety Evaluation Summary:**

During the condition when the clamp is installed, the HPCS system will remain inoperable; however, the containment isolation function will be maintained.

An engineering review determined that the clamp will maintain the valve disc in the closed and leak-tested condition, thus providing the containment isolation low leakage capability normally provided by the Limitorque actuator. The review considered the post-accident loading on the valve stem and disc resistance of the packing friction, and the additional resistance provided by the clamping friction force on the valve stem.

Based on the evaluation performed, it is concluded that this temporary change does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-060  
Implementation Document No.: N/A  
USAR Affected Pages: 9A.2-4, 9A.3-1a, 9A.3-2,  
9A.3-3, 9A.3-7, 9A.3-31,  
9A.3-31a, 9A.3-32, 9A.3-45;  
Tables 9A.3-4 Sht. 6,  
9A.3-9 Sht. 1  
System: Fire Protection  
Title of Change: USAR Fire Protection Update

**Description of Change:**

The Fire Hazards Analysis (USAR Appendix 9A) has been revised to reflect minor editorial corrections, program changes and fire loading revision to certain fire zones resulting from revised analysis.

**Safety Evaluation Summary:**

The editorial corrections, revised definitions, and USAR text provide consistency with implementing documents. Minor program realignment resulted in some descriptive text changes, and minor fire loading increases occurred in two existing fire zones. Additional fire loading was added to an area of the radwaste building where existing fire load is very minor. Additional fire loading was also evaluated for the main control room (MCR) to account for paper and ordinary combustibles due to office operations. The MCR is a heavily loaded fire area due to the large amount of wiring and control cabinets. However, the area is continuously occupied, provided with automatic fire detection and under floor fixed fire suppression for critical circuits. The fire loading increases were insignificant due to already existing fire protection considerations. None of the changes are inconsistent with the guidance provided in BTP CMEB 9.5-1 and no existing commitments are impacted.

Based on the evaluation performed, it is concluded that this change is in accordance with the operating license for Nine Mile Point Unit 2 and does not involve an unreviewed safety question.

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Safety Evaluation No.: 92-065

Implementation Document No.: LDCN U-1657

USAR Affected Pages: 7.5-7; Table 7.5-2 Shts. 1 through 48

System: Post-Accident Monitoring and Display Instrumentation (Reg. Guide 1.97)

Title of Change: Update of NMP-2 FSAR Section 7.5.2-1, "Compliance with Regulatory/Industry Standards," and FSAR Table 7.5-2, "Conformance to Regulatory Guide 1.97 (Revision 3)"

**Description of Change:**

This change updates FSAR Section 7.5.2.1 and Table 7.5-2 as follows:

1. Incorporates explicit reference to the basis of and justification for deviations from instrument design and qualification recommendations of Regulatory Guide 1.97 (RG 1.97), Revision 3, consistent with information previously transmitted by NMPC to the NRC and the results of the NRC's safety evaluation of NMP2 conformance to RG 1.97 (Appendix M of Supplement 4 to NUREG-1047, and Section 7.5.2.2 of Supplement 5 to NUREG-1047).
2. Corrects various entries in Table 7.5-2 as appropriate to reflect actual design features (range, location, power supply, etc.) and qualification status of listed RG 1.97 instrumentation.
3. Revises the format of the table to more clearly/concisely identify plant-specific variable type and instrument category.
4. Revises the format of the table to clearly identify display instrumentation ID numbers (rather than just identifying sensor ID numbers).

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Description of Change: (Cont'd)

5. Revises the format of the table to clearly show applicability of equipment qualification (EQ), seismic qualification (SQ), and safety classification to the entire instrument loop.
6. Adds instrument listings for the following plant parameters:
  - a. Main steam line radiation
  - b. Reactor vessel water level - shutdown range and upset range
  - c. Off-gas system effluent radioactivity
  - d. Standby gas treatment system effluent radioactivity
  - e. Turbine building ventilation effluent radioactivity
  - f. Reactor building ventilation above and below refuel floor effluent radioactivity
  - g. RHR heat exchanger bypass valve position

**Safety Evaluation Summary:**

This change will correct and enhance documentation concerning the implementation of RG 1.97 at Unit 2. This change does not physically add any new instrumentation, nor does it delete or in any way physically modify any existing (currently installed) instrumentation. Also, implementation of this change does not require making any changes to any plant operating, maintenance, or instrument calibration instructions specified in existing plant procedures.

This modification will have no adverse impact on the safe operation or shutdown of the plant.

Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

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USAR TEXT, TABLE AND FIGURE CHANGES  
(BASED ON PREVIOUSLY REPORTED SAFETY EVALUATIONS)

A number of text and figure revisions were made to the USAR to include additional changes that are based on previously reported safety evaluations. These changes are identified below.

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Safety Evaluation No.: 89-047, Rev. 1  
Previously Reported: 10/30/91

USAR Figure 9.3-10j has been updated, per EDCs 2M10092, 2M10108, 2M10095, 2M10094, and 2M10088, to reflect the completion of additional sump pump replacements included in Modification PN2Y89MX038, as described in Safety Evaluation 89-047, Rev. 1

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Safety Evaluation No.: 91-019, Rev. 1  
Previously Reported: 10/30/91

USAR Figure 9.2-1c has been revised to delete a flange shown on the piping to radiation monitoring cabinet 2SWP\*CAB23A, per EDC 2F00310A. This flange was removed when the original carbon steel piping was replaced with stainless steel, as described in Safety Evaluation 91-019, Rev. 1.

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