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The Northeast Utilities System

MAY 1 1997

Docket No. 50-423
B16360

Re: 10CFR50.59

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

Millstone Nuclear Power Station, Unit No. 3

Corrections to 1993 Annual Report of Changes
Pursuant to 10CFR50.59 and Section 6.9.1.2
of Appendix A to Operating License NPF-49

In our letter dated February 28, 1994, we submitted the 1993 Annual Report for Millstone Nuclear Power Station, Unit No. 3 describing changes made pursuant to 10CFR50.59 and Section 6.9.1.2 of Appendix A to Operating License NPF-49. Subsequent internal verification of the report has identifi

ed typographical errors involving numerical identification of eight plant changes and an error in the description of one of the design changes.

Attachment 1 to this letter provides correction of the subject errors in the original report. These corrections consist of replacement pages having the same page numbers as the original report, with the revised information indicated by change bars in the right-hand margin on each page.

Commitments

Attachment 2 provides the regulatory commitments in this submittal.

IE47 1/1

Should you have any questions regarding this submittal, please contact Mr. James Peschel at (860) 437-5840.

Very truly yours,
NORTHEAST NUCLEAR ENERGY COMPANY

M. H. Brothers
Vice President - Millstone Unit No. 3

cc: H. J. Miller, Region I Administrator
W. D. Travers, Ph.D, Director, Special Projects Office
J. W. Anderson, NRC Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

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

Docket No. 50-423
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Attachment 1

Millstone Nuclear Power Station, Unit No. 3

Corrections to 1993 Annual Report of Changes
Pursuant to 10CFR50.59 and Section 6.9.1.2
of Appendix A to Operating License NPF-49

May 1997

PLANT DESIGN CHANGES

PDCR Number	Title
3-92-119	Removal of Flow Switch Interlock from Auxiliary Building Filter Fan Controls
3-92-120	Vacuum Priming System Upgrade
3-92-124	Amertap System Removal, Condenser Discharge Piping Upgrade, Condensate System Pump Down line Piping Upgrade
3-92-129	Replacement of Reactor Plant Component Cooling Water (RPCCW) Heat Exchanger Cross-Connect Valve
3-93-003	Installation of Transfer Trip for Tripping Main Generator Output Breaker
3-93-006	Revised Rod Control System Compensation Parameters
3-93-008	Modifications to Heat Exchanger and Associated Piping for Charging Pumps Cooling and Safety Injection Pumps Cooling
3-93-009	Service Water System Piping Modifications
3-93-011	Modification to Generator Line Protection Pilot Wire Scheme
3-93-012	Technical Support Center Damper Modifications
3-93-013	Feedwater Flow Control Valves Position Indication
3-93-015	Replacement of Recirculation Spray Pump Suction Valves
3-93-016	Replacement of Snubbers Due to Functional Failures
3-93-023	Heated Junction Thermocouple Probe and Cable Replacement
3-93-024	Core Exit Thermocouple Connector and Cable Replacement

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-93-027	Reactor Coolant Pump (RCP) No. 1 Seal Leakoff High Flow Alarm Setpoint Change and Rescaling of All No.1 Seal High Range Leakoff Flow Transmitters
3-93-034	Permanent Reactor Cavity Seal Installation
3-93-037	Arcor Coating of the Inside Diameter of Service Water System Spools
3-93-044	Make Duct Support in Main Steam Valve Building Removable
3-93-050	Auxiliary Feedwater Pump Design Change
3-93-054	Reactor Flange Shield Installation
3-93-060	Installation of Forced Air Cooling Fans Within the 7300 System
3-93-061	Revising Motor Operated Valve's Motor Protection
3-93-062	Replacement of Turbine Driven Auxiliary Feedwater (TDAFW) Pump Valve Room Air Conditioning System
3-93-064	Rewire Relays in the Engineered Safety Features Actuation System (ESFAS)
3-93-067	Auxiliary Building Charging and Reactor Plant Component Cooling Water (CH/RPCCW) Area Ventilation Heaters
3-92-074	Unit 3 Ecolochem Building Installation
3-93-076	Modification to SIGMA Refueling Machine Control Console
3-93-082	Installation of Rockbestos Cable to Replace Thermo-Lag Fire Barrier
3-93-090	Reload Design for Millstone Unit No. 3 Cycle 5
3-93-092	Lube Oil Reservoir Tank Indication and Alarm
3-93-093	Boron Concentration Measurement System Removal

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-92-097	Addition of Isolation Valves to Containment Hatch for Local Leak Rate Testing (LLRT) Tube Connection
3-93-106	Rerouting of the Circuit Associated with a Reactor Plant Ventilation Temperature Switch to Comply with Separation Requirements
3-92-109	Motor Driven Auxiliary Feedwater Pump Trip Circuit Modification
3-93-116	Abandonment of Pressurizer Liquid Sample Line
3-93-120	Installation of Rigging Support for Auxiliary Feedwater Pump
3-93-121	Addition of Reactor Trip Interlock to Inadequate Core Cooling Monitor Annunciation
3-93-124	Engineered Safety Features (ESF) Building Ventilation Supply Fan Flow Switch Setpoint Revision
3-93-125	Replacement of the "A" Quench Spray Pump Anchor Bolts
3-93-126	Charging and Reactor Coolant System Valve Yoke Bolts Replacement
3-93-128	Modify Supports for "B" Safety Injection Pump Miniflow Isolation Valve to be Removable
3-93-129	Replacement of Service Water Inlet Isolation Valve to Reactor Plant Component Cooling Water Heat Exchanger
3-93-131	Modification to Reactor Head Vent Isolation Valves
3-93-133	Installation of Finer Cartridges for Letdown Reactor Coolant and Seal Water Injection Filters
3-93-135	Modifications to Diesel Jacket Water Expansion Line

PLANT DESIGN CHANGES (CONTINUED)

PDCR Number	Title
3-93-136	Jib Crane in the Reactor Head Storage Area
3-93-137	Canopy Seal Repair of Spare Capped Nozzles K-4 and E-9
3-93-140	Limitorque Actuator Spring Pack Replacement
3-91-146	Deletion of Radioactive Liquid Waste Conductivity Instruments
3-93-147	Substitution of Excess Flow check Valves with Steam Traps
3-93-14[illegible]	Steam Generator Tube U-Bend Stabilizer
3-93-153	Missile Shield Shim Modification for Control Rod Drive Mechanism (CRDM) Shroud Cooler
3-93-159	Repower Ventilation Flow Switches from Uninterruptible Safety-Related Power Sources
3-93-160	"C" Reactor Coolant Pump (RCP) Replacement
3-93-161	Limitorgue Actuator Gearing Changes
3-93-167	Fuel Transfer Tube Closure Bolt Reduction
3-93-174	"B" Reactor Coolant Pump (RCP) Replacement
3-93-177	"B" Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications
3-93-178	"A" Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications
3-93-180	"D" Reactor Coolant Pump (RCP) Charging System and Component Cooling Water System Piping and Support Modifications
3-93-184	Modification of Reactor Coolant Pumps (RCP) Oil Collection System

Plant Design Change Number 3-93-008

This change, entitled "Modifications to Heat Exchanger and Associated Piping for Charging Pumps Cooling (3CCE*E1A,B) and Safety Injection Pumps Cooling (3CCI*EIB)" is complete.

Description of Change

Modifications were made to the geometry of the Service Water piping on the Charging Pump Coolers (3CCE*EIA,B) and the Safety Injection Pump Cooler (3CCI*EIB). Also, the inlet and outlet isolation valves on 3CCI*EIB were replaced and relocated. In addition, several Service Water System pipe supports that were affected by the piping changes were modified.

Reason for Change

The piping modifications provided improved flow characteristics and eliminated rapid changes in direction which had been identified as contributing factors to pipe wall thinning. Unions were added for disassembly points to support heat exchanger inspections. The plug valves on 3CCI*E1B were replaced with full port ball valves to provide improved flow characteristics.

Safety Evaluation

The modifications to the Service Water piping, the isolation valves, and the associated pipe supports are safe. The changes do not adversely affect system performance, and the changes were made in accordance with the original ASME design criteria. The changes had no adverse impacts on the system design, operational, and functional characteristics.

Plant Design Change Number 3-92-074

This change, entitled "Unit 3 Ecolochem Building Installation," is complete.

Description of Change

A permanent pre-engineered metal building was installed adjacent to the Unit No. 3 Turbine Building. Fire protection was provided via a connection to the Unit No. 3 fire main.

Reason for Change

Provide a permanent structure for the vendor's water treatment equipment.

Safety Evaluation

The installation of the permanent building is consistent with the original plant design. No safety-related equipment is contained in the structure.

Plant Design Change Number 3-92-097

This change, entitled "Addition of Isolation Valves to Containment Hatch for Local Leak Rate Testing (LLRT) Tube Connection," is complete.

Description of Change

This change modified the LLRT system configuration of the containment personnel access hatch seals. The change removed a permanent testing system consisting of a pneumatic module, three solenoid valves, a remote control panel and associated tubing. The pneumatic unit was replaced with a portable test configuration. The new configuration consisted of tubing, two new valves and caps.

Reason for Change

The original test configuration was unreliable. The leak test was routinely performed by disconnecting the permanent unit and connecting the portable unit. The connecting and disconnecting of the portable equipment created a potential containment breach. Implementation of this change eliminated the potential containment breach.

Safety Evaluation

The new test system had been demonstrated through prior use to be more reliable and to meet containment integrity requirements.

Plant Design Change Number 3-92-109

This change, entitled "Motor Driven Auxiliary Feedwater Pump Trip Circuit Modification," is complete.

Description of Change

The change removed the low suction pressure trip for both of the Motor Driven Auxiliary Feedwater pumps, provided a low suction pressure trip alarm in the Control Room for each pump, and installed a surge suppressor in the sensing lines for the low suction pressure switches. This made a bypass jumper a permanent installation.

Reason for Change

The pump manufacturer, architect engineer, and Nuclear Steam System supplier recommended that no low suction pressure trip should be installed for the safety-related Auxiliary Feedwater pumps. Spurious trips could make a pump fail at the moment when it is required to feed a steam generator. An alarm replaced the trip function in order to alert the operator. Surge suppressors were installed in order to dampen the pressure spikes that were observed on the switches during previous testing.

Safety Evaluation

This change removed a possible failure mode which could cause an inadvertent pump trip. System design and line-up prevent an actual low suction pressure from occurring. Therefore, there was no need to have this trip function.

Plant Design Change Number 3-91-146

This change, entitled "Deletion of Radioactive Liquid Waste Conductivity Instruments," is complete.

Description of Change

This change removed two conductivity instrument in the Radioactive Liquid Waste System whose function had been previously eliminated from the plant by bypass Jumpers.

Reason for Change

The bypass jumpers were in place to eliminate the re-use of radioactive liquid waste distillate due to potential tritium build-up concerns. The need for the conductivity instruments was eliminated and this change permanently removed the control loops from the plant which allowed the elimination of the bypass jumpers.

Safety Evaluation

The deletion of the automatic control function of these instruments was compensated for by sampling of the distillate prior to discharge.

The system is non-safety-related and is not required for accident mitigation.

PROCEDURE CHANGES

Procedure Number	Title
OP 3301L, Rev. 1	Reactor Coolant Loop Stop Valves
OP 3314A, Rev. 12	Auxiliary Building Heating, and Ventilation Air Conditioning System
OP 3316A, Rev. 7	Main Steam System
OP 333OA, Rev. 7	Reactor Plant Component Cooling Water(RPCCW) System
OPS Form 3612B.4-127, Rev.1, Ch. 2 OPS Form 3612B.4-128, Rev.1, Ch. 2	Containment Local Leak Rate Test Type "C" Penetration 56
SP 3646B.8, Rev. 6	Emergency Generator Fuel Oil Particulate Sample Analysis
IC 3464I08, Rev. 0	Steam Generator Water Level Control Tuning
OPS Form 3273-3/4.3.2.4, Rev. 0	Quadrant Power Tilt Ratio (QPTR) Technical Specification Clarification
OPS Form 3273-3/4.3.1.1.3, Rev. 3	Moderator Temperature Coefficient
OPS Form 3273-3/4.3.1.3.5, Rev. 3	Shutdown Rod Insertion Limit
OPS Form 3273-3/4.3.1.3.6, Rev. 3	Control Rod Insertion Limit
OPS Form 3273-3/4.3.2.1.1, Rev. 2	Axial Flux Difference
OPS Form 3273-3/4.3.2.2.1, Rev. 3	Heat Flux Hot Channel Factor FQ(Z) Four Loop Operation
OPS Form 3273-3/4.3.2.3.1, Rev. 2	Reactor Coolant System Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor Four Loops Operating
OPS Form 3273-3/4.3.1.2.2, Rev. 0	Charging Flow Paths-Operating
OPS Form 3273-3/4.3.1.2.4, Rev. 1	Charging Pumps - Operating
OPS Form 3273-3/4.3.5.2, Rev. 1	Emergency Core Cooling System Subsystems - T_{AVG} Greater Than or Equal to 350°F

Procedure Number	Title
OPS Form 3612B.4-127, Rev. 1, Ch. 2 OPS Form 3612B.4-128, Rev. 1, Ch. 2	Containment Local Leak Rate Test Type "C" Penetration 56

Description of Change

This change provided guidance to supply the Containment Building Fire Protection Water system from the Auxiliary Building Fire Protection Water system while performing the local leak rate test on the containment penetration for the fire protection system.

Reason for Change

The normal fire protection water supply to the Containment Building fire protection water supply to the Containment Building had been isolated for maintenance testing. Bypass jumpers had previously been used to provide the temporary source of fire protection water.

Safety Evaluation

All fire protection water flow and pressure requirements were evaluated and maintained throughout the maintenance testing evolution. Fire suppression to the Containment Building was not impacted by this change.

Procedure Number	Title
SP 3646B.8, Rev. 6	Emergency Generator Fuel Oil Particulate Sample Analysis

Description of Change

This procedure change modified the method for obtaining the monthly fuel oil sample.

Reason for Change

The previous method for obtaining the monthly fuel oil sample required defeating of an electrical interlock by using an electrical jumper device. By electrically cross-connecting the fuel oil follow transfer pumps while collecting the sample, the need to install an electrical jumper was eliminated.

Safety Evaluation

This change does not significantly increase the probability of a failure of the backup fuel oil transfer pumps. The system was designed to provide the required sample.

Procedure Number	Title
OPS Form 3273-3/4.3.2.4, Rev. 0	Quadrant Power Tilt Ratio (QPTR) Technical Specification Clarification

Description of Change

The revision clarifies three technical specification action statements. These action statements delineate actions required to be taken if QPTR exceeds 1.02.

Reason for Change

The subject action statements are unclear as to whether it is acceptable to increase or decrease power above 50% power when QPTR exceeds 1.02. The revision was written to clarify what power manipulations are allowed and the time frames in which they are allowed.

Safety Evaluation

The revision was a clarification of technical specification action statements; it did not change any operating limits.

TESTS

Test Number	Title
IST 3-92-032	Emergency Generator Load Sequencer (EGLS) Train "A" Power Supply Data Acquisition Procedure
IST 3-92-033	Emergency Generator Load Sequencer (EGLS) Train "B" Power Supply Data Acquisition Procedure
IST 3-93-001	Volume Control Tank (VCT) Temperature Reduction and Reactor Coolant Pump (RCP) No. 1 Seal Leakoff Flow Monitoring
IST 3-93-003	Seal Injection Filter Element Replacement
IST 3-93-004	Upper Plenum Anomaly (UPA) Data Collection
IST 3-93-005	Operation of the "A" Main Feedwater Pump Following Coupling Replacement
IST 3-93-006	Sodium Hypochlorite Jet Pumps Test
IST 3-93-007	Solid State Protection System Slave Relay Inservice Test
IST 3-93-009	Solid State Protection System Slave Relay Inserviced Test
IST 3-93-011	Overlap Test of Steam Generator Hi-Hi Level and Safety Injection Trip of Turbine and Main Feedwater Pump
IST 3-93-012	Charging System Filter Element Replacement
IST 3-91-013	Charging System Dynamic Valve Test
IST 3-93-014	Trip Actuating Device Operational Test of the Main Steam Isolation Actuation Relays
IST 3-93-016	Reactor Coolant Pump (RCP) Temporary Cooling System
IST 3-93-019	Early Boration

Test Number	Title
IST 3-91-013	Charging System Dynamic Valve Test

Description of Test

This test provided a procedure to perform dynamic testing of the motor operated valves associated with the Charging System. Specific motor operated valves were dynamically tested to prove their operability during design basis conditions. The test demonstrated that specific valves opened and closed under dynamic conditions. It also validated assumptions and factors used in motor operated valve thrust calculations.

Reason for Test

Dynamic testing is required for safety-related motor operated valves in accordance with Nuclear Regulatory Commission guidance.

Safety Evaluation

The test was performed on the Charging System when the Technical Specifications did not require the system to be operable. With the reactor head removed and the Reactor Coolant System loop drain valves open, no credible means existed for loop overpressurization.

Docket No. 50-423
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Attachment 2

Millstone Nuclear Power Station, Unit No. 3

NNECO's Commitments Associated With
Corrections to 1993 Annual Report of Changes
Pursuant to 10CFR50.59 and Section 6.9.1.2
of Appendix A to Operating License NPF-49

May 1997

Enclosure
List of Regulatory Commitments

The following table identities those actions committed to by NNECO in this document. Any other actions discussed in the submittal represent intended or planned actions by NNECO. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager - Nuclear Licensing at the Millstone Nuclear Power Station Unit No. 3 of any questions regarding this document or any associated regulatory commitments.

Number	Commitment	Due
	(NONE IDENTIFIED)	