Enclosure to
NMP2L 1596

NINE MILE POINT - UNIT 2

SAFETY EVALUATION SUMMARY REPORT

1995

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

Safety Evaluation No.: 87-046

Implementation Document No.: Mod. PN2Y87MX063

USAR Affected Pages: Figure 10.1-5b

System: Condensate (CND)

Title of Change: An Addition of Cyclone Separator to Condensate Booster Pump Seal Water Injection Lines

Description of Change:

This modification installed two new cyclone separators on the seal injection water lines of each of the condensate booster pumps. Also, a new flow restriction orifice was installed upstream of each cyclone separator and associated valves.

Safety Evaluation Summary:

This modification is in accordance with ANSI B31.1-1973. This new equipment interfaces only with the CND system and has no impact to any other systems. This modification will ensure condensate booster pump reliability and prevent costly pump downtime for maintenance on mechanical seals.

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

Safety Evaluation No.: 89-075 Rev. 7 & 8

Implementation Document No.: Mod. PN2Y87MX038

USAR Affected Pages: Figures 9.5-8 Sh 1 & 2, 9.5-10 Sh 1, 9.5-24, 9.5-29

System: Communications (COJ, COP, COS)

Title of Change: Addition of Communication Equipment

Description of Change:

Modifications to the Gaitronics communications system were previously reported with USAR revisions dated October 30, 1991, October 29, 1992, and October 29, 1993, under Safety Evaluation 89-075, Revisions 3, 4, 5, 6 and 7.

Additional modifications to add/improve speakers, jacks, and associated equipment to the Gaitronics communications system have been made as evaluated under Safety Evaluation 89-075, Revisions 7 and 8.

Safety Evaluation Summary:

This modification will add/improve communication capabilities to meet the requirements in USAR Section 9.5.2. These changes will improve communication capabilities required for surveillance testing, personnel to respond to alarms in areas with high noise levels, and add communication equipment in areas that have been identified as needing communication capabilities.

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

Safety Evaluation No.: 91-003 Rev. 7, 8, & 9

Implementation Document No.: Calculation ES-269

USAR Affected Pages: N/A

System: Secondary Containment

Title of Change: Secondary Containment Drawdown Analysis

Description of Change:

Revisions 7, 8, and 9 of the safety evaluation evaluated plant operation for the remainder of the fourth fuel cycle. The following parameters were changed for the ΔT requirements for the fourth cycle as compared with the previous cycle.

1. <u>Spent fuel heat loads:</u>

   A spent fuel pool heat load of 4.49×10^6 Btu/hr corresponding to 50 days after reactor shutdown (DARS) was used to define the ΔT requirements for the fourth fuel cycle.

   In order to reduce the ΔT requirements and, hence, heating of the building during the summer months, a lower spent heat load of 2.31×10^6 Btu/hr corresponding to 180 DARS was used to define the ΔT requirements for the remainder of the fourth refueling cycle.

2. <u>Unit cooler performance</u>:

   Based on the performance tests performed during the 1992-93 time period, a 2% degradation of unit coolers 2HVR*UC413A & B and an average degradation of 30% (same as previous cycle) for the remaining drawdown related unit coolers was used for defining the ΔT requirements for the entire fourth operating cycle. This provides sufficient margin to account for any further degradation that may occur over the next operating cycle.

3. <u>Piping heat load reductions</u>:

   To reduce the ΔT requirements, piping heat loads have been reduced assuming a minimum temperature of 80°F (Curve 2) and 90°F (Curves 3 and 4) in the building.

Safety Evaluation No.: 91-003 Rev. 7, 8, & 9 (cont'd.)

Description of Change: (cont'd.)

4. <u>Cubicle ΔT</u>:

   For additional flexibility, drawdown analysis is done assuming emergency core cooling system, residual heat removal heat exchangers and reactor core isolation cooling cubicles in secondary containment are maintained at 8°F above the service water temperature.

   The secondary containment in-leakage test performed on October 27, 1993, indicated that in-leakage is less than 90% of the value used in the drawdown analysis. This provides a margin of about 10% (same as the previous cycle) for any potential degradation of in-leakage over the next cycle.

NOTE: These changes were superseded by Safety Evaluation 94-049 and associated License Amendment No. 56.

Safety Evaluation Summary:

The drawdown analysis (Calculation ES-269 and subsequent dispositions) provides four curves that define ΔT requirements for the entire fourth fuel cycle.

Based on the evaluation performed, it is concluded that the use of new ΔT curves does not involve an unreviewed safety question.

Safety Evaluation No.: 91-068

Implementation Document No.: Mod. PN2Y89MX080

USAR Affected Pages: Figures 9.3-12h, 9.3-12k, 10.1-8b

System: Turbine Building Miscellaneous Drains

Title of Change: Reboiler Steam Line Drain Valve Interlock

Description of Change:

The turbine plant miscellaneous drain system removes condensate buildup from the steam supply lines either through the drain valves 2DTM-AOV128 and 2DTM-AOV144 or through bypass lines around these drain valves through restricting orifices sized to pass condensate.

The original design required the drain valves to open whenever the auxiliary steam supply valves 2ASS-STV112 and/or 2ASS-STV143 close, or whenever turbine first-stage pressure indicated insufficient extraction steam was available.

The interlocks between the auxiliary steam supply valves and their corresponding drain valve have been removed. This modification allows operator control of the drain valves irrespective of the steam supply to the clean steam reboilers and/or the building heating intermediate heat exchangers within the boundaries allowed by the turbine first-stage pressure sensor.

Safety Evaluation Summary:

The drain valves are nonsafety related and are not required for safe operation or shutdown of the plant.

This modification provides additional increased operator control which will result in an enhancement to plant efficiency that will not impact the safe operation or shutdown capabilities of the plant.

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

Safety Evaluation No.: 91-080

Implementation Document No.: Mod. PN2Y88MX193

USAR Affected Pages: 8.3-10

System: Low-Voltage Molded-Case Circuit Breakers for Power Distribution

Title of Change: Replacement of Obsolete ITE Molded-Case Circuit Breakers

Description of Change:

This modification replaced six distribution panels in their entirety and various obsolete molded-case circuit breakers in motor control centers and other distribution panels. These breakers provide circuit protection for the low-voltage power distribution at Unit 2.

Safety Evaluation Summary:

This modification does not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the USAR. Replacement breakers are properly coordinated and adequately sized to their application in accordance with the standard ratings for the molded-case circuit breakers. Replacement of obsolete breakers and panels will preclude system outages, LCOs, and plant outage due to unavailable spares should any of these components fail in-service or surveillance testing.

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

Safety Evaluation No.: 91-089

Implementation Document No.: Simple Design Change SC2-0140-90

USAR Affected Pages: Figures 11.2-1d, 11.2-1g

System: Liquid Radwaste Management (LWS)

Title of Change: Retire Non-functional Conductivity Monitors

Description of Change:

This simple design change retired in place conductivity elements and conductivity indicating transmitters from the floor drain collector subsystem and the regenerant waste subsystem. These instruments provide display input only and have no logic function. Sparing the conductivity monitoring equipment eliminates repetitive maintenance and calibration. Grab samples are used for determining effluent conductivity in those areas where electronic monitoring is disabled.

Safety Evaluation Summary:

The LWS system provides diverse options for the processing of waste depending on the quality of the waste. However, in no case can the waste bypass a filtration or evaporation process. The conductivity of the waste is used to aid in the selection of a process method. Although grab samples will have to be used in lieu of electronic monitoring, no sacrifice to the integrity or function of the LWS will occur from the proposed change.

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

Safety Evaluation No.: 92-006 Rev. 1, 2, 4 & 5

Implementation Document No.: Mod. PN2Y89MX077

USAR Affected Pages: 4A-2, 5.2-21, 5.2-21a; Tables 3B-3 Sh 2, 6.2-56 Sh 7, 9.4-1 Sh 4; Figures 1.2-7 Sh 2, 1.2-11 Sh 3, 5.4-2b, 5.4-16a, 9.3-5g, 12.3-7, 12.3-40

System: Crack Arrest Verification

Title of Change: Installation of the Crack Arrest Verification System and RWCU Extension Tie-In

Description of Change:

This modification implemented the following changes:

1. Installation of the crack arrest verification system (CAVS) included a crack length monitor, water chemistry station, electrochemical potential monitor and tubing. The suction side of the CAVS was connected to the reactor recirculation system (RCS) sample line, downstream from the outboard isolation valve, 2RCS*SOV105, beyond the Class 2 line classification (i.e., connection will be made where the line is designated as Class 4). The return line of the CAVS was connected to the reactor water cleanup (RWCU) extension tie-in which is downstream from outboard isolation valve 2WCS*MOV112.

2. The RWCU extension tie-in begins at line 2-WCS-008-88-3, which is downstream of outboard containment isolation valve 2WCS*MOV112 in the RWCU valve cubicle located on elevation 240', secondary containment. Existing valves 2WCS-V45 and 2WCS-V46 were replaced with 3/4-inch pipe. The test connection, which is used during the leak rate testing of valve 2WCS*MOV112, was maintained by adding a threaded cap and two new valves, 2WCS-V431 and 2WCS-V432. The RWCU extension tie-in also included the 3/4-inch pipe run, including one isolation valve, 2WCS-V390, and a check valve, 2WCS-V392, which are inside the RWCU valve cubicle, a penetration (i.e., W-7512-C) through the cubicle wall, and an isolation valve, 2WCS-V391, outside the cubicle, which were added via Temporary Modification 90-054 and are now permanent per this modification.

Safety Evaluation No.: 92-006 Rev. 1, 2, 4 & 5 (cont'd.)

Safety Evaluation Summary:

The CAVS is not a safety-related system nor does it perform any safety-related function, and its addition to the Unit 2 design does not affect the safety and reliability of Unit 2. The system's function is to collect data to provide an indication of the performance of plant materials in the boiling water reactor environment. The Class 3 section of RWCU extension tie-in is considered Q and the Class 4 section of the RWCU extension tie-in is nonsafety related. Both of these sections are properly designed and will not affect the safe operation or safe shutdown capability of the plant.

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

Safety Evaluation No.: 92-033 Rev. 2

Implementation Document No.: Procedure N2-OSP-RHS-R@009

USAR Affected Pages: N/A

System: Residual Heat Removal (RHS)

Title of Change: Procedure N2-OSP-RHS-R@009

Description of Change:

This safety evaluation evaluated changes to procedure N2-OSP-RHS-R@009, which allows 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 delineate 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 outages.

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 functions are not required in operational conditions 4 or 5 and the RHS system safety functions are unaffected.

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

Safety Evaluation No.: 93-004

Implementation Document No.: Temporary Mod. 93-008

USAR Affected Pages: N/A

System: 4.16-kV and 600-V Normal Ac Distribution

Title of Change: Alternate Feed to Transformer 2NJS-X1F

Description of Change:

The 600-V unit subdistribution transformer 2NJS-X1F (feeder to unit sub 2NJS-US5) was temporarily powered from 4.16-kV stub bus 2NNS-SWG015 instead of its normal source, 2NNS-SWG014, which was out of service for repair of a cracked bushing in cubicle 14-6.

Safety Evaluation Summary:

This temporary modification does not affect or deal with any safety-related equipment in the plant. An Engineering review of the USAR, Technical Specifications and related documents indicates that this temporary modification is acceptable with procedural controls and limitations. With this change in place, the normal ac distribution will continue performing its intended function.

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

Safety Evaluation No.: 93-008

Implementation Document No.: Simple Design Change SC2-0008-93

USAR Affected Pages: Figure 10.1-6c

System: Feedwater (FWP)

Title of Change: Seal Water Injection Strainer Drain Valve and Pressure Gage

Description of Change:

This simple design change added a differential pressure indicator and drain valves to the seal injection duplex strainers to facilitate drainage for periodic maintenance of the strainers.

Safety Evaluation Summary:

This change is specific to the FWP system. No significant effects on any other plant systems and/or interlocks are being introduced.

The addition of the differential pressure gage and drain valves will improve the performance of maintenance on the subject strainers. Therefore, the system maintainability/availability is improved due to the ease in periodic changeout of the strainer baskets.

The system structural integrity will not be significantly affected by this change because the weight of the valves and the differential pressure gage are negligible relative to the piping size and schedule.

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

Safety Evaluation No.:	93-011
Implementation Document No.:	Temporary Mod. 93-015
USAR Affected Pages:	N/A
System:	Auxiliary Boiler
Title of Change:	Defeat Seal Cooling Low Flow Trip for Auxiliary Boiler Recirculation Pumps

Description of Change:

This temporary modification jumpered the auxiliary boiler recirculation pump seal cooling water flow switches in order to allow the pumps to run when seal water flow is throttled back. Throttling the seal water is done to reduce the frequency of required boiler blowdowns.

Safety Evaluation Summary:

Although this temporary change may result in damage to the pump seals, there is no nuclear safety significance to the proposed change since the auxiliary boilers are not required for the safe shutdown of the reactor. The risk associated with operating the boiler with the low flow trip defeated is acceptable since plant impact will be limited to boiler operability. Implementation of the proposed change does not constitute an unreviewed safety question.

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

Safety Evaluation No.: 93-017 Rev. 2

Implementation Document No.: Simple Design Change SC2-0375-91

USAR Affected Pages: N/A

System: Standby Liquid Control (SLS)

Title of Change: RRCS Logic Change to Preclude SLS Inoperability

Description of Change:

This modification changed the storage tank level zero logic in the redundant reactivity control system (RRCS) panels from deenergize-to-trip to energize-to-trip. Previously, if a RRCS panel was taken out of service, the respective SLS loop would become inoperable because SLS identified the RRCS out-of-service signal as a SLS tank level zero. The SLS tank level zero interlock disables the SLS pumps to protect them from damage due to running them dry. Previously, temporary jumpers needed to be installed if the RRCS panels were taken out of service to maintain SLS operability. This change eliminates the need for these jumpers and provides annunciation in the main control room to alert the operators if the storage tank level zero alarm is activated. This modification was incorporated into SLS and RRCS by changing the logic in RRCS panels 2CEC*P001 and 2CEC*P002 from deenergize-to-trip to energize-to-trip. This was done by minor panel wiring changes to the ac load driver printed circuit board in which it will no longer invert the alarm signal. Because of this logic change, new programmable read only memory integrated circuits were installed in the self test circuitry of RRCS. Minor wiring changes to the storage tank level zero interlock circuitry were made in panels 2CEC*PNL618 and 2CEC*PNL629 to accommodate this logic change in the RRCS panels to energize-to-trip.

Safety Evaluation Summary:

This modification will keep SLS operable regardless of the status of the RRCS panels and without the need for temporary jumpers. This modification has no impact on the safe operation or shutdown of the plant. Nuclear safety is enhanced in that temporary jumpers now do not need to be installed when a RRCS panel is out of service to keep SLS operable.

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

Safety Evaluation No.: 93-018 Rev. 0, 1, 2 & 3

Implementation Document No.: Mod. PN2Y91MX054

USAR Affected Pages: 5.4-44, 5.4-45; Figure 5.4-16f

System: Reactor Water Cleanup (WCS)

Title of Change: RWCU F/D Improvements

Description of Change:

This modification changed the reactor water cleanup (RWCU) filter demineralizer (F/D) system as follows:

1. Replaced septa in F/D vessels A, B, C, D with a new design.

2. Revised resin feed system to include replacement of the metering feed pumps with an eductor arrangement.

Safety Evaluation Summary:

The changes to the RWCU F/D system will enhance the system by making it easier for the operator to control and provide more operator options thereby increasing flexibility, and improve precoating of the F/D vessels. Ultimately, the system run cycles will increase and better utilization of precoat material will be achieved.

The proposed changes are nonsafety related and will have no impact on the safe operation or shutdown of the plant. Reactor water chemistry limits outlined in Regulatory Guide 1.56 Rev. 1, Table 1, and specified in Technical Specifications Table 3.4.4-1, will be maintained.

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

Safety Evaluation No.: 93-020

Implementation Document No.: Simple Design Change SC2-0049-93

USAR Affected Pages: Figure 9.4-10e

System: Radwaste Building Ventilation

Title of Change: Radwaste Control Room Noise Improvement

Description of Change:

Safety Evaluation 93-020 was previously reported in October 1994 when the Unit 2 USAR was revised to reflect replacement of the 7.5 hp return/exhaust air fans with new 3.0 hp fans.

This revision to the USAR revises the flow diagram to show a reduced flow of 10,700 cfm.

Safety Evaluation Summary:

This design change will improve environmental and working conditions in the Radwaste Control Room by reducing noise levels. The proposed change does not affect or involve any safety-related equipment.

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

Safety Evaluation No.:	93-027 Rev. 2
Implementation Document No.:	N/A
USAR Affected Pages:	9.1-25, 9.1-44
System:	FNR
Title of Change:	Fuel-Preparation Machine Full-Up-Stop Settings

Description of Change:

This safety evaluation addresses changing the position of the west fuel preparation machine (FPM) full-up-stops. This change will reduce the time/exposure spent during the transfer of new fuel to the spent fuel pool. Additionally, this will reduce the potential for personal contamination and plant contamination.

The west (2FNR*TL1B) FPM will be changed so that its normal configuration will be:

- Full-up-stops permanently removed
- Motive power removed (air-supply line disconnected and blocked)
- To be activated and used only with new nonirradiated fuel under administrative controls and then deactivated after completion of nonirradiated fuel handling

The west (2FNR*TL1B) FPM will have its full-up-stops removed such that a new fuel assembly loaded into its carriage will have its bail handle above the spent fuel pool water level. Positive stopping of the FPM carriage is performed by the end stops on roller chain mechanism. After the crane is disconnected from the new fuel assembly, which is sitting in the FPM, the assembly will be transferred by the refueling platform to its temporary storage location in the spent fuel storage rack.

Safety Evaluation Summary:

The function of the full-up-stops is to provide enough water shielding when using a FPM to handle irradiated fuel assemblies. When a FPM is used to transfer a nonirradiated fuel assembly into the spent fuel pool, a specific full-up limit is not required because its specific function (i.e., provide water shielding) is not required.

Safety Evaluation No.: **93-027 Rev. 2 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

Therefore, this safety evaluation is intended to allow the west (2FNR*TL1B) FPM to be configured to support the application appropriate for new fuel receipt/transfer activities and does not involve an unreviewed safety question.

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

Safety Evaluation No.: 93-037

Implementation Document No.: Simple Design Change SC2-0251-92

USAR Affected Pages: 7A.1-5

System: Reactor Building Heating and Ventilation (HVR), Containment Isolation (ISC), Main Steam (MSS), Residual Heat Removal (RHS), Reactor Protection (RPS), Standby Liquid Control (SLC), Service Water (SWP)

Title of Change: Replace P&B MDR Relays

Description of Change:

This simple design change replaced existing Potter and Brumfield (P&B) Model MDR relays that have been used as an isolation device to isolate nonsafety-related circuits from safety-related circuits, or to isolate redundant safety-related circuits.

Safety Evaluation Summary:

This change enhances the functionality of P&B MDR relays used as an isolation device in the systems listed above because the new P&B MDR relays are designed to preclude the failure modes of these relays.

Replacement relays will be qualified to the same requirements as the old relays. The new relays will be of the same form and fit such that they can replace the old relays as one for one replacement without requiring any major modifications during the installation.

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

Safety Evaluation No.:	93-044
Implementation Document No.:	Simple Design Change SC2-0102-91
USAR Affected Pages:	N/A
System:	Neutron Monitoring System (NMS)
Title of Change:	APRM Upscale Alarm and Rod Block (SDC SC2-0102-91)

Description of Change:

This change replaced the neutron flux input signal to the average power range monitor (APRM) upscale alarm and rod block circuit with the filtered simulated thermal power signal. The purpose was to filter out and reduce the noise levels of the neutron flux signal, which in turn allows Unit 2 operational entry into the Extended Load Line Limit Analysis (ELLLA) region of the power flow map along with a reduction in nuisance upscale alarms and rod withdrawal blocks.

Safety Evaluation Summary:

This modification allows operational entry into the ELLLA region of the power flow map which was prohibited by nuisance rod blocks. This modification 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.

Safety Evaluation No.: 93-055

Implementation Document No.: Simple Design Change SC2-0342-92

USAR Affected Pages: 9.5-4, 9A.3-46; Figure 9.5-1b

System: Fire Protection Water (FPW)

Title of Change: Install Curb Boxes for 2FPW-V1060 and 2FPW-V1061

Description of Change:

This change installed curb boxes (valve boxes) for two underground sectional isolation valves in the fire main.

Safety Evaluation Summary:

The subject valves were added during the construction of the site cafeteria building. During final construction activities, the valves were inadvertently covered prior to the installation of curb boxes as was intended. This change does not affect the piping and will allow for use of the two valves as key-operated, sectional isolation valves in accordance with 10CFR50 Appendix R, Section III.B. Normal construction activities involving excavation and fill are required for this installation. While this change will disrupt normal traffic flow in the area of installation, no impact to system or safe plant operation will result, and the ability to safely shut down the plant in the event of a fire is not impacted.

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

Safety Evaluation No.: 93-056 Rev. 1, 2 & 3

Implementation Document No.: Simple Design Change SC2-0328-92

USAR Affected Pages: Figures 1.2-1, 2.4-1

System: N/A

Title of Change: Construct a Spare Transformer Facility

Description of Change:

The spare transformer facility was constructed southwest of the Unit 2 345-kV switchyard. This facility will be used for the storage of the additional spare transformer for Unit 2.

Safety Evaluation Summary:

The construction of the spare transformer facility does not impact the pertinent licensing issues that are associated with hydrological engineering; i.e., flooding, local intense precipitation (probable maximum precipitation), and the impact on the air intake accident X/Q (Chi/Q), the atmospheric dispersion coefficient.

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

Safety Evaluation No.: 93-058 Rev. 2

Implementation Document No.: Calculation H21C-027

USAR Affected Pages: 9.1-39

System: FHS

Title of Change: Removal of Reactor Cavity Shield Plugs A, B, C and D at 40% or Less Reactor Power

Description of Change:

This safety evaluation evaluated the removal of reactor cavity shield plugs A, B, C and D at 40 percent or less reactor power.

Safety Evaluation Summary:

The removal of the reactor cavity shield plugs A, B, C and D at 40 percent or less reactor power does not affect the structural integrity of the shield plug barrier. The radiological effects of the proposed change have been calculated and determined to be negligible for radiological consequences to the refueling operators during normal refueling operations.

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

Safety Evaluation No.: 93-060

Implementation Document No.: Temporary Mod. 93-038

USAR Affected Pages: N/A

System: Reactor Building Ventilation (HVR)

Title of Change: Temporary Cooling for RWCU Pump Rooms

Description of Change:

This modification installed a temporary air conditioning unit outside the reactor water cleanup (RWCU) pump rooms to provide additional cooling to help alleviate high temperature conditions in the rooms. The air conditioning unit is powered from a welding receptacle fed from distribution panel 2WPS-PNL200.

Safety Evaluation Summary:

This modification does not affect any safety-related equipment, system, building or structure required to perform its safety function during normal operation or following a loss-of-coolant accident. An analysis of calculations indicates that a slight increase in the general area temperature is insignificant enough to cause any effect on the performance or the response time of a safety-related equipment or system to perform its intended function.

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

Safety Evaluation No.: 93-065

Implementation Document No.: Temporary Mod. 93-043

USAR Affected Pages: N/A

System: Ventilation Chilled Water (HVN)

Title of Change: Temporary Removal of 2HVN-TC17C

Description of Change:

The thermocouple well pipe connection for the thermocouple bulb from transmitter 2HVN-TC17C was leaking. The thermowell connection was temporarily removed and replaced with an isolation valve and pipe components until permanent replacement and maintenance was performed.

Safety Evaluation Summary:

This temporary modification will have no impact on the safe operation or capability to keep the plant in the safe shutdown condition.

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

Safety Evaluation No.: 93-075 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0014-93

USAR Affected Pages: 8.3-72; Figure 8.3-10

System: Safety-Related 125-V dc Battery System

Title of Change: Battery Charger Output Bifurcation

Description of Change:

This simple design change facilitates periodic testing of the Division I and II battery chargers, as required by Technical Specifications Section 4.8.2.1, with minimal impact to plant operations. This simple design change relocated the battery charger electrical connections to separate cubicles within their associated switchgear. This bifurcation was done utilizing the existing electrical power cabling between the battery chargers and the 125-V dc switchgear, and reterminating the cabling to individual cubicle load stabs within the switchgear. Only one charger was connected to the 125-V dc switchgear bus at a time. This was accomplished by using a breaker alternately between the breaker charger switchgear cubicles or installing a breaker in both of the battery charger switchgear breaker cubicles. In the event the 125-V dc switchgear breakers are installed for both chargers (of the same Division), one of the breakers shall be placed in A/C "Disconnect" position and locked out while the other breaker is closed. Although there will be an additional switchgear cubicle/breaker interface for battery charger connections to the switchgear, alarms and off-normal status displays will be maintained at those locations which currently provide such indications.

Safety Evaluation Summary:

This simple design change enhances the testability of the battery chargers by eliminating the need for lifting leads to perform the surveillance testing. This design 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.

Safety Evaluation No.: 93-081 Rev. 0, 1, 2 & 3

Implementation Document No.: Simple Design Change SC2-0022-93

USAR Affected Pages: 4.6-14; Figures 4.6-5c, 9.3-9a, 9.3-9b

System: Control Rod Drive (RDS), Reactor Building Equipment Drain (DER), Residual Heat (RHS), Reactor Building Ventilation (HVR), Reactor Core Isolation Cooling (ICS)

Title of Change: Elimination of Steam Emission from the Reactor Building Equipment Drains and RDS Scram Discharge Volume Collection Tank Installation

Description of Change:

This simple design change involved the following changes:

1. Isolated the hot pressurized drain lines from the cool gravity drains.

2. Added a pressure relief device in each of the Reactor Building drain loops to prevent overpressurization of the drain header in the event that the drain cooler inlet valves are inadvertently closed.

3. Rerouted the RDS scram discharge header vent line to the HVR system via 2DER-TK2B. This bypasses drain cooler 2DER-E2B and eliminates a possible blockage of the vent which would inhibit the scram discharge volume (SDV) drain flow.

4. Separated the RDS SDV drain line from the RHS and ICS pressurized steam-condensing header, and rerouted the drain to a new vented collection tank. SDV water entering the new collection tank post-scram is cooled by mixing with the existing water in the tank. The new tank then drains, via an overflow line, into the gravity drain header to the equipment drain tank, 2DER-TK2A.

The hot, pressurized drain lines within the "A" loop (i.e., ICS, RHS, and SDV drains) have been separated from the cool gravity drains, solving the ALARA concern. The "B" loop, hot pressurized drains from RHS and the main

Safety Evaluation No.: 93-081 Rev. 0, 1, 2 & 3 (cont'd.)

Description of Change: (cont'd.)

steam system were separated from the DER system, the RDS vent was rerouted, and the pressure relief devices were installed during Refuel Outage 4.

Safety Evaluation Summary:

All work associated with this change will be performed in the secondary containment elevations 175'-0" and 196'-0", in accordance with approved site Work Control and Radiation Protection procedures. The constructibility aspects of this change have been reviewed, and appropriate work sequencing instructions included within the applicable Work Orders. The use of construction aids, i.e., tank level tygon tube, pipe bladder, catch containments, flexible hose, etc., to facilitate installation of permanent piping have been reviewed and found adequate for system pressure retention and structural integrity. Temporary removal of pipe spools is required; replacement back to the original design, as required, will be controlled within the work order package. Temporary diversion of Reactor Building equipment drain effluent to the Reactor Building floor drain system has been approved and will be monitored by the Radwaste Department.

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

Safety Evaluation No.: 93-097

Implementation Document No.: Simple Design Change SC2-0078-93

USAR Affected Pages: Tables 9.3-1, 11.5-2 Sh 2; Figures 9.3-5c, 10.1-9e

System: Process Sampling

Title of Change: Deletion of Process Sample Points for URC Effluents

Description of Change:

This simple design change deleted process sample points for the condensate demineralizer system (CND) at the ultrasonic resin cleaner (URC) effluent, the URC resin effluent, and the URC resin receiver tank effluent. These process sample lines and associated valves are nonsafety related.

Safety Evaluation Summary:

Each of the above process sample points begins at a root valve in the CND system and proceeds via 1/4" tubing to sample sink SAS4. The 1/4" tubing was removed (prior to issuance of the plant Operating License) by modification PN2Y86MX044 in order to replace it with 1/2" tubing to alleviate plugging of the smaller diameter tubing. The modification was subsequently canceled and closed out before installing the 1/2" tubing.

The Standard Review Plan describes sample points for performance monitoring at the inlet and outlet of the condensate polishing system and sample points for radiological analysis of URC waste liquid effluent and by resin capacity analysis at panels located between the demineralizer and the URC process.

Sample points are provided for the common influent and common effluent of the CND system. In addition, Chemistry monitors URC performance by conductivity analysis at the resin mix and hold tank effluent and by resin capacity analysis at panels located between the demineralizers and the URC process.

The waste water from the URC is sent to the low conductivity waste tank, along with other liquid effluent from the CND system. The discharge from the low conductivity waste tank is provided with a sample point before being sent to either the anion regeneration tank or liquid radwaste. Therefore, a sample point exists for radiological analysis of common CND waste effluent, including URC waste water,

Safety Evaluation No.: 93-097 (cont'd.)

Safety Evaluation Summary: (cont'd.)

before discharge from the low conductivity waste tank. In addition, the sample root valves still remain so that temporary connections could be made to monitor the URC process.

Therefore, elimination of the URC effluent, URC resin effluent and the URC resin receiver tank effluent sample points does not violate any design bases or plant requirements.

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

Safety Evaluation No.: 93-113

Implementation Document No.: Unit 1 Simple Design Change SC1-0173-91

USAR Affected Pages: Table 2.3-4a

System: Weather Station

Title of Change: Replacement of 30′ Level Dewpoint Monitoring System at the Main Meteorological Tower

Description of Change:

This simple design change replaced the 30′ level dewpoint monitoring system at the main meteorological tower. A General Eastern Model 'E1' monitor with a 1211 HMP sensor and 175′ of interconnecting cable were procured for this change. The dewpoint temperature measurement is made with a direct-measuring sensor utilizing a Peltier-cooled mirror, automatically held at the dewpoint temperature by a photo-sensing, condensate-detecting, optical system incorporating a solid state LED light source and direct and bias photo detectors. The mirror temperature, if above freezing, measures the true dewpoint temperature and, if below, measures the frost point temperature. The temperature is measured by an embedded linear thermistor sensor.

Safety Evaluation Summary:

The frequency of the repairs on the old model dewpoint has caused the need to replace the model. The new model dewpoint system is as accurate as the old system and more reliable. The location of the new dewpoint sensor is independent of the 30-ft. level boom and was determined to be located on the southeast leg of the tower. This location was chosen because of existing bolt holes in the tower steel. The relocation will not affect the accuracy or validity of the data provided. The holes are located at the same level as the boom. Putting the dewpoint sensor at the same level as the boom instruments is required for consistency in instrument readings. Maintaining the surge protection factor is required to protect the new controller/monitor. Therefore, new surge protection boards were procured and will be installed in the monitoring system circuit.

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

Safety Evaluation No.: 93-129

Implementation Document No.: N/A

USAR Affected Pages: Figures 1.2-1, 2.4-1, 9A.3-1

System: N/A

Title of Change: Construction of the New Engineering Services Building

Description of Change:

The Engineering Services Building has been constructed outside the protected area, north of the P-Building where the R-Building and North Olympic Building stand.

The Engineering Services Building is a two-story, nonsafety-related structure with a slab on grade. This facility provides additional space requirements for departments relocated from the Salina Meadows facility. This building has a total area of approximately 45,000 square feet and provides office space for about 250 personnel.

Safety Evaluation Summary:

Based on the evaluation performed, it is concluded that construction of the Engineering Services Building does not involve an unreviewed safety question.

Safety Evaluation No.: 94-001

Implementation Document No.: Simple Design Change SC2-0255-91

USAR Affected Pages: Figure 9.2-5e

System: Makeup Water Treating System (WTS)

Title of Change: Ecolochem Filtered and Purge Water Connections

Description of Change:

To continue the use of the Ecolochem portable demineralized trailer, permanent filtered and purge water connections were added to the existing WTS system piping. Temporary Modification 91-093 was employed providing a connection for the purge water from the Ecolochem to the makeup waste neutralizing tank (2WTS-TK1). This change made the connection permanent as installed. In addition, a new connection was installed from the water treating filter drain line, 2-WTS-002-134-4, to supply the Ecolochem trailer. Makeup water from the Ecolochem demineralized trailer is controlled in accordance with procedure N2-OP-15.

Safety Evaluation Summary:

An engineering review of the change found that installing additional connections to facilitate the Ecolochem demineralized water process will improve the system performance without causing any safety or operability issues.

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

Safety Evaluation No.: 94-006

Implementation Document No.: DER 2-91Q-1718

USAR Affected Pages: 11.4-1 through 11.4-6; Table 11.4-4 Sh 1 & 2

System: Solid Radwaste

Title of Change: Abandonment In-Place of Asphalt Solidification Equipment

Description of Change:

This change abandoned in-place selected portions of the original asphalt-based solid radwaste processing system.

Safety Evaluation Summary:

The original plant design for radwaste solidification (i.e., removal of free water from miscellaneous wet wastes) utilized the Werner & Pfleiderer (WasteChem) asphalt volume reduction system addressed by Topical Reports WPC-VRS-001 and WPC-VRS-002. Due to various deficiencies, process problems, and offsite disposal facility burial criteria associated with the use of this system, the original asphalt-based solidification system was "abandoned in-place." The abandonment in-place of the asphalt-based solidification system will have minimal impact on radwaste processing, since a radwaste dewatering process providing an acceptable method of volume reduction utilizing methodology and equipment addressed in Chem Nuclear Systems, Inc., Topical Report RDS-25506-01-P/NP (reviewed and approved by the NRC) will be utilized. Abandonment in-place was accomplished in such a manner to assure proper pressure boundary confinement of all process applications.

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

Safety Evaluation No.: 94-009 Rev. 0, 1, 2 & 3

Implementation Document No.: Simple Design Change SC2-0015-94

USAR Affected Pages: 10.4-33, 10.4-34; Figure 10.1-6c

System: Feedwater (FWP)

Title of Change: Install Throttle Valves in Feed Pumps Seal Water Injection Lines

Description of Change:

This change installed throttle valves in feed pumps seal water injection lines. The addition of the throttle valves allows each seal water inboard and outboard injection line to be equally balanced, providing greater reliability of the feed pump seals.

Safety Evaluation Summary:

An engineering review of this change has been performed. This review, which included the effects of the change on the system's operability, reliability, maintainability, structural integrity, and system interactions, has found that the implementation of this change will enhance system reliability/maintainability without causing any significant safety or operability issues.

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

Safety Evaluation No.: 94-012

Implementation Document No.: N/A

USAR Affected Pages: Table 9A.3-15 Sh 3; Figures 1.2-1, 2.4-1, 9.5-1b, 9A.3-1

System: N/A

Title of Change: New Unit 2 Maintenance Building

Description of Change:

The Unit 2 Maintenance Building has been constructed inside the protected area, south of the Unit 2 Access Control Building and north of the new Operations Building. This building consolidates maintenance facilities into a new single structure which is located closer to existing plant accessways, enhancing the Maintenance Department's overall efficiency.

The building is a two-story, nonsafety-related structure with approximately 42,000 square feet of floor area. The structure has a slab on grade and provides shop areas for Electrical, Mechanical, and Instrumentation and Controls Maintenance Groups. Additional areas for locker rooms, material issue, and office spaces for Maintenance Management and Support personnel are provided. Also, a portion of the building provides high bay vehicular access equipped with overhead cranes. The new Maintenance Building and Access Control Building are connected, and an elevated walkway between the Maintenance Building and the Operations Building has been constructed.

Safety Evaluation Summary:

The pertinent safety issues identified in this Safety Evaluation are flooding and the impact on the Control Room fresh air intake radiological atmospheric dispersion coefficient. The Maintenance Building location provides adequate separation from safety-related systems and structures to preclude any adverse impact from any compressed gases or chemicals stored in the building.

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

Safety Evaluation No.:	94-013
Implementation Document No.:	Simple Design Change SC2-0035-94
USAR Affected Pages:	Figures 10.1-9a, 10.1-9b, 10.1-9c, 10.4-9 Sh 7, 8, 9
System:	Condensate Demineralizer (CND)
Title of Change:	Condensate Demineralizer Flow Recorders Upgrade

Description of Change:

This change replaced five condensate demineralizer flow recorders and the resin strainer differential pressure meters associated with each condensate demineralizer with new recorders that are designed for improved reliability. This change also replaced a sixth recorder which monitors the total differential pressure across all of the condensate demineralizers, and an additional meter which monitors resin recycle strainer differential pressure with a new recorder that performs these combined functions.

Safety Evaluation Summary:

Upon implementation of this simple design change, new recorders will have been installed that provide improved reliability of monitoring of flow through the condensate demineralizers as well as adequate monitoring of the strainers differential pressure. The CND 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 change does not involve an unreviewed safety question.

Safety Evaluation No.:	94-017
Implementation Document No.:	Simple Design Change SC2-0004-91
USAR Affected Pages:	Tables 6.2-56 Sh 7, 9A.3-15 Sh 5; Figures 7.3-10 Sh 1, 9.5-1g
System:	Fire Protection Monitoring (FPM), Fire Protection Water (FPW)
Title of Change:	Remove Abandoned FPW Pipe Systems from FPM Monitoring

Description of Change:

Two abandoned water deluge piping systems originally designed to suppress fires at the reactor recirculation pumps were removed from the FPM system. This change removed nuisance spurious alarms, trouble signals, horns, annunciations, and computer inputs from two piping systems which were never functional and not required. This change also disconnected cabling to spared devices in the plant, removed fuses and relays in the local fire control panel, and included the removal of deactivated switches and indication lights in the Main Control Room.

Safety Evaluation Summary:

Since the two affected water deluge piping systems have been inactive and capped prior to plant operation, the associated components perform no useful function. The primary containment does not require fire protection systems during normal operation since it is inerted.

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

Safety Evaluation No.: 94-018

Implementation Document No.: Temporary Mod. 94-020

USAR Affected Pages: N/A

System: High-Pressure Core Spray (CSH)

Title of Change: Jumper Control Signal for 2CSH*MOV118

Description of Change:

A temporary jumper was installed in the control circuit of the high-pressure core spray (HPCS) suppression pool suction valve, 2CSH*MOV118, to simulate a closed valve signal from HPCS test return valve 2CSH*MOV112. This provided a permissive signal for 2CSH*MOV118 to open even though valve 2CSH*MOV112 was deenergized and/or being stroked open (not closed). With 2CSH*MOV118 capable of opening, the HPCS was capable of transferring water from the suppression pool to the reactor vessel and met the requirements of Technical Specification 3/4.5.1.c. The HPCS was declared operable without 2CSH*MOV112 functioning, which allowed it to receive maintenance and be VOTES tested prior to the refueling outage.

Safety Evaluation Summary:

The HPCS system can be considered operable since this jumper installation will allow it to perform its designed functions without any impact from 2CSH*MOV112 on the system's flow rates, pressures, response times, flow paths, or setpoints. The jumper will not affect any other components or systems. The repairs and testing of 2CSH*MOV112 can be performed safely prior to the refueling outage.

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

Safety Evaluation No.: 94-020

Implementation Document No.: Simple Design Change SC2-0405-91

USAR Affected Pages: Figure 10.1-5b

System: Condensate (CNM)

Title of Change: Condensate Booster Pump Mechanical Seal Cavity Drains

Description of Change:

This simple design change added drain lines to the existing inboard and outboard mechanical seal cavity connections. In addition, the existing drain from the skid was removed and the connection capped.

Safety Evaluation Summary:

All drain lines, whether new or existing, are nonsafety related and will not impact the safe operation of the plant. The change does not affect the operation of the CNM system, Turbine Building equipment and floor drain systems (DET, DFT), nor does it affect the safe shutdown of the plant. Both systems are designed to handle influent from oily or nonoily waste from radioactive and potentially radioactive sources. Both systems pump waste from their respective collection tanks or sumps to radwaste for processing. The condensate pumps will continue to function as designed because this change involves routing water that may pass through the mechanical seals to the DFT system without impacting pump performance characteristics. The existing skid drain lines 2-CNM-150-330-4, 2CNM-150-331-4 and 2CNM-150-332-4 will be removed and a short nipple and cap will be installed. Any water or oil that may collect on the skid may be drained through the capped connection or wiped away.

Should a water leak develop around the condensate booster pumps and said flow was sufficient to overflow the pump skid containment, the DFT system would collect the added volume.

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

Safety Evaluation No.: 94-021 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0031-94

USAR Affected Pages: Table 3.9A-12 Sh 12; Figure 9.2-1f

System: Service Water (SWP)

Title of Change: IST-SWP Check Valve Internals Removal

Description of Change:

This simple design change removed the internals from check valves 2SWP*V800A, B and V802A, B. Removal of the internals will preclude sedimentation within the value and preclude test failures during in-service testing.

Safety Evaluation Summary:

This change will have no impact on the safe operation or capability to keep the plant in a safe shutdown condition.

Deletion of the check valve internals will not prevent the SWP system from performing its intended safety function, nor will the system pressure integrity be degraded during any mode of system or plant operation.

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

Safety Evaluation No.:	94-023
Implementation Document No.:	Simple Design Change SC2-0174-93
USAR Affected Pages:	Table 7.6-6
System:	Neutron Monitoring System (NMS)
Title of Change:	Revise APRM Flow-Biased Rod Block Setpoint SDC SC2-0174-93

Description of Change:

This design change revised the average power range monitor (APRM) flow-biased simulated thermal power (STP) scram setpoint from 0.66(W-ΔW) + 51% to 0.58(W-ΔW) + 59% (the APRM flow-biased STP upscale scram setpoint was analyzed under Technical Specification Amendment No. 51), and the APRM flow-biased rod block setpoint from 0.66(W-ΔW) + 42% to 0.58(W-ΔW) + 50%. This change allows Unit 2 to better utilize the extended load line limit analysis (ELLLA) region of the power/flow map.

Safety Evaluation Summary:

This change allows Operations to enter the ELLLA region of the power/flow map. Operation in the ELLLA region is restricted because at lower flows the APRM flow-biased scram and the APRM flow-biased rod block encroach on the ELLLA region. 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.

Safety Evaluation No.: 94-024 Rev. 1

Implementation Document No.: Simple Design Change SC2-0361-91

USAR Affected Pages: Tables 1.8-1 Sh 52, 7.5-2 Sh 1 & 8

System: Regulatory Guide 1.97 Monitoring and Display Instrumentation

Title of Change: Identification/Marking of Regulatory Guide 1.97 Display Instrumentation on Panels in the Control Room

Description of Change:

This change added to the panels in the Main Control Room a red plastic trim plate around the component identification label of the Regulatory Guide (RG) 1.97 Category 1 and Category 2 display devices for Type A, B, and C variables.

This change will assist the Control Room operators and supervisors in quickly locating the most important RG 1.97 display instruments (i.e., those expected to be the most useful for monitoring, assessing, and responding to postaccident conditions).

This change implements and conforms to a recommendation specified in RG 1.97 with the following exceptions: (1) the position indicating lights for the primary containment isolation valves were not marked with the red trim plate, and (2) the method used to identify the RG 1.97 display devices is the same as that used to identify several other important system control switches and display instruments.

Safety Evaluation Summary:

This change does not modify in any way the operation or performance of any plant systems or structures, nor does it require that any changes be made to any instructions currently specified in any plant operating, maintenance, or calibration procedures. This change does not require changing the currently specified safety classification or qualification criteria of any system component, and has no adverse impact on the safe operation or shutdown of the plant.

Also, the structural integrity of the reactor coolant system pressure boundary, the primary containment pressure boundary, and the secondary containment pressure boundary is in no way affected by the installation of the proposed change.

Safety Evaluation No.: 94-024 Rev. 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

The two noted deviations from full conformance with the subject recommendations of RG 1.97 have each been evaluated, and both have been determined to be acceptable on a plant-specific basis.

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

Safety Evaluation No.: 94-026

Implementation Document No.: Simple Design Change SC2-0134-93

USAR Affected Pages: Figure 9.2-8b

System: Domestic Water (DWS)

Title of Change: Domestic Hot Water Recirculation Pump Abandonment

Description of Change:

This change abandoned in place domestic water system recirculation pump 2DWS-P1 and associated motor and electrical equipment. The associated annunciator in the control room was also removed. Continuing problems concerning pump leakage and motor overloading were resulting in annunciator indication in the control room. Pump abandonment included necessary changes to associated equipment.

Safety Evaluation Summary:

The DWS system is not safety related, is not connected to any potentially radioactive process systems, and is nonseismic except in the Control Building, where appropriate design measures have been implemented.

System water pressure is provided by the normal source of domestic water, Oswego City Water, and is not affected by the recirculation pump. By abandoning the pump/motor and associated equipment in-place, an unnecessary annunciator will be removed and the need for pump repair/maintenance, which has proven to be quite extensive in the past, will be eliminated while not adversely affecting system operation.

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

Safety Evaluation No.: 94-027

Implementation Document No.: DER 2-94-0157

USAR Affected Pages: 9.3-20

System: Turbine Plant Sampling (SST), Reactor Plant Sampling (SSR)

Title of Change: Tolerance Change for Isobath Temperature in Sample Panels

Description of Change:

The temperature of the constant isothermal baths is maintained at 77°F ± 1°. The tight tolerance for the temperature requires constant change in refrigeration mode and this results in excessive wear and tear on the refrigeration units. This change provided for a wider temperature range (77°F ± 5°) to be maintained at the sample sink constant baths, thereby reducing the constant switching of refrigeration modes and the wear and tear on the units.

Safety Evaluation Summary:

The proposed change would reduce the wear and tear on the refrigeration units by expanding the tolerance of the allowable constant bath temperature and not significantly affect the accuracy of the conductivity measuring instrumentation.

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

Safety Evaluation No.: 94-028

Implementation Document No.: Simple Design Change SC2-0020-94

USAR Affected Pages: 9.5-84; Figures 9.5-52a, 9.5-52c

System: Auxiliary Boiler Systems (ABD, ABF, and ABH)

Title of Change: Addition of Auxiliary Boiler Chemical Injection Piping and Boiler Feed and Blowdown Sample Connections

Description of Change:

This change provided a means of adding sodium sulfite directly to the auxiliary boilers when the boilers are in a hot standby condition. In addition, water chemistry sample connections were added to facilitate boiler feedwater and blowdown analysis. All sample piping has been routed to a new sample sink for convenience. Restricting orifice 2ABF-RO128 bore dimension has been decreased to eliminate excessive steam loss from the auxiliary boiler deaerator.

Safety Evaluation Summary:

This change upgrades the auxiliary boiler system to improve system reliability and its capability to support plant operations. The auxiliary boiler system and the impacted boiler subsystems are classified as nonsafety related. These changes will have no impact on the safe operation or shutdown of the plant since the hardware changes have been designed in accordance with the original plant design basis, and have no effect on the functional capability of the auxiliary boiler systems.

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

Safety Evaluation No.: 94-029

Implementation Document No.: Temporary Mod. 94-022

USAR Affected Pages: N/A

System: Makeup Water Storage (MWS), Chilled Water Ventilation (HVN)

Title of Change: Temporary Makeup Water to the HVN System

Description of Change:

This temporary change provided an alternate source of makeup water to the chilled water ventilation system. The new makeup water is from the MWS system in lieu of the water treatment (WTS) system. The WTS system is experiencing a reduction of flow due to piping degradation and is not able to supply the required demand. MWS water will be routed from/to existing connections via temporary hose and associated components.

Safety Evaluation Summary:

The alternate makeup from the MWS system will be sufficient through a hose of equal size as a minimum. The new source of makeup water is demineralized water in lieu of filtered water, water quality is enhanced, and supply will be adequate to meet demand. All hoses and associated components shall be rated for their intended service conditions and will be adequately secured. The 60 gph of water from the MWS system will not affect the makeup water system capacity to feed water to its originally intended systems. The use of MWS water in lieu of the existing WTS water will not cause any adverse safety or operability issues.

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

Safety Evaluation No.: 94-032

Implementation Document No.: NUREG-0123

USAR Affected Pages: 9A.3-31, 9A.3-53, 9A.3-56, 9A.3-58

System: N/A

Title of Change: Changes to the UFSAR/USAR Actions Required for Inoperable Fire Protection Systems

Description of Change:

This change modified the UFSAR/USAR action statements for inoperable fire barriers, water-based extinguishing systems, Halon systems and carbon dioxide systems. In addition, the definition of fire watch patrol was changed in the Unit 1 UFSAR to reflect the action statement changes.

Safety Evaluation Summary:

The safety evaluation analyzes the current action statements and augments the options for compensatory measures with additional options to account for areas where fire detection systems are installed and operable. Further, the expanded use of engineering evaluation for impairments, which is currently recognized within the Unit 2 USAR, is expanded for application within the Unit 1 and Unit 2 action statements. Such impairment provisions allow greater flexibility in dealing with system impairments without adversely affecting the Fire Protection Program. The existing action statement options also remain as potential compensatory measures.

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

Safety Evaluation No.: 94-033

Implementation Document No.: Simple Design Change SC2-0164-93

USAR Affected Pages: 9C.8-5; Appendix 9C Tables 3-1, 3-4, 4-1 Sh 2; Appendix 9C Figure 5-1

System: Main Steam (MSS)

Title of Change: Replace SRV Crane 2MHR-CRN66

Description of Change:

The following changes were implemented by this simple design change:

1. Retired crane 2MHR-CRN66 and provided a replacement crane. This replacement crane is an electrical trolley and chain hoist and is designated as crane 2MHR-CRN66X.

2. Reworked and repaired electrical trolley and bus-bar for replacement crane 2MHR-CRN66X.

3. Provided an additional weld (nonstructural) for SRV crane 2MHR-CRN65X monorail splice at azimuth 240° to improve crane trolley performance.

Safety Evaluation Summary:

Replacement crane 2MHR-CRN66X is being supplied nonseismic and will be removed from the primary containment during plant operations to meet commitments made under the Guidelines for the Control of Heavy Loads (NUREG-0612) and USAR Appendix 9C at Unit 2. The load path has not changed and has been previously evaluated such that the failure of the crane during a seismic event will not affect plant safety. Replacement crane 2MHR-CRN66X is considered and included in the Control of Heavy Loads Analysis. The replacement crane and installation conditions meet requirements for seismic evaluation of nonsafety-related components in safety-related areas (inside primary containment) and does not affect the safety and reliability of Unit 2.

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

Safety Evaluation No.:	94-034 Rev. 1
Implementation Document No.:	N/A
USAR Affected Pages:	8.2-1, 8.2-28; Figures 8.1-1, 8.2-1, 8.2-1a, 8.2-1b, 8.2-9
System:	345-kV Transmission Output, 115-kV Offsite Power Sources
Title of Change:	Independence/Scriba 345-kV Transmission Line

Description of Change:

This change added a sixth 345-kV transmission line to Scriba Station through two new 345-kV circuit breakers. The two new 345-kV circuit breakers are the same electrical rating as the other eight 345-kV circuit breakers. Construction work included the electrical interconnection of one of the two 345-kV circuit breakers in the Spring of 1994 while Unit 1 and Unit 2 were running. In addition to the energization of this breaker, relay testing was also performed. The electrical interconnection of the second breaker and associated relay testing took place during the Unit 2 refuel outage in the Spring of 1995.

Safety Evaluation Summary:

This safety evaluation addresses the impacts on Unit 1 and Unit 2 resulting from Scriba Substation construction activities. It also analyzes the effect on the transmission system due to increased generation.

Worst-case scenarios were identified and found to be bounded by previous accidents and transients analyzed in both the Unit 1 and Unit 2 UFSARs. A Probabilistic Risk Assessment was performed to quantify the risks associated with the line outages, construction activities and operation of the new transmission lines. The results show the relative change in core damage frequency is small and is considered acceptable.

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

Safety Evaluation No.: 94-035 Rev. 2

Implementation Document No.: N/A

USAR Affected Pages: 8.1-3, 8.2-2, 8.2-7, 8.2-24; Figures 8.2-1, 8.2-1b, 8.2-4a, 8.2-6d through 8.2-6u

System: 115-kV Offsite Power Source

Title of Change: Alternate 115-kV Transmission Supply NMP2

Description of Change:

This modification allows the Unit 2 115-kV transmission line, No. 5 or No. 6, to be energized from the 115-kV transmission system instead of Scriba Substation. Either 115-kV transmission line No. 5 or line No. 6 would be energized from NMPC's 115-kV transmission system's line No. 2. Either transmission line will be connected to the Scriba Station 115-kV main bus (C for 5 line, D for 6 line) and will provide 115-kV offsite supply through existing 115-kV feeder breakers R50 or R60. No changes to protective trip schemes at Unit 2 would be required. Since existing 115-kV circuit breakers would still be energized, relay protective trip signals at Unit 2 will be functional.

Safety Evaluation Summary:

This safety evaluation addresses the impacts on Unit 1 and Unit 2 resulting from providing an electrical offsite power supply to Unit 2 from the 115-kV transmission system. It also analyzes the effect on the transmission system due to the increase in electrical load.

Worst-case scenarios were identified and found to be bounded by previous accidents and transients analyzed in both the Unit 1 and Unit 2 USARs. The analysis performed shows that the 115-kV transmission line No. 2 can be used as an alternate supply to the 115-kV No. 5 or No. 6 line under worst-case loading conditions as long as certain administrative controls are maintained.

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

Safety Evaluation No.: 94-036

Implementation Document No.: Simple Design Change SC2-0040-94

USAR Affected Pages: Table 10.2-1 Sh 2; Figure 10.2-3 Sh 1

System: Electro-Hydraulic Control (EHC)

Title of Change: Keylock Switch Addition to the Turbine Backup Overspeed Test Circuit

Description of Change:

This change added a keylock switch to the turbine backup overspeed test circuit. Redundant switch contacts were necessary to prevent the original potentially faulty test push button from tripping the turbine during normal testing. The new switch, in the test position, disables the trip relay and serves as a permissive for the test. Initiation of the test continues to be controlled by the push button only after the new switch is placed in the test position. In the normal position the new switch has no impact on the backup overspeed trip circuit.

Safety Evaluation Summary:

The turbine generator, designed to minimize the possibility of a failure that could produce high-energy missiles, is not required to trip for nuclear steam supply upsets but does so to protect itself from conditions that may cause damage. The turbine generator 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 change does not involve an unreviewed safety question.

Safety Evaluation No.: 94-039

Implementation Document No.: Simple Design Change SC2-0113-94

USAR Affected Pages: Figure 10.1-9c

System: Condensate Demineralizer (CND)

Title of Change: Condensate Demineralizer System Improvements: Replacement of Valve 2CND-PV188 and Removal of 2CND-RV278

Description of Change:

This simple design change replaced valve 2CND-PV188, a 1-1/2", 300# flanged Tufline plug valve, actuator and positioner with a Fisher Controls 2", 300# flanged globe control valve with actuator and positioner. The Tufline plug valve was not adequate for pressure control and controlled erratically. Valve 2CND-RV278 was removed from the system. The valve leaked, adversely impacting system performance. Valve RV278 is redundant and system overpressurization was provided by 2CND-RV352. In addition, pressure indicators 2CND-PI282, PI303 and PI304 were replaced with a larger scale gauge.

Safety Evaluation Summary:

This change does not affect any system, equipment or component of the plant which performs a safety-related function. Nuclear safety will not be affected as the change impacts the nonsafety-related CND system.

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

Safety Evaluation No.: 94-043 Rev. 2

Implementation Document No.: NEP-POL-0101

USAR Affected Pages: 13.1-4, 13.1-5; Figure 13.1-3

System: N/A

Title of Change: Engineering Technical Support Organization Changes

Description of Change:

The following changes were made in the Unit 2 Engineering organization:

1. General Supervisor Nuclear Design - position abolished.

2. Site Engineering - name changed to Plant Support.

3. Supervisor Safety Analysis - name changed to Supervisor Analysis.

4. Supervisor Chemistry/RP Support - position abolished; RP Support function moved under Supervisor Analysis (see 3 above) and the chemistry function moved to Supervisor Environmental Protection.

5. Lead Engineer Inspection Program - position abolished and the function integrated into Mechanical Design.

6. Lead Engineer Special Programs - position abolished and function integrated under Supervisor Analysis (see 3 above).

7. General Supervisor Engineering Performance Services - function integrated under Manager Unit 2 Engineering.

8. Supervisor Engineering Performance - cost estimating and scheduling functions integrated under Supervisor Project Management Unit 2.

9. Supervisor Administrative Services - position abolished; each Engineering Supervisor will oversee their own administrative staff.

10. Associate Senior Staff Tech Building Services - position upgraded and moved under Manager Information Management as "Supervisor Building Services."

Safety Evaluation No.: 94-043 Rev. 2 (cont'd.)

Description of Change: (cont'd.)

11. Program Director Independent Safety Engineering Group - word "program" deleted from the position title. The function remains unchanged.

12. Supervisor Document Control (site), Supervisor Document Control (Salina), Supervisor Records Management and Supervisor Resource Centers - these positions have been abolished and their functions have been transferred into a new position, "Supervisor Document Control/ Records Management."

13. Supervisor Software Development - this is a new position reporting to Manager Information Management.

Safety Evaluation Summary:

After implementation of these changes, adequate resources will exist to provide Engineering support for safe operation and maintenance of the facility under both normal and off-normal conditions. Consequently, the safe operation and maintenance of the facility is not adversely affected.

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

Safety Evaluation No.: 94-044

Implementation Document No.: DER 2-94-0036

USAR Affected Pages: N/A

System: N/A

Title of Change: Boraflex Coupon Removal with NETCO Procedure SEP-093-01

Description of Change:

A visual inspection was made and measurements taken of the full-length surveillance assembly (SA) at Unit 2. The boraflex sheets, or coupons, from the short-length SA were removed and sent to a qualified laboratory (Penn State) for testing and analysis. This analysis was used to establish a baseline (the "original" data described in the USAR) to compare future coupon tests against. The coupons will also be compared to unirradiated coupons taken from the original lot of Boraflex used to manufacture Unit 2's SAs.

Safety Evaluation Summary:

Performing a baseline characterization of the Boraflex coupons installed in the spent fuel racks is necessary to develop a Boraflex Poison Surveillance Program to track boron depletion. The lack of preinstallation baseline characterization will not have a significant impact on the development of a long-term surveillance program and will not pose a safety concern.

Standard industry practice and statistical studies show that removing all of the short-length coupons from the spent fuel racks, for the time period required to take this baseline data, will not have a significant impact on the neutron and gamma exposures seen by these coupons, and future coupon surveillance will be accurate.

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

Safety Evaluation No.: 94-045

Implementation Document No.: N/A

USAR Affected Pages: 3.1-24, 6.2-55, 6.2-99, 6.3-20, 6.4-6, 7.3-26, 7.3-34, 8.3-2, 8.3-48, 8.3-75, 9.1-18, 9.2-6, 9.2-16, 9.3-8, 9.3-11, 9.3-16, 9.3-29, 9.4-8, 9.4-24, 9.4-48, 9.4-54, 9.4-64, 9.5-32, 9.5-49, 9.5-61, 9.5-73, 9.5-81, 11.5-13; FMEA Volumes 1 and 2

System: N/A

Title of Change: Removal of the Failure Modes and Effects Analysis (FMEA), Book 1 and 2, from the USAR

Description of Change:

The FMEA was originally submitted to the NRC in 1983 as part of the Operating License application, and it documented the single-failure analyses for safety-related systems at that time. The FMEA is a very detailed, component-level, computer-based fault tree analysis.

The two FMEA volumes have been removed from the USAR and are retained as a separate engineering document which is referenced in the USAR.

Safety Evaluation Summary:

This is a documentation-only change which has no effect on the plant, its systems or procedures, and does not affect the safe operation or shutdown of the plant, nor does it affect the requirement to consider single-failure criterion as a normal part of the design process. Removing the FMEA from the USAR eliminates the requirement to update this document annually with the USAR.

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

Safety Evaluation No.: 94-046 Rev. 1

Implementation Document No.: Simple Design Change SC2-0099-93

USAR Affected Pages: 10.2-4, 10.2-5; Figure 1.2-40

System: Generator Hydrogen Supply (GMH)

Title of Change: Bulk Hydrogen Control Cabinet

Description of Change:

This change replaced all piping, valves, and controls associated with the existing bulk hydrogen storage unit. Changes included the replacement of all cylinder isolation valves, fabrication of a new stainless steel discharge manifold, installation of a new tube trailer discharge station, installation of a vendor-supplied (Air Products) standard pressure control station, and replacement of the excess flow check valve with a properly-sized unit. In addition, the discharge height for safety relief vents was increased.

Safety Evaluation Summary:

This change was made to address leakage and safety concerns with the previous piping arrangement, and to modify the system to provide adequate makeup flow rate for generator replenishment without defeating the protective features of the excess flow check valve. The design flow rate of the excess flow check valve was not changed.

The main generator hydrogen supply system is a nonsafety-related system that is used to provide hydrogen to the main generator after an outage or on an as-needed basis to make up for hydrogen loss from the generator. The system consists of a vendor-supplied bulk hydrogen storage unit with pressure-reducing controls located in the yard area between Unit 2 and Unit 1, and a network of distribution piping and controls which convey the hydrogen into the turbine building where it is used for generator makeup. The system's purpose, function, method of performing its function, and design basis was not changed.

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

Safety Evaluation No.: 94-048

Implementation Document No.: Simple Design Change SC2-0062-92

USAR Affected Pages: 7.7-33

System: Plant Process Computer System (PMS)

Title of Change: Remove Balance of Plant Performance Calculations (BOPCALC) and Vessel Temperature Rate of Change (VTC) Software from the Plant Process Computer System (PMS)

Description of Change:

This change disabled the current BOPCALC and VTC functions by removing the associated software programs from the PMS computer.

Safety Evaluation Summary:

Removing the BOPCALC and VTC software will have no impact on the safe operation or shutdown of the plant. The PMS computer will remain as a system to provide operators with the means to monitor nuclear steam supply system (NSSS) and BOP events.

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

Safety Evaluation No.: 94-049 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y89MX146

USAR Affected Pages: 6.2-66 through 6.2-71; Table 6.2-54; Figures 6.2-77, 6.2-95A through 6.2-95D

System: Secondary Containment and Standby Gas Treatment (SGTS)

Title of Change: 1-Hour Drawdown Analysis

Description of Change:

This safety evaluation supports plant operation for 1-hour drawdown time. The following parameters have been changed for the reduced ΔT requirements as compared with the fourth operating cycle. These parameters are discussed below:

1. Spent Fuel Heat Loads: A design basis spent fuel pool heat load (16 batches of fuel with 12 days of cooling and power uprate) is used for the ΔT requirements. The use of the design basis heat load is conservative.

2. Unit Cooler Performance: Forty percent degradation for all unit coolers is assumed. Based on the performance tests performed during the 1992-93 time period, the overall degradation of all unit coolers including 2HVR*UC413A & B is 13%. This provides sufficient margin to account for any further degradation that may occur in the future.

3. Secondary Containment In-leakage Rate: The maximum allowable secondary containment in-leakage is 2,670 cfm to support 1-hour drawdown time and reduced ΔT requirement. This is 17% higher than the fourth operating cycle but still meets the SRP Section 6.2.3 guideline.

4. Decay Heat Removal Flow Reduction: The 2,670 cfm in-leakage selection will be made such that it will permit a flow diversion up to 300 cfm for the decay heat cooling after 5 hours into an accident. The 300 cfm flow division is more than the cooling flow requirement of 145 cfm.

5. Elimination of ΔT Annunciation: Existing four-hour surveillance program is sufficient to ensure that the ΔT requirement will be met. The ΔT annunciation is no longer required because of significantly lower ΔT requirement, and periodic surveillance is adequate. Therefore, the ΔT annunciation can be eliminated.

Safety Evaluation No.: 94-049 Rev. 0 & 1 (cont'd.)

Description of Change: (cont'd.)

6. Surveillance Acceptance Criteria: The secondary containment and SGTS system surveillance acceptance criteria are revised to reflect 2,670 cfm in-leakage rate. The analysis methods are the same as those used previously.

7. Use of Electric Heaters: Up to 45 kW of electric heaters can be used to maintain proper emergency core cooling system room temperature (high-pressure core spray room excluded). This change does not adversely affect 60 minutes drawdown capacity. Use of additional heaters may be allowed, following Engineering evaluation and with Applicability Review.

NOTE: Revision 1 to the Safety Evaluation evaluated the use of the electric heaters as described in item 7. The electric heaters were prohibited from use in Revision 0.

Safety Evaluation Summary:

The drawdown analyses (Calc. ES-271, Rev. 0 and ES-259, Rev. 02) provide a curve that defines ΔT requirements based on 1-hour drawdown time for the remainder of the plant life. The ΔT requirement varies from 5 to 10°F during summer months. Because of low ΔT requirement, building heating is not anticipated. During winter months, the available ΔT will be more than the maximum ΔT requirement of 20°F.

The safety evaluation concludes that no safety concerns are involved and no unreviewed safety questions exist if the ΔT requirements of Figure 1 and other requirements as stipulated in the safety evaluation are adhered to.

Safety Evaluation No.:	94-050
Implementation Document No.:	EDC 2E10933
USAR Affected Pages:	Figures 5.4-13d, 5.4-13g
System:	Residual Heat System (RHS)
Title of Change:	Revise Safety Class of Control Components for RHS Steam Condensing Pressure Reducing Valves 2RHS*PV21A/B from SR to Q

Description of Change:

This change revised the safety class from SR (Safety Related) to Q (Quality) for components of the RHS steam-condensing pressure-reducing valve instrument loops which perform no safety function. The safety classification was changed for the pressure indicating controller, pressure indicator and the current to pneumatic converter for 2RHS*PV21B and the current to pneumatic converter for 2RHS*PV21A.

Safety Evaluation Summary:

Changing the safety classification, from SR to Q, of components which perform no safety function, 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.

Safety Evaluation No.:	94-053 Rev. 1
Implementation Document No.:	Simple Design Change SC2-0283-91
USAR Affected Pages:	Figure 7.4-1 Sh 1
System:	ICS - Reactor Core Isolation Cooling (RCIC)
Title of Change:	Add Time Delay in RCIC Initiated Turbine Trip

Description of Change:

The RCIC system at Unit 2 is designed in such a way that RCIC initiation provides automatic signal to trip the main turbine instantaneously, regardless of the cause of RCIC actuation. Therefore, any inadvertent RCIC actuation due to human error or equipment malfunction will cause an unnecessary trip of the main turbine. If at that moment the reactor is running at 35% power or higher, the reactor scram will follow.

To resolve this discrepancy, a time delay was added to the turbine trip signal initiated by RCIC. This change allows the operator to verify the cause of starting RCIC prior to the turbine trip and take appropriate actions.

To provide this time delay, the nonsafety-related auxiliary Agastat relay was replaced with a nonsafety-related time delay Agastat relay.

Safety Evaluation Summary:

General Electric (GE) report GE-NE-E51-00171-01, dated June 1994, and GE letters, dated 8/29/94, 8/30/94, and 9/13/94, provided requested analysis of the proposed change based on calculated moisture level in the steam and steam piping configuration.

The GE report concluded that a time delay of up to five minutes will not compromise the turbine protection and can be introduced to the RCIC initiated turbine trip, providing the total accumulated time of RCIC operation is monitored and does not exceed eight minutes per year.

It is concluded that this change will not alter the design or function of the main steam system or main turbine performance in a way that adversely affects the turbine protection or system performance or plant nuclear safety. The addition of the time delay will not affect the RCIC performance.

Safety Evaluation No.: **94-053 Rev. 1 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

Based on the analysis performed, it is concluded that the proposed change does not alter design, function, or method of performing the function of the safety-related system and is in compliance with NRC requirements.

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

Safety Evaluation No.: 94-055 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y94MX004

USAR Affected Pages: Figure 6.2-71a

System: Containment Atmosphere Monitoring (CMS)

Title of Change: Eliminate Moisture From H_2/O_2 Analyzers

Description of Change:

Water intrusion in the sample lines has had a consistent deleterious effect on the performance of both Train A and Train B hydrogen/oxygen (H_2/O_2) analyzer panels (2CMS*PNL66A and 2CMS*PNL66B). Problems range from water in the analyzing components (which result in inaccurate outputs) to equipment failures (sample pumps, analyzers, etc.). The source of water intrusion was determined to be due to the high humidity (in the primary containment and the area above the suppression pool) being carried into the sample suction lines.

Surveillance durations have increased as a result of having to repair/replace various components which are prone to water incursion damage. In addition, a 7-day LCO is started whenever a surveillance/maintenance is initiated on either H_2/O_2 train.

Safety Evaluation Summary:

This modification installed a moisture collector in both the sample inlet and return lines for both Trains A and B.

While the moisture collectors are identified as safety related, they have no permissive or control functions. They function only to prevent water intrusion into the H_2/O_2 analyzer system components. The modification will increase the systems' reliability and availability and decrease maintenance.

Most of the modification was installed prior to Refueling Outage 4 (RFO-4) with final tie-ins to the system during RFO-4.

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

Safety Evaluation No.:	94-056
Implementation Document No.:	DER 2-94-0217
USAR Affected Pages:	6.4-2
System:	N/A
Title of Change:	Change in NMP2 Control Room Supplies Requirement

Description of Change:

This change allows for the provision of food, sleeping facilities, and other personal comfort supplies from outside the Control Room vital area envelope on an as-needed basis. Providing food and sleeping facilities from either onsite or offsite sources can be readily performed when UFSAR-identified access routes are considered.

Safety Evaluation Summary:

Providing personal comfort supplies to the Control Room from outside the vital area envelope during design basis accident (DBA) conditions is in accordance with previously evaluated access routes described in the USAR. Habitability of the Control Room envelope without these supplies has been evaluated as consistent with the guidelines set by 10CFR50 Appendix A, General Design Criteria 19. This change does not affect any equipment important to safety previously evaluated in the USAR and has no impact on the safe operation or shutdown of the plant. This change has no impact on radiological effluents or nonradiological consequences to the environment.

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

Safety Evaluation No.: 94-058

Implementation Document No.: Drawing EY-8S

USAR Affected Pages: Figure 1.2-1

System: N/A

Title of Change: New Structures to Connect Unit 2 Access Control Building to the Plant

Description of Change:

The Unit 2 Access Control Building (Phase 1) was constructed in 1993. This structure was connected to the Reactor Building via temporary wooden structure. This wooden structure has been replaced with a permanent structure and an additional enclosed walkway from this structure to the Cardox Room/Auxiliary Services Building. These structures provide additional entry paths to both the radiologically-controlled areas and the nonradiologically controlled areas of Unit 2. The new passageway to the Cardox Room is outside the radiologically-controlled zone.

Safety Evaluation Summary:

The pertinent safety issues identified in this safety evaluation are impact on the Control Room fresh air intake, impact on the Flood Analysis, additional loads on Auxiliary Bay roof, Reactor Building and Control Building walls, impact on the CO_2 tank rupture analysis, and accessibility for the removal of the Auxiliary Bay roof plugs.

Based on the evaluation performed, it is concluded that construction of these interconnecting structures between the Access Control Building and the Unit 2 Plant structures does not involve an unreviewed safety question.

Safety Evaluation No.: 94-059 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y93MX004

USAR Affected Pages: N/A

Systems: Common Electrical (CES), Moisture Separator Vents/Drains (DSM), Moisture Separator Reheater Vents/Drains (DSR), Feedwater Pump Recirculation (FWR), Feedwater (FWS), High Pressure Feedwater Heater Drains (HDH), Low Pressure Feedwater Heater Drains (HDL), Hot Reheat (HRS), Meteorological Monitoring (MMS), Main Steam (MSS), Reactor Water Cleanup (WCS)

Title of Change: Modified ASME PTC 6.1 Turbine Generator Performance Test Capability for Unit 2

Description of Change:

A modified ASME PTC 6.1 Turbine Generator Performance Test (ASME Test) was required in order for General Electric (GE) to warrantee recovering the electrical megawatts lost due to the removal of two L-1 stage wheels in low-pressure turbines B and C by replacing rotors for low-pressure turbines A, B, and C.

This modification installed 50 thermocouples, a data acquisition terminal, and 6 condenser (basket tip) backpressure sensing lines (2 per condenser). These items, in addition to existing plant instrumentation, enabled the ASME test to be performed. Through the outputs of the data acquisition terminal, the modification also enabled the temperatures (sensed by the thermocouples) to be monitored on the site meteorological computer (METVAX).

Safety Evaluation Summary:

This modification will add supplementary instrument/pressure inputs in order to enable a modified ASME Test to be performed at Unit 2. This test is required in order to evaluate the efficiency of the turbine generator before and after the installation of the new low-pressure turbine monoblock rotors and, subsequently, the efficiency due to implementing power uprate. In addition, the change will enable plant personnel to permanently monitor temperatures at the power cycle

Safety Evaluation No.: **94-059 Rev. 0 & 1 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

block valves (PCBV) close to the condensers. Significant increase in temperatures, when the valves are closed, would be indicative of PCBV leakage.

The changes will have no impact on safe operation or shutdown of the plant. The modified ASME test is not discussed in the Unit 2 USAR. The Unit 1 USAR, Section II.C., describes the meteorology requirements and will not be impacted by the data link inputted from Unit 2 recorder 2CES-TJR100. Unit 2 USAR Section 2.3.3.2.3 describes the METVAX as a weather data processing system and it will not be impacted.

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

Safety Evaluation No.: 94-061

Implementation Document No.: PN2Y94MX008

USAR Affected Pages: Figures 9.2-1c, 9.2-1e, 9.2-1f, 9.2-1g, 9.2-1j, 9.2-1p

System: Service Water (SWP)

Title of Change: Installation of Service Water System Chemical Cleaning Valve Tie-Ins

Description of Change:

This modification provided the isolation and interface tie-ins necessary to facilitate chemical cleaning of the small diameter service water piping (i.e., 3-inch NPS and smaller) in the Reactor and Control Buildings. The cleaning operation was implemented to remove corrosion product and silt deposition from the affected piping and associated unit cooler coils. The cleaning process was the first step in suppressing further aggressive corrosion attack of the pipe surface due to microbiologically influenced corrosion. This precludes future costly piping repairs due to excessive pitting. The scope of this safety evaluation addresses the isolation valve tie-ins necessary to accommodate future cleaning. The actual cleaning operation will be addressed in separate documentation.

Safety Evaluation Summary:

This design provides in-line isolation valves and fittings for use in a future chemical cleaning operation. The isolation valves segment the affected headers into six independent cleaning loops. During normal operation, the isolation valves are maintained in the full open position and perform no active throttling or isolation function. During the cleaning operation the valves are closed, isolating the affected loop from the main header. Consideration for component access was addressed in the design and placement of the new valves and fittings. The system changes have been reviewed against the existing system flow calculations and system pipe stress calculations. These changes do not change or impede the function of the original installation. The new design provides isolation capabilities not included in the original installation, enhancing the operability and maintainability of the system.

This installation does not change the operation of the system or its function. The cleaning valve tie-ins provide the capability to support a future chemical cleaning operation. The added isolation capability enhances system maintenance

Safety Evaluation No.: **94-061 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

capabilities. These additions to the system do not change any licensing or design bases requirements of the SWP system.

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

Safety Evaluation No.: 94-062

Implementation Document No.: PN2Y94MX009

USAR Affected Pages: Figures 9.2-1c, 9.2-1e, 9.2-1g, 9.2-1L, 9.2-1m, 9.2-1p

System: Service Water (SWP)

Title of Change: Resize/Reroute SWP Piping

Description of Change:

This modification upsized approximately 1,100 ft. of SWP piping in the Reactor Building and Auxiliary Bays to improve the hydraulic performance of selected unit coolers.

Safety Evaluation Summary:

This project will replace approximately 1,100 ft. of the safety-related pipe. This pipe was identified, through hydraulic analysis, to be undersized for the duty requirements of the system. This review identified portions of piping for unit coolers 2HVR*UC401A through F, UC406, UC407A, B and C, UC408A and B, UC410A, UC411C and UC414A and B as having marginal clean pipe hydraulic performance. This design changes the size of the currently installed piping and in most cases conforms to the original routing of the existing pipe. There are some sections of the piping that will require minor reroutes for greater accessibility. The piping changes have been reviewed against the existing system flow calculations and system pipe stress calculations. These changes do not affect the function of the original installation. The new design provides isolation capabilities not included in the original installation, enhancing the operability and maintainability of the system.

Portions of the new piping will be installed in parallel with the existing piping. The existing pipe will be abandoned in place following the system tie-ins. Most piping installation will take place during normal power operation. The remainder of the pipe and all the required system connections will be installed during Refueling Outage 4. Precautions will be taken to insure that installation activities will not interfere with plant operation or endanger the ability of plant systems to perform their necessary functions.

The piping size increase enhances the capability of the system to respond to plant needs. This is accomplished by increasing the capability of the system to supply

Safety Evaluation No.: 94-062 (cont'd.)

Safety Evaluation Summary: (cont'd.)

service water to the coolers. The increases in pipe size will not change or impact the function of the system. Considering the documents reviewed it has been determined that this modification complies with all of the design and licensing requirements applicable to the Unit 2 SWP system.

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

Safety Evaluation No.: 94-063

Implementation Document No.: Simple Design Change SC2-0029-94

USAR Affected Pages: Figure 9.4-9 Sh 1, 2 & 3

System: Reactor Building Ventilation (HVR)

Title of Change: HVR Fans Repeatedly Failed to Start

Description of Change:

Reactor Building normal ventilation system spare/standby supply and exhaust fans have repeatedly failed to auto start or manually start. The start circuits for the fans contain a permissive logic requiring respective discharge dampers to be greater than 40% open. When a fan start signal is received, a relay/timer is initiated. Also, the start signal initiates the opening of the associated discharge dampers. If the associated discharge damper takes longer than 10 seconds to open to 40% open position, the timer will time out and send a signal to close the discharge damper; the fan will not auto start.

Field testing demonstrated the start sequence was between 7.3 and 9.2 seconds with the relay/timer tripping in 10.1 seconds; relay/timer range was 1.5 to 15 seconds. The margin allowed between "Start" and "Fail to Start" was too restrictive and did not allow for any anomaly or variance of operation. Therefore, the relay/timer setpoint was increased to ≥12 seconds.

Safety Evaluation Summary:

This change will enhance the performance of the nonsafety-related portion of the Reactor Building normal heating, ventilating and air conditioning system and will not affect the operation of any systems important to the safe operation and shutdown of the plant.

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

Safety Evaluation No.:	94-064
Implementation Document No.:	Simple Design Change SC2-0118-94
USAR Affected Pages:	9.5-33, 9A.3-16, 9A.3-17, 9A.3-63; Table 9A.3-7 Sh 2 & 3; Figure 9A.3-5
System:	N/A
Title of Change:	Deletion of Fire Barrier Rating - Diesel Generator Day Tank Rooms

Description of Change:

This change deleted the requirement for a three-hour rated fire barrier separating each diesel fuel day tank from its associated diesel generator.

Safety Evaluation Summary:

The current Fire Hazards Analysis, as presented in USAR Section 9.A, postulates a fire in each diesel generator area which includes the entire inventory of diesel fuel contained within the day tank. Since the diesel generator will not function without the day tank supplying fuel and the day tank has no value without the availability of a diesel generator, the provision of a fire barrier to isolate the fuel supply from the diesel generator is of little value. The postulated fire does not credit the fire barrier with isolating the day tank from the diesel generator. Elimination of the fire barrier does not place the area outside of compliance with applicable criteria since diking and spill containment, in accordance with BTP CMEB 9.5-1 Position C.7.i.(2), is maintained.

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

Safety Evaluation No.: 94-065

Implementation Document No.: Simple Design Change SC2-0034-94

USAR Affected Pages: Table 3.9A-12 Sh 12 & 13; Figures 9.2-1e, 9.2-1f, 9.2-1j, 9.2-1L

System: Service Water (SWP)

Title of Change: SWP Check Valve Removal, Relocation or Replacement

Description of Change:

The SWP system was reviewed by the project team in response to industry concerns and Unit 2 performance problems noted during SWP check valve and unit cooler tests. A study was performed which noted several areas for improvement of the system performance. The scope of this project addressed one aspect of the proposed system improvements recommended by the study. This project addressed check valve performance enhancements which included:

- Removal of the internals from check valves 2SWP*V75A and 75B.
- Relocation of check valves 2SWP*V1024 and V1025, and the installation of blocking valves and drain provisions.
- Replacement of lift check valves 2SWP*V201A and 201B with nozzle check valves, and rerouting the associated piping.
- Replacement of swing check valves 2SWP*V240A and 240B with nozzle check valves.

Safety Evaluation Summary:

The check valve changes enhance the capability of the system to respond to plant needs. This is accomplished by improving valve reliability, eliminating unnecessary maintenance and testing, and increasing the capability of the system to supply service water to the associated heat loads. These changes do not adversely change or impact the function of the system and comply with all of the design and licensing requirements applicable to the Unit 2 SWP system.

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

Safety Evaluation No.: 94-066

Implementation Document No.: Temporary Mod. 94-039

USAR Affected Pages: N/A

System: Reactor Protection System (RPS), Nuclear Steam Supply System (NSSS), Main Steam System (MSS)

Title of Change: Defeat of Main Steam Line Rad Monitoring Trip Signal Channel B1

Description of Change:

This temporary modification installed a jumper in panel 2CEC*PNL633 Bay B in order to defeat a trip signal (Channel B1) which would normally be provided whenever detector 2MSS*RE46B is inoperable.

Safety Evaluation Summary:

This temporary modification allows for ample time for the replacement of the faulty detector without entering a LCO per Technical Specifications 3/4.3.1.a and 3/4.3.2.b.1.b. This change reduces the plant's vulnerability to a full scram by prohibiting the half scram signal to be present during the time period that the detector is being replaced. In the event of fuel damage, the remaining main steam line radiation monitors will function to detect the release of fission products and initiate the appropriate mitigating actions to limit the release and to shut down the plant. This change does not impact the remaining detectors from performing their safety functions as originally designed.

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

Safety Evaluation No.: 94-067

Implementation Document No.: DER 2-94-0036

USAR Affected Pages: 9.1-9, 9.1-10

System: N/A

Title of Change: Update UFSAR Description of a Revised Boraflex Surveillance Program and Use of New Surveillance Assemblies

Description of Change:

US Tool & Die (UST&D) originally supplied a surveillance sample consisting of 2-inch square pieces of Boraflex for the Unit 2 spent fuel storage racks. This was shipped in 1984 and was the standard surveillance sample supplied by UST&D at that time. However, these samples were lost and therefore never installed in the Unit 2 spent fuel pool. Replacement Boraflex surveillance samples were purchased in 1990. These coupons were installed prior to the racks being put into use and the hottest bundles from each reload have been placed around them. This safety evaluation updated the description in the SAR of the Boraflex coupon surveillance assemblies to be in agreement with the actual coupons which were installed in 1990. In addition, the long-term Boraflex Surveillance Program has been modified to be in agreement with the most recent industry guidance for Boraflex surveillance, issued in a report by the Electric Power Research Institute (EPRI).

Safety Evaluation Summary:

Performing periodic testing of the Boraflex coupons installed in Unit 2's spent fuel racks is necessary to provide assurance that the Boraflex material in the spent fuel racks continues to perform acceptably over the service life of the racks. The replacement coupons and their associated surveillance program will provide adequate assurance that the Boraflex material is performing as intended and will preclude the occurrence of a criticality accident due to degradation of the Boraflex material. This revised surveillance program will continue to implement the requirements of GDC 61 and is a more conservative program than the original one described in the USAR. Neither the Boraflex coupons or their associated surveillance program present a safety concern for Unit 2.

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

Safety Evaluation No.: 94-068 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0040-93

USAR Affected Pages: 9.1-23, 9.1-24, 9.1-40, 9C.3-4, 9C.3-5, 9C.8-1, 9C.8-2, 9C.8-3; Tables 3.9A-4 Sh 6, 3-1 Sh 1, 3-3 Sh 1; Figure 5-2

System: MHR

Title of Change: Upgrade of 125 Ton Polar Crane to 132 Ton Capacity

Description of Change:

This change to the polar crane's use and function upgraded the main hoist from 125-ton to 132-ton capacity to allow for the full utilization of the reactor pressure vessel (RPV) head carousel strongback which was initially employed during Refueling Outage 3.

The upgrading of the polar crane main hoist involved modifications and recertification load testing of the hoist at 125% of the new rated capacity during plant operation. The constructibility review, the load drop assessment calculation, and the load test procedure ensured that plant equipment was not affected by the work and testing. The upgrading of the polar crane main hoist reduces personnel exposure and saves critical path refuel outage time by permitting the removal and reinstallation of the RPV head, the stud tensioners, the 76 studs, nuts, and washers in one lifting operation.

Safety Evaluation Summary:

The design changes and reanalysis of the polar crane main hoist for the upgrade to 132 tons meets the same single-failure proof criteria of NUREG-0612, as did the existing equipment.

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

Safety Evaluation No.:	94-069
Implementation Document No.:	Procedure N2-CSP-2V
USAR Affected Pages:	10.4-2, 10.4-22 through 10.4-27
System:	Condensate Demineralizer System
Title of Change:	Condensate Demineralizer Water Purity Maintenance

Description of Change:

This change replaced condensate demineralizer resin at intervals which are based on inlet conductivity and its relationship to the composition of the circulating water, and condensate flow rate through the beds since regeneration of resin is no longer performed at Unit 2. This approach to the operation of the demineralizers eliminates the potential for contamination from the products formed during acid and caustic regeneration of the resin.

Safety Evaluation Summary:

The condensate demineralizer resin will be replaced such that adequate remaining capacity will exist to handle the postulated main condenser leak event within the time permitted for an orderly shutdown. The effluent quality of the demineralizer system will satisfy the acceptable limit found in Table 2 of Regulatory Guide 1.56, Rev. 1, July 1978. The replacement of condensate demineralizer resins at these frequencies is in compliance with paragraph C of the regulatory guide.

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

Safety Evaluation No.: 94-070

Implementation Document No.: Simple Design Change SC2-0028-94

USAR Affected Pages: Figure 10.1-6b

System: Feedwater (FWS)

Title of Change: Replacement of Drain Valves 2FWS*V89A&B and 2FWS*V90A&B

Description of Change:

This change replaced drain valves 2FWS*V89A&B and V90A&B for containment isolation valves 2FWS*V12A&B. These valves are 3/4" NPS and are normally closed. The valves are not active components and their only safety function is pressure retention.

These valves are located in the primary containment and were replaced to enhance leak-tightness and prevent leakage in the drywell.

Safety Evaluation Summary:

Installation of the replacement drain valves does not impact the design of the FWS system. The system can still provide its intended flow and new drain valves 2FWS*V89A&B and V90A&B will assure system integrity.

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

Safety Evaluation No.: 94-071

Implementation Document No.: N/A

USAR Affected Pages: 2.2-1, 2.2-3, 2.2-8; Table 2.2-3; Figure 2.1-2

System: N/A

Title of Change: Gas Pipeline #63 to Sithe Energies USA Plant and #58 to Indeck Energy

Description of Change:

Sithe Energies USA has constructed a 1000-MW natural gas-fired electrical generating station known as Independence Generation Plant. It is a cogeneration plant located in Oswego County, New York.

Two natural gas pipelines lie within 8 km (5 miles) of the Nine Mile Point Station. One pipeline (#63) supplies gas to the Sithe plant and the other pipeline (#58) to Indeck Energy.

Safety Evaluation Summary:

The nearest point of the pipelines is over 2 miles from Unit 2. Due to the distance between the pipelines and Nine Mile Station, atmospheric dispersion would conservatively reduce the natural gas concentration below its lower explosive limit more than 1 mile from Nine Mile Point. The detonation of an unconfined natural gas dispersal in air is not a credible event.

Due to the distance from the pipelines to the Unit 2 Control Room air intake (>2 miles), the resultant atmospheric dispersion would conservatively reduce the natural gas concentration at the intakes to less than 9 g/m^3. This is well within the natural gas toxicity limit of 287 g/m^3.

Based on the evaluation performed, it is concluded that installation of the gas pipelines does not involve an unreviewed safety question.

Safety Evaluation No.: 94-073

Implementation Document No.: Simple Design Change SC2-0142-94

USAR Affected Pages: 9.4-71; Figures 9.4-22b through d

System: Heating and Glycol (HVG), Water Treatment (WTS)

Title of Change: Water Treating Makeup Water Supply to HVG

Description of Change:

This change isolated the hot water heating and glycol (HVG) system from makeup water supplied by the makeup water treatment system (WTS) because WTS has been repeatedly contaminated by glycol intrusion through leaky HVG valves. The needed makeup water for HVG was supplied manually by Operations under procedural controls. The isolation was accomplished by removing the in-line check valves and blanking off the lines.

Safety Evaluation Summary:

Neither the HVG nor the WTS systems have any safety-related functions. Failure or malfunction of the systems or components will not compromise any safety-related systems or components or prevent a safe reactor shutdown.

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

Safety Evaluation No.: 94-076

Implementation Document No.: DER 2-93-1935

USAR Affected Pages: Table 6.2-56 Sh 2, 4, 7, 10, 11, 12, 20, 21

System: Primary Containment

Title of Change: Appendix J Discrepancies, DER 2-93-1935

Description of Change:

DER 2-93-1935 addressed discrepancies involving editorial changes to the UFSAR clarifying the Unit 2 Appendix J Program.

This safety evaluation addresses changes to USAR Table 6.2-56 as follows:

1. Deleted reference to Type C testing for the following valves:

   2RHS*MOV1A, 2RHS*MOV1B, 2RHS*MOV1C, 2CSH*MOV118, 2CSL*MOV112 and 2ICS*MOV136

2. Revised Notes 23, 24 and 25 to delete reference to Type A testing of the following relief, safety, check and vacuum breaker valves:

   2RHS*SV34A/B, 2RHS*SV62A/B, 2RHS*RV56A/B, 2RHS*V20, 2RHS*V19, 2RHS*V117, 2RHS*V118, 2RHS*RVV35A/B, 2RHS*RVV36A/B, 2RHS*RV108, 2RHS*RV20A/B/C, 2RHS*RV61A/B/C, 2RHS*RV110, 2CSL*RV123, 2CSL*RV105, 2RHS*RV139, 2CSH*RV113 and 2CSH*RV114

3. Revised Table 6.2-56 to indicate a Type C test for valves 2CCP*RV170 and 2CCP*RV101.

Safety Evaluation Summary:

The valves in Item 1 are ECCS suction valves that take suction from the suppression pool at an elevation below minimum suppression pool water level of 199'-6" and, as such, are waterfilled post-LOCA. The valves in Item 2 are relief, safety, check and vacuum breaker valves that terminate in the suppression pool below the minimum water level of 199'-6". The suppression pool water effectively seals these containment isolation valves from the primary containment atmosphere, thereby preventing gaseous releases from the primary containment. Since these

Safety Evaluation No.: 94-076 (cont'd.)

Safety Evaluation Summary: (cont'd.)

valves do not see containment atmosphere post-LOCA, they do not represent potential containment atmospheric leakage paths and are not subject to leak testing as defined in Appendix J.

Relief valves 2CCP*RV170 and 2CCP*RV171 are located inside primary containment and their outlets terminate open ended inside the primary containment at an equipment drain and, as such, provide an atmospheric leak path from the primary containment. Therefore, these valves are considered containment isolation valves as defined in Appendix J and are Type C tested to satisfy the requirements of Appendix J.

This change revises USAR Table 6.2-56 to accurately denote the proper leak testing provisions for the valves in Items 1, 2 and 3. This is an editorial change and does not change leak testing requirements or leak testing methods of the Unit 2 Appendix J test program. The applicable containment isolation valves in Items 1, 2 and 3 will continue to be properly leak tested per existing procedures to ensure leak-tight integrity as required by 10CFR50 Appendix J and ASME XI.

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

Safety Evaluation No.: 94-077

Implementation Document No.: Procedure N2-OP-100B

USAR Affected Pages: 9.5-43, 9.5-69

System: Emergency Diesel Generator Lube Oil

Title of Change: Div. 3 EDG Lube Oil Temperature

Description of Change:

The lube oil temperature for the Div. 3 emergency diesel generator (2EGS*EG2) was stated in the USAR to be above 120°F during standby conditions. However, actual lube oil temperature was observed to be between 90°F and 110°F during standby conditions. With the vendor's concurrence, it has been established that the minimum standby lube oil temperature requirement for the Div. 3 diesel generator is 85°F.

Safety Evaluation Summary:

Establishing the requirement of 85°F as the minimum lube oil temperature during standby conditions is consistent with the acceptance criteria in NUREG-0800, that the temperature of the lubricating oil is maintained above a minimum value to enhance the "first-try" starting reliability of the engine in the standby condition. This change is also consistent with the vendor's recommendation for minimum lube oil temperature and will supply proper standby lubrication to the engine. Therefore, the actual lube oil temperature of between 90°F and 110°F is sufficient to verify that engine lube oil temperature requirements are met under standby conditions.

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

Safety Evaluation No.: 94-078

Implementation Document No.: Procedure N2-OPS-RPS-W001

USAR Affected Pages: N/A

System: MSS, HRS

Title of Change: Turbine Steam Valve Surveillance Test Interval Extension

Description of Change:

This safety evaluation evaluates changing the testing requirement for turbine control, stop, and intercept valves from weekly to monthly.

(NOTE: Subsequent changes have been evaluated under Safety Evaluation 95-032).

Safety Evaluation Summary:

The purpose of this testing is to discover any valve malfunctions that could contribute to a turbine overspeed event causing turbine components to become high-energy debris (missiles) capable of striking and damaging safety-related equipment.

The NRC Safety Evaluation Report for Unit 2 (NUREG-1047, Section 3.5.1.3.10), with regard to the turbine missile issue, concluded that the probability of unacceptable damage to safety-related structures, systems and components due to turbine missiles is acceptably low (i.e., 10^{-7} per year), provided that the total turbine missile generation probability is such that conformance with the NRC criteria (i.e., $P1 < 10^{-4}$ for favorably oriented turbines, $P1 < 10^{-5}$ for unfavorably oriented turbines) is maintained throughout the life of the plant by acceptable inspection and test programs. In reaching the conclusion, the NRC staff factored into consideration the favorable orientation of the Unit 2 turbine generator. Also in the Unit 2 Safety Evaluation, the NRC identified that the relevant General Electric (GE) missile probability analysis may be used in determining the inspection interval for the turbine discs at Unit 2.

The existing requirement for surveillance testing of turbine stop valves (TSVs), turbine control valves (TCVs) and turbine contained stop and intercept valves (CIVs) is to perform these tests on a weekly basis. In order to assure plant availability and decrease any potential of plant scrams, the surveillance testing

Safety Evaluation No.: 94-078 (cont'd.)

Safety Evaluation Summary: (cont'd.)

frequency of these valves is being temporarily extended (up to Refuel Outage 4 (RF04)) from a weekly to a monthly interval. Justification for this change in frequency is provided below. During RF04, the low-pressure turbine rotors are being replaced by monoblock rotors. Missile generation is not a concern for monoblock rotors. As part of the monoblock rotor modification, a separate safety evaluation will be prepared which will identify surveillance testing requirements for TSVs, TCVs and CIVs. It is anticipated that with the replacement of the existing "built-up" low-pressure rotors with monoblock rotors, surveillance testing frequency of TSVs, TCVs and CIVs can further be reduced (from monthly to quarterly).

In response to Niagara Mohawk's request, GE recalculated wheel missile probabilities for the Unit 2 low-pressure turbine rotors. These new calculations were based on the revised calculation procedure that 1) included updated failure rate data on the primary steam valves of GE nuclear units, and 2) included the capability of calculating wheel missile probabilities for extended time intervals between the GE normally recommended functional tests of the steam valves.

GE evaluation indicates that, based on NRC criteria (P1 < 1×10^{-4} for favorably oriented units), Q/Q/Q testing, and with no pre-warming, the inspection interval is reduced to 5.7 years for the A rotor. Considering pre-warming, M/Q/M testing, and based on GE recommendation (P1 < 1×10^{-5} for the unit), the A rotor inspection interval is reduced to 2.8 years. The existing inspection interval is 6 years. Based on existing frequency of testing (W/W/W) and test schedule of M/M/M (conservatively, Unit 2 shall utilize M/M/M schedule instead of M/Q/M analyzed by GE), which is anticipated to be utilized for a very short duration of the present operating cycle, the reduced inspection interval of 2.8 years would require A rotor inspection by R5.

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

Safety Evaluation No.: 94-079

Implementation Document No.: Procedures N2-FSP-FPM-A001-1 through 5

USAR Affected Pages: 9.5-2, 9A.3-43, 9A.3-44; Table 9.5-3, Sh 7 & 8

System: Fire Detection System

Title of Change: Performance Based Fire Detector Surveillance Testing

Description of Change:

This change added a clarifying paragraph that indicates that subsequent editions of NFPA codes and standards may be used for subsequent plant modifications and program revisions. Clarification was also added that a deviation from NFPA-72 specified fire detector test frequencies is utilized for fire zones that do not contain any equipment considered important to safety. Also, a deviation from NFPA-72 code specified testing requirements in fire zones containing safety-related equipment has been adopted. This deviation is based on obtaining equivalent reliability between test intervals as allowed by NFPA code equivalency provisions. The revised testing scheme uses a 10%, 20% expanded, total zone expanded, rotating test sample population for testing that is conducted on an annual basis.

Safety Evaluation Summary:

The clarifications are information additions that do not affect safety-related equipment and are not changes from present operating policies. The change to the testing scheme for fire detection in safety-related equipment areas is based on the plant-specific failure rate (failure to alarm under simulated fire conditions) of fire detection instruments. The scheme adopted provides an equivalent or reduced probability of a detector failure between test intervals than that generally assured by NFPA-72 annual test intervals using generic failure probability. The testing scheme change does not increase the probability of a postulated fire in the Fire Hazards Analysis, nor does it increase or decrease the severity of the fire.

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

Safety Evaluation No.: 94-080

Implementation Document No.: Mod. PN2Y94MX006

USAR Affected Pages: 3.7B-17, 3.9A-3, 3.9A-4, 3.9B-49; Table 3.9B-2m Sh 1 & 2

System: Hydrogen Recombiner (HCS), Reactor Water Recirculation (RCS), Residual Heat Removal (RHS), Service Water (SWP)

Title of Change: NMP2 Snubber Reduction

Description of Change:

This modification reduced the number of mechanical snubbers on Unit 2 safety-related piping systems by reanalyzing the piping systems for snubber removal or snubber replacement with struts.

Safety Evaluation Summary:

Due to failure rates associated with snubbers, snubber removal results in piping systems that are more reliable. Other benefits of the program include reduced long-term maintenance, inspection and test requirements.

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

Safety Evaluation No.: 94-082

Implementation Document No.: DER 2-93-2060

USAR Affected Pages: Figures 10.1-9e, 10.4-9 Sh 10

System: Condensate Demineralizer (CND)

Title of Change: CND Ultrasonic Resin Cleaner (URC) Level Element 2CND-LE225 Not Installed

Description of Change:

This safety evaluation documents the as-built plant condition for the CND system ultrasonic resin cleaner tank as not having a level element installed and the corresponding level alarms inoperative. Additionally, the associated sluicing water flow control valve has been maintained in the full open position since plant startup with approved holdout tag preventing misoperation.

Safety Evaluation Summary:

The ultrasonic resin cleaning equipment does not interface with or affect any equipment important to safety, and the CND system is not required to effect or support safe shutdown of the reactor or to perform in the operation of any reactor safety features.

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

Safety Evaluation No.: 94-083

Implementation Document No.: Procedure N2-FSP-FPM-R001

USAR Affected Pages: Table 9A.3-18 Sh 2

System: Fire Detection (FPM)

Title of Change: Elimination of Periodic Test Requirements for Thermal Fire Detectors in Fire Zone 252SW

Description of Change:

This change removed four thermal fire detectors from the scope of USAR-specified periodic testing. The thermal detectors are located in the SFP Phase Separator Tank Room on Reactor Building elevation 289'-0" and immediately outside the room. The area outside the room is also provided detection coverage by ionization detectors in another loop of the detection zone. The combustible loading within the room is insufficient to warrant fire detection. This change left the thermal detectors installed but will not require periodic testing in order for the zone to be considered operable. In addition, should one of the detectors go into alarm due to some future failure, this change allows the loop to be bypassed in the panel without declaring the zone of detection inoperable. This change was implemented since very high radiation levels normally present in the SFP Phase Separator Tank Room prevent testing in accordance with previous requirements.

Safety Evaluation Summary:

The reliability and margin of safety discussed in USAR Section 9A.3.6.1.7 will be maintained by this change since redundant ionization detection in the area will be maintained. The combustible loading within the SFP Phase Separator Tank Room is insufficient to warrant detection and has been documented in a fire protection engineering evaluation. The safe shutdown analysis is unaffected by this change as is the Fire Hazards Analysis.

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

Safety Evaluation No.: 94-087

Implementation Document No.: NUREG-0654, NUREG-0696, NUREG-0737

USAR Affected Pages: 13.3-2

System: N/A

Title of Change: Eliminate Corporate Emergency Operations Center

Description of Change:

NUREGS 0654, 0696, and 0737 contain criteria pertaining to emergency response facilities. There is no requirement specific to maintaining an offsite Corporate Emergency Operations Center (CEOC). There is, however, a requirement to maintain a backup Emergency Operations Facility (EOF) should the onsite EOF become uninhabitable. The offsite location previously used to obtain Engineering support was referred to as the CEOC in the Site Emergency Plan. The CEOC has historically provided a convenient location for obtaining Engineering support during emergency scenarios. The CEOC contained resources (drawings, calculations, personal references) typically used by Engineering personnel. The location was outside the ten-mile emergency zone and dedicated phone lines were used to ensure communication with the Technical Support Center (TSC) and EOF. The Engineering Department recently relocated to a building on site and could be affected by evacuation requirements. For emergency events which do not require evacuation, communication via phone lines ensures access to the same level of technical support previously available. However, if site evacuation is required, access to the resources contained in the Engineering Building could be lost. Under these scenarios, technical support would come solely from the TSC and EOF and would be dependent on the amount of technical information available in those facilities.

Safety Evaluation Summary:

Conformance with applicable criteria is assured since: 1) there is not a regulatory requirement for a CEOC, and 2) except under scenarios requiring evacuation, the proximity of Engineering resources to the plant will improve support. Therefore, there is no net negative impact from relocating the Engineering Department to the site and eliminating the CEOC.

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

Safety Evaluation No.: 94-088

Implementation Document No.: Simple Design Change SC2-0167-94

USAR Affected Pages: Page 7.7-20; Figure 7.7-6 Sh 3

System: Reactor Coolant System (RCS)

Title of Change: Recirculation Flow Control Valve Minimum Position Change

Description of Change:

The reactor recirculation flow control valves can become stuck at minimum position due to the differential pressure across the valve after the respective pump is transferred to high speed. This change now permits increasing the valve position to a maximum of 22% open (hot indicated) with the valve limit switch bypassed while the first pump was upshifted, and a maximum of 20% open (hot indicated) while the second pump was upshifted.

Safety Evaluation Summary:

The peak neutron flux that will result from the increased flow when the recirculation pumps are upshifted is conservatively below the high neutron flux scram.

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

Safety Evaluation No.: 94-089

Implementation Document No.: Simple Design Change SC2-0019-94

USAR Affected Pages: Figures 10.4-1, 10.4-2a

System: Condensate Air Removal (ARC)

Title of Change: Design Change to 2ARC-AOV104

Description of Change:

Valve 2ARC-AOV104 failed to open when required. The root cause for this deviation was determined to be improperly sized spring for the design condition. The most cost-effective repair was to retrofit this actuator to open (and close) on air. This design change deleted the fail open requirement for this valve.

Safety Evaluation Summary:

Since the valve is nonsafety related, no safety concerns exist in allowing this valve to open on air. The air will provide the necessary force to break the valve away from its seat. Additionally, should the valve fail once open, it will remain open maintaining condenser vacuum. Allowing the valve to open and close on air will not adversely affect nuclear safety.

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

Safety Evaluation No.: 95-005

Implementation Document No.: Procedure NSAS-POL-01, Procedure NEP-POL-0101

USAR Affected Pages: 13.1-4, 13.1-5, 13.1-7; Figures 13.1-3, 13.1-5

System: N/A

Title of Change: Reorganization of the Information Management Branch to the Nuclear Safety Assessment and Support Department

Description of Change:

The functions of the Information Management Branch have been relocated from the Nuclear Engineering organization to the Nuclear Safety Assessment and Support (NSAS) organization.

Safety Evaluation Summary:

Relocation of the administrative support functions provided by the Information Management Branch to the NSAS Department is consistent with the charter and responsibilities of that department and maintains clear management control and effective lines of authority and communication between the organizational units involved in the management, operation, and technical support for the operation of the facility.

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

Safety Evaluation No.:	95-030
Implementation Document No.:	Procedures S-RTP-165, S-RPIP-3.11
USAR Affected Pages:	Tables 1.8-2 Sh 6, 1.9-1 Sh 49 & 50, 12.5-3
System:	N/A
Title of Change:	Use of Audible Alarm Dosimeters and Personnel Air Samplers

Description of Change:

This change revised the USAR to agree with current Radiation Protection Program procedures for the use of audible alarm dosimeters and personnel air samplers.

Safety Evaluation Summary:

The proposed changes to the Unit 2 USAR will meet the intent of 10CFR20 and comply with applicable portions of regulatory guidelines.

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

Safety Evaluation No.: 95-032 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y93MX005

USAR Affected Pages: 3.5-8, 3.5-9, 3.5-16, 10.2-1, 10.2-8, 10.2-9, 10.2-10, 10.2-11, 10.2-12, 10.2-13; Tables 3.5-3, 3.5-6, 3.5-9, 3.5-12, 3.5-15, 3.5-16, 10.2-1 Sh 1 & 2

System: TMS

Title of Change: Low-Pressure Turbine Monoblock Rotor Replacement

Description of Change:

This modification replaced the three existing low-pressure turbine rotors (2TMS-T2A,B,C) with General Electric (GE) monoblock design rotors. The previous low-pressure turbine rotors were of a built-up design (shrunk-on wheels). The shrunk-on wheel design has a potential of developing cracking in the keyway, web and hub area due to stress corrosion cracking (SCC). The monoblock rotor design has been adopted as a corrective measure against SCC.

The benefit of replacement of the existing low-pressure turbine rotors with monoblock rotor is as follows:

a. Recovery of lost MWe due to wheel removal. GE guarantees a 28.3 MWe recovery.

b. Reduced low-pressure turbine rotor inspections. The recommended inspection frequency reduces from 6 to 10 years.

c. Reduced turbine valve testing.

d. Replacement monoblock rotors support reduced outage durations.

e. Cobalt reduction. The previous low-pressure turbine last-stage buckets utilized stellite erosion shields. The replacement buckets are flame hardened. The replacement last-stage buckets result in a reduction in radiation exposure to plant personnel.

Safety Evaluation No.: 95-032 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary:

The monoblock rotors were designed to meet the requirements of the previous design/operating conditions, including transient operating conditions. At the time of scheduled installation (1995), Unit 2 will be undergoing a power uprate. The monoblock rotors, therefore, conform to the design requirements established for power uprate. The requirements include the following:

Guaranteed Rating:	1,210,902 kW
Initial Steam Conditions:	1003 psia
Exhaust Pressure:	2.0" Hg Abs backpressure
Guaranteed Flow:	13,583,244 lbs/hr

The turbine rotors are designed with 5 percent flow margin above the flow required to meet the maximum guaranteed output for power uprate. The turbine generator is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features. The turbine generator is, however, designed to minimize the possibility of turbine rotor failure that might produce a high-energy missile that could damage a safety-related component.

Replacement with monoblock rotors reduces the probability of missile generation by removing the potential for SCC at the interface between a rotor and wheel via the use of a monoblock construction. The overall probability of damage by turbine missiles (for monoblock rotors) will be within the acceptance value of 10^{-6}/yr, as outlined in SRP 2.2.3, and the acceptance value of 10^{-7}/yr, as specified in Regulatory Guide 1.115. The existing overspeed protection controls will prevent the rotor from exceeding the maximum transient speed of 120 percent (design overspeed) of rated turbine speed. This safety evaluation also re-evaluated testing requirements for various turbine devices based on monoblock rotor replacement.

Adequate procedural controls shall be maintained such that there will be no adverse effect to nuclear safety both during the transportation and installation of the monoblock rotors and during storage of the removed rotors.

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

Safety Evaluation No.: 95-033

Implementation Document No.: Mod. PN2Y94MX003

USAR Affected Pages: Figures 9.2-8b, 9.2-9b

System: Domestic Water (DWS), Sanitary Plumbing (PBS), Auxiliary Service Building HVAC (HVL), Paging System (COP)

Title of Change: Auxiliary Building Renovation, RFO4 Scope

Description of Change:

This change renovated the Auxiliary Service Building elevation 261'-0" to allow use as controlled personnel ingress and egress to/from the Turbine and Reactor Buildings via the linkway during Refueling Outage 4 (RFO4). This change involved making an opening in the 13 line wall at elevation 261' near the entrance to the south electrical tunnel stairwell, installation of an additional 1.5-hr. fire-rated door (ET262-6) for stairway isolation, removal of lockers, removal of the drinking fountain and wash basins, the capping of floor drain(s) in the temporary access passageway, and the removal of door AS261-7 for improved access. Also, door ET262-4 was removed while the area was being used only for access and egress during RFO4. The temporary access passageway was created by installing painted gypsum wallboard partitions. The ceiling tile grid and associated services were revised in the area of the passageway. These changes are partial scope for this modification. After the 1995 refuel outage (RFO4), the modification was resumed to provide a radiation protection calibration laboratory, storage room, separate male/female personnel decontamination facilities, removal of the 1,980-gallon hot water heater and replace it with a 120-gallon capacity, install a different design door for ET262-4, and install an equipment lift from Turbine Building elevation 250'-0" and elevation 261'-0" of the Auxiliary Service Building. The changes to be made after completion of RFO4 will be addressed in a subsequent safety evaluation.

Safety Evaluation Summary:

The changes being made to the south electrical tunnel stairwell were addressed for conformance with General Design Criterion (GDC) 2 and no adverse impact was created. Potential impact to adjacent safety-related areas and conformance to GDC 3 and 10CFR50 Appendix R were evaluated and conformance was maintained. Building services were revised and did not impact any operation of equipment important to safety. Based on the evaluation performed, it is concluded that this change does not involve an unreviewed safety question.

Safety Evaluation No.:	95-034
Implementation Document No.:	Calculation ES-216-00B Calculation H21C-038-01B
USAR Affected Pages:	Tables 15.6-13 Sh 2 through 11, 15.6-16b
System:	Reactor Containment Purge (CPS), Gas-Nitrogen (GSN)
Title of Change:	Revised Bypass Leakage Design Basis 2GSN*V205

Description of Change:

This change revised the design basis analyses for determination of the bypass leakage through the CPS wetwell and drywell supply lines and resultant doses due to the increased leakage. The increased leakage was due to removal of leakage mitigation credit for check valve 2GSN*V205. Since this valve is not part of a leak test program, and will not be added to one, credit cannot be taken for leakage reduction following a design basis LOCA, as had been assumed in the original design basis analyses.

Safety Evaluation Summary:

This safety evaluation concludes that an unreviewed safety question does not exist as a result of removing leakage mitigation credit for check valve 2GSN*V205. This conclusion is based on the calculation of the additional leakage attributable to deleting credit for the check valve, and determination that the resultant doses will not cause the limits of 10CFR100 or 10CFR50 GDC 19 to be exceeded. The calculated doses at the exclusion area boundary, low population zone, and in the Control Room increase as a result of this change; however, they remain below the limits of 10CFR100 and GDC 19. Since Unit 2 is licensed to the limits of 10CFR100 and GDC 19, the consequences of a design basis accident are determined not to have increased.

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

Safety Evaluation No.: 95-035

Implementation Document No.: Simple Design Change SC2-0148-94

USAR Affected Pages: Figures 9.2-6a, 9.2-17b

System: Condensate Makeup and Drawoff (CNS), Makeup Water (MWS)

Title of Change: Elimination of Hose Connection Headers for MWS and CNS Systems

Description of Change:

This change eliminated hose connection headers and installed new permanent hose connections to allow movement of large equipment into the decontamination room of the Dirty Workshop on elevation 261′.

Safety Evaluation Summary:

An engineering review found that eliminating hose connection headers in the decontamination room and installing new permanent hose connections will improve movement of large equipment and still supply CNS and MWS to the decontamination room.

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

Safety Evaluation No.: 95-036

Implementation Document No.: Simple Design Change SC2-0157-94

USAR Affected Pages: 9.5-17, 9.5-18, 9.5-21; Table 8.3-1 Sh 3 & 4; Figures 8.3-2, 8.3-3, 8.3-4, 8.3-6 Sh 5 & 27

System: LAR, EJS, NJS

Title of Change: Change Power Supply to 2LAR-PNL200

Description of Change:

This change separated the nondivisional Reactor Building normal lighting system from its present Divisional Class 1E power source 2EJS*US1 and connected it to its originally designed nondivisional source 2NJS-US2. This change also disconnected a loss-of-coolant accident signal circuitry to 2EJS*US1 which provided for tripping the breaker feeding the lighting panel and the associated computer points and annunciation circuitry to the plant process computer and to panel 2CEC*PNL852, respectively. Due to a NRC approved extension in the amount of time allowed for the Reactor Building drawdown, a new drawdown analysis has determined that the heat load generated from the Reactor Building normal lighting system would no longer prevent the drawdown from being achieved.

Safety Evaluation Summary:

The work scope is minimal and involves the disconnection and connection of existing cables and functional testing. No new cables or raceways will be installed and those cables to be spared will be abandoned in place. Separation criteria, Appendix R requirements, and electrical protection will be maintained.

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

Safety Evaluation No.: 95-037

Implementation Document No.: Simple Design Change SC2-0019-95

USAR Affected Pages: Figure 9.5-42

System: EGS

Title of Change: Reroute of the Governor Oil Cooling Line

Description of Change:

The cooling arrangement for the governor oil cooler was found to be inadequate to keep the governor's oil temperature to the vendor's recommended values. The cause of this condition was attributed to inadequate cooling water flow rate through the cooler. This simple design change improved this cooling arrangement by rerouting the return line from the governor oil cooler to a low pressure point of the jacket water system to increase the differential pressure and the flow rate.

Safety Evaluation Summary:

This change will maintain diesel generator reliability by providing proper cooling water flow to the governor oil cooler. In addition, a throttle valve will be added to the return line to obtain optimum governor cooler temperature. Operation of this valve will be controlled by procedure N2-OP-100A to maintain governor temperature between 120°F and 200°F.

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

Safety Evaluation No.: 95-038

Implementation Document No.: Calculation MS-4361

USAR Affected Pages: 9.1-12, 9.1-37, 9.1-38

System: FHS

Title of Change: Clarification of Design Basis for Spent Fuel Pool Rack External Loading

Description of Change:

This change corrected the discrepancy between the design basis calculation, fuel handling procedures, and Section 9.1 of the USAR. The design basis calculation was revised to include the case for the fuel bundle dropping onto the spent fuel pool racks while being moved with the polar crane 1/2-ton hoist. Additionally, the calculation revision addressed the case of a fuel bundle and grapple being dropped over the spent fuel pool racks from a maximum height of 30 inches above the racks. Use of the 25-ton auxiliary hoist for transfer of new fuel bundles and the position of the fuel bundle crate when opened now better describe the actual new fuel receipt activities. The required changes indicate the use of the 1/2-ton hoist for the transfer of new fuel bundles to either the new fuel storage vault or the new fuel inspection stand. Also, the new fuel bundle crate may be opened in the horizontal position provided that the crate still functions to support the fuel bundles.

Safety Evaluation Summary:

Clarification between the procedure and the design basis documents for the spent fuel storage racks' external loading was done in accordance with the design basis for the spent fuel pool racks as referenced in USAR Section 9.1. The correction of USAR Section 9.1 regarding new fuel receipt activities and associated procedures was done in accordance with the heavy load commitments per NUREG-0612, as referenced in USAR Appendix 9C.

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

Safety Evaluation No.: 95-043

Implementation Document No.: LDCR 2-95-UFS-038

USAR Affected Pages: 15.7-8 through 15.7-14

System: N/A

Title of Change: Update SAR Description of the Bundle Drop Accident

Description of Change:

The fuel handling accident analysis for the bundle drop accident in Section 15.7.4 of the Unit 2 USAR has been revised to incorporate several changes in the assumptions of the analysis. The accident occurs during a refueling operation when a fuel assembly is moved over the top of the core. While the fuel grapple is in the overhoist condition (bottom of the assembly 32.95 feet above the top of the core), a main hoist cable fails allowing the assembly, the fuel grapple mast, and head to fall on top of the core impacting a group of four assemblies. The grapple head and mast are fixed vertically to the dropped assembly such that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly are subjected to bending. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel.

Safety Evaluation Summary:

The number of failed fuel rods for the bundle drop accident is determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The dropped assembly is considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason it is assumed that all the rods in the dropped assembly fail. Therefore, the total number of failed rods on initial impact is 62+33=95. The assembly is assumed to tip over and impact horizontally on the top of the core. The energy from this second impact will result in 9 more failed rods. Consequently, the total number of failed rods from both impacts is determined to be 104. This compares with 124 failed rods from the analysis presented in the current USAR. Since the new analysis shows fewer failed rods, the radiological consequences are bounded by those of the previous analysis. The revised bundle drop accident methodology incorporates several conservative assumptions (i.e., including the weight of the fuel

Safety Evaluation No.: **95-043 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

assembly and the mast and assuming a greater drop height) while maintaining the radiological consequences of this accident within the limits of the current analysis in the USAR. This methodology is the standard methodology used by General Electric (GE) for the licensing of all new fuel types and is included in GE's Standard Application for Reactor Fuel (GESTAR-II).

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

Safety Evaluation No.: 95-049 Rev. 0 & 1

Implementation Document No.: Procedure N2-OP-52

USAR Affected Pages: 6.2-66, 6.2-68, 6.2-69; Figure 6.2-77

System: Standby Gas Treatment (GTS)

Title of Change: 1-Hour Drawdown Analysis

Description of Change:

This safety evaluation addresses the revision to the drawdown analysis and procedural changes to reflect the analysis. The revised drawdown analysis removed existing conservatism (reduce the spent fuel pool heat load, and reduce allowable 2HVR*UC413A, B degradation from 40% to 20%) for the following six improvements:

1. Reduce ΔT requirement and eliminate periodic ΔT monitoring
2. Eliminate the normal lighting trip upon LOCA signal
3. Use only one of the two ECCS pump room unit coolers
4. Eliminate ΔT penalty curve for general area unit coolers
5. Restore use of electrical heaters in the ECCS cubicles and allow heating in secondary containment based on specific engineering evaluation
6. Increase the GTS and service water (SWP) systems initiation time following LOCA

These changes were made to improve plant flexibility and ease of maintenance work. The revised drawdown parameters are as follows:

- The longest calculated drawdown time is 57 minutes.
- Emergency unit coolers 2HVR*UC413A, B degradation can be as high as 20%.
- The GTS and SWP/unit coolers system initiation time following a loss-of-coolant accident (LOCA) can be as high as 60 and 90 seconds, respectively, from drawdown considerations only.

Safety Evaluation Summary:

The consequences of these changes have been evaluated against the current requirements. It is concluded that the 1-hour drawdown time requirement is not impacted.

Safety Evaluation No.: 95-049 Rev. 0 & 1 (cont'd.)

Safety Evaluation Summary: (cont'd.)

These improvements do not impact GTS/HVR systems capacity to restore and maintain required vacuum in the secondary containment following a LOCA.

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

Safety Evaluation No.: 95-050

Implementation Document No.: N/A

USAR Affected Pages: N/A

System: High-Pressure Core Spray (CSH)

Title of Change: Temporary Enclosure for CSH Strainer

Description of Change:

The high-pressure core spray (HPCS) system takes water from the suppression pool through suction strainer 2CSH*STR1, penetration Z-12, and suction valve 2CSH*MOV118. Penetration Z-12 and suction valve 2CSH*MOV118 are at elevation 194'-11 15/16" and suction strainer 2CSH*STR1 is at elevation 189'-8". With minimum suppression pool water level at elevation 199'-6", maintenance/repair work on 2CSH*MOV118 cannot be performed without isolating the suppression pool.

This safety evaluation was issued to address the installation of a temporary enclosure on the suction strainer, 2CSH*STR1, in order to support subsequent repair and maintenance work to be performed on 2CSH*MOV118. The enclosure on the strainer will ensure sufficient isolation of the suppression pool from suction valve 2CSH*MOV118.

Safety Evaluation Summary:

The enclosure will be utilized only for repair and maintenance activities on 2CSH*MOV118. Administrative controls per the work order shall be in place as part of the maintenance work package, which will not allow for work to be done on 2CSH*MOV118 if the enclosure leaks.

The plant will be in Mode 5 with the reactor vessel head removed, the cavity flooded, the spent fuel pool gates removed, and the water level maintained within the limits of Technical Specifications 3.9.8 and 3.9.9. Therefore, HPCS will not be required to be operational per Technical Specification 3.5.2.

The suppression pool is not required to be operable during this activity per Technical Specification 3.5.3. However, suppression pool level will be maintained between elevation 199'-6" and 201'-0" to ensure adequate net positive suction head for emergency core cooling system pumps needed for Shutdown Safety Criteria N + 1.

Safety Evaluation No.: **95-050 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

The valve pit wall elevation is equal to maximum suppression pool water level. Therefore, should the temporary enclosure fail, the valve pit will contain leakage from the suppression pool. Based on the size of the valve pit, the total leakage from the suppression pool would amount to approximately 5,723 gallons. This would result in lowering the suppression pool water level by approximately 1-1/2". Therefore, the availability of the suppression pool for water inventory control will not be affected.

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

Safety Evaluation No.: 95-051 Rev. 0 & 1

Implementation Document No.: Mod. PN2Y94MX013

USAR Affected Pages: 8.3-15; Tables 3.9A-12 Sh 8, 6.2-56 Sh 2, 8.3-1 Sh 17 & 20, 8.3-2 Sh 16, 17, 20, 8.3-4 Sh 8, 15, 8.3-5 Sh 1, 2, 3, 4, 8.3-6 Sh 2, 3, 4

System: Residual Heat Removal (RHS)

Title of Change: New Limitorque Actuators for 2RHS*MOV15A/B and MOV25A/B

Description of Change:

Based on revised sizing calculations due to changes made to the Unit 2 motor-operated valve (MOV) sizing calculation methodology, the motor output torque/thrust capability for containment spray isolation valves 2RHS*MOV15A/B and MOV25A/B under reduced voltage condition was not adequate to close the valves against the maximum expected differential pressure.

Containment spray injection valves 2RHS*MOV15A/B and MOV25A/B required replacement of their SMB-1-40 Limitorque motor/actuators with SMB-2-80 motor/actuators. The new motor/actuators are rated at 5.2 HP and 80 ft-lbs. The new motor/actuators meet valve operation test and evaluation system (VOTES) testing requirements per Generic Letter 89-10 and verify operation under design bases conditions.

Safety Evaluation Summary:

Replacement of the Limitorque motor/actuators for valves 2RHS*MOV15A/B and MOV25A/B with larger size motor/actuators will provide an acceptable torque switch setting thrust range to allow the valve to operate as intended during design basis conditions. This new range will also accommodate the use of the VOTES diagnostic test equipment and allow for actuator degradation.

Qualification for the new Limitorque motor/actuators has been performed to ensure continued structural integrity and operability of the modified valve assembly.

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

Safety Evaluation No.: 95-053

Implementation Document No.: Procedure N2-EMP-GEN-660

USAR Affected Pages: 8.3-74, 8.3-81; Table 1.8-1 Sh 63

System: BYS

Title of Change: Change of IEEE-Std 484 Year of Issue

Description of Change:

Unit 2 replaced the Div. I safety-related dc battery during Refuel Outage 3. The Div. II battery was replaced during Refuel Outage 4. Unit 2 is committed to comply with IEEE-Std 484-1975, "IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations," for new battery installation. Since the time of the installation, the standard has been revised several times. The 1987 issue is now in effect. According to IEEE, the latest issue of the standard reflects the current state of the art and is recommended for use. The criteria provided in the 1987 issue of the standard generally encompass or exceed the criteria of the 1975 issue. The new criteria will increase safety during installation and testing and reduce the installation time. The 1987 issue provides a wider range of acceptance criterion for the intercell connectors resistance that may facilitate installation and testing. This change has no impact on battery characteristics or performance. Unit 2 dc system design criterion is to maintain 105 V dc minimum at the battery terminals regardless of the intercell connection resistance. This criterion is satisfied.

Safety Evaluation Summary:

The analysis performed revealed that the new resistance criteria for intercell connections does not compromise the ability of the battery to perform the safety-related function as designed and as described in the USAR. Engineering calculation performed for the most loaded battery determined that the impact of the new resistance criterion on the total battery voltage during discharge cycle is negligible. The battery's capacity, short circuit capability, and heat release are not affected either. Technical Specification operability criteria and surveillance requirements are also satisfied.

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

Safety Evaluation No.: 95-055 Rev. 0 & 1

Implementation Document No.: Simple Design Change SC2-0107-94

USAR Affected Pages: 9.2-14; Figure 9.2-3C;

System: Reactor Building Closed Loop Cooling Water (CCP)

Title of Change: Alternate Drywell Cooling

Description of Change:

In order to provide an alternate drywell cooling system to be used during outages, this simple design change added two permanent changes:

- Two piping penetrations through the southeast quadrant of the Reactor Building wall
- New 4" hose connection on the CCP supply and return headers

During outages, a chiller (located in the yard) will be connected to the Reactor Building penetrations. Hoses will be routed from the Reactor Building penetrations through emergency air lock to the CCP connections in the drywell.

Safety Evaluation Summary:

The permanent changes are designed in accordance with design criteria for CCP. The Reactor Building penetrations are designed to ASME III NC-3600 requirements and include redundant spring-loaded check valves/blind flanges to assure that secondary containment integrity is maintained when alternate drywell cooling is operating/secured. The hoses will be routed so as to prevent physical interaction with safety-related items in the event of connector failure. All potentially affected essential equipment or systems are designed for flood or spray.

The implementation of this change will ensure that drywell temperature is controlled during an outage such that personnel stay times are maximized.

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

Safety Evaluation No.: 95-056

Implementation Document No.: Simple Design Change SC2-0114-94

USAR Affected Pages: Table 6.2-56, Sh 4

System: High-Pressure Core Spray (CSH)

Title of Change: Replace Valve Stem, Disc and Operator Gear Set for 2CSH*MOV105

Description of Change:

This simple design change changed the operator gear set and replaced the valve stem and disc for safety-related motor-operated valve (MOV) 2CSH*MOV105. The new gear set will increase the actuator output capacity under reduced voltage conditions and the new stem and disc will provide higher ASME allowable stresses. These changes, in turn, will increase the thrust window to accommodate diagnostic test equipment for torque switch setting as required by the Generic Letter 89-10 program. As a result of the actuator gear set change, the valve closure time will be increased.

Safety Evaluation Summary:

An engineering evaluation of the proposed change concluded that the replacement of the stem and disc with a higher ASME allowable and an increase in the stroke time due to the replacement gear set for the subject valve has no impact on the containment isolation requirements and the high-pressure core spray system operation as described in the USAR. The higher stroke time is still within the design basis of the system requirements. The MOV as well as the system will perform its intended safety function during and after an accident.

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

Safety Evaluation No.: 95-057

Implementation Document No.: Simple Design Change SC2-0095-94

USAR Affected Pages: 3B-2, 3B-3, 3B-5, 3B-6, 5.4-24; Table 6.2-56 Sh 5

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Actuator Gear Set Changes for 2ICS*MOV121 and 2ICS*MOV128

Description of Change:

This change replaced the actuator gear sets for the subject valves in order to provide a sufficient thrust window for the valve operation test and evaluation system (VOTES) diagnostic equipment. As a result of the gear set change, the stroke time for these valves will increase from 15 seconds to 30 seconds.

Safety Evaluation Summary:

An engineering review of the requested change, which includes the effects of the change on the system's operability, reliability, maintainability, structural integrity and system interactions, has found that the implementation of this change will have no change on the safety or operability of the ICS system.

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

Safety Evaluation No.:	95-059 Rev. 0 & 1
Implementation Document No.:	Procedures N2-TSP-CNT-@001, N2-TSP-CNT-@003, N2-TDP-IIT-0201, N2-TTP-CNT-001
USAR Affected Pages:	6.2-104; Figures 6.2-71a, 6.2-71b, 6.2-73a
System:	N/A
Title of Change:	Primary Containment Integrated Leak Rate Test

Description of Change:

The procedures for the integrated leak rate test were revised for the Type A test to be performed in Refueling Outage 4 (RFO4). The changes are as follows:

Item 1 (Rev. 0)

To allow the use of ANSI/ANS-56.8-1987, Containment System Leakage Testing Requirements, for the "Mass Point" method. NUREG-1047, Safety Evaluation Report related to the operation of Nine Mile Point Nuclear Station, Unit No. 2, Section 6.2.6, states the Type A test data will be analyzed using the "Mass Point" method in ANSI/ANS-56.8-1981. The "Mass Point" method will still be used to analyze the Type A test data, but a more current revision to ANSI/ANS-56.8 was used. The reason for the change is that 10CFR50 Appendix J was revised in 1988 to accept the "Mass Point" method described in ANSI/ANS-56.8-1987 but Nine Mile Point did not update the licensing base to reflect the change in 10CFR50 Appendix J.

Item 2 (Rev. 0)

To allow the installation and use of temporary instrumentation to monitor drywell parameters during the Type A test. USAR Section 6.2.6 states that two independent quartz digital-type absolute pressure manometers are connected to the leakage monitoring system (LMS) to monitor primary containment pressure during the Type A test. USAR Section 6.2.6 also states that 18 temperature elements and 6 humidity analyzers are provided in the containment atmosphere monitoring system (CMS) to monitor dry-bulb and dewpoint temperatures, respectively. The temporary instrumentation will be placed in the same locations as the permanent plant equipment.

Safety Evaluation No.: 95-059 Rev. 0 & 1 (cont'd.)

Description of Change: (cont'd.)

The reason for the change is that the instrumentation provides the test data for the Type A test. Advances in electronic technology have resulted in more reliable and reduced installation times over conventional instrumentation. The result is reduced costs in man-hours and man-rem during the installation and removal phases.

Item 3 (Rev. 1)

Procedure N2-TSP-CNT-@001 is being revised to allow the installation of temporary depressurization flanges on 2-CPS-014-9-4 and piping penetration 2PCB*Z74. These flanges will be used during the Type A test to reduce primary containment pressure. The reason for the change is to allow a safe and controlled depressurization of the primary containment.

Safety Evaluation Summary:

This safety evaluation has concluded that an unreviewed safety question does not exist as a result of evaluating a 24-hour Type A leakage rate in accordance with the "Mass Point" method described in ANSI/ANS-56.8-1987. ANSI/ANS-56.8-1987 provides recommendations for the Type A test instrumentation. These recommendations include calibration requirements, in-situ checks, and minimum quantities and loss criteria. N2-TTP-CNT-001 and N2-TSP-CNT-@001 were written to ensure that the recommendations of ANSI/ANS-56.8-1987 are met. Also, the temporary instrumentation will be placed in the corresponding locations of the permanent plant equipment. Therefore, an unreviewed safety question does not exist as a result of using temporary instrumentation to monitor primary containment parameters during the Type A test. The temporary flanges will be installed only in Operational Conditions 4 or 5 and are bounded by the USAR load combinations and stress limits for pipes and pipe penetrations. Therefore, this safety evaluation has concluded that an unreviewed safety question does not exist as a result of connecting temporary flanges to 2-CPS-014-9-4 and 2PCB*Z74.

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

Safety Evaluation No.: 95-060

Implementation Document No.: Simple Design Change SC2-0028-95

USAR Affected Pages: 10.4-30; Figure 8.3-1

System: Feedwater (FWS)

Title of Change: Feedwater Pump Motors HP Upgrade

Description of Change:

On January 17, 1995, feedwater pump motor 2FWS-M1B tripped while running at approximately full power. The trip occurred due to the action of the motor relay protection. The investigation of the event revealed that the insulation of the stator winding of the motor failed causing the action of the relay protection and motor trip. The motor was sent to Monarch Electric Service Co. for repair. Root cause evaluation performed by Monarch Co. identified that the motor insulation failure occurred due to corona erosion of ground wall insulation. The motor was rewound and returned to Unit 2. In the process of repair, a new type of insulation was used and the HP rating of the motor was increased from 12,000 to 14,100.

Since the motors 2FWS-M1A and 2FWS-M1C may also be susceptible to the same mode of failure, the decision was made to rewind these motors and upgrade the HP rating.

Safety Evaluation Summary:

The upgrading of HP of the feedwater pump motors satisfies functional requirements of the system. The performance of the pumps is not affected. The system and components will perform as designed and as described in the USAR. The upgraded HP of the motors is adequate for the power uprate of the plant.

The upgraded HP of the motors does not adversely affect the mechanical interface systems. According to NMPC Mechanical Engineering evaluation, the maximum HP requirement for the pump for power uprate condition is 13,190 HP. Therefore, the upgraded 14,100 HP is adequate.

The electrical equipment such as cables, circuit breakers, current transformers, and relays ratings were evaluated for the upgraded HP of the motors and were found to be adequate.

Safety Evaluation No.: **95-060 (cont'd.)**

Safety Evaluation Summary: (cont'd.)

The feedwater pumps are nonsafety-related components and are not required for safe shutdown of the plant. The upgraded HP of the motors has no impact on safety-related systems and components.

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

Safety Evaluation No.:	95-062
Implementation Document No.:	Simple Design Change SC2-0029-95
USAR Affected Pages:	Appendix 9C Table 3-2
System:	MHR
Title of Change:	Alternate Means for the SRV Removal (Replacing 2MHR-CRN200 With 2MHR-CRN200A and 2MHR-CRN200B)

Description of Change:

The purpose of this simple design change is to provide an interchangeable handling system for the removal and replacement of the safety relief valves (SRVs). Hoists 2MHR-CRN200A and 2MHR-CRN200B will replace crane 2MHR-CRN200 for the SRV removal activity. The SRVs were originally shipped to the site in the horizontal position and could be handled with a single 4-ton hoist. The SRVs now arrive in the vertical position, requiring a second hoist to safely remove the valves from their shipping container and reposition the valve into the horizontal position. Upon completion of the removal and replacement of the SRVs, hoists 2MHR-CRN200A and 2MHR-CRN200B can be removed from the monorail and crane 2MHR-CRN200 can be reinstalled.

Safety Evaluation Summary:

The improvement being made by this simple design change with the use of alternate hoists 2MHR-CRN200A and 2MHR-CRN200B in the place of crane 2MHR-CRN200 meets the requirements of the seismic evaluation of nonsafety-related components in safety-related areas and does not affect the safety and reliability of Unit 2. There is no safety-related equipment that would be affected by a load drop involving hoists 2MHR-CRN200A or 2MHR-CRN200B.

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

Safety Evaluation No.:	95-064
Implementation Document No.:	Calculations EC-044 Rev. 11, EC-045, Rev. 7, EC-046 Rev. 5, EC-097 Rev. 2
USAR Affected Pages:	8.3-76; Tables 8.3-11 through 8.3-15
System:	BYS, BWS
Title of Change:	Nonsafety-Related Batteries Load Profile Update

Description of Change:

The battery sizing calculations were updated to reflect as-built dc loads of the nonsafety-related system, and to account for the plant modification which was implemented without revising these calculations.

Safety Evaluation Summary:

The revised battery sizing calculations conclude that the change of the dc loads is within the capabilities of the batteries and the chargers. The nonsafety-related dc system will continue to perform as designed and as described in the USAR, with the updated loads each battery is still capable of performing its duty cycle following the loss of charger while fully charged at 65°F, and with capacity deteriorated to 80 percent. Each battery can start and operate all required loads for the duration of the discharge cycle according to the battery load profile without battery terminal voltage falling below 105 V for 125 V dc system, and 21 V for 24 V dc system.

Each battery charger can still supply the continuous updated load on the battery while recharging the battery from the designed minimum charge state to the fully charged state in less than 24 hours.

The impact of this change on the plant response to station blackout event (SBO) has also been evaluated. Based on additional battery calculations performed for the new revised loads, the conclusion is made that nonsafety-related batteries still meet the 4-hour capability requirement as specified in NUMARC 87-00 and Regulatory Guide 1.155, and as demonstrated in the SBO study performed by General Electric.

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

Safety Evaluation No.: 95-066

Implementation Document No.: Temporary Mod. 95-011

USAR Affected Pages: N/A

System: Reactor Core Isolation Cooling (ICS)

Title of Change: Manual Operation of 2ICS*AOV130

Description of Change:

The ICS system is designed to assure sufficient reactor water inventory is maintained in the reactor vessel to permit adequate reactor core cooling. This temporary modification manually opened valve 2ICS*AOV130 and maintained it in the open position until the next system outage because the valve actuator is not capable of keeping the valve open due to diaphragm failure. This valve is one of the two normally open valves in series on the drain pot drain line of ICS en-route to the Reactor Building equipment drain system (DER).

Safety Evaluation Summary:

Based on a review of the system design bases and configuration, there is no specific reason in the USAR for the double isolation arrangement. A system design review by General Electric determined that double isolation arrangement was intended to provide redundancy of the drain valve closure during ICS operation. This redundancy was to minimize the spread of contamination and radiation release in the Reactor Building in case of high radiation levels in the steam supply line to the ICS turbine. Based on the reviews performed, it has been determined that this isolation capability can still be maintained via a single valve with no impact on nuclear safety.

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

Safety Evaluation No.: 95-068

Implementation Document No.: Procedure N2-MPM-GEN-R901

USAR Affected Pages: 9.1-39, 9.1-41

System: FHP

Title of Change: Revision to Fuel Pool Gate Removal Process in USAR Section 9.1

Description of Change:

This change revised the USAR to indicate the option to remove both the inner and the outer spent fuel pool gates after completion of flood-up activities prior to refueling, as described in procedure N2-MPM-GEN-R901, Rev. 1.

Safety Evaluation Summary:

The revision of the USAR to indicate the option to remove both the inner and the outer spent fuel pool gates after the completion of flood-up activities results in a more conservative plant configuration during reactor vessel disassembly activities. This was performed in accordance with the Guidelines for the Control of Heavy Loads (NUREG-0612) as described in USAR Appendix 9C.

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

Safety Evaluation No.: 95-069

Implementation Document No.: N/A

USAR Affected Pages: N/A

System: 345-kV Transmission Output, 115-kV Offsite Power Sources

Title of Change: Scriba Station, 345-kV B Bus Connection

Description of Change:

A sixth 345-kV transmission line was added to Scriba Station and connected to the 345-kV A bus in June 1994. Scriba Station is a 345-kV breaker and 1/2 station with an A and a B bus. The new transmission line was connected to the 345-kV B bus.

Safety Evaluation Summary:

The plant will be shut down for refuel during the period when the work will take place. All applicable Technical Specification requirements will be met. The work and the schedule have been reviewed for safe shutdown criteria.

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

Safety Evaluation No.: 95-071

Implementation Document No.: N/A

USAR Affected Pages: A.0-1, A.4.3-1, A.4.4-3, A.5.2-1, A.5.2-2, A.5.2-4, A.6-1, A.6-2, A.15.0-2, A.15.0-7, A.15.1-4, A.15.1-9, A.15.2-5, A.15.2-12, A.15.4-9, A.15B-1, A.15D-1; Tables A.5.2-1, A.5.2-2, A.6-2, A.15.0-4 Sh 1, 2, 3

System: Various

Title of Change: Operation of NMP2 Reload 4/Cycle 5

Description of Change:

This change added new fuel bundles and established a new core loading pattern for Reload 4/Cycle 5 operation of Unit 2. Two hundred forty-eight (248) new fuel bundles of the GE11 design were loaded. Also, 32 twice-burned GE6B bundles that were discharged at the end of Reload 1/Cycle 2 were re-inserted. All 124 of the GE6B bundles from Reload 3/Cycle 4, and 156 of 196 GE9B bundles (P8CWB299), were discharged to the spent fuel pool. Various evaluations and analyses were performed to establish appropriate operating limits for the reload core. These cycle-specific limits were documented in the Core Operating Limits Report.

Safety Evaluation Summary:

The reload analyses and evaluations are performed based on the General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-10 and NEDE 24011-P-A-10-US (GESTAR II). This document describes the fuel licensing acceptance criteria; the fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases; and the safety analysis methodology. For Reload 4, the evaluations included transients and accidents likely to limit operation because of minimum critical power ratio considerations; overpressurization events; loss-of-coolant accident; and stability analysis. Appropriate consideration of equipment out of service was included. Limits on plant operation were established to assure that applicable fuel and reactor coolant system safety limits are not exceeded.

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

Safety Evaluation No.: 95-077

Implementation Document No.: Calculations H21C-038-01C, H21C-043-00B, A10.1-E-130

USAR Affected Pages: 15.6-13; Tables 15.6-13 Sh 10 & 11, 15.6-16b

System: Residual Heat (RHS)

Title of Change: Revise/Delete the Leak Rate Acceptance Criteria and Test Frequency for RHS Valves 2RHS*MOV142, MOV149, SOV35A/B and SOV36A/B

Description of Change:

This change revised the leak rate acceptance criteria and test frequency for valves 2RHS*MOV142, MOV149, SOV35A/B, and SOV36A/B. The leakage acceptance criteria of less than or equal to 1 gpm times the number of hydrostatically tested valves was increased to 20 gpm for 2RHS*SOV35A/B and SOV36A/B, and to 10 gpm for 2RHS*MOV142 and MOV149 at normal system operating pressure. The test frequency was revised from once every 18 months to once every 2 years. Leak testing requirements for the valves remain in the IST testing program; however, changes to NIP-DES-04 (by revising the footnote "m") and supporting operations procedures were required to implement the new leakage criteria. Implementation of simple design change SC2-0046-95 to install ASME Class 2 reducers to replace the leakage control function of the solenoid-operated valves (SOVs) was determined to be an acceptable alternative to leak testing the SOVs.

Safety Evaluation Summary:

This safety evaluation has concluded that an unreviewed safety question does not result from the proposed change. This conclusion is based on the ability to demonstrate RHS system leakage boundary integrity by satisfying the functional requirements of the low-pressure coolant injection system with the increased leakage, and determining that the consequences of the increased leakage into secondary containment post-LOCA are radiologically acceptable.

Safety Evaluation No.: 95-078

Implementation Document No.: Procedure N2-FHP-021

USAR Affected Pages: N/A

System: FNR, FNS, GTS

Title of Change: Revision to Control Blade Movement Procedure

Description of Change:

This safety evaluation was written to revise procedure N2-FHP-021, "Control Rod Uncoupling, Removal, and Installation." This revision allows control rod uncoupling, removal, and installation without secondary containment integrity and SGTS operability, provided seven days have elapsed since reactor shutdown and all movements are of objects totalling less than 617 pounds (the estimated weight of a fuel bundle).

Safety Evaluation Summary:

The revised control blade movement procedure provides the same level of safety to the Control Room and public as was previously available. The change does not alter Technical Specifications, or guidance provided by the vendor. Radiological analysis has shown that the proposal allows Unit 2 to meet 10CFR100 limits and remain in compliance with the plant safety analysis report.

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

Safety Evaluation No.: 95-079

Implementation Document No.: DER 2-95-1183

USAR Affected Pages: 8.3-11, 8.3-12

System: VBB

Title of Change: Revise UPS 2VBB-UPS1C/1D Acceptable Voltage Output Criteria

Description of Change:

Uninterruptible power supplies (UPSs) 2VBB-UPS1C and 2VBB-UPS1D feed selected lighting and communications loads. The original UPS units installed during construction were purchased from Exide. In 1991, these units were replaced with units purchased from HDR under Modification PN2Y89MX042. During the replacement, it was discovered that these units did not meet the output voltage acceptance criteria of Specification E-147, i.e., ±2% of output voltage variation. Engineering evaluation of the deficiency was performed and units were accepted as supplied by the vendor with voltage output acceptable up to ±3% based on the type of loads these units are feeding. The engineering specification E-147 was revised to allow the new acceptance criteria for UPSs 2VBB-UPS1C and 1D.

Safety Evaluation Summary:

The proposed change of accepting the ±3% of output voltage variation does not affect the performance of the connected loads. The purpose of specifying precise output voltage regulation for the UPSs is to meet the requirements of the precision instrumentation and control equipment which they feed. Most of this equipment will require power supply voltage variation not to exceed 2%. The two UPSs involved in this change, 2VBB-UPS1C and 2VBB-UPS1D provide power supply only to the essential lighting, egress lighting, and page party/public address (PP/PA) communication system loads. The essential and egress lighting system equipment are designed for ±10% supply voltage variation. The PP/PA communication system equipment are designed for 90-140 V as shown in Gaitronic specification. Therefore, an output voltage variation of ±3% for 2VBB-UPS1C and 2VBB-UPS1D will not adversely affect operation of any of their connected equipment and is acceptable.

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

Safety Evaluation No.: 95-080 Rev. 0 & 1

Implementation Document No.: Procedure N2-PM-S012

USAR Affected Pages: 3C-25, 3C-28, 3C-29

System: N/A

Title of Change: Change the Visual Leak/Flood Detection Walkdowns to a Minimum of Once per Calendar Day

Description of Change:

As depicted in the USAR, area walkdowns by plant personnel for visual leak/flood detection were performed once every 8-hour shift. In the past, plant operations personnel were scheduled for three 8-hour shifts. Plant operations has revised their shift work schedules from three 8-hour shifts to two 12-hour shifts.

Calculations were revised to reflect the increase in water levels due to the change of the visual leak/flood detections from 12 hours to a minimum of once per calendar day, not to exceed a 24-hour time period (36-hour time period for the Control Building basement) between inspections.

Safety Evaluation Summary:

The revision to the USAR to change the time intervals of the visual flood/leak detection walkdowns from every 12½ hours to a minimum of once per calendar day, not to exceed a 24-hour time period (36-hour time period for the Control Building basement) between inspections, does not affect the safety and reliability of Unit 2. The loss of safety-related equipment due to flooding has already been evaluated in the USAR Appendix 3C Spray/Flooding Evaluation and will not be changed.

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