NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
NORTHEAST UTILITIES SERVICE COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

MEMO

February 28, 1994

Docket Nos. 50-213
50-245
50-336
50-423
B14760

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

Haddam Neck Plant
Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3
Annual Report

Pursuant to the provisions of 10CFR50.59; Sections 6.9.1.4 and 6.9.1.5 of Appendix A to DPR-21, DPR-61, and DPR-65; and Section 6.9.1.2 of Appendix A to NPF-49, this report is submitted covering operations for the period January 1, 1993, to December 31, 1993. Additionally, this report contains a summary of the challenges to relief/safety valves as committed in a letter dated June 10, 1980.[1]

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY
NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Executive Vice President

Enclosure

cc: See Page 2

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(1) W. G. Counsil letter to D. G. Eisenhut, "Five Additional TMI-2 Related Requirements to Operating Reactors," dated June 10, 1980.

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Director, Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
Attention: REIRS Project Manager

HADDAM NECK

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INTRODUCTION

None of the plant design changes, procedure changes, jumpers-lifted leads-bypasses, setpoint changes, tests, or experiments described herein constitute (or constituted) an unreviewed safety question per the criteria of 10CFR50.59.

PLANT DESIGN CHANGES

PDCR Number	Title
1137	Connecticut Yankee NaOCl System Modifications: Installation of a Jet Pump for Service Water System Chlorination
1248	Connecticut Yankee Containment Isolation Valve Replacement CH-TV-240 (241)
1285	Terry Turbine Building (TTB) Ventilation Modification
1305	Repower Battery Charger BC-1-1A
1317	Separation Modifications
1320	Cycle 17 Refueling Outage Appendix J Modifications
1321	NG-SOV-470 Replacement
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PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
1434	MCC-5 Automatic Bus Transfer Re-design

Plant Design Change Record Number 1137

This change entitled, "Connecticut Yankee NaOCl System Modifications : Installation of a Jet Pump for Service Water System Chlorination," is complete.

Description of Change

The modification installed a jet pump in service water system chlorination line 1"-CHY-155C-29, located in the Screenwell Building.

Reason for Change

Installation of the jet pump ensures a positive driving force using service water to ensure a continuous flow of NaOCl to the service water pump suction bells with minimal operator adjustment. There were problems associated with the continuity of the gravity feed flow and the system requires regular attention to ensure proper hypochlorite flow rates.

Safety Evaluation

The potential for creating a new chemical spray hazard in a safety related building has been resolved as acceptable since no safety related equipment exists in the potential spray area. The jet pump installation improves the service water chlorination system and minimizes nuclear safety related heat exchanger heat transfer surface fouling. The improved chlorination also minimizes the potential for piping cloggage due to macro and micro-fouling. Therefore, this modification improves overall plant nuclear safety.

The physical protective boundaries do not exceed their acceptance limits so the Margin of Safety is not reduced. There is no effect on the probability of occurrence of a previously evaluated accident or of a previously evaluated malfunction of equipment important to safety. The consequences of a previously evaluated event is not adversely affected by this modification.

Plant Design Change Record Number 1248

This change, entitled "Connecticut Yankee Containment Isolation Valve Replacement CH-TV-240 (241)," is complete.

Description of Change

This modification replaced Reactor Coolant Pump Seal Water return containment isolation valves CH-TV-240 and CH-TV-241. The respective solenoid valves CH-SOV-240 and CH-SOV-241 and limit switches 33-TV-240U & L and 33-TV-241U & L were also replaced.

Reason for Change

The previously installed Contromatics ball valves had a soft seat that was determined to be operable for Cycle 17. However, they did not meet the design basis for containment. The design change eliminated the soft seat material from valves CH-TV-240 and CH-TV-241. The new butterfly valves have a metal to metal seating surface.

Safety Evaluation

The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. The isolation function, control logic, and failure modes of the replacement valve are identical to the previously installed valve. Also, the probability of a malfunction of the stainless steel seating surface is less than for the soft tefzel material.

Plant Design Change Record Number 1285

This change, entitled "Terry Turbine Building (TTB) Ventilation Modification," is complete.

Description of Change

This modification added four louvers into the TTB north and south walls and doors and four penthouses with throat mounted dampers into the TTB roof. Two local temperature gages were installed in the vicinity of the Auxiliary Feedwater (AFW) pumps/turbines.

Reason for Change

The addition of the louvers, dampers and penthouses will provide natural circulation in the TTB to prevent excessive ambient conditions during AFW pump/turbine operation. The temperature gages provide indication for opening and closing the dampers and louvers.

Safety Evaluation

The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. The louver, damper and penthouse designs are simple and seismically rugged and do not represent a seismic concern since they are adequately anchored. The components were not added to a safety related system and are manually operated and designed such that inadvertent opening and closing is not credible. The Margin of Safety is not affected by these modifications.

Plant Design Change Record Number 1305

This change, entitled "Repower Battery Charger BC-1-1A," is complete.

Description of Change

The 480VAC power feed to battery charger BC-1-1A was repowered from motor control center MCC-5 to Train "A" related MCC-13. A new cable run was installed. A short section of conduit was installed from MCC-13 to an existing cable tray.

Reason for Change

Since the battery charger BC-1-1A is a Train "A" related load only, it was repowered from Class 1E Train "A" 480V MCC-13, thus decreasing the modeled load on diesel generator "B" in the load calculations. Additionally, this modification was performed because a reduction in the Diesel Generator "B" load profile will support a higher combustion air inlet temperature to the diesel generator cubicle.

Safety Evaluation

The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. New tested equipment was used in the repowering and applied within its design margins. All the raceway and equipment involved is seismically qualified and maintained in a Quality Assurance manner.

Plant Design Change Record Number 1317

This change, entitled "Separation Modifications," is complete.

Description of Change

This change repowered the following : RH-MOV-33B, the main lube oil pump P-149-1A for charging pump "A," PR-MOV-569, PR-AOV-570, and the 480VAC distribution panel EGG-2A.

Reason for Change

RH-MOV-33B, the main lube oil pump P-149-1A, and PR-MOV-569 were repowered as a result of Northeast Utilities Services Company Probabilistic Risk Assessment Document "Electrical Separation Study: Connecticut Yankee ISAP Topic #1.64 'System Dependencies on Motor Control Center 5.'"

PR-AOV-570 was repowered as a result of NUSCO Letter GIC-91-113, "Potential Nuclear Safety Concern Regarding Pressurizer Overpressure (NUREG 1218)."

480VAC Distribution Panel EGG-2A was repowered from MCC-5 (Train "A" or Train "B" related) to MCC-13 (Train "A" related only) to decrease the modeled loading on MCC-5 and Diesel Generator "B." This reduction in the Diesel Generator "B" load profile supports a higher combustion air inlet temperature to the diesel generator cubicle, per NUSCO memo PSCY-92-362 "CY Emergency Diesel Generator Ventilation Operability Limitations," dated July 24, 1992.

Safety Evaluation

Repowering the loads from redundant power supplies does not increase the probability of the power supply malfunctions or occurrence of an accident. The modifications do not degrade affected systems performance, alter any assumptions made in the accident analyses or degrade any fission product barriers. Thus, there will be no adverse impact on the consequences of an accident.

There is a decrease in the consequences of a loss of MCC-5 or Semi Vital AC since the equipment failure is no longer a single failure of both trains of a system. The equipment loads which were repowered have redundant counterparts which remain powered by the original source.

The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

Plant Design Change Record Number 1320

This change, entitled "Cycle 17 Refueling Outage Appendix J Modifications," is complete.

Description of Change

This design change modified ten containment penetrations in order to permit local leak rate testing using air or nitrogen as a test medium. The modifications are grouped into several basic categories. These are: (a) add test boundary valve, (b) add test connection(s), (c) add testable blank flanges and (d) upgrade a test valve inside containment to containment isolation valve. More than one type of modification was made to several of the penetrations.

Reason for Change

The modifications permitted local leak rate testing in accordance with the requirements of 10CFR50 Appendix J.

Safety Evaluation

The modifications to these penetrations consisted of adding a manual test boundary valve, adding a test connection to sections of piping inside of containment, sealing penetrations with a testable blank flange immediately outside of containment, and replacing a check valve with a new Quality Assurance valve to serve as a containment isolation valve. The primary safety function for these penetrations is to isolate and remain leak tight during an accident which releases radioactivity inside of containment. These modifications enhanced safety by enabling the penetrations to be tested in a manner consistent with 10CFR50 Appendix J requirements.

The plant changes do not increase the probability of occurrence of an accident, increase the probability of occurrence of a malfunction of equipment, or increase the consequences of an accident evaluated previously in the safety analysis report. No other plant equipment was affected by the modifications.

Plant Design Change Record Number 1321

This change, entitled "NG-SOV-470 Replacement," is complete.

Description of Change

The Target Rock solenoid operated containment isolation valve NG-SOV-470 was removed and replaced with a new solenoid operated valve manufactured by Valcor. The Valcor valve provides the same functions as the previously installed valve but was constructed with a bolted bonnet instead of a welded bonnet to allow for ease of maintenance.

Reason for Change

Containment isolation valve NG-SOV-470 failed its local leak rate test, probably due to normal wear and tear, and repeated attempts to repair the leakage failed. During the repair process, the seal weld was removed and the valve re-welded, causing the valve to warp. The Valcor valve provides the same functions as the Target Rock valve except that the new valve has a bolted body to bonnet instead of a seal weld. This new design prevents damage due to any repairs due to normal wear and tear on the valve.

Safety Evaluation

The replacement of the Target Rock solenoid operated valve with the Valcor valve does not impact previously evaluated accidents. The plant change does not increase the probability of occurrence of an accident, increase the probability of occurrence of a malfunction of equipment, or increase the consequences of an accident evaluated previously in the safety analysis report.

The Valcor valve performs the same functions as the previously installed valve and is designed to ANSI B31.1. The valve meets the design, materials, fabrication, and non-destructive examination requirements of ASME Section III, Class 2, and was manufactured in accordance with the requirements of 10CFR50 Appendix B. The materials of construction and the design of the valve are consistent with the design requirements for the pressurizer relief tank nitrogen gas lines set forth in the Haddam Neck Piping Specification CYS-1550, "Shop Fabricated Nuclear Piping," revised July 21,1965.

Plant Design Change Record Number 1331

This change, entitled "Modernize Feedwater Controls," is complete.

Description of Change

This modification continued the Connecticut Yankee instrumentation modernization effort by replacing the obsolete Hagan Feedwater Control System instrumentation with a state of the art digital upgrade. The safety related portions of this upgrade included complete loop replacements consisting of transmitters, reactor trip alarm bistables, and reactor trip coincidence logic circuitry. This change utilized Foxboro Spec 200 analog equipment for much of the loop and Foxboro Spec 200 MICRO equipment for the decision making functions. The safety related functions performed by the digital equipment are limited to algebraic signal conditioning, alarm, and gate logic functions which are contained in the programmable read only memory and cannot be changed.

Reason for Change

The modifications replaced obsolete instrumentation and increased plant design adherence with present regulatory standards.

Safety Evaluation

The malfunctions that were evaluated included the failure of the new inverters and their associated static switch, and the new regulating transformers. These failures were evaluated from the standpoint of loss of Vital AC supply power, since that is the sole safety related function of the units. The new Vital AC power supplies replaced the equipment that was becoming obsolete with new and advanced equipment. By virtue of their newness and advanced design, the reliability of supply of Vital AC power has been increased. The changes do not increase the probability of occurrence of an accident, increase the probability of occurrence of a malfunction of equipment, or increase the consequences of an accident previously evaluated.

Plant Design Change Record Number 1332

This change, entitled "Modification of Main Control Board Sections 'G' and 'H' for Addition of Emergency Diesel Generator (EDG) 2A and 2B kW and Voltage Indications and the Resolution of Human Engineering Discrepancies," is complete.

Description of Change

This change added EDG 2A and 2B voltage and kilowatt indication to the main control board section "G." Various indicators were relocated on panels "G" and "H" and eight selector switches for phase "A," "B," and "C" amperage indication were removed. These phase "A," "B," and "C" ammeters were installed on their respective switch gear cubicles. This modification required cutting and patching of the subject panels and replacement of the Bus 11 ammeter on panel "G." First-out annunciator panel H-2 was replaced with a physically identical panel which serves as a normal annunciator panel.

Reason for Change

The additions of Main Control Board display of EDG voltage and kilowatt indications are necessary to correct discrepancies identified during the performance of the CY Control Room Design Review as documented in ISAP Topic 1.19.

Safety Evaluation

These modifications do not alter the functional characteristics of the electrical distribution system. The new components are electrically isolated from the other circuitry in accordance with IEEE 384-1981, hence, no credible failure of these components can propagate to safety related portions of their associated circuits. The relocated equipment are maintained and tested on a regularly scheduled basis and have satisfactorily operated over the current life of the plant so there is no reason to believe them to be deficient for their intended function.

This modification enhances the operator's ability to view and interpret the information provided by the modified displays. In the course of an accident, the operator will be able to take corrective actions in less time.

Plant Design Change Record Number 1339

This change, entitled "Emergency Diesel Generator (EDG) Cubicle Ventilation Modifications," is complete.

Description of Change

This plant design change installed fixed louvers and backdraft dampers into the EDG intake duct. Twelve fixed bladed louvers were installed into the intake penthouse located on the roof of the EDG building. A backdraft damper was installed into each EDG cubicle intake duct and security grating located in the discharge of the EDG exhaust fans was removed.

Reason for Change

Testing during the summer of 1992 determined that the ventilation flows to each cubicle were less than design flows. The modifications were performed to increase the overall flows. The installed louvers increased the intake free area which increased the flow to each EDG cubicle. The dampers which provided winter protection were removed to increase these flows. The backdraft dampers were installed to provide smoke protection and prevent excessive heat loss from each cubicle during winter. The new security grating was installed such that the flow reduction due to the grating was lessened.

Safety Evaluation

The modifications do not increase the probability of occurrence of an accident, increase the probability of occurrence of a malfunction of equipment, or increase the consequences of an accident evaluated previously in the safety analysis.

The probability of failure of the backdraft dampers to open or close on demand is minimal. They are not motorized so a failure mode has been removed. The backdraft dampers open under flow conditions (i.e. fan or diesel operation) and close when the diesel and fan are secured. Testing will ensure proper operation of the damper and will measure actual ventilation flow to compare with design flows. Also, the dampers were purchased and installed to meet the seismic requirements of the EDG Building.

The louver blades are fixed in one position, thus, no failure mode is associated with them, and relocation of the security grating does not impact the operation of the plant.

Plant Design Change Record Number 1341

This change, entitled "Connecticut Yankee Containment Isolation Valve Replacements CC-TV-917 (920)," is complete.

Description of Change

The design change replaced Component Cooling Water containment isolation valves CC-TV-917 and CC-TV-920, as well as the respective solenoid valves CC-SOV-917 and CC-SOV-920 and limit switches.

Reason for Change

The previously installed Contromatics ball valves had a soft seat material made of reinforced teflon which was determined operable for Cycle 17. However, they did not meet the design basis for containment. The modification eliminated the soft seat material and replaced them with butterfly valves with a metal to metal seating surface.

Safety Evaluation

The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. The isolation function, control logic, and failure modes of the replacement valve are identical to the previously installed valve. Also, the probability of a malfunction of the stainless steel seating surface is less than for the soft teflon material.

Plant Design Change Record Number 1347

This change, entitled "Modification of Containment Air Recirculation (CAR) Fan Safety Injection (SI) Actuation Circuits," is complete.

Description of Change

This modification installed a five minute time delay relay in place of the 48 second time delay relay in the CAR Fan auto start on SI circuitry. A ten minute time delay relay was also installed on each train which will annunciate on the main control board ten minutes after SI actuation and only if less than four CAR Fans are running.

Reason for Change

This plant design change was initiated to change the Emergency Diesel Generator (EDG) loading profile to eliminate combustion air temperature as a factor on EDG loading. The modification helped the EDG loading profile without violating safety criteria and alleviated the operators of the responsibility to manually start the second CAR Fan on each train during Emergency Operations. The annunciators serve as reminders to the operator that verification of CAR Fan operation is required.

Safety Evaluation

The modification does not affect the circuitry involved with the start of the first CAR Fan on each train. Any failure of the auto start circuitry can be overridden by manual start of the CAR Fans. Verification of CAR Fan operation is an emergency operating procedure requirement. In addition, the four relays that are installed are seismically qualified to not fail to perform their safety function during a postulated seismic event.

The failure modes associated with the modifications cannot be an initiating event for design basis accidents, thus the changes do not increase the possibility of occurrence of these accidents. The consequences of a previously evaluated accident and the probability of malfunction of equipment is no greater with the modifications than with the original design.

Plant Design Change Record Number 1348

This change, entitled "Replace Station Service Transformers 389 and 399 with Automatic Load Tap Changing Transformers," is complete.

Description of Change

Two 115-4.16 kV station service transformers located in the Connecticut Yankee 12R 115 kV switchyard were replaced with automatic load tap changing transformers.

Reason for Change

Installation of the load tap changing transformers allows a constant voltage to be maintained with a wide fluctuation of input voltage or a change in system loading. This provides an economical savings to Northeast Utilities for not having to "must run" a Middletown Unit, and it negates the need to procure a spare transformer for the Haddam Neck Plant or a new one for the Mt. Tom Station.

Safety Evaluation

If the transformer output voltage is inadequate to supply safety related loads, alarms will annunciate, and should a safety injection signal then occur, the offsite supply will be tripped and the on-site Class 1E diesel generators will energize the accident loads. There is no increase in the probability of occurrence of previously evaluated accidents nor is there any adverse effect on the probability of occurrence of a previously evaluated malfunction of equipment important to safety. The replacement of the transformers does not change the function of the offsite or on-site power systems. The potential of a new unanalyzed accident has not been increased.

Plant Design Change Record Number 1373

This change, entitled "Main Steam/Feedwater (MS/FW) Pipe Bridge Systematic Evaluation Program (SEP) Modifications," is complete.

Description of Change

This design change modified four existing structural connections and added two new structural connections to the MS/FW Pipe Bridge.

Reason for Change

The modifications to the MS/FW Pipe Bridge were required as part of SEP Topic III-6, Seismic Design Considerations. The modifications to the Pipe Bridge decouple the Auxiliary Feedwater System Pumphouse (Terry Turbine Building) from the Turbine Building/Service Building and upgrade the Pipe Bridge for loads from a seismic event.

Safety Evaluation

The modifications do not affect the ability of any safety related structure, system, or component to perform its required safety function. Also, the structural integrity of the pipe bridge was maintained at all times. The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

Plant Design Change Record Number 1378

This change, entitled "Installation of a Back-up 20 Auto Stop Trip Turbine Trip Solenoid Valve," is complete.

Description of Change

The design change added a second 20/AST (Auto Stop Trip) Solenoid Valve ASCO 1 No. EFHCX8223G12 to the high pressure oil supply side of the test handle. The new solenoid valve also acts as a backup to the existing 20/AST during power operation.

Reason for Change

After a turbine at a nuclear power facility experienced an overspeed incident, Westinghouse provided an advisory memo to all owners of Westinghouse turbines. Westinghouse Customer Advisory Letter (CAL) 92-02 provided recommendations for operation, maintenance and testing of all turbine control system solenoids. The CAL recommended the installation of an additional 20/AST solenoid valve. This allows for electrical trip signals to be effective during testing of the mechanical turbine trip block.

Safety Evaluation

The design change insignificantly increases the probability of a trip of the turbine generator, which is not an "important to safety" component. Failure of the new 20/AST would not cause a new malfunction of a different type than previously analyzed. The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

Plant Design Change Record Number 1390

This change, entitled "Connecticut Yankee Steam Generator Tube/Plug Repairs," is complete.

Description of Change

This modification installed mechanical plugs in steam generator tubes which, after eddy current testing, exhibited defects that exceeded the limits of the Haddam Neck Plant Technical Specifications. It also installed mechanical plug retainers (PAPs) in existing cold leg Westinghouse ribbed plugs produced from Heat No. NX-3513.

Reason for Change

The plugs were installed to comply with the requirements of the Section 3/4.4.5, "Steam Generators," of the Haddam Neck Plant Technical Specifications.

Bulletin Number 89-01, "Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs," identifies certain heats of the Westinghouse 3/4" diameter ribbed plugs to be susceptible to stress corrosion cracking. To obtain complete assurance, all plugs from Heat No. NX-3513 were repaired.

Safety Evaluation

The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. The change does not constitute an unreviewed safety question because plugs and PAPs are designed, manufactured and tested to withstand the original design conditions, and because generic and exceptional failures are essentially precluded due to extensive review and control of each phase of the repair program.

Plant Design Change Record Number 1399

This change, entitled "Cycle 18 Reload," is complete.

Description of Change

The reactor core was reconfigured for Cycle 18 four loop operation. To reflect the Cycle 18 operation, the Technical Report Supporting Cycle Operation was modified.

Reason for Change

There was no reactivity left in the core following Cycle 17. This modification provided additional fuel to be added to allow for continued operation and to reconfigure the core for Cycle 18 operation.

Safety Evaluation

The fuel design and fuel handling equipment and procedures for the Cycle 18 fuel reloading are basically the same as that for the previous outage. Hence, the modifications do not increase the probability of occurrence or the consequences of an unanalyzed accident or malfunction of equipment important to safety.

The only accident affected by this modification is the boron dilution accident. The shutdown margin was increased to compensate for this accident and preserves the required operator action times to assure that the reactor will not go critical following this event. Therefore, the margin of safety is preserved.

Plant Design Change Record Number 1400

This change, entitled "Roll Expansion Repair of Steam Generator Tubes," is complete.

Description of Change

This design change repaired steam generator tubes by roll expansion of the tube above the existing hardroll.

Reason for Change

The tubes were repaired utilizing the roll expansion process to comply with the requirements of Section 3/4.4.5, "Steam Generators," of the Haddam Neck Plant Technical Specifications. The tubes that were repaired meet the plugging criteria of the Haddam Neck Plant Technical Specifications and may remain in service.

Safety Evaluation

The roll expansion repair restores original design basis margins to steam generator tubes and has no effect on the consequences of an accident or malfunction of equipment. The roll expansion repair does not create the possibility of a multiple tube rupture incident nor does it create any new equipment malfunction scenarios.

Plant Design Change Record Number 1434

This change, entitled "MCC-5 Automatic Bus Transfer Re-design," is complete.

Description of Change

This modification redesigned the 480V MCC-5 Automatic Bus Transfer (ABT) scheme. The major features of the new scheme are:

- 480V Bus 5 breaker 9C was selected to normally supply MCC-5.
- Upon a total loss of offsite power, MCC-5 will be energized from the first available source and remain aligned to it, unless the source is subsequently lost or operators take manual control to re-transfer.

Two breaker control switches and one control relay were installed.

Reason for Change

Testing of the scheme that was performed during the Cycle 17 refueling outage uncovered vulnerabilities and design deficiencies. The modified scheme is more reliable in that there are less challenges to breaker operations. The new scheme also reduces the overall core melt frequency and thus has a positive impact on corporate nuclear safety goals.

Safety Evaluation

The reliability of the ABT is improved significantly and several single failure modes associated with the previous design of the ABT have been eliminated. The design change does not have any adverse affect on the Margin of Safety.

The modifications result in a decrease in MCC-5 failure probability. The modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

PROCEDURE CHANGES

The following changes in procedures as described in the Updated Final Safety Analysis Report were made during 1993:

Procedure Number	Title
AOP 3.2-12, Revision 10	Loss of Residual Heat Removal System
AOP 3.2-31, Revision 7	Reactor Coolant System Leak (1, 2, 3, or 4)
AOP 3.2-59, Revision 0	Loss of Spent Fuel Pool Cooling
EOP 3.1-0 , Revision 19	Emergency Response Procedures
EOP 3.1-10, Revision 16	Partial Loss of AC
EOP 3.1-12, Revision 9	Emergency Boration
EOP 3.1-21, Revision 10	Pressurizer Spray Valve Malfunction
EOP 3.1-34, Revision 12	Complete Loss of Control Air Supply
EOP 3.1-46, Revision 12	Total Loss of Semi Vital Power
EOP 3.1-49, Revision 11	Partial Loss of DC
EOP 3.1-50, Revision 6	Loss of MCC-5
SPL 10.1-36, Revision 0	Installation, Testing and Use of a Temporary Fire Pump
SPL 10.1-37, Revision 0	Installation, Testing and Use of a Temporary Fire Pump
SPL 10.3-29, Revision 0	Changing Plant Load for Rod Position Indication System Testing and Turbine Control Valve Test
SPL 10.3-30, Revision 1	Control Rod Maneuvers for Rod Position Indication System Testing and Flux Mapping
SPL 10.3-31, Revision 0	Control Rod Maneuvers for Rod Position Indication System Data Collection Instrumentation Check-out
SUR 5.7-148B, Revision 3	Substantial Flow Test of "A," "B," "C," and "D" Service Water Pumps

Procedure Number | Title

AOP 3.2-12, Rev. 10 — Loss of Residual Heat Removal System

Description of Change

Major modifications were made to broaden the procedural guidance with respect to possible initial plant conditions and to reasons for loss of the Residual Heat Removal system.

Reason for Change

To provide guidance for accident mitigation and equipment usage.

Safety Evaluation

Since the procedure provides guidance for assuring adequate core cooling and establishing containment integrity when necessary, it assures that the margin of safety is not adversely impacted.

The mitigation strategies have been found to provide adequate core cooling. The review of the strategies has also found them to appropriately utilize equipment and prolong the need to take actions to assure long term core cooling.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number | Title

AOP 3.2-31, Rev. 7 | Reactor Coolant System Leak (1, 2, 3 or 4)

Description of Change

The change proposed the implementation of a new procedure, AOP 3.2-31A. This new procedure provides guidance for responding to excessive reactor coolant system or refueling cavity leakage. Concomitant with the implementation of this new procedure is the deletion of Modes 5 and 6 from those that AOP 3.2-31 is applicable.

Reason for Change

To provide guidance to the operators in identifying and isolating a leakage.

Safety Evaluation

The change proposed using a new procedure for reactor coolant system and/or cavity leakage during Modes 5 or 6 instead of AOP 3.2-31. The new procedure provides a guidance more appropriate for responding to leakage during Modes 5 or 6.

The change provides guidance to the operators. This refers the operators to the appropriate plant procedures for conditions where the leakage has the potential to result in a challenge to any protective boundary. Based on this, the change does not adversely impact the margin of safety.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
AOP 3.2-59, Rev. 0	Loss of Spent Fuel Pool Cooling

Description of Change

Implementation of a new plant procedure, Loss of Spent Fuel Pool Cooling. In conjunction with the implementation of the new plant procedure, sections pertaining to alternate spent fuel pool cooling are being deleted from NOP 2.10-1.

Reason for Change

To provide guidance to the operators for mitigating a total loss of spent fuel pool cooling.

Safety Evaluation

The procedure provides guidance for mitigating a loss of spent fuel pool cooling. The guidance was found to be appropriate and to not adversely impact equipment. The guidance is also consistent with the safety evaluation for the spent fuel pool cooling system. Also, the implementation of this procedure change permitted the deletion of the use of alternate cooling in NOP 2.10-1.

The acceptability of the license amendment is based on the fact that there is sufficient time for operators to take actions to either restore spent fuel pool cooling or to maintain adequate pool inventory. The specified actions are consistent with those required to support the license amendment as well as being suitable from a mechanical and electrical standpoint. Based on this, the procedure does not negatively impact the margin of safety.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
EOP 3.1-0, Rev. 19	Emergency Response Procedures

Description of Change

The proposed change modified the EOPs listed below:

ECA-0.0	Rev. 11	ES-0.0	Rev. 4	F-0.6	Rev. 4
ECA-0.1	Rev. 8	ES-0.1	Rev. 9	FR-C.1	Rev. 9
ECA-0.2	Rev. 10	ES-0.2	Rev. 10	FR-C.2	Rev. 8
ECA-1.1	Rev. 10	ES-0.3	Rev. 9	FR-C.3	Rev. 5
ECA-1.2	Rev. 4	ES-0.4	Rev. 9	FR-H.1	Rev. 1
ECA-2.1	Rev. 10	ES-1.1	Rev. 9	FR-H.2	Rev. 5
ECA-3.1	Rev. 9	ES-1.2	Rev. 8	FR-H.3	Rev. 8
ECA-3.2	Rev. 9	ES-1.3	Rev. 12	FR-H.4	Rev. 6
ECA-3.3	Rev. 10	ES-1.4	Rev. 12	FR-H.5	Rev. 7
EOP 3.1-0	Rev. 19	ES-3.1	Rev. 8	FR-I.1	Rev. 6
EOP 3.1-10	Rev. 16	ES-3.2	Rev. 8	FR-I.2	Rev. 5
EOP 3.1-12	Rev. 9	ES-3.3	Rev. 7	FR-I.3	Rev. 9
EOP 3.1-21	Rev. 10	F-0.1	Rev. 4	FR-P.1	Rev. 1
EOP 3.1-34	Rev. 12	F-0.2	Rev. 3		
EOP 3.1-46	Rev. 12	F-0.3	Rev 4		
EOP 3.1-49	Rev. 11	F-0.4	Rev. 4		
EOP 3.1-50	Rev. 6	F-0.5	Rev. 4		

Reason for Change

The changes were made to reflect the following:

- Installation of a Condensate Storage Tank under PDCR 1271
- Modification to the Containment Air Recirculation Fan Safety Injection Circuit under PDCR 1347
- Replacement of the Highest Lift Pressure Safety Valve on each Steam Line with a Safety/Relief Valve Under PDCR 1316
- Repowering of Electrical Equipment Under PDCR 1317
- Replacement of the Reactor Vessel Level Indication System Probes Under PDCR 1338
- Removal of the "C" and "D" DC Panels Under PDCR 1336

Safety Evaluation

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries. The procedure change does not adversely impact the design basis accidents, accident mitigation strategy, nor equipment usage.

Procedure Number Title

EOP 3.1-10, Rev. 16 Partial Loss of AC

Description of Change

The change adds a step in the procedure to ensure that at least one AC emergency bus is energized. If not, and power can not be restored, the procedure directs the operator to a AOP 3.2-61.

Reason for Change

The change addresses power restoration for scenarios in Modes 5 or 6 where this procedure is entered with loss of both AC emergency buses.

Safety Evaluation

There are no design basis accidents postulated to occur in Modes 5 or 6 coincident with loss of both emergency AC trains. Since the proposed change will only be utilized for scenarios that are not considered to be design basis accidents, the changes can not affect the probability of occurrence of previously evaluated accidents.

The procedures and changes provide appropriate guidance for handling a total loss of AC in Modes 5 or 6. Therefore, they can not adversely affect any protective boundary. This means that the margin of safety for the protective boundaries is not adversely impacted.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number   Title

EOP 3.1-12, Rev. 9   Emergency Boration

Description of Change

The change reversed the order of Steps 4 and 5. Also, the Refueling Water Storage Tank was added to Step 5.

Reason f r Change

The change provides additional guidance to operations personnel, and increase the consistency with EOP 3.1-10.

Safety Evaluation

The change does not affect design basis accident mitigation. They do not negatively impact equipment usage, nor do they change the intent of the existing steps. Therefore, there is no impact on the margin of safety.

The procedure change provides appropriate guidance to operations personnel for mitigation of the initiating event. Also, the procedure change does not adversely imnpact the design basis accidents, accident mitigation strategy, or equipment usage.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number | Title

EOP 3.1-21, Rev. 10 Pressurizer Spray Valve Malfunction

Description of Change

The change modifies the format of the procedure steps without changing the mitigation strategy. Also, the caution instructing the operators to trip the reactor and go to procedure E-0 if Reactor Coolant System (RCS) pressure approaches 1800 psig was expanded to include RCS pressure approaching a variable low pressure reactor trip setpoint. Step 15 and Substep 12a were added to provide additional guidance to the operators.

Reason for Change

The change provides additional guidance to the operators, and it increases consistency with EOP 3.1-0. If the actions taken in the procedure have stabilized plant conditions, additional actions may be taken to isolate a failed spray valve. This ensures that the procedural steps to reduce power and stop the reactor coolant pump will be performed under conditions for which containment entry and spray valve isolation will not be successful.

Safety Evaluation

The changes do not affect design basis accident mitigation. They do not negatively impact equipment usage nor change the intent of the existing steps. Therefore, there is no impact on the margin of safety.

The procedure change provides appropriate guidance to the operators for mitigation of the initiating event. Also, the procedure change does not adversely impact the design basis accidents, accident mitigation strategy, nor equipment usage. Therefore, the changes are safe.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
EOP 3.1-34, Rev. 12	Complete Loss of Control Air Supply

Description of Change

Step No. 1 of the procedure was revised to guide operators in checking the control air pressure to determine if it is stable at greater than 50 psig or increasing. If the control air pressure is decreasing or operation of the air-operated valves is unstable, E-0, "Reactor Trip or Safety Injection," should be entered. While efforts are ongoing to locate and isolate the source of the leak, it is acceptable for control air pressure to be stable.

Reason for Change

The change provides additional guidance to operators and increases the consistency with EOP 3.1-0. The change helps to lesson the likelihood that an unnecessary reactor trip will occur. The addition of manually adjusting the speed of the Auxiliary Feedwater Water (AFW) pumps only provides additional AFW flow. The check of control air pressure that was revised is acceptable.

Safety Evaluation

The changes do not affect design basis accident mitigation. They do not negatively impact equipment usage nor change the intent of the existing steps. Therefore, there is no impact on the margin of safety.

The procedure change provides appropriate guidance to the operators for mitigation of the initiating event. Also, the procedure change does not adversely impact the design basis accidents, accident mitigation strategy, nor equipment usage. Therefore, the changes are safe.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number | Title

EOP 3.1-46, Rev. 12 Total Loss of Semi Vital Power

Description of Change

The change revised Steps 2e, 5, and 6. Also, the change added an additional caution statement.

Reason for Change

To provide additional guidance to operators and increase the consistency with EOP 3.1-0. To instruct operators on how to throttle charging flow since the selected charging flow control valve fails open on loss of semi-vital power. The change provides greater assurance that the actions to control pressurizer level are successful. The change to Step 6 adds closing of the seal water return trip valves, CH-TV-240 and 241, to closing of the seal water return motor operated valves. The additional caution will alert the operator to the fact that high pressure steam dump will actuate upon restoration of semi-vital power if reactor coolant system temperature is above 545 degrees Fahrenheit. The change to Step 5 provides additional/better guidance to the operators to respond to a loss of semi-vital power.

Safety Evaluation

The changes do not affect design basis accident mitigation. They do not negatively impact equipment usage nor change the intent of the existing steps. Therefore, there is no impact on the margin of safety.

The procedure change provides appropriate guidance to the operators for mitigation of the initiating event. Also, the procedure change does not adversely impact the design basis accidents, accident mitigation strategy, nor equipment usage. Therefore, the changes are safe.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number Title

EOP 3.1-49, Rev. 11 Partial Loss of DC

Description of Change

As a result of the modifications made under PDCR 1331, the loss of DC Bus "A" will no longer directly result in the loss of the "A" and "B" vital panels. This means that charging flow control valve CH-FCV-110A will no longer fail open. Also, the steam generator (SG) feedwater air-operated bypass valves for SGs #1 and #2 will no longer fail open nor will the associated SG narrow range level channels fail. This procedure was modified to delete the actions associated with these failures, which should no longer occur, and to verify operation of the static transfer switches. The other changes to the procedure provide additional guidance. The addition of attachments for power restoration should provide additional assurance that these actions are successful.

Reason for Change

This change incorporates plant modifications associated with PDCR 1331, provides additional guidance to the operators, and increases consistency with procedure EOP 3.1-0.

Safety Evaluation

The changes do not affect design basis accident mitigation. They do not negatively impact equipment usage nor change the intent of the existing steps. Therefore, there is no impact on the margin of safety.

The procedure change provides appropriate guidance to the operators for mitigation of the initiating event. Also, the procedure change does not adversely impact the design basis accidents, accident mitigation strategy, nor equipment usage. Therefore, the changes are safe.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number Title

EOP 3.1-50, Rev. 6 Loss of MCC-5

Description of Change

The modifications made under PDCR 1317 included: 1) moving the power supply for the charging pump "A" Main Lube Oil Pump, P-149-1A, from MCC-5 to MCC-12; and 2) moving the power supply for battery charger BC-1-1A from MCC-5 to MCC-13. Step 4 was rewritten to incorporate that a loss of MCC-5 would not cause the "A" charging pump's main lube oil pump to lose power. The "A" charging pump is now the preferred charging pump, since manual actions may be required to start an auxiliary lube oil pump for the "B" charging pump. The "response not obtained" action for this step was modified to provide the actions required to start the "B" charging pump if MCC-12 is unavailable.

In addition, this change deletes the alarm associated with BC-1-1A, since a loss of MCC-5 will no longer result in a loss of power to the "A" charging pump's auxiliary lube oil pump.

Note, these changes reflect the repowering performed under PDCR 1317.

Reason for Change

This change incorporates the plant modifications associated with PDCR 1317, provides additional guidance to the operators, and increases the consistency with procedure EOP 3.1-0.

Safety Evaluation

The changes do not affect design basis accident mitigation. They do not negatively impact equipment usage nor change the intent of the existing steps. Therefore, there is no impact on the margin of safety.

The procedure change provides appropriate guidance to the operators for mitigation of the initiating event. Also, the procedure change does not adversely impact the design basis accidents, accident mitigation strategy, nor equipment usage.

Implementation of this procedure does not affect protective boundaries or the safety
limits of the protective boundaries.

Procedure Number	Title
SPL 10.1-36, Rev. 0	Installation, Testing and Use of a Temporary Fire Pump

Description of Change

The change revises the procedure to include a temporary fire pump system.

Reason for Change

The change provides a backup/alternate water supply for temporary replacement of the permanent fire pump supply system.

Safety Evaluation

The temporary modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. This procedure tests the auto-start capability and develops a pump capability curve for comparison with the permanent fire pump curves. In addition, the system will be tested at the normal system operating pressure to assure pressure integrity.

The temporary modification does not create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, nor does it reduce the margin of safety as defined in the basis for any technical specification.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
SPL 10.1-37, Rev. 0	Installation, Testing and Use of a Temporary Fire Pump

Description of Change

The change revises the procedure to include a temporary fire pump system.

Reason for Change

The change provides a backup/alternate water supply for temporary replacement of the permanent fire pump supply system.

Safety Evaluation

The temporary modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. This procedure tests the auto-start capability and develops a pump capability curve for comparison with the permanent fire pump curves. In addition, the system will be tested at the normal system operating pressure to assure pressure integrity.

The temporary modification does not create a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, nor does it reduce the margin of safety as defined in the basis for any technical specification.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
SPL 10.3-29, Rev. 0	Changing Plant Load for Rod Position Indication System Testing and Turbine Control Valve Test

Description of Change

This procedure was created to: (1) collect control rod position, low voltage differential transformer voltage/current, reactor coolant system temperature, and reactor power data at preselected reactor power levels and Bank "B" positions, (2) demonstrate main control board and coil stack temperature effects on new rod position indication drawer power supply, (3) determine reactor coolant system temperature effects on the analog rod position indication system, and (4) accomplish load change necessary to perform PMP 9.1-1, "Turbine Control Valve Test."

Reason for Change

The procedure collects data which will be used to analyze IRPI system response and its correlation to changes in reactor power level and rod position.

Safety Evaluation

Although the Rod Drop Load Runback feature was disabled during the test, the short time duration limits the risk associated with a rod drop without turbine runback. Further, Dropped Rod Accident Analysis was revised in Cycle 15 to eliminate the credit for turbine runback. United States Nuclear Regulatory Commission approval was received for elimination of this credit. Thus, disabling the Rod Drop Load Runback feature was consistent with the design basis accident analysis. Finally, while no longer credited in the design basis accident analysis, the Rod Drop Load Runback feature does serve to mitigate a rod drop accident. Temporary disabling of the feature is necessary to perform the required IRPI data collection.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
SPL 10.3-30, Rev. 1	Control Rod Maneuvers for Rod Position Indication System Testing and Flux Mapping

Description of Change

Clerical changes to update the procedure for the Cycle 18 Refueling Outage.

Reason for Change

The procedure collects data which will be used to analyze IRPI system response and its correlation to changes in reactor power level and rod position.

Safety Evaluation

Although the Rod Drop Load Runback feature was disabled during the test, the short time duration limits the risk associated with a rod drop without turbine runback. Further, the Dropped Rod Accident Analysis was revised in Cycle 15 to eliminate the credit for turbine runback. United States Nuclear Regulatory Commission approval was received for elimination of this credit. Thus, disabling the Rod Drop Load Runback feature was consistent with the design basis accident analysis. Finally, while no longer credited in the design basis accident analysis, the Rod Drop Load Runback feature does serve to mitigate a rod drop accident. Temporary disabling of the feature is necessary to collect data regarding the Individual Rod Position Indicators.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
SPL 10.3-31, Rev. 0	Control Rod Maneuvers for Rod Position Indication System Data Collection Instrumentation Check-out

Description of Change

Temporarily remove from service at least three control rod position indicators with the Load Runback mercoid jumpered if reactor power is above 90%.

Reason for Change

To collect Individual Rod Position Indicator (IRPI) data. This procedure is designed to test the feasibility of procedure SPL 10.3-29.

Safety Evaluation

A test box jumper device with "test" and "operational" switch positions was installed to collect IRPI data. With the switch in the "operational" position, IRPI will be operable. During Bank B rod motion, the test switch will be placed in the "operational" position. The test box jumper device has no affect on the operability or performance of the system.

The proposed procedure does not adversely effect the margin of safety incorporated in the design basis event analysis. The consequences associated with the identified evaluated accidents have already taken into account the unavailability of automatic load runback.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Procedure Number	Title
SUR 5.7-148B, Rev. 3	Substantial Flow Test of "A," "B," "C," and "D" Service Water Pumps

Description of Change

The change allows the operable train's pair of service water pumps to be placed in the auto position (pump stopped but ready to start on manual or auto signal) with the other (Technical Specification inoperable) train's pumps running.

Reason for Change

The purpose of this revision to SUR 5.7-148B is to allow a full flow test to be performed on one of the service water pumps while in Mode 1, 2, 3, or 4. Prior to this revision, the procedure only allowed testing in only Modes 5 and 6.

Safety Evaluation

These changes do not increase the probability of occurrence of an accident evaluated previously in the safety analysis report and do not increase the probability of occurrence of a malfunction of equipment important to safety evaluated previously in the safety analysis report.

The performance of the procedure will have no impact on the availability of offsite power. Furthermore, the performance of this procedure does not increase the chance of failure of the service water system on loss of AC. While there will only be a minimum of two pumps running during the test, the pumps in the operable train will remain operable and available for use for the duration of the test.

These changes do not increase the consequences of an accident evaluated previously in the safety analysis report. The consequences of the loss of offsite power accident are not increased. By maintaining the availability of both pumps on the operable train, this procedure does not prohibit any of the operable pumps from sequencing onto the diesels. Therefore, it does not reduce service water availability in a loss of offsite power scenario.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

JUMPERS-LIFTED LEADS-BYPASSES

J-LL-B Number	Title
92-035*	Replacement Fastener for Panel Door
93-007	Temporary Installation of a 2" x 4" Wooden Wedge in the Operator of SW-V-101A
93-015	Shutdown Risk Reduction Diesel Generator
93-016	Replacement of Erosion/Corrosion Suspect Piping
93-020	Temporary Load Cell with Digital Indicator
93-021	Temporary Powering of Vital Buses "A" and "B" from Semi-Vital System
93-025	Removal of EG2A Excitation Panel Upper Real Panel Covers
93-030	Test Spare GSU 345 kV Transformer Trip Circuit on the 27Y/1-3 Lockout Relay on Auxiliary Board 4
93-033	Bypass the Signal from the Bad Sensor on Train "A" Reactor Vessel Level Indication System (RVLIS) (Sensor 6)
93-036	1st Point Heater Extraction Steam Line Isolation Valve MS-V-560
93-037	Installation of Clamp on Bonnet
93-039	Supplying Underground Storage Tanks on Demand While the Above Ground Storage Tank is Empty
93-040	Supplying Fuel Oil to the Boilers While the Above Storage Tank is Empty
93-041	Connection of a Power Line Disturbance Monitor

* Note: This jumper was accidentally omitted from the 10CFR50.59 1992 report.

Jumper - Lifted Lead - Bypass Number 92-035

This bypass jumper, entitled "Replacement Fastener for Panel Door," has been removed.

Description of Jumper-Lifted Lead-Bypass

A 1/4" x 1" carbon steel bar was used to restrain the control card access doors on the "A" and "B" Auxiliary Feedwater Pump Digital Control System panels.

Reason for Jumper-Lifted Lead-Bypass

The carbon steel bar ensures the panels' seismic qualification was maintained. The access doors fit up against the cards and are credited with preventing the cards from dislocating during a seismic event. Originally, six 1/4 turn fasteners were provided on each door to hold it closed. Several of these fasteners had failed in the past from excessive use thereby altering the condition of the panel from that in which it was originally seismically qualified.

Safety Evaluation

The temporary restraints have no effect on the Auxiliary Feedwater system operability in any mode of operation other than during a seismic event. They do not increase the probability or consequences of any design basis accident nor do they increase the probability of failure of any safety system.

Jumper - Lifted Lead - Bypass Number 93-007

This bypass jumper, entitled "Installation of a 2" x 4" Wooden Wedge in the Operator of SW-V-101A," has been removed.

Description of Jumper-Lifted Lead-Bypass

A 2" x 4" wooden wedge was temporarily installed in the operator of SW-V-101A.

Reason for Jumper-Lifted Lead-Bypass

The wooden wedge will prevent valve closure during valve operator maintenance. It will allow the "A" service water pump to remain operable by preventing SW-V-101A (butterfly valve) from inadvertently closing due to full flow forces and valve operator disassembly. The valve operator is disassembled to replace bearings and washers which are worn to the point of jeopardizing the ability to reopen the valve after the strainer cleaning and maintenance.

Safety Evaluation

The addition of a wooden wedge will not alter the valves operability as defined in the Haddam Neck Plant Technical Specifications, since the required open position will be maintained and easily verified. If by chance the valve were to inadvertently close, the wedge could easily be removed and the operator reassembled within the 72 hour action statement of Section 3.7.3 of the Haddam Neck Plant Technical Specifications.

Since the valve operator will not be functional, the addition of the wedge will cause the discharge valve to perform differently from the description in Section 9.2.1 of the Haddam Neck Plant Final Safety Analysis Report. It states: "the service water system accommodates individual isolation of all pumps, strainers, etc., to maintain system operation during equipment repair and maintenance periods." However, the valve is wedged in its required safety position.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-015

This bypass jumper, entitled "Shutdown Risk Reduction Diesel Generator," has been removed.

Description of Jumper-Lifted Lead-Bypass

This jumper installed a temporary diesel generator (D/G) along the north fence of the spare (6X) G.S.U. transformer yard (12-R).

Reason for Jumper-Lifted Lead-Bypass

The purpose of this D/G is to provide a supplemental power source during the refueling outage in accordance with NUMARC item 4.3.1.4 and 4.3.1.6 recommendations for shut down risk reduction.

Safety Evaluation

The D/G trailer is a self supporting unit housing all necessary equipment for operation. This includes lighting (AC and DC), batteries (24v), battery charger, switchgear, sound proof control and monitoring equipment room, engine block heaters, fire detection and suppression system, ventilation, and fuel tank (1000 gallons).

The jumper has not been found to be an unreviewed safety question or a significant hazard consideration since it does not: (1) increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated in the safety analyses report (the D/G will not be connected directly to any safety equipment or busses), (2) create the possibility for an accident or malfunction (when the D/G is used to power any safety related equipment, the normal protection devices for that equipment will be in service in addition to the diesels own protection devices), (3) reduce the margin of safety as defined in the Haddam Neck Plant Technical Specifications

There will not be a direct connection to any safety related equipment or busses without their normal protective devices in service.

Jumper - Lifted Lead - Bypass Number 93-016

This bypass jumper, entitled "Replacement of Erosion/Corrosion Suspect Piping," has been removed.

Description of Jumper-Lifted Lead-Bypass

Installation of temporary power for the replacement of erosion/corrosion suspect piping (per PDCR 1296).

Reason for Jumper-Lifted Lead-Bypass

480 volt power is needed throughout the turbine building to operate Cooper heaters (pre and post heat treatment), grinders, and welding machines.

Safety Evaluation

The temporary power load has no adverse effect on existing in-service loads of MCC-3 or MCC-4. Since the work was done during plant shutdown, the motor control centers (MCC) were lightly loaded. All wires were disconnected/reconnected in accordance with PMP 9.5-192, "Disconnection and Reconnection of Electrical Breakers." In addition, all circuit breakers that were installed were tested in accordance with PMP 9.8-88, "Testing of Molded Case Circuit Breakers and Motor Circuit Protectors."

The jumper does not affect the operability of any plant equipment or system as defined in the Technical Specifications. The jumper will only affect the electrical operation of MS-MOV-36. Temporary power was energized only after plant shutdown. The temporary equipment is protected by recently tested circuit breakers which will protect plant systems (safety and non-safety related) in the event of an equipment malfunction.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-020

The bypass jumper, entitled "Temporary Load Cell with Digital Indicator," has been removed.

Description of Jumper-Lifted Lead-Bypass

Installation of a temporary load cell with digital indicator to bypass the load cell on the Spent Fuel Building north crane (CR-5-1A).

Reason for Jumper-Lifted Lead-Bypass

To replace an existing load cell which was inoperable.

Safety Evaluation

The temporary load cell will not cause the north crane to perform differently from the description in the Haddam Neck Plant Final Safety Analysis Report (FSAR). In the FSAR, the safety requirement for handling of fuel states that during this operation "the design of the equipment & structures result in the handling of assemblies under a minimum of 8′ of water." This criterion was not affected.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-021

This bypass jumper, entitled "Temporary Powering of Vital Buses 'A' and 'B' from Semi-Vital System," has been removed.

Description of Jumper-Lifted Lead-Bypass

Temporary powering of 120VAC Vital Buses "A" and "B" while inverters "A" and "B" were replaced during the Cycle 17 Refueling Outage. The work was performed under a Jumper Bypass Procedure while the plant was between Modes 5 and 6.

Reason for Jumper-Lifted Lead-Bypass

Temporary power was required to vital panels while inverters "A" and "B" were replaced.

Safety Evaluation

Section 3.8.3.2.C of the Haddam Neck Plant Technical Specifications states that either 120VAC Vital Buses "A" and "B" or Vital Buses "C," "C1," "D" and "D1" shall be energized from their associated inverters. Vital Buses "C," "C1," "D," and "D1" were energized in this manner during the time the "A" and "B" inverters were out of service. To allow more flexibility, vital buses "A" and "B" were temporarily powered from the semi-vital system.

In either case, adequate coordination of molded case circuit breakers was maintained such that a fault on vital bus "A" and "B" would not affect loads on the semi-vital system.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-025

This bypass jumper, entitled "Removal of EG2A Excitation Panel Upper Real Panel Covers," has been removed.

Description of Jumper-Lifted Lead-Bypass

The jumper removed the cabinet covers on the excitation panel for the "B" Emergency Diesel Generator (EDG).

Reason for Jumper-Lifted Lead-Bypass

To allow sufficient cooling of the cabinet internal components while the normal ventilation fan is out of service.

Safety Evaluation

The jumper will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety. The EDG was not connected to any safety equipment or busses. If the EDG was used to power any safety related equipment, the normal protection devices for that equipment will be in service in addition to the EDG's own protective devices.

Jumper - Lifted Lead - Bypass Number 93-030

This bypass jumper, entitled "Test Spare GSU 345 kV Transformer Trip Circuit on the 27Y/1-3 Lockout Relay on Auxiliary Board 4," has been removed.

Description of Jumper-Lifted Lead-Bypass

To open all flexi-test fingers on device FT-27Y/1-3 (A), with the exception of finger "I." Also, to open all flexi-test fingers on device FT-27Y/1-3 (B), with the exception of finger "B" and "A."

Reason for Jumper-Lifted Lead-Bypass

To ensure that there will not be an inadvertent trip of P-31-1A, P-33-1A, P-34-1C, SS1A, Warehouse Feeder, P-35-1A and 3T1A, during testing of the spare GSU trip circuit on the 27Y/1-3 lockout relay.

Safety Evaluation

The test involved functional verification of finger "I" of device FT-27Y/1-3 (A) and fingers "A" and "B" of device FT-27Y/1-3 (B), and testing of the spare GSU 345 kV transformer trip circuit.

The 27Y/1-3 undervoltage trip is blocked for the aforementioned components for the duration of the red-line procedure (approximately one hour). Therefore, if there was an undervoltage condition on Buses 1-2 and 1-3, the above loads will not be shed because the trip and block close circuits were isolated via the opening of the flex-test fingers.

It was highly unlikely that an undervoltage condition would occur due to the reliability of the off-site power sources. Even if an undervoltage condition were to occur, these loads would have to be manually tripped from the control room.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-033

This bypass jumper, entitled "Bypass the Signal from the Bad Sensor on Train 'A' Reactor Vessel Level Indication System (RVLIS) (Sensor 6)," has been removed.

Description of Jumper-Lifted Lead-Bypass

To lift wire TB2-T23 (Heated Thermocouple #6) and move wire TB2-T24 to T26. Also, to remove jumper between T26 and T23 and install twisted thermocouple wire between terminal T23 and T24.

Reason for Jumper-Lifted Lead-Bypass

There was a bad connection in the cabling for sensor 6 (heated thermocouple) which was disabling train "A" RVLIS.

Safety Evaluation

With this jumper installed, the "A" train RVLIS will not indicate reactor coolant inventory below 14% even if inventory is actually lower. Decreasing inventory will not be affected by this jumper since level will trend toward 14% as other operable sensors are uncovered. This jumper will not alter operator response since at least 15% level will be displayed if the lowest sensor is covered and less than 15% if it is not. Since sensor 6 contributes only 14% of level and the emergency operating procedures require at least 15% inventory is verified before entering a "response not obtained" action, this jumper does not require that any procedures be changed.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-036

This bypass jumper, entitled "1st Point Heater Extraction Steam Line Isolation Valve MS-V-560," has been removed.

Description of Jumper-Lifted Lead-Bypass

Replacement of MS-V-560 with a spool piece.

Reason for Jumper-Lifted Lead-Bypass

To provide a turbine extraction steam path to 1A and 1B feedwater heaters. MS-V-560 had developed a through-wall leak and would not be isolated.

Safety Evaluation

The 1st Point Heater Extraction Steam Line Isolation Valve (MS-V-560) removed under this jumper does not support the operation of any safety related system. The valve is not utilized in any station normal operating procedure or emergency oeprating procedure other than to ensure correct valve line-up. In the event of a small line break, the system would be isolated by the Main Turbine Stop Trip valves.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-037

This bypass jumper, entitled "Installation of Clamp on Bonnet," has been removed.

Description of Jumper-Lifted Lead-Bypass

Installation of clamp on the #1 Feedwater Regulating Valve (FRV) bonnet.

Reason for Jumper-Lifted Lead-Bypass

To provide additional restraint for a 1/4" threaded plug in the stuffing box. This plug, installed in place of a Furmanite fitting, was exhibiting a minor leak around the threads. The temporary clamp is installed to reduce the potential for plug failure and to provide a secondary leak barrier in the event of plug failure.

Safety Evaluation

The temporary clamp was installed external to the valve and has no impact on valve operation. The clamp reinforces the integrity of the threaded plug providing a secondary assurance that the plug will remain in place. Failure of the clamp will not cause the plug or any other part on the FRV to fail.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-039

This bypass jumper, entitled "Supplying Underground Storage Tanks on Demand While the Above Ground Storage Tank is Empty," has been completed.

Description of Jumper-Lifted Lead-Bypass

Diesel fuel truck hooked into fuel lines 1-1/2"-OFH-151-1/2 between FO-LCV-1700A/B and underground storage tank.

Reason for Jumper-Lifted Lead-Bypass

Above ground storage tank TK-33-1A was drained for cleaning and inspection.

Safety Evaluation

The above ground storage tank is not credited in either the Haddam Neck Plant Technical Specifications or the design basis. The original design basis summary concluded that the underground storage tanks are sufficient to provide 24 hours of diesel operation. This change was not an unreviewed safety question since the Haddam Neck Plant Technical Specifications and design basis documents were not compromised.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-040

This bypass jumper, entitled "Supplying Fuel Oil to the Boilers While the Above Ground Storage Tank is Empty," has been removed.

Description of Jumper-Lifted Lead-Bypass

The diesel fuel truck was hooked into the fuel transfer pump P-44-1A/1B suction line and boiler return line (2"-OFH-151-47) at above ground storage tank valve station.

Reason for Jumper-Lifted Lead-Bypass

The boric acid mix tank would not maintain the minimum required temperature if the heating system was not available.

Safety Evaluation

The above ground storage tank is not credited in either the Haddam Neck Plant Technical Specifications or the design basis. This jumper does not affect jumper #93-039 which supplies a temporary fuel oil source to the underground storage tank TK-33-2A/2B. This change was not an unreviewed safety question since the Haddam Neck Plant Technical Specifications and design basis documents were not compromised.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety.

Jumper - Lifted Lead - Bypass Number 93-041

This bypass jumper, entitled "Connection of a Power Line Disturbance Monitor," has been removed.

Description of Jumper-Lifted Lead-Bypass

Connection of a Power Line Disturbance Monitor (PLDM) to the 120VAC Individual Rod Position Indicators (IRPI) power source.

Reason for Jumper-Lifted Lead-Bypass

To monitor the voltage supplied to IRPI drawers and to determine the voltage affects on IRPI.

Safety Evaluation

All possible accidents were evaluated. The only functions affected were the IRPI and the load runback circuits which were not credited in any design basis accident analysis. Therefore, the proposed jumper will not affect any assumption or conclusion nor will it increase the probability of occurrence of an accident evaluated previously in the safety analysis report.

The change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the safety analysis report since IRPI is non-Quality Assurance, and has no safety related inputs.

The jumper will not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, nor will it reduce the margin of safety. The affected system performs as if nothing has changed.

SETPOINT CHANGES

Setpoint Change Number	Title
93-011	Diesel Generator Unloading
93-012	Shutdown Risk Reduction Diesel Generator
93-014	Motor Circuit Protectors for Control Air Compressors C-3-1A and C-3-1B
93-016	Motor Circuit Protectors for "B" Control Air Compressor C-3-1B
93-026	Motor Circuit Protectors
93-037	Motor Circuit Protectors
93-038	Motor Circuit Protectors
93-047	Motor Circuit Protectors
93-048	Motor Circuit Protectors
93-049	Low Voltage MCC-B Replacement - Wastinghouse Type HMCP Motor Circuit Protector Replacement

Setpoint Change Request Number 93-011

This change, entitled "Diesel Generator Unloading," is complete.

Description of Change

The change revises the setpoints to 5 minutes +/- 30 seconds for relays 62-7 & 62-10 and to add two relays (74-1 & 74-2) so that they can annunciate an alarm 10 minutes after Safety Injection Actuation Signal (SIAS) if less than four Containment Air Recirculation (CAR) Fans are operating.

Reason for Change

This change alleviated operators from having to manually start the second CAR Fan on each train following SIAS. Also, it provided annunciation of Fan starts at five minutes and ten minutes after SIAS for verification of CAR Fan operation.

Safety Evaluation

During an event which has initiated an SIAS, it is required for one CAR Fan to be operable within one minute of the SIAS, two CAR Fans within ten minutes of the SIAS, and three Car Fans within 15 minutes of the SIAS. Table 15.3.7 of the Haddam Neck Plant Final Safety Analysis Report assumes that one CAR Fan will be operable within one minute of a pipe rupture, and three CAR Fans will be available within 15 minutes. These criteria have not been compromised by this modification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-012

This change, entitled "Shutdown Risk Reduction Diesel Generator," is complete.

Description of Change

This change temporarily revised the time dial setting from 3 to 10 on the Over Current Trip Relays located in Switchgear Cubicle #28 (spare GSU feed).

Reason for Change

For shutdown risk reduction, a temporary D/G (1750 kW) would be powering Bus 1-3. Primary protection would be provided to the downstream cable and the loads by the D/G breaker. Having the time dial setting adjusted for cell 28 breaker would allow coordination and back-up protection in supporting jumper 93-015.

Safety Evaluation

Per the safety evaluation for Jumper-Lifted Lead-Bypass number 93-015, it was determined that this modification would not increase the probability of occurrence of an accident or malfunction of equipment important to safety, nor would it create the possibility for an accident or malfunction or reduce the margin of safety as defined in the Haddam Neck Plant Technical Specifications.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-014

This change, entitled "Motor Circuit Protectors for Control Air Compressors C-3-1A & C-3-1B," is complete.

Description of Change

To change the setting of the control air compressor's circuit protectors from C-500% to E-700%.

Reason for Change

The original settings were resulting in random nuisance tripping. Resetting would allow for more operating margin while still providing adequate motor protection. Original settings were not based on the most conservative field data.

Safety Evaluation

The change in setpoint from the original 500% of the breaker rating to 700% of the breaker rating was in response to the random tripping of Control Air Compressors C-3-1A & C-3-1B, while energized. The Control Air Compressors are non-Quality Assurance: the Motor Circuit Protectors provide for the isolation of Class 1E and non-Class 1E systems, in accordance with IEEE 384, "Independence of Class 1E Equipment and Systems." The safety significance in the revised setpoints is that the associated motor circuit protectors coordinate with their upstream devices so as not to cause overtripping of the MCC-5 feeder breakers. These devices will not trip at a fault value of 700 amps which still provides for adequate protection of the compressor motors.

Based on the aforementioned, the setpoint change does not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-016

This change, entitled "Motor Circuit Protectors for 'B' Control Air Compressors C-3-1B," is complete.

Description of Change

To revise the trip settings associated with the Westinghouse Type HMCP Motor Circuit Protectors for Control Air Compressor C-3-1B and Auxiliary Boiler Fuel Transfer Pump P-44-1A by adding additional conservative factor of one CAM (dial) setting.

Reason for Change

Field operation had shown that additional conservatism was needed to address uncertainties.

Safety Evaluation

The change in setpoint 800% of the breaker rating (800 Amps for the Air Compressor and 56 amps for the Fuel Oil Pump) was in response to the random tripping of the "B" Control Air Compressor and Auxiliary Boiler Fuel Transfer Pump P-44-1A, while energized, and to apply additional margin to account for inaccuracies in the nameplate data. The Control Air Compressor and the Fuel Oil Transfer Pump are non-Quality Assurance: the Motor Circuit Protectors provide for the isolation of Class 1E and non-Class 1E systems, in accordance with IEEE 384, "Independence of Class 1E Equipment and Systems." The safety significance in the revised setpoint is that the associated motor circuit protectors coordinate with their upstream devices so as not to cause overtripping of the MCC5 feeder breakers. These devices will now trip at a fault well above the levels expected during normal motor operation, which still provides for adequate protection of the circuits.

Based on the aforementioned, the setpoint change does not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-026

This change, entitled "Motor Circuit Protectors," is complete.

Description of Change

Delete the settings for the motor circuit protectors installed in MCC-5 for the following devices:

SI-MOV-871A	MCC-8, component 3FG
SI-MOV-871B	MCC-8, component 5FF
BA-MOV-373	MCC-12, component 1L
F-36-1A - PAB Upper Level Supply Fan	MCC-12, component 3J
F-25-1A - PAB Lower Level Supply Fan	MCC-13, component 1J

Reason for Change

The settings were revised to include a margin to account for potential inaccuracies associated with motor nameplate and vendor provided data. Accounting for the margin requires the trip settings of all five branch circuit breakers to be adjusted one cam setting higher.

Safety Evaluation

The motor circuit protectors are all located on safety related buses. By increasing their settings, the level of current required to trip the device was increased, thereby allowing a greater margin between normal motor starting current requirements and the level that will trip the breaker. This resulted in a greater margin of operability for the load. The safety criteria for these devices does not allow branch fault currents to cause overtripping which would therefore trip the motor control center feeder breakers. For all the devices mentioned, the proposed settings will in no way infringe upon this criteria.

The setpoint revisions do not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-037

This change, entitled "Motor Circuit Protectors," is complete.

Description of Change

Delete the settings for the motor circuit protectors installed in MCC-5 for the following devices:

FW-MOV-11	SI-MOV-871A
FW-MOV-12	SI-MOV-871B
FW-MOV-13	C-3-1A
FW-MOV-14	C-3-1B

Reason for Change

The settings were deleted because the associated Westinghouse Type HMCP Motor Circuit Protectors were replaced with nonsettable thermal magnetic molded case breakers.

Safety Evaluation

The setpoint deletions do not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,
- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or
- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-038

This change, entitled "Motor Circuit Protectors," is complete.

Description of Change

Revise the settings for the motor circuit protectors installed in MCC-5 and MCC-12 for the following devices:

BA-MOV-32	CH-MOV-292B	BA-MOV-373	SI-MOV-861C
CH-MOV-257	CH-MOV-292C	SI-MOV-861A	SI-MOV-861D
CH-MOV-257B	H-MOV-310	SI-MOV-861B	P-149-1B

Reason for Change

The settings are revised because the loads are susceptible to jogging/plugging and/or other switching transients. The dynamics involved with this type of operation requires that more margin be employed in setting Westinghouse Type HMCP Motor Circuit Protector trip devices in order to account for the higher in-rush currents thereby allowing a greater operating margin.

Safety Evaluation

The safety related function of the above mentioned devices is to provide short circuit protection for the motor circuit and to allow the loads to perform their intended function.

The setting revisions do not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-047

This change, entitled "Motor Circuit Protectors," is complete.

Description of Change

Delete the settings for the motor circuit protectors installed in MCC-5, MCC 12 and MCC-13 for the following devices:

MCC-5		MCC-12	MCC-13
SI-MOV-861A	CH-MOV-292C	CH-MOV-292B	SW-MOV-837B
SI-MOV-861B	SI-MOV-861C	CH-MOV-257B	
SW-MOV-1	SI-MOV-861D	BA-MOV-373	
SW-MOV-2	LD-MOV-200	SW-MOV-837A	
CH-MOV-331	CH-MOV-257		

Reason for Change

The settings are deleted because the associated Westinghouse Type HMCP Motor Circuit Protectors were replaced with non-settable thermal magnetic molded case circuit breakers.

Safety Evaluation

The setting deletions do not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-048

This change, entitled "Motor Circuit Protectors," is complete.

Description of Change

Revise the settings for the motor circuit protectors installed in MCC-5, MCC-6, MCC-7, MCC-8, MCC-12, and MCC-13 for the following devices:

FW-MOV-15	RC-MOV-528	RH-MOV-780	BA-MOV-366
RC-MOV-538	CH-MOV-298	RH-MOV-804	RC-MOV-501
RC-MOV-546	SI-MOV-873	FW-MOV-16	RC-MOV-512
RC-MOV-577	RH-MOV-21	RC-MOV-513	RC-MOV-510
SW-MOV-5	RH-MOV-34	RC-MOV-524	RH-MOV-23
FH-MOV-578	FH-MOV-535	RC-MOV-515	RH-MOV-22
DH-MOV-544	DH-MOV-534	SW-MOV-3	FH-MOV-508
CH-MOV-311	CH-MOV-312	SW-MOV-4	DH-MOV-507
BA-MOV-386	PR-MOV-567	SW-MOV-6	CH-MOV-314
BA-MOV-349	RH-MOV-31	FH-MOV-522	DH-MOV-562
RC-MOV-526	PW-MOV-555	DH-MOV-521	RH-MOV-781
RC-MOV-537	RH-MOV-33A	CH-MOV-313	RH-MOV-803
FH-MOV-344	SW-MOV-30	SI-MOV-24	DH-MOV-310
P-44-1B	CH-MOV-331	F-16-1A	CD-MOV-17
P-82-1A	P-73-1B	P-22-1A	CH-MOV-257
P-63-1A	P-20-1B	P-26-1A	P-44-1A
P-12-1A	P-10-1A	P-19-1A	P-19-1B
P-48-1B	P-28-1B	P-20-1A	P-95-1A
P-29-1B	P-22-1B	P-10-1B	MOV-854
AP-24-1A	P-26-1B	P-46-1A	MOV-901
P-23-1A	P-48-1A	MOV-854B	MOV-903
P-25-1A	P-25-1A	MOV-902	MOV-33B
F-36-1A	P-29-1A	MOV-904	MOV-569
P-149-1A			

Reason for Change

The settings are revised because the loads are susceptible to switching transients. The dynamics involved with this type of operation requires that more margin be employed in setting Westinghouse Type HMCP Motor Circuit Protector trip devices in order to account for the higher in-rush currents thereby allowing a greater operating margin.

Setpoint Change Request Number 93-048 (Continued)

Safety Evaluation

The safety related function of the above mentioned devices is to provide short circuit protection for the motor circuit and to allow the loads to perform their intended function.

The setting revisions do not:

- Increase the probability of an occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report,

- Introduce the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or

- Reduce the margin of safety as defined in the basis for any technical specification.

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

Setpoint Change Request Number 93-049

This change, entitled "Low Voltage MCC-B Replacement - Westinghouse Type HMCP Motor Circuit Protector Replacement," is complete.

Description of Change

Change Westinghouse Type HMCP Motor Circuit Protector setting for motor circuit prctector P-149-1B.

Reason for Change

Resetting trip point allows for more operating margin while still providing adequate motor protection.

Safety Evaluation

There is no impact on the protective boundaries nor on the safety limits of protective boundaries.

TESTS

Test Number	Title
ST 11.3-4, Revision 0	Control Rod Coupling Verification at Power
ST 11.7-107, Revision 0	Emergency Diesel Generator (EDG) EG-2B 24 Hour Endurance Test and 2950 kW Load Test
ST 11.7-108, Revision 0	Emergency Diesel Generator (EDG) EG-2A 24 Hour Endurance Test and 2950 kW Load Test

Number	Title
ST 11.3-4, Rev. 0	Control Rod Coupling Verification at Power

Description of Test

The test procedure will step each control rod in approximately 24 steps (individually) and look for a change in flux as measured by the flux map detector. Five of the six rods will be deliberately inserted below the rod insertion limit applicable at the current power level.

Reason for Test

The procedure was written to verify that the six control rods in the vicinity of core location E-5 are coupled to their drive shafts.

Safety Evaluation

Note, this test is essentially the same in terms of significance as the routine monthly control rod motion checks, which drive an entire bank at a time to below the rod insertion limit. Thus, the safety impact is insignificant.

The procedure does not constitute an unreviewed safety question in that it will not:

- result in an increase in the probability or consequences of an accident. The procedure cannot cause an accident, and since shutdown margin will still be maintained, it cannot increase the consequences of an accident.

- result in a new or different type of accident. The proposed procedure cannot cause any type of accident. The increased potential to drop a rod during the test is insignificant.

- decrease the margin of safety as defined in the basis of any technical specification. Shutdown margin requirements will be maintained. Also, the duration of this procedure will be very short.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Number	Title
ST 11.7-107, Rev. 0	Emergency Diesel Generator (EDG) EG-2B 24 Hour Endurance Test and 2950 kW Load Test

Description of Test

The test was written to test the associated EDG's ability to produce power at the maximum emergency loading, as determined in the associated EDG automatic and Manual Loading Calculations.

Reason for Test

The procedure was written to perform EDG testing similar to EDG's monthly surveillance and to determine the power necessary for the maximum loading expected during a design basis accident with loss of offsite power.

Safety Evaluation

This test will envelop the worst case accident load and will maintain the EDG within its design rating and mode of operation. This test is written to be performed on the EDG and does not challenge the EDG in a way not originally designed and is maintained within its design ratings. This test is considered safe for these reasons.

It is therefore concluded that this test does not:

- Create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis,

- Increase the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety, or

- Reduce the margin of safety as defined in the basis for any technical specification.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

Number	Title
ST 11.7-108, Rev. 0	Emergency Diesel Generator (EDG) EG-2A 24 Hour Endurance Test and 2950 kW Load Test

Description of Test

The test was written to test the associated EDG's ability to produce power at the maximum emergency loading, as determined in the associated EDG automatic and Manual Loading Calculations.

Reason for Test

The procedure was written to perform EDG testing similar to the way EDG is tested during its monthly surveillance and to determine the power necessary for the maximum loading expected during a design basis accident with loss of offsite power.

Safety Evaluation

This test will envelop the worst case accident load and will maintain the EDG within its design rating and mode of operation. This test is written to be performed on the EDG and does not challenge the EDG in a way not originally designed and is maintained within its design ratings. This test is considered safe for these reasons.

It is therefore concluded that this test does not:

- Create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis,

- Increase the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety, or

- Reduce the margin of safety as defined in the basis for any technical specification.

Implementation of this procedure does not affect protective boundaries or the safety limits of the protective boundaries.

EXPERIMENTS

There were no experiments performed under the provisions of Title 10, Code of Federal Regulations, Section 50.59 during 1993.

PRIMARY COOLANT IODINE SPIKING

There was no primary coolant iodine spiking in 1993 that exceeded the one micro curie per gram limit set in the technical specifications.

CHALLENGES TO RELIEF/SAFETY VALVES

In accordance with the commitment made under Item II.K.3.3 of NUREG 0737 (TMI Action Plan) in the W.G. Counsil letter to D.G. Eisenhut, dated June 10, 1980, the following is a report of challenges to Relief/Safety Valves during 1993.

There were no challenges made to the Primary Relief/Safety Valves in 1993.

REGULATORY GUIDE 1.16 REPORT FOR 1993

REGULATORY GUIDE 1.16 REPORT FOR 1993
CONN. YANKEE ATOMIC POWER CO

DATE: 1/ 4/94

WORK & JOB FUNCTION	NUMBER OF PERSONNEL (>100 MREM) STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES	TOTAL MAN-REM STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES
REACTOR OPERATIONS & SURVEILLANCE						
MAINTENANCE PERSONNEL	3	1	0	2.85	0.28	0.24
OPERATING PERSONNEL	36	0	1	17.99	0.01	0.23
HEALTH PHYSICS PERSONNEL	14	0	34	5.59	0.03	15.03
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	1	0	0	0.42	0.12	0.24
ROUTINE MAINTENANCE						
MAINTENANCE PERSONNEL	42	27	100	18.14	6.33	43.35
OPERATING PERSONNEL	0	0	1	0.36	0.02	0.22
HEALTH PHYSICS PERSONNEL	7	0	3	1.80	0.08	1.73
SUPERVISORY PERSONNEL	0	0	0	0.03	0.00	0.08
ENGINEERING PERSONNEL	2	1	5	0.93	1.02	2.06
INSERVICE INSPECTION						
MAINTENANCE PERSONNEL	10	6	88	7.45	2.43	47.03
OPERATING PERSONNEL	0	0	2	0.24	0.00	0.48
HEALTH PHYSICS PERSONNEL	1	0	17	0.29	0.00	4.30
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	3	3	115	1.87	1.41	111.98
SPECIAL MAINTENANCE						
MAINTENANCE PERSONNEL	7	12	96	2.31	3.13	30.24
OPERATING PERSONNEL	0	0	1	0.07	0.00	0.20
HEALTH PHYSICS PERSONNEL	3	0	1	0.65	0.00	0.81
SUPERVISORY PERSONNEL	0	0	1	0.00	0.00	0.29
ENGINEERING PERSONNEL	1	1	8	0.63	0.64	1.64
WASTE PROCESSING						
MAINTENANCE PERSONNEL	0	0	1	0.01	0.00	0.18
OPERATING PERSONNEL	0	0	0	0.04	0.00	0.00
HEALTH PHYSICS PERSONNEL	12	0	20	7.11	0.00	8.13
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.00	0.04	0.00
REFUELING						
MAINTENANCE PERSONNEL	17	5	61	6.83	1.13	34.17
OPERATING PERSONNEL	19	0	0	4.31	0.00	0.09
HEALTH PHYSICS PERSONNEL	8	0	39	2.06	0.02	14.61
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	2	11	0.08	0.48	5.57
TOTAL						
MAINTENANCE PERSONNEL	79	51	346	38.09	13.34	155.74
OPERATING PERSONNEL	55	0	5	23.16	0.03	1.23
HEALTH PHYSICS PERSONNEL	45	0	114	18.20	0.13	45.29
SUPERVISORY PERSONNEL	0	0	1	0.03	0.00	0.37
ENGINEERING PERSONNEL	7	7	139	4.00	3.71	122.07
GRAND TOTAL	186	58	605	83.50	17.21	324.69

REGULATORY GUIDE 1.16 REPORT FOR 1993

REGULATORY GUIDE 1.16 REPORT FOR 1993
CONN. YANKEE ATOMIC POWER CO

DATE 1/ 4/94

WORK & JOB FUNCTION	NUMBER OF PERSONNEL (>100 MREM)			TOTAL MAN-REM		
	STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES	STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES
REACTOR OPERATIONS & SURVEILLANCE						
MAINTENANCE PERSONNEL	3	1	0	2.85	0.28	0.24
OPERATING PERSONNEL	36	0	1	17.99	0.01	0.23
HEALTH PHYSICS PERSONNEL	14	0	34	5.59	0.03	15.03
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	1	0	0	0.42	0.12	0.24
ROUTINE MAINTENANCE						
MAINTENANCE PERSONNEL	42	27	100	18.14	6.33	43.35
OPERATING PERSONNEL	0	0	1	0.34	0.02	0.22
HEALTH PHYSICS PERSONNEL	7	0	3	1.80	0.08	1.73
SUPERVISORY PERSONNEL	0	0	0	0.03	0.00	0.08
ENGINEERING PERSONNEL	2	1	5	0.93	1.02	2.06
INSERVICE INSPECTION						
MAINTENANCE PERSONNEL	10	6	88	7.45	2.43	47.03
OPERATING PERSONNEL	0	0	2	0.24	0.00	0.48
HEALTH PHYSICS PERSONNEL	1	0	17	0.29	0.00	4.30
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	3	3	115	1.87	1.41	111.98
SPECIAL MAINTENANCE						
MAINTENANCE PERSONNEL	7	12	96	2.31	3.13	30.24
OPERATING PERSONNEL	0	0	1	0.07	0.00	0.20
HEALTH PHYSICS PERSONNEL	3	0	1	0.65	0.00	0.81
SUPERVISORY PERSONNEL	0	0	1	0.00	0.00	0.29
ENGINEERING PERSONNEL	1	1	8	0.62	0.69	1.64
WASTE PROCESSING						
MAINTENANCE PERSONNEL	0	0	1	0.01	0.00	0.18
OPERATING PERSONNEL	0	0	0	0.04	0.00	0.00
HEALTH PHYSICS PERSONNEL	12	0	20	7.11	0.00	8.13
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.00	0.04	0.00
REFUELING						
MAINTENANCE PERSONNEL	17	5	61	6.83	1.13	34.17
OPERATING PERSONNEL	19	0	0	4.31	0.00	0.09
HEALTH PHYSICS PERSONNEL	8	0	39	2.06	0.02	14.61
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	2	11	0.08	0.48	5.57
TOTAL						
MAINTENANCE PERSONNEL	79	51	346	38.09	13.34	155.74
OPERATING PERSONNEL	55	0	5	23.16	0.03	1.23
HEALTH PHYSICS PERSONNEL	45	0	114	18.20	0.13	45.29
SUPERVISORY PERSONNEL	0	0	1	0.03	0.00	0.37
ENGINEERING PERSONNEL	7	7	139	4.00	3.71	122.07
GRAND TOTAL	186	58	605	83.50	17.21	324.69

Docket Nos. 50-213
50-245
50-336
50-423

Haddam Neck Plant and Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3

Annual Report

January 1, 1993 to December 31, 1993

MILLSTONE UNIT NO. 1

TABLE OF CONTENTS

INTRODUCTION

None of the plant design changes, procedure changes, jumpers-lifted leads-bypasses, tests or experiments described herein constitute (or constituted) an unreviewed safety question per the criteria of 10CFR50.59.

PLANT DESIGN CHANGES

PDCR Number	Title
1-009-90	Plant Heating Steam System Modifications
1-010-90	Electric Heater Installations and Associated Ventilation System Modifications
1-026-92	Chemistry Lab Heating, Ventilation, and Air Conditioning (HVAC) Modifications
1-132-92	345 kV Transmission System Special Protection System
1-133-92	Millstone Unit No. 1 Intake Structure Platform E-Bay
1-026-93	Reactor Water Cleanup Isolation Setpoint Change
1-083-93	Condenser Low Vacuum Scram Pressure Switch Tubing Modifications

Plant Design Change Number 1-009-90

This change, entitled, "Plant Heating Steam System Modifications," is complete.

Description of Change

Plant Heating Steam System piping and components have been removed from the Heating, Ventilation, and Air Conditioning (HVAC) Room and Switchgear Room and rerouted outside of these rooms. Pipe whip restraints and jet impingement protection have been installed on this piping.

Reason for Change

This modification was implemented to ensure that all vital components and equipment contained in the HVAC Room and Switchgear Room are isolated from harsh environments and dynamic effects resulting from potential high energy line breaks in the Plant Heating Steam System.

Safety Evaluation

This modification ensures that a potential line break in the Plant Heating Steam System (a non-Quality Assurance system) will not adversely affect the vital equipment contained in the HVAC Room and the Switchgear Room. A line break in the supply lines will not affect these rooms since the lines are either outside the rooms or provided with protective devices such as whip restraints, jet shields and encapsulation devices. A line break in the remaining steam heating coils inside the plenum will not affect these rooms since the plenum communicates with the outside environment via air louvers on the external surface of the plenums. Existing plenum dampers on the inlet to these steam heating coils have been locked open to ensure the vent path to the environment is always provided.

The normal operating parameters, design bases and function of the Plant Heating Steam System remain unchanged.

All building and structural loads remain within design bases.

Plant Design Change Number 1-010-90

This change, entitled, "Electric Heater Installations and Associated Ventilation System Modifications," is partially complete.

Description of Change

Electric heaters have been installed in the Heating, Ventilation, and Air Conditioning (HVAC) Room to replace the Plant Heating Steam System components removed under a separate design change. Additionally, electric duct heaters have been installed in the Switchgear Room to replace the Plant Heating Steam System components removed under a separate design change. However, the new dampers and penthouse have not yet been installed for this system; thus, the Switchgear Room heating and ventilating system currently requires manual operation during the winter mode of operation.

Reason for Change

This modification is being implemented to ensure that all vital components and equipment contained in the HVAC Room and Switchgear Room are isolated from harsh environments and dynamic effects resulting from potential high energy line breaks in the Plant Heating Steam System.

Safety Evaluation

This modification ensures that a potential line break in the plant heating steam system will not adversely affect the vital equipment contained in the HVAC room and the Switchgear Room.

Attachments to Quality Assurance (QA) Category 1 building structures have been designed and installed to preclude any adverse effect on the integrity of the building and support structures. New and existing equipment and supports affected by this change have been designed for seismic loads where they could otherwise potentially interact with safety related equipment.

Loss of the new non-QA electric heating coils or unit heaters is equivalent to loss of the existing non-QA Plant Heating Steam System, thus there is no change presented by this plant modification with respect to the design basis of the heating coils.

The Class 1E power supply system is adequately protected against electrical faults on the heater/cables and inadvertent heater loads on the Emergency Diesel Generator.

Plant Design Change Number 1-026-92

This change, entitled "Chemistry Lab Heating, Ventilation, and Air Conditioning (HVAC) Modifications," is complete.

Description of Change

The Millstone Unit No. 1 Chemistry Lab HVAC system has been modified. Three new split air conditioning systems and three new bench type laboratory hoods have been installed. The associated power supplies and control circuits have been modified as required. Duct smoke detectors have been installed.

Reason for Change

The main office HVAC system supplied conditioned air to the Office Area, Counting Room, and the Lab. Over time, the distribution and circulation of air to the Chemistry Lab has degraded, adversely affecting the operability of the laboratory equipment.

Safety Evaluation

This modification enhances the capability of the HVAC system to provide conditioned air to the Laboratory Area. The only safety related aspects of this change are: 1) the penetrations installed in the wall of a Quality Assurance (QA) Category I wall; 2) the installation of expansion anchor inserts into the ceiling of a QA Category I ceiling; and 3) the installation of non-QA relays, terminal block and fuse in a QA Category I HVAC panel. These modifications were designed and installed to preclude adverse effects to the QA Category I equipment and structure design margins.

Plant Design Change Number 1-132-92

This change, entitled, "345 kV Transmission System Special Protection System," is complete.

Description of Change

A protective relaying scheme was installed to detect simultaneous or successive faults occurring within a one second time span on the Millstone-Montville 371 transmission line and the Millstone-Card 383 transmission line and to trip Millstone Unit No. 1 by opening the 345 KV switchyard breakers 5T and 6T to preclude system instability.

Reason for Change

Northeast Utilities System Planning Department has completed an analysis which demonstrates the potential for transmission system instability following postulated faults on the 345 KV transmission system in the vicinity of Millstone Station. In response to this potential instability, procedures were developed to limit net generation from Millstone Station to ensure stability.

This modification allows Millstone Station generation at design rating while ensuring stability should the postulated faults occur.

Safety Evaluation

The design of this protective relaying scheme is in accordance with Northeast Power Coordinating Council Basic Design Criteria for normal contingencies. This modification enhances the stability of the 345 KV transmission system, thereby reducing the probability of a loss of offsite/normal power. The modification will increase the frequency of a loss of external load by approximately 0.01 per year. This is insignificant. No credible single failure will result in a trip of Millstone Unit No. 1 due to inadvertent actuation of the proposed design.

Plant Design Change Number 1-133-92

This change, entitled, "Millstone Unit No. 1 Intake Structure Platform E-Bay," is complete.

Description of Change

A platform and a personnel safety cable were installed along the wall of the E-Bay Intake Structure.

Reason for Change

This modification was implemented to facilitate safe quarterly inspection of the Emergency Service Water Headers.

Safety Evaluation

The platform and safety cable facilitate quarterly inspection of the Emergency Service Water Headers. They do not perform any safety related function or affect any safety related system, component, or structure. The possibility of interaction with the nearby safety related piping headers and Emergency Service Water pump intake lines has been reviewed. Also, the potential obstruction of the pump intake lines has been reviewed. It has been concluded that interaction or obstruction is not credible.

Plant Design Change Number 1-026-93

This Change, entitled, "Reactor Water Cleanup Isolation Setpoint Change," is complete.

Description of Change

The setpoint of the Reactor Water Cleanup non-regenerative heat exchanger high temperature alarm and Reactor Water Cleanup isolation has been lowered from 150 degrees Fahrenheit to 135 +/-5 degrees Fahrenheit (increasing).

Reason for Change

The Reactor Water Cleanup demineralizer is designed for a maximum operating temperature of 140 degrees Fahrenheit. Operation of the demineralizer above 140 degrees Fahrenheit may result in breakdown of the resin bed, resulting in potential release of particulate impurities back into the Reactor Water Cleanup system and into the reactor. Lowering the setpoint from 150 degrees Fahrenheit to 135 +/-5 degrees Fahrenheit eliminates the potential for demineralizer resin bed breakdown.

Safety Evaluation

This modification will not result in a change in the probability of occurrence or consequences of previously evaluated accidents or malfunctions of equipment important to safety and does not create the possibility of an accident or malfunction of a different type then previously evaluated. This modification only changes the temperature at which the Reactor Water Cleanup isolation signal occurs and at which the inlet and outlet isolation valves close. Operation of these valves (part of the reactor coolant pressure boundary) is not affected. The only effect of this modification is to help maintain proper reactor water chemistry, by minimizing the potential for demineralizer resin bed breakdown.

Plant Design Change Number 1-083-93

This change, entitled,"Condenser Low Vacuum Scram Pressure Switch Tubing Modifications," is complete.

Description of Change

A portion of the rigid tubing to each of the Condenser Low Vacuum Scram Pressure Switches has been replaced with flexible tubing.

Reason for Change

The replacement flexible tubing reduces the probability of a reactor scram due to high vibration of the Main Condenser Low Vacuum Switches.

Safety Evaluation

The replacement tubing meets the pressure and temperature requirements for this application. It has been designed and installed to preclude the presence of low points in the tubing and minimize the potential for kinking. Therefore, the possibility of condensation blocking the sensing line is minimized. A seismic and structural review has concluded that the resultant loading remains unchanged. The flexible tubing minimizes the probability of a reactor scram resulting from high vibration of the Main Condenser Low Vacuum Switches.

PROCEDURE CHANGES

Procedure Number	Title
ONP 525D, Rev. 6	Degraded Fire in Reactor Building (Except Shutdown Cooling Pump Room)
EOP 570, Rev. 0	Reactor Pressure Vessel (RPV) Control Flow Charts
EOP 575, Rev. 0	Failure to Scram Flow Charts
EOP 580, Rev. 0	Primary Containment Control Flow Chart
EOP 585, Rev. 0	Secondary Containment Control and Radioactive Release Control Flow Chart
EOP 590.1, Rev. 2	Maximizing Control Rod Drive (CRD) Flow
EOP 590.2, Rev. 1	Injecting Sodium Pentaborate Using the Control Rod Drive (CRD) System
EOP 590.4, Rev. 2	Injecting Sodium Pentaborate Using the Standby Liquid Control Injection Line Drain
EOP 590.6, Rev. 2	Alternate Reactor Pressure Vessel (RPV) Makeup Using Emergency Core Cooling System (ECCS) Keepfill
EOP 590.7, Rev. 3	Alternate Reactor Pressure Vessel (RPV) Makeup Using Fire Water
EOP 590.8, Rev. 4	Primary Containment Spray
EOP 590.10, Rev. 2	Shifting Low Pressure Coolant Injection (LPCI) Pump Suctions from the Torus to the Condensate Storage Tank (CST)
EOP 590.11, Rev. 2	Shifting Core Spray (CS) Pump Suctions from the Torus to the Condensate Storage Tank (CST)
EOP 590.12, Rev. 1	Primary Containment Flooding Using Emergency Service Water (ESW)
EOP 590.14, Rev. 3	Bypassing Group 1 Isolation Signal on Reactor Pressure Vessel (RPV) Low-Low Water Level
EOP 590.15, Rev. 2	Bypassing Group 4 Isolation Signals
EOP 590.16, Rev. 3	Bypassing Cleanup System Isolation Signals
EOP 590.17, Rev. 3	Bypassing Reactor Building Heating, Ventilation, and Air Conditioning (HVAC) Group 2 Isolation Signals

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
EOP 590.18, Rev. 2	Alternate Methods of Control Rod Insertion
EOP 590.19, Rev. 3	Bypassing All Group 1 Isolation Signals Except High-High Steam Line Radiation
EOP 590.20, Rev. 3	Bypassing All Group 1 Isolation Signals
EOP 590.23, Rev. 4	Containment Vent Procedure
EOP 590.24, Rev. 1	Alternate Shutdown Cooling
EOP 590.26, Rev. 5	Containment Cooling During Accident Conditions
EOP 590.27, Rev. 5	Containment Cooling During Anticipated Transients Without Scram (ATWS) Conditions
SP 92-1-35, Rev. 0	Low Pressure Coolant Injection (LPCI) System Check Valve Closing Procedure
SP 93-1-01, Rev. 0	Bypassing Main Steamline Radiation Detectors While Placing Condensate Demineralizer in Service Following Ultrasonic Resin Cleaning (URC)
SP 93-1-03, Rev. 0	Penetration X-25/202D Online Leak Rate Test
SP 93-1-04, Rev. 0	On-Line Local Leak Rate Test (LLRT) - Electrical Penetrations/Control Rod Drive Hatch
SP 93-1-11, Rev. 0	Intake Structure Loss of Ventilation Test
SP 93-1-13, Rev. 0	Functional Test of Reactor Protection System (RPS) Scram Logic Associated with Loss of Normal Power (LNP)
SP 93-1-17, Rev. 0	Replace Breaker in 4160 Volt Bus 14C to Reserve Station Service Transformer (RSST) Cubicle
SP 93-1-20, Rev. 0	Service Water Piping Replacement
SP 93-1-25, Rev. 0	Implementation of Final Feedwater Temperature Reduction During Cycle 14 Coastdown
SP 93-1-26, Rev. 0	Spent Fuel Pool (SFP) Filter Backwash Line 6"-RWR-45

Procedure Number	Title
ONP 525D, Rev. 6	Degraded Fire in Reactor Building (Except Shutdown Cooling Pump Room)

Description of Change

This change deletes guidance for restoration of Reactor Building Closed Cooling Water flow using cross-tie hoses and replaces it with guidance for restoration using cross-tie hard pipes.

Reason for Change

This procedure was revised to reflect replacement of temporary cross-tie hoses with permanent hard, fixed, cross-tie lines.

Safety Evaluation

Implementation of this procedural change recognizes the improved reliability provided by permanent (hard pipe) versus temporary (hoses) cross-ties. The procedure changes have no adverse impact on Appendix R Shutdown requirements.

Procedure Number	Title
EOP 570, Rev. 0	Reactor Pressure Vessel (RPV) Control Flow Charts

Description of Change

This new flow chart oriented procedure supersedes the original text based EOP 570.

Reason for Change

This change was processed to enhance operators' ability to implement Emergency Operating Procedures.

Safety Evaluation

The procedure was converted from text to flow charts. The format change was made to enhance the operators' ability to implement this procedure. The Primary Containment Flooding part of the procedure was changed to reflect a strategy approved by the Nuclear Regulatory Commission. In addition, minor changes were made to improve mitigation of beyond design basis accident scenarios.

The flow charts were validated on the Millstone Unit No. 1 simulator. The validation showed that the flow charts can be effectively used to mitigate both design basis and beyond design basis accident scenarios. The flow charts introduced no new malfunctions and did not adversely impact the margin of safety.

Procedure Number	Title
EOP 575, Rev. 0	Failure to Scram Flow Charts

Description of Change

This new flow chart oriented procedure supersedes the original text based EOP 575.

Reason for Change

This change was processed to enhance operators' ability to implement Emergency Operating Procedures.

Safety Evaluation

The procedure was converted from text to flow charts. The format change was made to enhance the operators' ability to implement this procedure. The Primary Containment Flooding part of the procedure was changed to reflect a strategy approved by the Nuclear Regulatory Commission. In addition, minor changes were made to improve mitigation of beyond design basis accident scenarios.

The flow charts were validated on the Millstone Unit No. 1 simulator. The validation showed that the flow charts can be effectively used to mitigate both design basis and beyond design basis accident scenarios. The flow charts introduced no new malfunctions and did not adversely impact the margin of safety.

Procedure Number Title

EOP 580, Rev. 0 Primary Containment Control Flow Chart

Description of Change

This new flow chart oriented procedure supersedes the original text based EOP 580.

Reason for Change

This change was processed to enhance operators' ability to implement Emergency Operating Procedures.

Safety Evaluation

The procedure was converted from text to flow charts. The format change was made to enhance the operators' ability to implement this procedure. The Primary Containment Flooding part of the procedure was changed to reflect a strategy approved by the Nuclear Regulatory Commission. In addition, minor changes were made to improve mitigation of beyond design basis accident scenarios.

The flow charts were validated on the Millstone Unit No. 1 simulator. The validation showed that the flow charts can be effectively used to mitigate both design basis and beyond design basis accident scenarios. The flow charts introduced no new malfunctions and did not adversely impact the margin of safety.

Procedure Number	Title
EOP 585, Rev. 0	Secondary Containment Control and Radioactive Release Control Flow Chart

Description of Change

This new flow chart oriented procedure supersedes the original text based EOP 585.

Reason for Change

This change was processed to enhance operators' ability to implement Emergency Operating Procedures.

Safety Evaluation

The procedure was converted from text to flow charts. The format change was made to enhance the operators' ability to implement this procedure. The Primary Containment Flooding part of the procedure was changed to reflect a strategy approved by the Nuclear Regulatory Commission. In addition, minor changes were made to improve mitigation of beyond design basis accident scenarios.

The flow charts were validated on the Millstone Unit No. 1 simulator. The validation showed that the flow charts can be effectively used to mitigate both design basis and beyond design basis accident scenarios. The flow charts introduced no new malfunctions and did not adversely impact the margin of safety.

Procedure Number	Title
EOP 590.1, Rev. 2	Maximizing Control Rod Drive (CRD) Flow

Description of Change

This change reorganizes the procedure so that all control room operations are completed prior to directing actions outside the control room.

Reason for Change

This change was implemented to simplify procedural steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575 580, and 585).

Procedure Number | Title

EOP 590.2, Rev. ? | Injecting Sodium Pentaborate Using the Control Rod Drive (CRD) System

Description of Change

This change expands upon the injection of Sodium Pentaborate using various methods. Also, it adds steps to open the CRD suction filter bypass valve to preclude blockage of the CRD suction filter during injection of Sodium Pentaborate solution.

Reason for Change

This change was implemented to simplify procedural sections and steps for ease of performance. Also, instructions to open the CRD suction filter bypass valve during injection were included.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

EOP 590.2, Rev. 1 (CONTINUED)

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.4, Rev. 2	Injecting Sodium Pentaborate Using the Standby Liquid Control Injection Line Drain

Description of Change

This change expands upon the injection of Sodium Pentaborate using various methods.

Reason for Change

This change was implemented to simplify procedural sections and steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.6, Rev. 2	Alternate Reactor Pressure Vessel (RPV) Makeup Using Emergency Core Cooling System (ECCS) Keepfill

Description of Change

This change revises the procedure to delineate the steps required for alternate RPV make-up using the ECCS keepfill system.

Reason for Change

This change was implemented to increase the RPV makeup capability using the ECCS keepfill system.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.7, Rev. 3	Alternate Reactor Pressure Vessel (RPV) Makeup Using Fire Water

Description of Change

This change was processed to align RPV makeup using the Fire Water system via the Condensate Transfer/Core Spray systems and via the Standby Liquid Control Injection Line Drain.

Reason for Change

This change was implemented to simplify procedural sections and steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.8, Rev. 4	Primary Containment Spray

Description of Change

This change reformats the procedure section to ensure that Primary Containment Spray is normally implemented. Figure 6.1 is added to illustrate fire water containment spray flow paths.

Reason for Change

This change was implemented to simplify procedural sections and steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number | Title

EOP 590.10, Rev. 2 — Shifting Low Pressure Coolant Injection (LPCI) Pump Suctions from the Torus to the Condensate Storage Tank (CST)

Description of Change

This revision incorporates those changes to the Emergency Operating Procedure (EOP) Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the EOP's supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.11, Rev. 2	Shifting Core Spray (CS) Pump Suctions from the Torus to the Condensate Storage Tank (CST)

Description of Change

This revision incorporates changes from the Emergency Operating Procedure Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.12, Rev. 1	Primary Containment Flooding Using Emergency Service Water (ESW)

Description of Change

This change retitles the procedure and divides it into two sections. One section for Torus alignment, and the other section for Condensate Storage Tank alignment.

Reason for Change

This change was implemented to simplify procedural steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.14, Rev. 3	Bypassing Group 1 Isolation Signal on Reactor Pressure Vessel (RPV) Low-Low Water Level

Description of Change

This change deletes steps to close or verify closed IC-6&7 and RR-36 and RR-37 prior to bypassing the Group 1 isolation signal.

Reason for Change

This change was implemented to reduce the time necessary to bypass the Low-Low water level isolation.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number

Title

EOP 590.15, Rev. 2 Bypassing Group 4 Isolation Signals

Description of Change

This revision changes the sequence in which IC-2, IC-1, IC-3 and IC-4 are opened.

Reason for Change

This change was implemented to optimize the capability of IC-1, IC-2, IC-3 and IC-4 versus the operating conditions these valves experience.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.16, Rev. 3	Bypassing Cleanup System Isolation Signals

Description of Change

This revision incorporates changes from the Emergency Operating Procedure (EOP) Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the EOP's supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title

Procedure Number	Title
EOP 590.17, Rev. 3	Bypassing Reactor Building Heating, Ventilation, and Air Conditioning (HVAC) Group 2 Isolation Signals

Description of Change

This revision incorporates changes from the Emergency Operating Procedure (EOP) Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the EOP's supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.18, Rev. 2	Alternate Methods of Control Rod Insertion

Description of Change

This change prioritizes sections for the performance of alternate rod insertion. Additional steps have been included to only de-energize channels of Anticipated Transients Without Scram (ATWS) divisions "C" and "D," to allow monitoring of ATWS panel Reactor Pressure Vessel level and pressure indications.

Reason for Change

This change was implemented to simplify procedural steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.19, Rev. 3	Bypassing All Group 1 Isolation Signals Except High-High Steam Line Radiation

Description of Change

This revision incorporates changes from the Emergency Operating Procedures (EOP) Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the EOP's supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.20, Rev. 3	Bypassing All Group 1 Isolation Signals

Description of Change

This revision incorporates changes from the Emergency Operating Procedure (EOP) Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the EOP's supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number Title

EOP 590.23, R v. 4 Containment Vent Procedure

Description of Change

This revision incorporates changes from the Emergency Operating Procedure (EOP) Writers Guide which was modified to include comments from the verification and validation process.

Reason for Change

This change was processed to make the procedure consistent with the EOP Writers Guide.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the EOP's supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.24, Rev. 1	Alternate Shutdown Cooling

Description of Change

This change was implemented to ensure closure of IC-1, IC-2, IC-3 and IC-4 prior to flooding the Reactor Pressure Vessel above the Isolation Condenser line.

Reason for Change

This change was implemented to simplify procedural steps for ease of performance.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.26, Rev. 5	Containment Cooling During Accident Conditions

Description of Change

This revision incorporates minor editorial changes.

Reason for Change

This change was processed to expedite implementation of Containment Cooling.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
EOP 590.27, Rev. 5	Containment Cooling During Anticipated Transients Without Scram (ATWS) Conditions

Description of Change

This revision incorporates minor editorial changes.

Reason for Change

This change was processed to expedite implementation of Containment Cooling.

Safety Evaluation

The procedure changes will not increase the probability of occurrence of an accident since the Emergency Operating Procedure's (EOP) supplemental procedures are implemented via the EOPs which are entered after an accident has already occurred.

The procedure changes will not increase the probability of a malfunction of equipment important to safety. EOP 590.1, EOP 590.8, and EOP 590.26 would be implemented following a design Basis Accident (DBA). These procedures continue to use the systems within the specified design requirements.

The procedure changes will not increase the consequences of an accident previously evaluated since the systems are expected to function as assumed in the main EOP Flowcharts.

The procedure changes will not increase the consequences of a malfunction of equipment important to safety since the Verification and Validation process has concluded that these systems are expected to function in the manner assumed in the main EOP Flowcharts.

The procedure changes will not create an accident different from those previously analyzed since adequate DEFENSE IN DEPTH is provided. In addition, the systems and components are expected to function as described in the main EOPs.

The procedure changes will not cause a different malfunction of equipment important to safety than those already considered since the systems which would be used following a DBA (EOP 590.1, EOP 590.8, EOP 590.10, and EOP 590.26) will not cause any new malfunction of equipment important to safety.

The procedure change will not reduce the margin of safety as defined in the basis for any technical specification. The margin of safety of the accident mitigation strategy is directed by the main EOP Flowcharts (EOPs 570, 575, 580, and 585).

Procedure Number	Title
SP 92-1-35, Rev. 0	Low Pressure Coolant Injection (LPCI) System Check Valve Closing Procedure

Description of Change

This procedure was developed to provide the steps necessary to close the LPCI pump discharge check valves if they are found leaking.

Reason for Change

A leaking LPCI pump discharge check valve will result in the containment spray header no longer being filled. This procedure will facilitate the reseating of the previously unseated check valve, thus maintaining proper keepfill pressure and assuring the containment spray headers are filled.

Safety Evaluation

The performance of this procedure does not challenge any primary system boundary.

This procedure requires double verification of the proper isolation and subsequent restoration of the pump, check valve and automatic pressure relief switch.

During periods where system alignment differs from normal conditions, the LPCI and Automatic Pressure Relief systems will operate as designed and as described in design documents.

Procedure Number	Title
SP 93-1-01, Rev. 0	Bypassing Main Steamline Radiation Detectors While Placing Condensate Demineralizer in Service Following Ultrasonic Resin Cleaning (URC)

Description of Change

This procedure was created to allow the Main Steam Line Radiation Monitors trip function to be bypassed while a condensate demineralizer is placed in service following a URC, during power operation.

Reason for Change

Recent plant experience indicates that a spurious plant trip may be initiated due to undetermined causes while placing the condensate demineralizers in service. This procedure will improve plant reliability by reducing the probability of unnecessary plant transients resulting from inadvertent Main Steam Isolation Valve (MSIV) closures at 100 percent power.

Safety Evaluation

The probability of fuel failure during the time the Main Steam Line Radiation Monitors trip function will be bypassed is low.

The Off-Gas Radiation monitor will ensure proper containment of any main steam activity, should a fuel failure occur during the time the Main Steam Line Radiation Monitors trip function is bypassed. Furthermore, the control room operator is directed to scram the reactor, close the Off-Gas Isolation Valve and close the MSIVs if the trip setpoints are exceeded for both the Main Steam Line Radiation Monitor and the Off-Gas Radiation Monitor.

The Millstone Unit No. 1 design basis accident analysis does not take credit for the Main Steam Line Radiation Monitors trip function; therefore, bypassing the trip and isolation functions has no affect on the consequences of previously analyzed accidents.

Procedure Number	Title
SP 93-1-03, Rev. 0	Penetration X-25/202D Online Leak Rate Test

Description of Change

This procedure was created to address the performance of a Local Leak Rate Test (LLRT) on Penetration X-25/202D while the plant is at power.

Reason for Change

Existing procedures address the performance of LLRTs while the reactor is not at power. No procedures were available for performing the test with the plant at power. Performing the test at power was required to meet an United States Nuclear Regulatory Commission commitment in lieu of performing the complete test program at the interval required by the Technical Specifications.

Safety Evaluation

The basis for the Plant Technical Specifications allow plant operation with only one of the two (inner and outer) test boundaries performing the isolation function. The inner boundary will be in position to perform this function (isolation valves will be closed).

The amount of air required to perform this test is far less than that required to significantly affect the oxygen concentration in the drywell. Furthermore, administrative controls in the procedure require securing the LLRT when the oxygen concentration reaches 3%. The test will also be terminated if drywell pressure falls below 1 psig or reaches 1.2 psig.

Procedure Number	Title
SP 93-1-04, Rev. 0	On-Line Local Leak Rate Test (LLRT) - Electrical Penetrations/Control Rod Drive Hatch

Description of Change

This procedure was created to address the performance of LLRTs on selected primary containment electrical penetrations and the double gasketed Control Rod Removal Hatch while the plant is at power.

Reason for Change

Existing procedures address the performance of LLRTs when the reactor is not at power. No procedures were available for performing the tests at power.

Safety Evaluation

The basis for the Plant Technical Specifications allows plant operation with only one of the two (inner and outer) test boundaries performing the isolation function. The inner boundary will be in position to perform this function.

The amount of air required to perform these tests is far less than that required to significantly affect the oxygen concentration in the drywell atmosphere. Furthermore, the Containment Atmospheric Control System is available for purging and venting the drywell to ensure that the Technical Specification for oxygen concentration is not violated. The oxygen concentration will be monitored during pressurization of the penetrations.

Procedure Number	Title
SP 93-1-11, Rev. 0	Intake Structure Loss of Ventilation Test

Description of Change

This procedure was created to document the steps required to perform a test which measures Intake Structure ambient temperature with normal ventilation out of service.

Reason for Change

The test is being performed to benchmark calculations which were performed to disposition Reportability Evaluation Form 93-45, "Intake Structure Ventilation."

Safety Evaluation

During performance of this test, ambient temperatures will be monitored. If temperatures reach the maximum allowable temperatures for the equipment housed in the Intake Structure (Motor Control Center, Service Water Pumps, Emergency Service Water Pumps, Circulatory Water Pumps), the test will be terminated and ventilation will be restored. In the event the exhaust fans fail to restart, the Circulating Pumps will be secured and doors and hatches opened if necessary.

This test will be performed when the plant is in cold shutdown. The only design basis accident analyzed for this condition is the fuel handling accident. The temperature testing will have no impact on this accident or its consequences.

Procedure Number	Title
SP 93-1-13, Rev. 0	Functional Test of Reactor Protection System (RPS) Scram Logic Associated with Loss of Normal Power (LNP)

Description of Changes

This procedure was created to provide the steps to functionally test the RPS scram logic associated with an LNP to the 4160 Volt Busses 14A, 14B, 14C and 14D.

Reason for Change

This test procedure was created to help determine the cause of a recent reactor scram.

Safety Evaluation

The test will be performed with the plant in cold shutdown and all control rods fully inserted into the reactor vessel core; thus, an inadvertent full scram signal will not result in the damaging of its associated components.

Contacts that may affect plant conditions will be jumped or blocked so that associated equipment will not be affected.

The only analyzed accident applicable to the cold shutdown condition is the fuel handling accident. Implementation of this procedure will have no effect on the probability or consequences of a fuel handling accident.

This test will be performed on the bus undervoltage logic associated with RPS. This portion of the logic will not affect the portion of the logic which would initiate an LNP.

Operators will be prepared to transfer equipment to an alternate bus should a loss of bus voltage occur.

Procedure Number	Title
SP 93-1-17, Rev.0	Replace Breaker in 4160 Volt Bus 14C to Reserve Station Service Transformer (RSST) Cubicle

Description of Change

This procedure was created to provide the steps necessary to remove and replace the breaker in the 4160 Volt Bus 14C to RSST cubicle and to provide guidance to restore busses 14C and 14E from the RSST, Emergency Station Shutdown Transformer (ESST), and the Gas Turbine if needed.

Reason for Change

This change provides a written procedure for the removal and replacement of the breaker. The procedure provides a consistent, approved series of steps for a sequence of steps previously performed without a written procedure.

Safety Evaluation

Measures have been incorporated into the procedure to prevent overload conditions.

This procedure will only be implemented during cold shutdown. The only applicable analyzed accident for this plant mode is the fuel handling accident. This procedure will not affect the probability or consequences of this accident, since the reactor head will not be removed and no fuel pool activities will be ongoing.

In the event of a Loss of Normal Power, all electrical line-ups will be restored to normal.

Procedure Number	Title
SP 93-1-20, Rev. 0	Service Water Piping Replacement

Description of Change

This procedure provides the steps required to safely replace the service water piping downstream of the "A" and "B" Reactor Building Closed Cooling Water (RBCCW) heat exchangers.

Reason for Change

This procedure was created to provide a written, approved and consistent sequence of steps to replace the service water piping. Previously, no written procedure existed.

Safety Evaluation

The service water will remain in operation throughout the procedure. In the unlikely event that service water to the RBCCW heat exchangers is secured, adequate defense in depth is provided for decay heat removal, reactor vessel inventory control, containment control and electrical power source capabilities.

Implementation of this procedure requires that the plant be in cold shutdown. The only design basis accident applicable to a shutdown condition is a fuel handling accident. It is not applicable to implementation of this special procedure since fuel handling will not take place during the implementation of this procedure.

Procedure Number	Title
SP 93-1-25, Rev. 0	Implementation of Final Feedwater Temperature Reduction During Cycle 14 Coastdown

Description of Change

This procedure was created to provide guidance for removing the high pressure heaters from service during Cycle 14 coastdown.

Reason for Change

Final Feedwater Temperature Reduction (FWTR) provides a method for extending the operating cycle beyond normal end-of-full-power life.

Safety Evaluation

A generic safety analysis has been performed by General Electric and approved by the Nuclear Regulatory Commission. It analyzed the effects of FWTR on loss-of-coolant, rod withdrawal error, rod drop accident, fuel loading error, turbine trip, generator load rejection without bypass and feedwater controller failure, and showed that FWTR is not limiting on BWR-3 plants. This mode of operation, therefore, is within the licensing basis of the plant and does not constitute a design change or a license change.

Procedure Number	Title
SP 93-1-26, Rev. 0	Spent Fuel Pool (SFP) Filter Backwash Line 6"-RWR-45

Description of Change

This procedure was created to provide the requirements for performing a pressurized fluid backwash test of the SFP filter M4-24.

Reason for Change

Current operating procedures provide instructions for performing an air backwash of the filter. This special procedure tests a change in this process. The valve open/close sequence is changed such that the water contained within the filter is expelled in addition to the injected air. It is expected that this change will improve the effectiveness of the backwash. The test will help determine the need for additional piping support.

Safety Evaluation

The components related to the backwash system are categorized as non-Quality Assurance, non seismic. There is no direct interface between these components and equipment important to safety.

Piping loads have been determined to be much lower than those experienced previously and the capacity of the piping; thus, there is no potential for interaction with nearby safety related equipment.

In the unlikely event of a piping rupture, the volume of radioactive material released is limited to 2.8 cubic feet. This material will be contained and managed using standard Health Physics (HP) procedures. Any accumulation of fluid in piping low points may result in radioactive hot spots. These can be handled using standard HP procedures.

STAND ALONE SAFETY EVALUATIONS

Safety Evaluation Number	Title
No Number	Safety Evaluation in Support of a Change to Technical Specification Basis 3.5.C - Feedwater Coolant Injection Subsystem
No Number	Safety Evaluation in Support of a Change to Fire Protection Technical Requirements - Revision 1
No Number	Safety Evaluation in Support of Changes to Technical Specification Basis 3.5.A, Core Spray and Low Pressure Coolant Injection (LPCI), and Basis 3.5.B, Containment Cooling Subsystems

Safety Evaluation Number	Title
No Number	Safety Evaluation in Support of a Change to Technical Specification Basis 3.5.C - Feedwater Coolant Inspection Subsystem

Description of Safety Evaluation

The changes to Bases Section 3.5.C clarified the basis for Feedwater Coolant Injection (FWCI) Subsystem operation, and identified the components comprising the FWCI Subsystem.

Reason for Safety Evaluation

The updated basis clarifies that the design basis of the FWCI Subsystem is to minimize the probability of core uncovery, and to ensure that the FWCI Subsystem can provide flow to the core at high pressure in scenarios such as the closure of a main steamline isolation valve or small break loss of coolant accident (LOCA). The changes clarify the redundant emergency core cooling systems (i.e, LPCI, core spray, automatic pressure relief, and isolation condenser) which are available to mitigate any size LOCA, should FWCI fail.

Safety Evaluation

The basis change did not impact the manner in which FWCI is operated or maintained. Additionally, the way the FWCI Subsystem functions was not modified (i.e., the system flowrates, start times, and supporting systems were not changed). Clarification of the bases neither degrades the performance nor increases the probability of failure of the FWCI subsystem.

Safety Evaluation Number	Title
No Number	Safety Evaluation in Support of a Change to Fire Protection Technical Requirements - Revision 1

Description of Safety Evaluation

This revision to the Fire Protection Technical Requirements changed two sections of fire watch requirements, one surveillance requirement, and other non-intent changes.

Reason for Safety Evaluation

The fire protection requirements were modified to more accurately reflect the actions necessary to ensure safe operation of the plant against fire hazards.

Safety Evalaution

The changes to the fire protection requirements were commensurate with standard industry accepted practices. The new requirements ensure adequate fire protection for all areas of the plant for all situations. These changes do not reduce the availability or reliability of systems associated with achieving and maintaining safe shutdown conditions when considering the effects of a fire.

Safety Evaluation Number	Title
No Number	Safety Evaluation in Support of Changes to Technical Specification Basis 3.5.A, Core Spray and Low Pressure Coolant Injection (LPCI), and Basis 3.5.B, Containment Cooling Subsystems

Description of Safety Evaluation

The changes to Bases Section 3.5.A and 3.5.B remove the limitation of only utilizing the "A" and "B" LPCI pumps for core and containment cooling in the long-term.

Reason for Safety Evaluation

The bases needed to be revised to reflect the replacement of the water-cooled LPCI pump motors ("C" and"D") with air-cooled motors qualified to provide long-term post-accident core and containment cooling.

Safety Evaluation

As a result of LPCI pumps "C" and "D" being equipped with air-cooled motors, all the LPCI pumps can be used for core and containment cooling in the long-term and the limitation of only using the "A" and "B" pumps is no longer needed. The change also deletes the interpretation that the corresponding containment cooling subsystem is considered inoperable and in a 4-day limiting condition for operation (LCO) when the "A" or "B" LPCI pump is inoperable. This requirement was in place since the long-term post-accident operation of water-cooled LPCI pumps could not be assured. Since all four LPCI pumps are now equipped with air-cooled motors, the four day LCO for pumps "A" and "B" is no longer needed and the 30-day LCO stated in the Technical Specifications is appropriate.

JUMPERS-LIFTED LEADS-BYPASSES

J-LL-B	Title
1-93-023	"A" Recirculation Motor Generator (MG) Set Lube Oil System Portable Fluid Purifier
1-93-025	"B" Recirculation Motor Generator (MG) Set Lube Oil System Portable Fluid Purifier
1-93-038	Refueling Bridge and Platform Inspection
1-93-047	Auxiliary Cleanup Pump Temperature Element Jumper

Jumper-Lifted Lead-Bypass Change Number 1-93-023

This bypass jumper, entitled, "'A' Recirculation Motor Generator (MG) Set Lube Oil System Portable Fluid Purifier," has been removed.

Description of Jumper-Lifted Lead-Bypass

A portable fluid purifier had been installed on the "A" Recirculation Pump MG Set Lube Oil System sump. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This installation reduced the amount of entrained particulate matter and water which could result in lube oil pump suction strainer failing.

Safety Evaluation

The portable fluid purifier and associated hoses meet or exceed the normal MG set operating temperature and pressure requirements. Failure of the pressure retaining capability of the fluid purifier and hoses, if not detected, could result in a trip of the MG set on low lube oil pressure. However, this event (considered to be of moderate frequency) is bounded by existing accident analyses and the consequences of such an event would not be increased. Plant personnel have been instructed to secure and isolate the fluid purifier in the event of any malfunctions.

The portable fluid purifier was installed to remove impurities from the MG set lube oil system. This improves MG set reliability and reduces downtime resulting from plugged lube oil pump suction strainers.

Jumper Lifted Lead-Bypass Change Number 1-93-025

This bypass jumper, entitled, "'B' Recirculation Motor Generator (MG) Set Lube Oil System Portable Fluid Purifier," has been removed.

Description of Jumper-Lifted Lead-Bypass

A portable fluid purifier had been installed on the "B" Recirculation Pump MG Set Lube Oil System sump. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This installation reduced the amount of entrained particulate matter and water which could result in lube oil pump suction strainer failing.

Safety Evaluation

The portable fluid purifier and associated hoses meet or exceed the normal MG set operating temperature and pressure requirements. Failure of the pressure retaining capability of the fluid purifier and hoses, in not detected, could result in a trip of the MG set on low lube oil pressure. However, this event (considered to be of moderate frequency) is bounded by existing accident analyses and the consequences of such an event would not be increased. Plant personnel have been instructed to secure and isolate the fluid purifier in the event of any malfunctions.

The portable fluid purifier was installed to remove impurities from the MG set lube oil system. This improves MG set reliability and reduces downtime resulting from plugged lube oil pump suction strainers.

Jumper-Lifted Lead-Bypass Change Number 1-93-038

This bypass jumper entitled, "Refueling Bridge and Platform Inspection," has been removed.

Description of Jumper-Lifted Lead-Bypass

The limit switch arms for two refueling bridge position switches have been removed. These switches initiate control rod blocks when the refueling bridge platform is positioned over the reactor vessel in order to prevent inadvertent criticality during refueling operations. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

Implementation of this bypass jumper allowed the refueling bridge platform to be moved over the reactor vessel while at power to permit the performance of refueling bridge and platform inspections.

Safety Evaluation

The sole purpose of the two refueling bridge position switches was to prevent inadvertent criticality during refueling operations. These switches have no effect on any accidents or the performance of any equipment important to safety when the plant is in any mode of operation other than refueling. Reinstallation of the limit switch arms prior to refueling was assured by an operating procedure which requires testing of the refueling interlocks prior to fuel movement.

Jumper-Lifted Lead-Bypass Change Number 1-93-047

This bypass jumper, entitled, "Auxiliary Cleanup Pump Temperature Element Jumper," has been removed.

Description of Jumper-Lifted Lead-Bypass

The temperature indicating switch, which monitors the cooling water temperature on the Auxiliary Cleanup Pump bearing housing and isolates the cleanup system on high cooling water temperature, was jumpered. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The current location of the temperature indicating switch wiring and conduit interferes with the removal of the spent fuel pool heat exchangers. This bypass jumper permitted the disconnection and rerouting of this wiring and conduit without generating a cleanup system isolation signal.

Safety Evaluation

The Reactor Water Cleanup System is used during operations to maintain reactor water purity. During power operation, the system uses the Main Cleanup Pump. While the plant is being shut down, the Auxiliary Cleanup Pump is put into service because of insufficient net positive suction head for the main pump (reactor pressure below 100 psig). The temperature indicating switch monitors the cooling water temperature on the Auxiliary Cleanup Pump bearing housing and isolates the cleanup system on high cooling water temperature.

It has been determined that the Auxiliary Cleanup Pump is not needed until six hours after the event that would require the plant to be brought to a cold shutdown. It is expected that the disconnection of the wiring and its removal from the conduit, followed by reconnection of the wire in its original configuration, would require less than three hours, well within the time window available should the pump be required.

The installation of this bypass jumper will not have any effect on the Cleanup System valves required for containment isolation on a group 5 isolation signal.

Existing operating procedures deal with high conductivity in the reactor water.

TESTS

Test Number	Title
T-93-1-01, Rev. 0	Degraded Voltage Testing of Motor Operated Valve (MOV) Brakes

Test Number	Title
T-93-1-01, Rev. 0	Degraded Voltage Testing of Motor Operated Valve (MOV) Brakes

Description of Test

The procedure covered the testing of MOV brakes installed on 1-LP-43A, 1-LP-43B, and 1-CS-43. The test involved applying a degraded voltage to the brake coil to verify pick-up during degraded voltage conditions.

Reason for Test

This test was performed to ensure brakes can energize and allow the valve to operate in a degraded voltage condition.

Safety Evaluation

The test was performed on one valve at a time, and the associated Low Pressure Coolant Injection loop under test was declared inoperable, and the Technical Specification Limiting Condition for Operation was entered.

EXPERIMENTS

There were no experiments performed under the provisions of Title 10, Code of Federal Regulations, Section 50.59 during 1993.

CHALLENGES TO RELIEF/SAFETY VALVES

In accordance with the commitment made under Item II.K.3.3 of NUREG 0737 (TMI Action Plan) in the W. G. Counsil letter to D. G. Eisenhut dated June 10, 1980, the following is a report of challenges to relief/safety valves during 1993.

There were no challenges to relief/safety valves during 1993.

PRIMARY COOLANT IODINE SPIKING

During 1993, the specific activity of the primary coolant did not exceed the limits stated in the Technical Specifications.

REGULATORY GUIDE 1.16 REPORT FOR 1993

REGULATORY GUIDE 1.16 REPORT FOR 1993
NORTHEAST NUCLEAR ENERGY CO. UNIT 1

DATE: 2/ 3/94

WORK & JOB FUNCTION	NUMBER OF PERSONNEL(>100 MREM) STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES	TOTAL MAN-REM STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES
REACTOR OPERATIONS & SURVEILLANCE						
MAINTENANCE PERSONNEL	10	0	2	3.41	0.35	1.70
OPERATING PERSONNEL	24	1	1	6.48	0.20	1.53
HEALTH PHYSICS PERSONNEL	17	0	5	5.21	0.00	1.66
SUPERVISORY PERSONNEL	0	0	0	0.01	0.00	0.01
ENGINEERING PERSONNEL	0	0	1	0.06	0.11	0.30
ROUTINE MAINTENANCE						
MAINTENANCE PERSONNEL	0	0	0	0.27	0.04	0.05
OPERATING PERSONNEL	0	0	0	0.00	0.00	0.00
HEALTH PHYSICS PERSONNEL	0	0	0	0.25	0.00	0.11
SUPERVISOR PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.05	0.00	0.00
INSERVICE INSPECTION						
MAINTENANCE PERSONNEL	0	0	0	0.07	0.00	0.16
OPERATING PERSONNEL	0	0	0	0.06	0.00	0.00
HEALTH PHYSICS PERSONNEL	0	0	0	0.10	0.00	0.01
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	1	1	0	0.20	0.28	0.24
SPECIAL MAINTENANCE						
MAINTENANCE PERSONNEL	34	0	27	11.97	0.18	11.99
OPERATING PERSONNEL	1	0	3	1.30	0.00	0.64
HEALTH PHYSICS PERSONNEL	14	0	4	4.11	0.00	1.81
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.01
ENGINEERING PERSONNEL	2	5	6	0.50	1.59	2.13
WASTE PROCESSING						
MAINTENANCE PERSONNEL	0	0	0	0.04	0.00	0.22
OPERATING PERSONNEL	1	0	1	0.35	0.01	0.18
HEALTH PHYSICS PERSONNEL	3	0	8	1.46	0.00	2.35
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.00	0.05	0.00
REFUELING						
MAINTENANCE PERSONNEL	0	0	0	0.00	0.00	0.00
OPERATING PERSONNEL	0	0	0	0.00	0.00	0.00
HEALTH PHYSICS PERSONNEL	0	0	0	0.00	0.00	0.00
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.00	0.00	0.00
TOTAL						
MAINTENANCE PERSONNEL	44	0	29	15.77	0.57	14.13
OPERATING PERSONNEL	26	1	5	8.21	0.21	2.36
HEALTH PHYSICS PERSONNEL	34	0	17	11.14	0.00	[illegible].94
SUPERVISORY PERSONNEL	0	0	0	0.01	0.00	0.02
ENGINEERING PERSONNEL	3	6	7	0.81	2.04	2.68
GRAND TOTAL	107	7	58	35.94	2.83	25.13

MILLSTONE UNIT NO. 2

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INTRODUCTION

None of the plant design changes, procedure changes, jumpers-lifted leads-bypasses, tests or experiments described herein constitute (or constituted) an unreviewed safety question per the criteria of 10CFR50.59.

PLANT DESIGN CHANGES

PDCE Number	Title
MP2-89-088	Energy, Inc., Isolator Model Change
MP2-90-034	Replacement of Check Valve Retaining Block Bolts, Low Pressure Safety Injection (LPSI), High Pressure Safety Injection (HPSI), Containment Spray (CS) and Spent Fuel Pool Cooling and Cleanup (RW) Valves

PDCR Number	Title
2-015-87	Backfeed Interlock Switch, Millstone Unit Nos. 1 and 2
2-007-88	Auxiliary Feedwater System Steam Bypass Line
2-021-89	Charging Pump Discharge Relief Valve Flanges
2-001-90	Steam Jet Air Ejector Radiation Monitor Replacement
2-027-91	Replacement of Two Rotor Limit Switches with Four Rotor Limit Switches in Limitorque Motor Operators
2-036-91	Millstone Unit No. 2 Steam Generator Replacement
2-054-91	Replace Solenoid Operated Valves for Reactor Building Closed Cooling Water (RBCCW) System Pump Header Suction Valves
2-061-91	Steam Generator Replacement Project Wide Range Level Replacement
2-071-91	Loose Parts Monitor Upgrade
2-107-91	Line Service Water Strainer Bottom Cover with PVC
2-115-91	Drains Cooler Replacement Project
2-124-91	Electro-Hydraulic Control (EHC) System - Control Valve Pressure Switches Relocation
2-006-92	Replacement of ASCO Solenoid Valves Under-rated for the Instrument Air System Pressure
2-024-92	Control Element Assembly (CEA) Position Display System Replacement (Metrascope Replacement)
2-025-92	Main Generator Pilot Wire Circuit Modifications

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
2-028-92	Containment Pressure Signal Addition to Main Steam Isolation (MSI) Function
2-030-92	Installation of Pressure Gauges in Service Water System
2-037-92	Replacement of Heater Drains System Spargers
2-068-92	Cycle 12 Reload of Fuel
2-078-92	Reactor Protection System (RPS) Pressurizer Pressure Alarm Modification - Power Operated Relief Valve (PORV) Actuation Logic Reversal
2-079-92	Condensate Storage Tank (CST) Nitrogen Blanket
2-094-92	Reactor Coolant Loop Resistance Temperature Detectors (RTD) Termination Modification
2-103-92	Heating, Ventilation, and Air Conditioning (HVAC) Control Cabinet Air Supply Relief Valves
2-112-92	Static O-Ring Pump Suction Pressure Switch Replacement in the Chemical and Volume Control System (CVCS) and Reactor Building Closed Cooling Water (RBCCW) System
2-114-92	Required Modifications for Main Steam Line Break Scenario
2-181-92	Chemical Feed to Tank T72 - Ammonium Hydroxide Solution Tank
2-186-92	Replace Service Water Pump P5C Discharge Head
2-189-92	Diesel Generator Room Temperature Setpoint Change
2-208-92	Coat Service Water Pump P5C Columns with Belzona
2-005-93	2-HV-171 Motor Replacement

Plant Design Change Evaluation Number MP2-89-088

This change, entitled "Energy, Inc., Isolator Model Change," is complete.

Description of Change

This change replaced Energy Incorporated series SC-993 analog isolation devices with improved series SCA-101 devices. This change applied to containment and main steam line radiation monitors, the containment hydrogen analyzers and the reactor coolant system loop number 1 temperature monitors.

Reason for Change

This change was required due to a high failure rate experienced with the series 993 devices, caused by the elevated ground potential at Millstone Unit 2. The new isolators were modified by the manufacturer to operate with this elevated ground.

Safety Evaluation

The new isolators were tested and certified to exceed the safety criteria requirements of Millstone Unit 2. In all cases, the application of the credible failures to the isolators' non-safety related output had an insignificant affect on the safety related input. The new isolators are the same size and weight as the replaced units, and were certified by the manufacturer as "usable as a direct replacement."

Plant Design Change Evaluation Number MP2-90-034

This change, entitled "Replacement of Check Valve Retaining Block Bolts, Low Pressure Safety Injection (LPSI), High Pressure Safety Injection (HPSI), Containment Spray (CS) and Spent Fuel Pool Cooling and Cleanup (RW) Valves," is complete

Description of Change

This change installed substitute retaining block hold down bolts/studs (and/or clapper arm studs) in several LPSI, HPSI, CS and Spent Fuel Pool Cooling system check valves.

Reason for Change

This change addressed concerns about the susceptibility of the original material to intergranular stress corrosion cracking. The replacement bolts/studs were approved for use by the valve manufacturers.

Safety Evaluation

This change was entirely internal to the affected check valves. The function of the fasteners, which were replaced, is to hold the hinge pin in place (or hold together the swing arm) in a swing check valve. By reducing the susceptibility of the fasteners to intergranular stress corrosion, the change increases the reliability of the valves to function as designed. The change has no impact on any analyzed scenario. Due to the relatively simple nature of the change, it does not create the possibility of an accident or malfunction of a different type than any previously evaluated, nor reduce the safety margin as defined in any technical specifications.

Plant Design Change Record Number 2-015-87

This change, entitled "Backfeed Interlock Switch, Millstone Unit Nos. 1 and 2," is complete.

Description of Change

This change added switches in the 4.16KV main feeder breaker cubicle which defeated interlocks associated with the main feeder breaker and two tie feeder breakers.

Reason for Change

The original purpose of the interlocks was to prevent a crosstie of Unit 2 AC supplies with the Unit 1 Reserve Station Service Transformer. Since that time, a separate change installed an isolation breaker between the backfeed and the Unit 1 Reserve Station Service Transformer, so the crosstie concern is no longer valid. Removal of the interlock function allows Unit 2 to supply power to Unit 1 using the Normal Station Service Transformer, the Reserve Station Service Transformer or the diesel generator. The change fulfills requirements of Appendix R.

Safety Evaluation

This change has no impact on Unit 2. There are no credible failure Modes associated with the change since installation of the switches will not affect protective trip functions of any of the associated breakers. The sole purpose of this change is to allow full use of the Appendix R Backfeed. The result is that the reliability of the onsite power system at Unit 1 is increased.

Plant Design Change Record Number 2-007-88

This change, entitled "Auxiliary Feedwater System Steam Bypass Line," is complete.

Description of Change

This change added a bypass line around the two auxiliary feedwater pump steam supply isolation valves. This change also added pressure indicators to the auxiliary feedwater steam supply piping.

Reason for Change

The bypass line provided a drainage path for condensate that collects in the line during plant start-up and normal operation when one steam supply isolation valve is shut. The pressure indicators are used during in-service leak testing of the auxiliary feedwater system steam supply check valves.

Safety Evaluation

The bypass line was added to improve reliability of the main steam and auxiliary feedwater systems by reducing the potential for water hammer damage during operation of the steam supply isolation valves. This line is isolated during normal plant operation. It sees only intermittent service under controlled conditions, with an operator present whenever the valve is in use. There is no credible mechanism by which this change could impact any analyzed scenario or create a new, unanalyzed scenario.

The pressure indicators are only used during in-service testing, and are installed in a non-safety related section of the system.

Plant Design Change Record Number 2-021-89

This change, entitled "Charging Pump Discharge Relief Valve Flanges," is complete.

Description of Change

This change substituted a lap-joint flange for one of the pair of socket weld flanges on the inlet side of the charging pump discharge relief valves.

Reason for Change

The charging pump relief valves were assembled by threading the valve internals (to which the inlet side piping is welded) into the valve body. This threaded joint does not necessarily reach the proper torque value at the same relative rotation each time, making bolt hole alignment difficult during reassembly. The new flanges are free to rotate, reducing the time, expense and exposure needed to perform valve maintenance.

Safety Evaluation

This change had no impact on any analyzed scenario. Leakage at the relief valves could not result in a depressurization or loss of coolant type accident, because each train is isolated from the reactor coolant system by a check valve and a manual isolation valve. No other type of failure resulting from this change is credible.

Plant Design Change Record Number 2-001-90

This change, entitled "Steam Jet Air Ejector Radiation Monitor Replacement," is complete.

Description of Change

This change involved replacement of existing radiation monitor components (detector/shield assembly and control room electronics) with microprocessor electronics.

Reason for Change

Equipment upgrade.

Safety Evaluation

This is a non-Quality Assurance radiation monitor. It is a "trend" monitor, designed to provide early warning of a primary to secondary leak (steam generator tube leak). It is backed up by an additional monitor, which provides the same information and steam generator blowdown isolation signal. Seismic integrity and electrical separation were maintained during installation of this change. The change has no impact, direct or indirect, on any analyzed scenario.

Plant Design Change Record Number 2-027-91

This change, entitled "Replacement of Two Rotor Limit Switches with Four Rotor Limit Switches in Limitorque Motor Operators," is complete.

Description of Change

This change replaced 2 rotor geared limit switches with 4 rotor geared limit switches on 45 safety related motor operated valves. It also installed larger limit switch covers on some of the smaller actuators, to accommodate the larger 4 rotor geared limit switches.

Reason for Change

This change improved the performance of the safety related valves by separating the torque switch by-pass function from the remote valve position indication. These two functions are now controlled from separate rotors, rather than from the same rotor. This allows the valve position indication to be set to indicate full open or closed, and the torque switch by-pass to be set as required to ensure that the valve won't torque out and stop prematurely.

Safety Evaluation

There was no change in the basic design or function of any systems or components. There was no control logic change to the actuators of any of the motor operated valves involved in this change. All wiring changes were made using Quality Assurance wire and enviromentally qualified connections and splices. The actuator weight increases were evaluated and found to have no adverse affect on the seismic qualification of any of the piping systems involved. The failure of any one of the modified motor operated valves would have no impact on any analyzed scenario, since that would fall within the single failure criterion; i.e., unaffected redundant equipment would be available.

Plant Design Change Record Number 2-036-91

This change, entitled "Millstone Unit No. 2 Steam Generator Replacement," is complete.

Description of Change

This change replaced both steam generators. The replacement component included the channel heads, tube sheet with the tube bundle and the secondary shell up to the midpoint on the transition cone. A girth cut separated the steam drum from the lower assembly. The steam drums were then inverted and refurbished with new dryers and separators as well as a new feedwater ring. Several permanent changes were made in support systems to enhance the operational and maintenance capabilities of the new steam generators. Also, a number of temporary modifications had to be made to support the physical removal of the old generators and installation of the new units.

Reason for Change

The steam generators had experienced various types of degradation, in the form of denting due to tube support corrosion, pitting due to localized corrosive attack on the external tube surfaces and cracking due to high stress in an aggressive environment. The decision to replace the steam generators was made to : (1) reduce total radiation dose to workers involved in the testing and preventive measures utilized with the existing steam generators, (2) increase unit reliability by minimizing the frequency of forced outages, and (3) increase unit output by recovering heat transfer losses caused by the large quantity of plugged tubes.

Safety Evaluation

The Safety Evaluation and the Integrated Safety Evaluation prepared by Northeast Utilities personnel in support of this change are over 50 pages in length. These supplement the evaluations, studies and calculations performed by Northeast Utilities, Combustion Engineering and other specialty contractors in preparation for this change. The result of this effort was a conclusion that this change had no significant impact on any previously evaluated accidents or safety equipment malfunctions, no potential for a new unanalyzed accident or malfunction, and no impact on the margin of safety. The replacement design was evaluated against the criteria of 10CFR50.59 and was found to involve no unreviewed safety question, nor any change in public risk.

Plant Design Change Record Number 2-054-91

This change, entitled "Replace Solenoid Operated Valves for Reactor Building Closed Cooling Water (RBCCW) Pump Header Suction Valves," is complete.

Description of Change

This change replaced the ASCO solenoid valves on all six of the RBBCW pump inlet valves (2-RB-211A, B, C, D, E, and F).

Reason for Change

The solenoids were replaced with a newer Model due to high failure rate.

Safety Evaluation

The replacement solenoids were identical to the original units in form, fit and function, and certified as Seismic Category 1 by the vendor. There are no new credible failures or failure probabilities or consequences associated with this change.

Plant Design Change Record Number 2-061-91

This change, entitled "Steam Generator Replacement Project Wide Range Level Replacement," is complete.

Description of Change

This change installed four new instruments to monitor steam generator wide range level, provided four new input signals to the plant integrated computer system from the existing feedwater control system level instruments, and installed two new resistance temperature detectors to monitor the steam generator secondary side water temperature.

Reason for Change

This change enhanced the operation of the steam generator instrumentation by improving both the accuracy and displays of the existing equipment, and providing a new wide range level system. It also satisfied the last remaining Control Room Design Review Human Engineering Discrepancy.

Safety Evaluation

This change had no adverse affect, direct or indirect, on the operation of any safety systems, nor any impact on any analyzed scenario. All changes were implemented using proper electrical separation and isolation, and correct seismic restraints. This ensures that operation of the plant systems is not adversely affected, and that a single failure will not degrade redundant trains. This change enhances performance of the steam generator water level control system by:

- Providing new instrumentation which will display steam generator level through the normal and accident ranges.
- Improving the accuracy and response time of the steam generator temperature instrument.
- Providing new displays using the integrated computer system for the existing narrow range level instruments.

The new instrument loops provide control panel indication and integrated computer system displays only. There are no automatic safety related functions associated with this equipment, and redundant displays are available. The plant computer is non-safety related, is not required for a safe plant shutdown and is not utilized in the analysis of any design basis accident.

Plant Design Change Record Number 2-071-91

This change, entitled "Loose Parts Monitor Upgrade," is complete.

Description of Change

This change upgraded the Loose Parts Monitor system.

Reason for Change

This upgrade included the sensors, conduit and cables in containment, penetration feed-throughs in the containment wall, cables in the penetration room up to the charge amplifiers, the charge amplifiers and the monitoring electronics in the control room. An additional improvement included the method of attachment to the reactor vessel and the steam generators.

Safety Evaluation

The Loose Parts Monitor is not a safety-related system, nor is it credited in any of the accident analyses. The only potential affect of this change was in the work performed in its installation. Installation of the accelerometers to the lifting lugs of the reactor vessel and the steam generators, in accordance with approved welding procedures, assured proper welding techniques were employed and appropriate inspections were conducted so that no degradation occurred to the upper head of the vessel or to the steam generators during installation. Installation of the containment electrical penetrations, in accordance with approved procedures, assured that proper techniques were employed and appropriate inspections were conducted so that containment integrity was not degraded during installation of the feed-throughs. Periodic leak testing of the penetrations ensures that containment integrity is maintained. The engineering evaluation concluded that the seismic integrity of the loose parts monitor equipment enclosures, cable trays, and conduit will not be degraded during a design basis earthquake.

Plant Design Change Record Number 2-107-91

This change, entitled "Line Service Water Strainer Bottom Cover with PVC," is complete.

Description of Change

This change lined the bottom covers of the service water strainers with PVC instead of the then existing coal tar epoxy lining.

Reason for Change

The PVC lining is expected to provide better protection for the carbon steel covers, and thereby improve the service water strainer reliability and increase the cover life.

Safety Evaluation

The same PVC coating is currently used as a lining in many service water system spool pieces, and has been used at Millstone Unit 2 since 1988 with good results. The PVC coating exhibits the following positive characteristics: the coating is seam free and baked onto the base material, the bonding agent between the base material and the PVC is stronger than the PVC (precluding delamination of the coating) and PVC is impervious to attack by saltwater and chlorine. Because the strainer bottom covers are on the upstream side of the service water strainer septums, they would not be able to pass through the strainer, and could not affect downstream components, even if the PVC lining does peel or delaminate. Also, because the strainers have an automatic backwash feature, loose material would backwash when the strainer delta P reaches 2 psid. A high delta P alarms in the control room at 4 psig.

Plant Design Change Record Number 2-115-91

This change, entitled "Drains Cooler Replacement Project," is complete.

Description of Change

This change replaced Drains Coolers 7A and 7B. These are two parallel flow heat exchangers in the feedwater heater drains system, located between the sixth point feedwater heaters and the main condenser. In addition to the heater replacement, the sixth point heater level control valves were relocated to a point closer to the main condenser, and a shell side vent was installed on the new drains coolers.

Reason for Change

This change removes the last remaining major copper bearing components from the feedwater system. The tubes in the new drains coolers are stainless steel. The level control valves were relocated as part of an overall plan to correct a two-phase flow condition which existed between the control valves and the main condenser. The new shell side vents were installed to enable operators to ensure that the shell side of the coolers are properly flooded during startup.

Safety Evaluation

Drains coolers 7A and 7B are non-safety related, non-seismic components. Criteria for the procurement, installation, inspection and testing of the replacement heat exchangers and associated piping met or exceeded the original design criteria and all applicable code requirements. Drain cooler replacement did not change operation of the level control valves, nor change any operation of the condensate or heater drain systems. Relocation of the level control valves did not change operation of the sixth point heater level control valves. Installation of the shell side vents did not change operation of the sixth point heater level control valves or operation of the drain coolers. No Technical Specification changes were required as a result of these changes.

Plant Design Change Record Number 2-124-91

This change, entitled "Electro-Hydraulic Control (EHC) System - Control Valve Pressure Switches Relocation," is complete.

Description of Change

This change relocated four pressure switches in the Electro-Hydraulic Control system from the control valves, which the switches service, to a seismic instrument rack.

Reason for Change

The control valves on which the pressure switches were originally mounted are subject to vibration during plant operation. The pressure switches were relocated to preclude the possibility of malfunction due to vibration.

Safety Evaluation

The switch relocation does not affect system design or operation, nor have any impact on the original design basis or any previously reviewed scenario. The new sensing lines are properly supported and routed away from loose parts/equipment and traffic, ruling out any potential for creating a new, unanalyzed accident.

Plant Design Change Record Number 2-006-92

This change, entitled "Replacement of ASCO Solenoid Valves Underrated for the Instrument Air System Pressure," is complete.

Description of Change

This change replaced solenoid valves which had a maximum operating differential pressure of less than 115 psig with valves which are rated at 115 psig or greater.

Reason for Change

The systems involved were upgraded. The upgrade will enable the Category 1 solenoid valves to perform their design function, in the event of a failure of upstream, non-Category 1 regulator valves.

Safety Evaluation

The replacement solenoid valves perform the same function, in the same manner, as the replaced units. The change does not affect the design function of the component valve or damper controlled by the solenoid, nor the design function of any system in which the controlled valve or damper is installed. Each installation was evaluated for its specific requirements, in terms of seismic, electrical separation, environmental qualification, etc., and found to be satisfactory. This change has no impact on any previously evaluated scenario, and does not create any new accident or malfunction.

Plant Design Change Record Number 2-024-92

This change, entitled "Control Element Assembly (CEA) Position Display System Replacement (Metrascope Replacement)," is complete.

Description of Change

This change replaced the existing analog CEA position display system with a new digital system.

Reason for Change

The new equipment replaced an obsolete system that had a poor reliability record. The new system incorporates a digital computer, color monitor and multiple displays to provide an improved read-out scheme for CEA position.

Safety Evaluation

This system (both new and old) does not perform any protection or control functions, and hence is not "important to safety" equipment. The new system provides the same functions as the old system, using the same inputs. The Metrascope system is a backup to manual control of the CEAs by plant operators using the inferred (pulse counting) CEA indication system. Operator recognition of a malfunction is aided by other backup systems, such as the core mimic on the control panel and incore instrumentation. In-place surveillance procedures ensure that system functions are maintained in a normal operating condition. The outcome of malfunctions of the digital and analog systems are identical; hence, there is no impact on any analyzed scenario.

Plant Design Change Record Number 2-025-92

This change, entitled "Main Generator Pilot Wire Circuit Modifications," is complete.

Description of Change

This change modified the main generator primary pilot wire transfer trip scheme to decrease the time it takes for a trip signal from the generator lockout relays to reach the switchyard breakers 8T and 9T.

Reason for Change

As the pilot wire transfer scheme existed prior to this change, specific trip conditions could result in an unstable 345 KV electrical transmission system. This change was designed to maintain distribution system stability for electrical faults in accordance with Northeast Power Coordinating Council criteria.

Safety Evaluation

The pilot wire scheme and generator lockout relays are not safety related. This change does not introduce any new failure Modes, nor will it adversely affect any Class 1E equipment or systems, or any technical specification conditions. This change is an improvement in the electrical protection system.

Plant Design Change Record Number 2-028-92

This change, entitled "Containment Pressure Signal Addition to Main Steam Isolation (MSI) Function," is complete.

Description of Change

This change provided an input from the high containment pressure signal to the MSI Actuation Modules, so that an MSI actuation would occur on either High Containment Pressure or Low Steam Generator Pressure. Also, the Containment Isolation Actuation System (CIAS) automatic close signals to the Feedwater Block Valves 2-FW-42A & B were replaced with the modified MSI signal. As a result of these changes, new auxiliary logic modules for the Automatic Test Insertion (ATI) signal were required to maintain proper operation of the ATI system.

Reason for Change

This change was required to maintain containment design conditions as a result of main steam line break reanalysis, including a range of main steam line breaks which result in an immediate high containment pressure without a low steam generator pressure condition. Specifically, this change provides for:

- Direct trip of the main feed pumps and closure of the feedwater regulating and block valves via the modified safety grade MSI signal.
- Reduction of the pressure differential which the feedwater block valve motor operated valves (MOV) experience during feed isolation, because this change trips both the feedwater pumps and the MOVs with the same signal.
- Elimination of the potential for causing an inadvertent main feed isolation event during CIAS testing.
- Returning the plant to its original configuration, where the main feedwater block and regulating valves and pumps receive the same trip signal.

Safety Evaluation

The change met or exceeded all existing plant design criteria. All failure Modes associated with this change were bounded by the existing plant design. This change did not significantly impact the plant's response to any previously evaluated accident, nor create the possibility of an unevaluated accident. The existing main steam line break analysis was favorably affected by the change. The change did introduce a failure Mode which increased the probability of a loss of feedwater event - a spurious partial main steam isolation resulting in automatic closure of the main feed block valve and the opposite feed regulating valve. This was judged insignificant when compared to all other possible initiators of a loss of feedwater event.

Plant Design Change Record Number 2-030-92

This change, entitled "Installation of Pressure Gauges in Service Water System," is complete.

Description of Change

This change installed 14 pressure taps in the service water system. Five differential pressure gauges were permanently installed, using ten of the 14 taps. The remaining four taps are capped during normal operation, and are used for temporary gauge installation only.

Reason for Change

This change provides the instrumentation to verify service water flow rates in accordance with Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment."

Safety Evaluation

Design specifications and seismic criteria for this change meet or exceed the original plant specifications and seismic requirements. The change has no impact on any evaluated scenario. Even the rupture of all of the newly installed pressure tap pipes would not lead to the occurrence of any of the accidents discussed in the Final Safety Analysis Report.

Plant Design Change Record Number 2-037-92

This change, entitled "Replacement of Heater Drains System Spargers," is complete.

Description of Change

This change replaced the spargers in the heater drains system. The spargers are located downstream of the drains coolers and are installed inside the main condensers.

Reason for Change

This change was designed to increase the back pressure in the line from the sixth point heater level control valves to the main condenser. This change, in conjunction with plant design change record number 2-115-91, was implemented to correct a two phase flow condition between the control valves and the main condenser.

Safety Evaluation

The spargers are non-safety related equipment and do not perform any safety function. Hence, they have no impact on any design basis accidents. There is no failure Mode for the spargers. The new spargers are made of P5 Chrome-Moly, rather than the original carbon steel, and so are much more resistant to erosion and corrosion.

Plant Design Change Record Number 2-068-92

This change, entitled "Cycle 12 Reload of Fuel," is complete.

Description of Change

The Cycle 12 reload consisted of loading 72 new Siemens fuel assemblies and one twice burned Westinghouse batch "G" assembly from the spent fuel pool, while removing 73 Westinghouse assemblies. Also, the last six control element assembly flow plugs were removed.

Reason for Change

This change was implemented to provide a safe, economical, extended cycle core load for Cycle 12.

Safety Evaluation

Safety analysis changes were made to take advantage of the increased reactor coolant system flow resulting from replacement of the steam generators. For cycle 12, the changed parameters were the integrated radial peaking factor, linear heat rate, reactor coolant system flow, bypass flow, and steam generator tube plugging. All changes were accommodated within the required acceptance criteria.

Accidents involving core flow and reactivity change were reanalyzed. Specific analyses performed by Siemens showed that all the accidents reanalyzed used the same acceptance criteria and methods as before, and results were acceptable. In each case, the increase in reactor coolant system flow (from the new steam generators) was sufficient to cover removal of the control element assembly flow plugs and the increased integrated radial peaking factor. Additional analysis demonstrated that the core design acceptance criteria for power distribution, shutdown margin, Moderator temperature coefficient, and fuel assembly burnup limits have been met.

Plant Design Change Record Number 2-078-92

This change, entitled "Reactor Protection System (RPS) Pressurizer Pressure Alarm Modification - Power Operated Releif Valves (PORV) Actuation Logic Reversal," is complete.

Description of Change

This change modified the high pressurizer pressure trip alarm function to make the alarm function independent from the pressurizer pressure trip inhibit scheme.

Reason for Change

This modification separated the alarm operation from the inhibit function such that the alarm is now operaticnal during testing as well as during normal operation. It also precluded the false generation of a PORV actuation signal caused by failure of DC Bus "A."

Safety Evaluation

This change adds relays to the affected circuits which are identical to relays already used in those circuits. Therefore, the failure Modes for the modified circuits are essentially the same as for the unmodified circuits. Since this change does add a component (relay), it results in a slight increase in the probability of a half PORV actuation signal due to relay failure. This risk increase is considered negligible. A failure of either the existing or new relays triggers an alarm, alerting the operators of potential failure. Since the new relays are normally deenergized, relay life should be greatly extended, reducing the risk of failure. In addition, both the PORV function and the alarm function affected by this change are non-safety related. Quality Assurance rated relays were used for this change because they provide interface isolation between safety grade (Reactor Protection System Control Power) and non-safety grade (PORV & Annunciator) sections.

Plant Design Change Record Number 2-079-92

This change, entitled "Condensate Storage Tank (CST) Nitrogen Blanket," is complete.

Description of Change

This change involved installing a nitrogen blanketing system for the vapor space of the existing CST, changing the tank from vented to atmosphere to a pressurized design. This required structural reinforcement of the tank, as well as installation of a pressure and vacuum relief scheme. Additional support installations included:

- New Cryogenic Nitrogen Storage and Supply System
- CST Nitrogen Sparger System and Oxygen Analyzer
- Relief Equipment Environmental Enclosure and Heat Tracing
- Indication and Alarm System

Reason for Change

This change was made to enhance secondary chemistry control in conjunction with the installation of the new steam generators.

Safety Evaluation

This change was evaluated extensively from several aspects, which are summarized below:
This change was examined for possible impact on:

- Loss of feedwater/auxiliary feedwater
- Steam generator tube rupture
- Feedwater line break

The only effect of this change is that the chemistry of the water entering the feed system will have a lower oxygen concentration.

The evaluation concluded that the change would not:

- Degrade the performance of safety systems or preventive actions assumed in any accident analysis.
- Alter any of the assumptions made in the analyses that increase the consequences of an accident.
- Degrade any fission product barriers.
- Affect the consequences of:
    * Tornado Missiles
    * Tornado Winds
    * Seismic Events
    * Tank Overfills
    * Failures of Plant Nitrogen Distribution Regulator
    * Failures of Nitrogen Pressure Control Valve
    * Losses of Nitrogen Storage and Supply System.

Plant Design Change Record Number 2-094-92

This change, entitled "Reactor Coolant Loop Resistance Temperature Detectors (RTD) Termination Modification," is complete.

Description of Change

This change replaced twenty reactor coolant system RTDs with the same Model RTD sensor which has a different method of termination. The new units use a quick disconnect connector instead of a terminal block inside the RTD head.

Reason for Change

This change provides an environmentally qualified connection for the RTDs being modified.

Safety Evaluation

Installation of the new RTDs only changed the method of terminating the field cable. The RTD sensor did not change. Electrically and functionally the RTD instrument loop is the same. This change has no impact on any previously evaluated scenario, and cannot result in any new or different scenario not previously evaluated.

Plant Design Change Record Number 2-103-92

This change, entitled "Heating, Ventilation, and Air Conditioning (HVAC) Control Cabinet Air Supply Relief Valves," is complete.

Description of Change

This change adds two relief valves, set at 22 psig, to the outlet of HVAC cabinets C25A and C25B solenoid pressure regulators.

Reason for Change

This change was implemented in response to Information Notice 88-24, which alerted addressees of potential problems with air-operated valves in safety related systems. It provided suggestions to consider regarding the protection of safety related solenoid valves with maximum operating pressure differentials (MODP) less than that of full instrument air pressure. Full instrument air pressure is 115 psig, and the MODP of the HVAC solenoid valves is 25 psig.

Safety Evaluation

This change ensured that the control room HVAC damper solenoids will not be pressurized above their MODP following failure of the upstream pressure regulating valves. The new relief valves are seismically mounted, and hence pose no threat to the control circuits or components in the vicinity. The change increases the reliability of the solenoid valves, and has no adverse impact on any previously analyzed accident or equipment important to safety.

Plant Design Change Record Number 2-112-92

This change, entitled "Static O-Ring Pump Suction Pressure Switch Replacement in the Chemical and Volume Control System (CVCS) and Reactor Building Closed Cooling Water (RBCCW) System," is complete.

Description of Change

This change replaced pressure and vacuum switches in the RBCCW system and CVCS. A total of four switches were involved, three in the CVCS and one in the RBCCW system.

Reason for Change

This change was performed in response to a 10 CFR Part 21 notification from Static O-Ring, which reported potential failure of pressure switches that utilize a Kapton primary diaphragm.

Safety Evaluation

The diaphragm material in the replacement switches was changed in response to the Part 21 notification. The weight of the new switches is the same as the original switches, and hence does not impose any changed affect on pipe loading or electrical systems. The replacements were considered a one-for-one replacement with components that were equal to, or better than, the original switches. The new diaphragm material is more resistant to corrosion, and is considered a product improvement. The change has no impact on any previously analyzed scenario.

Plant Design Change Record Number 2-114-92

This change, entitled "Required Modifications for Main Steam Line Break (MSLB) Scenario," is complete.

Description of Change

This change modified plant systems/setpoints as follows:

A) The emergency diesel start signal was modified to actuate on a safety injection actuation signal (SIAS).
B) The Containment hi-hi pressure setpoint was reduced from 27 psig to 9.48 psig.
C) The feedwater regulating valves were powered from VA-10/20 rather than from VR-11/21.
D) The Main Steam Isolation (MSI) signals from trains A & B were added to the feed pump discharge valves and feed block valves. Redundant MSI signals were provided to the feed regulating valves, feed bypass valves and feed pumps.

Reason for Change

The changes were required to quickly isolate feedwater and initiate containment protection systems to limit peak containment pressure below the 54 psig design limit following a postulated MSLB or loss of coolant accident (LOCA) in containment. Specific reasons for changing each item were:

A) To restore the system to operate as it did at plant startup, with loss of normal power and safety injection actuation signal starting the diesel.
B) To initiate spray actuation earlier.
C) To provide a more reliable source of power.
D) To ensure isolation of the feedwater system.

Safety Evaluation

This change ensured that an adequate margin of safety is maintained for the MSLB and LOCA accidents. No other analyzed accident scenarios are impacted. While this change adds to the quantity of equipment required to isolate feedwater, analysis has shown that there is no significant increase in the probability of an inadvertent feedwater isolation. Repowering the feed regulating valves increased their reliability, since the new power source is more reliable. This change is safe since all safety analysis criteria are met. The probability of exceeding containment design pressure was reduced, reactor trip frequency increased slightly and frequency of loss of all feedwater increased slightly. Core melt frequency increased less than 3E-8/reactor year; therefore, this increase was considered negligible.

Plant Design Change Record Number 2-181-92

This change, entitled "Chemical Feed to Tank T72 - Ammonium Hydroxide Solution Tank," is complete.

Description of Change

This change installs a transfer pump and associated hardware for measuring tank T-72 (Calgon K-35 feed tank). It also provides nitrogen from the station nitrogen header to blanket the portable chemical bins.

Reason for Change

This change permits filling the T-72 measuring tank, and provides the capability for blanketing the portable chemical bins during chemical feed into T-72 measuring tank and T-73 measuring tank (hydrazine solution tank).

Safety Evaluation

There are no accident scenarios applicable to the ammonium hydroxide solution tank, nor to the station nitrogen system in the vicinity of the secondary side chemical addition stations. The system performs no safety function nor does it support any equipment that performs a safety function. The change provides personnel protection during chemical transfer.

Plant Design Change Record Number 2-186-92

This change, entitled "Replace Service Water Pump P5C Discharge Head," is complete.

Description of Change

This change replaced the existing cast stainless steel discharge head and carbon steel motor support stand with an integral stainless steel assembly.

Reason for Change

The discharge head was replaced because of corrosion related degradation. Head replacement was selected, in lieu of a Code weld repair, as an opportunity to resolve a long standing vibration condition associated with this pump. This change also eliminated the intermediate motor support stand and stiffened the assembly, increasing the natural frequency of the assembly enough to alleviate the vibration condition.

Safety Evaluation

The service water system function is not affected by this change. The design of the new pump discharge head is consistent with the original pump design. The new discharge head has been designed for both normal and accident conditions, and has been seismically qualified. The worst case accident scenario associated with this change, failure of the new discharge head, is bounded by the existing accident analysis which assumes one pump fails. This change improves the reliability of the pump, providing additional assurance of operability of two service water headers, as required by the Technical Specifications.

Plant Design Change Record Number 2-189-92

This change, entitled "Diesel Generator Room Temperature Setpoint Change," is complete.

Description of Change

This change raised the diesel generator room low temperature alarm from 55 degrees Fahrenheit to 60 degrees Fahrenheit, and lowered the high temperature alarm from 120 degrees Fahrenheit to 110 degrees Fahrenheit.

Reason for Change

Previously, the alarms were set at the calculated design temperatures - 55 and 120 degrees Fahrenheit. This did not provide any "band" for operator action to correct the situation prior to exceeding the design limit. The new settings give operations personnel an opportunity to respond.

Safety Evaluation

The diesel generator room temperature alarm has no automatic actions. This alarm is a warning to the operators that a potential problem may exist which requires operator action. This change alerts the operator to low or high diesel generator room temperature earlier, and is therefore a conservative setpoint change.

Plant Design Change Record Number 2-208-92

This change, entitled "Coat Service Water Pump P5C Columns with Belzona," is complete.

Description of Change

This change applied Belzona S-Metal coating over the entire inside surface of the service water pump P5C column.

Reason for Change

This coating had previously been applied over the column weld joint areas to minimize corrosion in the heat affected zones. The coating had protected the welds, but the edges of the coating had been prone to crevice corrosion. Coating the entire inside surface eliminated the coated/uncoated interface, which was prone to corrosion.

Safety Evaluation

Experience with Belzona coating in service water application to date has been totally satisfactory. Any coating loss which could occur would be captured in the system strainer, and automatically backwashed. A total failure of the coating (sheet delamination) could be postulated to clog the strainer, resulting in a temporary loss of the affected train of service water. The resulting consequences would be no different than previously evaluated service water pump failures. The change has no impact on any previously analyzed scenario, and actually increases the reliability of the service water pump.

Plant Design Change Record Number 2-005-93

This change, entitled "2-HV-171 Motor Replacement," is complete.

Description of Change

The motor for ventilation damper 2-HV-171, previously a Honeywell Model M644A-E, was replaced with a Honeywell Model M640A-1022. The new damper motor required a different mounting configuration.

Reason for Change

Damper upgrade.

Safety Evaluation

Dampers 2-HV-170 and 2-HV-171 are in series, and provide redundant isolation of the main exhaust from the spent fuel pool area on auxiliary exhaust actuation signal initiated by two-out-of-four "high" radiation logic. This was essentially a one for one exchange, with a mounting change to account for the new configuration. The replacement motor was fully qualified and seismically tested to the same operating requirements as the previously installed unit. No modifications were made to the initiation logic, damper location or system configuration. Therefore, there is no impact on any evaluated scenario.

PROCEDURE CHANGES

Procedure Number	Title
SP-92-2-20	Incore Instruments (ICI) Flange Assembly Using HYTORC Hydraulic Torque Wrench
SP-93-2-3	Poison Rodlets in Spent Fuel Pool
SP-93-03	Recovered Boric Acid Storage Tank Chloride Cleanup
OP-2301A	Reactor Coolant System Quench Tank Operation
OP-2304A	Volume Control Portion of the Chemical and Volume Control System
EOP-2525	Standard Post Trip Actions
EOP-2532	Loss of Primary Coolant
EOP-2540C	Functional Recovery of Reactor Coolant System Inventory and Pressure
EOP-2540D	Functional Recovery of Heat Removal
AOP-2568	Reactor Coolant System Leak
AOP-2572	Loss of Shutdown Cooling
AOP-2582	Loss of Spent Fuel Pool Cooling
AOP-2583	Loss of All AC Power during Shutdown
SP-2411A	Control Element Assembly Inhibit Verification
IC-2418Z	Loose Parts Detection Calibration
IC-2434B	Reactor Coolant Pump Motor Instrument Removal and Replacement Guide
EOP-2526	Reactor Trip Recovery
MP-2702A5	Auxiliary Feedwater Control Valve Overhaul
MP-2702A6	Condensate Recirculation Control Valve Overhaul
MP-2702A7	Condensate Level Control Valve Overhaul
MP-2702E4	Morris Butterfly Valve Overhaul

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
MP-2723D	Operation of Pall Fluid Purifier
MP-2723E	Electro-Hydraulic Control (EHC) System Skid Maintenance and Periodic Test
EN-21004B	Reactor Startup Following Refueling (Initial Criticality)
EN-21004H	Excore Detector Shape Annealing Factor Measurements
SP-21027	Special Test Exceptions
EN-21036	Spent Incore Instruments (ICI)
SP-21128	Operational Readiness Testing of the Chilled Water System Valves
EN-21129	Containment Isolation Valves Operational Readiness Test
EN-21130	Reactor Building Closed Cooling Water (RBCCW) System Valves Operational Readiness Test
EN-21131	Chemical and Volume Control System Valves Operational Readiness Test
EN-21132	Service Water System Valves Operational Readiness Test
EN-21133	Reactor Coolant System Valves Operational Readiness Test
EN-21134	Main Steam Valves Operational Readiness Test
EN-21135	Main and Auxiliary Feedwater System Valves Operational Readiness Test
EN-21136	Safety Injection and Containment Spray System Valves Operational Readiness Test
SP-21160	Auxiliary and Main Feedwater System Leakage Test
SP-21198	Containment Spray System Leakage Test
SP-21199	Low Pressure Safety Injection/Shutdown Cooling Heat Exchanger Leakage Test

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
SP-21208	Containment Integrated Leak Rate Test (Type "A")

Procedure Number	Title
SP-92-2-20	Incore Instrument (ICI) Flange Assembly Using HYTORC Hydraulic Torque Wrench

Description of Change

This procedure provides detailed instructions for torquing ICI flange fasteners using a HYTORC hydraulic torque wrench.

Reason for Change

This procedure provides an alternate to the previously used manual method for torquing an ICI flange, with the goal of lowering personnel exposure.

Safety Evaluation

This procedure is performed during refueling shutdown, and only changes the method for torquing the ICI flanges. There is no credible mechanism by which proper performance of this procedure could impact any previously analyzed scenario or create a new or unanalyzed accident or malfunction.

Procedure Number	Title
SP-93-2-3	Poison Rodlets in Spent Fuel Pool

Description of Change

This special procedure installed a total of nine poison rodlets in three fuel assemblies in the spent fuel pool (three rodlets per assembly).

Reason for Change

This procedure is a part of the program to increase the storage capacity of the spent fuel pool.

Safety Evaluation

This procedure ensures that all mechanical design aspects of the fuel racks to store fuel and maintain the fuel assemblies coolable and in a subcritical configuration, with Keff less than 0.95, remain valid, unaffected and unchanged. This procedure does not move any fuel assemblies nor remove any cell blockers.

Procedure Number	Title
SP-93-03	Recovered Boric Acid Storage Tank Chloride Cleanup

Description of Change

This procedure provides guidance for removal of chlorides from the inventory of the Recovered Boric Acid Storage Tanks.

Reason for Change

Performance of this procedure allows re-use of the boric acid reclaimed by cycling the borated water through a mixed bed demineralizer.

Safety Evaluation

This procedure does not impact any safety related components, based upon evaluation of the two possible malfunctions of resin breakdown due to high temperature borated water solution and hose failure.

Procedure Number	Title
OP-2301A	Reactor Coolant System Quench Tank Operation
OP-2304A	Volume Control Portion of the Chemical and Volume Control System
EOP-2525	Standard Post Trip Actions
EOP-2532	Loss of Primary Coolant
EOP-2540C	Functional Recovery of Reactor Coolant System Inventory and Pressure
EOP-2540D	Functional Recovery of Heat Removal
AOP-2568	Reactor Coolant System Leak

Description of Change

The procedures listed above were changed to impose an upper limit on the reactor coolant system pressure at which the power operated relief valve block valves can be closed.

Reason for Change

This change compensates for degradation of the power operated relief block valves relative to their design basis.

Safety Evaluation

This change effectively prevents the isolation of a leaking power operated relief valve without a controlled plant shutdown and depressurization. Since the restrictions on isolating a leaking power operated relief valve result in a situation similar to any other unisolable reactor coolant system leak in containment, there is no change to any previously evaluated scenario, nor does the change create any new accident or malfunction situation.

Procedure Number	Title
AOP-2572	Loss of Shutdown Cooling

Description of Change

This procedure was revised to incorporate the following guidance:
- elimination of automatic closure of safety injection valves
- addition of the reactor building closed cooling water system as a potential leak path
- precautions concerning gravity feed of the reactor water storage tank
- action items for calculations on reactor coolant system time to boil and time to uncover the core

Reason for Change

This procedure provided enhanced guidance to assist the plant operators in determining the time to boil and time to uncover the core following a loss of shutdown cooling, and specified additional containment closure requirements.

Safety Evaluation

This procedure provides guidance for actions to be taken after an accident (loss of shutdown cooling) has occurred. It has no affect on the probability of occurrence of that (or any other) accident or malfunction. The purpose of the procedure is to mitigate the effects of, and recover from, a specific accident situation. It does not create any new or unanalyzed scenario.

Procedure Number	Title
AOP-2582	Loss of Spent Fuel Pool Cooling

Description of Change

This new procedure provides guidance to plant operators in the event of a loss of spent fuel pool cooling.

Reason for Change

This procedure consolidates into a single document guidance and information which was previously found in multiple sources.

Safety Evaluation

This procedure provides guidance for actions to be taken after an accident (loss of spent fuel pool cooling) has occurred. It has no affect on the probability of occurrence of that (or any other) accident or malfunction. The purpose of the procedure is to mitigate the effects of, and recover from, a specific accident situation. It does not create any new or unanalyzed scenario.

Procedure Number	Title
AOP-2583	Loss of All AC Power during Shutdown

Description of Change

This new procedure provides guidance to plant operators in the event of a loss of all AC power during shutdown conditions.

Reason for Change

This procedure consolidates into a single document guidance and information which was previously found in multiple sources.

Safety Evaluation

This procedure provides guidance for actions to be taken after an accident (loss of all AC power during shutdown) has occurred. It has no affect on the probability of occurrence of that (or any other) accident or malfunction. The purpose of the procedure is to mitigate the effects of, and recover from, a specific accident situation. It does not create any new or unanalyzed scenario.

Procedure Number	Title

Procedure Number	Title
SP-2411A	Control Element Assembly Inhibit Verification

Description of Change

This procedure provides a means to satisfy Technical Specification requirements, by requiring that control element assembly motion inhibit be demonstrated operable at least once per 31 days by a functional test of the control element assembly group deviation circuit.

Reason for Change

This procedure reflects deletion of the metrascope position display system and installation of the computer based control element assembly position display system.

Safety Evaluation

This procedure simulates control element assembly movement, using the test Mode of the control element assembly position display system, in order to initiate control motion inhibit. The control motion inhibit will be confirmed by challenging the inhibit signal, and then moving the control element assembly in bypass. The control element assemblies will not exceed Technical Specification limits during the performance of the procedure and only the equipment's normal operating Modes will be tested by this procedure

Procedure Number | Title

IC-2418Z | Loose Parts Detection Calibration

Description of Change

This procedure details the methods for calibrating the loose parts monitoring system.

Reason for Change

This procedure was developed to provide the necessary guidance to calibrate the non-safety related loose parts monitoring system.

Safety Evaluation

This procedure requires performance in accordance with accepted industry standards. As noted above, the equipment being calibrated is not safety related, and there is ro credible mechanism by which performance of this procedure can impact any safety related component, system or structure.

Procedure Number Title

IC-2434B Reactor Coolant Pump Motor Instrument Removal and Replacement Guide

Description of Change

This procedure is a guide for removal and reinstallation of reactor coolant pump motor instrumentation.

Reason for Change

This procedure provides a means of formally tracking the removal and reassembly of instrumentation associated with the reactor coolant pump motors.

Safety Evaluation

This procedure provides a means of controlling and tracking instrumentation as it is removed and reinstalled on the reactor coolant pump motors. All work is completed in accordance with accepted industry standards, with this procedure providing additional control over the work being done. There is no credible mechanism by which performance of this procedure could impact any previously analyzed scenario.

Procedure Number	Title
EOP-2526	Reactor Trip Recovery

Description of Change

This change deleted reference to the use of primary makeup water as a makeup source to the condensate storage tank.

Reason for Change

This change was required as a follow-up to a design change which cut and capped the line between the primary makeup water system and the condensate storage tank/auxiliary feedwater suction piping.

Safety Evaluation

This change was necessary to reflect a piping system change where the use of primary makeup water to restore condensate storage tank level had been eliminated. Several other methods for restoring condensate storage tank level remain available. The change had no impact on any previously evaluated scenario, or on any safety related equipment or system.

Procedure Number	Title
MP-2702A5	Auxiliary Feedwater Control Valve Overhaul

Description of Change

This procedure provides instructions for preventive and corrective maintenance of a Copes Vulcan 4 inch, 600 psi, Model D100 control valve and its associated actuator.

Reason for Change

This procedure provides explicit guidance specific to this type and size of valve.

Safety Evaluation

This procedure does not address any modification to the valve or the system in which the valve is installed. Therefore, its use does not change any failure modes and effects analysis. Hence, operability of the auxiliary feedwater system will not be affected by performance of this procedure.

Procedure Number	Title
MP-2702A6	Condensate Recirculation Control Valve Overhaul

Description of Change

This procedure provides instructions for preventive and corrective maintenance of a Copes Vulcan 6 inch, 300 psi, Model D100 control valve and its associated actuator.

Reason for Change

This procedure provides explicit guidance specific to this type and size of valve.

Safety Evaluation

This procedure does not address any modification to the valve or the system in which the valve is installed. Therefore, its use does not change any failure modes and effects analysis. Hence, operability of the auxiliary feedwater system will not be affected by performance of this procedure.

Procedure Number	Title
MP-2702A7	Condensate Level Control Valve Overhaul

Description of Change

This procedure provides instructions for preventive and corrective maintenance of a Copes Vulcan 4 inch, 300 psi, Model D100 control valve and its associated actuator.

Reason for Change

This procedure provides explicit guidance specific to this type and size of valve.

Safety Evaluation

This procedure does not address any modification to the valve or the system in which the valve is installed. Therefore, its use does not change any failure modes and effects analysis. Hence, operability of the condensate system will not be affected by performance of this procedure.

Procedure Number	Title
MP-2702E4	Norris Butterfly Valve Overhaul

Description of Change

This procedure provides instruction for overhauling Norris butterfly valves.

Reason for Change

This procedure provides details for all aspects of valve and actuator removal, reinstallation and testing of Norris butterfly valves.

Safety Evaluation

These valves, used in the service water system, are fail-open air-actuated valves. This procedure provides formal control and documentation of work performed. It does not modify any service water system function or design basis. There is no credible mechanism by which performance of this procedure could affect any previously evaluated scenario.

Procedure Number	Title

Procedure Number	Title
MP-2723D	Operation of Pall Fluid Purifier

Description of Change

This procedure provides instruction for purifier startup, operation, shutdown, and maintenance.

Reason for Change

The procedure provides instructions for use of the purifier, which is a piece of temporary maintenance equipment used to control moisture and impurities in the Electro-Hydraulic Control (EHC) system fluid.

Safety Evaluation

The bounding case of EHC failure specified in the Final Safety Analysis Report states that "the (EHC) system is designed to respond to a loss of fluid pressure for any reason, and leads to turbine inlet valve closing and consequent turbine generator shutdown." Operation and potential failure modes of the temporary EHC purifier are therefore bounded.

Procedure Number	Title
MP-2723E	Electro-Hydraulic Control (EHC) System Skid Maintenance and Periodic Test

Description of Change

This procedure provides instructions for proper maintenance and periodic testing of the EHC skid components.

Reason for Change

This procedure provides details on various skid operating parameters.

Safety Evaluation

There is no credible mechanism by which performance of this procedure can impact any previously analyzed failure modes of the EHC system, or introduce any unanalyzed scenarios. Proper maintenance and testing should actually increase the operational reliability of the EHC system.

Procedure Number | Title

Procedure Number	Title
EN-21004B	Reactor Startup Following Refueling (Initial Criticality)

Description of Change

This procedure detailed the steps necessary to perform reactor startup following refueling (initial criticality of the new core).

Reason for Change

This procedure consolidates requirements from a previous startup test procedure plus several other sources into a single document. The procedure also specifies termination criteria and contingency actions for operations and reactor engineering personnel to address any unusual situations during procedure performance.

Safety Evaluation

All reactivity manipulations and equipment operations were performed in accordance with normal plant operating procedures. This procedure provided a guide to ensure a controlled, monitored approach to criticality. The procedure had no impact on any analyzed scenario, nor did it create any new unevaluated accident or malfunction.

Procedure Number	Title
EN-21006H	Excore Detector Shape Annealing Factor Measurements

Description of Change

This procedure details the proper methods of obtaining incore and excore data necessary for determining shape annealing factors.

Reason for Change

This procedure provides for measurement of xenon oscillation at 65% reactor power. This measurement was previously performed in Cycles 1 and 6.

Safety Evaluation

Siemens Nuclear Power has evaluated the performance of this procedure with respect to the fuel preconditioning guidelines, and has concluded that no violation of the fuel precondition guidelines occurs. Performance of this procedure will not impact any of the plant's protective barriers (fuel cladding/matrix, reactor coolant system pressure boundary or reactor containment).

Procedure Number	Title
SP-21027	Special Test Exceptions

Description of Change

This procedure provides details on performance of surveillances associated with the Technical Specifications "Special Test Exceptions," which are used during rise to power and physics testing following refueling.

Reason for Change

This procedure was developed to provide surveillance requirements which were previously contained in the body of the Low Power Physics In-Service Test procedures.

Safety Evaluation

This procedure records plant data to ensure compliance with Technical Specification Special Test Exceptions surveillances. Use of the Special Test Exceptions is similar to use of an action statement, in that it is intended to be used for a limited set of controlled conditions (e.g., physics testing). The surveillance requirements imposed provide assurance that any unusual conditions will be quickly identified and corrected. Performance of this procedure cannot create any malfunctions or alter the assumptions of any accident analyses.

Procedure Number	Title
EN-21036	Spent Incore Instruments (ICI)

Description of Change

This procedure describes the steps necessary to cut up the spent incore instruments for storage and disposal.

Reason for Change

This procedure details the use of a hydraulic cutter to cut the incore instruments into small pieces and drop them into a spent fuel pool debris container or a shipping liner. It also specifies the particular hydraulic fluid to be used in the Unit No. 2 spent fuel pool.

Safety Evaluation

While there is a potential for personnel overexposure or generation of loose parts in the spent fuel pool while performing this procedure, these risks are minimized by the administrative (i.e., procedural) and Health Physics controls in place during the performance period. The hydraulic fluid specified will not create any spent fuel pool chemistry problems. There is no impact on the margin of safety since this operation will not adversely affect the fuel/cladding matrix, the reactor coolant pressure boundary or the containment.

Procedure Number	Title
SP-21128	Operational Readiness Testing of the Chilled Water System Valves
EN-21129	Containment Isolation Valves Operational Readiness Test
EN-21130	Reactor Building Closed Cooling Water (RBCCW) System Valves Operational Readiness Test
EN-21131	Chemical and Volume Control System Valves Operational Readiness Test
EN-21132	Service Water System Valves Operational Readiness Test
EN-21133	Reactor Coolant System Valves Operational Readiness Test
EN-21134	Main Steam Valves Operational Readiness Test
EN-21135	Main and Auxiliary Feedwater System Valves Operational Readiness Test
EN-21136	Safety Injection and Containment Spray System Valves Operational Readiness Test

Description of Change

This change alters the method of depressurizing the air operating cylinders of the valves under test, by moving the vent point from the upstream side of the air regulator serving the valve to the downstream side of the air regulator.

Reason for Change

It was found that some of the air regulators serving these valves would not allow air back-flow under certain conditions, thus preventing the valve from moving to its specified fail-safe position.

Safety Evaluation

The revised test method assures that the affected valves will travel to the fail-safe position when air pressure is lost to the actuator. This is consistent with the accident position of the valve. Air trapped in the actuator/solenoid valve assembly by the pressure regulator will hold the valve in the pressurized position until leakage or actuation of the solenoid allows the valve to travel to the fail-safe position. This is consistent with normal system operation, and does not constitute a degradation of the safety function of the affected valves. The change assures that the valves will be tested under the same conditions in which they are required to function in an accident situation.

Procedure Number	Title
SP-21160	Auxiliary and Main Feedwater System Leakage Test

Description of Change

This one-time change reduced the scope of the normal in-service leak test to two specific sections required to be inspected for the 10 year leak test, and provided controls and limits for the pressure applied to these sections.

Reason for Change

This one-time change allowed for leak testing to verify the integrity of piping between the auxiliary feed regulating valves and their immediate downstream isolation valves.

Safety Evaluation

This test provided a reasonable and safe method of verifying auxiliary feedwater piping integrity in a specific area. The test had no impact on any equipment important to safety, nor did it create new scenarios or change previously evaluated accidents.

Procedure Number　　　　Title

SP-21198　　　　Containment Spray System Leakage Test

Description of Change

This procedure provides instructions for performing ASME Section XI functional leakage tests of the containment spray system piping and components.

Reason for Change

This procedure replaces prior guidance. The new guidance provides more detailed and enhanced guidance for performance of the subject test.

Safety Evaluation

This procedure directs operation of plant systems and equipment in accordance with approved operating procedures. It does not modify or revise any previously approved plant or system alignment. Hence, it has no impact on any analyzed scenarios, nor does it introduce any new, unanalyzed conditions.

Procedure Number	Title
SP-21199	Low Pressure Safety Injection/Shutdown Cooling Heat Exchanger Leakage Test

Description of Change

This procedure provides instructions for performing ASME Section XI functional leakage tests of the Low Pressure Safety Injection system and Shutdown Cooling heat exchanger piping and components.

Reason for Change

This procedure replaces prior guidance. The new guidance provides more detailed and enhanced guidance for performance of the subject test.

Safety Evaluation

This procedure directs operation of plant systems and equipment in accordance with approved operating procedures. It does not modify or revise any previously approved plant or system alignment. Hence, it has no impact on any analyzed scenarios, nor does it introduce any new, unanalyzed conditions.

Procedure Number	Title
SP-21208	Containment Integrated Leak Rate Test (Type "A")

Description of Change

This procedure provides test methods, procedures and acceptance criteria for performing the Type "A" integrated leak rate test on the reactor containment structure.

Reason for Change

This procedure consolidated information previously available from multiple sources, and was issued as a Special Procedure to reflect a change of responsibility for performance direction from Operations to Engineering.

Safety Evaluation

This procedure combined guidance formerly available in several documents. It did not change or modify in any way the basic methods or procedures which have been used successfully during previous performances of this test. Test performance has no impact on any previously analyzed scenario, nor does it create any new, unevaluated accident or malfunction.

JUMPERS-LIFTED LEADS-BYPASSES

Jumper-Lifted Lead-Bypass	Title
2-93-3	Lift Leads from Relays 52X-1 and 52X-2
2-93-12	"D" Reactor Coolant Pump Vapor Seal Leakoff Line: Install New Line with Relocated Check Valve
2-93-13	Installation of Wire Mesh Screens on the Containment Air Recirculation Coolers
2-93-16	Eliminate Hot Junction Thermocouple Detector #4 and Unheated Thermocouple #6 for Channel "A" of the Reactor Vessel Level Indication System
2-93-25	Install Engineered Clamp on Service Water Supply Line 2"-HUD-130, West 480 Volt Switchgear Room Cooler Service Water Supply Line
2-93-27	Temporary Seismic Supports for Removal of Valve 2-SW-9B
2-93-41	Temporary Seismic Support for Removal of Valve 2-SW-9B
2-93-45	Lift the Upper Electrical Limit Lead for Control Element Drive Mechanism (CEDM) 43
2-93-50	Installation of Temporary Variator on 2-CH-089
2-93-51	Inverter #3 Synchronization Failure Alarm
2-93-52	Installation of Temporary Variator on 2-CH-089
2-93-53	Installation of Temporary Variator on 2-CH-515
2-93-56	Temporary Seismic Support for Removal of Valve 2-SW-8.1C and Associated Piping

JUMPERS-LIFTED LEADS-BYPASSES (CONTINUED)

Jumper-Lifted Lead-Bypass	Title
2-93-57	Provide Test Connections for "B" Emergency Diesel Generator
2-93-61	Freeze Seal on Letdown Line 2"-CCA-16 Piping
2-93-67	Temporary Seismic Supports for Removal of Valve 2-SW-8.1A and Upstream and Downstream Pipe Spools
2-93-69	Security Grating Installation
2-93-70	Concrete Block Removal to Facilitate Piping Modifications
2-93-72	Spool Piece Installed for 2-CH-223
2-93-85	Temporary Instrumentation for the Electro-Hydraulic Control System
2-93-86	Reactor Protection System Channel "A" Low Flow

Jumper-Lifted Lead-Bypass Number 2-93-3

This bypass jumper, entitled "Lift Leads from Relays 52X-1 and 52X-2," is installed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper consists of a Caution Tag on the "B" High Pressure Safety Injection Pump start switch, and two lifted leads-one each from relays 52X-1 and 52X-2. The lifted leads disconnect the "open" signal that the "C" Engineered Safety Features room supply and exhaust dampers receive, when the "B" high pressure safety injection pump is started. These dampers can then be operated and controlled from panel C-01 in the control room. This bypass jumper is still active.

Reason for Jumper-Lifted Lead-Bypass

This change addresses a potential design deficiency in the Engineered Safety Features (ESF) Ventilation System. In its current configuration, when the "B" High Pressure Safety Injection (HPSI) pump starts, the air flow to the "A" and "B" ESF rooms is automatically throttled to 70% of the pre-Safety Injection Actuation Signal (SIAS) flow. The remaining 30% of the air flow is diverted to the "C" ESF room. With respect to the "A" and "B" ESF rooms, depending on the specific lineup in place at the time of the event, the 70% flow is approximately 67% of the flow required by one room, and approximately 200% of the flow required by the other room. This change allows the operator to select the appropriate source for air flow to the "C" ESF room during the event.

Safety Evaluation

This installation will not in any way increase the probability of occurrence or the effects of any previously evaluated scenario, nor will it introduce any new, unaddressed situation. It will not impede the reaction of any other system to either a SIAS or a CSAS. Operator action is already required, both in the control room and in the plant, when aligning the "B" HPSI pump to either facility. A plant design change record is being prepared to correct this situation. The design change is scheduled for installation during the 1994 Refueling Outage.

Jumper-Lifted Lead-Bypass Number 2-93-12

This bypass jumper, entitled "'D' Reactor Coolant Pump Vapor Seal Leakoff Line: Install New Line with Relocated Check Valve," is installed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed a new, temporary drain line between the "D" reactor coolant pump seal and the primary drain tank. This new line has a check valve located near the tie-in to valve 2-LRR-363 to prevent the primary drain tank gas pressure from acting on the pump seal. This check valve has a very low (1 psig) cracking pressure, such that the available head between the pump seal and the valve (approximately 20 feet) is more than adequate to open the valve. This bypass jumper is still active.

Reason for Jumper-Lifted Lead-Bypass

The purpose of this bypass jumper is to bypass the existing drain line. Leakage from the "D" reactor coolant pump vapor seal, which was replaced during the steam generator replacement outage, is not at a high enough pressure to open valve 2-RC-27D, the piston check valve located downstream of the pump seal.

Safety Evaluation

This bypass jumper will not affect the design function of the leakoff line. The reactor coolant pump and the primary drain tank will continue to perform their normal functions. The new line is consistent with the requirements for the existing leakoff line (pressure, stress, material requirements, etc.). The temporary line will continue to provide the same function, in the same degree, as the original arrangement.

Jumper-Lifted Lead-Bypass Number 2-93-13

This bypass jumper, entitled "Installation of Wire Mesh Screens on the Containment Air Recirculation Coolers," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed 1/2-inch wire mesh screens on the exterior frames of each of the 4 Containment Air Recirculation Coolers. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The purpose of this bypass jumper was to protect the cooling coil fins of the coolers from any damage during startup and initial operation, following the steam generator replacement outage.

Safety Evaluation

A Containment Air Recirculation Fan Flow Test verified that the wire mesh screens did not affect the overall air flow through the coolers. The method used to attach the screens to the coolers, together with the overall physical dimensions and mesh size of the screens, made it highly unlikely that an individual screen could be separated from its cooler and transported to the containment sump, in a manner which could adversely affect post loss of coolant accident operation of the containment recirculation mode of operation during containment sump recirculation. Therefore, installation of these screens did not affect the heat removal capability of the coolers, nor did it have any impact on the operating characteristics or safety related requirements of the coolers.

Jumper-Lifted Lead-Bypass Number 2-93-16

This bypass jumper, entitled "Eliminate Hot Junction Thermocouple Detector #4 and Unheated Thermocouple #6 for Channel 'A' of the Reactor Vessel Level Indication System," is installed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper lifted leads to remove the thermocouple input from detectors #4 and #6. This bypass jumper is still active.

Reason for Jumper-Lifted Lead-Bypass

The purpose of this bypass jumper was to remove the listed detectors from the circuit, as both were giving inaccurate readings and spurious alarms.

Safety Evaluation

Channel "A" of the Reactor Vessel Level Indication System consists of a total of eight sensors, four upper sensors and four lower sensors. The Channel is defined as operable if two or more sensors are operable in each sector (upper and lower). Installation of this Bypass-Jumper left three sensors operable in each sector. These six sensors are sufficient to adequately monitor water level in the reactor vessel.

Jumper-Lifted Lead-Bypass Number 2-93-25

This bypass jumper, entitled "Install Engineered Clamp on Service Water Supply Line 2"-HUD-130, West 480 Volt Switchgear Room Cooler Service Water Supply Line," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed an "engineered clamp" on the supply piping to the west switchgear room cooler X-181A. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The clamp was required to meet the structural integrity requirements of the ASME code. A weld joint on the service water inlet line leaked during a hydrostatic test performed during a unit startup. This resulted in system structural integrity requirements not being met. This clamp was installed as an acceptable alternate repair, as part of an United States Nuclear Regulatory Commission Notice of Enforcement Discretion. During a subsequent unit shutdown, the clamp was removed, the system repaired, and successfully hydrostatically tested pursuant to ASME code requirements. (It is appropriate to note that the weld joint did not leak at normal system operating pressure.)

Safety Evaluation

Prior to installation, Northeast Utilities Service Company stress analysis personnel determined that the existing pipe weld was structurally adequate for all design loads, and concluded that the engineered clamp was structurally adequate to react to piping loads that would occur if the defective weld were to fail completely. The engineered clamp satisfied all Technical Specification structural integrity requirements. Thus, this bypass jumper had no affect on normal operation of the service water system or the switchgear cooling system. (It was noted that the normal room cooling requirements exceed the accident condition requirements, and system realignment during an accident is not required).

Jumper-Lifted Lead-Bypass Number 2-93-27

This bypass jumper, entitled "Temporary Seismic Supports for Removal of Valve 2-SW-9B," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed two temporary supports on the service water discharge piping from the "B" Reactor Building Closed Cooling Water system heat exchanger. In one case, an existing spring support was locked in place prior to draining the system. In the other case, a temporary support adequate for a 1,000 lb. load was installed on the service water return header. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was installed to permit removal of 2-SW-9B, the manual service water discharge valve from the RBCCW HX, for repairs.

Safety Evaluation

The temporary configuration outlined above was analyzed by Northeast Utilities Services Company Pipe Stress Engineering, and found to be adequate to maintain the seismic integrity of the Service Water system while valve 2-SW-9B was removed. Operation of the Service Water system for the "A" and "C" Reactor Building Closed Cooling Water system heat exchangers was not affected by the work on the "B" heat exchanger piping. Therefore, two trains of Service Water and two trains of Reactor Building Closed Cooling Water remained operable for all Modes of operation, as required by Technical Specifications.

Jumper-Lifted Lead-Bypass Number 2-93-41

This bypass jumper, entitled "Temporary Seismic Support for Removal of Valve 2-SW-9B," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed a temporary support on the service water discharge piping from the "B" Reactor Building Closed Cooling Water (RBCCW) system heat exchanger. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was installed to permit unbolting of 2-SW-9B, the manual service water discharge valve from the RBCCW heat exchanger. When the valve was unbolted, upstream piping between that valve and the RBCCW system "B" heat exchanger was structurally disconnected from the service water header, thus permitting repair of flange leaks in the piping between the valve and the heat exchanger.

Safety Evaluation

The temporary configuration outlined above was analyzed by Northeast Utilities Services Company Pipe Stress Engineering, and found to be adequate to maintain the seismic integrity of the Service Water System while valve 2-SW-9B was unbolted. Operation of the Service Water System for the "A" and "C" RBCCW system heat exchangers was not affected by the work on the "B" heat exchanger piping. Therefore, two trains of Service Water and two trains of Reactor Building Closed Cooling Water remained operable for all Modes of operation, as required by Technical Specifications.

Jumper-Lifted Lead-Bypass Number 2-93-45

This bypass jumper, entitled "Lift the Upper Electrical Limit Lead for Control Element Drive Mechanism (CEDM) 43," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper documented the lifting and taping of leads for the Upper Electrical Limit reed switch on CEDM number 43. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The purpose of this bypass jumper was to allow the Control Element Assembly (CEA) to be withdrawn for reactor startup. This reed switch was believed to be stuck in the closed position due to residual magnetization.

Safety Evaluation

The Upper Electrical Limit reed switch provides an alternate means of indication that the CEA is at the "full out" position, as well as providing an interlock which stops outward motion of the CEA. Lifting this lead does not affect other position indicators (i.e. - CEA position display system reed switches and pulse counters), and thus does not increase the probability of occurrence of any previously evaluated accidents. In addition, it does not affect any of the plant protective barriers (fuel cladding/matrix, reactor coolant system pressure boundary or reactor containment), and therefore has no impact on the margin of safety as defined in the basis of any Technical Specification.

Jumper-Lifted Lead-Bypass Number 2-93-50

This bypass jumper, entitled "Installation of Temporary Variator on 2-CH-089," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed a temporary variator and calibrated test gauge in the air supply line upstream of the solenoid valve which provides the Containment Isolation Actuation Signal (CIAS) for valve 2-CH-089. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was used to operate the valve during troubleshooting to determine the cause of valve seat leakage.

Safety Evaluation

The primary concern was the affect of the bypass jumper installation on valve operation. In any accident scenario which would result in a CIAS, valve 2-CH-089 would be required to close. Installation of this bypass jumper did not have any impact on the control circuit for the solenoid valve in the operating line. Therefore, a CIAS signal would operate the solenoid valve, permitting the air in the line to the 2-CH-089 operator to vent off normally through the solenoid valve, to close 2-CH-089. The isolation capability of 2-CH-089 was maintained at all times.

Jumper-Lifted Lead-Bypass Number 2-93-51

This bypass jumper, entitled "Inverter #3 Synchronization Failure Alarm," is installed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper removes from service the inverter #3 synchronization failure alarm contact that triggers the inverter #3 trouble annunciator at control board C08. This synchronization failure alarm is used to alert the operator to the fact that this inverter's alternate power source is out of synchronization. This bypass jumper is still active.

Reason for Jumper-Lifted Lead-Bypass

Random noise occurs frequently in the 60 Hertz power supply from the 120 VAC bus. This noise is sensed by the inverter's synchronization check circuit as an out of synchronization condition, resulting in frequent spurious #3 inverter trouble alarm annunciations. This bypass jumper was installed to eliminate this nuisance condition.

Safety Evaluation

This bypass jumper removed the synchronization alarm from the control room common alarm for Inverter #3. This is a non-safety related alarm, which has no effect on the operation of the Inverter or Static Switch. The local alarm at the inverter and the interlock associated with the static switch for transferring to an alternate source will not be affected. This bypass jumper only affects the common alarm for one Inverter (#3), which supplies power to VA30 which, in turn, powers the engineered safeguards actuation signal and the reactor protection system. If VA30 was lost, a one-out-of-four logic trip would occur, which is within the design basis of the plant. A two-out-of-four logic trip is needed for an actuation to occur. Since the worst case event is the loss of VA30, there are no accidents which can occur as a result of this bypass jumper.

Jumper-Lifted Lead-Bypass Number 2-93-52

This bypass jumper, entitled "Installation of Temporary Variator on 2-CH-089," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed a temporary variator and calibrated test gauge in the air supply line upstream of the solenoid valve which provides the Containment Isolation Actuation Signal (CIAS) for valve 2-CH-089. In addition, a temporary dial indicator was installed on the valve stem, using a magnet as the mounting device, to detect stem movement. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was used to operate the valve during troubleshooting to adjust the air operator spring force.

Safety Evaluation

The primary concern was the affect of the bypass jumper installation on valve operation. In any accident scenario which would result in a CIAS, valve 2-CH-089 would be required to close. Installation of this bypass jumper did not have any impact on the control circuit for the solenoid valve in the operating line. Therefore, a CIAS signal would operate the solenoid valve, permitting the air in the line to the 2-CH-089 operator to vent off normally through the solenoid valve, to close 2-CH-089. The isolation capability of 2-CH-089 was maintained at all times.

Jumper-Lifted Lead-Bypass Number 2-93-53

This bypass jumper, entitled "Installation of Temporary Variator on 2-CH-515," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed a temporary variator and calibrated test gauge in the air supply line upstream of the solenoid valve which provides the Safety Injection Actuation Signal (SIAS) for valve 2-CH-515. In addition, a temporary dial indicator was installed on the valve stem,, using a magnet as the mounting device, to detect stem movement. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was used to operate the valve during maintenance activities to adjust the actuator spring.

Safety Evaluation

The primary concern was the affect of the bypass jumper installation on valve operation. In any accident scenario which would result in a SIAS, valve 2-CH-515 would be required to close. Installation of this bypass jumper did not have any impact on the control circuit for the solenoid valve in the operating line. Therefore, an SIAS would operate the solenoid valve, permitting the air in the line to the 2-CH-515 operator to vent off normally through the solenoid valve to 2-CH-515, to close 2-CH-515. The isolation capability of 2-CH-515 was maintained at all times.

Jumper-Lifted Lead-Bypass Number 2-93-56

This bypass jumper, entitled "Temporary Seismic Support for Removal of Valve 2-SW-8.1C and Associated Piping," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper documented the installation of temporary supports on the Service Water discharge piping from the "C" Reactor Building Closed Cooling Water heat exchanger. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper permitted removal of valve 2-SW-8.1C and associated piping for repairs to the internal coating of the piping.

Safety Evaluation

The temporary configuration outlined above was analyzed by Northeast Utilities Services Company Pipe Stress Engineering, and found to be adequate to maintain the seismic integrity of the Service Water system while valve 2-SW-8.1C was removed. Operation of the Service Water system for the "A" and "B" Reactor Building Closed Cooling Water system heat exchangers was not affected by the work on the "C" heat exchanger piping. Therefore, two trains of Service Water and two trains of Reactor Building Closed Cooling Water remained operable for all Modes of operation, as required by Technical Specifications.

Jumper-Lifted-Lead-Bypass Number 2-93-57

This bypass jumper, entitled "Provide Test Connections for 'B' Emergency Diesel Generator," is installed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper installed test jacks inside the "B" emergency diesel generator control cabinet (C39), to permit connection of ten electrical test points. It included mounting and wiring two 12-point terminal blocks and ten banana style test jacks. The assembly was mounted on the bottom of the cabinet, in accordance with approved seismic requirements. Quality Assurance Category I fire retardant shielded instrument cable with wire lugs was used to connect the control devices to the test blocks. This bypass jumper is still active.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was installed to permit monitoring of the performance of the diesel's governor controls during performance of regular surveillances.

Safety Evaluation

This bypass jumper is in use only during performance of weekly surveillance testing, during which the diesel is declared inoperable. During normal operation of the diesel, the test jacks will be open circuits protected from electronic noise by shielded wire. The wire, cable, terminal blocks, wire lugs and plug jacks are compatible with the existing circuits. The assembly is securely mounted at the bottom of the control cabinet, precluding the possibility of it becoming detached, falling, and damaging safety related equipment. The installation has no impact on any of the diesel controls.

Jumper-Lifted Lead-Bypass Number 2-93-61

This bypass jumper, entitled "Freeze Seal on Letdown Line 2"-CCA-16 Piping," has been removed.

Description of Jumper-Lifted Lead-Bypass

Charging Letdown Valve 2-CH-442 developed a leak, which had to be repaired. This valve is located on letdown line 2"-CCA-16, which originates at the loop 2B reactor coolant system cold leg. The valve is the first valve in this line, located below the centerline of the hot leg, and therefore must be isolated by installation of a freeze seal. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper isolated 2-CH-442 to permit maintenance (valve replacement) to be performed to correct the leaking condition.

Safety Evaluation

The most probable failure associated with the freeze seal is the gradual thawing of the ice plug. In the event of the loss of liquid nitrogen cooling to the freeze chamber, gradual thawing of the ice plug would result. Insertion of the plug provided by the contingency plan would prevent flow through the letdown line. This type of failure would have no affect on the reactor coolant system level. Sudden failure of the ice plug is also extremely unlikely. The water in the pipe expands and exerts an outward force against the pipe wall as the ice plug forms. The freeze seal was located such that there was a 90 degree elbow between the ice plug and the valve. Complete failure of the 2" pipe is extremely unlikely. The freeze seal temperature (as low as -320 degrees Fahrenheit) will not degrade the stainless steel pipe material (2 inch schedule 160 pipe, A376 gr 316 S.S.). Dimensional and non-destructive testing was performed on the piping before and after the freeze seal was installed to verify piping system integrity was not compromised.

Jumper-Lifted Lead-Bypass Number 2-93-67

This bypass jumper, entitled "Temporary Seismic Supports for Removal of Valve 2-SW-8.1A and Upstream and Downstream Pipe Spools," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper documents the installation of one temporary support and temporary modification to two existing supports on the service water discharge piping from the "A" reactor building closed cooling water heat exchanger, and temporary modification to one existing support on the service water discharge piping from the "B" reactor building closed cooling water heat exchanger. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper permitted the removal of valve 2-SW-8.1A and its associated upstream and downstream spool pieces for repair.

Safety Evaluation

The temporary configuration outlined above was analyzed by Northeast Utilities Services Company Pipe Stress Engineering, and found to be adequate to maintain the seismic integrity of the Service Water system while valve 2-SW-8.1A was removed. Operation of the Service Water system for the "B" and "C" Reactor Building Closed Cooling Water system heat exchangers was not affected by the work on the "A" heat exchanger piping. Therefore, two trains of Service Water and two trains of Reactor Building Closed Cooling Water remained operable for all Modes of operation, as required by Technical Specifications.

Jumper-Lifted Lead-Bypass Number 2-93-69

This bypass jumper, entitled "Security Grating Installation," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper added steel grating around the area where the auxiliary feedwater suction lines leave the isolation valves, off the condensate storage tank, and go into the underground pipe trench that leads to the turbine building. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The steel grating installed by this bypass jumper provides a security barrier in lieu of the concrete blocks removed to provide access for maintenance work in the CST pipe trench. See Jumper-Lifted Lead-Bypass Number 2-93-70.

Safety Evaluation

There is no credible method by which the steel grating installed by this bypass jumper could present a hazard to the auxiliary feedwater piping in the condensate storage tank trench. The grating was restrained and could not physically impinge on either of the two feedwater pipes. No weight was added to the piping supports, since the grating rested on the concrete floor of the trench. Therefore, the grating has no affect on any safety features.

Jumper-Lifted Lead-Bypass Number 2-93-70

This bypass jumper, entitled "Concrete Block Removal to Facilitate Piping Modifications," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper authorized the temporary removal of up to six concrete blocks covering the underground pipe trench between the condensate storage tank and the turbine building (in the transformer yard). This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The blocks were removed to permit access to the pipes in the trench to replace the existing heat trace with electric space heaters, and to replace damaged insulation.

Safety Evaluation

The equipment of interest in this bypass jumper is the two independent auxiliary feedwater lines which transport water from the condensate storage tank to the auxiliary feedwater pumps. The concrete blocks were handled in a carefully controlled fashion, and the area was kept clear of any material which might have impacted or affected the piping integrity. Administrative controls were in place which required replacement of the concrete blocks prior to being affected by winds of significant strength. A probabilistic risk assessment was performed, and concluded that the probability of a break in both feedwater lines was insignificantly increased for the short time duration (approximately one month) that the concrete block sections would be removed. In addition, a backup system (station fire protection system) was available.

The initial limit of six concrete blocks removed at one time was based on a maintenance crew being "on call" to reinstall them if a high wind warning, or other impending hazard, was identified. This was later amended to allow 12 blocks to be removed at one time, based on a maintenance crew being on site at all times to replace the blocks if circumstances dictated.

Jumper-Lifted Lead-Bypass Number 2-93-72

This bypass jumper, entitled "Spool Piece Installed for 2-CH-223," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper documented the temporary installation of a spool piece in place of 2-CH-223, the letdown heat exchanger cooling water outlet temperature control valve. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The temperature control valve required maintenance. Removing the valve from the system facilitated performance of the maintenance. It is preferred that charging and letdown not be isolated for more than four hours, and the work associated with this valve was estimated to take more than four hours.

Safety Evaluation

Review of the licensing basis accidents for the plant indicates that no licensing basis accidents are affected by this bypass jumper. During the relatively brief periods of time when the charging and letdown system was isolated to permit installation and removal of this bypass jumper, personnel were on hand to restore operability and/or repair leaks as necessary. While the bypass jumper was installed, the letdown system was operated with the letdown heat exchanger cooling water outlet valve, 2-RB-108, throttled as necessary to control letdown temperature. Valve 2-CH-223 is not credited with any specific function in any accident analysis. Therefore, its removal and replacement with a spool piece and manual control of temperature has no design basis impact.

Jumper-Lifted Lead-Bypass Number 2-93-85

This bypass jumper, entitled "Temporary Instrumentation for the Electro-Hydraulic Control System," has been removed.

Description of Jumper-Lifted Lead-Bypass

A digital voltmeter and a strip chart recorder were connected, via shielded cables with an isolated power supply, to monitor the 125 VDC and 24 VDC inputs to the KT106 relay in the electro-hydraulic control unit cabinet. The KT106 relay actuates relays on demand to power the master trip relay, the speed control logic and load control, as well as other affiliated relays. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was installed to monitor control functions of the system to perform functional diagnostic testing.

Safety Evaluation

The equipment installed by this bypass jumper was passive in nature. It did not generate or transmit any signal which could interfere with the normal function of the electro-hydraulic control system. All safety related systems and equipment used to prevent, or mitigate the consequences of any analyzed accident were isolated from the voltage monitoring equipment installed by this bypass jumper. Therefore, this bypass jumper had no impact on fuel, reactor coolant systems boundaries or containment systems required by technical specifications.

Jumper-Lifted Lead-Bypass Number 2-93-86

This bypass jumper, entitled "Reactor Protection System Channel 'A' Low Flow," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper documented the installation of a Fluke multimeter in the reactor protection system, to measure and record the voltage input to the channel "A" low flow bistable. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This bypass jumper was installed to monitor voltage input to determine the cause of intermittent low flow pre-trip signals which had been experienced in channel "A" of the reactor protection system.

Safety Evaluation

For this installation, the worst case scenario was considered to be tripping of the channel "A" low flow bistable. The test meter will be isolated from the rest of the safety loop by a voltage to current isolation card. If the meter were to short out and trip the low flow bistable, the computer indication and local indication for the channel will be unaffected due to the isolation card. There is no fault in the test meter measuring circuit which could prevent a reactor protection system trip.

TESTS

Test Number	Title
T-92-25	Diesel Engine Mechanical Governor Shaft Position Indicator Test
T-92-27	"A" Emergency Diesel Generator (EDG) Digital Reference Unit (DRU) Test
T-92-46	Dynamic Test of 2-SV-4188, 2-MS-201 and 2-MS-202
T-92-53	Nitrogen System Startup Testing
T-92-54	Z2 Actuation of Feedwater Isolation Valves and Pumps
T-92-55	Z1 Actuation of Feedwater Isolation Valves and Pumps
T-92-57	Functional Test of Automatic Test Inserter/Sump Recirculation Actuation Signal (ATI/SRAS) Modifications
T-92-58	Condensate Storage Tank (CST) Nitrogen Overpressure Operability Test
T-92-60	Diesel Engine Speed Switches Loop Test
T-92-62	"A" Emergency Diesel Generator (EDG) Shutdown Solenoid Test
T-92-64	"A" Emergency Diesel Generator (EDG) Electromagnetic Interference (EMI) Functional Test
T-92-67	Simulated Loss of Two Vital AC Panels
T-92-71	Electro-Hydraulic Controls System (EHC) Hydraulic Power Unit Functional Test
T-92-72	Control Element Assembly (CEA) Monitoring Response Time Inservice Test
T-92-73	Steam Generator Replacement Project Pipe Support Inservice Test
T-92-74	Steam Generator Moisture Carryover Test
T-92-75	Cycle 12 Low Power Physics Tests
T-92-76	Cycle 12 Power Ascension Tests

TESTS (CONTINUED)

Test Number	Title
T-93-05	Heater Drain Flow to Drains Coolers Flow Test
T-93-08	Mussel Cook - Intake Bay Temperature Monitoring
T-93-12	Motor Operated Valves (MOV) Brake Removal Test

Number	Title
T-92-25	Diesel Engine Mechanical Governor Shaft Position Indicator Test

Description of Test

This test provided the method to functionally test and calibrate the newly installed governor shaft position indication transmitter.

Reason for Test

This test was designed to permit documentation of accurate mechanical governor output shaft position from zero to 100 percent of shaft rotation and allow for indication adjustment as required for the desired accuracy.

Safety Evaluation

The test was performed without running the engine and the governor shaft was not actually moved. All shaft "movement" was simulated by moving the linkage through its full range. Neither the diesel generator nor the generator shaft were adversely impacted. The non-vital 125 VDC panel D11, breaker 02, protected by a downstream fuse and intervening power supply for the position indication transmitter, supplied power for the test. No safety related circuits were adversely impacted. There was no credible mechanism by which performance of this test could initiate, or change the consequences of any analyzed accident, or create an accident or malfunction of a different type than previously evaluated. The margin of safety was not impacted since the test would not adversely affect any plant system. During performance of this test, the diesel was inoperable, as the air start isolation valves were closed. However, the alternate emergency diesel generator was available for fulfilling safety functions.

Number	Title
T-92-27	"A" Emergency Diesel Generator (EDG) Digital Reference Unit (DRU) Test

Description of Test

This test provided the guidance to document the correct function of the newly installed "A" EDG DRU when its raise/lower input control switches were varied and the Emergency Start Signal relay was actuated.

Reason for Test

This test was performed to record the analog output and function of the new "A" EDG governor DRU to verify its correct function with local and remote governor raise/lower switches and Agastat timer.

Safety Evaluation

Performance of this test was limited to the DRU within the "A" EDG governor system and without the engine running. The diesel generator was not adversely impacted. There was no adverse impact to the station battery supply or wiring because the DRU was powered from DV10 circuit 20 on a 100 amp breaker and further protected with an existing local fuse. There was no credible mechanism by which performance of this test could initiate, or change the consequences of any analyzed accident, or create an accident or malfunction of a different type than previously evaluated. The margin of safety was not impacted since the test would not adversely affect any EDG system. During performance of this test, the EDG was inoperable, as the air start isolation valves were closed. However, the alternate EDG was available for fulfilling safety functions.

Number	Title
T-92-46	Dynamic Test of 2-SV-4188, 2-MS-201 and 2-MS-202

Description of Test

Diagnostic equipment was installed on the three (3) motor operated valves (MOVs). The valves were then stroke tested under maximum practical in-situ conditions.

Reason for Test

This test provided verification of the operability of the three MOVs per the requirements of Generic Letter 89-10.

Safety Evaluation

This test was designed to be performed with the plant in Hot Standby (Mode 3), with the Main Steam Isolation Valves closed with steam header pressure being controlled via the Atmospheric Dump Valves. Use of the Turbine Driven Auxiliary Feedwater Pump was required to develop the necessary flow test conditions. Technical Specifications require that all three Auxiliary Feedwater Pumps be operable in Mode 3, therefore, feedwater would be made available via manual handwheel operation should both Motor Driven Feedwater Pumps fail. Since the Auxiliary Feedwater System is typically used during startup and shutdown under similar loads, the test did not exceed the design basis nor modify any protective features. Test personnel, in communication with the Control Room, would be stationed at the valve and could position the valve as required by the Control Room if the diagnostic equipment adversely affected remote operation of the MOVs. Equipment was verified to function properly prior to returning the system to service. The test did not increase or decrease the rate of feedwater addition to the steam generators nor did it increase the times assumed for initiation of auxiliary feedwater. Any unisolable line breaks as a result of MOVs affected by this test are bounded by the Safety Analysis line break. Therefore, the margin of safety was not impacted during the performance of this test.

Number	Title
T-92-53	Nitrogen System Startup Testing

Description of Test

This procedure provided direction for testing the newly installed nitrogen cryogenic unit, plant supply reducing station, and piping.

Reason for Test

This test was performed to clean, pneumatic pressure test, and verify proper operation of the new equipment.

Safety Evaluation

The nitrogen system is non-safety related and its loss does not adversely affect any safety related components. On-line portions of the nitrogen system were isolated from the sections being tested. Overpressurization of non-safety related instrumentation was prevented by isolation and removal, or isolation with a vent path. Four (4) redundant, 100 percent capacity relief devices (breather valves and rupture disks) prevented overpressurization of the safety related Condensate Storage Tank (CST). The relief devices were sized to pass the full open capacity of the nitrogen supply regulator or the sparger supply regulator. An additional vent path was provided downstream of the sparger test boundary valve. Test nitrogen supplies to the CST were isolated by the valve lineup. The test was performed prior to entry into Mode 3 and since CST availability is required only for Modes 1, 2, and 3, there was no impact on the margin of safety.

Number	Title
T-92-54	Z2 Actuation of Feedwater Isolation Valves and Pumps

Description of Test

This procedure provided specific guidance for testing newly installed operational features in the Main Steam Isolation (MSI) portion of the Engineered Safeguards Actuation Signal (ESAS) control room circuitry. Additional guidance was provided to test modifications performed on the control room Safety Injection Actuation Signal (SIAS) circuitry.

Reason for Test

The test was performed to ensure that valves 2-FW-42A and B, 2-FW-38A and B, 2-FW-41A and B, and 2-FW-51A and B would close automatically and that main feedwater pump turbines H5A and H5B would trip when a signal was initiated from their associated actuation modules within ESAS. Normal operation of the MSI manual initiation and associated valves was verified as functioning properly. The override capability was demonstrated for valves 2-FW-42A and B, 2-FW-38A and B, 2-FW-41A and B, and 2-FW-51A and B after a closure was initiated by the ESAS. The test also verified that the emergency diesel generators would start during a SIAS.

Safety Evaluation

The test was performed prior to entry into Mode 4 and no accidents relating to the MSI actuation apply in either Mode 5 or 6. Since MSI actuation is not required to be operable during Modes 5 or 6, there was no credible mechanism by which performance of this test could initiate, or change the consequences of, any analyzed accident, or create an accident or malfunction of a different type than previously evaluated. The test did not decrease the margin of safety as defined in the basis for any technical specification because it ensured that the newly installed operational features perform as designed to maintain containment integrity during a main steam line break.

Number	Title
T-92-55	Z1 Actuation of Feedwater Isolation Valves and Pumps

Description of Test

This procedure provided specific guidance for testing newly installed operational features in the Main Steam Isolation (MSI) portion of the Engineered Safeguards Actuation Signal (ESAS) control room circuitry. Additional guidance was provided to test modifications performed on the control room Safety Injection Actuation Signal (SIAS) circuitry.

Reason for Test

The test was performed to ensure that valves 2-FW-42A and B, 2-FW-38A and B, 2-FW-41A and B, and 2-FW-51A and B would close automatically and that main feedwater pump turbines H5A and H5B would trip when a signal was initiated from their associated actuation modules within ESAS. Normal operation of the MSI manual initiation and associated valves was verified as functioning properly. The override capability was demonstrated for valves 2-FW-42A and B, 2-FW-38A and B, 2-FW-41A and B, and 2-FW-51A and B after a closure was initiated by the ESAS. The test also verified that the emergency diesel generators would start during a SIAS.

Safety Evaluation

The test was performed prior to entry into Mode 4 and no accidents relating to the MSI actuation apply in either Mode 5 or 6. Since MSI actuation is not required to be operable during Modes 5 or 6, there was no credible mechanism by which performance of this test could initiate, or change the consequences of, any analyzed accident, or create an accident or malfunction of a different type than previously evaluated. The test did not decrease the margin of safety as defined in the basis for any technical specification because it ensured that the newly installed operational features perform as designed to maintain containment integrity during a main steam line break.

Number	Title
T-92-57	Functional Test of Automatic Test Inserter/Sump Recirculation Actuation Signal (ATI/SRAS) Modifications

Description of Test

This procedure provided direction for testing new modifications to the SRAS trip logic and the ATI alarm logic of the Engineered Safeguards Actuation Signal (ESAS). Guidance was also provided for testing the newly installed ATI power supply disable circuit.

Reason for Test

The test provided verification of proper trip signal processing at the SRAS actuation modules including the SRAS Actuation Signal annunciators for matrix combinations AB, AD, BC, and CD, since combinations AC and BD were removed from the matrix. This test verified proper functioning of the modified ATI logic and proper operation of the ATI de-energizing circuit. The test also demonstrated the ability of the ATI to test the SRAS logic.

Safety Evaluation

The test was performed in Mode 6 with the core fully offloaded, which limited the consequences of an inadvertent ESAS actuation. Removal of actuation relays and jumper installations prevented SRAS and under voltage actuation which prevented the trip of the operating Low Pressure Safety Injection (LPSI) Pump and a Loss of Normal Power (LNP). Procedures were in place to instruct operations personnel how to manually start and load the Emergency Diesel Generator if an LNP were to occur. Inadvertant actuation of Safety Injection Actuation Signal, Containment Isolation Actuation Signal, Enclosure Building Filtration Actuation Signal, and Main Steam Isolation have no affect in Mode 6. There was no impact on the margin of safety since the portions of ESAS required in Mode 6 remained in service and there was no potential for interruption of LPSI pump service as the SRAS function was blocked.

Number	Title
T-92-58	Condensate Storage Tank (CST) Nitrogen Overpressure Operability Test

Description of Test

This test provided a method to pressure and vacuum test the CST.

Reason for Test

The test was performed to verify the adequacy of structural reinforcements and the operability of pressure control components and alarms which were installed during the CST conversion from a vented to a pressurized tank.

Safety Evaluation

The test had no adverse impact on the loss of Auxiliary Feedwater Supply (AFW) because it was restricted to Modes not requiring the availability of the CST (Modes 5 and 6). Controls were established to maintain positive and negative pressures sufficiently below design limits during the test. Installed relief devices were sufficient to prevent exceeding the design limits of the CST during both the test and normal operation. Failure of the CST due to pressure or vacuum is bounded by the loss of AFW accident. Since the CST availability was not required, the test did not adversely impact the margin of safety.

Number	Title
T-92-60	Diesel Engine Speed Switches Loop Test

Description of Test

A portable signal generator was connected to the Emergency Diesel Generator (EDG) speed control circuitry to test the new "Dynalco" speed switch and new "Kilovac" relays.

Reason for Test

The test verified the performance of the speed switch and proper actuation of the relays.

Safety Evaluation

The test was performed without the engine running, therefore, the EDG was not adversely impacted. Existing Low Speed Relay (LSR) and High Speed Relay (HSR) devices were not changed in any way. Test prerequisites established controls to prevent any adverse impact on related components from LSR and HSR actuation. The station battery would not be adversely impacted because the 125 VDC system was sufficiently isolated. Since the test frequency signal was isolated and separated from all safety related systems, there was no impact on the margin of safety. During performance of this test, the EDG was inoperable, as the air start isolation valves were closed. However, the alternate EDG was available for fulfilling safety functions.

Number	Title
T-92-62	"A" Emergency Diesel Generator (EDG) Shutdown Solenoid Test

Description of Test

This test provided a method to verify the new governor shutdown solenoid.

Reason for Test

This test documented proper functioning of the shutdown solenoid by verifying proper continuity and that it energized when the control relay energized it.

Safety Evaluation

The test was limited to the evaluation of the new EGB-13P governor shutdown solenoid without the EDG running, therefore, the diesel generator was not adversely affected. There was no adverse impact to the station battery cabling since the new speed switch wiring is powered from DV10 circuit 20 on a 100 amp breaker and is, further, protected with an existing local fuse. This test had no impact on the margin of safety as this test would not adversely affect any EDG systems and thus did not affect the basis of any of the Technical Specifications. No unreviewed safety questions were introduced. During performance of this test, the EDG was inoperable, as the air start isolation valves were closed. However, the alternate EDG was available for fulfilling safety functions.

Number	Title
T-92-64	"A" Emergency Diesel Generator (EDG) Electromagnetic Interference (EMI) Functional Test

Description of Test

This test provided a method to verify the new "A" EDG governor control and speed switch function against EMI.

Reason for Test

This test documented the correct function of the governor and speed switches in response to EMI from hand-held walkie-talkie operation.

Safety Evaluation

Diesel operation would not be affected since the output breaker was open and it was not considered operable during this test. Testing was limited to evaluation of the new EDG control equipment with the EDG breaker open. This test had no impact on the margin of safety as this test would not adversely affect any EDG systems and thus did not affect the basis of any of the Technical Specifications. No unreviewed safety questions were introduced.

Number	Title
T-92-67	Simulated Loss of Two Vital AC Panels

Description of Test

This test outlined the method to simulate a loss of two vital AC panels.

Reason for Test

This test demonstrated, under simulated conditions, that a loss of two vital AC panels would not create a simultaneous Sump Recirculation Actuation Signal and Safety Injection Actuation Signal conflict, and would not create a recurring load shedding condition in an undervoltage Loss of Normal Power response.

Safety Evaluation

Testing was limited to one facility at a time with the other facility equipment and associated Emergency Diesel Generator available for manual actuation. Components being subjected to the test were carefully limited. Manual operation and/or alignment of systems required for nuclear fuel movement was established. The tested facility was fully restored prior to commencing the next facility test. Testing was performed prior to the Mode requirement for Engineered Safeguards Actuation Signal availability and did not affect the normal control features of the equipment. Therefore, the test did not reduce the margin of safety of the bases of any Technical Specifications.

Number	Title
T-92-71	Electro-Hydraulic Controls System (EHC) Hydraulic Power Unit Functional Test

Description of Test

This test provided verification of the modified EHC System Hydraulic Power Unit.

Reason for Test

The test documented proper operation of components associated with the EHC System Hydraulic Power Unit.

Safety Evaluation

The test was performed with the turbine off-line and the steam lines depressurized. Previously analyzed failure Modes of the EHC system, leakage and inability to properly control the overspeeding of the turbine, were not changed and would not be affected by the test.

Number	Title
T-92-72	Control Element Assembly (CEA) Monitoring Response Time Inservice Test

Description of Test

This test verified the operation of the new CEA Position Display System (CEAPDS) and the Plant Process Computer (PPC) CEA monitoring program.

Reason for Test

This test energized the Control Element Drive System (CEDS) to withdraw control rods as required to test the CEDS interlocks provided by the CEAPDS and the PPC.

Safety Evaluation

This test was performed prior to entry into Mode 2 and the CEDS was operated in accordance with normal plant operating procedures. The Technical Specification required Reactor Coolant System boron concentration for Modes 3, 4, and 5 was maintained to ensure the Shutdown Margin was adequate, even if all CEAs were fully withdrawn from the core. No other types of reactivity changes were required by the test. The test did not challenge any plant protective boundaries, nor did it affect any parameters which form the basis for any Technical Specification. Therefore, performance of the test was not an unreviewed safety question.

Number	Title
T-92-73	Steam Generator Replacement Project Pipe Support Inservice Test

Description of Test

This test was developed to visually inspect new and modified pipe supports associated with the Main Steam, Feedwater, Blowdown, Wet Lay-Up, and Nitrogen lines.

Reason for Test

Visual examinations were conducted to determine the general mechanical and structural conditions of the pipe supports and piping due to thermal growth.

Safety Evaluation

This test was performed prior to entry into Mode 2. Visual verification of pipe support structural integrity has neither a direct nor indirect affect on any plant system. Performance of this test was a verification of design requirements and, therefore, had no affect on plant systems. The test had no affect on the consequences of the current analysis, the malfunction of safety systems or creation of a new or different accident.

Number	Title
T-92-74	Steam Generator Moisture Carryover Test

Description of Test

This test provided the instructions to inject a liquid radiotracer (Na-24)into the plant secondary system.

Reason for Test

This test determined the moisture carryover from each of the new steam generators via Na-24 transport on water droplets.

Safety Evaluation

The test was performed at 100 percent power and all secondary systems would operate as designed and within their normal pressure parameters. Similar tests have been performed at other plants. It was concluded that the 5 ppb expected increase of stable sodium over the short test duration would have no long-term affect on the quality or performance of the steam generators. Performance of the test was bounded by the Increase in Steam Flow, Steam Line Break and Steam Generator Tube Failure Analyses in the Final Safety Analysis Report. The test did not change any system design or operating parameter. The unit would meet the Limiting Condition for Operation and would not be required to enter any action statement during the tracer injection and test equipment setup. This test did not affect the consequences of the current analysis, the malfunction of safety systems, or creation of a new or different accident.

Number	Title
T-92-75	Cycle 12 Low Power Physics Tests

Description of Test

This test performed post-refueling Low Power Physics Testing (LPPT) of the reactor.

Reason for Test

The testing measured the following reactivity parameters: Isothermal Temperature Coefficient, Control Rod Worths, Rodded and Unrodded Critical Boron Concentrations. The testing also completed the Technical Specification surveillance for Shutdown Margin and Moderator Temperature Coefficient required after each refueling.

Safety Evaluation

This test was performed at a power level below 1 percent and within the requirements of the Technical Specification Special Test Exceptions. Testing methods utilized the normal operating controls for the plant sampling, atmospheric dump valve, Control Element Drive and the Chemical Volume and Control systems. These tests were bounded by the assumptions and consequences of the accidents evaluated in the safety analyses and would not challenge any of the protective barriers.

Number	Title
T-92-76	Cycle 12 Power Ascension Tests

Description of Test

This test performed physics testing during the transition to full power operation from Lower Power Physics testing.

Reason for Test

This test was performed to verify that the actual measured plant parameters were in agreement with the Siemens Nuclear Power (nuclear fuel supplier) core physics design model. The test also completed reactivity and power distribution related Technical Specification surveillances and industry standard tests in the proper sequence and at appropriate power levels.

Safety Evaluation

All testing activities were performed within the assumptions of the safety analyses and did not alter or affect the normal operation of any plant equipment. Performance of the tests were within the Technical Specification Limiting Conditions for Operation. The tests did not affect the results of any Final Safety Analysis Report accident analyses and did not challenge any of the protective boundaries.

Number	Title
T-93-05	Heater Drain Flow to Drains Coolers Flow Test

Description of Test

This test provided the instructions and controls to throttle the 6A (6B) feedwater heater level control valve outlet isolation valve.

Reason for Test

This test manually throttled the outlet stop valves on both Drains Coolers X7A and X7B, separately, to determine an optimum valve position for creating sufficient back pressure to allow single phase flow (water) through the Drains Coolers.

Safety Evaluation

The Drains Coolers do not perform an active safety function. No components were added or replaced during the test. Flow through the Drains Cooler shell side was the process parameter which could have been affected by the test. However, loss of this flow path due to full closure of the manual valves, was determined to be within the bounds of the original design. Manual throttling of the outlet stop valves did not change the operation of the feedwater heaters. Feedwater heaters were not bypassed during the test. The test also established controls to manipulate the Drains Coolers' Normal Level Control Valves. Normal Level Control Valves were returned to automatic level control once the outlet isolation valve optimum position was determined. The High Level Dump Valves remained in automatic control to respond to changing 6A and 6B feedwater heater levels.

Number	Title
T-93-08	Mussel Cook - Intake Bay Temperature Monitoring

Description of Test

This test provided a means of obtaining temperature profiles within Intake Structure bays during a mussel cook evolution.

Reason for Test

Data obtained from this test would be used in the planning of future mussel cook evolutions.

Safety Evaluation

This test introduced weighted thermocouple assemblies in Intake bay locations approximately 12-14 feet away from the Service Water pump suction and 23-25 feet away from the Circulating Water pump suction. Thermocouple assembly fabrication and controls (which limited installation of the thermocouple assemblies to one bay with an operating Service Water pump at any one time) would not impact any previously analyzed scenario, nor increase the possibility of a different accident or malfunction of equipment important to safety. The only possible malfunction associated with this test is a thermocouple assembly entering a Service or Circulating Water pump suction and damaging a pump. Each thermocouple assembly weighs 15 pounds. The water velocity at the entrance to the bay is 0.81 feet per second which, based on engineering judgement, is not sufficient to carry the weight into the pump suction.

Number	Title
T-93-12	Motor Operated Valve (MOV) Brake Removal Test

Description of Test

This test provided the instructions to stop Feedwater MOVs 2-FW-38A and B, and 2-FW-42A and B, in a partially closed position.

Reason for Test

This test was performed to verify that the valve disc and stem would not drift (after having their motor brakes removed) under a line pressure of at least 834 psig and essentially no differential pressure.

Safety Evaluation

This test was performed during standard plant start-up operations. Neither the Main Steam Line Break Event or the Loss of Feedwater event were considered applicable, since the feed regulator valves, feed regulator by-pass valves, and back-up isolation valves were shut. Therefore, steam line isolation was established and controls were in place to re-open the test valve if required. Since partial closing of the valves did not impact the Feedwater pump recirculation flow, pump operation would not be affected. Stopping the valve in a partially closed position with no flow is less severe than full closure, due to a Main Steam Isolation signal which could occur with full feedwater flow. The margin of safety was not impacted because none of the protective boundaries would be affected by the test.

EXPERIMENTS

There were no experiments conducted by Millstone Unit No. 2 in 1993.

CHALLENGES TO RELIEF/SAFETY VALVES

In accordance with Technical Specification 6.9.1.5, the following is a report of challenges to relief/safety valves during 1993.

On June 3, 1993, at 1624 hours, with the plant in Mode 1 at 100% power, the main turbine generator Electro-Hydraulic Control system initiated a signal that caused the main turbine intercept valves and control valves to close. The main turbine load rapidly decreased. The resulting load imbalance between the reactor plant and the steam plant caused an increase in reactor temperature and pressure, opening both Power Operated Relief valves (PORVs) and several steam generator safety valves. The reactor tripped on high pressurizer pressure. The pressurizer PORVs and steam generator safety valves properly reseated. This incident was reported by Licensee Event Report LER-93-013, dated July 2, 1993.

STEAM GENERATOR TUBING INSERVICE INSPECTION

The new Millstone Unit No. 2 Steam Generators began service in January, 1993, with plant restart following the steam generator replacement outage. There have been no tubing inservice inspections performed during 1993.

PRIMARY COOLANT IODINE SPIKING

During 1993, the specific activity of the primary coolant did not exceed the limits stated in the Technical Specifications.

REGULATORY GUIDE 1.16 REPORT FOR 1993

REGULATORY GUIDE 1.16 REPORT FOR 1993
NORTHEAST NUCLEAR ENERGY CO. UNIT 2

DATE: 2/ 3/94

WORK & JOB FUNCTION	NUMBER OF PERSONNEL(>100 MREM) STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES	TOTAL MAN-REM STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES
REACTOR OPERATIONS & SURVEILLANCE						
MAINTENANCE PERSONNEL	8	0	1	2.93	0.07	1.49
OPERATING PERSONNEL	22	1	0	7.43	0.13	0.06
HEALTH PHYSICS PERSONNEL	18	0	5	5.33	0.01	1.98
SUPERVISORY PERSONNEL	0	0	0	0.08	0.00	0.01
ENGINEERING PERSONNEL	2	0	0	0.63	0.24	0.17
ROUTINE MAINTENANCE						
MAINTENANCE PERSONNEL	15	0	4	3.23	0.01	2.29
OPERATING PERSONNEL	0	0	0	0.16	0.00	0.12
HEALTH PHYSICS PERSONNEL	2	0	2	1.19	0.00	0.37
SUPERVISORY PERSONNEL	0	0	0	0.01	0.00	0.00
ENGINEERING PERSONNEL	0	1	0	0.15	0.32	0.18
INSERVICE INSPECTION						
MAINTENANCE PERSONNEL	0	0	0	0.00	0.00	0.00
OPERATING PERSONNEL	0	0	0	0.00	0.00	0.00
HEALTH PHYSICS PERSONNEL	0	0	0	0.00	0.00	0.01
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.00	0.00	0.01
SPECIAL MAINTENANCE						
MAINTENANCE PERSONNEL	18	0	7	6.00	0.00	3.46
OPERATING PERSONNEL	3	0	0	1.49	0.08	0.02
HEALTH PHYSICS PERSONNEL	1	0	0	0.99	0.00	0.02
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	2	1	6	0.63	0.28	1.65
WASTE PROCESSING						
MAINTENANCE PERSONNEL	0	0	0	0.29	0.00	0.30
OPERATING PERSONNEL	1	0	0	0.36	0.00	0.02
HEALTH PHYSICS PERSONNEL	4	0	6	2.02	0.00	1.28
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.02	0.04	0.00
REFUELING						
MAINTENANCE PERSONNEL	18	0	11	2.90	0.07	3.63
OPERATING PERSONNEL	0	0	0	0.20	0.00	0.05
HEALTH PHYSICS PERSONNEL	2	0	2	0.06	0.00	0.11
SUPERVISORY PERSONNEL	0	0	0	0.01	0.00	0.03
ENGINEERING PERSONNEL	2	0	1	0.35	0.03	0.27
TOTAL						
MAINTENANCE PERSONNEL	59	0	23	17.43	0.15	11.91
OPERATING PERSONNEL	26	1	0	9.80	0.21	0.33
HEALTH PHYSICS PERSONNEL	27	0	15	10.68	0.01	4.42
SUPERVISORY PERSONNEL	0	0	0	0.12	0.00	0.04
ENGINEERING PERSONNEL	6	2	7	2.00	1.01	2.42
GRAND TOTAL	118	3	45	40.03	1.40	19.13

REGULATORY GUIDE 1.16 REPORT FOR 1993

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NORTHEAST NUCLEAR ENERGY CO. UNIT 2

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SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
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GRAND TOTAL	118	3	45	40.03	1.40	19.13

MILLSTONE UNIT NO. 3

TABLE OF CONTENTS

INTRODUCTION

None of the plant design changes, procedure changes, jumpers-lifted leads-bypasses, tests, or experiments described herein constitute (or constituted) an unreviewed safety question per the criteria of 10CFR50.59.

PLANT DESIGN CHANGES

PDCR Number	Title
3-85-006	Supports Required for Seismic Interaction
3-85-014	Supports Required for Seismic Interaction
3-90-243	Replacement of Letdown Heat Exchanger Studs
3-91-024	Service Water Pump Material Changeout
3-91-114	T_{avg} and Auctioneered T_{avg} Deviation Alarm Setpoint Change
3-91-116	Deletion of Manual CO_2 Discharge from the Main Control Room
3-91-170	Deletion of Reactor Coolant System (RCS) Loop Relief Lines Flow Indication
3-92-010	Extension of the Class 3 Boundary For the Service Water System Supply to the Post Accident Sampling System (PASS)
3-92-030	Sodium Hypochlorite Support Modifications
3-92-060	Installation of Domestic Water Strainer and Changeout of Backflow Preventer
3-92-061	Power Operated Relief Valve (PORV) Position Indication Cover Gasket Replacement
3-92-077	Modification of the Condenser Available Interlock Circuit
3-92-090	Millstone Unit No. 3 Station Blackout Diesel Generator Fire Suppression System Installation
3-92-091	Service Water Pump Cubicle Sump Pump Modification
3-92-092	Modifications to Service Water Inlet Piping to Charging Pump Cooling Heat Exchanger
3-92-095	Phase 1 Station Blackout Diesel Stand Alone Integration
3-92-096	Phase II Alternate AC Diesel Generator Integration
3-92-106	Loose Parts Monitoring Upgrade

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-92-119	Removal of Flow Switch Interlock from Auxiliary Building Filter Fan Controls
3-92-120	Vacuum Priming System Upgrade
3-92-124	Amertap System Removal, Condenser Discharge Piping Upgrade, Condensate System Pump Down Line Piping Upgrade
3-92-129	Replacement of Reactor Plant Component Cooling Water (RPCCW) Heat Exchanger Cross-Connect Valve
3-93-003	Installation of Transfer Trip for Tripping Main Generator Output Breaker
3-93-006	Revised Rod Control System Compensation Parameters
3-93-008	Arcor Coating of the Inside of Service Water System Spools
3-93-009	Service Water System Piping Modifications
3-93-011	Modification to Generator Line Protection Pilot Wire Scheme
3-93-012	Technical Support Center Damper Modifications
3-93-013	Feedwater Flow Control Valves Position Indication
3-93-015	Replacement of Recirculation Spray Pump Suction Valves
3-93-016	Replacement of Snubbers Due to Functional Failures
3-93-023	Heated Junction Thermocouple Probe and Cable Replacement
3-93-024	Core Exit Thermocouple Connector and Cable Replacement

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-93-027	Reactor Coolant Pump (RCP) No. 1 Seal Leakoff High Flow Alarm Setpoint Change and Rescaling of All No. 1 Seal High Range Leakoff Flow Transmitters
3-93-034	Permanent Reactor Cavity Seal Installation
3-93-037	Arcor Coating of the Inside Diameter of Service Water System Spools
3-93-044	Make Duct Support in Main Steam Valve Building Removable
3-93-050	Auxiliary Feedwater Pump Design Change
3-93-054	Reactor Flange Shield Installation
3-93-060	Installation of Forced Air Cooling Fans Within the 7300 System
3-93-061	Revising Motor Operated Valve's Motor Protection
3-93-062	Replacement of Turbine Driven Auxiliary Feedwater (TDAFW) Pump Valve Room Air Conditioning System
3-93-064	Rewire Relays in the Engineered Safety Features Actuation System (ESFAS)
3-93-067	Auxiliary Building Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Area Ventilation Heaters
3-93-074	Unit 3 Ecolochem Building Installation
3-93-076	Modification to SIGMA Refueling Machine Control Console
3-93-082	Installation of Rockbestos Cable to Replace Thermo-Lag Fire Barrier
3-93-090	Reload Design for Millstone Unit No. 3 Cycle 5
3-93-092	Lube Oil Reservoir Tank Indication and Alarm
3-93-093	Boron Concentration Measurement System Removal

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-93-097	Addition of Isolation Valves to Containment Hatch for Local Leak Rate Testing (LLRT) Tube Connection
3-93-106	Rerouting of the Circuit Associated with a Reactor Plant Ventilation Temperature Switch to Comply with Separation Requirements
3-93-109	Motor Driven Auxiliary Feedwater Pump Trip Circuit Modification
3-93-116	Abandonment of Pressurizer Liquid Sample Line
3-93-120	Installation of Rigging Support for Auxiliary Feedwater Pump
3-93-121	Addition of Reactor Trip Interlock to Inadequate Core Cooling Monitor Annunciation
3-93-124	Engineered Safety Features (ESF) Building Ventilation Supply Fan Flow Switch Setpoint Revision
3-93-125	Replacement of the "A" Quench Spray Pump Anchor Bolts
3-93-126	Charging and Reactor Coolant System Valve Yoke Bolts Replacement
3-93-128	Modify Supports for "B" Safety Injection Pump Miniflow Isolation Valve to be Removable
3-93-129	Replacement of Service Water Inlet Isolation Valve to Reactor Plant Component Cooling Water Heat Exchanger
3-93-131	Modification to Reactor Head Vent Isolation Valves
3-93-133	Installation of Finer Cartridges for Letdown Reactor Coolant and Seal Water Injection Filters
3-93-135	Modifications to Diesel Jacket Water Expansion Line

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-93-136	Jib Crane in the Reactor Head Storage Area
3-93-137	Canopy Seal Repair of Spare Capped Nozzles K-4 and E-9
3-93-140	Limitorque Actuator Spring Pack Replacement
3-93-146	Deletion of Radioactive Liquid Waste Conductivity Instruments
3-93-147	Substitution of Excess Flow check Valves with Steam Traps
3-93-149	Steam Generator Tube U-Bend Stabilizer
3-93-153	Missile Shield Shim Modification for Control Rod Drive Mechanism (CRDM) Shroud Cooler
3-93-159	Repower Ventilation Flow Switches from Uninterruptible Safety-Related Power Sources
3-93-160	"C" Reactor Coolant Pump (RCP) Replacement
3-93-161	Limitorque Actuator Gearing Changes
3-93-167	Fuel Transfer Tube Closure Bolt Reduction
3-93-174	"B" Reactor Coolant Pump (RCP) Replacement
3-93-177	"B" Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications
3-93-178	"A" Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications
3-93-180	"D" Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications
3-93-184	Modification of Reactor Coolant Pumps (RCP) Oil Collection System

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-93-185	"A" Reactor Coolant Pump (RCP) Replacement
3-93-186	"D" Reactor Coolant Pump (RCP) Replacement
3-93-200	Auxiliary Building Filtration System and Supplemental Leak Collection and Release System (SLCRS) Enhancements
3-93-204	Support Modification to Portion of Main Steam Line
3-93-205	Auxiliary Building Filtration System and Supplemental Leak Collection and Release System (SLCRS) Enhancements
3-93-210	Revision to Flow Switch Setpoints for Supplemental Leak Collection and Release System (SLCRS) Fans
3-93-218	Over Power Delta-T (OPDT) and Over Temperature Delta-T (OTDT) Turbine Runback Setpoint Revision

Software Number	Title
M3-91-23333	Supplementary Leak Collection and Release System (SLCRS) Normal Range Radiation Monitor Noise Spike Filter Software Modification
3-92-17454	Steam Generator Blowdown Sample Radiation Monitor Software Modification
M3-93-01575	Waste Neutralization Sump Radiation Monitor Low Flow Alarm and Low Pressure Alarm Delays
M3-93-10641	Reactor Plant Component Cooling Water (RPCCW) Radiation Monitor Noise Spike Filter and Low Flow and Low Pressure Alarm Delays
M3-93-15026	Inadequate Core Cooling Monitor (ICCM) Software Implementation Package for ICCM Alarm Unit
3-93-20623	Modify Redundant Measurement Program to Correct Reactor Coolant System (RCS) Leakage Program

Plant Design Change Number 3-85-006

This change, entitled "Supports Required for Seismic Interaction," is complete.

Description of Change

This change provided for the installation of seismic pipe restraints on portions of non-safety related piping systems which were not physically isolated from Seismic Category I systems. This change had been held open pending documentation of several installations.

Reason for Change

These changes were required to meet the license requirement to resolve the interaction of seismic and non-seismic systems.

Safety Evaluation

The location of the restraints meets the original design basis of the plant. The design of the restraints was based on similar types of restraints for similar pipe sizes in Category I systems. Fabrication, installation, and inspection of the restraints was in accordance with original specifications.

Plant Design Change Number 3-85-014

This change, entitled "Supports Required for Seismic Interaction," is complete.

Description of Change

This change provided for the installation of seismic pipe restraints on portions of non-safety related piping systems which were not physically isolated from Seismic Category I systems. This change had been held open pending documentation of several installations.

Reason for Change

These changes were required to meet the license requirement to resolve the interaction of seismic and non-seismic systems.

Safety Evaluation

The location of the restraints meets the original design basis of the plant. The design of the restraints was based on similar types of restraints for similar pipe sizes in Category I systems. Fabrication, installation, and inspection of the restraints was in accordance with original specifications.

Plant Design Change Number 3-90-243

This change, entitled "Replacement of Letdown Heat Exchanger Studs," is complete.

Description of Change

This change replaced 21 of 28 originally installed studs and nuts with studs and nuts manufactured from a more corrosion resistant material. The new studs and nuts are torque tight to 280 foot-pounds in lieu of the previous 250 foot-pounds.

Reason for Change

The originally installed bolting material is very susceptible to boric acid corrosion from leakage occurring at the tube side flange of the heat exchanger. The new material is a more corrosion resistant material. Because of corrosion, seven of the old studs could not be removed using recommended torques. Excessive torque was not applied for fear that they would fracture.

Safety Evaluation

An evaluation determined the structural integrity was well within the design limits for the heat exchanger with only 21 of the bolts being used. No credit was taken for the seven remaining carbon steel bolts as it was conservatively assumed that they would eventually corrode and lose all load bearing capability.

Plant Design Change Number 3-91-024

This change, entitled "Service Water Pump Material Changeout," is complete.

Description of Change

The impeller, wear rings, upper, intermediate, and lower shafts, shaft couplings, and various wear rings on the "A" Service Water pump were replaced with new materials. Additionally, the pump columns, discharge elbow, diffuser, and suction bowl were coated with Arcor S-30 for galvanic protection.

Reason for Change

The existing pump internals have shown excessive wear and the new materials are more resistant to the harsh service. The new materials are estimated to increase the service life of the pump by a factor of three.

Safety Evaluation

This modification to the Service Water pump internals will not adversely impact the operation of the Service Water system.

Plant Design Change Number 3-91-114

This change, entitled "T_{avg} and Auctioneered T_{avg} Deviation Alarm Setpoint Change," is complete.

Description of Change

The setpoints of eight T_{avg} and auctioneered T_{avg} deviation alarm bistables were changed from 2°F to 3°F.

Reason for Change

The referenced setpoints were expected to be adjusted during startup and subsequent operation. The subsequent adjustments that were made placed the setpoint range just beyond the normal operating variations. These alarms are intended to provide an indication of instrument failure. The previous setpoints failed to fulfill the intended function. This change was required because Millstone Unit No. 3 was experiencing an upper plenum anomaly which caused fluctuations in the T_{avg} signal. These fluctuations resulted in numerous nuisance alarms.

Safety Evaluation

The safety evaluation addressed the increase in the alarm setpoint from 2°F to 3°F. Equipment and system performance were not adversely affected by the change. The reliability and function of the alarms were not impacted by this change.

The T_{avg} deviation is not utilized in any of the accident analysis to direct the operator to take any mitigation action. No explicit credit is taken for this alarm in the plant's safety analysis.

Plant Design Change Number 3-91-116

This change, entitled "Deletion of Manual CO_2 Discharge from the Main Control Room," is complete.

Description of Change

This change removed the ability to manually generate a CO_2 discharge from the fire protection console located in the Control Room.

Reason for Change

This change was a personal safety enhancement. It eliminated the potential for an inadvertent CO_2 discharge from the fire protection console. The automatic and local manual CO_2 discharge features were not impacted by this change.

Safety Evaluation

This change did not impact the availability of the CO_2 fire suppression system.

Plant Design Change Number 3-91-170

This change, entitled "Deletion of Reactor Coolant System (RCS) Loop Relief Lines Flow Indication," is complete.

Description of Change

Isolation valves for RCS loop relief lines flow indication were isolated and will remain isolated since the flow indication switches are no longer in use. Cables to the switches were disconnected and equipment was abandoned in place.

Reason for Change

The switches are no longer required or used.

Safety Evaluation

This change isolates valves and disconnects wiring to flow switches that are not in service.

Plant Design Change Number 3-92-010

This change, entitled "Extension of the Class 3 Boundary for the Service Water System Supply to the Post Accident Sampling System (PASS)," is complete.

Description of Change

The seismic and safety classification boundary for the Service Water supply and return for the PASS cooler was moved. The isolation valves closer to the cooler now mark the boundary as opposed to the isolation valves further downstream.

Reason for Change

The original Service Water isolation valves for the PASS cooler were determined to be inaccessible post-accident. Moving the boundary enables post-accident access to these valves.

Safety Evaluation

The piping which is now included in the boundary has been seismically qualified. The design of the Service Water system has not been changed.

Plant Design Change Number 3-92-030

This change, entitled "Sodium Hypochlorite Support Modifications," is complete.

Description of Change

The existing piping supports for the sodium hypochlorite injection piping in the Circulating Water bays were modified. The anchor nuts were changed from carbon steel to stainless steel to eliminate the potential for galvanic corrosion.

Reason for Change

The original pipe supports for the sodium hypochlorite injection piping failed due to galvanic corrosion between parts that were not identified as being carbon steel. The design change ensured that all the materials employed were made of the same material and that where possible, supports which were located in the water were removed.

Safety Evaluation

The change ensures that the piping and supports will not fail in the marine environment. The change results in the system being more reliable and does not introduce any new accidents.

Plant Design Change Number 3-92-060

This change, entitled "Installation of Domestic Water Strainer and Changeout of Backflow Preventer," is complete.

Description of Change

This change added a strainer for the demineralized water storage tank (complete with blowdown valve) upstream of the backflow preventer. Also, it replaced the existing backflow preventer and associated isolation valves with new models. An existing temporary support was replaced with permanent pipe supports and necessary piping layout changes were made.

Reason for Change

The strainer was installed to prevent fouling of the check valves in the backflow preventer. The backflow preventer was replaced because it was not similar to others installed in the plant. Replacement allowed a reduction in the required spare parts inventory.

Safety Evaluation

The safety evaluation addressed the affects of the loads imposed on the existing demineralized water storage tank blockhouse structure by the new supports and the affects of a malfunction of components added by this modification. The modifications performed did not have any effect on any design basis accident or its consequences.

Plant Design Change Number 3-92-061

This change, entitled "Power Operated Relief Valve (PORV) Position Indication Cover Gasket Replacement," is complete.

Description of Change

This change replaced the solenoid housing gasket and the limit switch housing gasket for the PORVs with Grafoil gasket material.

Reason for Change

The actual temperatures in the area of the PORVs was higher than originally expected. The gasket material was not designed to operate in this environment. Therefore, the Grafoil material was selected.

Safety Evaluation

Replacement of the gasket material satisfied the original design of the PORVs. This change reduces the likelihood of PORV failure.

Plant Design Change Number 3-92-077

This change, entitled "Modification of the Condenser Available Interlock Circuit," is complete.

Description of Change

The condenser available interlock allows steam to bypass the turbine and be dumped directly to the condenser.

The condenser circulating water pump circuit breaker auxiliary contacts were interlocked with the condenser vacuum switches to prevent actuation of turbine steam bypass unless one circulating water pump motor per condenser bay is operating.

Reason for Change

This modification reduced the probability of damage to the condensers due to loss of condenser vacuum because of loss of circulating water cooling to the condensers. Without a circulating pump running, the condenser will eventually lose vacuum and become unavailable. During a plant trip in 1991, this condition occurred and resulted in the main condenser rupture disks actuating.

Safety Evaluation

The safety evaluation addressed the affect of the design change on Millstone Unit No. 3 electrical systems. This design change augmented an existing control system and did not change electrical system loading.

The turbine steam bypass is not required for safe shutdown and is not a safety related system. The design change did not alter a Class 1E electrical system, did not alter electrical separation, or add a heat source to existing electrical or mechanical systems.

Plant Design Change Number 3-92-090

This change, entitled "Millstone Unit No. 3 Station Blackout Diesel Generator Fire Suppression System Installation," is complete.

Description of Change

A fire suppression system was installed in the new station blackout diesel generator enclosures. The system utilizes a combination of smoke and heat detectors for fire detection, which serve to actuate a fire water valve feeding numerous sprinkler heads located throughout the enclosure. The new fire suppression system is fed from an existing fire water supply header in the vicinity of the station blackout diesel generator enclosures.

Reason for Change

A new 2600 kW diesel generator was installed under Plant Design Change Number 3-92-095 as an alternate AC power supply to meet the requirements of the station blackout rule. As a result of this new installation, a fire suppression system for personnel and equipment safety was necessary for the new station blackout diesel generator enclosure.

Safety Evaluation

The safety evaluation addressed the impact of the new fire suppression system on the existing station fire protection system and concluded that it was acceptable with no adverse impact. The header used to supply the diesel has sufficient capacity and the areas covered are separated to prevent fire spreading.

Plant Design Change Number 3-92-091

This change, entitled "Service Water Pump Cubicle Sump Pump Modification," is complete.

Description of Change

A hole was corebored in each of the Service Water cubicle sumps to provide a drainage path into the intake for the Service Water pump seal leakoff. A pipe sleeve and safety-related valve were installed in each hole to ensure the drainage path could be closed during severe weather or flooding. A vital powered receptacle was installed to provide power for the sump pumps for use in severe weather. Modifications were made to the piping to allow easy changeout of the pump and a replacement pump was mounted in a nearby storage box for use if the installed pump should fail.

Reason for Change

The Service Water pump cubicles are designed as watertight rooms that are protected from external flooding sources by watertight doors. With the doors closed, there exists the potential for internal flooding of the cubicles if the Service Water pump seal leakoff path is isolated. The normal source of leakage is the Service Water pump seal leakoff and Service Water pump strainer leakage. Service Water piping leakage is another potential source of water in the cubicle. The modification provided a more reliable means of removing water from the cubicle.

Safety Evaluation

The effects of the corebore were reviewed and found that the corebore did not alter the ability of the slab section to carry the design loads and did not reduce the margin of safety. The installation of the safety-related valve, which is administratively closed in times of severe weather, accomplishes the same positive isolation as the original slab. Therefore, there is no increase to the probability of the occurrence of a malfunction of the equipment important to safety.

Plant Design Change Number 3-92-092

This change, entitled "Modifications to Service Water Inlet Piping to Charging Pump Cooling Heat Exchanger," is complete.

Description of Change

The inlet Service Water piping to the "A" Charging Pump Cooling heat exchanger relocated and changed the inlet isolation valve.

Reason for Change

The piping geometry change improved the flow characteristics of the inlet piping. The valve change similarly improved the flow characteristics of the system.

Safety Evaluation

This modification will not affect the intended function of any safety-related system. The change meets or exceeds all original design parameters.

Plant Design Change Number 3-92-095

This change, entitled "Phase 1 Station Blackout Diesel Stand Alone Integration," is complete.

Description of Change

This modification was the first of a two phase project to install a new 2600 kW diesel generator as an alternate AC power supply for Millstone Unit No. 3. This first phase included the engineering, design, and construction activities necessary to install the diesel generator unit onto its foundation, complete with its auxiliary support equipment and enclosures. Phase 1 work did not include activities for integration of the diesel generator with the Millstone Unit No. 3 plant, as that was covered under Phase 2 of the project.

Reason for Change

Millstone Unit No. 3 chose to install an alternate AC power supply to comply with the requirements of the station blackout rule.

Safety Evaluation

This modification involved installation of a new non safety-related diesel generator unit, with essentially no connections made to plant systems during Phase 1. Since there was no interface with plant safety systems, the modification could not impact any equipment required for accident mitigation, nor could it create an accident.

Plant Design Change Number 3-92-096

This change, entitled "Phase II Alternate AC Diesel Generator Integration," is complete.

Description of Change

This change was the second of a two-phase project to install a new 2600 kW diesel generator as an Alternate AC (AAC) power supply for Unit No. 3. This phase of the change encompassed those tasks necessary to integrate the AAC output with the existing 4160 volt buses and to provide control of the diesel from the Control Room.

Reason for Change

Unit No. 3 chose to install an AAC power supply to comply with the requirements of the station blackout rule.

Safety Evaluation

The AAC source is only used to mitigate a station blackout event which is not part of the original design basis of the plant. The AAC source is not normally in use. The AAC source is only used to supply power to safety-relate components in the event that the other source of power cannot be used. Existing protective breaker trip functions allowed for safety testing of the AAC source after installation.

Plant Design Change Number 3-92-106

This change, entitled "Loose Parts Monitoring Upgrade," is complete.

Description of Change

This change replaced damaged Channel 10 in-containment cable and accelerometer. It also improved Loose Parts Monitor operations by establishing an independent signal common for all 12 channels and establishing a charge-amp single-ended configuration for all channels except Channel 10.

Reason for Change

Channel 10 was not operable because of cable degradation and required replacement. The other design changes reduced channel noise and improved the operation and reliability of the Loose Parts Monitor.

Safety Evaluation

This modification does not degrade system operation or affect design basis of the Loose Parts Monitor. The design change improves the Loose Parts Monitoring System so that Channel 10 will operate, and Channels 1 through 9, 11, and 12 will have improved performance and reliability.

Plant Design Change Number 3-92-119

This change, entitled "Removal of Flow Switch Interlock from Auxiliary Building Filter Fan Controls," is complete.

Description of Change

This change removed a flow switch from the "B" Auxiliary Building Filter fan circuitry. The flow switch served as a fan failure detection device to auto start the "B" train fan on a failure of the "A" train fan. The flow switch was disconnected from the circuit and abandoned in place.

Reason for Change

Under a certain failure scenario, the flow switch would not detect the fan failure. A plenum pressure failure detection scheme was installed to sense fan failure and auto start the "B" train fan.

Safety Evaluation

The change permanently disconnects an unreliable plant component based on replacement by a more reliable component. A failure of the new detection scheme does not result in unacceptable conditions. This change does not adversely affect the performance of the Auxiliary Building Filtration system.

Plant Design Change Number 3-92-120

This change, entitled "Vacuum Priming System Upgrade," is complete.

Description of Change

The existing carbon steel Vacuum Priming System piping was replaced with copper nickel piping and the piping was rerouted to include standpipe loops between the condenser Vacuum Priming domes and the station Vacuum Priming tank. Additionally, twelve level control valves were removed and butterfly isolation valves were installed. The piping at the Vacuum Priming pumps was modified to include dirt pockets and strainers.

Reason for Change

The piping changes were required to minimize corrosion and to provide more reliable automatic waterbox level control system. Additionally, the new piping eliminated the flow of Circulating Water to the Vacuum Priming pumps.

Safety Evaluation

The upgrade of the non safety-related Vacuum Priming System improved overall system operation.

Plant Design Change Number 3-92-124

This change, entitled "Amertap System Removal, Condenser Discharge Piping Upgrade, Condensate System Pump Down Line Piping Upgrade," is complete.

Description of Change

The change replaced the Amertap strainer spools with flanged in situ form-lined carbon steel spools. The change also removed and spared 18 Limitorque operators and removed associated conduit and cables. Additionally, all Amertap piping, valves, pumps, and instrumentation were removed. The condenser discharge piping was coated as required to repair degrading cladding. The change also repaired the condenser outlet valves and installed a permanent condensate system pumpdown line.

Reason for Change

The Amertap system was abandoned in place and the strainers were detericrating and damaging the outlet valves. The coating was performed to ensure cladding degradation and was stopped prior to through wall leaks developing. The Condensate System pump down line was added to replace a temporary line which was no longer serviceable.

Safety Evaluation

The removal of the Amertap system, the modifications to the Circulating Water system, and the installation of the Condensate system pump down line will not contribute to any previously evaluated accident or malfunction of equipment important to safety or its consequences. The changes have no affect on the safe shutdown capability of the plant. The changes meet all original design specifications and, as a result, are bounded by the original safety analysis.

Plant Design Change Number 3-92-129

This change, entitled "Replacement of Reactor Plant Component Cooling Water (RPCCW) Heat Exchanger Cross-Connect Valve," is complete.

Description of Change

This change replaced an existing 18" rubber lined butterfly cross connect isolation valve on the discharge of the third (swing) RPCCW heat exchanger. The new valve is unlined and features a new design consisting of a rubber seat ring attached to the disc.

Reason for Change

The original valve was very difficult to operate and could not be closed completely. It was suspected that the rubber lining had become detached from the valve body and was preventing the disc from closing. Also, the rubber lining which serves as the seating surface for the valve disc could not be repaired in the field. The new design will enable valve maintenance to be performed on-site.

Safety Evaluation

The isolation valve is a passive component which is not required to operate to accomplish a safe shutdown. Since it is not credited in any accident analysis, this change will not affect the probability or consequences of an accident or failure of equipment important to safety.

Plant Design Change Number 3-93-003

This change, entitled "Installation of Transfer Trip for Tripping Main Generator Output Breaker," is complete.

Description of Change

This change installed an audio tone tripping scheme which will trip the main electrical generator output breaker upon receipt of a trip signal from the Millstone Station electrical switchyard.

Reason for Change

This modification represents a portion of the special protection system designed to detect electrical disturbances within the vicinity of Millstone Station. The special protection system reduces the potential for Millstone Station electrical switchyard instability in the case of faults occurring in the vicinity of Millstone Station.

Safety Evaluation

This new trip function may cause the plant to trip earlier in response to a fault on the grid. However, the instability resulting from the fault would reach the station switchyard and cause a trip anyway.

Plant Design Change Number 3-93-006

This change, entitled "Revised Rod Control System Compensation Parameters," is complete.

Description of Change

This change revised lead and lag time constants within the Rod Control System.

Reason for Change

Millstone Unit No. 3 is one of several Westinghouse plants experiencing a flow anomaly in the upper plenum. This anomaly was leading to periodic fluctuations in hot leg temperatures and consequently, loop average temperature. Relatively small but rapid changes in loop average temperature were leading to spurious stepping of the control rods with the Rod Control System in automatic. The revised lead and lag compensation parameters allowed the Rod Control System to filter out the spurious fluctuations in loop average temperature.

Safety Evaluation

A failure in the Rod Control System could initiate an inadvertent rod or bank withdrawal event. However, the changes being made did not change the overall performance characteristics of the system or the fundamentals of how it responds to a change in one of its inputs. No new failure mechanisms are being introduced that would increase the probability of an inadvertent rod or bank withdrawal accident.

The Rod Control System is a control grade system not assumed to function to mitigate the consequences of any licensing or design basis accidents. Operation of the Rod Control System is assumed if it increases the severity of an event. The changes to the compensation parameters do not alter system performance to the point where the consequences of any licensing or design basis accidents are increased.

Plant Design Change Number 3-93-008

This change, entitled "Arcor Coating of the Inside of Service Water System Spools," is complete.

Description of Change

Arcor S-30 was applied to the entire internal diameter of several Service Water system spools.

Reason for Change

Arcor S-30 was applied to the pipe spools to increase the erosion and corrosion resistance of the piping. The coating serves as an inert layer of erosion and corrosion protection between the copper nickel, carbon steel pipe and the sea water.

Safety Evaluation

Past operating experience with Arcor has shown it tends to fall in small flakes (if at all). Since the Service Water system is very tolerant of small debris, there will be no effect on any downstream components. The changes have no effect on the safe shutdown capability of the plant. The changes meet all original design specifications and, as a result, are bounded by the original safety analysis.

Plant Design Change Number 3-93-009

This change, entitled "Service Water System Piping Modifications," is complete.

Description of Change

The Service Water piping in areas which had experienced wall thinning was modified. These areas include the following:

- suction and discharge piping of Control Building Air Conditioning Booster Pumps

- supply and return headers to Emergency Safety Features Building ventilation units.

- the return piping from the Control Building Air Conditioning Water Chillers

- install a flange pair in the Service Water Access Enclosure

- modify Service Water system supports to accommodate the piping changes

Reason for Change

The piping changes reduced flow turbulence which has been identified as a cause of wall thinning and through wall flaws. The change of material to Monel will prolong piping service life. The installation of flanges will enable easier rigging and access in future piping inspections.

Safety Evaluation

The changes have no effect on the safe shutdown capability of the plant. The changes meet all original design specifications and as a result are bounded by the original safety analysis.

Plant Design Change Record Number 3-93-011

This change, entitled "Modification to Generator Line Protection Pilot Wire Scheme," is complete.

Description of Change

This change allowed switchyard breakers associated with the generator protection pilot wire scheme to open more quickly upon the actuation of two input relays.

Reason for Change

The previous design could result in instability on the 345 KV transmission system in response to a fault in the main step up transformer.

Safety Evaluation

This change will decrease the plants response time to a fault in the main step up transformer. However, the overall response to this event remains unchanged.

Plant Design Change Number 3-93-012

This change, entitled "Technical Support Center Damper Modifications," is complete.

Description of Change

The change permanently locked open and removed the associated actuators, cabling and controls from two dampers in the Technical Support Center Ventilation System. These dampers were motor operated dampers which were previously disabled to the open position during system start-up testing to satisfy the design criteria of the system. This change made that temporary modification permanent.

Reason for Change

It was discovered during start-up testing for the Technical Support Center Ventilation System that two motor operated dampers could possibly fail to the closed position and shut the suction path of the air conditioning unit. This would cause equipment damage. Testing confirmed that the design flow rates could be met by locking these dampers to the open position. This change permanently locks these dampers to the open position.

Safety Evaluation

The safety evaluation stated that the Technical Support Center Ventilation System is not a safety-related system and that it cannot adversely affect any other safety-related systems. Therefore, the damper modifications can not affect the plant's safety-related systems.

Plant Design Change Number 3-93-013

This change, entitled "Feedwater Flow Control Valves Position Indication," is complete.

Description of Change

This modification permanently incorporated a bypass jumper which provided position indication for the feedwater flow control valves. Permanent incorporation required enhancing the existing design, installing a recorder on the back of Main Board 5, and providing input to the plant process computer.

Reason for Change

The reasons were threefold: to provide input for trouble shooting the feedwater flow control valves, to provide a historical record of valve position, and to support the plant's commitment to reduce the number of bypass jumpers installed in the field.

Safety Evaluation

The position indication for the feedwater flow control valves is not safety related and is not required to mitigate the consequences of an accident. This modification does not degrade the performance of equipment important to safety and does not introduce new malfunctions of equipment important to safety.

Plant Design Change Number 3-93-015

This change, entitled "Replacement of Recirculation Spray Pump Suction Valves," is complete.

Description of Change

This change replaced the suction valve for each of the four pumps in the Containment Recirculation Spray system. The new valves have an improved seat design.

Reason for Change

The original seats had become detached. This resulted in the valves not closing properly and failing local leak rate testing.

Safety Evaluation

The new valves reduce the probability of the valves failing to close completely. The new valves meet the original design criteria. The valves are open during accident conditions.

Plant Design Change Number 3-93-016

This change, entitled "Replacement of Snubbers Due to Functional Failures," is complete.

Description of Change

This change replaced small model snubbers that were identified as functional failures during past refueling outages. Larger model snubbers were installed in place of smaller model snubbers to eliminate the high failure rate. Also, snubber supports were modified to allow for the larger style snubbers.

Reason for Change

This change was performed to alleviate the recurring functional failures of small model snubbers and to prevent future failures. The past failures were a result of unanticipated loading, such as external torsion, side loading, and impacts caused by mishandling. The small model snubbers were replaced with larger model snubbers to increase their reliability. The larger model snubbers are inherently more rugged.

Safety Evaluation

The safety evaluation addressed the affect of replacing snubbers on the Reactor Coolant System, Low Pressure Safety Injection System, High Pressure Safety Injection System, Containment Recirculation System, and the Steam Generator Blowdown System.

The purpose of the snubbers is to allow for thermal growth of the piping during normal operation and to control rapid dynamic response due to fluid transients of seismically induced loadings during accident conditions. Replacing the existing snubbers with more durable ones decreases the likelihood of snubber failure which could result in piping failure.

Plant Design Change Number 3-93-023

This change, entitled "Heated Junction Thermocouple Probe and Cable Replacement," is complete.

Description of Change

The heated junction thermocouple probe and associated reactor head cable in the "B" train Reactor Vessel Level Indicating System (RVLIS) were replaced with a new design liquid level probe and transition cable.

Reason for Change

The "B" train heated junction thermocouple probe had two failed heaters, one in the upper head region and one in the upper plenum region, resulting in non-conservative inaccuracies in the indicated reactor vessel upper head and upper plenum levels. The heaters could not be repaired.

The heated junction thermocouple probe was replaced with a new design liquid level probe in order to improve the reliability of the "B" train RVLIS.

Safety Evaluation

The safety evaluation addressed the fact that the replacement probe was functionally identical to the original probe.

The only malfunction which required evaluation is a failure of the probe or cable connector, which would render the entire probe inoperable. The plant Technical Specifications allow continued plant operation in the event of the failure of a probe, if repairs are not feasible, with increased emphasis on the use of core exit temperature monitoring during a loss of coolant accident.

Plant Design Change Number 3-93-024

This change, entitled "Core Exit Thermocouple Connector and Cable Replacement," is complete.

Description of Change

The reactor head cable connectors and cables for 44 of 50 core exit thermocouples were removed and replaced with new design connectors and upgraded cables.

Reason for Change

This modification improved the reliability of core exit temperature indication and greatly reduced the time and exposure needed to disconnect and reconnect the thermocouples when refueling.

During the previous cycle, 13 of 50 core exit thermocouples were inoperable due to degradation of the thermocouple and/or connector. Because the connectors could not be repaired, new upgraded connectors were installed.

Safety Evaluation

The safety evaluation addressed the failure of one complete 14 pin connector and thus six or seven core exit thermocouples in one train.

Plant Technical Specifications require that a minimum of two core exit thermocouples per train per core quadrant be operable to provide indication of core exit temperature and adequate core cooling. The failure of one connector would not violate this requirement. The core exit temperature monitoring system has no control or actuation functions and provides indication only.

Plant Design Change Number 3-93-027

This change, entitled "Reactor Coolant Pump (RCP) No. 1 Seal Leakoff High Flow Alarm Setpoint Change and Rescaling of All No. 1 Seal High Range Leakoff Flow Transmitters," is complete.

Description of Change

The change was initiated in two parts.

1. The high alarm setpoint for the "B" RCP No. 1 seal leakoff was temporarily raised from 5.7 gallons per minute (gpm) to 7.2 gpm.

2. A review of the calibration data for the "B" RCP No. 1 seal leakoff flow indication revealed that the scaling needed to be adjusted to obtain the design accuracy for the flow element. The flow transmitters for all RCPs are of the same design, so an adjustment was made for all transmitters.

Reason for Change

The "B" RCP No. 1 seal leakoff was increasing slowly. In order to continue operating the pump for the four months remaining until the scheduled refueling outage shutdown, the alarm limit was raised to a value below the maximum value allowed by the RCP manufacturer. During the shutdown, the No. 1 seal was replaced and the alarm was restored to its design value.

The rescaling was necessary in order to obtain the design accuracy for the flow elements installed in the seal injection lines.

Safety Evaluation

The increase in alarm point was done with the concurrence of the RCP manufacturer. The limits on No. 1 seal leakage are based, in part, on the 8 gpm capacity of the RCP thermal barrier heat exchanger. At 0.8 gpm or less, a loss of normal seal injection will not cause a loss of cooling capability for the thermal barrier heat exchanger. The 0.8 gpm difference between the alarm point and the maximum allowed 8 gpm includes margin for instrument error and some time for the operators to respond to the alarm in a reasonable manner so the 8 gpm limit is not exceeded while the plant is at power.

The rescaling of the flow transmitters was performed based on verified calculations, and with concurrence from the RCP manufacturer.

Plant Design Change Number 3-93-034

This change, entitled "Permanent Reactor Cavity Seal Installation," is complete.

Description of Change

The change replaced a temporary reactor cavity pit seal with a permanent seal which was welded in place. Access to the nuclear instrumentation is provided by eight access holes. During plant operation, the holes are open to allow ventilation; chimney covers are in place to prevent direct spray or moisture from entering the cavity. Covers are installed in the access holes and air tested for a proper seal prior to flooding up the reactor cavity. When the cavity is drained prior to returning to power, the covers are replaced with the chimney covers.

Reason for Change

The temporary seal required critical path polar crane time to install and remove during refueling outages. It also leaked significantly during floodup and draindown whenever water level was within about six feet of the top of the temporary cover.

Safety Evaluation

The flexible membrane is designed in accordance with industry code and is analytically qualified for a bounding number of normal operating and faulted load cycles. All work, including welding and weld inspections, was performed using approved procedures. All welding and subsequent inspection techniques were reviewed to ensure structural integrity of the installation. The ventilation was checked before and after installation of the permanent pit seal, with chimney covers installed, in order to ensure proper ventilation was maintained.

During a safe shutdown earthquake, the flexible membrane accommodates relative seismic motion of the reactor vessel. Horizontal motion is limited by the reactor vessel supports and neutron shield tank. Installation of the permanent cavity seal does not impact any barriers which affect public health and safety.

Plant Design Change Number 3-93-037

This change, entitled "Arcor Coating of the Inside Diameter of Service Water System Spools," is complete.

Description of Change

Arcor S-30 was applied to the internal diameter of several Service Water system spools.

Reason for Change

Arcor S-30 was applied to the pipe spools to increase the erosion/corrosion resistance of the piping. The coating serves as an inert layer of erosion/corrosion protection between the copper nickel, carbon steel pipe, and the sea water.

Safety Evaluation

Past operating experience with Arcor has shown that it tends to fail in small flakes (if at all). Since the Service Water system is very tolerant of small debris, there will be no effect on any downstream components. The changes have no effect on the safe shutdown capability of the plant. The changes meet all original design specifications and, as a result, are bounded by the original safety analysis.

Plant Design Change Number 3-93-044

This change, entitled "Make Duct Support in Main Steam Valve Building Removable," is complete.

Description of Change

This change modified a duct support in the area of the "A" Main Steam Isolation Valve (MSIV) to make it removable by replacing the welded attachment to the I-Beam with bolting hardware.

Reason for Change

This change will facilitate maintenance on the "A" MSIV. The surrounding duct work was welded in place making removal and reinstallation time consuming.

Safety Evaluation

The duct work is not safety related. The new fastening method is equivalent to the previous welded attachment and meets the original design criteria.

Plant Design Change Number 3-93-050

This change, entitled "Auxiliary Feedwater Pump Design Change," is complete.

Description of Change

This change replaced the rotating elements of the "A" Auxiliary Feedwater pump. The new rotating element has impellers with an integral chrome plated wear surface and shaft sleeves made from a material not susceptible to intergrannular stress corrosion cracking.

Reason for Change

The wear rings and shrink fit center and throttle shaft sleeves were susceptible to intergrannular stress corrosion cracking.

Safety Evaluation

This change reduced the likelihood of a failure of the "A" Auxiliary Feedwater pump.

Plant Design Change Number 3-93-054

This change, entitled "Reactor Flange Shield Installation," is complete.

Description of Change

The change installed a cylindrical shield inside the reactor vessel internals lift rig. During plant operation, the shield rests on the upper internals storage stand along with the lift rig. As the lift rig is raised, placed over the reactor vessel, and then lowered, the shield rests on the lift rig torus. As the lift rig comes to rest over the vessel, the shield is supported on the lip of the upper internals.

Reason for Change

Radiation levels near the reactor vessel at the bottom of the refueling cavity are extremely high. The shield significantly reduces the radiation exposure to personnel required to work near the vessel during refueling. Also, the cylindrical shield provides an excellent support for additional lead blankets which further reduce the radiation levels.

Safety Evaluation

An analysis determined that the addition of a 39,000 lb. shield to the lift rig did not exceed allowable load limits. The addition of the flange shield does introduce a new load path in the lifting rig. When lifting the rig and the flange shield, the load path is through the torus, a support plate, bolts, and dowel pins. Except for the dowel pins, the lift rig with shield is in conformance with the design criteria. The dowel pins were replaced with higher strength material in order to satisfy the loading criteria.

Calculations were performed to assure the additional weight could be supported by the upper internals storage stand, the lower internals storage stand, and construction laydown areas on which the lift rig and flange shield would be placed. In addition, the additional load was reviewed for adverse affects on postulated heavy load drop accidents and found to be within acceptable criteria.

Plant Design Change Number 3-93-060

This change, entitled "Installation of Forced Air Cooling Fans Within the 7300 System," is complete.

Description of Change

Forced air cooling was installed in the 7300 process control cabinets.

Reason for Change

The 7300 process control cabinets were experiencing heat build-up due to normal operations. The forced air cooling was added to the cabinets in an effort to facilitate cooling flow within the cabinets.

Reducing the heat build-up within the cabinets would significantly reduce the effects of heat as a factor in the accelerated aging of the electrical components on the system cards.

Safety Evaluation

The cooling fans were installed within the 7300 cabinets in such a manner that they would no impact the electrical power supplies to the cabinets should a fan unit failure occur.

Increased cooling of the cabinets reduces the likelihood of component failure.

Plant Design Change Number 3-93-061

This change, entitled "Revising Motor Operated Valve's Motor Protection," is complete.

Description of Change

The magnetic trip coil setting and/or thermal overload heater size in 159 motor operated valves were changed.

Reason for Change

This change was instituted to comply with industry standards and regulatory guidance.

Safety Evaluation

The changes to each motor optimized motor protection while minimizing nuisance tripping. There were no changes to motor control circuitry so accident response and the probability of an accident are unaffected.

Plant Design Change Number 3-93-062

This change, entitled "Replacement of Turbine Driven Auxiliary Feedwater (TDAFW) Pump Valve Room Air Conditioning System," is complete.

Description of Change

The change replaced the existing non-safety-related two ton air conditioning unit in the TDAFW pump valve room with an upgraded three ton unit.

Reason for Change

The original two ton air conditioning unit failed to maintain the design temperature in the TDAFW pump valve room. Operator action was required during the warmer times of the year to prevent the space temperature from exceeding the Electrical Environmental Qualification (EEQ) limits of the equipment in the room. The new three ton unit provides increased cooling capacity to maintain temperatures below the EEQ limits.

Safety Evaluation

This modification upgraded the capacity of the TDAFW pump air conditioning unit from two to three tons. The upgrade of this AC unit allows the system to maintain the space temperature below the EEQ limits of the equipment in the room. The air conditioning system is not a safety-related system and this change will only improve the conditions within the space where safety-related equipment is located.

Plant Design Change Number 3-93-064

This change, entitled "Rewire Relays in the Engineered Safety Features Actuation System (ESFAS)," is complete.

Description of Change

This change rewired the ESFAS circuitry to shift the following loads from a relay which actuates on a containment isolation phase "B" (CIB) signal to a relay which actuates on a containment depressurization (CDA) signal:

- six Auxiliary Feedwater System valves

- a 4.16 KV Auxiliary Circuit

In addition, an alarm on one relay which actuates on a CIB was transferred to a different relay which actuates on the same signal.

Reason for Change

The change was made to resolve a known testing deficiency and to resolve a discrepancy between various design documents.

Safety Evaluation

Rewiring the CIB relays to CDA relays conform to the original plant design. Because a CDA signal will also initiate a CIB, the shifting of loads from one type of contact to another did not alter the plant response to any accident conditions.

Rewiring the alarm from one relay to a similar relay had no affect on plant operation.

Plant Design Change Number 3-93-067

This change, entitled "Auxiliary Building Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Area Ventilation Heaters," is complete.

Description of Change

This modification installed a permanent, dual train, safety-related heating system in the CH/RPCCW area. Eight Category I electric unit heaters with associated thermostats and supply breakers were installed and supplied power from Emergency 480 VAC Motor Control Centers.

Reason for Change

All temperature control capabilities for the CH/RPCCW area had been removed previously. Consequently, eight non-Category I heaters (four per train), fed from emergency load centers through isolation breakers to 480 VAC distribution panels, were installed in the CH/RPCCW area under a bypass jumper to meet the design temperature limits of the area. This modification replaced the temporary non-Category I heating system with a permanent safety-related heating system.

Safety Evaluation

The heaters are powered from Emergency Control Centers that receive power from the Emergency Diesel Generators during a Loss of Power Accident. The Emergency Diesel Generators were verified to be capable of carrying the increased load.

The eight heaters are sufficient to maintain 65°F in the area with an outside temperature of 0°F and the Auxiliary Building Ventilation System in the winter mode alignment.

The change was installed to meet the spatial separation requirements between redundant safety-related electrical trains as well as between safety-related and non safety-related electrical trains. The electrical cables and loading were analyzed and are within the capacity of the electrical distribution system. Additionally, the heaters, Motor Control Center retrofit breakers/buckets are seismically qualified and mounted. The associated cabling is run via seismically qualified raceways.

Plant Design Change Number 3-93-074

This change, entitled "Unit 3 Ecolochem Building Installation," is complete.

Description of Change

A permanent pre-engineered metal building was installed adjacent to the Unit No. 3 Turbine Building. Fire protection was provided via a connection to the Unit No. 3 fire main.

Reason for Change

Provide a permanent structure for the vendor's water treatment equipment.

Safety Evaluation

The installation of the permanent building is consistent with the original plant design. No safety-related equipment is contained in the structure.

Plant Design Change Number 3-93-076

This change, entitled "Modification to SIGMA Refueling Machine Control Console," is complete.

Description of Change

This change added a computer reset switch and single point low impedance ground path to the SIGMA Refueling Machine control console.

Reason for Change

Prior to this change, the only way to reset the computer in the control console was to de-energize the entire console. This put unnecessary wear and tear on the electronic components in the console. The lack of a single point low impedance ground in the console contributed to noise in the control system, which in turn led to unreliable operation. Installation of a reset switch and low impedance ground will lead to improved reliability of the SIGMA refueling machine.

Safety Evaluation

Installation of the reset switch and ground does not affect or alter the operation of the SIGMA refueling machine, including its ability to positively latch and hold a fuel assembly. Failure of the reset switch or ground would also not affect operation of SIGMA.

The consequences of the design basis fuel handling accident are not increased, since the SIGMA Refueling Machine only handles one fuel assembly at a time.

Plant Design Change Number 3-93-082

This change, entitled "Installation of Rockbestos Cable to Replace Thermo-Lag Fire Barrier," has been completed.

Description of Change

A fire rated cable in a conduit was installed in place of a non-fire rated cable housed in an externally coated fire barrier conduit.

Reason for Change

The externally applied fire barrier material was deemed indeterminate in its ability to retard fire.

Safety Evaluation

This change improved the fire resistance of the cable.

Plant Design Change Number 3-93-090

This change, entitled "Reload Design for Millstone Unit No. 3 Cycle 5," is complete.

Description of Change

This change addressed the reloading of the reactor core with fresh and previously burned fuel into a configuration which could be licensed to allow operation for Cycle 5. The reload consisted of 84 fresh VANTAGE 5H fuel assemblies, and 109 previously burned VANTAGE 5 and 5H fuel assemblies. Feed assemblies incorporated the following design enhancements:

(i) Removal of the cusp and keyhole in the top nozzle assembly.
(ii) Shorter guide tubes, instrumentation tubes, and fuel rods.
(iii) ZIRLO fuel rod cladding, guide tubes, and instrument tubes, and ZIRLO mid-spare and IFM-grids on 36 fuel assemblies.
(iv) Shorter plenum springs.
(v) Annular axial blankets.
(vi) IFBA coated fuel pellets containing increased boron loading in the diboride coating.

In addition, all remaining Hafnium control rods were replaced with chrome plated, enhanced performance control rods, and thimble plugs were reinserted. The nuclear design was not substantially changed from Cycle 4.

Reason for Change

Cycle 4 had reached the end of its design operating life, requiring refueling. The changes in the mechanical design of the fuel assemblies were made for the following reasons:

(i) Prevent top nozzle spring hang-up on the nozzle cusp during operation.
(ii) Allow for increased fuel assembly growth during longer fuel cycles.
(iii) ZIRLO is less susceptible to corrosion, providing additional margin to allow for longer cycle length.
(iv) Shorter springs are required to account for the shorter fuel rods.
(v) Reduce neutron leakage and provide additional void volume to accommodate increased fission gas release as a result of longer cycles.
(vi) Provides for power peaking and Moderator Temperature Coefficient control.

Hafnium control rods were replaced due to a problem with hafnium hydriding leading to bulges in the rodlets, as well as cracks at the end plugs. Thimble plugs were reinserted to increase Reactor Coolant system flow through the hot channels in an attempt to reduce subcooled boiling on the fuel rods.

Plant Design Change Number 3-93-090 (Continued)

Safety Evaluation

Design basis accidents are typically by events outside the reactor core. The changes in the mechanical design of the fuel assemblies does not alter or affect the interface between the fuel assemblies and the fuel handling equipment. Therefore, the probability of occurrence of previously evaluated accidents is not increased.

The changes in the mechanical design of the fuel does not degrade its ability to withstand design and licensing basis accidents, nor does it degrade fuel performance during normal operation. Since the response of the fuel to design and licensing basis accidents is not affected, the consequences of previously evaluated accidents is not affected. Since the structural characteristics of the assembly is not changed, the probability of occurrence of a previously evaluated malfunction (cladding breach) of equipment important to safety (fuel assembly) is not increased.

The nuclear design of Cycle 5 is essentially identical to Cycle 4. Given this, and the fact the changes in mechanical design do not affect fuel assembly performance during normal or accident conditions, the probability of a new accident or malfunction is not created.

Plant Design Change Number 3-93-092

This change, entitled "Lube Oil Reservoir Tank Indication and Alarm," is complete.

Description of Change

This change deleted an existing lube oil level switch, added alarm capabilities and computer points to a more reliable level transmitter and added a local sight glass.

Reason for Change

The old lube oil reservoir level instrumentation was unreliable.

Safety Evaluation

The safety evaluation evaluated the possibility of a loss of the turbine lube oil level transmitter during the implementation of this change. The likelihood of a loss of the turbine lube oil level transmitter while the turbine was off-line and the lube oil reservoir was drained was considered insignificant, with no impact to equipment important to nuclear safety.

Plant Design Change Number 3-93-093

This change, entitled "Boron Concentration Measurement System Removal," is complete.

Description of Change

The boron concentration measurement system was abandoned in place by permanently closing the inlet and outlet isolation valves. In addition, accessible support equipment and cables were removed or de-terminated and spared in place.

Reason for Change

The installed boron concentration measurement system was extremely unreliable. As a result, chemical analysis is used to determine the concentration of the boron 10 isotope rather than the installed system. Maintenance of the system requires unnecessary manpower and radiation exposure. Since chemical analysis is sufficient, it is not cost effective or radiologically prudent to install a new, more reliable system.

Safety Evaluation

The system was not part of a control element or control system, nor is it designed for this use. Chemical analysis is used to determine the relative concentration of the boron 10 isotope.

Plant Design Change Number 3-93-097

This change, entitled "Addition of Isolation Valves to Containment Hatch for Local Leak Rate Testing (LLRT) Tube Connection," is complete.

Description of Change

This change modified the LLRT system configuration of the containment personnel access hatch seals. The change removed a permanent testing system consisting of a pneumatic module, three solenoid valves, a remote control panel and associated tubing. The pneumatic unit was replaced with a portable test configuration. The new configuration consisted of tubing, two new valves and caps.

Reason for Change

The original test configuration was unreliable. The leak test was routinely performed by disconnecting the permanent unit and connecting the portable unit. The connecting and disconnecting of the portable equipment created a potential containment breach. Implementation of this change eliminated the potential containment breach.

Safety Evaluation

The new test system had been demonstrated through prior use to be more reliable and to meet containment integrity requirements.

Plant Design Change Number 3-93-106

This change, entitled "Rerouting of the Circuit Associated with a Reactor Plant Ventilation Temperature Switch to Comply with Separation Requirements," is complete.

Description of Change

This change relocated cabling which controls a heating and ventilation unit. The operation or performance of the unit was not impacted by this change.

Reason for Change

A fire in the location where the cabling had resided had the potential to affect redundant components.

Safety Evaluation

This change ensured that a single fire would not incapacitate redundant components.

Plant Design Change Number 3-93-109

This change, entitled "Motor Driven Auxiliary Feedwater Pump Trip Circuit Modification," is complete.

Description of Change

The change removed the low suction pressure trip for both of the Motor Driven Auxiliary Feedwater pumps, provided a low suction pressure trip alarm in the Control Room for each pump, and installed a surge suppressor in the sensing lines for the low suction pressure switches. This made a bypass jumper a permanent installation.

Reason for Change

The pump manufacturer, architect engineer, and Nuclear Steam System supplier recommended that no low suction pressure trip should be installed for the safety-related Auxiliary Feedwater pumps. Spurious trips could make a pump fail at the moment when it is required to feed a steam generator. An alarm replaced the trip function in order to alert the operator. Surge suppressors were installed in order to dampen the pressure spikes that were observed on the switches during previous testing.

Safety Evaluation

This change removed a possible failure mode which could cause an inadvertent pump trip. System design and line-up prevent an actual low suction pressure from occurring. Therefore, there was no need to have this trip function.

Plant Design Change Number 3-93-116

This change, entitled "Abandonment of Pressurizer Liquid Sample Line," is complete.

Description of Change

This change abandoned the pressurizer liquid sample line and associated containment isolation valves. Abandonment consists of:

- De-termination of electric power, control and indication for containment isolation valves, and removal of control switches and indication.

- Cut and cap line inside and outside of containment and modify support.

Reason for Change

The pressurizer liquid sample line mixes with condensed steam from pressurizer relief valve loop seal drains. This mixing dilutes the pressurizer sample and results in unrepresentative chemistry, therefore, the pressurizer liquid sample line is no longer used.

Safety Evaluation

The safety evaluation looked at the effects on containment integrity, potential for Reactor Coolant system (RCS) leak via sample piping and the ability to ensure that boron concentration differences between the Pressurizer and RCS are acceptable as required by operating procedures. The capped piping exceeds design requirements for qualification as containment pressure boundary and was evaluated as structurally and seismically acceptable.

A reactivity addition accident due to boron concentration differences between the RCS and the pressurizer is prevented by inducing pressurizer spray to equalize boron concentration whenever RCS boron concentration changes by 50 ppm.

Plant Design Change Number 3-93-120

This change, entitled "Installation of Rigging Support for Auxiliary Feedwater Pump," is complete.

Description of Change

This change installed rigging to allow the "A" Auxiliary Feedwater System pump to be modified as described in Plant Design Change Number 3-93-050. The rigging installation required modifying the supports for a lube oil pressure switch.

Reason for Change

The rigging was necessary to remove the shaft of the "A" Auxiliary Feedwater Pump. The supports for a lube oil switch were interfering with the rigging.

Safety Evaluation

The operation of the pressure switch was not impacted by the modifications of the supports.

Plant Design Change Number 3-93-121

This change, entitled "Addition of Reactor Trip Interlock to Inadequate Core Cooling Monitor Annunciation," is complete.

Description of Change

Inadequate Core Cooling Monitor annunciator outputs for Saturation Trouble Train A(B) and Core Exit Temperature High Train A(B) have been interlocked with the status of the reactor trip breakers such that these annunciator points are only enabled post-trip.

Reason for Change

The annunciators have been continuously locked in during normal plant operation due to a new fuel design and had become a nuisance to the operators. They are primarily intended for use in accident scenarios that can lead to inadequate core cooling, where a reactor trip will have occurred.

Safety Evaluation

The safety evaluation addressed the failure of the subject annunciator points to function when required, or to function inadvertently.

The Inadequate Core Cooling Monitoring system information is available from safety related indicators and from the Safety Parameter Display system. The modification affects annunciator points and auxiliary relays which have no control, actuation, or safety-related functions, and will not result in a challenge to protective boundaries.

Plant Design Change Number 3-93-124

This change, entitled "Engineered Safety Features (ESF) Building Ventilation Supply Fan Flow Switch Setpoint Revision," is complete.

Description of Change

The design change revised the setpoint for the flow switches of the ESF Building Emergency Ventilation System supply fans from 500 fpm to 60 fpm. It also extended the time delay on the follow-up fan starting from 15 seconds to 40 seconds.

Reason for Change

The purpose of the flow switches is to confirm operation of the lead fan and provide a signal to energize the redundant follow-up fan in case of lead fan failure. The original setpoint of 500 fpm was arbitrarily selected as sufficiently low to indicate failure of the fan. Normal velocity is approximately 1800 fpm. The instrument location in the corner of the duct and in close vicinity of the axial fan suction, caused the instrument to provide low readings as compared to the actual tested flow rates and caused fan cycling. The lower setpoint will adequately support the system design function and will prevent unnecessary fan cycling.

System tests also indicated that it takes longer than 15 seconds (approximately 40 to 50 seconds) for the air flow to stabilize, and therefore, for the instruments to provide correct readings. The time delay was extended to 40 seconds to assure that stable air flow conditions exist prior to the system checking for proper fan operation.

Safety Evaluation

The ESF Building Emergency Ventilation System function is to provide a suitable environment for personnel and equipment operation and to prevent or minimize the spread or release of airborne radioactive material to the atmosphere. The changes to the setpoints result in the system being more reliable and do not introduce any new accidents.

Plant Design Change Number 3-93-125

This change, entitled "Replacement of the 'A' Quench Spray Pump Anchor Bolts," is complete.

Description of Change

The existing anchor bolts on the Quench Spray Pumps were replaced with anchor bolts made from material which is not susceptible to stress corrosion cracking. The new anchor bolts are bar stock, threaded at each end, and covered with RayChem heat shrink tubing.

Reason for Change

To preclude stress corrosion cracking as a common mode failure of the anchor bolts. One bolt had failed and the analysis indicated stress corrosion cracking was the failure mechanism.

Safety Evaluation

The seismic and structural integrity of the pump and its foundation remained unchanged, so the pump remained operable to depressurize containment following an accident. There are no changes or impacts to the normal or emergency operation, controls, or safety/protection features associated with the Quench Spray pumps, other Quench Spray equipment, or overall Quench Spray system operation.

Plant Design Change Number 3-93-126

This change, entitled "Charging and Reactor Coolant System Valve Yoke Bolts Replacement," is complete.

Description of Change

Valve bonnet to yoke bolts were replaced on the Charging Flow Control Isolation Valve, the Charging Flow Controller Isolation Valve, and both Pressurizer Power Operated Relief Valve (PORV) Isolation Valves. The original material was replaced with a stronger material. In addition, the original nuts were replaced. Washers under bolt heads and nuts were added to aid torquing and to minimize embedding due to increased torquing.

Reason for Change

Replacement of the bolts and nuts was required in order to increase the maximum allowable seismic valve limit and increase the valves thrust capacity. The bonnet to yoke bolt to nut connection was the most limiting component in the valve and valve motor assembly. The new bolts are of a higher tensile and yield strength than previously installed.

Safety Evaluation

The safety evaluation addressed the operation of the subject valves with new bonnet to yoke bolts. The replacement of valve bonnet to yoke bolt and nuts on the Charging Flow Control Isolation Valve, the Charging Flow Controller Isolation Valve, and both Pressurizer PORV Isolation Valves increased the seismic and thrust capacity of the valves.

Plant Design Change Number 3-93-128

This change, entitled "Modify Supports for 'B' Safety Injection Pump Miniflow Isolation Valve to be Removable," is complete.

Description of Change

The change modified gang hanger support for the "B" Safety Injection Pump Miniflow Isolation Valve and associated nearby pipe to be removable by adding splice plates and a pipe strap to the hanger.

Reason for Change

Repair or replacement of the "B" Safety Injection Pump Miniflow Valve was extremely difficult due to the configuration of the original hanger support. This change made the hanger portable to allow for easier repair and replacement of the valve.

Safety Evaluation

This safety evaluation addressed the addition of splice plates and a bolted pipe strap to the hanger for the "B" Safety Injection Pump Miniflow Valve and associated piping. The new hanger configuration was analyzed and found to be seismically qualified. The support characteristics of the support remained unchanged.

Plant Design Change Number 3-93-129

This change, entitled "Replacement of Service Water Inlet Isolation Valve to Reactor Plant Component Cooling Water Heat Exchanger," is complete.

Description of Change

The Service Water supply isolation valve to the "C" Reactor Plant Component Cooling Water heat exchanger was replaced with a new valve with an improved seat design. The existing butterfly valve had a rubber-lined carbon steel body. The new valve has a stainless steel body and a disc mounted resilient seat.

Reason for Change

Operations had reported that the original valve was leaking by its seat. The new valve is an improved design furnished by the original manufacturer.

Safety Evaluation

The new valve is very similar in design to the original valve and meets design specifications for the Service Water system. The form, fit and function of the valve are essentially unchanged and all previous accident analyses apply.

The changes have no affect on the safe shutdown capability of the plant. The changes meet all original design specifications and as a result are bounded by the original safety analysis.

Plant Design Change Number 3-93-131

This change, entitled "Modification to Reactor Head Vent Isolation Valves," is complete.

Description of Change

This change fabricated two identical valve spool pieces containing Target Rock isolation valves and block valves. Four manual globe valves were installed and removable hangers replaced the existing welded supports.

Reason for Change

The Target Rock Head Vent Isolation Valves have had a history of high maintenance activity due to leak-by. The installation of manual valves provide for isolation capabilities of the Reactor Head Vent Line upstream of the Target Rock Isolation Valves. This in turn will allow for changeout of the existing Target Rock Valve spools at 100 percent power. The installation of flanged supports minimizes personnel time in a radiological area. Spool with identical dimensions allows the "A" and "B" train to be replaced with a spare spool.

Safety Evaluation

This change meets all the original design criteria for the Reactor Head Vent line. Dead weight, thermal, system transient loads and seismic analyses were performed for the modified system by Westinghouse Electric Corporation.

Plant Design Change Number 3-93-133

This change, entitled "Installation of Finer Cartridges for Letdown Reactor Coolant and Seal Water Injection Filters," is complete.

Description of Change

The modification changed the size of the Letdown Reactor Coolant and Seal Injection Filters as follows:

Letdown	6 microns or larger
Reactor Coolant	0.2 microns to 25 microns or 6/40
Seal Injection	0.2 microns to 5 microns

Filter sizes are based upon the results of IST 3-93-003 and IST 3-93-012. The filter material did not change.

Reason for Change

Changes in micron size of the cartridges are recommended to maintain the Reactor Coolant System as clean as possible, to reduce the particle loading on the Seal Water Injection filters, and to prevent smaller particles from plating out on the reactor coolant pump seals.

Safety Evaluation

Changing the filter cartridge size of these filters will not affect any design basis accident or its consequences. This change was evaluated for its impact on the charging system operation and on Reactor Coolant Pump seal water injection.

Plant Design Change Number 3-93-135

This change, entitled "Modifications to Diesel Jacket Water Expansion Line," is complete.

Description of Change

This change added a flange in the "A" Emergency Diesel Generator (EDG) jacket water expansion line.

Reason for Change

Maintenance on the EDG cylinders requires that the expansion line be disassembled. This change made the removal of this line easier.

Safety Evaluation

The addition of the flange had no impact on the seismic qualification of the EDG. The performance of the jacket water cooling system is not affected by the addition of the flange.

Plant Design Change Number 3-93-136

This change, entitled "Jib Crane in the Reactor Head Storage Area," is complete.

Description of Change

A jib crane was installed in the reactor head storage area of the containment. A temporary bracket to secure the crane is installed under jumper 3-93-167.

Reason for Change

The crane was installed to enable repair of the Control Rod Drive Mechanism canopy seals.

Safety Evaluation

The crane was seismically analyzed and determined to be satisfactory. Additional administrative controls ensure that loads will not be carried above safety-related piping. Therefore, no new malfunctions or accidents were created.

Plant Design Change Number 3-93-137

This change, entitled "Canopy Seal Repair of Spare Capped Nozzles K-4 and E-9," is complete.

Description of Change

The design change installed Canopy Seal Clamp Assemblies (CSCA) on two reactor head spare capped nozzles, at the K-4 and E-9 locations. Each spare nozzle is capped with a threaded cap which is then seal welded. The two caps at the K-4 and E-9 locations were found to have very small leaks as identified by boric acid deposits on adjacent nozzles. Each CSCA includes split Graphoil seals, split seal carrier, housing, top plate cap screws, belleville washers and dummy can spacer adapter. The CSCA seals using standard packing and clamping action does not involve any welding.

Reason for Change

Spare capped nozzles E-9 and K-4 had been identified to exhibit small leaks. These leaks would only grow in magnitude and therefore must be stopped. This is a generic Westinghouse Reactor problem. The CSCAs installed have been used at a number of other facilities and is a standard fix for this type of leak. The fix provides an effective seal which can be installed in a short period of time and is very "As Low As Reasonably Achievable" prudent. The fix is also advantageous in that it changes the stress distribution in the seal weld and therefore helps minimize future crack growth.

Safety Evaluation

The safety evaluation evaluated the effects of the additional weight of the clamp assembly on the nozzle loadings. No other aspects were considered because the change does not change or affect the pressure boundary of the Reactor Coolant System. No malfunctions were evaluated as even with the additional weight from the CSCAs the nozzle loads remain within acceptable limits. The CSCA is a passive device with no active function.

The change resulted in the system being more reliable and did not introduce any new accidents.

Plant Design Change Number 3-93-140

This change, entitled "Limitorque Actuator Spring Pack Replacement," is complete.

Description of Change

This change replaced the Limitorque Actuator spring packs in specific motor operated valves (MOVs) in the Charging System, Containment Atmosphere Monitoring System, Main Steam System, Reactor Coolant System, Containment Recirculation System, and Low Pressure Safety Injection System.

Reason for Change

This change supported work associated with the safety-related MOV testing and surveillance generic letter. Replacement of the existing spring packs allowed the actuators to conform to established thrust and torque requirements.

Safety Evaluation

The safety evaluation reviewed the impact on the MOVs ability to perform their design basis functions. Structural, electrical and mechanical aspects were considered.

This change was consistent with the original design of the various MOVs and ensured that they would function properly in normal and accident conditions.

Plant Design Change Number 3-93-146

This change, entitled "Deletion of Radioactive Liquid Waste Conductivity Instruments," is complete.

Description of Change

This change removed two conductivity instruments in the Radioactive Liquid Waste System whose function had been previously eliminated from the plant by bypass jumpers.

Reason for Change

The bypass jumpers were in place to eliminate the re-use of radioactive liquid waste distillate due to potential tritium build-up concerns. The need for the conductivity instruments was eliminated and this change permanently removed the control loops from the plant which allowed the elimination of the bypass jumpers.

Safety Evaluation

The deletion of the automatic control function of these instruments was compensated for by sampling of the distillate prior to discharge.

The system is non-safety-related and is not required for accident mitigation.

Plant Design Change Number 3-93-147

This change, entitled "Substitution of Excess Flow Check Valves with Steam Traps," is complete.

Description of Change

Replace two excess flow valves installed in series in the exhaust leg drain line of each of four atmospheric steam dumps with a single steam trap. The excess flow valves were safety-related. The steam traps are not safety-related.

The steam trap assembly includes an integral bypass line and valves, strainer, blowdown port and test ports.

Move the safety class interface point between the safety-related Main Steam System and the non safety-related drain system.

Reason for Change

These changes restored the capability to drain the exhaust leg of the atmospheric steam dump valves. The trap is more reliable and not susceptible to clogging as the original excess flow checks were.

Safety Evaluation

The steam traps are more reliable draining devices and will therefore decrease the probability of malfunction of the atmospheric dump valves.

The relocation of the safety class boundary was acceptable because the failure of a steam trap would not result in the failure of the associated atmospheric steam [illegible] nor would it cause a failure of the drain system.

Plant Design Change Number 3-93-149

This change, entitled "Steam Generator Tube U-Bend Stabilizer," is complete.

Description of Change

The change installed a stabilizer in a single tube in row 50, column 95 of the "A" steam generator. The stabilizer assembly consists of a spiral wound stainless steel hose and a stainless steel wire rope inside the hose. Both ends of the hose are equipped with a bullet nose which eased the stabilizer around the tube U-Bend. The stabilizer provides inside support for the length of the tube. The stabilizer remains permanently inside the tube with both ends of the tube plugged.

Reason for Change

Testing determined that the tube has a throughwall defect. It is postulated that a complete severance of the tube could occur during plant operation due to tube vibration and fretting against a tube anti-vibration bar. It is also postulated that the free ends of the severed tube could potentially damage any of the adjacent tubes, causing a primary to secondary side leak. The stabilizer dampens and minimizes flow induced tube vibrations. It is also designed to limit the movement of the free ends in the event the tube becomes completely severed.

Safety Evaluation

All aspects of the seismic qualification of the stabilizer have been evaluated as meeting design criteria. Materials compatibility has also been reviewed and found acceptable. Severance of the stabilizer spiral wound stainless hose combined with significant displacement of the free ends could possibly free the wire rope from the hose sheathing resulting in loose material on the secondary side of the steam generator. In this unlikely event, the rope would remain in the steam generator. The results of this occurrence could be a rupture of the steam generator tube which has already been evaluated. It is not credible to postulate that the stabilizer could be carried by steam outside the steam generator.

Plant Design Change Number 3-93-153

This change, entitled "Missile Shield Shim Modification for Control Rod Drive Mechanism (CRDM) Shroud Cooler," is complete.

Description of Change

This change modifies the CRDM Missile Shield shim design to facilitate removal and reinstallation of the shield. The change also deleted an interior bolt from the shield footing detail.

Reason for Change

The change was implemented because the existing shims were difficult to remove and one of the hold down bolts on the footing was hard to reach for removing and re-installing the missile shield.

Safety Evaluation

The change did not significantly alter system design or impact previously evaluated accidents. The modified shims and removed bolt did not affect the seismic qualification or structural integrity of the missile shield. The missile shield met all design requirements with the modified footings. The change was implemented during the refueling outage.

Plant Design Change Number 3-93-159

This change, entitled "Repower Ventilation Flow Switches from Uninterruptible Safety-Related Power Sources," is complete.

Description of Change

This change repowered flow switches and the "B" Auxiliary Building Filter fan auxiliary circuit from uninterruptible power supplies. The change affects the Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Pump Area ventilation system and the Auxiliary Building Filtration System. The Engineered Safety Features (ESF) Building Emergency Ventilation system circuit was also changed to increase the time delay for auto starting on a low flow condition. This change also modified the control circuit monitoring of these systems to ensure Control Room annunciation upon loss of power to the control circuits.

Reason for Change

The change was implemented due to an inherent delay in flow switch operation after a loss of power event. The flow switch utilizes a heated element which would cool down on a loss of power and would not function properly until power was restored and the element reached normal temperature. This characteristic caused unacceptable delays in starting fans in the CH/RPCCW and Auxiliary Building Filtration systems. Cycling problems may have occurred in the ESF Building Emergency Ventilation system due to a loss of power to the flow switch element.

Safety Evaluation

The change corrected a known deficiency and, therefore, increased system reliability.

Plant Design Change Number 3-93-160

This change, entitled "'C' Reactor Coolant Pump (RCP) Replacement," is complete.

Description of Change

The installed RCP was replaced with a new pump modified as follows:

- The turning vane bolts were replaced with upgraded bolts made of cold worked 316 stainless steel.

- The turning vane locking cups were replaced with locking cups which are split to allow expansion of the cup into the turning vane bolt counterbore, and which have more substantial retainer tabs to reduce the likelihood of movement out of the counterbore holes.

- The diffuser adapter bolts were also replaced with bolts made of cold worked 316 stainless steel.

Reason for Change

An inspection of the installed RCPs revealed that the locking cups could become loose and be torn from the counterbore, despite the existence of lock tabs/restraints designed to keep them from falling out. In addition, two turning vane bolt heads from the "B" RCP were found severed or almost severed. The locking cups loosened and fell out due to improper design. The turning vane bolts fractured because of intergranular stress corrosion cracking.

Since the design of "C" RCP is identical to that of "B" RCP, a pump with the upgraded design bolts and locking cups was installed. The new style locking cups have proven successful in other plants which have the same model RCP and the replacement turning vane bolt material is not susceptible to cracking.

The diffuser adapter bolts were also replaced because there have been occurrences of fractured diffuser bolts attributed to stress corrosion cracking due to the presence of chlorides.

Safety Evaluation

The replacement pump is basically identical to the original pump except for minor design differences which have no adverse affect on the pump performance. The turning vane and diffuser adapter bolts are of different material which is not susceptible to stress corrosion cracking. The upgraded locking cups incorporate design improvements recommended by the manufacturer. There is a slight difference in the replacement pump turning vane in that it does not have the stacked thermal cans found in the original RCP. The manufacturer has indicated that these cans were not needed and their absence does not affect pump performance. The hydraulic performance of the replacement pump was tested during plant return to power and found to satisfy the stipulated safety requirements.

Plant Design Change Number 3-93-161

This change, entitled "Limitorque Actuator Gearing Changes," is complete.

Description of Change

This change replaced the motor pinion and worn shaft gearing on the Power Operated Relief Valves (PORV) block valves.

Reason for Change

The valves were identified as being in an over thrust condition. The change reduced the speed of the valve closure which in turn eliminated the over thrust condition.

Safety Evaluation

The Safety Evaluation reviewed the valves ability to perform their design basis function. Structural, electrical, and mechanical aspects were considered. The isolation valves are normally open and they are only closed to isolate a leaking PORV. Eliminating the possible over thrust condition if the valve needed to be closed reduced the probability of valve failure.

Plant Design Change Number 3-93-167

This change, entitled "Fuel Transfer Tube Closure Bolt Reduction," is complete.

Description of Change

This change reduced the number of bolts required to bolt the Fuel Transfer Tube blind flange to the Transfer Tube in Containment from twenty to a minimum of four.

Reason for Change

The Fuel Transfer Tube blind flange uses two Quad Ring Gaskets which require a minimum of four bolts to adequately seal the blind flange to the tube. The end of the tube is in a harsh environment and installing twenty bolts is time consuming. The reduction of bolts from twenty to a minimum of four will result in a time savings as well as a reduction in personnel exposure during both installation and removal of the blind flange each refueling.

Safety Evaluation

The blind flange with a minimum of four bolts installed remains within its design criteria and will be capable of fulfilling its design function, which is to isolate Containment from the Fuel Transfer Tube.

Plant Design Change Number 3-93-174

This change, entitled "'B' Reactor Coolant Pump (RCP) Replacement," is complete.

Description of Change

The installed RCP was replaced with a new pump modified as follows:

- The turning vane bolts were replaced with upgraded bolts made of cold worked 316 stainless steel.
- The turning vane locking cups were replaced with locking cups which are split to allow expansion of the cup into the turning vane bolt counterbore, and which have more substantial retainer tabs to reduce the likelihood of movement out of the counterbore holes.
- The diffuser adapter bolts were also replaced with bolts made of cold worked 316 stainless steel.

Reason for Change

An inspection of the installed RCPs revealed that the locking cups could become loose and be torn from the counterbore, despite the existence of lock tabs/restraints designed to keep them from falling out. In addition, two turning vane bolt heads from the "B" RCP were found severed or almost severed. The locking cups loosened and fell out due to improper design. The turning vane bolts fractured because of intergranular stress corrosion cracking.

A pump with the upgraded design bolts and locking cups was installed.
The new style locking cups have proven successful in other plants which have the same model RCP and the replacement turning vane bolt material is not susceptible to cracking.

The diffuser adapter bolts were also replaced because there have been occurrences of fractured diffuser bolts attributed to stress corrosion cracking due to the presence of chlorides.

Safety Evaluation

The replacement pump is basically identical to the original pump except for minor design differences which have no adverse affect on the pump performance. The turning vane and diffuser adapter bolts are of different material which is not susceptible to stress corrosion cracking. The upgraded locking cups incorporate design improvements recommended by the manufacturer. There is a slight difference in the replacement pump turning vane in that it does not have the stacked thermal cans found in the original RCP. The manufacturer has indicated that these cans were not needed and their absence does not affect pump performance. The hydraulic performance of the replacement pump was tested during plant return to power and found to satisfy the stipulated safety requirements.

Plant Design Change Number 3-93-177

This change, entitled "'B' Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications," is complete.

Description of Change

A 2-inch elbow extension was welded onto the Reactor Plant Component Cooling Water (RPCCW) system inlet and outlet connections for the "B" RCP thermal barrier cooler. Supports associated with the piping change were relocated in order to accommodate the extensions. A 2-inch spacer was also installed in the RCP NO. 1 seal injection inlet.

Reason for Change

The RCP was replaced under Design Change Number 3-93-174. Cooling water and seal injection stubs were shorter than those on the original pump. 2-inch extensions were required to enable connections to be made to the plant piping.

Safety Evaluation

The Charging System and RPCCW piping modifications were performed in accordance with ASME requirements. The piping connections were schedule 160 rather than schedule 80 per the original design. However, the thicker pipe did not result in any reduction in flow. The supports were relocated per design in order to accommodate the piping changes.

Plant Design Change Number 3-93-178

This change, entitled "'A' Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications," is complete.

Description of Change

A 2-inch elbow extension was welded onto the Reactor Plant Component Cooling Water (RPCCW) system inlet and outlet connections for the "A" RCP thermal barrier cooler. Supports associated with the piping change were relocated in order to accommodate the extensions. A 2-inch spacer was also installed in the RCP NO. 1 seal injection inlet.

Reason for Change

The RCP was replaced under Design Change Number 3-93-185. Cooling water and seal injection stubs were shorter than those on the original pump. 2-inch extensions were required to enable connections to be made to the plant piping.

Safety Evaluation

The Charging System and RPCCW System piping modifications were performed in accordance with ASME requirements. The piping connections were schedule 160 rather than schedule 80 per the original design. However, the thicker pipe did not result in any reduction in flow. The supports were relocated per design in order to accommodate the piping changes.

Plant Design Change Number 3-93-180

This change, entitled "'D' Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications," is complete.

Description of Change

A 2-inch elbow extension was welded onto the Reactor Plant Component Cooling Water (RPCCW)system inlet and outlet connections for the "D" RCP thermal barrier cooler. Supports associated with the piping change were relocated in order to accommodate the extensions. A 2-inch spacer was also installed in the RCP NO. 1 seal injection inlet.

Reason for Change

The RCP was replaced under Design Change Number 3-93-186. Cooling water and seal injection stubs were shorter than those on the original pump. 2-inch extensions were required to enable connections to be made to the plant piping.

Safety Evaluation

The Charging System and RPCCW System piping modifications were performed in accordance with ASME requirements. The piping connections were schedule 160 rather than schedule 80 per the original design. However, the thicker pipe did not result in any reduction in flow. The supports were relocated per design in order to accommodate the piping changes.

Plant Design Change Number 3-93-184

This change, entitled "Modification of Reactor Coolant Pumps (RCP) Oil Collection System," is complete.

Description of Change

The oil collection system for each RCP was modified in the following manner:

RCP A — Rerouted oil collection piping. Modified two supports to accommodate the new piping locations and to enable axial pipe movement during a seismic event.

RCP B & C — Added spacers to a support to enable axial pipe movement during a seismic event.

RCP D — Rerouted oil collection piping and added a support to accommodate new piping locations. Added a spacer to a support to enable axial pipe movement during a seismic event.

Reason for Change

An evaluation during RCP pump replacements revealed that the potential loading on some nozzles exceeded current operability limits. The modifications resolved the loading issues.

Safety Evaluation

The changes to the piping and supports were required to ensure the RCP nozzle loads are not exceeded during a seismic event. The pipe rerouting and support modifications do not affect the operation of the RCP oil collection system.

Plant Design Change Number 3-93-185

This change, entitled "'A' Reactor Coolant Pump (RCP) Replacement," is complete.

Description of Change

The installed RCP was replaced with a new pump modified as follows:

- The turning vane bolts were replaced with upgraded bolts made of cold worked 316 stainless steel.
- The turning vane locking cups were replaced with locking cups which are split to allow expansion of the cup into the turning vane bolt counterbore, and which have more substantial retainer tabs to reduce the likelihood of movement out of the counterbore holes.
- The diffuser adapter bolts were also replaced with bolts made of cold worked 316 stainless steel.

Reason for Change

An inspection of the installed RCPs revealed that the locking cups could become loose and be torn from the counterbore, despite the existence of lock tabs/restraints designed to keep them from falling out. In addition, two turning vane bolt heads from the "B" RCP were found severed or almost severed. The locking cups loosened and fell out due to improper design. The turning vane bolts fractured because of intergranular stress corrosion cracking.

Since the design of "A" RCP is identical to that of "B" RCP, a pump with the upgraded design bolts and locking cups was installed. The new style locking cups have proven successful in other plants which have the same model RCP and the replacement turning vane bolt material is not susceptible to cracking.

The diffuser adapter bolts were also replaced because there have been occurrences of fractured diffuser bolts attributed to stress corrosion cracking due to the presence of chlorides.

Safety Evaluation

The replacement pump is basically identical to the original pump except for minor design differences which have no adverse affect on the pump performance. The turning vane and diffuser adapter bolts are of different material which is not susceptible to stress corrosion cracking. The upgraded locking cups incorporate design improvements recommended by the manufacturer. There is a slight difference in the replacement pump turning vane in that it does not have the stacked thermal cans found in the original RCP. The manufacturer has indicated that these cans were not needed and their absence does not affect pump performance. The hydraulic performance of the replacement pump was tested during plant return to power and found to satisfy the stipulated safety requirements.

Plant Design Change Number 3-93-186

This change, entitled "'D' Reactor Coolant Pump (RCP) Replacement," is complete.

Description of Change

The installed RCP was replaced with a new pump modified as follows:

- The turning vane bolts were replaced with upgraded bolts made of cold worked 316 stainless steel.
- The turning vane locking cups were replaced with locking cups which are split to allow expansion of the cup into the turning vane bolt counterbore, and which have more substantial retainer tabs to reduce the likelihood of movement out of the counterbore holes.
- The diffuser adapter bolts were also replaced with bolts made of cold worked 316 stainless steel.

Reason for Change

An inspection of the installed RCPs revealed that the locking cups could become loose and be torn from the counterbore, despite the existence of lock tabs/restraints designed to keep them from falling out. In addition, two turning vane bolt heads from the "B" RCP were found severed or almost severed. The locking cups loosened and fell out due to improper design. The turning vane bolts fractured because of intergranular stress corrosion cracking.

Because the "D" RCP had the same design locking cups, locking cup tabs/restraints, and turning vane bolts, a pump with the upgraded design bolts and locking cups was installed. The new style locking cups have proven successful in other plants which have the same model RCPs and the replacement turning vane bolt material is not susceptible to cracking.

The diffuser adapter bolts were also replaced because there have been occurrences of fractured diffuser bolts attributed to stress corrosion cracking due to the presence of chlorides.

Safety Evaluation

The replacement pump is basically identical to the original pump except for minor design differences which have no adverse affect on the pump performance. The turning vane and diffuser adapter bolts are of different material which is not susceptible to stress corrosion cracking. The upgraded locking cups incorporate design improvements recommended by the manufacturer. There is a slight difference in the replacement pump turning vane in that it does not have the stacked thermal cans found in the original RCP. The manufacturer has indicated that these cans were not needed and their absence does not affect pump performance. The hydraulic performance of the replacement pump was tested during plant return to power and found to satisfy the stipulated safety requirements.

Plant Design Change Number 3-93-200

This change, entitled "Auxiliary Building Filtration System Supplemental Leak Collection and Release System (SLCRS) Enhancements," is complete.

Description of Change

This change revised the "B" Train Auxiliary Building Ventilation Spec 200 Microprocessor logic to initiate the 30 second timer immediately upon receipt of an accident signal. Additionally, a reset feature was added to require two deliberate operator actions to shutdown the "B" Auxiliary Building filter fan after an accident condition. The time delays associated with low flow detection for the Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Pump Area ventilation were decreased from 20 seconds to 10 seconds. The setpoints for the associated flow switches were also changed to provide for consistent response times. The final aspect of the change was to implement a new control scheme for the CH/RPCCW ventilation system. The control switches are in AUTO with one fan running and one in standby. Use of the ON position is no longer allowed.

Reason for Change

The change was implemented to ensure adequate system response to meet the drawdown criteria of 0.4 inches water gauge within 2 minutes. The control switch scheme was changed to alleviate single failure concerns with the CH/RPCCW ventilation system.

Safety Evaluation

This change implemented modifications which enhance the ability of SLCRS and Auxiliary Building Filtration systems to achieve negative pressure requirements. This ensures the safety of the public by maintaining postulated offsite releases within acceptable limits.

Plant Design Change Number 3-93-204

This change, entitled "Support Modification to Portion of Main Steam Line," is complete.

Description of Change

This change modified supports on the main steam line going to the Turbine Driven Auxiliary Feedwater (TDAFW) pump. A snubber was replaced and its associated support was modified to allow for the replacement. Also, a spring-can was replaced and its associated support was modified.

Reason for Change

A re-analysis of the supports on the main steam line to the TDAFW pump resulted in higher support loads. This change made modifications to the supports on the line to accommodate for the higher loads.

Safety Evaluation

This change was consistent with the design of the steam supply to the TDAFW pump. The changes allowed the increased load to be accommodated.

Plant Design Change Number 3-93-205

This change, entitled "Auxiliary Building Filtration System Supplemental Leak Collection and Release System (SLCRS) Enhancements," is complete.

Description of Change

This change revised the "B" Auxiliary Building filter fan trip circuit by replacing a timing relay with an auxiliary relay. The change will cause the "B" filter fan to shut down and the "A" filter fan to start first during an accident. The system is not designed to auto start the Train "A" fan on Train "B" failure.

Reason for Change

The previous Auxiliary Building Filtration system design made the Train A fan inoperable whenever the Train B fan was running. This change improved system availability by allowing the Train "A" fan to start first with the Train "B" as backup.

Safety Evaluation

The change replaced a 20 second timing relay with an auxiliary relay which in effect allows the "B" filter fan to shut down faster when an accident signal is generated. This would allow the Train A fan to start first. This meets the original design intent of the system and is an improvement in safety function of the current system operation.

Plant Design Change Number 3-93-210

This change, entitled "Revision to Flow Switch Setpoints for Supplemental Leak Collection and Release System (SLCRS) Fans," is complete.

Description of Change

This change revised the flow switch setpoints for the SLCRS fans. The flow switches do not affect the response of the SLCRS fans to an accident signal. Both fans will automatically start upon receipt of a Safety Injection signal, however, if one fan shuts down and the running fan subsequently fails, the flow switch will start the standby fan.

Reason for Change

During monthly surveillance testing of the SLCRS fans, the standby fan was starting after one SLCRS fan was manually started. The operators were unable to shut down the standby fan to complete surveillance testing on one SLCRS fan. This change will prevent the standby fan from starting or at least allow the standby fan to be shutdown after the first SLCRS fan gets up to speed.

Safety Evaluation

The change revised the setpoints of SLCRS fan flow switches to eliminate the potential for false starting of the standby SLCRS fan. This change does not adversely affect the SLCRS performance or design function. The setpoint change to the flow switches has no adverse impact on other safety-related systems or equipment.

Plant Design Change Number 3-93-218

This change, entitled "Over Power Delta-T (OPDT) and Over Temperature Delta-T (OTDT) Turbine Runback Setpoint Revision," is complete.

Description of Change

This change reduced the OPDT and OTDT turbine runback and rod block setpoint signal from 3% to 1%.

Reason for Change

Due to periodic hot leg temperature fluctuations which were believed to originate in the upper plenum, spurious turbine runbacks were occurring.

The existing 3% setpoint was allowing turbine runback signals to be generated if the margin to the OTDT or OPDT trip setpoint fell to less than 3%. Reducing the setpoint to 1% increased the margin and eliminated turbine runbacks. The plant was able to proceed to 100% operating power.

Safety Evaluation

The OPDT and OTDT runback and rod block signals are not credited for accident protection.

Software Installation Plan M3-91-23333

This change, entitled "Supplemental Leak Collection and Release System (SLCRS) Normal Range Radiation Monitor Noise Spike Filter Software Modification," is complete.

Description

The change incorporates a software filter which "holds up" the processing of signals which change by more than a threshold amount. If the signal remains high until the next sample cycle, the signal is considered valid and passed along to be processed for alarms. The radiation monitor detects gaseous activity downstream of the SLCRS filter.

Reason for Change

During normal plant operations, this consists of monitoring the discharge point for the containment vacuum pumps, condenser air removal, reactor plant gaseous vents, and reactor plant aerated vents. No automatic isolation function is associated with this radiation monitor.

The change eliminated spurious alarms which resulted from noise spikes. The spikes are due to induced electro-magnetic interference, and the noise spike filter is cost-effective compared with relocating cables or replacing them with better shielded ones.

Safety Evaluation

The change increases instrument response times by one to two seconds. This increase is trivial compared to the 30 minute operator response time assumed in the Final Safety Analysis Report. The increase in radiological consequence of an additional two second delay in operator response is insignificant.

Software Implementation Package 3-92-17454

This change, entitled "Steam Generator Blowdown Sample Radiation Monitor Software Modification," is complete.

Description of Test

This change incorporated a 20 second delay into the low flow alarm processing for the Steam Generator Blowdown Sampling Monitor.

Reason for Test

The Steam Generator Blowdown Sampling Monitor had experienced spurious alarms which resulted from brief flow variations. The modification ensured that a low flow condition had to exist for 20 consecutive seconds before the monitor would produce a low flow alarm.

Safety Evaluation

This radiation monitor is not required for accident mitigation. The automatic isolation of blowdown upon high activity alarm is not affected by this change.

Software Installation Plan M3-93-01575

This change, entitled "Waste Neutralization Sump Radiation Monitor Low Flow Alarm and Low Pressure Alarm Delays," is complete.

Description of Change

This change installed a time delay in the alarms for low flow and low pressure from the Waste Neutralization Sump Radiation Monitor.

Reason for Change

Normal fluctuations in flow and pressure were causing numerous alarms. By lengthening the time that a low flow or low pressure condition had to exist before an alarm is generated, the number of nuisance alarms was decreased.

Safety Evaluation

This radiation monitor is not required for mitigation of any accident condition.

The automatic isolation of effluent discharge upon a high activity alarm is not affected by this change.

Software Installation Plan M3-93-10641

This change, entitled "Reactor Plant Component Cooling Water (RPCCW) Radiation Monitor Noise Spike Filter and Low Flow and Low Pressure Alarm Delays," is complete.

Description of Change

This change installed a 20 second time delay in the alarms for low flow and low pressure from the RPCCW radiation monitor and a electronic filter for noise spikes.

Reason for Change

Normal fluctuations in flow and pressure were causing numerous alarms. By lengthening the time that a low flow or low pressure condition had to exist before an alarm is generated, the number of nuisance alarms was decreased.

Induced noise spikes were causing spurious high radiation alarms. The filter was more cost effective than installing new cabling.

Safety Evaluation

This radiation monitor is not required for mitigation of any accident condition. Since the RPCCW system is a closed cooling system, there is normally no leakage to the environment. There are no automatic isolation features associated with this radiation monitor.

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

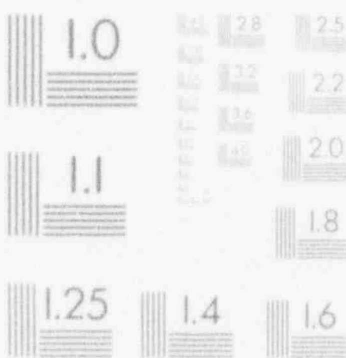
IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)



150mm

6″

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

Software Implementation Package 3-93-15026

This change, entitled "Inadequate Core Cooling Monitor (ICCM) Software Implementation Package for ICCM Alarm Unit," is complete.

Description of Change

This change corrected the logic for the upper head and plenum level alarm and provided a method to positively reset the alarm when the alarm condition clears.

Reason for Change

Prior to the modification, there was a deficiency in the code such that with the temperature margin between 15 and 20°F, the ICCM reactor upper head and plenum level alarm annunciators would cycle in and out every 4 seconds. Also, there was no provision in the code to positively reset the alarm when the condition cleared.

Safety Evaluation

There are no malfunctions associated with the software modification. The change was implemented by installing modules which had been programmed with the new code, and the system was subsequently tested to verify proper operation prior to being placed into service. As such, there are no credible failure modes relating to the ICCM alarm module.

Software Implementation Plan 3-93-20623

This change, entitled "Modify Redundant Measurement Program to Correct Reactor Coolant System (RCS) Leakage Program," is complete.

Description of Change

The calculation of the average pressurizer pressure by the process computer was based upon the average of the four narrow range pressurizer pressure indicators. The narrow range indicators "Bottom out" at 1700 psia. Between 0 and 1700 psia, the average pressure is calculated by the average of the two wide range pressurizer pressure indicators.

Reason for Change

The RCS leakage calculation program gave invalid results when pressurizer pressure was below 1700 psia. This change ensures proper pressure will be used when determining RCS leakage using the process computer with pressure below 1700 psia.

Safety Evaluation

The safety evaluation evaluated automatic substitution of wide range pressure vice manual entry of RCS pressure and its effect on the RCS leakage calculation. The operation of the plant does not rely solely on the plant process computer. The computer does not provide any control signals and there are other plant indicators that the Operators use to control and operate the plant.

PROCEDURE CHANGES

Procedure Number	Title
OP 3301L, Rev. 1	Reactor Coolant Loop Stop Valves
OP 3314A, Rev. 12	Auxiliary Building Heating, and Ventilation Air Conditioning System
OP 3316A, Rev. 7	Main Steam System
OP 3330A, Rev. 7	Reactor Plant Component Cooling Water (RPCCW) System
OP 3612B.4, Rev. 10	Containment Local Leak Rate Test Type "C" Penetration
OP 3646B.8, Rev. 6	Emergency Generator Fuel Oil Particulate Sample Analysis
IC 3464I08, Rev. 0	Steam Generator Water Level Control Tuning
OPS Form 3273-3/4.2.3.4, Rev. 0	Quadrant Power Tilt Ratio (QPTR) Technical Specification Clarification
OPS Form 3273-3/4.3.1.1.3, Rev. 3	Moderator Temperature Coefficient
OPS Form 3273-3/4.3.1.3.5, Rev. 3	Shutdown Rod Insertion Limit
OPS Form 3273-3/4.3.1.3.6, Rev. 3	Control Rod Insertion Limit
OPS Form 3273-3/4.3.2.1.1, Rev. 2	Axial Flux Difference
OPS Form 3273-3/4.3.2.2.1, Rev. 3	Heat Flux Hot Channel Factor FQ(Z) Four Loop Operation
OPS Form 3273-3/4.3.2.3.1, Rev. 2	Reactor Coolant System Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor Four Loops Operating
OPS Form 3273-3/4.3.1.2.2, Rev. 0	Charging Flow Paths-Operating
OPS Form 3273-3/4.3.1.2.4, Rev. 1	Charging Pumps - Operating
OPS Form 3273-3/4.3.5.2, Rev. 1	Emergency Core Cooling System Subsystems - T_{AVG} Greater Than or Equal to 350°F

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
OPS Form 3273-3/4.3.5.3, Rev. 1	Emergency Core Cooling System Subsystems - T_{AVG} Less Than 350°F
OPS Form 3273-3/4.3.6.3, Rev. 2	Containment Isolation Valve Table
OPS Form 3273-3/4.3.6.3, Rev. 4	Containment Isolation Valve Table
OPS Form 3273-3/4.3.6.3, Rev. 5	Containment Isolation Valve Table
OPS Form 3273-3/4.3.6.6.1, Rev. 1	Supplemental Leak Collection and Release System (SLCRS)
OPS Form 3273-3/4.3.7.4, Rev. 1	Service Water System
OPS Form 3273-3/4.3.7.7, Rev. 0	Control Room Emergency Ventilation System
OPS Form 3273-3/4.3.7.8, Rev. 1	Control Room Envelope Pressurization System Technical Specification Clarification
OPS Form 3273-3/4.3.7.9, Rev. 1	Auxiliary Building Filter System
OPS Form 3273-3/4.3.7.9, Rev. 2	Auxiliary Building Filter System
OPS Form 3273-3/4.3.9.12, Rev. 0	Fuel Building Exhaust Filter System
OPS Form 3273-3/4.3.4.9.1, Rev. 0	Pressure and Temperature Limits
OPS Form 3612B.4-124, Rev. 1	Proceduralizing Local Leak Rate Test (LLRT) for Penetration 52
OPS Form 3612B.4-125, Rev. 1	Proceduralizing Local Leak Rate Test (LLRT) for Penetration 52
SP 31103, Rev. 2	Integrated Leak Rate Test
SP 3448E51, Rev. 1	Diesel Sequencer Train "A" Actuation Timer Test
SP 3448E52, Rev. 0	Diesel Sequencer Train "B" Actuation Timer Test

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
SP 3446F31, Rev. 4	Solid State Protection System Refuel Test
SP 3442J12, Rev. 2, and SP 3442J12, Rev. 1	Cold Overpressure Protection System (COPPS) Operational Test (Train "A" and "B")
SP 3646A.8, Rev. 10, Ch. 3	Slave Relay Test - "A" Train
SP 3646A.9, Rev. 10, Ch. 4	Slave Relay Test - "B" Train
OPS Form 3646A.8-1, Rev. 7, Ch. 6	Slave Relay Test - "A" Train
OPS Form 3646.9-1, Rev. 8, Ch. 2	Slave Relay Test - "B" Train
SP 3674.1, Rev. 3	Motor Operated Valve Thermal Overload Bypass Testing
OPS Form 3674.1-5, Rev. 0	Motor Operated Valve Thermal Overload Bypass Testing for Charging System Isolation Valve
EOP 3501, Rev. 1	Loss of All AC Power (Modes 5, 6, and 0)
EOP 3505, Rev. 4	Loss of Shutdown Cooling and/or Reactor Coolant System (RCS) Inventory
EOP 3505A, Rev. 0	Loss of Spent Fuel Pool Cooling
EOP ECA 0.0, Rev. 7	Loss of All AC Power
EOP 3503, Rev. 8	Shutdown Outside Control Room
EOP 3501, Rev. 4	Loss of All AC Power (Modes 5, 6, and 0)
EOP 35 E1-1, Rev. 10	Loss of Reactor or Secondary Coolant
EOP 35 FR-C.1, Rev. 6	Response to Inadequate Core Cooling
Spec. Proc. 91-3-010	Temporary Service Water to Turbine Plant Component Cooling Water Heat Exchanger
Spec. Proc. 93-3-006	Supplementary Leak Collection and Release System (SLCRS) Rebalancing

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
Spec. Proc. 93-3-019	Infrequently Performed Test for Evolution from Releasing SIGMA Mast Fuel Assembly F38
Spec. Proc. 93-3-023	Release of SIGMA Gripper Mast from Fuel Assembly F38
Spec. Proc. 93-3-039	Supplemental Leak Collection and Release System (SLCRS) Drawdown Only-Trains "A" and "B"
Spec. Proc. 93-3-040	Low Pressure Turbine Pre-Warming Procedure
Spec. Proc. 93-3-045	"D" Reactor Coolant Pump (RCP) Oil Addition to Operating Motor
SP 3613F.2-1, Rev. 3	Containment Purge and Exhaust Isolation Valve Operability Test
OPS Form 3273-3/4.3.6.3, Rev. 3	Containment Isolation Valve Table
SP 3614I.3, Rev. 7	Supplemental Leak Collection and Release System Negative Pressure Verification
OPS Form 3614I.3-1&2, Rev. 4	Supplemental Leak Collection and Release System Negative Pressure Verification Forms
AOP 3555, Rev. 3	Reactor Coolant Leak
AOP 3554, Rev. 4	Reactor Coolant Pump (RCP) Trip or Stopping an RCP at Power
AOP 3559, Rev. 3	Loss of Condenser Vacuum
AOP 3575, Rev. 0	Rapid Downpower
No Number	Safety Evaluation for the Leak Repair of an Auxiliary Feedwater System Check Valve
No Number	Safety Evaluation for the Leak Repair of Three Auxiliary Feedwater Check Valves
No Number	Safety Evaluation for the Tightening of the Bonnet Clamp of 3FWA*MOV35A
EOP 35 E-3, Rev. 10	Steam Generator Tube Rupture

PROCEDURE CHANGES (CONTINUED)

Procedure Number	Title
EOP 35 ES-1.2, Rev. 7	Post LOCA Cooldown and Depressurization
EOP 35 ES-1.4, Rev. 6	Transfer to Hot Leg Recirculation
EOP 35 ES-3.2, Rev. 6	Post Steam Generator Tube Rupture Cooldown Using Steam Blowdown
EOP 35 ES-3.3, Rev. 7	Post Steam Generator Tube Rupture Cooldown Using Steam Dump
EOP 35 FR-H.1, Rev. 7	Response to Loss of Secondary Heat Sink
EOP 35 FR-P.1, Rev. 7	Response to Imminent Pressurized Thermal Shock Condition
EOP 35 FR-P.2, Rev. 6	Response to Anticipated Pressurized Thermal Shock Condition
EOP 35 FR-S.1, Rev. 8	Response to Nuclear Power Generation/ ATWS
EOP 35 FR-Z.1, Rev. 7	Response to High Containment Pressure
EOP 3501, Rev. 7	Loss of All AC Power (Modes 5, 6 and Zero)
EOP 35 ECA-0.0, Rev. 9	Loss of All AC Power
EOP 35 ECA-0.1, Rev. 7	Loss of All AC Power - Recovery without Safety Injection (SI) Required
EOP 35 ECA-0.2, Rev. 6	Loss of All AC Power - Recovery with SI Required
EOP 35 ECA-0.3, Rev. 0	Loss of All AC Power - Recovery with the Station Blackout (SBO) Diesel
EOP 35 ECA-1.1, Rev. 6	Loss of Emergency Coolant Recirculation
EOP 35 ECA-3.1, Rev. 9	Steam Generator Tube Rupture with Loss of Reactor Coolant - Subcooled Recovery Desired
EOP 35 ECA-3.2, Rev. 6	Steam Generator Tube Rupture with Loss of Reactor Coolant - Saturated Recovery Desired
EOP 35 ECA-3.3, Rev. 7	Steam Generator Tube Rupture without Pressurizer Pressure Control
Vendor Procedure FP-NEU-1, Rev. 0	Fuel Reconstitution

Procedure Number	Title
OP 3301L, Rev. 1	Reactor Coolant Loop Stop Valves

Description of Change

This procedure revision provides guidance for stroking the cold leg stop valves while the plant is in a shutdown condition.

Reason for Change

Loop stop valves were closed during the refueling outage to support maintenance activities in the loop. The cold leg stop valves may not completely close on the initial attempt. Experience has shown that cycling the valve several times while lubricating the stem will allow the valve to fully close. The procedure was revised to provide the necessary guidance to accomplish this.

Safety Evaluation

Stroking of the cold leg stop valve accomplished with the hot leg stop valve closed and the reactor coolant pump off. Loop temperature and boron concentration were verified to be consistent with the other loops prior to stroking the valve. Therefore, the probability of occurrence and consequences of an inadvertent dilution or startup of an inactive loop accident were not affected. Also, this change does not create the possibility of a different type of accident since all equipment will be operated within its design capabilities and technical specification requirements.

This procedure operates all equipment in accordance with their design capabilities and in accordance with existing procedures. Only the plant conditions under which the valves are to be stroked is changing.

Procedure Number	Title
OP 3314A, Rev. 12	Auxiliary Building Heating, Ventilation and Air Conditioning System

Description of Change

This change provided guidance for stationing a dedicated operator at the circuit breaker for the motor operated damper at the discharge of the "B" Auxiliary Building filter fan when the "B" train fan is being run. The change also provided specific instructions for the operator in the event of an accident signal. Opening the breaker will cause the damper to close which will shut down the "B" train filter fan.

Reason for Change

The "A" train Auxiliary Building filter fan is the designed lead fan in response to an accident signal. If the "A" train fan fails, the "B" train fan will start on high plenum pressure. The system design does not ensure that the 'A" train fan will start for all postulated "B" train fan failures. For this reason the "A" train fan is considered inoperable whenever the "B" train fan is operating. When the "B" train fan is running, a dedicated operator is utilized to ensure that one train of Auxiliary Building Filtration is available.

Safety Evaluation

This change requires that a dedicated operator perform a manual action to ensure that a safety related component will automatically respond to an accident signal. Based on the simplicity of the manual action, it is reasonable to assume that adequate system response will be maintained. This change provides guidance for very rare conditions.

The only malfunction evaluated for this change is the failure of the damper to close by spring action. This action is incorporated in normal system operation and has been reliable. The probability for this failure is minimal. The change does not adversely affect the system and will allow the system to operate to mitigate the consequences of potential radiological releases.

Procedure Number	Title
OP 3316A, Rev. 7	Main Steam System

Description of Change

A section was added to control gagging of a steam generator safety valve. This section requires that all technical specifications be observed.

Reason for Change

This section is to be used when performing hydrostatic tests, setting the relief valves and at the discretion of the Shift Supervisor when a safety is stuck open.

Safety Evaluation

This change ensures that the plant is operated in accordance with technical specifications in the event that a main steam safety needs to be gagged.

Procedure Number	Title
OP 3330A, Rev. 7	Reactor Plant Component Cooling Water (RPCCW) System

Description of Change

The Train "A" system was aligned to supply Train "B" non-safety-related header by opening available cross-connect valves.

Reason for Change

Train "B" Service Water outage removed cooling from Train "B" RPCCW system coolers. The cross-connection was made in order to enable Train "A" RPCCW to supply cooling to the seal water heat exchanger which is normally supplied by Train "B" RPCCW.

Safety Evaluation

An assessment determined that the Train "A" RPCCW pump and heat exchanger were capable of supplying the additional loads without exceeding the maximum allowed flow of 8100 gallons per minute. Also, if a problem were to occur in Train "B" then the cross-connect valves could be closed immediately so the Train "A" problem would be readily isolated from Train "B."

Procedure Number	Title
OP 3612B.4, Rev. 10	Containment Local Leak Rate Test Type "C" Penetration

Description of Change

This change provided guidance to supply the Containment Building Fire Protection Water system from the Auxiliary Building Fire Protection Water system while performing the local leak rate test on the containment penetration for the fire protection system.

Reason for Change

The normal fire protection water supply to the Containment Building fire protection water supply to the Containment Building had been isolated for maintenance testing. Bypass jumpers had previously been used to provide the temporary source of fire protection water.

Safety Evaluation

All fire protection water flow and pressure requirements were evaluated and maintained throughout the maintenance testing evolution. Fire suppression to the Containment Building was not impactad by this change.

Procedure Number	Title
OP 3646B.8, Rev. 6	Emergency Generator Fuel Oil Particulate Sample Analysis

Description of Change

This procedure change modified the method for obtaining the monthly fuel oil sample.

Reason for Change

The previous method for obtaining the monthly fuel oil sample required defeating of an electrical interlock by using an electrical jumper device. By electrically cross-connecting the fuel oil follow transfer pumps while collecting the sample, the need to install an electrical jumper was eliminated.

Safety Evaluation

This change does not significantly increase the probability of a failure of the backup fuel oil transfer pumps. The system was designed to provide the required sample.

Procedure Number	Title

Procedure Number	Title
IC 3464I08, Rev. 0	Steam Generator Water Level Control Tuning

Description of Change

This procedure was implemented to collect data which was used to optimize the Steam Generator Water Level Control system.

Reason for Change

This test was performed to optimize the Steam Generator Water Level Control system. The Feedwater Regulator Valves were opened to approximately 70 to 75% of complete travel. Data was collected for various parameters to determine system performance. System gain adjustments were determined which were used to tune the system.

Safety Evaluation

Plant Technical Specifications requires that the Steam Generator Water Level Control system be operable during start-up and at-power operations. This procedure did not impact the operability of the Steam Generator Water Control system. The precautions utilized for this procedure ensured that the master speed controller was placed in manual prior to calibrating or replacing any 7300 Process Control system cards, or connecting/disconnecting any test equipment. The Unit No. 3 Operations Department determined how long system oscillations could persist during testing prior to taking corrective action with the master speed controller to regain system stability. Implementation of this procedure did not impact the design function of the system.

Procedure Number	Title
OPS Form 3273-3/4.2.3.4, Rev. 0	Quadrant Power Tilt Ratio (QPTR) Technical Specification Clarification

Description of Change

The revision clarifies three technical specification action statements. These action statements delineate actions required to be taken if QPTR exceeds 1.02.

Reason for Change

The subject action statements are unclear as to whether it is acceptable to increase or decrease power above 50% power when QPTR exceeds 1.02. The revision was written to clarify what power manipulations are allowed and the time frames in which they are allowed.

Safety Evaluation

The revision was a clarification of technical specification action statements; it did not change any operating limits.

Procedure Number	Title
OPS Form 3273-3/4.3.1.1.3, Rev. 3	Moderator Temperature Coefficient
OPS Form 3273-3/4.3.1.3.5, Rev. 3	Shutdown Rod Insertion Limit
OPS Form 3273-3/4.3.1.3.6, Rev. 3	Control Rod Insertion Limit
OPS Form 3273-3/4.3.2.1.1, Rev. 2	Axial Flux Difference
OPS Form 3273-3/4.3.2.2.1, Rev. 3	Heat Flux Hot Channel Factor FQ(Z)Four Loop Operation
OPS Form 3273-3/4.3.2.3.1, Rev. 2	Reactor Coolant System Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor Four Loops Operating

Description of Change

These revisions incorporated Cycle 5 core operating limits into the Core Operating Limits Report (COLR). In addition, the L(Z) term was removed from the equation for calculating the Heat Flux Hot Channel Factor.

Reason for Change

Operating limits on Moderator Temperature Coefficient, Rod Insertion Limit, Axial Flux Difference, and peaking factors change on a cycle-to-cycle basis. These limits have been removed from technical specifications and are contained in the COLR. These revisions update the COLR to incorporate the Cycle 5 limits. The L(Z) term has been incorporated into the W(Z) constants. Leaving the L(Z) in the equation for the Heat Flux Hot Channel Factor would result in a double penalty to this limit.

Safety Evaluation

Operation of the Cycle 5 core within the new limits will ensure that the plant is maintained within the initial conditions assumed in the accident analysis.

The COLR specifies limits within which the plant must be operated; operation outside these limits requires the plant be either shut down, or placed in a condition wherein the limits can be met.

Procedure Number	Title
OPS Form 3273-3/4.3.1.2.2, Rev. 0	Charging Flow Paths-Operating

Description of Change

This change provides information in the Technical Requirements Manual to administratively control the safety-related heaters in the area. Eight heaters, four in each train, were installed to ensure minimum acceptable temperatures were maintained in the Charging Pump cubicles.

Reason for Change

When outdoor air temperature reaches 17°F or below, the heaters are required to maintain minimum temperatures in the Charging Pump cubicles in the event of a loss of normal power. This change implemented administrative controls to ensure that the heaters are available if required for Charging Pump operability.

Safety Evaluation

The change implemented additional administrative controls to ensure that support systems and components are available when required. These administrative controls will ensure Charging Pump operability at design basis temperature conditions. There are no adverse affects as a result of these administrative controls.

Procedure Number	Title
OPS Form 3273-3/4.3.1.2.4, Rev. 1	Charging Pumps - Operating

Description of Change

This change provided information in the Technical Requirements Manual to administratively control the safety-related heaters in the Charging Pump and Reactor Plant Component Cooling Water pump area. Eight heaters, four in each train, were installed to ensure minimum acceptable temperatures were maintained in the Charging Pump cubicles.

Reason for Change

When outdoor air temperature reaches 17°F or below, the heaters are required to maintain minimum temperatures in the Charging Pump cubicles in the event of a loss of normal power. This change implemented administrative controls to ensure that the heaters are available if required for Charging Pump operability.

Safety Evaluation

The change implemented additional administrative controls to ensure that support systems and components are available when required. These administrative controls will ensure Charging Pump operability at design basis temperature conditions. There are no adverse effects as a result of these administrative controls.

Procedure Number | Title

OPS Form 3273-3/4.3.5.2, Rev. 1 Emergency Core Cooling System Subsystems - T_{AVG} Greater Than or Equal to 350°F

Description of Change

This change provides information in the Technical Requirements Manual to administratively control the safety-related heaters in the Charging and Reactor Plant Component Cooling Water pump area. Eight heaters, four in each train, were installed to ensure minimum acceptable temperatures were maintained in the Charging Pump cubicles.

Reason for Change

When outdoor air temperature reaches 17°F or below, the heaters are required to maintain minimum temperatures in the Charging Pump cubicles in the event of a loss of normal power. This change implemented administrative controls to ensure that the heaters are available if required for Charging Pump operability.

Safety Evaluation

The change implemented additional administrative controls to ensure that support systems and components are available when required. These administrative controls will ensure Charging Pump operability at design basis temperature conditions. There are no adverse affects as a result of these administrative controls.

Procedure Number	Title
OPS Form 3273-3/4.3.5.3, Rev 1	Emergency Core Cooling System Subsystems - T_{AVG} Less Than 350°F

Description of Change

This change provided information in the Technical Requirements Manual to administratively control the safety-related heaters in the Charging and Reactor Plant Component Cooling Water pump area. Eight heaters, four in each train, were installed to ensure minimum acceptable temperatures were maintained in the Charging Pump cubicles.

Reason for Change

When outdoor air temperature reaches 17°F or below, the heaters are required to maintain minimum temperatures in the Charging Pump cubicles in the event of a loss of normal power. This change implemented administrative controls to ensure that the heaters are available if required for Charging Pump operability.

Safety Evaluation

The change implemented additional administrative controls to ensure that support systems and components are available when required. These administrative controls will ensure Charging Pump operability at design basis temperature conditions. There are no adverse affects as a result of these administrative controls.

Procedure Number | Title

OPS Form 3273-3/4.3.6.3, Rev. 2 | Containment Isolation Valve Table

Description of Change

This change added the manual discharge isolation valve for the "B" Motor Driven Auxiliary Feedwater pump as an acceptable replacement for the motor operated isolation valve which is the normal containment isolation valve. The manual valve was closed and locked in position to provide containment integrity while the actuator for the motor operated valve was removed and repaired.

Reason for Change

To allow the required repair of the motor operated valve actuator. This change fulfilled of the requirements for containment integrity.

Safety Evaluation

The use of the manual valve meets the design criteria for containment isolation. Therefore, the use may be considered the same as if the motor operated valve was closed. There is no additional impact.

Procedure Number	Title
OP 3273-3/4.3.6.3, Rev. 4	Containment Isolation Valve Table

Description of Change

This procedure change deleted the containment isolation valves for the pressurizer liquid sample line penetration and deleted check valves in the suction line for the Hydrogen Recombiner System.

Reason for Change

The liquid sample line had been cut and capped by Plant Design Change Number 3-93-116. The containment penetration was also cut and capped. This procedure change incorporated this modification in the isolation valve table.

The check valves in the suction lines of the Hydrogen Recombiner System allow flow out of containment, and therefore, have no containment isolation function. Therefore, they were deleted from the isolation valve table.

Safety Evaluation

The pressurizer sample line was no longer in use. The caps on each end of the containment penetration exceed the containment isolation requirements.

The check valve in the Hydrogen Recombiner System suction line did not provide a containment isolation function.

Procedure Number | Title

OPS Form 3273-3/4.3.6.3, Rev. 5 | Containment Isolation Valve Table

Description of Change

This change deleted the isolation valves for the four atmospheric steam dumps from the list of containment isolation valves.

Reason for Change

During the licensing of Millstone Unit No. 3, the containment isolation valves were identified both in the Final Safety Analysis Report (FSAR) and Technical Specifications. Neither of these tables identified the isolation valves for the atmospheric steam dumps as containment isolation valves. However, during the removal of the Containment isolation valve table from technical specifications, they were inadvertently incorporated into the Operation's Department form.

Safety Evaluation

This change made the procedure consistent with the intent of Technical Specifications. These valves were not originally designed as containment isolation valves as demonstrated by their exclusion from both the FSAR and Technical Specification tables.

Procedure Number	Title
OPS Form 3273-3/4.3.6.6.1, Rev. 1	Supplemental Leak Collection and Release System (SLCRS)

Description of Change

This change provided guidance for determining operability of the SLCRS filters after 720 hours of operation. Specifically, the change states that the charcoal sample surveillance requirement at 720 hours of operation can be obtained at any time up to 720 hours + 25% or 900 hours.

Reason for Change

This change is being implemented to alleviate confusion regarding the operability of the filters after 720 hours of operation. Technical Specifications require a charcoal sample be removed from the filter after 720 hours of filter operation and sent out for laboratory analysis. Guidance was required concerning operability of the filters if they operated for more than 720 hours prior to a charcoal sample being removed.

Safety Evaluation

A review of regulatory guidance indicates the 720 hours was arbitrarily selected because it reflected approximately one month of filter operation. The charcoal sample testing vendor was consulted to determine any impact in prolonging filter operation prior to obtaining a charcoal sample. Based on past sample tests the extended run time would have an insignificant impact on the charcoal.

Procedure Number	Title
OPS Form 3273-3/4.3.7.4, Rev. 1	Service Water System

Description of Change

The Technical Specification clarification provides guidance to Operations with regard to the operability of the Service Water system when a leak occurs within a Service Water system train. The Service Water system and its supplied loads will remain operable unless it is determined the effects of the leak are detrimental to safe continued plant operation.

Reason for Change

The change provided guidance to Operations to ensure the Service Water system is maintained operable unless the severity and location of the leak prevents continued, safe operation of the plant.

Safety Evaluation

The system is designed to provide cooling water through separate supply and discharge lines to redundant safety-related components. Thus, assuming a single failure of one of redundant supply or discharge line, the system safety function is still achieved. A Service Water piping break is considered unlikely due to its design.

The minor leaks which occur do not threaten the ability of the system to fulfill its safety function.

Procedure Number | Title

OPS Form 3273-3/4.3.7.7, Rev. 0 — Control Room Emergency Ventilation System

Description of Change

This change provided guidance for determining operability of the Auxiliary Building Filter System filters after 720 hours of operation. Specifically, the change states that the charcoal sample surveillance requirement at 720 hours of operation can be obtained at any time up to 720 hours + 25% or 900 hours.

Reason for Change

This change is being implemented to alleviate confusion regarding the operability of the filters after 720 hours of operation. Technical Specifications require a charcoal sample be removed from the filter after 720 hours of filter operation and sent out for laboratory analysis. Guidance was required concerning operability of the filters if they operated for more than 720 hours prior to a charcoal sample being removed.

Safety Evaluation

A review of regulatory guidance indicates the 720 hours was arbitrarily selected, because it reflected approximately one month of filter operation. The charcoal sample testing vendor was consulted to determine any impact in prolonging filter operation prior to obtaining a charcoal sample. Based on past sample tests, the extended run time would have an insignificant impact on the charcoal.

Procedure Number	Title
OPS Form 3273-3/4.3.7.8, Rev. 1	Control Room Envelope Pressurization System Technical Specification Clarification

Description of Change

This change clarifies the surveillance requirement to maintain greater than 0.125" water gauge (wg) positive pressure for one hour in the Control Room Envelope. The clarification provides guidance for acceptability of momentary pressure spikes.

Reason for Change

Test data had determined that the differential pressure measuring device was subject to spikes caused by doors opening and closing. The acceptance criteria was based on maintaining a positive pressure in the Control Room Envelope during transients caused by winds and other factors. Due to the location of the differential pressure measuring device, localized effects of door movements were causing momentary spikes which could cause dips below the acceptance criteria.

Safety Evaluation

This system pressurizes the Control Room Envelope to greater than 0.125" wg for one hour to minimize radiation exposure to Control Room operators in the event of a Design Basis Accident. Pressurizing the envelope to greater than 0.125" wg above the initial atmospheric pressure ensures that it will remain at a relative positive pressure during subsequent changes in outside conditions.

The intent of the surveillance test is not changed in that the system is still verified to pressurize and maintain greater than 0.125" wg differential pressure above the initial atmospheric pressure.

Procedure Number | Title

OPS Form 3273-3/4.3.7.9, Rev. 1 | Auxiliary Building Filter System

Description of Change

This change provided guidance for determining operability of the Auxiliary Building Filter System filters after 720 hours of operation. Specifically, the change states that the charcoal sample surveillance requirement at 720 hours of operation can be obtained at any time up to 720 hours + 25% or 900 hours.

Reason for Change

This change is being implemented to alleviate confusion regarding the operability of the filters after 720 hours of operation. Technical Specifications require a charcoal sample be removed from the filter after 720 hours of filter operation and sent out for laboratory analysis. Guidance was required concerning operability of the filters if they operated for more than 720 hours prior to a charcoal sample being removed.

Safety Evaluation

A review of regulatory guidance indicates the 720 hours was arbitrarily selected, because it reflected approximately one month of filter operation. The charcoal sample testing vendor was consulted to determine any impact in prolonging filter operation prior to obtaining a charcoal sample. Based on past sample tests, the extended run time would have an insignificant impact on the charcoal.

Procedure Number	Title
OPS Form 3273-3/4.3.7.9, Rev. 2	Auxiliary Building Filter System

Description of Change

This change adds a paragraph to the Limiting Condition For Operation section of the form to clarify that if a train of Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Area Ventilation is unavailable, the associated train of Auxiliary Building Filtration is also inoperable. An explanation was also added to the bases section of the form.

Reason for Change

The Auxiliary Building Filtration system will not operate properly without the CH/RPCCW Area Ventilation system. The administrative controls implemented by this change ensure that the Auxiliary Building Filtration system meets the requirements of the plant's Technical Specifications.

Safety Evaluation

The change implemented additional administrative controls to ensure safety system operability and availability. There is no adverse impact on normal or emergency plant operations.

Procedure Number	Title
OPS Form 3273-3/4.3.9.12, Rev. 0	Fuel Building Exhaust Filter System

Description of Change

This change provided guidance for determining operability of the Auxiliary Building Filter System filters after 720 hours of operation. Specifically, the change states that the charcoal sample surveillance requirement at 720 hours of operation can be obtained at any time up to 720 hours + 25% or 900 hours.

Reason for Change

This change is being implemented to alleviate confusion regarding the operability of the filters after 720 hours of operation. Technical Specifications require a charcoal sample be removed from the filter after 720 hours of filter operation and sent out for laboratory analysis. Guidance was required concerning operability of the filters if they operated for more than 720 hours prior to a charcoal sample being removed.

Safety Evaluation

A review of regulatory guidance indicates the 720 hours was arbitrarily selected, because it reflected approximately one month of filter operation. The charcoal sample testing vendor was consulted to determine any impact in prolonging filter operation prior to obtaining a charcoal sample. Based on past sample tests, the extended run time would have an insignificant impact on the charcoal.

Procedure Number	Title
OPS Form 3273-3/4.3.4.9.1, Rev. 0	Pressure and Temperature Limits

Description of Change

This clarification to Technical Specifications limits Reactor Coolant System (RCS) Heatup to a maximum of 60°F/Hr.

Reason for Change

The wording in the Technical Specifications indicated the limit was 100°F/Hr. However, the curves were developed assuming a limit of 60°F/Hr. Since 60°F/Hr is more restrictive and consistent with the development of the curves, it was stipulated to be the maximum heatup limit.

Safety Evaluation

A heatup limit of 60°F/Hr. is the most restrictive of the two limits implied or stated in Technical Specification. Therefore, limiting RCS heatup to a maximum of 60°F/Hr is conservative.

Procedure Number	Title
OPS Form 3612B.4-124, Rev. 1 OPS Form 3612B.4-125, Rev. 1	Proceduralizing Local Leak Rate Test(LLRT) for Penetration 52

Description of Change

This change installed two temporary Service Air hoses from two locations in the Auxiliary Building through the personnel access hatch to two locations inside containment.

Reason for Change

The purpose of this change was to perform a LLRT for the containment penetration for the Service Air System when containment integrity was not required. This change proceduralized the task to allow future installation without a bypass jumper.

A continuous supply of Service Air was required for various loads inside containment during the refueling outage.

Safety Evaluation

This jumper was installed while containment integrity was not required.

Procedure Number	Title
SP 31103, Rev. 2	Integrated Leak Rate Test

Description of Change

The location of the verification test for the Integrated Leak Rate Test was moved from the containment penetration for the Service Air system to the personnel access airlock. Equalizing valves on the airlock doors were blocked open and tygon t ng was attached to the valves and run between the doors. This pro ded a flowpath for the verification test.

Reason for Change

During the previous test, the verification flow test rig had become clogged and the entire test had to be repeated. Selection of another location was required to avoid this problem.

Safety Evaluation

The personnel access airlock contains no safety-related equipment. Therefore, tube failure could not damage any equipment. Since containment was already pressurized, use of the airlock was already prohibited.

Number	Title
SP 3448E51, Rev. 1	Diesel Sequencer Train "A" Actuation Timer Test
SP 3448E52, Rev. 0	Diesel Sequencer Train "B" Actuation Timer Test

Description of Change

These procedures were changed to incorporate the use of a video camera to record the response time testing of both the Train "A" and Train "B" actuation signals.

Reason for Change

The former procedure required that the safety actuation signal input leads be lifted and attached to a data logger. Upon completion of the testing the potential for relanding these input leads on the wrong terminal existed. The revision of these procedures eliminated the need to lift leads on the input relays.

Safety Evaluation

By not requiring the input leads to be lifted and attached to the data logger, system reliability was increased.

Procedure Number	Title
SP 3446F31, Rev. 4	Solid State Protection System Refuel Test

Description of Change

This procedure was changed to incorporate mainboard annunciators which would provide the appropriate response for Solid State Protection System testing during refueling operations.

Reason for Change

The procedure was revised to include mainboard annunciators which were multiplexed from the Solid State Protection System which would be available to provide the appropriate response during testing. These mainboard annunciators would be lit during refueling operations.

Safety Evaluation

Revising this procedure to use the available annunciator windows did not impact the systems capabilities to mitigate the consequences of an accident.

Procedure Number	Title
SP 3442J11, Rev. 2 SP 3442J12, Rev. 1	Cold Overpressure Protection System (COPPS) Operational Test (Train A and B)

Description of Change

This new procedure created an overall test of the COPPS from a transmitter simulator through the logic up to but not including valve operation. The Power Operated Relief Valve (PORV) block valve was shut to prevent loss of reactor coolant and the output slave relay was blocked to prevent an open signal from reaching the valve.

Reason for Change

The test was performed to meet the frequency of a surveillance requirement. Prior to this procedure, the surveillance requirement was addressed by several separate tests which were scheduled at varying frequencies but did not satisfy the specified frequency.

Safety Evaluation

The new procedure to test COPPS involves blocking the slave relay output and shutting the PORV block valve to prevent loss of reactor coolant due to inadvertent opening of the PORV. The methods used in the test procedure are more operationally limiting in that the high pressurizer pressure relief function of the PORV is disabled. The PORV is not credited to provide any relief during at power over pressurization accidents and PORV isolation is allowed by Technical Specifications.

Number	Title
SP 3646A.8, Rev. 10, Ch. 3	Slave Relay Test - "A" Train
SP 3646A.9, Rev. 10, Ch. 4	Slave Relay Test - "B" Train
OPS Form 3646A.8-1, Rev. 7, Ch. 6	Slave Relay Test - "A" Train
OPS Form 3646A.9-1, Rev. 8, Ch. 2	Slave Relay Test - "B" Train

Description of Change

This change allowed for the energization of relays supplying control circuit power to four low pressure safety injection accumulator isolation valves. The thermal overload protection devices to the valve motors were removed prior to prevent inadvertent valve operation.

Reason for Change

This test evaluated the control circuits capability to supply power to the four accumulator isolation valves. This circuit had not been tested previously.

Safety Evaluation

The removal of the valves thermal overload devices ensured that required valve positioning was maintained throughout the testing evolution.

Number	Title
SP 3674.1, Rev. 3	Motor Operated Valve Thermal Overload Bypass Testing
OPS Form 3674.1-5, Rev. 0	Motor Operated Valve Thermal Overload Bypass Testing for Charging System Isolation Valve

Description of Change

This change added the Charging System isolation valve to the procedure for MOVs which cannot be tested during Engineered Safeguards Features testing.

The test method is consistent with that used for other MOVs.

Reason for Change

This circuit had not been tested previously.

Safety Evaluation

The thermal overload protection device was removed during the test which prevented the valve from inadvertently closing and stopping charging flow. The plant status was maintained as required by plant Technical Specifications.

Procedure Number	Title
EOP 3501, Rev. 1	Loss of All AC Power (Modes 5, 6, and 0)
EOP 3505, Rev. 4	Loss of Shutdown Cooling and/or Reactor Coolant System (RCS) Inventory
EOP 3505A, Rev. 0	Loss of Spent Fuel Pool Cooling
EOP ECA 0.0, Rev. 7	Loss of All AC Power
EOP 3503, Rev. 8	Shutdown Outside Control Room

Description of Change

EOP 3501 was revised to include the actions to take when refueling operations are in progress. This primarily concerns action to be taken if a fuel assembly is being moved when the loss of AC occurs.

EOP 3505 was revised to include response to a loss of RCS inventory while the plant is shutdown.

EOP 3505A was created to specifically handle a loss of spent fuel pool cooling.

EOP ECA 0.0 and EOP 3503 were revised to delete resetting of two relays. One locks out individual loads and the other locks out an entire bus after a fault is sensed on a bus.

Reason for Change

EOP 3501 and 3505 were changed to implement industry recommendations, update setpoints and improve the flow of the procedures.

EOP 3505A was created to address an issue as recommended by the industry.

EOP 35 ECA 0.0 and EOP 3503 were changed due to equipment and personnel safety concerns. The procedure previously allowed resetting the relays without investigation of why they had actuated.

Safety Evaluation

These procedures are utilized only after an accident has occurred; therefore, they can have no impact on the probability of an accident.

These procedures are not used in response to design basis accidents; therefore, do not impact the consequences of these accidents.

Procedure Number	Title
EOP 3501, Rev. 4	Loss of All AC Power (Modes 5, 6, and 0)

Description of Change

This EOP has been revised to:

- direct restoration of offsite power to emergency busses, and
- provide instructions and limitations for operation of the Station Blackout Diesel.

Reason for Change

Incorporate use of the Station Blackout Diesel.

Safety Evaluation

The loss of all AC power is beyond the design basis of the plant. In addition, this procedure is used only after the loss of all AC has happened.

The use of the Station Blackout Diesel mitigates the impact of this event.

Procedure Number	Title
EOP 35 E-1, Rev. 10	Loss of Reactor or Secondary Coolant
EOP 35 FR-C.1, Rev. 6	Response to Inadequate Core Cooling

Description of Change

EOP 35 E-1 was revised to

- reset all Engineered Safety Features signals when appropriate,
- allow shifting steps for checking hydrogen concentration and placing hydrogen recombiners in service,
- differentiate between power lockouts on two main control boards.

EOP 35 FR-C.1 was revised to

- direct continued attempts to establish a heat sink to at least one steam generator while performing the remainder of the procedure, and
- notify operator that the main steam line isolation signal is active when the low steam generator line pressure safety injection signal has been reset.

Reason for Change

Incorporate changes recommended by the Westinghouse owner's group and from the training department.

Safety Evaluation

The changes make the procedures clearer and more complete. The overall strategy in the procedures has not changed.

Procedure Number	Title
Spec. Proc. 91-3-010	Temporary Service Water to Turbine Plant Component Cooling Water Heat Exchanger

Description of Change

The procedure provided instructions to Operations for the operation of temporary Service Water to a Turbine Plant Component Cooling Water system heat exchanger during the recent refueling outage.

Reason for Change

Temporary cooling water was required during Refueling Outage 4 to maintain the Turbine Plant Component Cooling Water system operational to enable running various turbine support systems.

Safety Evaluation

The procedure was in effect only for the duration Bypass Jumper 3-93-144 was installed. The plant was in Mode 5 for the duration of this period. The procedure and bypass jumper affect only the Turbine Plant Component Cooling Water system, Vacuum Priming, Stator Cooling, Instrument Air and Circulating Water Systems. The procedure affects only the temporary configuration of the plant and will have no impact on any safety-related systems or equipment.

Procedure Number	Title
Spec. Proc. 93-3-006	Supplemental Leak Collection and Release System (SLCRS) Rebalancing

Description of Change

This procedure provided instructions to rebalance the SLCRS to achieve optimum system performance. The procedure provided instructions to adjust each of the four branch balancing dampers on the suction of the SLCRS fans. The adjustments were made in small increments so the system performance would not be significantly affected.

Reason for Change

The SLCRS was initially balanced in the fall of 1985. Since that time flow rates have fallen off each summer and then returned to higher flow rates each fall. This change was implemented to maximize system performance during the summer months when flow rates have dropped to Technical Specification limits.

Safety Evaluation

Constant system monitoring and small incremental adjustments ensured the SLCRS was at, or better, than as-found condition during the balancing. No other systems were impacted by this change.

Procedure Number	Title
Spec. Proc. 93-3-19,	Infrequently Performed Test Evolution for Releasing SIGMA Mast from Fuel Assembly F38
Spec. Proc. 93-3-23,	Release of SIGMA Gripper Mast from Fuel Assembly F38

Description of Change

These special procedures were written to provide guidance for manually disengaging the SIGMA refueling machine from fuel assembly F38 in the reactor vessel.

Reason for Change

These procedures were required due to the failure of one of the four gripper fingers on the SIGMA refueling machine. This failure resulted in the finger becoming detached from the gripper assembly, and engaging the top nozzle of the fuel assembly. The gripper finger could not be disengaged from the assembly using the normal controls.

Safety Evaluation

These procedures were written to provide guidance on the manual disengaging of the gripper finger from the fuel assembly. The technique for accomplishing this did not require lifting the fuel assembly. Therefore, the probability of a fuel handling accident was not increased.

The refueling machine was stuck on only one fuel assembly, and all assemblies adjacent to, and in the vicinity of, the affected assembly had already been removed from the reactor vessel. Even if the assembly were inadvertently lifted and dropped, the consequences would not be increased.

Procedure Number	Title
Spec. Proc. 93-3-039	Supplemental Leak Collection and Release System (SLCRS) Drawdown Only-Trains "A" and "B"

Description of Change

This procedure provided instructions to perform a drawdown test with only one SLCRS fan. The test is similar to a SLCRS drawdown surveillance test except that no other fans run during the test except the one SLCRS fan being tested.

Normal plant ventilation systems within the SLCRS boundary are shut down during the test. The only safety-related system shutdown is the Charging and Reactor Plant Cooling Water Area Ventilation System.

Reason for Change

The test provided results which can be compared to past test results to determine SLCRS boundary degradation.

Safety Evaluation

Based on previous test data the Charging Pump cubicle temperature does not change significantly without the ventilation system running. The affected area temperatures are continuously monitored by existing data systems. The test was performed with the plant in Mode 5. The safety-related equipment supported by ventilation systems affected by this test will remain operable.

Number	Title
Spec. Proc. 93-3-040	Low Pressure Turbine Pre-Warming Procedure

Description of Change

This special procedure was performed to pre-warm the low pressure turbines by degrading condenser back pressure until the turbine reached 25% rated electrical output.

Reason for Change

This special procedure pre-warmed the low pressure turbines to prevent small cracks in the "A" and "C" main turbine rotors from propagating. This was recommended by General Electric as a method of maintaining the reinspection frequency for these rotors. The cracks were found during the refueling outage.

Safety Evaluation

The safety evaluation evaluated the possibility of a turbine trip caused by the loss of condenser vacuum. The likelihood of this event occurring was determined to be very low. This is based on the precautions that were instituted during the performance of the procedure, to ensure that the proper condenser vacuum was maintained.

Procedure Number	Title
Spec. Proc 93-3-045	"D" Reactor Coolant Pump (RCP) Oil Addition to Operating Motor

Description of Change

This procedure provides instruction for adding oil to the upper motor bearing oil reservoir of the "D" RCP motor while the motor is operating.

Reason for Change

Monitoring of the oil levels in the RCP motors found that there was a gradual decrease in the upper bearing oil level in the "D" pump. This Special Procedure ensured that the oil reservoir is not overfilled and provides instructions to ensure the task is performed correctly and efficiently.

Safety Evaluation

The safety evaluation for this Special Procedure was covered in the safety evaluation of the Bypass Jumper 3-93-179.

Procedure Number	Title
SP 3613F.2-1, Rev. 3	Containment Purge and Exhaust Isolation Valve Operability Test
OPS Form 3273-3/4.3.6.3, Rev. 3	Containment Isolation Valve Table

Description of Change

The procedure was changed to reflect the change in the closure time of the Containment Purge and Exhaust System Containment Isolation Valves from 3.37 seconds back to 3.0 seconds.

Reason for Change

In a previous change to the surveillance procedure, the stroke time for the subject valves increased from 3.0 seconds to 3.37 seconds. A review of this change determined that while it was technically justified, it was not in agreement with the Final Safety Analysis Report table for Containment Isolation Valves. Therefore, this change was required to bring the surveillance into agreement with the FSAR.

Safety Evaluation

These valves are required to be closed in all modes of operation other than refueling. In the event of a fuel handling accident, the valves must be closed prior to any release of radiation occurring. A review of the closure time required for the valves showed that 3.89 seconds was available for valve closure time when instrument response time was taken into account. Since the new valve closure time is 3.0 seconds, the change was made in a conservative direction.

Procedure Number	Title
SP 3614I.3, Rev. 7	Supplemental Leak Collection and Release System Negative Pressure Verification
OPS Form 3614I.3-1 & 2, Rev. 4	Supplemental Leak Collection and Release System Negative Pressure Verification Forms

Description of Change

The surveillance procedure and forms were changed to reflect new criterion for negative pressure verification. The new criteria is 0.4 inches water gauge within 120 seconds measured at the ground floor of the Auxiliary Building. This change also incorporated a more conservative testing methodology which simulates a failure of the operating Charging Pump/Reactor Plant Component Cooling Water Pump (CH/RPCCW) ventilation train at the time of the event.

Reason for Change

This procedure revision incorporated changes in plant Technical Specifications. The change in test methodology was made to ensure system performance in worst case scenarios.

Safety Evaluation

The changes associated with the Technical Specifications were evaluated as part of the change process. The new methodology is incorporated by switching trains of CH/RPCCW ventilation in accordance with the operating procedure at time zero of the test. The changes do not adversely impact operation of any safety related system.

Procedure Number	Title
AOP 3555, Rev. 3	Reactor Coolant Leak

Description of Change

Possible leakage into the steam generators is evaluated prior to checking other less significant sources of Reactor Coolant System (RCS) leakage. The actions for a confirmed leak were modified to include reference to a radiation monitor abnormal operating procedure, require notification of Millstone Unit No. 1, consider shifting auxiliary steam to the auxiliary boilers, and consult with the Duty Officer regarding power reduction or shutdown.

Reason for Change

Use of the procedure in simulator sessions indicated that steps related to identifying primary leakage into a steam generator needed to be re-organized.

Changes to the actions for a confirmed primary leak into the steam generators was made to address generic regulatory concerns.

Safety Evaluation

The changes in sequence of some of the procedure steps facilitates the execution of these steps. This reduces the likelihood of an accident or malfunction.

The changes in response to a confirmed leak only come into play after a problem and can only reduce the impact of the event.

Procedure Number	Title
AOP 3554, Rev. 4	Reactor Coolant Pump (RCP) Trip or Stopping an RCP at Power
AOP 3559, Rev. 3	Loss of Condenser Vacuum
AOP 3575, Rev. 0	Rapid Downpower

Description of Change

AOP 3554 was revised to include transitions to the rapid downpower adnormal operating procedure. This was an administrative change.

AOP 3559 was revised to trip the reactor instead of the turbine if no steam dumps are available. In addition, two administrative changes were made to include transitions to the rapid downpower abnormal operating procedure and to consult with the Duty Officer if a certain plant response is not observed.

AOP 3575 was written to provide detailed instructions for performing a rapid downpower.

Reason for Change

A normal operating procedure had been used for conducting downpower maneuvers. Certain conditions required rapid operator response; therefore, they were more suited to the abnormal operating procedure format. Therefore, AOP 3575 was created to deal with this concern.

AOP 3559 trip criteria was modified to reduce the heat load on the steam generators after a trip when the steam dumps were unavailable. By tripping the reactor first, the heat input is reduced before the turbine trips. This reduces the likelihood that the steam generator safety valves will lift.

Safety Evaluation

In relation to the old method of conducting a downpower, the new rapid downpower procedure decreases the probability of damaging equipment during an abnormal condition. Therefore, no new malfunctions are introduced.

The revision to the loss of condenser procedure reduces the likelihood of challenging the main steam safeties.

Procedure Number	Title
No Number	Safety Evaluation for the Leak Repair of an Auxiliary Feedwater System Check Valve

Description of Repair

This valve is a 4" 900# tilting disc check valve made by Anchor/Darling. The valve uses a tapered, pressure seal gasket to seal the body to bonnet joint. The repair consisted of tightening the bonnet studs per an approved procedure.

Reason for Repair

The body to bonnet connection was leaking when first subjected to main feedwater pressure during startup from the recent refueling outage.

Safety Evaluation

The repair process was performed following standard practices for torquing of bolts. The designated industry standard safety factor was maintained so the probability of equipment failure was not increased. This repair had no affect on the ability of the Auxiliary Feedwater system to mitigate the consequences of a previously evaluated accident.

Procedure Number	Title
No Number	Safety Evaluation for the Leak Repair of Three Auxiliary Feedwater Check Valves

Description of Repair

The Auxiliary Feedwater to "D" steam generator check valve is a 4" 900# tilting disc check valve made by Pacific Valve Incorporated. The motor driven Auxiliary Feedwater pump discharge check valve and the "B" Turbine Drive Auxiliary Feedwater pump to "B" steam generator check valve are 6" and 3" (respectively) tilting disc check valves made by Anchor/Darling. These valves use a tapered, pressure seal gasket to seal the body to bonnet joint. The repairs consisted of tightening the bonnet studs per approved procedures.

Reason for Repair

The body to bonnet connections were leaking when first subjected to main feedwater pressure during startup from the recent refueling outage.

Safety Evaluation

Each repair process was performed following standard practices for torquing of bolts. The designated industry standard safety factor was maintained so the probability of equipment failure was not increased. The repairs were performed one-at-a-time to preclude the potential for multiple failures of Auxiliary Feedwater system equipment. These repairs had no affect on the ability of the Auxiliary Feedwater system to mitigate the consequences of a previously evaluated accident.

Procedure Number	Title

Procedure Number	Title
No Number	Safety Evaluation for the Tightening of the Bonnet Clamp of 3FWA*MOV35A

Description of Repair

The valve bonnet clamp on the outboard containment isolation valve for the "A" Motor Driven Auxiliary Feedwater Pump was tightened per an approved procedure. The tightening of the bonnet clamp pulled up on the pressure seal, thereby sealing off the leak.

Reason for Repair

The valve was leaking at the body to bonnet interface.

Safety Evaluation

The valve repair tightened the bonnet clamp. The act of tightening the valve bonnet clamp did not significantly increase the potential for valve failure.

Procedure Number	Title
EOP 35 E-3, Rev. 10	Steam Generator Tube Rupture
EOP 35 ES-1.2, Rev. 7	Post LOCA Cooldown and Depressurization
EOP 35 ES-1.4, Rev. 6	Transfer to Hot Leg Recirculation
EOP 35 ES-3.2, Rev. 6	Post Steam Generator Tube Rupture Cooldown Using Steam Blowdown
EOP 35 ES-3.3, Rev. 7	Post Steam Generator Tube Rupture Cooldown Using Steam Dump
EOP 35 FR-H.1, Rev. 7	Response to Loss of Secondary Heat Sink
EOP 35 FR-P.1, Rev. 7	Response to Imminent Pressurized Thermal Shock Condition
EOP 35 FR-P.2, Rev. 6	Response to Anticipated Pressurized Thermal Shock Condition
EOP 35 FR-S.1, Rev. 8	Response to Nuclear Power Generation/ATWS
EOP 35 FR-Z.1, Rev. 7	Response to High Containment Pressure
EOP 3501, Rev. 7	Loss of All AC Power (Modes 5, 6 and Zero)
EOP 35 ECA-0.0, Rev. 9	Loss of All AC Power
EOP 35 ECA-0.1, Rev. 7	Loss of All AC Power - Recovery without Safety Injection (SI) Required
EOP 35 ECA-0.2, Rev. 6	Loss of All AC Power - Recovery with SI Required
EOP 35 ECA-0.3, Rev. 0	Loss of All AC Power - Recovery with the Station Blackout (SBO) Diesel
EOP 35 ECA-1.1, Rev. 6	Loss of Emergency Coolant Recirculation
EOP 35 ECA-3.1, Rev. 9	Steam Generator Tube Rupture with Loss of Reactor Coolant - Subcooled Recovery Desired
EOP 35 ECA-3.2, Rev. 6	Steam Generator Tube Rupture with Loss of Reactor Coolant - Saturated Recovery Desired
EOP 35 ECA-3.3, Rev. 7	Steam Generator Tube Rupture without Pressurizer Pressure Control

Description of Change

The changes are summarized below:

*A step is added or changed so that the Reactor Coolant Pump (RCP) overcurrent trip is set to COLD setting to allow continued operation of the RCP during Reactor Coolant System (RCS) cooldown.

*A check is added for hydrogen concentration following breaks in the containment with/without guidance for the operation of recombiners and containment purge.

*Allowance for filling the Demineralized Water Storage Tank (DWST) from the Ecolochem Trailer is made, thus providing an additional source of demineralized water during RCS cooldown and ensuring a secondary heat sink.

*A note to identify that main steam isolation will occur on high steam pressure rate setpoint after blocking the low steam line pressure safety injection so that consequences of the event are not adversely affected.

*A caution to prohibit the first RCP being started in the ruptured loop after natural circulation cooldown, to prevent inadvertent criticality and ensure continued core cooling.

*A step to reset safety injection if necessary, so that later actions can be carried out.

*A step to restore power to the communication console following Station Blackout to facilitate mitigation of the event.

*Additional assurances in the form of cautions, notes or steps are made so that proper action or transition to the proper procedure is made if a safety injection signal is generated during an evolution.

*Additional checks of Refueling Water Storage Tank level for the switchover to recirculation and of DWST level for the Auxiliary Feedwater (AFW) switchover to the CST so that operability of the pumps is not affected.

*Clarification that loss of heat sink is based or pressurizer Power Operated Relief Valve (PORV) opening pressure in conjunction with increasing of core exit temperature and verification that the RCPs have stopped for bleed and feed. This ensures that the proper symptom is identified, the operator is not mislead, and heat input to the RCS from the RCPs is eliminated.

*Change of the minimum/maximum steam generator level during RCS cooldown is made to avoid AFW actuation/isolation setpoint while cooling down the ruptured steam generator.

Description of Change (Continued)

The changes are summarized below:

*Caution to ensure that feed flow is isolated to ruptured steam generator following steam generator tube rupture and clarification that the ruptured SG level indicates off scale high are added to further assure that safety analysis assumptions are not violated.

*Modifications are made to feed no more than 100 gpm to only one SG when all SGs are dry following loss of secondary heat sink. This is done to minimize damage to the SGs due to thermal shock while allowing one SG to be refilled in a safe manner.

*Verification that the pressurizer PORV block valve is open is made for the RCS bleed path. This is done to ensure success of bleed and feed operation.

*Allowance is made for initiating the pressurizer auxiliary sprays if the PORV fails to open so that pressurization of the RCS is stopped and depressurization initiated.

Reason for Change

Incorporate changes recommended by Westinghouse or improvements suggested by Training and Operations personnel. Incorporate use of the station blackout diesel in response to loss of all AC power events.

Safety Evaluation

Only EOP ES-1.2, ES-1.4, ES-3.2, and ES-3.3 are used in response to design basis accidents. All others are used in response to beyond design basis events.

The overall strategy of the four procedures which are used to mitigate design basis accidents has not changed. The changes improved the verification of automatic functions, the verification of the occurrence of a malfunction, or the mitigation of the consequences of a malfunction.

Procedure Number | Title

Vendor Procedure FP-NEU-1, Rev. 0 | Fuel Reconstitution

Description of Change

This procedure provides guidance and direction for the disassembly and reassembly of a top nozzle reconstitutable fuel assembly and visual inspection of individual fuel rods.

Reason for Change

Industry experience with Westinghouse VANTAGE 5H fuel indicated a potential for flow induced vibration of the fuel assembly leading to grid-to-rod fretting. During the 1991 refueling outage, it was decided to disassemble two VANTAGE 5H fuel assemblies to inspect for signs of fretting wear on susceptible rods. This procedure provided the requisite instructions for performing this work.

Safety Evaluation

Fuel assembly disassembly, fuel rod removal and reinsertion and assembly reassembly were performed with the fuel assembly in the New Fuel Elevator. The physical arrangement of the elevator, coupled with tool design and administrative controls, ensured at least 10.5' of water were maintained over the fuel assembly and/or remove rod. Therefore, there was no adverse affect on radiation levels in and around the Spent Fuel Pool.

The assembly was transported to and from the elevator using the normal Spent Fuel Handling Tool, therefore, this evolution did not increase the probability of a design basis fuel handling accident, nor create the possibility of a new type of accident. Handling of an individual fuel rod does not create the possibility of a new type of accident due to its very small mass and radionuclide inventory compared to an entire fuel assembly.

Since only one fuel assembly was out of its normal storage position at any given time during this process, the consequences of a design basis fuel handling accident, had it occurred, would not have been affected.

JUMPERS-LIFTED LEADS-BYPASSES

Jumper-Lifted Lead-Bypass	Title
3-93-011	Temporary System for Collecting and Pumping the Turbine Building Floor Drains
3-93-014	Install Spool Piece for Repair of Engineered Safety Features (ESF) Ventilation Valcor Pressure Regulating Valve
3-93-025	Install Mechanical Gag on Main Steam Safety
3-93-041	Blind Flanges on Outlet of Main Steam Safety to Maintain Supplemental Leak Collection and Release System (SLCRS) Boundary Integrity
3-93-042	Installation of Blank Flange on Inlet to a Main Steam Safety Valve
3-93-043	Installation of Data Acquisition System in the Electro-Hydraulic Control (EHC) System
3-93-060	Gagging Diesel Enclosure Motor Operated Damper to the Full Open Position During Repair
3-93-062	Installation of a Cap Downstream of Safety Injection Accumulator Containment Isolation Valve Sample Line
3-93-067	Reactor Coolant Pump Temporary Cooling System
3-93-070	Setpoint Change of Service Water Pump Lubrication Water Pressure
3-93-071	Conversion of Hypochlorite System from Educator to Metering Pump Supply
3-93-072	Temporary Jumper to Blow Condensate Flow Transmitter Instrument Lines into the Condensate System from Feedwater
3-93-074	Bypass Sensing Line of Condensate Flow Transmitter

JUMPERS-LIFTED LEADS-BYPASSES (CONTINUED)

Jumper-Lifted Lead-Bypass	Title
3-93-078	Replace Cation Resins in Cesium Removal Ion Exchangers with Mixed Resins
3-93-083	Temporary Fire Protection to a Temporary Trailer
3-93-085	Installation of Temporary Instrument Air Compressor
3-93-093	Temporary Vacuum Priming for Two Circulating Water Pumps
3-93-094	Repair of Instrument Air Line in Auxiliary Building
3-93-095	Temporary Vacuum Priming for Two Circulating Water Pumps
3-93-104	Block Open "B" Train Containment Purge and Exhaust Valves during "A" Train Electrical Outage
3-93-111	Block Open Auxiliary Feedwater Cross Connect Valve
3-93-112	Head Vent Piping Flush
3-93-114	Block Open Outside Containment Penetration for Instrument Air System
3-93-122	Lifted Leads from Volume Control Tank Level Transmitters
3-93-124	Control Building Chiller Resistance Temperature Detector (RTD) Replaced
3-93-125	Temporary Shielding - Reactor Cavity Flange Area to Reduce Radiation Levels During Mode 0 Work in Cavity
3-93-133	Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Ventilation System Temporary Ductwork
3-93-137	Blind Flange Over "A" Steam Generator Supply to Turbine Drive Auxiliary Feedwater Pump Valve Body

JUMPERS-LIFTED LEADS-BYPASSES (CONTINUED)

Jumper-Lifted Lead-Bypass	Title
3-93-138	Containment Video System
3-93-144	Operation of Turbine Plant Component Cooling Water (TPCCW) System with a Limited Heat Sink
3-93-145	Installation of a Temporary Blank Flange on "B" Train of Service Water
3-93-150	Installation of a Freeze Seal on the Service Water Return Piping of the "A" Charging Pump Cooling Heat Exchanger
3-93-151	Installation of a Freeze Seal on the Service Water Return Piping of the "B" Charging Pump Cooling Heat Exchanger
3-93-152	Installation of Temperature Detectors on Reactor Plant Component Cooling Water Lines
3-93-155	Enclosure Wall in Fuel Building Railroad Access
3-93-164	Train "A" Reactor Vessel Level Indication System Heater No. 8 Bypassed
3-93-165	Install Blind Flange in Place of Reactor Plant Gaseous Drain Relief Valve
3-93-167	Temporary Restraint for Jib Crane
3-93-170	Direct Atmospheric Pressure for Supplemental Leak Collection and Release System Testing
3-93-171	Plywood Shield For "A" Quench Spray Pump
3-93-172	Reverse Lifted Lead Termination on Rod C11 Lift Coil
3-93-173	Bypass of Low Pressure/Low Flow Output to Hydrogen Analyzer Common Trouble Alarm

JUMPERS-LIFTED LEADS-BYPASS (CONTINUED)

Jumper-Lifted Lead-Bypass	Title
3-93-174	Temporary Shielding - Auxiliary Building Boronometer Cubicle
3-93-179	Addition of Oil Add connection to "D" Reactor Coolant Pump (RCP)
3-93-184	Temporary Drain Lines for "B" and "D" Steam Supplies to the Turbine Driven Auxiliary Feedwater Pump (TDAFW)

Jumper-Lifted Lead-Bypass Number 3-93-011

This jumper, entitled "Temporary System for Collecting and Pumping the Turbine Building Floor Drains," has been removed.

Description of Jumper-Lifted Lead-Bypass

This bypass jumper provided a temporary means of collecting and pumping the Turbine Building floor drains while cleaning was performed on an oil/water separator. The jumper has been removed.

Reason for Jumper-Lifter Lead-Bypass

An alternative method of collecting and pumping the Turbine Building floor drains was required while performing preventive maintenance on the oil/water separator.

Safety Evaluation

This evaluation addressed the radiological aspects of bypassing the normal radioactive liquid effluent monitoring instrumentation and determined that with the short duration, sampling requirements, and secondary coolant activity level there was a minimal affect on analyzed accidents.

Jumper-Lifted Lead-Bypass Number 3-93-014

This jumper entitled, "Install Spool Piece for Repair of Engineered Safety Features (ESF) Ventilation Valcor Pressure Regulating Valve," is installed.

Description of Jumper-Lifted Lead-Bypass

A spool piece consisting of a length of pipe, flanges, a blank, and a lead sheet was installed in place of a Valcor Pressure Regulating Valve, which was removed for development of a repair kit. The pipe was blanked to duplicate the flow characteristics of the existing failed shut valve. The lead sheet was added to duplicate the seismic attributes of the existing installation. The bypass jumper is presently installed.

Reason for Jumper-Lifted Bypass

The Valcor valves for the ESF Building air conditioning units were incorrectly designed and have been bypassed via a separate bypass jumper. This change removed a valve from one of the four ESF Building air conditioning units for shipment to the vendor and development of a repair kit. This repair kit will be developed for all eight Valcor valves (two per air conditioning unit). When installed, it will return the air conditioning units to their intended design and allow for the removal of the bypass jumper.

Safety Evaluation

The safety evaluation addressed the Service Water Flow requirements for the ESF Building air conditioning unit and the seismic attributes of the installed spool piece. The blanked spool piece duplicated the Service Water flow characteristics of the failed shut Valcor valve. The seismic attributes of the system were maintained by adding mass (lead sheet) to the spoolpiece and ensuring that the weight was comparable to the removed valve.

Jumper-Lifted Lead-Bypass Number 3-93-025

This jumper, entitled "Install Mechanical Gag on Main Steam Safety," has bee removed.

Description of Jumper-Lifted Lead-Bypass

A mechanical gag was temporarily installed on one of the Main Steam Safety valves for the "D" steam generator. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The installation of the gag on this relief was necessary to stop an ongoing steam release following the March 31, 1993, plant trip. Following the trip, the safety valve did not completely reseat after lifting. Installing a gag allowed the operators to better control the plant cooldown rate.

Safety Evaluation

This jumper was evaluated for the potential impact on a main steam line overpressurization with subsequent rupture. The probability of a main steam line overpressurization was determined to be very low. This was based on the plant being cooled down with reduced stored energy available to challenge the system. The other relief valves were available to mitigate an overpressurization situation.

Jumper-Lifted Lead-Bypass Change Number 3-93-041

This jumper, entitled "Blind Flanges on Outlet of Main Steam Safety to Maintain Supplemental Leak Collection and Release System (SLCRS) Boundary Integrity," has been removed.

Description of Jumper-Lifted Lead-Bypass

This jumper installed blind flanges on the relief valve outlet pipes for the Main Steam Safety valve for the "D" Steam Generator. The flanges were installed while the valve was removed for maintenance. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The jumper was required to maintain the integrity of the SLCRS boundary while the safety valve was removed for maintenance and the plant was not in Mode 5 or 6.

Safety Evaluation

There were no new malfunctions or accidents created by this change. The change was necessary to ensure integrity of the SLCRS boundary while the safety valve was removed for maintenance.

Jumper-Lifted Lead-Bypass Number 3-93-042

This jumper, entitled "Installation of Blank Flange on Inlet to a Main Steam Safety Valve," has been removed.

Description of Jumper-Lifted Lead-Bypass

A blank flange on the inlet to a main steam safety valve for the "D" steam generator was installed while the valve was removed. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The safety valve was removed for repair. Flange installation was required to satisfy requirements for containment isolation while the valve was removed.

Safety Evaluation

The safety evaluation considered the temporary condition when a blank was installed on the inlet to one main steam safety valve while the plant was shutdown. The flange was analyzed to verify that it would withstand the maximum Main Steam header pressure possible while shutdown and satisfy the required pressure for containment pressurization.

Jumper-Lifted Lead-Bypass Number 3-93-043

This jumper, entitled "Installation of Data Acquisition System in the Electro-Hydraulic Control (EHC) System," has been removed.

Description of Jumper-Lifted Lead-Bypass

A data acquisition system was installed in the EHC system electronics cabinet. This data acquisition system periodically scanned thirty EHC system parameters at 200 millisecond intervals. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The data acquisition system gathered data from thirty EHC parameters to assist in any future post transient analysis, relating to the EHC system.

Safety Evaluation

This jumper was evaluated for the potential impact on a turbine trip being generated by an input from the data acquisition system via its connections to the EHC system. The probability of a turbine trip initiated from the data acquisition system connected to the EHC system was determined to be very low. This was based on the precautions taken with the connection cable characteristics, such as minimization of length, shielded cable, and low impedance.

Jumper-Lifted Lead-Bypass Number 3-93-060

This jumper, entitled "Gagging Diesel Enclosure Motor Operated Damper to the Full Open Position During Repair," has been removed.

Description of Jumper-Lifted Lead-Bypass

The inlet motor operated damper for the "B" Diesel Enclosure was mechanically locked to the full open position during the repair of the damper's actuator. The locking device was a seismically evaluated device previously used in similar applications. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The actuator for the inlet damper for the "B" Diesel Enclosure had to be removed for repair. During this time frame the damper had to be held in the full open position to allow for maximum ventilation for the "B" Diesel Enclosure area. A locking device was used to mechanically lock the damper open.

Safety Evaluation

The safety evaluation addressed the ventilation requirements for operating The "B" Emergency Diesel Generator and the seismic consideration for the locking device used to lock the damper open. The full open position of the damper allowed for maximum cooling of the diesel enclosure during the warm weather time frame when the jumper was installed and provided sufficient ventilation to maintain the operability of the diesel. The locking device was evaluated as being seismically qualified for this application.

Jumper-Lifted Lead-Bypass Number 3-93-062

This jumper, entitled "Installation of a Cap Downstream of Safety Injection Accumulator Containment Isolation Valve Sample Line," has been removed.

Description of Jumper-Lifted Lead-Bypass

This jumper approved the installation of a blind flange on the Safety Injection Accumulator Sample line downstream of the containment isolation valve. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The containment isolation valve was removed for repair and the blind flange provided the required containment integrity.

Safety Evaluation

The flange's pressure and temperature ratings exceed those of the sample line.

Jumper-Lifted Lead-Bypass Number 3-93-067

This jumper, entitled "Reactor Coolant Pump Temporary Cooling System," has been removed.

Description of Jumper-Lifted Lead-Bypass

Cooling water jackets were installed on the common seal injection line in the Auxiliary Building. A chiller was installed outside of the Auxiliary Building with cooling lines running through an abandoned Supplemental Leak Collection and Release System (SLCRS) penetration. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The seal leak off from the Reactor Coolant Pumps had been increasing slowly during the last portion of the fuel cycle. In order to reduce leak off, cooler water was supplied to the pump seals.

Safety Evaluation

This jumper was evaluated for potential impact on the SLCRS boundary during a seismic event. The likelihood of cooling line rupture on both sides of the penetration was determined to be very low.

Jumper-Lifted Lead-Bypass Number 3-93-070

This jumper, entitled "Setpoint Change of Service Water Pump Lubrication Water Pressure," has been removed.

Description of Jumper-Lifted Lead-Bypass

The bypass jumper changed the setpoint of the low lubricating water pressure for the "A" Service Water pump from 23 psig to 10 psig. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The change was required to allow the continued operation of the "A" Service Water pump with reduced lube water pressure without compromising the ability of Operations to maintain vigilance over a potential loss of lube water to the pump.

Safety Evaluation

The setpoint change did not degrade pump performance nor did it affect operator response to low lube water flow. The actuation point for the pressure switch did change; however, this change remained invisible to the operator and the circuitry actuated to inform the operator of any degraded condition.

Jumper-Lifted Lead-Bypass Number 3-93-071

This jumper, entitled "Conversion of Hypochlorite System from Educator to Metering Pump Supply," is installed.

Description of Jumper-Lifted Lead-Bypass

The removal of the existing educator, associated components, installation of metering pumps, flow meters and piping was required to return system to a metered pump configuration.

Four existing flow elements were removed and replaced by the originally installed 1-1/2" flow elements. "V" groove ball assemblies were removed and replaced with full port ball assemblies in four valves. The jumper will remain installed until permanent design changes are completed.

Reason for Jumper-Lifted Lead Bypass

The recently installed educator based sodium hypochlorite was found to be incapable of providing required flow. Therefore, the old metered pump design was re-installed. It resulted in a higher mixed flow rate at the suction bell area of the Service Water pumps. The increased flow forces the flow toward the pump suction to help assure treated water enters the pump.

Safety Evaluation

The hypochlorite system is not safety-related. However, the loss of hypochlorite flow could eventually result in failure to chlorinate and possible fouling of the Service Water components. This change improved the operation of the system.

Jumper-Lifted Lead-Bypass Number 3-93-072

This jumper, entitled "Temporary Jumper to Blow Condensate Flow Transmitter Instrument Lines into the Condensate System from Feedwater," has been removed

Description of Jumper-Lifted Lead-Bypass

This jumper allowed the connection of tubing to the first point feedwater heater bypass line. This tubing was then sequentially connected to the high pressure and low pressure instrument lines for a condensate flow transmitter to blow down the piping into the Condensate System. The transmitter provides input to the "B" Turbine Driven Auxiliary Feedwater pump recirculation flow control valve. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This jumper provided a temporary source of high pressure water to blowdown the condensate instrument lines which were providing erratic indication.

Safety Evaluation

This evaluation addressed the impact on previously evaluated feedwater accidents. In addition, it addressed the implications of a failure of the jumper while it was installed and the consequences of a flow path bypassing the feedwater pump.

Jumper-Lifted Lead-Bypass Number 3-93-074

This jumper, entitled "Bypass Sensing Line of Condensate Flow Transmitter," has been removed.

Description of Jumper-Lifted Lead-Bypass

This jumper bypassed lines between the test connections on the instrumentation block for the normal "B" train condensate flow transmitter and the backup transmitter. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The capability for automatic operation of the feedwater circulation valve for the "B" Main Feed pump was restored by this jumper. The valve had been maintained in manual due to pressure fluctuations in the low pressure line to the normal flow transmitter.

Safety Evaluation

The safety evaluation addressed the automatic operation of the feedwater recirculation valve and the potential failure of jumper components. No new malfunctions were created by the installation of the bypass jumper. All jumper components used were in accordance with applicable specifications.

Jumper-Lifted Lead-Bypass Change Number 3-93-078

This jumper, entitled "Replace Cation Resins in Cesium Removal Ion Exchangers with Mixed Resins," is installed.

Description of Jumper-Lifted Lead-Bypass

A mixed bed resin was placed in the Cesium Removal Ion Exchangers in place of the cation bed resins. This jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

The reason for the jumper is to reduce the concentration of chloride ions in the boron recovery tanks. Use of a mixed bed resin in place of the normal cation bed will allow removal of fission products, as designed, and also remove chloride ions. High chloride concentrations have required the discharging of the contents of the boric acid storage tank which increases the overall amount of rad waste for the unit.

Safety Evaluation

The bypass jumper is required as the Final Safety Analysis Report states that a cation removal resin be installed in these ion exchangers. By changing the type of resin in the demineralizer, there is no change in the function of the demineralizer. The boron recovery system is not safety related. The failures of concern are out leakage and system failure to function.

The installation of mixed bed resin has no affect on the overall operation of the demineralizer. The change allows the boron recovery system to operate in an optimum configuration for the present plant conditions. The boron recovery system is not used to mitigate the consequences of any accident.

Jumper-Lifted Lead-Bypass Number 3-93-083

This jumper, entitled "Temporary Fire Protection to a Temporary Trailer," has been removed.

Description of Jumper-Lifted Lead-Bypass

Temporary fire water was connected to an outside break trailer for turbine outage personnel during the recent refueling outage. A wye connection was installed to a fire department connection in the Turbine Building at a hose rack location. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The fire water was installed to temporarily provide fire protection to the temporary break trailer for approximately three months.

Safety Evaluation

This jumper was evaluated for the potential impact on fire protection water flow demand in the Turbine Building with the temporary fire protection connected to the trailer. The impact of connecting fire protection water to the trailer was determined to have minimal impact on the fire protection water flow capacity in the area of concern of the Turbine Building.

Jumper-Lifted Lead-Bypass Number 3-93-085

This jumper, entitled "Installation of Temporary Instrument Air Compressor," has been removed

Description of Jumper-Lifted Lead-Bypass

A temporary Instrument Air compressor was installed to supplement loads while cooling to normal Instrument Air compressors was limited by high cooling water temperature. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

High cooling water temperatures limited capacity of normal Instrument Air compressors.

Safety Evaluation

The Instrument Air System is not safety related. The safety evaluation considered plant design maximum air loads, failure of the temporary compressor, and loss of Instrument Air.

Instrument Air components were designed to fail to their safety position on a loss of Instrument Air. All jumper components met plant Instrument Air System pressure requirements. Failure of the temporary compressor would not cause a loss of all Instrument Air. Service Air automatically provides a backup source of Instrument Air in the event the temporary and normal compressors were lost.

Jumper-Lifted Lead-Bypass Number 3-93-093

This jumper, entitled "Temporary Vacuum Priming For Two Circulating Water Pumps," has been removed.

Description of Jumper-Lifted Lead-Bypass

The bypass jumper installed a temporary configuration of the station Vacuum Priming System to enable the running of the "D" and "F" Circulating Water pumps. A temporary hose was run between the inlet flange of the "D" waterbox and the "E" waterbox isolation to the Vacuum Priming tank. Additionally, pancakes were installed at the inlet and outlet flanges of the "C" waterbox, the outlet flange of the "E" waterbox, and the north tank nozzles of the Vacuum Priming tank. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

Two Circulating Water pumps were needed to allow discharging of low level radwaste during the recent refueling outage.

Safety Evaluation

The Circulating Water system is not safety-related. Due to the plant mode in which the jumper was installed, no evaluated accidents apply. Also, no new accidents were created.

Jumper-Lifted Lead-Bypass Number 3-93-094

This jumper, entitled "Repair of Instrument Air Line in Auxiliary Building," is complete.

Description of Jumper-Lifted Lead-Bypass

A temporary hose with tee connections was installed between a spare valve and six points in the Instrument Air header within the Auxiliary Building. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This jumper provided an alternate Instrument Air supply while a section of the system was isolated for repair.

Safety Evaluation

The Instrument Air system is not safety related. All components fail to a safe position if Instrument Air is lost.

Jumper-Lifted Lead-Bypass Number 3-93-095

This jumper, entitled "Temporary Vacuum Priming for Two Circulating Water Pumps," has ben removed.

Description of Jumper-Lifted Lead-Bypass

The bypass jumper installed a temporary configuration of the station Vacuum Priming System to enable the running of the "A" and "C" Circulating Water pumps. A temporary hose was run between the outlet flange of the "C" waterbox and the "B" waterbox and isolation to the north standpipe loop. Additionally, pancakes were installed at the inlet and outlet flanges of the "D" waterbox, the south tank nozzles of the Vacuum Priming tank. A blind flange was installed at the "B" outlet flange. This bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

Two Circulating Water pumps were needed to allow discharging of low level radwaste during the recent refueling outage.

Safety Evaluation

The Circulating Water system is not safety-related. —due to the plant mode in which the jumper was installed, no evaluated accidents apply. Also, no new accidents were created.

Jumper-Lifted Lead-Bypass Number 3-93-104

This jumper, entitled "Block Open "B" Train Containment Purge and Exhaust Valves during "A" Train Electrical Outage," has been removed.

Description of Jumper-Lifted Lead-Bypass

The "B" train Containment Purge and Exhaust Valves were blocked open during the "A" train electrical outage. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

A ventilation path for containment was required during the refueling outage. When the containment radiation monitor is de-energized, all four Containment Purge and Exhaust Valves fail closed. This would block the ventilation path. Since the radiation monitor is powered from the "A" train, the "B" train valves were blocked open.

Safety Evaluation

The jumper was installed during a period when no core alterations were in progress. The Containment Purge and Exhaust System was demonstrated to be operable within 100 hours of fuel movement as required by Technical Specifications.

Jumper-Lifted Lead-Bypass Change Number 3-93-111

This jumper, entitled "Block Open Auxiliary Feedwater Cross Connect Valve," has been removed.

Description of Jumper-Lifted Lead-Bypass

This jumper blocked open the Motor Driven Auxiliary Feedwater (MDAFW) pump discharge cross-connect valve to ensure that the "B" MDAFW pump would be available to supply steam generators on either the "A" or "B" trains while the plant was shutdown. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

This jumper ensured that a MDAFW pump was available to supply water to two steam generators as required by shutdown risk management. One operable steam generator was on the "A" train and the other operable steam generator was on the "B" train. The bypass jumper was installed to provide positive assurance of auxiliary feedwater availability during breaker testing.

Safety Evaluation

The jumper provided additional assurance that auxiliary feedwater would be available to two steam generators as required by shutdown risk management.

Jumper-Lifted Lead-Bypass Number 3-93-112

This jumper, entitled "Head Vent Piping Flush," has ben removed.

Description of Jumper-Lifted Lead-Bypass

This jumper installed a pump, hoses, and fittings to the high point of the reactor vessel head vent piping. This equipment was then used to back flush the head vent piping. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The head vent piping had very high levels of contamination. Flushing the piping back to the reactor vessel and to the Pressurizer Relief Tank significantly reduced the radiation levels.

Safety Evaluation

The flushing evolution was performed while the plant was shut down with one residual heat removal pump in operation. Flushing was also limited to under 550 gallons. Under these conditions, an evaluation concluded that dilution was minimal and did not significantly affect shutdown margin. The restrictions also assured no overpressurization would occur, and any debris which might be in the line flushed to the vessel would not significantly add to the present concentration, and would be removed by letdown system filters.

Jumper-Lifted Lead-Bypass Number 3-93-114

This jumper, entitled "Block Open Outside Containment Penetration for Instrument Air System," has been removed

Description of Jumper-Lifted Lead-Bypass

The outside containment penetration isolation valve for the Instrument Air System was blocked open during inspection of the switchgear which supplies power to the valve actuator. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

Supply of Instrument Air to loads inside containment is needed for normal operation during refueling outage.

Safety Evaluation

The jumper was installed while the plant was in a condition that does not require full containment isolation capability. In addition, no direct path from inside containment to outside atmosphere was created by blocking the valve open because the Instrument Air System was intact.

Jumper-Lifted Bypass-Jumper Number 3-93-122

This jumper, entitled "Lifted Leads from Volume Control Tank Level Transmitters," has been removed.

Description of Jumper-Lifted Lead-Bypass

This change modified an electrical circuit to prevent the reactor cavity from filling via the Refueling Water Storage Tank (RWST) during reactor refueling maintenance activities. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

Isolating the RWST supply to the reactor cavity during refueling sequence was required for maintenance activities to the Reactor Coolant System volume control tank.

Safety Evaluation

The system removed from service was not required during the refueling activity.

Jumper-Lifted Lead-Bypass Number 3-93-124

This jumper, entitled "Control Building Chiller Resistance Temperature Detector (RTD) Replaced," has been removed.

Description of Jumper-Lifted Lead-Bypass

An 82 Ohm resistor was installed across the bearing temperature RTD for the "A" Control Building Chiller. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The chiller was tripping off due to erroneous output from this circuit. The installation of this jumper enabled the chiller to operate during the "B" train outage.

Safety Evaluation

An assessment determined that operation of the chiller without the bearing temperature trip was satisfactory until the bearing RTD was repaired and that there was no increased probability of chiller failure with the bypass jumper installed.

Jumper-Lifted Lead-Bypass Number 3-93-125

This jumper, entitled "Temporary Shielding - Reactor Cavity Flange Area to Reduce Radiation Levels During Mode 0 Work in Cavity," has been removed.

Description of Jumper-Lifted Lead-Bypass

During refueling operations, 2000 pounds of temporary shielding were installed on the reactor core barrel flange against the upper internals shield. The jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The use of the temporary shielding reduced the radiation exposure received by personnel working on the reactor vessel flange assembly.

Safety Evaluation

The safety evaluation addressed the potential affect the 2000 pounds of temporary lead shielding had on the operability of the Reactor Coolant System. The safety evaluation determined that the temporary shielding was acceptable because the total weight of the temporary shielding plus the weight of the upper internals lift rig and shielding was less than the total weight seen when the reactor head was installed. The reactor vessel and its supports were well within their design loading for this arrangement.

Jumper-Lifted Lead-Bypass Number 3-93-133

This jumper, entitled "Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Ventilation System Temporary Ductwork," has been removed

Description of Jumper-Lifted Lead-Bypass

A temporary duct spool piece was installed at the backdraft damper for the discharge of the "A" train exhaust fan. The CH RPCCW ventilation system was shut down for short periods to remove the old damper and install the spool piece and then to remove the spool piece and install the new damper. The spool piece was fabricated to the same dimensions as the damper and bolted in place. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

During replacement of the backdraft damper the old damper was used as a template for the new damper bolt holes. In order to operate the ventilation system during this time, the spool piece was installed in place of the damper.

Safety Evaluation

The damper replacement was performed during the refueling outage when all fuel is removed from the reactor. An RPCCW heat exchanger was in operation to support the spent fuel pool cooling system. A calculation conservatively shows that the pump and heat exchanger area would not exceed maximum temperatures for 90 minutes without ventilation. The ventilation system was off for less than 90 minutes for the damper replacement. This change had no adverse affects on any safety related systems or components.

Jumper-Lifted Lead-Bypass Number 3-93-137

This jumper, entitled "Blind Flange Over 'A' Steam Generator Supply to Turbine Drive Auxiliary Feedwater Pump Valve Body," has been removed.

Description of Jumper-Lifted Lead-Bypass

A blind flange and gasket was installed in place of the upper valve body for the "A" Steam Generator supply to the Turbine Drive Auxiliary Feedwater Pump. The blind flange was held in place to the valve body with four, 1/2" cap screws. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The valve was being repaired when the need for Containment integrity was required. By installing a plate over the open valve body, the pressure integrity of the piping system was restored and Containment integrity was obtained.

Safety Evaluation

The safety evaluation looked at the capability of the blind flange to ensure Containment integrity. As the plant was in Mode 6, two accidents were evaluated. These are the design basis Fuel Handling and Loss of Decay Heat Removal accidents. The maximum Containment pressure postulated for a loss of decay heat removal accident is 15 psig. The blind flange was analyzed as more than capable of handling a 15 psig differential pressure.

Jumper-Lifted Lead-Bypass Number 3-93-138

This jumper, entitled "Containment Video System," is installed.

Description of Jumper-Lifted Lead-Bypass

This jumper installed a video system inside the Millstone Unit No. 3 Containment to enable remote, continuous monitoring of various components and general areas during power operations. This jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

The video system was installed to provide the capability to remotely monitor conditions inside Containment during power operations. The ability to remotely monitor conditions and personnel inside Containment increases the Unit's ability to detect potential problems early and to continuously monitor specific areas and or components as necessary, without requiring a Containment personnel entry.

Safety Evaluation

The 120 VAC power required for system operation, provided by non-vital Containment lighting electrical outlets, is minimal and within the capacity of the individual lighting panels. As such, there is no adverse impact on voltage studies. The cameras do not contribute any short circuit current and there was no impact on this analysis.

All cable is qualified and is suitable for permanent use inside the Containment.

The maximum fault currents associated with the video system are limited by electronic components such that there would not be sufficient fault current available on the cables to damage the electrical penetration.

The cables are physically located and qualified to standards that ensure that fire protection standards are not jeopardized.

The location and mounting of the video system cable and components inside Containment was done in a manner so as not to introduce any seismic interaction concerns with other safety-related equipment.

Jumper-Lifted Lead-Bypass Number 3-93-144

This jumper, entitled "Operation of Turbine Plant Component Cooling Water (TPCCW) System with a Limited Heat Sink," has been removed.

Description of Jumper-Lifted Lead-Bypass

A temporary cooling water supply to the TPCCW System heat exchangers from the manway on an inlet Circulating Waterbox was provided. The water was pumped through the heat exchanger using fire hoses and a temporary pump via a temporary manway cover and temporary heat exchanger inlet channel cover. The cooling capacity of one heat exchanger with specific heat loads was verified.

Special Procedure 91-3-010 ensured that only specific heat loads were placed on the heat exchanger utilizing this bypass jumper. The procedure also ensured the heat exchanger was not overpressurized during this evolution. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The change was implemented to maintain the TPCCW system in operation while its normal cooling water source (Service Water) was out of service for inspections.

Safety Evaluation

The failure of the pump or any of the temporary TPCCW equipment utilized in this bypass jumper would result in reduced cooling to the on-line heat exchanger, thereby reducing the heat sink for those loads requiring cooling. Existing plant instrumentation allowed constant monitoring of the temperature and, upon receipt of an alarm, operations' personnel would have been able to take action to mitigate any equipment concerns. The change affected no safety-related equipment and had no impact on the safe shutdown capability of the plant.

Jumper-Lifted Lead-Bypass Number 3-93-145

This jumper, entitled "Installation of a Temporary Blank Flange on 'B' Train of Service Water," has been removed

Description of Jumper-Lifted Lead-Bypass

The bypass jumper installed a temporary blank flange in place of "B" train Service Water supply to Turbine Plant Component Cooling Water (TPCCW). The flange maintained piping integrity for the "B" train of Service Water. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The installation of the flange enabled continued operation of the "A" and "B" trains of Service Water while repairs were made to the downstream flange of the "B" train Service Water supply to the TPCCW system during the recent refueling outage. The "A" train Service Water supply valve provided isolation for the "A" train of Service Water.

Safety Evaluation

The flange did not impact the ability of the Service Water system to perform its safety functions. The blank flange met all design inputs and was evaluated to be structurally and seismically satisfactory.

Jumper-Lifted Lead-Bypass Number 3-93-150

This jumper, entitled "Installation of a Freeze Seal on the Service Water Return Piping of the 'A' Charging Pump Cooling Heat Exchanger," has been removed.

Description of Jumper-Lifted Lead-Bypass

A freeze seal was installed on the Service Water return piping of the "A" Charging Pump Cooling heat exchanger. The freeze seal isolated the main header from the return piping of the outlet isolation valve. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The freeze seal was required to repair the piping downstream of the outlet valve of the "A" heat exchanger while the "A" train of Service Water was maintained operable.

Safety Evaluation

The "A" train of Service Water remained operable while the seal was installed. The installation of the seal caused the "A" train of Charging and Charging Pump Cooling to be inoperable. The freeze seal was installed while the plant was in Mode 5 and the "A" train of Charging was not required for any safe shutdown capability. Inoperability of the "A" train of the Charging System and its associated cooling system did not affect plant operation in Mode 5. No shutdown risk requirements were affected by the change. Contingencies were implemented to ensure both trains of Service Water remained operable if the freeze seal were failed or the pipe which was frozen cracked.

Jumper-Lifted Lead-Bypass Number 3-93-151

This jumper, entitled "Installation of a Freeze Seal on the Service Water Return Piping of the 'B' Charging Pump Cooling Heat Exchanger," has been removed.

Description of Jumper-Lifted Lead-Bypass

A freeze seal was installed on the Service Water return piping of the "B" Charging Pump Cooling heat exchanger. The freeze seal isolated the main header from the return piping of the outlet isolation valve. This jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The freeze seal was required to repair the piping downstream of the outlet valve of the "B" heat exchanger while the "B" train of Service Water was maintained operable.

Safety Evaluation

The "B" train of Service Water remained operable while the seal was installed. The installation of the seal caused the "B" train of Charging and Charging Pump Cooling to be inoperable. The freeze seal was installed while the plant was in Mode 5 and the "B" train of Charging was not required for any safe shutdown capability. Inoperability of the "B" train of the Charging System and its associated cooling system did not affect plant operation in Mode 5. No shutdown risk requirements were affected by the change. Contingencies were implemented to ensure both trains of Service Water remained operable if the freeze seal failed or the pipe which was frozen cracked.

Jumper-Lifted Lead-Bypass Number 3-93-152

This jumper, entitled "Installation of Temperature Detectors on Reactor Plant Component Cooling Water Lines," is installed.

Description of Jumper-Lifted Lead-Bypass

Temperature detectors and thermal insulating blankets were installed on the discharge lines of the Reactor Plant Component Cooling Water (RPCCW) lines for all four Reactor Coolant Pumps (RCPs). Cabling for the detectors was routed to a Netpac unit in the Containment annulus utilizing existing openings in the annulus wall. The bypass jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

The temperature detectors were installed to obtain data on the RPCCW temperature at the outlet of the RCPs. The information is used to monitor RCP performance and detect potential problems with the RCP thermal barrier.

Safety Evaluation

The potential for components being swept into the containment sumps during a loss of coolant accident and blocking or clogging the recirculation pumps was evaluated. Clogging was determine not to be a concern because the materials/components used do not have the potential of breaking up into pieces small enough to pass through the sumps' protective screening. The surface area of the components is not considered to be large enough to block the screens and prevent flow to the sump.

A seismic evaluation determined that the additional weight of the insulation and the temperature detectors, did not have any adverse affects on, or change, the seismic qualification of the RPCCW piping.

Jumper-Lifted Lead-Bypass Change Number 3-93-155

This jumper, entitled "Enclosure Wall in Fuel Building Railroad Access," is installed.

Description of Jumper-Lifted Lead-Bypass

This change installed a temporary wall in the railroad access of the Fuel Building. The wall formed an enclosure for the storage and overhaul for a used Reactor Coolant Pump (RCP). The wall was made of pre-stressed concrete blocks. The wall was approximately 17′ long x 12′ high x 2′ thick. This jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

This wall was installed to form an enclosure for storage and overhaul of a used reactor coolant pump.

Safety Evaluation

The safety evaluation concluded that the wall and storage of RCPs in this location would not adversely affect the Fuel Building.

Jumper-Lifted Lead-Bypass Number 3-93-164

This jumper, entitled "Train 'A' Reactor Vessel Level Indication System Heater No. 8 Bypassed," is installed.

Description of Jumper-Lifted Lead-Bypass

A resistor has been substituted for heater No. 8 in the "A" train of the Reactor Vessel Level Indication System (RVLIS). Electrical jumpers have been installed in the associated temperature control circuits. This jumper will remain installed for the duration of the fuel cycle.

Reason for Jumper-Lifted Lead-Bypass

Heater No. 8 in the "A" train of the RVLIS circuit was indicating an open circuit. The system will not operate properly with an open circuit. The cause of the open circuit is not known.

Safety Evaluation

The system remains operable with the bypass jumper installed. The "A" train reactor vessel level indication will be inaccurate in a non-conservative direction should actual level decrease below 19% in the upper plenum. This inaccuracy was acceptable in that the emergency operating procedures have been revised to direct the operators to take specific actions before this level is reached, i.e., at $\geq$ 32% rather than at $\geq$ 19%. This system is for indication only and has no control functions. "B" train was available and accurate.

Jumper-Lifted Lead-Bypass Number 3-93-165

This jumper, entitled "Install Blind Flange in Place of Reactor Plant Gaseous Drain Relief Valve," has been removed.

Description of Jumper-Lifted Lead-Bypass

A blind flange was installed in place of a thermal relief valve on the discharge line from the Containment Drains Transfer Pumps while the valve was removed for maintenance. The relief valve is located on the portion of the discharge line between the two containment isolation valves in the line. Should the penetration be isolated by the isolation valves and the penetration were to receive external heat input, the relief valve relieves excessive pressure build-up. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

While the plant was shutdown, the relief valve needed to be repaired. Plant conditions required that Containment Drains Transfer Tank be pumped every 8 hours. In order to perform the valve repair, the flange where the valve was located needed to be blanked off.

Safety Evaluation

The plant was in a condition in which there was no credible source of heat input to the penetration. Therefore, the relief valve was not required for penetration protection. The relief valve is not needed for any other reason in the operation of the system.

Jumper-Lifted Lead-Bypass Change Number 3-93-167

This jumper, entitled "Temporary Restraint for Jib Crane," is installed.

Description of Jumper-Lifted Lead-Bypass

This jumper installed a temporary restraint for jib crane 'A'. The restraint is required when the jib crane is not in use. The restraint will be removed during the next refueling outage and a permanent bracket installed.

Reason for Jumper-Lifted Lead-Bypass

A temporary restraint was installed as time constraints prevented installing a permanent bracket. The restraint will prevent horizontal movement of the crane boom if there is a seismic event.

Safety Evaluation

No new malfunctions were created and no previously evaluated accidents were affected as the temporary restraint satisfies the same criteria as the permanent bracket design.

Jumper-Lifted Lead-Bypass Number 3-93-170

This jumper, entitled "Direct Atmospheric Pressure for Supplemental Leak Collection and Release System Testing," is installed.

Description of Jumper-Lifted Lead-Bypass

Tubing was run from the normal location of Supplemental Leak Collection and Release System (SLCRS) test bottles in the "B" High Pressure Safety Injection (SIH) Cubicle, the Engineered Safety Features (ESF) Building, and the Main Steam Valve Building (MSVB) to the Enclosure Building. The tubing was run through the shake spaces between Containment and the ESF and MSVB structures to the Enclosure Building outer wall. A hole large enough for a 1/2" stainless steel tube was drilled through the wall of the Enclosure Building, the tube was installed and sealed to maintain the SLCRS boundary. For SLCRS testing, the tubing was connected to the penetrating pipe. When the testing was completed, the tubing was removed and the penetrating pipe capped with airtight caps. This arrangement allowed making building draw dow measurements directly, using outside air pressure as a reference. This bypass jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

The original test bottles for SLCRS drawn down test were developed in order not to make penetrations in existing buildings. The design of the system made it difficult to make differential pressure measurements. The modification allowed for the installation of two test gauges which directly measure the differential pressure between the outside and the "B" SIH Cubicle and MSVB.

Safety Evaluation

The added penetration to the Enclosure Building was evaluated with regard to its impact on the integrity of the SLCRS boundary during operation. The SLCRS boundary was maintained by having no direct air access when using the pressure gauge and air tight fittings installed after completion of the test.

Jumper-Lifted Lead-Bypass Number 3-93-171

This jumper, entitled "Plywood Shield for 'A' Quench Spray Pump," has been removed

Description of Jumper-Lifted Lead-Bypass

A plywood shield was constructed around the "A" Quench Spray pump. The bypass jumper has been removed.

Reason for Jumper-Lifted Lead-Bypass

The anchor bolts for the "A" Quench Spray pump were being removed due to possible stress corrosion cracking. The shield was installed to protect workers from injury in the event that the pump started automatically.

Safety Evaluation

The shield did not impact the operation of the pump. Ventilation for the pump was not significantly obstructed by the shield. The plywood did not increase the fire loading in the building. The shield was not attached to the pump so there was no impact on the seismic qualification of the pump.

Jumper-Lifted Lead-Bypass Change Number 3-93-172

This jumper, entitled "Reverse Lifted Lead Termination on Rod C11 Lift Coil," is installed.

Description of Jumper-Lifted Lead-Bypass

This change reversed the two lead wires for the Control Rod C11 Lift Coil. This ensures that all of the coils are of the same polarity. This jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

During startup testing from the recent refueling outage, it was noted that the traces obtained while performing control rod drop testing for the C11 rod were significantly weaker than for the other control rods. It was determined that by having the polarity reversed for the lift coil the residual magnetism in the rod was less and the trace generated as the rod dropped was significantly less. The reversed leads resulted in acceptable operation of the control rod drive system in the short term, but caused concerns in the long term on operating margin, wear, and stepping functions. The trace, while being weaker than normal was still accurate on drop time and readable.

Safety Evaluation

The change returned the system to its original design. The jumper was necessary as the leads were reversed in a location accessible at the time but not at the reactor head which is where the initial reversal occurred.

The possibility that the error in polarity reversal would have had an adverse affect on the rod control system in the past was also reviewed. While polarity reversal is not desirable in the long term, it has no adverse affect on the operation of the rod control system in the short term.

Jumper-Lifted Lead-Bypass Number 3-93-173

This jumper, entitled "Bypass of Low Pressure/Low Flow Output to Hydrogen Analyzer Common Trouble Alarm," is installed.

Description of Jumper-Lifted Lead-Bypass

The isolation relay output of low pressure/low flow was disconnected from the other six relay outputs actuating the common trouble alarm located in the Control Room.

Reason for Jumper-Lifted Lead-Bypass

The hydrogen monitoring system is normally turned off. This results in a constant alarm in the Control Room due to low flow and low pressure. This input was disconnected to clear this nuisance alarm.

Safety Evaluation

The hydrogen monitoring system is capable of performing its function with the low pressure/low flow alarm disconnected.

Jumper-Lifted Lead-Bypass Number 3-93-174

This jumper, entitled "Temporary Shielding - Auxiliary Building Boronometer Cubicle," is installed.

Description of Jumper-Lifted Lead-Bypass

Temporary shielding is installed on Charging System relief line piping in the Boronometer cubicle. The jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

This jumper is needed to reduce radiation levels to eliminate the need for the area to be classified and controlled as a Technical Specification locked high radiation area.

Safety Evaluation

The safety evaluation addressed the potential impact of placing 50 pounds of temporary lead shielding on the Charging System relief line off the Reactor Coolant Pump Seal Water Header. The temporary shielding represented a 20% load increase to the system. This was found to be within the systems calculated limits for deadload and seismic moments. To preclude interaction between the stainless steel piping and temporary sheet lead, the system piping would be protected by heat resistant fabric and vinyl to eliminate the lead/stainless steel interaction. Also, the temporary shielding would be affixed to the system piping at 6-inch intervals using industrial hose clamps or steel banding.

Jumper-Lifted Lead-Bypass Number 3-93-179

This jumper, entitled "Addition of Oil Add Connection to 'D' Reactor Coolant Pump (RCP)," is installed.

Description of Jumper-Lifted Lead-Bypass

A series of fittings, tubing, and an isolation valve were added at the oil fill valve for the "D" RCP to allow for the addition of oil from a remote location while the pump is in operation. This jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

Monitoring of the oil levels in the RCP motors found that there was a gradual decrease in the upper bearing oil level in the "D" pump. If oil could not be added, a point would be reached where bearing damage would occur and the pump would have to be stopped.

Safety Evaluation

The seismic review conducted by Westinghouse on the additional weight added to the oil fill connection concluded that it was acceptable. This review ensured that there is no increased probability of failure of the connection.

The new fittings have a slightly increased probability of leakage. The increased probability was acceptable as the oil pressure at the fitting is very low and leakage would not spray but just drip into the existing oil collection system. Additionally, any leakage would have to first leak past the existing isolation valve.

The residual oil left in the tube after filling was an insignificant amount and posed no safety concerns. The review concluded that the amount of oil being added at any one time was less than 4% of that allowed in transit in containment; therefore, it was within the bounds of the analysis.

Jumper-Lifted Lead-Bypass Change Number 3-93-184

This jumper, entitled "Temporary Drain Lines for 'B' and 'D' Steam Supplies to the Turbine Driven Auxiliary Feedwater Pump (TDAFW)," is installed.

Description of Jumper-Lifted Lead-Bypass

A temporary drain line and catch basin was installed for the drain lines upstream of the TDAFW pump steam isolation valves from the "B" and "D" steam generators. The temporary drain lines were attached downstream of the integral trap drains. The lines extended down into a 55 gallon drum filled with water. The water in the drum served as a quenching volume for the hot condensate. A red rubber hose was run from the 55 gallon drum to the near-by contaminated floor drain. A bronze isolation valve was installed at the drum to control flow to the floor drain. This jumper is installed.

Reason for Jumper-Lifted Lead-Bypass

This jumper prevented a water slug or water hammer from damaging the TDAFW pump or steam supply lines if the "B" or "D" steam supply valves automatically opened. A water slug could overspeed the turbine; so draining the condensate upstream from the isolation valve precluded the potential of a water slug that would result in overspeed of the turbine.

Safety Evaluation

The small amount of tubing had no impact on the seismic qualification of the existing piping. The stainless steel tubing is compatible with the existing piping. The temporary drains were downstream of the containment isolation valves which close on a steam line isolation signal. This prevents a pipe failure from diverting steam from the TDAFW pump.

Due to the location of drain on the "A" steam supply, it was inconvenient to provide this drain line with temporary piping. Therefore, for the duration of this by-pass, the "A" steam generator supply to the TDAFW was considered inoperable. The "A" steam supply isolation valve was tagged shut to ensure that an auto-start of the TDAFW did not occur from the "A" steam supply. Only two main steam supply lines are required to support the TDAFW pump operability, so isolating the "A" supply was acceptable.

TESTS

Test Number	Title
IST 3-92-032	Emergency Generator Load Sequencer (EGLS) Train "A" Power Supply Data Acquisition Procedure
IST 3-92-033	Emergency Generator Load Sequencer (EGLS) Train "B" Power Supply Data Acquisition Procedure
IST 3-93-001	Volume Control Tank (VCT) Temperature Reduction and Reactor Coolant Pump (RCP) No. 1 Seal Leakoff Flow Monitoring
IST 3-93-003	Seal Injection Filter Element Replacement
IST 3-93-004	Upper Plenum Anomaly (UPA) Data Collection
IST 3-93-005	Operation of the "A" Main Feedwater Pump Following Coupling Replacement
IST 3-93-006	Sodium Hypochlorite Jet Pumps Test
IST 3-93-007	Solid State Protection System Slave Relay Inservice Test
IST 3-93-009	Solid State Protection System Slave Relay Inservice Test
IST 3-93-011	Overlap Test of Steam Generator Hi-Hi Level and Safety Injection Trip of Turbine and Main Feedwater Pump
IST 3-93-012	Charging System Filter Element Replacement
IST 3-93-013	Charging System Dynamic Valve Test
IST 3-93-014	Trip Actuating Device Operational Test of the Main Steam Isolation Actuation Relays
IST 3-93-016	Reactor Coolant Pump (RCP) Temporary Cooling System
IST 3-93-019	Early Boration

TESTS (CONTINUED)

Test Number	Title
IST 3-93-020	Trip Actuating Device Operational Test of Turbine Trip on Reactor Trip
IST 3-93-022	Speed Sensor Noise Data Collection
IST 3-93-025	Technical Support Center Ventilation Modification Test Plan
IST 3-93-028	Safety Injection System Pump Discharge Valve Dynamic Test
IST 3-93-032	Charging System Valve Dynamic Test
IST 3-93-034	Auxiliary Feedwater System Motor Operator Valve Dynamic Test
IST 3-93-035	Auxiliary Feedwater System Motor Operator Valve Dynamic Test
IST 3-93-036	Auxiliary Feedwater System Motor Operator Valve Dynamic Test
IST 3-93-046	Supplemental Leak Collection and Release System (SLCRS) and Auxiliary Building Ventilation System Retest for Design Changes 3-93-200 and 3-93-205 in Winter Modes
IST 3-93-048	Speed Sensor Noise Data Collection
IST 3-93-049	Low Pressure Turbine Pre-Warming Procedure
IST 3-93-052	Over Pressure Delta Temperature (OPDT) and Over Temperature (OTDT) Delta Temperature Data Acquisition
IST 3-93-053	Electro-Hydraulic Control (EHC) System Control Valve Testing
IST 3-93-054	Electro-Hydraulic Control (EHC) System Control Valve Testing
Not Numbered	Slave Relay Interlock Contact Testing

Test Number	Title
IST 3-92-032	Emergency Generator Load Sequencer (EGLS) Train 'A' Power Supply Data Acquisition Procedure

Description of Test

This test was performed to verify the voltage and current characteristics of the EGLS output relay power supplies. Data acquisition equipment was attached to the power supply under test to monitor performance during normal EGLS surveillance testing. The testing equipment and methodology are not invasive and could not alter power supply performance or damage EGLS components in any way.

Reason for Test

Power supplies of similar design located in the EGLS power cabinet have failed due to component aging and other reasons. Even though the EGLS is under regular surveillance, testing of the EGLS output relay power supplies was considered prudent because of the identified failure pattern of these components

Safety Evaluation

The methodology of power supply surveillance could not cause an accident or worsen the consequences of an analyzed accident.

One train of the EGLS was operable at all times.

Test Number	Title
IST 3-92-033	Emergency Generator Load Sequencer (EGLS) Train 'B' Power Supply Data Acquisition Procedure

Description of Test

This test was performed to verify the voltage and current characteristics of the EGLS output relay power supplies. Data acquisition equipment was attached to the power supply under test to monitor performance during normal EGLS surveillance testing. The testing equipment and methodology are not invasive and could not alter power supply performance or damage EGLS components in any way.

Reason for Test

Power supplies of similar design located in the EGLS power cabinet have failed due to component aging and other reasons. Even though the EGLS is under regular surveillance, testing of the EGLS output relay power supplies was considered prudent because of the identified failure pattern of these components.

Safety Evaluation

The methodology of power supply surveillance could not cause an accident or worsen the consequences of an analyzed accident.

One train of the EGLS was operable at all times.

Test Number	Title
IST 3-93-001	Volume Control Tank (VCT) Temperature Reduction and Reactor Coolant Pump (RCP) No. 1 Seal Leakoff Flow Monitoring

Description of Test

The test temporarily lowered the temperature of the Reactor Plant Closed Cooling Water system to approximately 60°F. This results in cooler make-up water being supplied to the VCT. Previous experience has shown that cooler make-up water results in lower leakoff flows from the RCP Number 1 seals.

Reason for Test

This test was conducted to mitigate the increased leakoff flow from the "B" RCP Number 1 seal.

Safety Evaluation

The affected plant systems were operated within their design limits.

Test Number	Title
IST 3-93-003	Seal Injection Filter Element Replacement

Description of Test

This test placed a 0.2 micron filter element in either one of the seal injection filters. This filter was then placed inservice for normal duty. The standby filter had a 2.0 Micron filter element.

Reason for Test

The test installed finer filter cartridges as a means of preventing particulate from plating out on the Reactor Coolant Pump (RCP) Number 1 seals.

Safety Evaluation

The test ensured that the finer filters would be changed at the same differential pressure as the filters that were currently accepted for use. The standby filter had a 2.0 micron filter, the preferred backup filter size. Experience has shown that this provides assurance that seal injection flow to the RCPs will not be adversely impacted.

Test Number	Title
IST-3-93-004	Upper Plenum Anomaly (UPA) Data Collection

Description of Test

The test collected Reactor Coolant System temperature data to forward to Westinghouse. A data acquisition system was connected to the plant computer input cabinets and approximately eight hours of hot leg and cold leg temperature data was collected at 10 samples per second.

Reason for Test

The data was requested by Westinghouse in support of their UPA investigation. The data will be used to determine whether the UPA causes an unacceptable amount of cyclic stress to upper vessel internals, hot leg nozzle welds, or steam generator inlet nozzles.

Safety Evaluation

Test connections were all within a non-1E cabinet. Electrical separation between the connection point and safety grade signals are a design feature of the Process Protection System. A fault in the data acquisition system would not affect the protection system.

Test Number	Title
IST 3-93-005	Operation of the "A" Main Feedwater Pump Following Coupling Replacement

Description of Test

This test provided the sequence of steps to operate plant systems with the necessary hold points to acquire vibration and alignment data to determine satisfactory pump operation following changing the pump to turbine coupling from gear type to flexible disc type.

Reason for Test

This test provided a controlled startup of the "A" main feedwater pump to determine if acceptable operation is achievable following coupling replacement.

Safety Evaluation

The safety evaluation analyzed the replacement of the gear type coupling with one of a flexible disc design and the testing of the pump under a controlled evolution prior to making the piping misalignment and potential casing deformation repairs. There was no increase in the probability of occurrence of a loss of normal feedwater flow.

Test Number Title

IST 3-93-006 Sodium Hypochlorite Jet Pumps Test

Description of Test

The test assured that the pumps operated within the design limits prior to system operation and determined individual pump performance characteristics.

Reason for Test

The test assured that the jet pumps were capable of producing 0.2 gallons per minute as required for proper chlorination of the Service Water system without exceeding the free available chlorine 0.25 parts per million.

Safety Evaluation

The test did not affect the biofouling of the Service Water system because of the limited duration of no hypochlorite injection. Any biofouling which may have occurred due to a lack of hypochlorite injection was eliminated during the recent refueling outage when the Service Water system was fully drained and inspected.

Test Number	Title
IST 3-93-007	Solid State Protection System Slave Relay Inservice Test

Description of Test

This procedure tested the operability of the Solid State Protection System output relay for Containment Isolation Phase "B" for Trains "A" and "B". The associated valves were momentarily taken to their non-accident position to verify that they did stroke to their safety position in response to the slave relay actuation. This test did not prevent Emergency Diesel Generator operation if a valid undervoltage condition existed on the train under test.

Reason for Test

This output relay was not previously tested.

Safety Evaluation

This test was performed under the requirements stipulated by Plant Technical Specifications.

Test Number	Title
IST 3-93-009	Solid State Protection System Slave Relay Inservice Test

Description of Test

This test energized relays supplying control circuit power to four low pressure safety injection accumulator isolation valves. Thermal overload protection devices to the valve motors were removed prior to testing to ensure that the valves would not inadvertently open.

Reason for Test

This test evaluated the control circuits capability to supply power to the four accumulator isolation valves. This circuitry had not been tested previously.

Safety Evaluation

The removal of the valves thermal overload protection devices ensured that required valve positioning was maintained during the testing evolution. The plant was shut down so the injection capability of the accumulator was not required.

Test Number	Title
IST 3-93-011	Overlap Test of Steam Generator Hi-Hi Level and Safety Injection Trip of Turbine and Main Feedwater Pump

Description of Test

This test ensured that a Hi-Hi level in a steam generator or a safety injection signal would cause the turbine and the main feedwater pumps to trip.

Reason for Test

The Technical Specifications require a channel calibration of the Hi-Hi steam generator level trip function and an operational test of the safety injection signal. A relay in the turbine trip and main feedwater pump trip circuitry had not been included in previous surveillance testing.

Safety Evaluation

These protective trips are only used in response to an accident. The plant was already shutdown when the test was performed. Therefore, while the accidents mitigated by these functions could occur during the test, they are bounded by the at-power conditions. No safety-related equipment was involved in this test.

Test Number	Title
IST 3-93-012	Charging System Filter Element Replacement

Description of Test

This test evaluated the feasibility of installing finer filters for the Reactor Coolant System (RCS) and letdown filters. A 0.2 micron filter element was installed in place of the existing 25.0 micron filter element for the RCS filter. A 6.0 micron filter element was installed in place of the existing 25.0 micron filter element in the letdown filter.

Reason for Test

Filtering of particles as small as 0.2 microns was desired in order to prevent particles from plating out on the Reactor Coolant Pump seals. The intent is to maintain the RCS as clean as possible and to reduce the particle loading on the seal injection filters.

Safety Evaluation

Both filters are in the letdown portion of the Charging System. This portion of the system is isolated during a design basis accident and is not used for accident mitigation.

Test Number	Title
IST 3-93-013	Charging System Dynamic Valve Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Charging System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was performed on the Charging System when the Technical Specifications did not require the system to be operable. With the reactor head removed and the Reactor Coolant System loop drain valves open, no credible means existed for loop overpressurization.

Test Number	Title
IST 3-93-014	Trip Actuating Device Operational Test of the Main Steam Isolation Actuation Relays

Description of Test

This test ensured that a Main Steam Isolation (MSI) signal would cause the Main Steam Isolation Valves (MSIVs) to close. The test was performed at-power, so a jumper was installed to keep the MSIVs open despite the trip signal present from one train.

Reason for Test

Technical Specifications require an operational test of the MSI logic. A relay in the MSIV closure circuitry had not been included in previous surveillance testing.

Safety Evaluation

The test rendered one train of the MSI signal inoperable at a time. The Technical Specification time and plant condition limitations for this condition were observed. The opposite train MSI signal was always available to initiate closure. The MSI signal is used to mitigate certain accidents. Accidental closure of the MSIVs while at-power would cause a reactor trip but would not initiate an accident.

Test Number — Title

IST-3-93-016 — Reactor Coolant Pump (RCP) Temporary Cooling System

Description of Test

This test was performed to evaluate the effects and provide guidance on the operation of the temporary cooling system installed under bypass jumper 3-93-067. The temperature change on the seal injection line and the effect on the RCP seal leak off was monitored during the test.

Reason for Test

This temporary cooling system was installed to minimize RCP seal leak off by cooling seal injection water.

Safety Evaluation

The safety evaluation addressed the possibility of a failure of the cooling system and the impact of a failure on RCP seals. Once the cooling system was in operation, a postulated failure would result in a rapid rise in leak off and seal temperature. If the leak off exceeded limits, a plant shutdown would be required, but no damage to the seals would occur.

Test Number	Title
IST 3-93-019	Early Boration

Description of Test

This test directed the boration of the Reactor Coolant System to shutdown levels before reaching 400°F during the plant shutdown immediately preceding the refueling outage. This resulted in a crud burst which could be cleaned up before completion of the shutdown.

Reason for Test

Industry experience showed that early boration resulted in lower radiation levels inside containment during a subsequent refueling outage.

Safety Evaluation

Similar procedures have been used successfully at other plants. Increasing the boron concentration early in the shutdown process only increases the shutdown margin of the plant.

Test Number	Title
IST 3-93-020	Trip Actuating Device Operational Test of Turbine Trip on Reactor Trip

Description of Test

The procedure tested the turbine trip function resulting from a reactor trip breaker opening. A jumper was installed to simulate a reactor trip breaker opening. A digital multi-meter was used to monitor contact operation.

Reason for Test

The test was performed to meet the Technical Specification requirement of testing the reactor trip-turbine trip interlock.

Safety Evaluation

The contact which trips the turbine was removed from the circuit by means of a sliding link. Only one train of the interlock was removed from service at any time. The opposite train and the manual turbine trip were always operable during this time. The duration of the test was less than the allowed inoperable time defined in Technical Specifications.

Test Number	Title
IST 3-93-022	Speed Sensor Noise Data Collection

Description of Test

This test was performed to gather electronic noise data from the Main Turbine Speed Sensors as the turbine was ramping down in load and speed for the planned refueling outage.

Reason for Test

This test provided valuable data on where the electronic noise originated in the Main Turbine Electro-Hydraulic Control system. This information was used as a basis for the corrective action done during the refueling.

Safety Evaluation

The safety evaluation evaluated the possibility of a turbine trip caused by the loss of a single speed sensor or the loss of all speed sensors firing the data gathering evolution. The likelihood of these events occurring was determined to be very low. This is based on the precautions that were instituted during the performance of the test.

Test Number	Title
IST 3-93-025	Technical Support Center Ventilation Modification Test Plan

Description of Test

This test performed a retest for Plant Design Change Request 3-93-012, "Technical Support Center Ventilation Damper Modifications." The system was operated in all modes of operation and data was taken to ensure that all system design criteria were met after two motor operated dampers were modified to the permanent locked open positions.

Reason for Test

Plant Design Change Number 3-93-012 permanently locked open and removed the associated actuators, cabling, and controls from the two dampers in the Technical Support Center Ventilation System. This test verified that the system would operate per design after the modifications were completed.

Safety Evaluation

The Technical Support Center Ventilation System is not a safety-related system. However, certain of the system's modes of operation are initiated by a safety-related Control Room isolation actuation. To simulate this signal, a jumper was placed in a safety-related isolator cabinet. Since the jumper was placed in the non-safety output side of the cabinet, safety systems were not affected during the test.

Test Number	Title
IST 3-93-028	Safety Injection System Pump Discharge Valve Dynamic Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Safety Injection System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was performed on the Safety Injection System when the Technical Specifications did not require the system to be operable. With the reactor head removed and the Reactor Coolant System loop drain valves open, no credible means existed for loop overpressurization.

Test Number	Title
IST 3-93-032	Charging System Valve Dynamic Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Charging System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was safe. The test was performed on the Charging System when the Technical Specifications did not require the system to be operable. With the reactor head removed and the Reactor Coolant System loop drain valves open, no credible means existed for loop overpressurization.

Test Number	Title
IST 3-93-034	Auxiliary Feedwater System Motor Operator Valve Dynamic Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Auxiliary Feedwater System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was performed on the Auxiliary Feedwater System. The affect on safety while pumping water from the Condensate Storage Tank into the steam generators was evaluated. The Technical Specifications did not require the system to be operable and operation of the system did not impact required safety systems.

Test Number	Title
IST 3-93-035	Auxiliary Feedwater System Motor Operator Valve Dynamic Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Auxiliary Feedwater System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was performed on the Auxiliary Feedwater System and coincided with filling of the steam generators. The Technical Specifications did not require the system to be operable and operation of the system did not impact required safety systems.

Test Number	Title
IST 3-93-036	Auxiliary Feedwater System Motor Operator Valve Dynamic Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Auxiliary Feedwater System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was performed on the Auxiliary Feedwater System and coincided with filling of the steam generators. The affect on safety while pumping water from the Condensate Storage Tank into the steam generators was evaluated. The Technical Specifications did not require the system to be operable and operation of the system did not impact required safety systems.

Test Number	Title
IST 3-93-046	Supplemental Leak Collection and Release System (SLCRS) and Auxiliary Building Ventilation System Retest For Design Changes 3-93-200 and 3-93-205 in Winter Modes

Description of Test

This change provides instructions to perform the test in any plant operating mode in which SLCRS is not required to be operable. The inservice test was originally intended to be performed in Mode 5 or 6.

Reason for Test

Enforcement discretion was granted by the Nuclear Regulatory Commission for the plant to start up for low power testing without an operable SLCRS. This change was implemented to allow SLCRS testing during plant heatup.

Safety Evaluation

Because the plant had been shut down for over 80 days for refueling, virtually all iodine in the core had decayed away. No additional iodine was produced during Modes 3 through 5 because the reactor is subcritical. Any unfiltered release during Modes 3 through 5 will be bounded by the design basis accident analysis. The iodine produced by operating in Mode 2 for one week is also bounded by the iodine inventory assumed in the design basis accident analysis, thus any unfiltered release would also be bounded.

Test Number | Title

IST 3-93-048 | Speed Sensor Noise Data Collection

Description of Test

This test was performed to gather electronic noise in the Main Turbine Electro-Hydraulic Control system after the corrective action performed during the refueling outage.

Reason for Test

This test provided valuable data on where the electronic noise originated in the Main Turbine Electro-Hydraulic Control system. This information was used as a basis for the corrective action done during the refueling.

Safety Evaluation

The safety evaluation evaluated the possibility of a turbine trip caused by the loss of a single speed sensor or the loss of all speed sensors firing the data gathering evolution. The likelihood of these events occurring was determined to be very low. This is based on the precautions that were instituted during the performance of the test.

Test Number	Title
IST 3-93-049	Low Pressure Turbine Pre-Warming Procedure

Description of Test

This test was performed to pre-warm the low pressure turbines by degrading condenser back pressure until the turbine reached 25% rated electrical output.

Reason for Test

This test pre-warmed the low pressure turbines to prevent small cracks in the "A" and "C" Main Turbine rotors from propagating. This was recommended by General Electric as a method of maintaining the reinspection frequency for these rotors. The cracks had been found during the refueling outage.

Safety Evaluation

The safety evaluation evaluated the possibility of a turbine trip caused by the loss of condenser vacuum. The likelihood of this event occurring was determined to be very low. This is based on the precautions that were instituted during the performance of the test to ensure that the proper condenser vacuum was maintained.

Test Number	Title
IST 3-93-052	Over Power Delta Temperature and Over Temperature Delta Temperature Data Acquisition

Description of Test

The test collected data to determine available margin to OPDT and OTDT Runback and Reactor Trip setpoints with the plant at 100% power.

Reason for Test

The test determined whether the plant needs to downpower for Nuclear Instrumentation System and Delta-T/T_{avg} calibrations.

Safety Evaluation

There is no impact on previously evaluated accidents. Channel redundancy and test restrictions prevented a chart recorder failure from causing the loss of Delta-T/T_{avg} protection. The worst case impact of a chart recorder failure is no different than the impact due to routine channel surveillance testing.

Failure of the recorder during OTDT/OPDT margin data collection would only affect one channel, and in the worst case, result in a trip signal for that channel, changing the logic from 2/4 to 1/3.

This test was for monitoring only. There is no affect on the performance of systems that could lead to an impact on the margin of safety.

Monitoring the margin to OTDT and OPDT setpoints has no affect on any protection system, provided there are no test equipment failures. The only credible data acquisition failure would produce a conservative plant response. This failure probability is so low that it may be considered insignificant.

Test Number	Title
IST 3-93-053	Electro-Hydraulic Control (EHC) System Control Valve Testing

Description of Test

The test collected data on the EHC system operation during control valve testing.

Reason for Test

The test was for troubleshooting the EHC control valve circuitry in order to locate a possible fault that was discovered during control valve testing.

Safety Evaluation

The test produced a slight increase in the possibility of a malfunction in the EHC system. However, the slight increase in the resulting probability of a turbine trip has no impact on previously evaluated accidents. Performance of the test did not involve any additional risk, since technicians routinely monitor signals on line.

Test Number	Title
IST 3-93-054	Electro-Hydraulic Control (EHC) System Control Valve Testing

Description of Test

The test collected additional data on the EHC system operation during control valve testing.

Reason for Test

The test was for additional troubleshooting of the EHC control valve circuitry in order to locate a possible fault that was discovered during control valve testing. It also provided a means to correct the suspected cause of the problem if the test confirmed the cause.

Safety Evaluation

The test produced a slight increase in the possibility of a malfunction in the EHC system. However, the slight increase in the resulting probability of a Turbine trip has no impact on previously evaluated accidents. Performance of the test did not involve any additional risk, since technicians routinely monitor signals on line.

Test Number	Title
Not Numbered	Slave Relay Interlock Contact Testing

Description of Test

No components were tested. This item is included because a safety evaluation was performed to show that not testing certain interlock relays was safe.

Reason for Test

During review of the overlap of slave relay testing, several interlock contacts were identified as not requiring testing. These interlock relays prevent inadvertent operator realignment of the associated component to an unsafe configuration while the actuation signal is locked in.

Safety Evaluation

The failure of any of the identified interlock relays would not prevent its associated component from accomplishing its safety related function. While interlock failure could permit inadvertent operation realignment of the associated component without resetting the actuation signal, rigorous training in emergency operations procedures minimizes the likelihood of such operator action. During realignment after an accident, failure of an interlock relay may delay restoration of a component but would not prevent it.

EXPERIMENTS

There were no experiments performed under the provisions of Title 10, Code of Federal Regulations, Section 50.59 during 1993.

CHALLENGES TO RELIEF/SAFETY VALVES

In accordance with the commitment made under Item II.K.3.3 of NUREG 0737 (TMI Action Plan) in the W. G. Counsil letter to D. G. Eisenhut, dated June 10, 1980, the following is a report of challenges to Relief/Safety Valves during 1993.

There were no challenges made to the Primary Relief/Safety Valves in 1993.

PRIMARY COOLANT IODINE SPIKING

There was no primary coolant iodine spiking in 1993 that exceeded the one micro curie per gram limit set in the technical specifications.

REGULATORY GUIDE 1.16 REPORT FOR 1993

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NORTHEAST NUCLEAR ENERGY CO. UNIT 3

DATE: 2/ 3/9

WORK & JOB FUNCTION	NUMBER OF PERSONNEL (>100 MREM) STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES	TOTAL MAN-REM STATION EMPLOYEES	UTILITY EMPLOYEES	OTHER EMPLOYEES
REACTOR OPERATIONS & SURVEILLANCE						
MAINTENANCE PERSONNEL	2	0	2	0.74	0.01	0.51
OPERATING PERSONNEL	27	1	0	6.44	0.14	0.17
HEALTH PHYSICS PERSONNEL	18	1	10	4.53	0.29	3.98
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.01
ENGINEERING PERSONNEL	1	1	0	0.53	0.20	0.03
ROUTINE MAINTENANCE						
MAINTENANCE PERSONNEL	57	8	348	28.50	3.12	181.53
OPERATING PERSONNEL	24	1	4	6.29	0.76	1.50
HEALTH PHYSICS PERSONNEL	34	3	67	20.88	0.63	26.66
SUPERVISORY PERSONNEL	0	0	2	0.00	0.00	0.52
ENGINEERING PERSONNEL	8	[illegible]	55	3.68	3.23	25.16
INSERVICE INSPECTION						
MAINTENANCE PERSONNEL	0	0	48	0.00	0.00	20.40
OPERATING PERSONNEL	0	0	0	0.03	0.00	0.00
HEALTH PHYSICS PERSONNEL	1	1	3	0.44	0.35	1.16
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	1	10	0.11	0.27	4.09
SPECIAL MAINTENANCE						
MAINTENANCE PERSONNEL	8	0	11	2.54	0.49	4.79
OPERATING PERSONNEL	27	0	2	8.52	0.01	0.54
HEALTH PHYSICS PERSONNEL	6	2	19	2.03	0.32	11.67
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.01
ENGINEERING PERSONNEL	0	0	2	0.37	0.28	0.60
WASTE PROCESSING						
MAINTENANCE PERSONNEL	0	0	3	0.00	0.00	0.68
OPERATING PERSONNEL	0	0	0	0.00	0.00	0.00
HEALTH PHYSICS PERSONNEL	5	0	12	1.59	0.00	2.75
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.00
ENGINEERING PERSONNEL	0	0	0	0.00	0.03	0.01
REFUELING						
MAINTENANCE PERSONNEL	25	0	66	14.91	0.14	27.83
OPERATING PERSONNEL	3	0	2	1.21	0.01	0.40
HEALTH PHYSICS PERSONNEL	21	0	18	7.08	0.00	4.59
SUPERVISORY PERSONNEL	0	0	0	0.00	0.00	0.01
ENGINEERING PERSONNEL	4	0	16	1.26	0.34	5.73
TOTAL						
MAINTENANCE PERSONNEL	92	8	478	46.79	3.77	235.85
OPERATING PERSONNEL	81	2	8	22.63	0.93	2.62
HEALTH PHYSICS PERSONNEL	85	7	129	36.63	1.59	50.89
SUPERVISORY PERSONNEL	0	0	2	0.00	0.00	0.55
ENGINEERING PERSONNEL	13	10	83	5.96	4.35	35.64
GRAND TOTAL	271	27	700	112.00	10.63	325.55