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The Northeast Utilities System

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

Millstone Nuclear Power Station, Unit No. 1
Annual Reports

Pursuant to the provisions of 10CFR50.59, Northeast Nuclear Energy Company (NNECO) on behalf of Millstone Unit No.1 is reporting on facility changes for the period January 1, 1996 to December 31, 1996. NNECO is also including in this submittal changes to its NRC commitments. If the NRC Staff should have any questions or comments on these reports, please contact R. W. Walpole at (860) 440-2191.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

John P. McElwain

John P. McElwain
Millstone Unit No. 1 Recovery Officer

Enclosure

cc: H. J. Miller, Region I Administrator
S. Dembek, NRC Project Manager, Millstone Unit No. 1
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IE47

Attachment 1

Millstone Nuclear Power Station, Unit No. 1

10CFR50.59 Annual Report
January 1, 1996 through December 31, 1996

June 1997

MILLSTONE UNIT NO. 1
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INTRODUCTION

None of the plant design changes, procedure changes, jumpers-lifted leads-bypasses, tests, technical requirements or experiments described herein constitute (or constituted) an unreviewed safety question per the criteria of 10CFR50.59, with the exception of an "Operability Determination", done to address Boraflex Deterioration and was resolved through a Licensing Amendment, whereby gaps in the Boraflex were considered in the design.

PLANT DESIGN CHANGES

PDCP. Number	Title
1-083-95	GTG North Fuel Forwarding Pump (M8-85A) Modification
1-085-95	Reactor Building Sample Hood Modification And HVT-10 Interlock
1-087-95	RR Scoop Tube Positioner Replacement/Misc. Modifications
1-093-95	Spent Resin Tank Piping Modifications
1-095-95	Bypassing LPCI Keepfill Pressure Control Valves
1-099-95	Feedwater Coolant Injection (FWCI) System - Condensate Booster Pump Lube Oil Foundation And Related Electrical Component Anchorage
1-100-95	Installation Of Filter Assemblies In The Instrument Air Supplylines To Air Operated Valves With QA Cat. 1 Backup Air Supply
1-002-96	Spent Fuel Pool Bulk Temperature Monitoring System
1-003-96	Condensate Pump Seal Water Modification
1-006-96	New Unit Heater For Battery Room 18B

PDCR Number	Title
1-083-95	GTG North Fuel Forwarding Pump (M8-85A) Modification

Description of Change

This change is complete. The modification proposed under PDCR MP1-83-95 revision 1 will add a stiffener to shift a structural resonance frequency away from a coincident rotor critical frequency. The proposed stiffener will be installed between the AC motor stand and the tank nozzle to shift the structural resonance frequency of the AC motor stand.

Reason for Change

The modification proposed under PDCR MP1-83-95 revision 1 will enable the pump to run with acceptable Inservice Test levels (IST) and improve the overall reliability of the pump.

Safety Evaluation

The design change does not increase the risk to the public. The modification will improve the reliability of the Gas Turbine Generator (GTG) north fuel forwarding pump and does not adversely affect the pumps ability to fulfill its intended safety function of forwarding fuel for GTG operation. A seismic review has been performed and it has been determined that the seismic and mechanical loading added by this modification is acceptable. Testing will be performed to ensure that the modification is acceptable and the pump can perform its function. The modification does not adversely affect the probability of occurrence or consequences of a previously evaluated accident or malfunction of equipment important to safety. The modification does not adversely affect the possibility of an accident or a malfunction of a different type than previously evaluated. The modification does not adversely impact the existing margin of safety. The modification does not result in an Unreviewed Safety Question.

PDCR Number	Title
1-085-95	Reactor Building Sample Hood Modification And HVT-10 Interlock

Description of Change

This change is complete. The non-QA exhaust duct from the Reactor Building Cleanup System Sampling Hood was redirected to ensure the Sampling Hood is maintained under a negative pressure. The control switch for Reactor Building Transfer Fan HVT-10, was relocated to the HVAC-MCP, as well as the rewiring of nine Reactor Building Transfer Fans which are interlocked with Reactor Building Supply Fans. Normally closed auxiliary contacts from new QA relays which are energized by the initiation of the Standby Gas Treatment System are placed in the control circuits for the Reactor Building Transfer Fans. The control circuitry for the Reactor Building Isolation Dampers logic was corrected by adding a QA contact from relays energized by the opposite train of the Standby Gas Treatment System initiation. Four QA Keylock Bypass Switches are placed in the control circuitry for the Reactor Building Isolation Dampers and in the Steam Tunnel Isolation Dampers respectively. A non-QA differential pressure gauge was added to monitor Reactor Building pressure. The Reactor Building Supply Fans are interlocked with the Freeze Stats. Open and close position limit switches are added to the Steam Tunnel Isolation Dampers, which will drive indicating lights at both the control room and the HVAC-MCP.

Reason for Change

The rework of the Reactor Cleanup System Sampling Hood exhaust ductwork improves the systems performance and helps to maintain personnel safety. All control circuit changes have been implemented to align the systems with their current design basis. The added safety features (Reactor Building Isolation Damper Trips/ Reactor Building Transfer Fan Trips) will provide additional assurance that the safety function is maintained. The addition of isolation damper indicating lights will provide the operators with information on the performance of safety related equipment. The addition of test bypass switches will allow testing to be performed on the Steam Tunnel Isolation Dampers.

Safety Evaluation

These modifications are safe and do not create an unreviewed safety question. These modifications will improve accident mitigation. They will not increase the consequences of any accident. The changes do not increase the probability of an accident or malfunction. They will provide additional assurance that the Reactor Building Transfer Fans will be secured when required by the initiation Standby Gas Treatment System, and that the Steam System Tunnel Isolation Dampers have closed when required by the initiation Standby Gas Treatment System.

PDCR Number	Title
1-087-95	RR Scoop Tube Positioner Replacement/Misc. Modifications

Description of Change

This change is complete. The Bailey type RC electric positioner has been replaced with a Beck model 11-400 electric positioner. The indication system has been upgraded to provide Beck position and position demand on CRP 904. The Beck drive has been hardened to prevent EMI/RFI noise malfunctions. Cables, conduits and other equipment were installed to support full I/A System installation. The volts per hertz trip function is blocked during normal startup sequence. A resister was replaced internal to the voltage regulator as recommended in SIL 586.

Reason for Change

This design change provides an input monitor relay to the drive and verifies the position of the Beck drive local Auto/Manual switch. Should a loss of voltage to the drive occur, or if the local switch is taken out of the Auto position, the scoop tube will be locked into position.

Safety Evaluation

This change will have no negative impact on the margin of safety. The change does not increase the probability of recirculation flow malfunctions (increasing or decreasing flows). The consequences of the events remain bounding. Based on the above, the change will not adversely impact the margin of safety, and the design change is determined to be safe, and does not result in an unreviewed safety question.

PDCR Number	Title
1-093-95	Spent Resin Tank Piping Modifications

Description of Change

This change is complete. A 6 inch hole on the top of the Spent Resin Tank has been repaired by the installation of a roof patch and nozzle. The out of service suction line has been cut and plugged at the tank. Installation of new piping, valves and components were performed in accordance with the augmented QA Category RWQA requirements of Reg. Guide 1.143, including the installation of replacement Sludge Pump (M6-28). The 24 inch manhole and its associated inlet and outlet connections for the Spent Resin Tank has been modified to remove out of service and /or abandoned piping, valves and components and install new equipment which provides improved operational flow paths.

Reason for Change

Modifications consist of repairs to the Spent Resin Tank, restoration of transfer materials to augmented QA Category RWQA, revision of the tank function to allow storage of backwashed charcoal filter sludge as well as spent resin sludge, flow path revisions including removal of out of service lines and components and installation of new lines and valves.

Safety Evaluation

The modifications do not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are therefore safe and do not raise an unreviewed safety question.

PDCR Number	Title
1-095-95	Bypassing LPCI Keepfill Pressure Control Valves

Description of Change

This change is complete. The LPCI Keepfill Pressure Control Valves (PCV) have been bypassed. The inlet to the LPCI Keepfill PCVs (1-MW-82) and the outlet from the PCVs (1-MW-84) will be left normally closed and the bypass valve (1-MW-85) will be left normally open. No hardware modification is implemented.

Reason for Change

The Condensate Storage Transfer pumps are used to keep the LPCI system pressurized, thus ensuring the system is full and not susceptible to water hammer. Historically setpoints have drifted. This drifting has resulted in an Operations burden to maintain pressure manually, and an I&C/Maintenance equipment burden. Therefore it was proposed to bypass the regulators by changing the valve line-up.

Safety Evaluation

The change in the valve line-up, which will bypass the LPCI Keepfill pressure regulators and allow full Condensate Storage Transfer pressure is safe and not an unreviewed safety question.

PDCR Number	Title
1-099-95	Feedwater Coolant Injection (F[illegible]) System -Condensate Booster [illegible] Lube Oil Foundation And Related Electrical Component Anchorage

Description of Change

These modifications are complete. Components required to support the safety related function of the Feedwater Coolant Injection System (FWCI) have been structurally modified to ensure that their anchorage's are seismically adequate using the Generic Implementation Plan (GIP) approach.

Reason for Change

Components required to support the safety related function of the FWCI were found not to be Seismic Category I.

Safety Evaluation

The seismic qualification for all required equipment will not be affected by the change in methodology. Thus all the required equipment will continue to perform its safety function and there is no impact on the consequences of any previously evaluated accident. In addition, all equipment will remain seismically qualified as required. Thus the change does not create the possibility of an accident of a different type. The change in seismic evaluation methodology can only affect malfunctions due to seismic events. Thus it cannot introduce a different type of malfunction than previously evaluated. The use of GIP for the seismic evaluation of the FWCI components is safe and is not an unreviewed safety question.

PDCR Number	Title
1-100-95	Installation Of Filter Assemblies In The Instrument Air Supply lines To Air Operated Valves With QA Cat. 1 Backup Air Supply

Description of Change

This change is complete. The design change installs duplex filters in the Instrument Air supply lines, and will also install stainless steel flex hose, as well as additional in line check and test valves in the Instrument Air supply lines.

Reason for Change

Installation of the duplex filters in the Instrument Air supply line will improve the reliability of the air operated valves as well as the reliability of the check valves protecting backup air systems. The installation of the additional check valves in the Instrument Air supply lines will improve the reliability of the backup air systems, it will also provide design continuity with the latest backup air designs that require dual check valve installation. The existing rubber hoses and copper tubing at the connections to air cylinder are replaced with superior quality braided stainless steel hose and stainless steel tubing.

Safety Evaluation

The modification does not increase the probability of an accident or malfunction, and does not increase the consequences of any accident. The modifications improve the design of the system by providing clean, filtered air to the end use components. This modification is safe and does not create an unreviewed safety question.

PDCR Number	Title
1-002-96	Spent Fuel Pool Bulk Temperature Monitoring System

Description of Change

The installation of two thermocouples is complete. The two thermocouples are installed, one on the north wall approximately center line and one on the west wall approximately center line of the spent fuel pool. The thermocouples are "Type T", Copper-Constantan installed in thermal wells with the sensing unit located approximately 6 feet below the normal water level of the spent fuel pool. The thermal wells are constructed of 304 Stainless Steel and mounted seismically II/I to the side of the pool. New thermocouple cabling 2/C #16 is routed in flexible conduit from the thermocouples around the pool in a cable to existing conduit and cable trays. The cables are routed from the trays to the Control Room panel (CRP) 904 and existing temperature recorder TR-1040-2, channels 10 and 11 in the Control Room. The recorder has been programmed and the new channels are set to alarm at 125^{o} F and actuate the "SHTDN COOLING/FUEL POOL HITEMP" annunciator on CRP 904.

Reason for Change

Shortcomings of the fuel pool monitoring capabilities were identified by the Spent Fuel Storage Assessment Team. At issue, is the fact that equipment available to measure fuel pool temperature requires flow through the Fuel Pool Cooling system. Temperature measurement was provided on the suction side of the fuel pool pumps and on the discharge side of the fuel pool heat exchangers. Failure of the fuel pool pumps would leave the plant with no means of monitoring the fuel pool temperature.

Safety Evaluation

This design change is safe. The design prior to and after this modification provide the identical flow paths for the Fuel Pool Cooling System. This modification provides additional instrumentation relating to the Fuel Pool Cooling. This modification will not affect any previously analyzed accident or its consequences; nor will this change contribute to any new accident outside of those already analyzed. The margin of safety, as defined in the basis for any Technical Specifications, is not reduced. This modification is safe and does not create an unreviewed safety question.

PDCR Number	Title
1-003-96	Condensate Pump Seal Water Modification

Description of Change

These modifications have been completed. The modification upgraded the Condensate Pump Water Seal System from non-QA to QA Category 1. Hard piped the non-QA make-up water supply connection approximately 150 feet from the recently installed make-up water connection to one end of the abandoned seal water header above the "A" condensate booster pump. In addition, a line from the "C" booster pump suction header to the previously abandoned seal water supply line was hard piped. A pressure reducing station was removed and replaced with a new pressure reducing station which is connected to the seal supply pipe.

Reason for Change

The modifications improve the design of the existing system by providing cleaner water to the mechanical seals of the condensate pumps.

Safety Evaluation

These modifications have no impact on the accident analysis. The seal water supply system installation will have no adverse effect on the condensate pump performance. The margin of safety as defined in the basis for the Technical Specifications is not affected by these modifications of the seal water supply to the condensate pumps. The condensate pump design parameters will be unchanged and their design margin will not be altered. The modifications do not increase the probability of an accident or malfunction, and does not increase the consequences of any accident. These modifications are safe and do not create an unreviewed safety question.

PDCR Number	Title
1-006-96	New Unit Heater For Battery Room 18B

Description of Change

An explosion proof electric unit heater was installed in Battery Room "18B". The unit heater is seismically attached to the Battery Room wall. A power supply is provided from MCC D1. A fire seal is installed in the wall penetration. The unit is controlled by a separate, wall mounted thermostat, Chromalox Model WP-80EP, which meets standards for hazardous locations. The locally mounted thermostat is seismically attached to the Battery Room wall.

Reason for Change

Temperature in Battery Room "18B" has fallen below FSAR requirement of 65 deg. F., especially during cold weather outages. This condition results from the large volume of outdoor air required to dilute and exhaust the hydrogen generated from the battery charging. A dedicated unit heater with a locally wall mounted thermostat will ensure that FSAR requirements are not compromised.

Safety Evaluation

This design change has no adverse impact on the existing accident analysis. This change will have no effect on the margin of safety as defined in the basis for any Technical Specification. This design change is safe and does not create an unreviewed safety question. It will not affect accident mitigation nor will it increase the consequences of an accident or malfunction. It will provide additional assurance that the Battery Room temperature will be maintained within acceptable limits.

PROCEDURE CHANGES

Procedure Number	Title
EN 1066	Monitoring and Setting Reactor Building Negative Pressure
EOP 590.26	Containment Cooling During Accident Conditions
FO-OP-022-46955	Waste Water and Liner Gross Dewatering
HPP-50385-1	Blackness Testing
IC 409D	ATWS Power Supplies Functional and Calibration Test
IC 421G-1	Drywell Hydrogen (H2) and Oxygen (O2) Analyzer Functional Test.
ONP 503B	Loss of Station AC Power (LNP)
ONP 503B-S1	14E-S1 Loss of Normal Power
ONP 503B-S2	14E-S2 Loss of Normal Power
ONP 503B-12C	Loss of 480 Volt Bus 12C
ONP 503B-12D	Loss of 480 Volt Bus 12D
ONP 503B-12E	Loss of 480 Volt Bus 12E
ONP 503B-12F	Loss of 480 Volt Bus 12F
ONP 503B-14AB	Loss of 4160 Volt Buses 14A and/or 14B
ONP 503B-14E	Loss of 4160 Volt Bus 14E
ONP 503F	Degraded Voltage Conditions On Buses 14E/F
OP 302D	Lowering CRD Flow to the Reactor Vessel
OP 303D	Reactor Vessel Level Control With Cleanup System Out of Service and Fuel Pool Gates Installed

PROCEDURE CHANGES(Continued)

Procedure Number	Title
OP 305A	Operating Shutdown Cooling with Fuel Pool Cooling
OP 322	Emergency Service Water
OPS Form 319-1	Unit 1 Make-Up Water System
OP 335	LPCI Containment Cooling System
OPS Form 335-1	LPCI Valve Lineup
OP 328K	Filling Reactor Cavity With Fuel Gates Installed
SP 406PP	Liquid Radwaste Effluent Radiation Monitor Time Response Test
SP 617.1	Loss of Normal Power Relays
SP 671.5	Liquid Radwaste Discharge to LIS
SPROC 95 1 58	MOV Dynamic Testing of Blocking Valve 1-FW-4C
SPROC 96 1 11	Functional Test of Reactor Building HVT Fan Interlocks and Secondary Containment Isolation Valve Bypass Switches.
SPROC 96 1 15	MP1 Fuel Pool Rigging
SPROC 96-1-16-1	Fuse Control Plan Retest Requirements.
SPROC 96 1 19	SRV Electric Lift RFI Susceptibility
SPROC 96 1 21	Boraflex Blackness Testing Preparation
SPROC 96 1 25	Maintenance of Service Water Piping and Components Downstream of 1-SW-107 and repair of RBCCW Pump 'B'- IPTE (TB/RB Crosstie)

PROCEDURE CHANGES(Continued)

Procedure Number	Title
SPROC 95-1-25	Removal of Tri Nuc Filter from Spent Fuel Pool
SPROC 96-1-36	Control Rod Blade Recovery Plan - Control Rod Restoration and Tri Nuc Filter Removal
SPROC 96-1-27	Switchgear Overhaul Alignments for RFO15A
SPROC 96-1-03	Deenergization and Energization of busses 4160-14D and 4160-14F for RFO15A
SPROC 96-1-02	Deenergization and Energization of busses 4160-14C and 4160-14E for RFO15A
SPROC 95-1-41	Operator Actions for LNP Conditions During Implementation of PDCR 1-76-94 "LNP Logic Modification"
SPROC 96 1 31	Functional Test of Steam Tunnel Damper Solenoid Valve Installation
SPROC 96 1 35	Control Room Temperature Monitoring with Normal Ventilation Secured
SPROC 96 1 37	Control Rod Restoration -Uprighting
SPROC 96 1 38	Control Room In - Leakage Test
SPROC 96 1 49	Removal and Installation of Reactor Recirculation Pump Motor M8-51B(MTR)
MP 721.1	Overhaul of Reactor Recirculation Pumps
SPROC 96 1 133	Stored Control Rod Velocity Limiter Support Wire Manufacture
SPROC 96 1 134	Stored Control Rod Support Wire Manufacture and Installation

Procedure Number	Title
EN 1066	Monitoring and Setting Reactor Building Negative Pressure

Description of Change

This procedure provides instructions for Engineering personnel to perform a system walkdown and assessment, and to adjust Reactor Building negative pressure.

Reason for Change

Performance of this procedure will adjust the position of the Reactor Building supply fan inlet dampers to control building negative pressure.

Safety Evaluation

Adjusting the Reactor Building differential pressure will not affect the probability or consequences of an accident, the probability of a new accident or equipment malfunction, and that the margin of safety is not decreased, it is concluded that use of this procedure is safe and not an unreviewed safety question.

Procedure Number	Title
EOP 590.26	Containment Cooling During Accident Conditions

Description of Change

The current EOP 590.26 (Containment Cooling Accident Condition) requires that no more than one LPCI pump per train be running before the ESW pumps are started. This is achieved by tripping one LPCI pump if both pumps are running on the train. Additional minor changes are as follows:

Addition of an override which instructs the operator not to divert LPCI flow from injection to the vessel if adequate core cooling is not assured.

Addition of an override which instructs the operator to reenter the procedure if the conditions occur which realign the containment cooling. This override instructs the operator to restart torus cooling by reentering the procedure.

The current procedure has a step which requires adjustment of Diesel Generator (DG) and Gas Turbine Generator (GTG) loading to allow start of the ESW pumps. This statement is being replaced by allowing start of all available ESW pumps.

Reason for Change

These actions are taken to ensure that the DG and GT do not overload when ESW pumps are started. The procedure provides flexibility to the operator by not requiring that all LPCI pumps be tripped before starting ESW pumps.

Safety Evaluation

The changes do not adversely impact consequence of design basis accidents, including LOCA. The changes do not increase probability of failure of the DG or GT, nor have any adverse impact on the performance of the safety systems. It is concluded that the changes do not adversely impact the margin of safety. The changes are safe and not an unreviewed safety question.

Procedure Number	Title
FO-OP-022-46955	Waste Water and Liner Gross Dewatering

Description of Change

The ChemNuclear Systems Inc. (CNSI) mobile dewatering system was temporarily installed in the Millstone Unit 1 radwaste truck bay. The CNSI supplied mobile dewatering system will be used to partially dewater Ecodex® filter media that has been transferred into a radwaste liner. The CNSI system starts at the connection from the plant sludge transfer line at valve 1-SRW-51, and ends at the discharge of the waste water back to the plant at valve 1-SRW-61. The CNSI system consist of a dewatering/fill head assembly with temperature and level indications, high-high level switch, and a video camera. The system also includes a separate dewatering pump, a four level, valved, filter dewatering manifold with pressure gages and a decanting pump with wand.

Reason for Change

The CNSI equipment is being used to facilitate the emptying of the Filter Sludge Tank and Clean Up Filter Sludge Tank as part of the lower level radwaste remediation project.

Safety Evaluation

The changes will not affect any design basis accident, or its consequences. It will not contribute to any new accidents. The change is therefore safe and does not present an Unreviewed Safety Question.

Procedure Number	Title
HPP-50385-1	Spent Fuel Pool Blackness Test - 1st Campaign

Description of Change

In support of the Boraflex Surveillance Program at MP1, Blackness Testing will be performed on the Spent Fuel Racks. The testing will be performed in a selected number candidate storage locations that have experienced continuous fuel storage service.

Reason for Change

Blackness Testing will be performed on the spent fuel racks to confirm the presence, identify the gaps and evaluate the integrity of the Boraflex material for continued utilization as a neutron absorber.

Safety Evaluation

All of the design qualifications, attributes and parameters of the fuel racks to store nuclear spent fuel, maintain fuel assemblies and in a safe subcritical configuration of Keff < .90 remain valid, unaffected and unchanged. Additionally, this equipment has been successfully utilized at both MP2 and MP3. Therefore this test is safe and is not an unreviewed safety question.

Procedure Number	Title
IC 409D	ATWS Power Supplies Functional and Calibration Test.

Description of Change

I&C procedure IC409D, provides instructions for performing functional test and calibration of the ATWS input monitoring circuits and loss of power trip inhibit. Instructions are also provided for checking, and adjusting where possible all system power supplies.

Reason for Change

IC 409D is being performed as a retest to return a failed Division to an operating status. The performance of IC 409D will reduce the total time one Division is unavailable and the time the ATWS is prone to a single failure.

Safety Evaluation

IC 409D will have no adverse effect on the probability of occurrence of previously evaluated accidents nor will it introduce the possibility of an accident of a different type than previously analyzed. There is no impact on the margin of safety and has been determined to be safe and does not involve an unreviewed safety question.

Procedure Number	Title
IC 421G-1	Drywell Hydrogen (H_2) and Oxygen (O_2) Analyzer Functional Test.

Description of Change

Revision 15 of the subject procedure was developed to incorporate changes resulting from replacement of recorder R-9-75. The changes made to the procedure are,

- Deleted all reference to the "H2 RANGE" switch.
- Moved checking of CRP annunciation to span adjustment section.
- Deleted calibration of recorder indication and alarms due to recorder replacement.
- Added reference to manual for new recorder.
- Added attachment for surveillance briefing.
- Modified procedure to conform to format of DC2, rev 1.

Reason for Change

IC 421G "Drywell Hydrogen (H_2) and Oxygen (O_2) Analyzer Functional Test" provides instructions for the functional testing and calibration of the containment atmosphere H_2/O_2 analyzer, including the PAM 102 indicators, the H_2/O_2 recorder R-9-75 on CRP 925 and Control Room computer points.

Safety Evaluation

Based on the fact that existing plant conditions do not require the H_2/O_2 analyzer for safety (primary containment is open to atmosphere) and the fact that the operation of the analyzer in accordance with IC 421G, rev 15 "Drywell Hydrogen (H_2) and Oxygen (O_2) Analyzer Functional Test" will not effect the probability or consequence of a fuel handling accident or the probability of a new accident or equipment malfunction, and does not decrease the margin of safety, it is concluded that the conditions of the procedure are safe and do not increase public risk. Field validation of IC 421G, rev 15, is not an unreviewed safety question.

Procedure Number's	Title
ONP 503B	Loss of Station AC Power (LNP)
ONP 503B-S1	14E-S1 Loss of Normal Power
ONP 503B-S2	14E-S2 Loss of Normal Power
ONP 503B-12C	Loss of 480 Volt Bus 12C
ONP 503B-12D	Loss of 480 Volt Bus 12D
ONP 503B-12E	Loss of 480 Volt Bus 12E
ONP 503B-12F	Loss of 480 Volt Bus 12F
ONP 503B-14AB	Loss of 4160 Volt Buses 14A and/or 14B
ONP 503B-14E	Loss of 4160 Volt Bus 14E

Description of Change

In Cycle 15 refueling outage, the Loss of Normal Power (LNP) logic was modified to allow capability to mitigate a partial LNP. The above procedures are developed with a general philosophy on how and when to allow cross-tying two safety divisions or specific buses from two divisions.

Reason for Change

A need was identified to provide a comprehensive procedural guidance on how and when to cross-tie electrical power from one division to another.

Safety Evaluation

The procedures neither increase the probability of occurrence, nor consequences of design basis accidents. These procedures do not increase the probability of failure of the Diesel Generator or Gas Turbine Generator. No credible malfunctions are created. These procedures utilize the existing design feature (cross-tie breakers) to allow flexibility in mitigating scenarios involving multiple equipment failures. Considering all of these arguments, it is concluded that these procedures do not decrease the margin of safety, and do not constitute an Unreviewed Safety Question.

Procedure Number	Title
ONP 503F	Degraded Voltage Conditions On Buses 14E/F

Description of Change

The actions performed per ONP 503F are to respond to a degraded voltage condition. This procedure only applies to non-accident conditions, since a degraded voltage condition coincident with an accident signal will initiate a Loss of Normal Power and ONP 503B will apply. Additionally, a degraded on-site bus voltage condition due to degraded grid voltage is not considered credible with the main generator on-line and the on-site buses powered from the Normal Station Service Transformer.

Reason for Change

This is a new procedure. It is required to provide actions to be taken by the operator in the event of a degraded off-site power source under non-accident conditions and specify the time and voltages condition permitted before an operator action is required to disconnect the Class 1E buses from the degraded off-site source.

Safety Evaluation

Performance of this procedure does not increase the probability of a loss of AC power. This procedure has been independently reviewed and validated to ensure that the specified actions are safe and do not place the plant in an unsafe or unanalyzed condition. This procedure is safe and not an unreviewed safety question.

Procedure Number	Title
OP 302D	Lowering CRD Flow to the Reactor Vessel

Description of Change

OP302D, "Lowering CRD Flow to the Reactor Vessel," reduces system flow by throttling 1-CRD-11A/B and 1-CRD-7. OP302D allows the total flow to be decreased to 60 gpm by decreasing system flow to 40 gpm. This procedure is applicable during SHUTDOWN or REFUEL modes.

Reason for Change

When the plant is in SHUTDOWN or REFUEL modes, lowering the CRD pump flow may be desirable because the normal reject path through the Feedwater system may not be available. Any water added to the vessel via CRD would have to be rejected to the Waste Collector Tank to maintain vessel level. This limits plant operation as the processing capabilities of Radwaste can not keep up with the amount of water which would be rejected during normal operation. Limited CRD system flow is also desirable during the vessel operational leak test (SP681) as normal CRD flows would significantly increase vessel cooldown.

Safety Evaluation

The performance of OP302D, "Lowering CRD Flow to the Reactor Vessel," does not involve an unreviewed safety question. OP302D does not change the CRD system's ability to perform its safety function. The CRD system will be operated with the design boundaries of the equipment.

Procedure Number	Title
OP 303D	Reactor Vessel Level Control With Cleanup System Out of Service and Fuel Pool Gates Installed

Description of Change

The new procedure will use the Shutdown Cooling (SDC) and Spent Fuel Pool Cooling (SFPC) systems to remove excess water from the reactor vessel

Reason for Change

The procedure provides instructions for removing excess water from the reactor vessel via the SDC and SFPC systems when the plant is in shutdown with the Reactor Water Cleanup system out of service and the fuel pool gates installed.

Safety Evaluation

This procedure will improve the capability to remove excess water from the Spent Fuel Pool and the reactor vessel. While the procedure diverts cooling water from the Spent Fuel Pool it will maintain water level and pool temperature within required limits. Thus it is concluded that the changes are safe and not an unreviewed safety question, in that:

- the probability of occurrence or consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased;

- the possibility of an accident or malfunction of a different type than evaluated in the safety analysis report is not created;

- the margin of safety as defined in the basis for any technical specification is not reduced.

Procedure Number	Title
OP 305A	Operating Shutdown Cooling with Fuel Pool Cooling

Description of Change

These changes are being incorporated into section 4.2 of OP305A, rev 2:

> A section is being added to the procedure for the use of Shutdown Cooling Loop "B" to supplement Fuel Pool Cooling when the Shutdown Cooling System is initially out of service and the fuel pool gates are closed. This section is only applicable when (1) primary containment integrity is not required, (2) the fuel pool gates are installed and (3) all fuel is in the fuel pool has been reviewed.

Reason for Change

The change to the plant operating procedure permits greater flexibility in providing heat removal for the Spent Fuel Pool.

Safety Evaluation

This procedure change will improve the capability to remove decay heat from the Spent Fuel Pool. This change allows the flexibility to use either train of Shutdown Cooling to cool the Spent Fuel Pool when the fuel pool gates are installed and the primary containment integrity is not required. Thus it is concluded that the changes are safe and not an unreviewed safety question, in that:

- the probability of occurrence or consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased;
- the possibility of an accident or malfunction of a different type than evaluated in the safety analysis report is not created;
- the margin of safety as defined in the basis for any technical specification is not reduced.

Procedure Number	Title
OP 322	Emergency Service Water
OPS Form 319-1	Unit 1 Make-Up Water System
OP 335	LPCI Containment Cooling System
OPS Form 335-1	LPCI Valve Line-up

Description of Change

The Condensate Storage Transfer pumps are used to keep the LPCI system pressurized, thus ensuring the system is full and not susceptible to water hammer. In the current design, pressure regulators are used to reduce CST pump discharge pressure from ~100 psig to ~35 psig. This modification bypasses the regulators. This will be achieved by changing the valve line-up. The inlet to the LPCI Keepfill PCV's (1-MW-82) and the outlet from the PCV's (1-MW-84) will be left normally closed and the bypass valve (1-MW-85) will be left normally open. Essentially this will change the keepfill pressure for the LPCI system from ~35 psig to ~100 psig.

Reason for Change

Historically the pressure regulator setpoints have drifted, and have caused an Operations burden to maintain pressure manually, and an I&C/Maintenance equipment burden.

Safety Evaluation

This change in the valve line-up, which bypasses the LPCI keepfill pressure regulators and allows full Condensate Storage Transfer pressure is safe and not an Unreviewed Safety Question.

Procedure Number	Title
OP 328K	Filling Reactor Cavity With Fuel Gates Installed

Description of Change

This procedure uses the Shutdown Cooling and Condensate Transfer systems for adding water inventory to the reactor cavity.

Reason for Change

The procedure provides instructions for providing makeup to the reactor cavity using the Condensate Transfer System through the Shutdown Cooling System. The Shutdown Cooling System can either be in service to the reactor vessel or off during the performance of this procedure.

Safety Evaluation

This procedure change will improve the capability to add water to the reactor cavity when the fuel pool gates are installed. While the procedure will reduce the cooling flow from the reactor cavity during fill operations when shutdown cooling is in service, the effect of this reduction will be negligible on the bulk temperature of the reactor cavity. Thus it is concluded that the changes are safe and not an unreviewed safety question, in that:

- the probability of occurrence or consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased;
- the possibility of an accident or malfunction of a different type than evaluated in the safety analysis report is not created;
- the margin of safety as defined in the basis for any technical specification is not reduced.

Procedure Number

SP 406PP

Title

Liquid Radwaste Effluent Radiation Monitor Time Response Test

Description of Change

This is a new procedure, patterned after existing approved procedures. The detector is placed in a shepard calibrator with a radiation field as close to normal background as possible. The monitor high level trip is adjusted to 67% of the decade step change that will be initiated in the shepard calibrator. This will be approximately 1 time constant value. A one decade radiation field step change will be initiated in the shepard calibrator by quickly removing the X10 paddle. The time response of the meter obtained by measuring the time from when the X10 paddle is moved until high alarm is received and solving the equation for response time. Engineering will determine if the value is acceptable as defined in M1-EV-960024, rev. 0.

Reason for Change

I&C procedures SP406PP, "Liquid Radwaste Effluent Radiation Monitor Time Response Test" provides instructions for performing a time response test of the associated monitor.

Safety Evaluation

SP406PP, "Liquid Radwaste Effluent Radiation Monitor Time Response Test" has been reviewed and determined to be safe and not involve an unreviewed safety question. There is no impact on the margin of safety since the loss of this monitor was already evaluated and accepted by the NRC when the existing Technical Specifications and LCO was approved. This procedure does not introduce the possibility of an accident or malfunction of a different type than previously analyzed.

Procedure Number	Title
OPS SP 617.1	Loss of Normal Power Relays.

Description of Change

This procedure functionally test all relays, breakers, and contactors required to operate in a loss of normal power (LNP) condition and verifies automatic starting and load sequencing ability of the Diesel Generator (DG) and Gas Turbine Generator (GTG). The Changes are summarized as follows:

- Change relay designations and contacts
- Add test of auto-start of the "C" and "D" Service Water Pumps on a LNP to the opposite train.
- Add test of auto-start "A" and "B" Turbine Building Secondary Closed Cooling Water Pumps on a LNP to the opposite train.
- Provide separate S1 and S2 sections for verification of non-essential loads seal-out.
- Separate the LNP test for 14E-S1 and 14F-S2.
- Add test of the 12C to 12D tie breaker S1 LNP load shed trip.
- Add steps to remove fuel pool filter and demineralizer from service prior to performing Bus 12E and 12F load shed, and subsequent steps for placing fuel pool cooling and fuel pool filter or demineralizer back in service.

Reason for Change

The previous design required a loss of both Buses 14E and 14F to initiate an LNP. With the design changes complete, the S1 and S2 LNP systems will be separate and independent such that a loss of Bus 14E will generate a S1 LNP and a loss of 14F will generate a S2 LNP.

Safety Evaluation

The performance of LNP relay testing as specified in SP617.1 is determined safe and not an unreviewed safety question. There is no reduction in the margin of safety as required by the Technical Specifications.

Procedure Number	Title
SP 671.5	Liquid Radwaste Discharge to LIS

Description of Change

Surveillance Procedure, SP 671.5, provides instructions for the controlled and monitored discharges of radioactive waste effluent through designated flow paths. This procedure further provides instructions to test radiation effluent discharge monitors and to verify flow rates to ensure all equipment is operational prior to performing discharges.

Reason for Change

Controlling the effluent releases via this Surveillance Procedure ensures that all discharges are controlled and monitored to meet the licensing requirements.

Safety Evaluation

The issuance of this procedure will not affect any design basis accident, or its consequences. It will not contribute to any new accidents beyond those already analyzed. The procedure is therefore safe and does not present an unreviewed safety question.

Procedure Number	Title
SPROC 95 1 58	MOV Dynamic Testing of Blocking Valve 1-FW-4C

Description of Change

This procedure will pressurize the piping between 1-FW-4C and 1-FW-6C with 1-FW-5C open. Once the piping is pressurized to approximately 1894 psig the 1-FW-4C valve will be opened from the Control Room. Votes equipment will be used to collect thrust data as the 1-FW-4C strokes open. This test will be repeated until three tests with good data have been completed.

Reason for Change

Generic Letter 89-10 and Northeast Utilities Motor Operated Valve Program require dynamic testing of safety related valves.
The Design Basis Review (DBR), requires 1-FW-4C to close against a differential pressure of 1894 psig with a flowrate of 1736 GPM. PI-11, Determination of In-Situ test capability, requires a minimum of 50% of the specified differential pressure in the DBR to be attained before a dynamic test is considered valid. Because of the limitations on 1-FW-5C, Feedwater Regulating Valve, no more than 35% of the DBR differential pressure is attainable during the performance test of 1-FW-4C.

Safety Evaluation

The Feedwater Coolant Injection (FWCI) System is not required to be operable in cold shutdown, therefore this procedure has no effect on the probability of occurrence of a previously evaluated malfunction on equipment important to safety. The procedure does not perform any actions or expose any components to any conditions which would increase the consequences of a previously evaluated malfunction of equipment important to safety. There is no credible possibility of creating an accident of a different type than previously evaluated. This procedure does not impact the margin of safety. The procedure is safe and not an unreviewed safety question.

Procedure Number	Title
SPROC 96-1-11	Functional Test of Reactor Building HVT Fan Interlocks and Secondary Containment Isolation Valve Bypass Switches.

Description of Change

This tests the modifications by providing several initiating signals at different times and verifying the ventilation systems as necessary. It also verifies the isolation of the Reactor Building Ventilation System (RBVS) fans on a Freezestat actuation. Finally, it verifies proper operation of all component control circuits, including the newly installed bypass switches.

Reason for Change

SPROC 96 1 11, "Functional Test of Reactor Building HVT Fan Interlocks and Secondary Containment Isolation Valve Bypass Switches", provides instructions to functionally test modifications made to the control circuitry of several heating and ventilation components. The control circuits of the components were modified to provide reactor building (supply and transfer fans) and steam tunnel isolation upon receipt of a group 2 or a Freezestat actuation (RBVS fans only) signal.

Safety Evaluation

This procedure is safe and does not constitute an unreviewed safety question. There is no reduction in margin of safety as required by the Technical Specifications, and the possibility of a malfunction of a different type than previously evaluated is not created.

Procedure Number	Title
SPROC 96 1 15	MP1 Fuel Pool Rigging

Description of Change

This procedure, SPROC 96-1-15 is used for lifting the Fuel Pool Gates including Rigging Hardware, Crane Equipment and calculation methods, and fully complies with the single failure proof criterion of NUREG 0612.

Reason for Change

The MP1 Fuel Pool Gates are considered heavy load due to their weight being greater than that of a Fuel Assembly and its handling tool. Because the Gates must transverse an area over the Spent Fuel Pool where irradiated Spent Fuel is stored, and analysis was performed to develop a single failure proof load path system for lifting the gates.

Safety Evaluation

This procedure for movement of the fuel pool gates into the spent fuel pool utilizing a redundant load path methods is safe. This procedure will not contribute to any previously analyzed accident or it's probability or consequences; nor will the procedure contribute to any accident outside those already analyzed. The margin of safety as defined in the basis technical specifications is not reduced. There are no unreviewed safety questions associated with this procedure.

Procedure Number

SPROC 96 1 16-1

Title

Fuse Control Plan Retest Requirements.

Description of Change

The fuses are tested by inserting an initiating signal (loss of voltage) to the circuit, thus actuating all relays fed from the fuses in question and load shedding particular loads from busses 12C and 12F.

Reason for Change

SPROC 96 1 16, "Fuse Control Plan Retest Requirements" provides instructions for performing a functional test to verify the adequacy of fuses replaced under the Unit 1 Fuse Control Plan.

Safety Evaluation

The performance of the testing does not change any system or component design functions or parameters, therefore, the potential for a new unanalyzed accident is not created. By performing the fuse test on only one bus at a time, the opposite division remains operable and there is no reduction in the margin of safety as required by the Technical Specifications. It is concluded that SPROC 96 1 16 is safe and does not constitute an unreviewed safety question.

Procedure Number	Title
SPROC 96 1 19	SRV Electric Lift RFI Susceptibility

Description of Change

SPROC 96-1-19 establishes pressure to the Safety Relief Valves (SRV) electric lift transmitters at approximately 1055 psig. This establishes operating conditions with the expected threshold noise tolerance. SRV circuits are monitored with either clamp on or high impedance test equipment to prevent test equipment faults from affecting safety related circuits. Plant radios are keyed as directed by the test engineer and CHAR personnel and the impact on the SRV circuits noted.

Reason for Change

96-1-19, "SRV Electric Lift RFI Susceptibility" establishes plant conditions to allow safe testing of the SRV Electric Lift System for EMI/RFI.

Safety Evaluation

SPROC 96-1-19 will not adversely impact the consequences of any previously analyzed accidents. There are no new accidents postulated by the new test procedure, and there is no impact on the margin of safety. This procedure is safe and not an unreviewed safety question.

Procedure Number	Title
SPROC 96 1 21	Boraflex Blackness Testing Preparation

Description of Change

SPROC 96-1-21, Boraflex Blackness Testing Preparation, provides guidance for movement of fuel assemblies in the Spent Fuel Pool (SFP) to support performance of Blackness testing that is being performed during the current refueling outage.

Reason for Change

This procedure provides guidance for configuring the plant systems to meet license requirements for the movement of fuel assemblies in the Spent Fuel Pool. Guidance is provided to open the steam tunnel High Energy Line Break (HELB) doors during performance of the secondary containment tightness test and during fuel movement. This action ensures that the steam tunnel pressure is equalized with reactor building pressure during Standby Gas Treatment (SGBT) operation. The procedure will ensure that the Reactor Building is at the required negative pressure.

Safety Evaluation

SPROC 96-1-21, "Boraflex Blackness Testing Preparation", is safe and does not involve an unreviewed safety question.

Procedure Number

SPROC 96 1 25

Title

Maintenance of Service Water Piping and Components Downstream of 1-SW-107 and repair of RBCCW Pump 'B'- IPTE (TB/RB Crosstie)

Description of Change

The SPROC provides the necessary procedural guidance to align Turbine Building Closed Cooling Water (TBCCW) to supply Reactor Building Closed Cooling Water (RBCCW) with cooling water for the purpose of removing decay heat from the fuel pool while using the TBCCW to RBCCW cross-tie piping.

Reason for Change

Maintenance is required to be performed on Service Water piping and components downstream of RBCCW Service Water Supply Stop Valve, 1-SW-107 which will remove Service Water cooling from the RBCCW heat exchangers. During this evolution the 'B' RBCCW pump will be overhauled.

Safety Evaluation

Performance of this procedure will not increase the probability of a malfunction of equipment important to safety. The RBCCW does not provide or support any function or equipment necessary for accident mitigation or recovery except primary containment isolation. Performance of the SPROC is restricted to the cold shutdown condition when primary containment is not required. Administrative controls are contained within the SPROC to direct that sampling be performed to detect any inadvertent contamination of the RBCCW, TBCCW or SW systems. This will provide adequate assurance that performance of the SPROC will not increase the potential for unmonitored release. The performance of this procedure will not reduce the margin of safety as per the Technical Specifications. The performance of this procedure utilizing the TBCCW to RBCCW cross-tie is safe and does not involve an unreviewed safety question.

Procedure Number	Title
SPROC 95-1-25	Removal of Tri Nuc Filter from Spent Fuel Pool
SPROC 96-1-36	Control Rod Blade Recovery Plan - Control Rod Restoration and Tri Nuc Filter Removal

Description of Change

The work performed under SPROC 95-1-25 and SPROC 96-1-36 consist of the following:

1) Before lowering of the Control Rod Blades (CRB's) velocity limiter ends, a redundant rigging system will be placed around the lower group of CRB's velocity limiters. In addition, a cable grip will be attached to the cable that is tangled around the velocity limiters and supported from Tri-Nuclear skid to facilitate future removal operations.
2) CRB's velocity limiter ends will be lowered to the fuel pool floor using the come-alongs attached to the refueling platform while concurrently lowering the Tri-Nuclear filter skid with the overhead crane auxiliary hook.
3) After the CRB's are supported on the Fuel Pool floor, the temporary cables attached to the velocity limiter ends of the CRB's from the come-alongs will be removed from the come-along and securely attached to the east equipment rail for future removal operations. The cable from the Tri-Nuclear filter skid will then be cut and secured to the east equipment rail.
4) The Tri-Nuclear filter skid will then be removed from the pool

Reason for Change

During an attempt to remove the Tri-Nuclear skid from the floor of the Spent Fuel Pool, cables which were tangled in the skid apparently snagged some control rod blades and displaced them from their stored positions along the east wall of the pool.

Safety Evaluation

All of the design parameters required to assure that stored spent fuel is maintained in a safe, coolable, subcritical configuration remain within their acceptance limits. Therefore, this modification is safe and does not constitute an Unreviewed Safety Question.

Procedure Number	Title
SPROC 96-1-27	Switchgear Overhaul Alignments for RFO15A
SPROC 96-1-03	Deenergization and Energization of buses 4160-14D and 4160-14F for RFO15A
SPROC 96-1-02	Deenergization and Energization of buses 4160-14C and 4160-14E for RFO15A
SPROC 95-1-41	Operator Actions for LNP Conditions During Implementation of PDCR 1-76-94 "LNP Logic Modification"

Description of Change

The Loss of Normal Power (LNP) logic modifications involve splitting the existing interdependent logic, into separate independent S1 and S2 LNP systems. Since certain changes directly affect operability of both emergency power sources, the circuit modifications will be made in several phases to ensure Technical Specification and shutdown risk management requirements are met and to take advantage of the scheduled maintenance outages for emergency power sources.

Reason for Change

Temporary modifications are needed for the implementation of LNP design changes.

Safety Evaluation

It is concluded that the performance of this work under the controls and conditions that are in place, no unsafe conditions exist and no Unreviewed Safety Question is created.

Procedure Number	Title
SPROC 96 1 31	Functional Test of Steam Tunnel Damper Solenoid Valve Installation

Description of Change

SPROC 96-1-31, "Functional Test of Steam Tunnel Damper Solenoid Valve Installation" provides instructions to functionally test the steam tunnel damper solenoid valves replaced by DCN# DM1-S-0891-96.

Reason for Change

Proper operation of the replacement solenoid valves is verified by testing the ability of each valve to permit flow in the forward and reverse directions and also to vent.

Safety Evaluation

Based on the fact that SPROC 96-1-31 "Functional Test of Steam Tunnel Damper Solenoid Valve Installation" is performed when secondary containment integrity is not required and on the fact that the procedure will not affect the probability or consequence of a fuel handling accident or the probability of a new accident or equipment malfunction, and does not decrease the margin of safety, it is concluded that the conditions of the procedure are safe and do not increase public risk. Performance of SPROC 96-1-31 is not an unreviewed safety question.

Procedure Number

SPROC 96 1 35

Title

Control Room Temperature Monitoring with Normal Ventilation Secured

Description of Change

This procedure provides instructions on collecting Control Room temperature and damper closure data to ensure equipment reliability and Control Room habitability with the ventilation system isolated.

Reason for Change

Isolation of the ventilation system is being performed in order to determine damper closure times and Control Room heat-up times to support isolation of the Control Room ventilation system during a radiological event. Isolation of the ventilation system is necessary to ensure that the Control Room in-leakage during a Design Basis Accident is less than 1000 cfm.

Safety Evaluation

Based on this evaluation stopping and starting the control room ventilation system and closing and opening the outside air intake isolation damper will not affect the probability or consequences of an accident, the margin of safety is not decreased. It is therefore concluded that the use of this procedure is safe and not unreviewed safety question.

Procedure Number	Title
SPROC 96 1 37	Control Rod Restoration - Uprighting

Description of Change

The work performed under SPROC 96-1-37 consist of the following:

- Utilizing the refueling platform mounted auxiliary hoist, lower and attach grapple to the Control Rod Blade (CRB) lifting bail.
- Remove previously installed support wires attaching CRB lifting bail to the east equipment rail and those attaching the CRB velocity limiter to the refueling platform come-alongs.
- Cut the lanyard wire that is entangled around the CRB velocity limiter and raise the six CRB's to a vertical position over cask laydown area in Spent Fuel Pool (SFP).
- move the six CRB's from the cask laydown area while in a vertical position and suspended approximately two inches off the SFP floor and place the CRB's in their original storage position along the east wall of the SFP. Install a new support cable and hook to each lifting bail and secure the support cable to the east equipment rail.
- raise one CRB that is noticeably higher than the other CRB's in the north group and place it on the SFP floor. Install the new support cable and hook to this CRB and remove existing cable and hook attachment.
- Install new support cables and hooks to the remaining thirteen vertical CRB's stored along the east fuel pool wall and remove the existing cable and ring attachment.

Reason for Change

During an attempt to remove the Tri-Nuclear filter skid from the floor of the SFP, cables which were tangled in the skid apparently snagged some control rod blade and displaced them from their stored positions along the east wall of the pool.

Safety Evaluation

This modification does not introduce a new failure mode which could result in the possibility of an accident or malfunction of a different type than previously evaluated. There is no impact on the margin of safety. This procedure is safe and not an unreviewed safety question.

Procedure Number	Title
SPROC 96 1 38	Control Room In - Leakage Test

Description of Change

This procedure provides instructions on collecting Unit 1 Control Room in-leakage data using a tracer gas method to ensure equipment reliability and Control Room Habitability with the ventilation system isolated, and the recirculation mode with a maximum 500 cfm +/- 10% outside air intake.

Reason for Change

This test is being performed in order to determine the potential in-leakage into the Control Room during a radiological event.

Safety Evaluation

Performance of this procedure, stopping and starting the Control Room ventilation system and closing and opening the outside air intake isolation damper will not affect the probability or consequences of an accident or the probability of a new accident or equipment malfunction, and the margin of safety is not decreased. It is therefore concluded that the use of this procedure is safe and not an unreviewed safety question.

Procedure Number	Title
SPROC 96 1 49	Removal and Installation of Reactor Recirculation Pump Motor M8-51B(MTR)
MP 721.1	Overhaul of Reactor Recirculation Pumps

Description of Change

The disassembly of the pump and required relocation of the motor are considered to be maintenance activities, although the requirements to eliminate existing interference's to the motor movement and stabilization of the piping system to support the move require the implementation of interim design changes to the plant. The requirements to isolate the Reactor Building Closed Cooling Water (RBCCW) system during motor movement and to maintain the Reactor Recirculation piping seismically qualified at all times during the work will ensure that any safety related equipment will not be affected by this work.

Reason for Change

The pump disassembly is required to allow replacement of the casing gasket which was found to be leaking and allow the removal and replacement of the pump shaft to support a shaft cracking concern examination.

Safety Evaluation

These changes will in no way impact or compromise the safety of the plant and is safe. Maintenance is completed with the plant in the cold shutdown condition with the core offloaded, no equipment required for plant safety will be affected or compromised by this activity. The work will be completed with the isolation of the Reactor Recirculation pipework from the Reactor vessel and cavity by a single valve isolation and the Spent Fuel Pool will have a second isolation barrier from the work by the installation of the Spent Fuel Pool Gates at all times during the work activity. There are no accidents previously analyzed that would be affected by this maintenance activity nor will the performance of this activity generate a condition that has not been previously analyzed. The conduct of this maintenance activity will not result in challenges to any of the protective barriers to the plant or affect the margin of safety previously analyzed. All precautions have been taken to provide for maximum safety during the performance of this maintenance and the result of any component malfunction associated with the work will not affect any safety systems required in the cold shutdown core offloaded condition. This work is safe and does not constitute an Unreviewed Safety Question.

Procedure Number	Title
SPROC 96 1 133	Stored Control Rod Velocity Limiter Support Wire Manufacture

Description of Change

The velocity limiter end of a group of five CRB's are tied together and supported by a cable hanging from the filter skid. In an effort to place these CRB's into a more secure configuration pending return to their original storage locations, supplementary rigging is installed which ties the bottom ends of two of these CRB's off to one of the bridge girders of the refueling platform.

Reason for Change

During an attempt to remove the Tri-Nuclear filter skid from the floor of the Spent Fuel Pool, cables which were tangled in the skid apparently snagged some control rod blades and displaced them from their stored positions along the east wall of the pool.

Safety Evaluation

All of the design parameters required to assure that the stored spent fuel is maintained in a safe, coolable, subcritical configuration remain within their acceptance limits. Therefore, this modification is safe and does not constitute an Unreviewed Safety Question.

Procedure Number	Title
SPROC 96 1 134	Stored Control Rod Support Wire Manufacture and Installation

Description of Change

A minimum of seven new support wires will be installed to the lifting bale of the Control Rod Blades.

Reason for Change

The Control Rod Blades were repositioned due to the lifting of the Tri-Nuclear filter.

Safety Evaluation

All of the design parameters required to assure that stored spent fuel is maintained in a safe, coolable, subcritical configuration remain within their acceptance limits. Therefore this procedure is safe and does not constitute an unreviewed safety question.

TECHNICAL REQUIREMENTS MANUAL CHANGES

TRMCR Number	Title
OPS 273-7.4	Control Room Habitability Technical Requirements
OPS 273-7.5	Technical Requirements ATWS System.
OPS 273-3/4.5	Technical Requirements Manual - Safety Related Air Handling Units.
TRM 273-7.6	Reactor Building Negative TRM

TRMCR Number	Title
OPS 273-7.4	Control Room Habitability Technical Requirements

Description of Change

Incorporated are the following requirements associated with Self-Contained Breathing Apparatus (SCBA) and Control Room Habitability:

- Medical Requirements
- Training Requirements
- Number of SCBA's and Replacement Air Cylinders required
- Number of 1 hour replacement cylinders are being increased from 6 to 30 to provide a 6 hour air supply for six Control Room operators
- Use of Scott IIA SCBA's and 30 min. bottles for Scott IIA SCBA's is discontinued and all references removed from the Technical Requirements Manual (TRM)
- Voice amplification system is required to be installed on all SCBA's credited for each Control Room Habitability

Reason for Change

Implementation of the TRM will improve plant safety by providing additional administrative controls to better insure the Control Room Habitability SCBA's are available.

Safety Evaluation

Implementation of the TRM Section revision will improve the margin of safety by providing administrative controls to insure the SCBA are maintained available. The addition of the TRM Section revision is safe and not an unreviewed safety question.

TRMCR Number	Title
OPS 273-7.5	Technical Requirements ATWS System.

Description of Change

The ATWS system initiates an Alternate Rod Insertion (ARI) and a Recirculation Pump Trip (RPT) to aid in preserving the integrity of the fuel cladding following operational transients when a scram does not, but should occur.

Reason for Change

OPS Form 273-7.5, "Technical Requirements ATWS System" establishes operability requirements and Limiting Conditions for Operation as a new section of the Technical Requirements Manual. It ensures the reliability of the ATWS system and associated equipment required by Confirmatory Order to Millstone Unit 1 Technical Specifications and NRC regulation (10CFR50.62) but not addressed within plant Technical Specifications.

Safety Evaluation

The availability of the ATWS system is improved. The ATWS system lessens the consequences of previously evaluated malfunctions of equipment important to safety. The ATWS Technical Requirements do not introduce any malfunctions of a different type than previously analyzed. There are no new accidents postulated by the ATWS Technical Requirements. There is no impact on the margin of safety and does not involve an unreviewed safety question.

TRMCR Number	Title
OPS 273-3/4.5	Technical Requirements Manual - Safety Related Air Handling Units.

Description of Change

New Technical Requirements for the safety related air handling units for the Feedwater Coolant Injection system (FWCI), Emergency Diesel Generator (EDG), and LPCI and Core Spray Systems.

Reason for Change

Operations department personnel have been trained on these requirements for the past several years, however the requirements were not previously incorporated into the Technical Requirements Manual (TRM).

Safety Evaluation

The establishment of these Technical Requirements does not create the possibility of an accident or malfunction of a different type than previously evaluated. The margin of safety will not be decreased since these Technical Requirements will only ensure that equipment remains able to perform its safety function.

TRMCR Number	Title
TRM 273-7.6	Reactor Building Negative TRM

Description of Change

The TRM provides for operational requirements if the Reactor Building internal pressure is not at least -.5 in. wg. with respect to ambient when secondary containment is required. The TRM requires operators to restore normal ventilation system operation and building minimum negative pressure or to demonstrate secondary containment integrity with the Standby Gas Treatment (SBGT) System. If these requirements cannot be met, the TRM specifies to discontinue fuel movement and, if applicable, have the reactor in cold shutdown within 24 hours.

Reason for Change

The requirement to maintain secondary containment negative is necessary because the Unit 1 accident analysis assumes the Reactor Building is negative prior to SBGT initiation since it does not have adequate capacity to draw down the building in the required time if the building is positive.

Safety Evaluation

Based on the fact that conditions of the TRM will not affect the probability or consequences of an accident, or the probability of a new accident or equipment malfunction, and that the margin of safety is not decreased, it is concluded that the conditions of the TRM are safe and is not an unreviewed safety question.

SETPOINT CHANGES

Setpoint Change Number	Title
M1-95-027	IRM High Flux Setpoint Change
M1-95-031	Containment Spray Interlock Setpoint Change
M1-95-033	Drywell High Pressure RPS Switch Setpoint Change
M1-95-034	Scram Discharge Volume RPS Level Switch Setpoint
M1-95-036	LPCI/Core Spray 300-350 psi Interlock (PS-263-52A,B)
M1-96-003	APRM Hi Flux Clamp
M1-96-006	APRM Load Reject Setdown Setpoint
M1-96-037	LPCI/Core Spray 300-350 psi Interlock (PS-263-52C,D)

Setpoint Change Number

M1-95-027

Title

IRM High Flux Setpoint Change

Description of Change

This change will change the Hi Hi Flux setpoint on the Intermediate Range Monitors (IRM's) 750-7A,B,C,D,E,F,G,H to 115% from 118%. This setpoint provides input to the Reactor Protection System (RPS) and causes a scram on hi hi flux. The Technical Specifications limit is 120%.

Reason for Change

The setpoint verification program has included errors which were not previously incorporated into the setpoint determination. Inclusion of these errors requires that the setpoint be moved further away from the Technical Specifications limit of 120%.

Safety Evaluation

T' hange is determined to be safe and not an unreviewed safety question. The affects of the ne setpoint have already been enveloped by MP1 accident analysis since it is based on the T nical Specifications limit. The change introduces no new equipment failure modes and will not affect any accident scenarios.

Setpoint Change Number	Title
M1-95-031	Containment Spray Interlock Setpoint Change

Description of Change

This modification changes the calibration tolerances on PS-1501-83A,B,C, and D. Currently the As-Found and As-Left tolerances are + 0.4 psig. These will be changed to an As-Found value of + 0.16 psig and As-Left value of + 0.10 psig.

Reason for Change

These switches sense the relative pressure inside the drywell. If pressure increases above the setpoint, the contacts internal to the pressure switches close, causing associated ECCS relays associated with the containment cooling interlock to energize. When drywell pressure drops below the pressure switch reset, the interlock will discontinue containment spray cooling.

Safety Evaluation

This change is determined to be safe and not an unreviewed safety question. The affects of the new calibration tolerances have already been enveloped by the MP1 accident analysis since it is based on the Technical Specifications limits. The change introduces no new equipment failure modes and will not affect any accident scenarios.

Setpoint Change Number	Title
M1-95-033	Drywell High Pressure Reactor Protection System (RPS) Switch Setpoint Change

Description of Change

This modification will change the setpoint on PIS-1621A,B,C,D to 1.8 psig (50.0 in. wg.).

Reason for Change

These switches sense the relative pressure inside the drywell. A higher than normal pressure inside the drywell is an indication of a break in the reactor coolant boundary and resultant reduction in coolant inventory. If pressure increases above the setpoint, these switches provide a scram signal to the reactor protection system.

Safety Evaluation

It has been determined that this change is safe and not an unreviewed safety question. It will not impact plant safety and will not adversely affect the performance of plant equipment.

Setpoint Change Number	Title
M1-95-034	Scram Discharge Volume Reactor Protection System (RPS) Level Switch Setpoint

Description of Change

This modification will change setpoint upper limit on LS-302N-82A,B,C,D and LS-302S-A,B,C,D to 25 7/8 inches of water from 26 inches of water.

Reason for Change

These switches sense the two (North & South) scram discharge volumes (SDV). When the SDV water level increases above the setpoint the contacts internal to the level switches open, causing associated RPS sensor relays to de-energize. De-energization of these relays causes a reactor scram.

Safety Evaluation

This change is determined to be safe and not an unreviewed safety question. The affects of the new setpoint have already been enveloped by MP1 accident analysis since it is based on the Technical Specifications limit. The change introduces no new equipment failure modes and will not affect any accident scenarios.

Setpoint Change Number	Title
M1-95-036	Low Pressure Coolant Injection (LPCI)/Core Spray 300-350 psi Interlock (PS-263-52A,B)

Description of Change

This modification will change the setpoint on PS-263-52A and B from 338 psig to 328 psig. These switches sense pressure inside the reactor.

Reason for Change

These switches ensure that the Emergency Core Cooling System (ECCS) initiates early into an accident by being required to trip at 300 psig or above. In addition, these switches also ensure that the reactor pressure has decreased to below 350 psig prior to allowing ECCS initiation to protect the low pressure piping.

Safety Evaluation

This change is determined to be safe and not an unreviewed safety question. The affects of the new setpoint have already been enveloped by MP1 accident analysis since it is based on the Technical Specifications limit. The change introduces no new equipment failure modes and will not affect any accident scenarios.

Setpoint Change Number	Title
M1-96-003	Average Power Range Monitor (APRM) Hi Flux Clamp

Description of Change

This modification will change the APRM High Flux clamp setpoint from 117.5% to 117.0% reactor power. This existing AS-Found acceptance criteria will be decreased from + 2.5% to +2.0%. In addition, the AS-Left acceptance criteria will be given as +1.5%.

Reason for Change

APRMs 1,2,3,4,5,6 sense reactor power by averaging the signals provided by Local Power Range Monitors (LPRMs). As reactor power increases, the APRM scram setpoint increases as flow increases. Under high flow conditions, the high flux clamp circuit places an upper limit on the scram setpoint to ensure the Technical Specifications limit of 120% is not exceeded.

Safety Evaluation

This change is determined to be safe and not an unreviewed safety question. The affects of the new setpoint have already been enveloped by MP1 accident analysis since it is based on the Technical Specifications limit. The change introduces no new equipment failure modes and will not affect any accident scenarios.

Setpoint Change Number	Title
M1-96-006	Average Power Range Monitor (APRM) Load Reject Setdown Setpoint

Description of Change

This modification will change the APRM High Flux clamp setpoint from 87.5% to 87.0% reactor power. The existing As-Found acceptance criteria will be decreased from +2.5% to +2.0%. In addition, the As-Left acceptance criteria will be given as +1.5%.

Reason for Change

APRMs 1,2,3,4,5,6 sense reactor power by averaging the signals provided by Local Power Range Monitors (LPRMs). As reactor power increases, the APRM scram setpoint increases as flow increases. Under high flow conditions, the high flux clamp circuit places an upper limit on the scram setpoint to ensure the Technical Specifications limit of 120% is not exceeded. Following a generator load reject, a select rod insert occurs and reactor power drops to approximately 75% reactor power. The APRM clamp setpoint will drop to a Technical Specifications limit of 90% to ensure that the load reject was successful.

Safety Evaluation

This change is determined to be safe and not an unreviewed safety question. The affects of the new setpoint have already been enveloped by MP1 accident analysis since it is based on the Technical Specifications limit. The change introduces no new equipment failure modes and will not affect any accident scenarios.

Setpoint Change Number	Title
M1-96-037	Low Pressure Coolant Injection (LPCI)/Core Spray 300-350 psi Interlock (PS-263-52C,D)

Description of Change

This modification will change the setpoint PS-263-52C and D from 351 psig to 342 psig. These switches sense pressure inside the reactor.

Reason for Change

These switches ensure that the ECCS initiates early into an accident by being required to trip at 300 psig or above. In addition, the switches also ensure that reactor pressure has decreased below 350 psig prior to allowing ECCS initiation to protect the low pressure piping (it should be noted that the old setpoint is not a violation of Technical Specifications due to the head correction).

Safety Evaluation

This change is determined to be safe and not an unreviewed safety question. The affects of the new setpoint have already been enveloped by MP1 accident analysis since it is based on the Technical Specifications limit. The change introduces no new equipment failure modes and will not affect any accident scenarios.

BYPASS / JUMPER CHANGES

Bypass / Jumper Number	Title
1-95-145	Desludging of the Filter Sludge and Cleanup Filter Sludge Tanks
1-96-001	Bag Filter Substitution for Waste Collector Filter
1-96-005	Install a freeze seal on 3/4" -PRL-21 to provide isolation for 1-RR-35
1-96-13	Temporary heat to the "B" Battery Room
1-96-15 & 1-96-36	Alternate Ventilation System Control for Reactor Building
1-96-25	Capping 3"-RWR-19 from Floor Drain Filter to Spent Resin Tank
1-96-026	Fire Pumphouse Halon System Heat Detector Installation Jumper Bypass
1-96-034	Bypassing of Refuel Bridge Over Core Limit Switches
1-96-45	14A/B/C/D UV Scram Power
1-96-50	Gas Turbine Generator Trip Bypasses for A-46 and OD for ACR 9072
1-96-53	Gas Turbine
1-96-65	Blocking Off "A" Concentrator Exhaust Register in Radwaste Building
1-96-75	Reduction of Rx Build Supply Air Design Limit

BYPASS / JUMPER CHANGES(Continued)

Bypass / Jumper Number	Title
1-96-95 & 1-96-96 & 1-96-97	'A' & 'B' & 'C' TBCCW Heat Exchanger Inlet Strainer
1-96-98 & 1-96-99	'A' & 'B' TBSCCW Heat Exchanger Inlet Strainer
1-96-100 & 1-96-101 & 1-96-102	'A' & 'B' & 'C' RBCCW Heat Exchanger Inlet Strainer
1-96-108	Installation of Uninteruptable Power Supply for Kaman radiation monitor (RIT-1705-79) in the Stack Gas Building

Bypass / Jumper Number	Title
1-95-145	Desludging of the Filter Sludge and Cleanup Filter Sludge Tanks

Description of Change

A vendor supplied air operated diaphragm pump and flexible hoses will be used to transfer sludge from the filter sludge tank and cleanup filter sludge tank in lieu of the installed plant equipment.

Reason for Change

The augmented quality of the plant equipment (i.e.. pump and hoses) has not been maintained as radwaste quality assurance (RWQA) classified material. Specifically, periodic hydrostatic testing of hoses and pressure boundary material certifications have not been maintained.

Safety Evaluation

The modifications will not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are safe and do not raise an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-001	Bag Filter Substitution for Waste Collector Filter

Description of Change

A vendor supplied filtering skid (bag filter system) will be installed to provide filtering water processed through the equipment drains portion of the liquid radwaste system. The filtering skid will be tied into existing plant equipment via flanged connections and will bypass the Waste Collector Filter.

Reason for Change

The vendor supplied Bag Filtering System will be used to bypass the Waste Collection Filter

Safety Evaluation

The modifications will not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are safe and do not raise an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-005	Install a freeze seal on 3/4" -PRL-21 to provide isolation for 1-RR-35

Description of Change

The work involves replacing the gasket between the body and bonnet inside the valve (1-RR-35) during the refueling outage with the fuel out of the reactor vessel. The valve pressure boundaries will be open to atmosphere, therefore a freeze seal will be applied to isolate the valve from the reactor vessel.

Reason for Change

The work entails performing maintenance on Reactor Recirculation System valve 1-RR-35, Reactor Isolation Valve to the Post Accident Sampling system located in the Drywell.

Safety Evaluation

A detailed safety evaluation is not required as the evolution is deemed safe and not an unreviewed safety question. This maintenance has minimal potential for draining the vessel with redundant makeup capability and potential leakage anticipated being no higher than the estimated 6-8 gpm leak rate. Therefore the appropriate controls are in effect to prevent flooding in the drywell. The fuel will not be uncovered due to the slow potential leakage of 6-8 gpm and also because the fuel is not in the reactor, but in the fuel pool. Performance of the maintenance procedure is considered safe and not an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-13	Temporary heat to the "B" Battery Room

Description of Change

This change installs a temporary electric heater in the area outside of the "B" battery room door. The heater is to be installed in the vicinity of bus 14B/Cubicle 14B.

Reason for Change

This heater in conjunction with a temporary fan installed bypass jumper 1-96-2 will provide additional heat required to maintain the "B" battery room temperature above the 65°F as required by USFSAR Section 8.3.2.1.1.1 to ensure that electrolyte temperature remains above 60°F

Safety Evaluation

It is concluded that this jumper device is safe. It will not affect any previously evaluated accident, or its consequences, nor will it contribute to any new accidents. The margin of safety is not affected. The review concludes that the temporary jumper installation does not constitute an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-15 & 1-96-36	Alternate Ventilation System Control for Reactor Building

Description of Change

This jumper bypass disconnects the Reactor Building ventilation dampers from the operator and secures the damper in an open position, as normal operational control is disabled for associated modification implementation. Normal operating system protection functions are addressed by manual compensatory actions. This includes monitoring supply air plenum temperat re and building negative pressure.

Reason for Change

This jumper device is necessary to provide the Reactor Building with ventilation with the dampers disabled. This condition is necessary for the implementation of plant modifications PDCR # 1-85-95 and PDCR # 1-100-95.

Safety Evaluation

The modifications will not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are safe and do not raise an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-25	Capping 3"-RWR-19 from Floor Drain Filter to Spent Resin Tank

Description of Change

This modification involves cutting line 3"-RWR-19 in two places, removing the leaking portion of the line between the cut locations and installing pipe caps on both ends of the line remaining after the removal of the leaking section.

Reason for Change

Line 3"-RWR-19 has a leak in its bimetalic weld.

Safety Evaluation

The modifications will not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are safe and do not raise an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-026	Fire Pumphouse Halon System Heat Detector Installation Jumper Bypass

Description of Change

The change associated with this Jumper Bypass consist of the removal of an existing smoke detector (one of two), and replacing it with a heat detector. The new heat detector is rated at 225°F, with a rate compensating feature.

Reason for Change

To preclude Halon from prematurely discharging on smoke only - the system will now require both smoke and heat. The use of a heat detector is preferred because the possibility exist for the Diesel Fire Pump to exhaust minor amounts of smoke into the room exposing both existing smoke detectors. The heat detector is not sensitive to smoke, therefore ensuring that the Halon system is not subject to trigger on a single change in environment (not just smoke, not just heat).

Safety Evaluation

A review of the change associated with this bypass jumper has concluded that Halon system response may be slightly slower than if two smoke detectors remained in service. However, the use of a heat detector is appropriate for this application to preclude inadvertent operation, while still meeting NFPA and manufacturer's recommendations. A review of this change has determined:

1) This change does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the safety analysis.

2) The change does not create the possibility for an accident or malfunction of a different type than any evaluated previously in a safety analysis.

3) This change does not reduce the margin of safety, which forms the basis for MP-1's Technical Specifications (or Technical Requirements for Fire Protection).

Bypass / Jumper Number	Title
1-96-034	Bypassing of Refuel Bridge Over Core Limit Switches

Description of Change

This jumper will allow the removal of the #1 and #2 limit switch arms used to detect refuel bridge position with respect to the reactor core. The limit switches provide input to refueling interlock logic to restrict movement of the refuel bridge during refueling. The refueling interlocks are designed to prevent inadvertent criticality. This jumper is being installed with maximum negative reactivity in the core since the core will be offloaded with all control rods fully inserted and electrically disarmed. With control rods fully inserted and electrically disarmed, the refueling interlocks are not required to be operable per Technical Specification 3.10.A, Refueling Interlocks.

Reason for Change

This bypass jumper allows moving the refuel bridge over the core by removing refueling interlock limit switch arms. The refueling interlocks are provided to assure criticality does not occur inadvertently when there is a potential for altering the core. Technical Specification 3.10.A states that refueling interlocks are not required to be operable when all control rods are verified to be fully inserted and electrically disarmed. While this jumper is installed, no core alterations will be in progress and the core will be offloaded with control rods fully inserted and electrically disarmed.

Safety Evaluation

The modifications will not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are safe and do not raise an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-45	14A/B/C/D UV Scram Power

Description of Change

This jumper device requires the reactor to be in cold shutdown or refuel condition and the Reactor Protection System (RPS) to be inoperable and jumpered out in accordance with I&C procedure SP408U or IC408V. The leads to the 14A/B/C/D UV scram contacts are lifted to remove power from the improperly separated logic circuits. The jumpers maintaining the manual scram contactors energized are tagged to prevent removal since lifting the leads interrupts the manual scram circuits.

Reason for Change

Jumper Device 1-96-46, "14A/B/C/D UV Scram Power" eliminates the possible interaction between improperly separated circuits and ensures AC RPS power faults are limited to one supply. Inadequate separation was reported in ACR 11574 and LER 96-025-00.

Safety Evaluation

The modifications will not increase the probability of an accident or malfunction, affect accident mitigation or increase the consequences of an accident. The changes are safe and do not raise an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-50	Gas Turbine Generator Trip Bypasses for A-46 and OD for ACR 9072

Description of Change

These jumper devices are installed in the Gas Turbine Generator and Turbine Buildings. They are being installed for a limited time to support Blackness Testing in the Spent Fuel Pool.

Reason for Change

Unresolved Safety Issue A-46 studies lead to the conclusion that some electrical relays would perform badly in a seismic event. The purpose of these jumper devices is to ensure that seismic forces/motions do not spurious trip or lockout the Gas Turbine Generator or the Electrical Distribution System.

Safety Evaluation

A fuel handling accident is not more probable to occur because of the jumpers. Nor is the probability of occurrence of a previously evaluated malfunction of equipment important to safety increased. Also, the consequences of the previously evaluated fuel handling accident are not increased by the installation of these jumpers. The consequences of a previously evaluated malfunction of equipment important to safety are not changed. There is no possibility of an accident of a different type than previously evaluated. The jumpers do not affect the margin of safety with respect to the Secondary Containment. Therefore, it is concluded that the jumpers are safe and not an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-53	Gas Turbine

Description of Change

These jumper devices are installed in the Gas Turbine Generator (GTG) and Turbine Buildings. They are being installed for a limited time to support Blackness Testing in the Spent Fuel Pool.

Reason for Change

These jumper devices are being used to make the GTG and Electrical Distribution system operable for Blackness Testing.

Safety Evaluation

A fuel handling accident is not more probable to occur because of the jumpers. Nor is the probability of occurrence of a previously evaluated malfunction of equipment important to safety increased. Also, the consequences of the previously evaluated fuel handling accident are not increased by the installation of these jumpers. The consequences of a previously evaluated malfunction of equipment important to safety are not changed. There is no possibility of an accident of a different type than previously evaluated. The jumpers do not affect the margin of safety with respect to the Secondary Containment. Therefore, it is concluded that the jumpers are safe and not an unreviewed safety question.

Bypass / Jumper Number	Title
1-96-65	Blocking Off "A" Concentrator Exhaust Register in Radwaste Building

Description of Change

This Jumper Bypass 1-96-65 will install a taped closure plate to block off an 8"x12" exhaust register (600 cfm) located in the Waste Concentrator "A" Room in the Radwaste Building.

Reason for Change

Blocking off the exhaust register will be required during construction work related to asbestos abatement to prevent spreading of asbestos through the building ventilation system.

Safety Evaluation

The temporary changes do not effect any previously analyzed accidents or associated consequences, nor will they contribute to any new accidents. These modifications will not cause any increase in risk to public health and safety. The proposed temporary modifications are therefore safe and do not present an Unreviewed Safety Question.

Bypass / Jumper Number	Title
1-96-75	Reduction of Rx Build Supply Air Design Limit

Description of Change

This jumper bypass provides for temporarily reducing Reactor Building supply flow below design basis limits, to reestablish Reactor Building negative pressure.

Reason for Change

Due to a degraded exhaust flow condition a temporary reduction in Reactor Building Ventilation System (RBVS) supply flow is required to reestablish a negative Reactor Building pressure.

Safety Evaluation

Based on this evaluation that reducing RBVS supply air to reestablish Reactor Building differential pressure will not affect the probability or consequences of an accident, the probability of a new accident or equipment malfunction, Design Basis Analysis assumptions, and that the margin of safety is not decreased, it is concluded that this jumper bypass is safe during COLD SHUTDOWN and not an Unreviewed Safety Question.

Bypass / Jumper Number	Title
1-96-95 & 1-96-96 & 1-96-97	'A' & 'B' & 'C' TBCCW Heat Exchanger Inlet Strainer
1-96-98 & 1-96-99	'A' & 'B' TBSCCW Heat Exchanger Inlet Strainer
1-96-100 & 1-96-101 & 1-96-102	'A' & 'B' & 'C' RBCCW Heat Exchanger Inlet Strainer

Description of Change

This evaluation documents the temporary installation of full flow strainers on the Service Water inlet tube sheet of the Turbine Building Closed Cooling Water (TBCCW), Turbine Building Secondary Closed Cooling Water TBSCCW, and Reactor Building Closed Cooling Water RBCCW heat exchangers. ARCOR epoxy chips have been found in the TBSCCW and the TBCCW heat exchangers and is believed to be the cause of tube leaks. The installation of the full flow strainers will be in the cold shutdown mode only.

Reason for Change

The installation of these strainers will prevent heat exchanger tube damage from ARCOR chips. Installation of the strainers will improve conditions until a permanent repair can be implemented on the Service Water supply header.

Safety Evaluation

The installation of full flow Service Water strainers will not affect the probability or consequences of an accident, the probability of a new accident or equipment malfunction, and the margin of safety is not decreased. It is therefore concluded that the installation of the temporary strainers is safe and not an Unreviewed Safety Question.

Bypass / Jumper Number	Title
1-96-108	Installation of Uninteruptable Power Supply for Kaman radiation monitor (RIT-1705-79) in the Stack Gas Building

Description of Change

This jumper device installs an external uninteruptable power supply (UPS) to provide filtered, reliable power to the High Range Stack Noble Gas Radiation Monitor microcomputer.

Reason for Change

Jumper Device 1-96-108, "Installation of Uninteruptable Power Supply for Kaman radiation monitor (RIT-1705-79) in the Stack Gas Building" improves the reliability of the High Range Stack Noble Gas Radiation Monitor by providing reliable power to the monitor during momentary power interruptions. The High Range Stack Noble Gas Radiation Monitor is powered by lighting panel LP-4A which receives power from motor control center CD-6 via DP-13. System lockups and signal spiking have resulted from the transfer of the ABT feeding CD-6.

Safety Evaluation

Jumper Device 1-96-108, "Installation of Uninteruptable Power Supply for Kaman radiation monitor (RIT-1705-79) in the Stack Gas Building", has been reviewed and determined to be safe and does not involve an Unreviewed Safety Question.

MISCELLANEOUS CHANGES

Miscellaneous Number	Title
96-MP1-008	FSAR changes to Gas Turbine Generator to support Blackness Testing
96-MP1-009	FSAR changes to Area Radiation Monitors to support Blackness Testing
96-MP1-010	FSAR changes to Standby Gas Treatment System to support Blackness Testing
ISE/MP1-92-038	MP1 Reactor Wide-Range Level Instrumentation Reference Leg Fill Modification
OD MP1-208-96	Fuel Assemblies not Fully Seated in Spent Fuel Pool
Operability Determination	Operability Determination For Boraflex Deterioration At Millstone Unit 1
Software Program	DBVT Database System

Miscellaneous Number	Title
96-MP1-008	FSAR changes to Gas Turbine Generator to support Blackness Testing

Description of Change

The changes to the UFSAR will correct the setpoints for the (1) Gas Turbine (GT) low lube oil pressure trip, (2) Electric generator bearing header low lube oil trip, and (3) the GT generator lube oil air cooler louver control temperature controller (TIC-25-88FC-2). In addition there will also be changes made to the system operating description of the Gas Turbine Fuel Unloading system.

Reason for Change

These changes correct the UFSAR description to be consistent with the plant procedures and the design.

Safety Evaluation

The changes are neither an Unreviewed Safety Question nor a change which reduces the margin of safety.

Miscellaneous Number | Title

96-MP1-009

FSAR changes to Area Radiation Monitors to support Blackness Testing

Description of Change

Section 11.5.2.1 of the UFSAR describes the design of the Reactor Building Ventilation Exhaust, Main Steam Tunnel Ventilation Exhaust, and Refueling Floor Area Radiation Monitors. This section states that all three functions include continuous recording in the main control room. The design at Millstone Point Unit 1 does not include continuous recording of the Main Steam Tunnel Ventilation Exhaust or Refueling Floor Area Radiation Monitor readings.

Reason for Change

This change corrects the UFSAR description to match the design.

Safety Evaluation

The changes are neither an Unreviewed Safety Question nor a change which reduces the margin of safety.

Miscellaneous Number	Title
96-MP1-010	FSAR changes to Standby Gas Treatment System to support Blackness Testing

Description of Change

FSAR Section 6.5.1.1.2 (pg. 6.5-1) states "The exhaust duct radiation monitor provides a continuous indication of radioactivity entering the system and the stack monitor samples the effluent". This statement is incorrect, and will be deleted and replaced with, "Refer to Section 11.5 for information regarding the radiation monitors used in initiating the SBGT system and regarding the stack gas radiation monitors.

In FSAR Section 6.2.3.4 "Inspection and Testing", the last four paragraphs are deleted and in its place a referral to FSAR Section 6.5.1.1.4 for "SBGT Testing and Inspection" is added.

Reason for Change

These changes correct the UFSAR description to match the design.

Safety Evaluation

The changes will not affect any previously evaluated accident, or its consequences, nor will they contribute to any new accidents. The margin of safety is not affected. It is concluded that the changes are safe and does not constitute an Unreviewed Safety Question.

Miscellaneous Number	Title
ISE/MP1-92-038	MP1 Reactor Wide-Range Level Instrumentation Reference Leg Fill Modification

Description of Change

This modification will provide flow from the Control Rod Drive (CRD) system to the cold reference legs associated with the Fuel Zone Level Instrumentation. The CRD flow will be taken from the charging header. The tie-in point for the CRD flow will be in the instrument rack at 14'6" elevation. Flow would be continually provided at a rate of 4 lbm/hr through a needle valve, which will reduce the pressure from 1500 psig to about 1050 psig. Flow instrumentation would also be provided for local indication

Reason for Change

This change will restore the accuracy of level indicators implied in the accident analysis and in the assumed operator actions.

Safety Evaluation

It is concluded that this design change is safe and does not constitute an Unreviewed Safety Question.

Miscellaneous Number	Title
OD MP1-208-96	Fuel Assemblies not Fully Seated in Spent Fuel Pool

Description of Change

Subsequent to an inspection which identified a fuel assembly in the Spent Fuel Pool that was not fully seated in its storage rack, an inspection was performed on October 10, 1996 to identify any similar conditions. During the inspection 56 fuel assemblies were identified as not fully seated. A Safety Evaluation was performed for the non-conforming fuel assemblies.

Reason for Change

An inspection of the Millstone 1 Spent Fuel Pool identified 56 spent fuel assemblies that are not fully seated in their storage racks. This condition is non-conforming in that fuel is intended to be fully seated in the racks.

Safety Evaluation

The condition resulting from fuel assemblies being elevated from their fully seated storage locations in the current configuration of the Spent Fuel Pool is safe, and does not constitute an Unreviewed Safety Question.

Miscellaneous Number	Title
Operability Determination	Operability Determination For Boraflex Deterioration At Millstone Unit 1

Description of Change

Considered in this evaluation is the mechanical/material deterioration of the Boraflex in the Spent Fuel Pool Racks. The degradation of the neutron-absorbing material has an impact on the performance qualification of the Boraflex as a reactivity holddown device in the Spent Fuel Pool Rack design. Boraflex material is known to experience deleterious changes to the mechanical properties when subjected to gamma irradiation. The damage results in the formation of gaps and or erosion of the material.

Reason for Change

Gaps in the Boraflex material have been identified under inspection and examination of the Spent Fuel Pool Racks.

Safety Evaluation

This change does involve an Unreviewed Safety Question based on the determination that a new malfunction has been identified that was not addressed in the design basis of the Boraflex Spent Fuel Pool Racks.

Miscellaneous Number	Title
Software Program	DBVT Database System

Description of Change

The original data base for verifying the operating characteristics of MP-1 used Filemaker Pro software. As the verification effort grew, the data base needs increased. The Microsoft Access software provides the Graphic User Interface (GUI) to the higher performing Sybase data storage and retrieval system.

Reason for Change

The Access application provides security for the record number (also referred to as the log number), the description of the requirement, the initiator of the description, and the discussion items used to evaluate the discrepancy. Specifically, once the description, the log number and the discussion are entered into the data base, they cannot be removed or edited by the user community.

Safety Evaluation

Use cf this data base using the Access / Sybase platforms is safe and does not constitute an Unreviewed Safety Question.

Docket No. 50-245
B16517

Attachment 2

Millstone Nuclear Power Station, Unit No. 1

NRC Commitment/Changes

June 1997

TABLE 1
Millstone Unit No. 1
Changes in Schedule

No.	Description	Scheduled Completion Date	Remarks	Revised Completion Date
B15606-1[1]	Modify Technical Specification 3.6.H and Technical Specification Surveillance 4.6.H, in order to allow for surveillance of the recirculation pump flow mismatch by surveiling the recirculation pump M/G set speed.	Prior to startup for operating Cycle 16	PDCR 1-87-95 will provide indication of recirculation pump speed in the Control Room	Canceled
B15757-2[2]	Complete a review of the seismic design adequacy of safety-related plant piping. This review will cover both small bore and large bore piping, and will verify that all safety-related large bore lines have been addressed under the IEB 79-14 project.	11/1/96	Additional time required to finalize documentation. The date change does not change the current requirement that the activities be completed prior to declaring the individual systems operable.	Prior to Startup for Operating Cycle 16
B15757-3[2]	Supplement this LER [LER 96-040-00] with the results of the review of small and large bore piping.	11/1/96	See B15757-2	See B15757-2
B15917-2[2]	Supplement this LER [LER 96-051-01] with the results of this ongoing NUSOER 92-02 Program review.	1/30/97	See B15757-2	See B15757-2
B15784-3[3]	Resolve whether the PIV designation of valves 1-SD-1, SD-2A, 1-SD-2B, and 1-SD-5 is appropriate. The resolution, and any additional corrective actions related to it, will be	1/10/97	Additional time required to complete validation of all statements.	Prior to startup for Operating Cycle 16

TABLE 1
Millstone Unit No. 1
Changes in Schedule

No.	Description	Scheduled Completion Date	Remarks	Revised Completion Date
B15485-2[4]	Complete ventilation system design review	6/30/96	There is a duplication of commitments. B15485-2 is similar to B16035-1, which is scheduled for completion during Refueling Outage 15 as part of the 50.54(f) effort.	Canceled
B16211-1[5]	Complete (Prior to performing the next ILRT), the program administrative control document, "Containment Leak Rate Testing Program Administrative," which will identify the specific roles and responsibility of implementing and maintaining the program. This document will also list the primary containment isolation barriers with associated basis for each penetration.	5/31/97	Resources reallocated due to change in restart date	Prior to the startup for Operating Cycle 16
B16211-2[5]	Determine the Appendix J Program staffing requirements and obtain program management approval by May 31, 1997.	5/31/97	Resources reallocated due to change in restart date.	Prior to the startup for Operating Cycle 16
B16211-3[5]	Develop an Appendix J action plan which will identify the importance of conservative decisions making regarding implementation of the Appendix J requirements from all perspectives including design, operation, and engineering. NNECO will incorporate this plan into an administrative procedure, "Containment Leak Rate Testing	5/31/97	Resources reallocated due to change in restart date.	Prior to the startup for Operating Cycle 16

TABLE 1
Millstone Unit No. 1
Changes in Schedule

No.	Description	Scheduled Completion Date	Remarks	Revised Completion Date
	Program Administration."			
B16211-4[5]	Determine Appendix J programmatic enhancement training requirements and include in administrative procedure,* "Containment Leak Rate Testing Program Administration" and will implement this training.**	5/31/97* 9/30/97**	Resources reallocated due to change in restart date.	Prior to the startup for Operating Cycle 16
B16279-1[6]	Review the records for the past two cycles to verify that other Appendix J surveillances have not exceeded their interval.	7/15/97	Resources reallocated due to change in restart date.	Prior to the startup for Operating Cycle 16

TABLE 1
Millstone Unit No. 1
References

(1) W. J. Riffer letter to the U.S. Nuclear Regulatory Commission, "LER 96-015-00," dated March 22, 1996

(2) P. D. Hinnenkamp letter to the U.S. Nuclear Regulatory Commission, "Millstone Unit No. 1, Commitment Changes," dated January 30, 1997

(3) W. J. Riffer letter to the U.S. Nuclear Regulatory Commission, "LER 96-039-1," dated October 1, 1996.

(4) D. B. Miller, Jr. letter to the U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 1, Reply to Notice of Violation, Inspection Nos. 59-245/95-31, 50-336/95-31, and 50-423/95-31," dated Jan[illegible], 1996.

(5) P. D. Hinnenkamp letter to the U.S. Nuclear Regulatory Commission, "LER 96-046-03," dated March 3, 1997.

(6) P. D. Hinnenkamp letter to the U.S. Nuclear Regulatory Commission, "LER 97-011-00," dated March 17, 1997.

TABLE 2
Millstone Unit No. 1
Revised NRC Commitments

No.	Original Commitment	Revised Commitment
B15961-3[1]	Signs have been placed on Security Department file cabinets indicating the requirement to provide a "Safeguards Information Coversheet" on SGI that is transmitted out of the Department.	Signs have been placed on Security Department file cabinet, containing SGI, indicating the requirement to provide a "Safeguards Information Coversheet" on SGI that is transmitted out of the Department.
B15636-1[2]	Initiate a Technical Specification change, prior to startup for operating Cycle 16, which will make the Technical Specification for reactor anomaly surveillances consistent with the Standard Technical Specifications (NUREG 1433)	Initiate a Technical Specification change which will make the Technical Specifications for reactor anomaly surveillance consistent with the Standard Technical Specifications (NUREG 1433). This will be submitted as part of the Millstone 1 STS conversion project.
B15636-2[2]	Modify the Technical Specification surveillance procedures required for reactor anomaly surveillances, so that they will be consistent with the Technical Specification requirements, as modified by the Technical Specification change identified above, prior to startup for operating Cycle 16.	Prior to operating Cycle 16, modify the Technical Specification surveillance procedures required for the reactor anomaly surveillance so that they will be consistent with the current Technical Specification requirements.
B16175-5[3]	Increase description field size to allow complete documentation of work performed and field conditions which impacted the work by 5/1/97	Increase description field size to allow complete documentation of work to be performed and field conditions which could impact work by 8/1/97

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(1) T. L. Harpster letter to the U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3, Facility Operating License Nos. DPR-21, DPR-65, and NPF-49, Reply to a Notice of Violation, Inspection 50-245/96-06; 50-336/96-06; 50-423/96-06," dated November 8, 1996.

(2) W. J. Riffer letter to the U.S. Nuclear Regulatory Commission, "LER 96-018-00," dated April 4, 1996.

(3) J. P. McElwain letter to the U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 1, Maintenance Improvement Plan Revised Corrective Action Plan," dated January 22, 1997.