Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

JUL 2 2001

Docket Nos. 50-336
50-423
B18442

RE: 10 CFR 50.59

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

Millstone Nuclear Power Station, Unit Nos. 2 and 3
10 CFR 50.59 and Commitment Change Report for 2000

Pursuant to the provisions of 10 CFR 50.59(d)(2), the reports for Millstone Nuclear Power Station, Unit Nos. 2 and 3, are submitted in Enclosures 1 and 2 for changes made to the plant over the period January 1, 2000, to December 31, 2000. Enclosure 3 of this submittal is the 2000 Commitment Change Report for both Unit Nos. 2 and 3. The annual Commitment Change Report is being submitted consistent with the Millstone Nuclear Power Station's Regulatory Commitment Management Program.

There are no regulatory commitments contained within this letter.

Should you have any questions regarding these reports, please contact Mr. Paul R. Willoughby at (860) 447-1791, extension 3655.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price, Vice President
Nuclear Technical Services - Millstone

cc: See next page

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Enclosures (3)

cc: H. J. Miller, Region I Administrator
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Enclosure 1

Millstone Nuclear Power Station, Unit No. 2

10 CFR 50.59 Report for 2000

Millstone Unit No. 2
Plant Design Changes

Design Change Notice No.	Title	Safety Evaluation No.
DCN DM2-00-0084-00	Setpoint Change for PS-7985A and PS-7985C "A" Emergency Diesel Generator (EDG) AC and DC Starting Air Compressor Pressure Switch	S2-EV-00-0027
DCN DM2-00-0086-00	Setpoint Change for PS-7985B and PS-7985D "B" Emergency Diesel Generator (EDG) AC and DC Starting Air Compressor Pressure Switch	S2-EV-00-0028
DCN DM2-00-0285-00	Replacement of "A" Service Water Inspection Tee Cover/Vent Assembly with a Blind Flange	S2-EV-00-0050
DCN DM2-000-293-98	Auxiliary Steam and Condensate System Modification	S2-EV-00-0003
DCN DM2-00-0303-00	Reload Design for Millstone Unit No. 2 Cycle 14	E2-EV-00-0033
DCN DM2-00-0352-00	Gag Relief Valves 2-RB-308 and 2-RB-309	S2-EV-00-0062
DCN DM2-00-0355-00	Gag Reactor Building Closed Cooling Water (RBCCW) Relief Valves	S2-EV-00-0063
DCN DM2-02-0825-99	Reactor Coolant Pump (RCP) Vapor Stage Leak-off Re-Route to the Containment Sump	S2-EV-99-0135
DCN DM2-04-0346-97	Gag Relief Valve 2-RB-330	S2-EV-00-0057
DCN DM2-05-0586-99	Design Change for Main Generator Temperature Monitoring Recorder TR-4510	S2-EV-00-0052

Design Change Record No.	Title	Safety Evaluation No.
DCR M2-00006	Reload Design for Millstone Unit No. 2 Cycle 14	E2-EV-00-0033
DCR M2-00007	Millstone Unit No. 2 Spent Fuel Pool Cooling Analysis for 2R13 & 2C14	S2-EV-00-0002
DCR M2-97016	Reactor Building Closed Cooling Water (RBCCW) System - Relief Valve Replacement - Phase 2	S2-EV-97-0036

Millstone Unit No. 2
Plant Design Changes

Design Change Record No.	Title	Safety Evaluation No.
DCR M2-97053	Screen Wash Pumps (P-8A, B) Lubrication Water Modification	S2-EV-97-0221
DCR M2-99004	Safety Injection Tank Nitrogen System Modification	S2-EV-99-0033
DCR M2-99050	Safety Parameter Display (SPDS) Upgrade - Hardware	S2-EV-99-0142
DCR M2-99052	Containment Dew Point Temperature Monitoring Instrumentation Removal	S2-EV-99-0137
DCR M2-99059	Addition of Braided Expansion Joints to the Spent Fuel Pool (SFP) Cooling System	S2-EV-00-0024
DCR M2-99060	Hydrogen Bulk Storage Skid Relocation	S2-EV-99-0150
DCR M2-99065	Redesign of Existing Severe Line Outage Detector (SLOD) System and Master Supervisory Panel	S2-EV-99-0149
DCR M2-99066	Removal of Reactor Protection System Calibration and Indication Panel (RPSCIP) Flow Dependent Setpoint Selector Switch (FDSSS)	S2-EV-00-0001
DCR M2-99068	Millstone Unit No. 2 Fire Water Connections Cross-Tie Project	S2-EV-00-0004

FSARCR No.	Title	Safety Evaluation No.
FSARCR 99-MP2-38	Safety Injection Tank Nitrogen System Modification	S2-EV-99-0033
FSARCR 99-MP2-96	Reactor Coolant Pump (RCP) Vapor Stage Leak-off Re-Route to the Containment Sump	S2-EV-99-0135
FSARCR 00-MP2-3	Hydrogen Bulk Storage Skid Relocation	S2-EV-99-0150
FSARCR 00-MP2-4	Redesign of Existing Severe Line Outage Detector (SLOD) System and Master Supervisory Panel	S2-EV-99-0149
FSARCR 00-MP2-6	Millstone Unit No. 2 Spent Fuel Pool Cooling Analysis for 2R13 & 2C14	S2-EV-00-0002
FSARCR 00-MP2-7	Removal of Reactor Protection System Calibration and Indication Panel (RPSCIP) Flow Dependent Setpoint Selector Switch (FDSSS)	S2-EV-00-0001

Millstone Unit No. 2
Plant Design Changes

FSARCR No.	Title	Safety Evaluation No.
FSARCR 00-MP2-19	Addition of Braided Expansion Joints to the Spent Fuel Pool (SFP) Cooling System	S2-EV-00-0024
FSARCR 00-MP2-23	Reload Design for Millstone Unit No. 2 Cycle 14	E2-EV-00-0033
FSARCR 00-MP2-24	Setpoint Change for PS-7985A and PS-7985C "A" Emergency Diesel Generator (EDG) AC and DC Starting Air Compressor Pressure Switch	S2-EV-00-0027
	Setpoint Change for PS-7985B and PS-7985D "B" Emergency Diesel Generator (EDG) AC and DC Starting Air Compressor	S2-EV-00-0028
FSARCR 00-MP2-27	Millstone Unit No. 2 Spent Fuel Pool Cooling Analysis for 2R13 & 2C14	S2-EV-00-0002

Minor Modification No.	Title	Safety Evaluation No.
MMOD M2-99017	Steam Generator Blowdown Flow Instrumentation	S2-EV-99-0052
MMOD M2-99-043	Nitrogen Supply Line Modifications to Quench Tank and Primary Drain Tank (PDT)	S2-EV-99-0130
MMOD M2-99047	Reactor Coolant Pump (RCP) Vapor Stage Leak-off Re-Route to the Containment Sump	S2-EV-99-0135
MMOD M2-99048	Reactor Coolant Pump (RCP) Seal Controlled Bleed-off Relief Valve Replacement	S2-EV-99-0133
MMOD M2-99057	Modification of Main Condenser Steam Dump Valves and Turbine Bypass Valve, (2-MS-206, 207, 208 and 209)	S2-EV-99-0151

Procedure No.	Title	Safety Evaluation No.
OP2335A	Clean Liquid Radwaste System	S2-EV-00-0061
OP2335A	Clean Liquid Radwaste System	S2-EV-00-0066
OP2338A	Solid Radwaste System - Filters	S2-EV-00-0066
OP2338D	Temporary Demineralizer Operation	S2-EV-00-0061
SP2605P.3A	Installation of Freeze Seal to Support Leak Test of 2-SI-460	S2-EV-00-0007

Millstone Unit No. 2
Plant Design Changes

Procedure No.	Title	Safety Evaluation No.
SPROC ENG99-2-11	Withdrawal/Insertion of Boraflex Poison Box	S2-EV-00-0073

Technical Evaluation No.	Title	Safety Evaluation No.
M2-EV-00-0001	Auxiliary, Turbine & Enclosure Buildings Auxiliary Steam Heater Replacements	S2-EV-00-0006

Temporary Modification No.	Title	Safety Evaluation No.
Temp. Mod. 2-00-002	Installation of Freeze Seal to Support Leak Test of 2-SI-460	S2-EV-00-0007
Temp. Mod. 2-00-003	Installation of Freeze Seal to Relief Valve Setpoint Testing of 2-SI-469	S2-EV-00-0016
Temp. Mod. 2-00-006	Containment (CTMT) Closure During R13 for Steam Generator (SG) Activities	S2-EV-00-0017
Temp. Mod. 2-00-013 Revision 0	Clean Liquid Radwaste Monitor Tank Recirculation	S2-EV-00-0065
Temp. Mod. 2-00-013 Revision 1	Clean Radwaste Sea Water Removal	S2-EV-00-0061
Temp. Mod. 2-00-014	Temporary Filtration System for Clean Waste Monitor Tank Discharge	S2-EV-00-0066
Temp. Mod. 2-00-015	Monitor "A" Motor Generator Output Voltage, Exciter Field Current, and Control Relay "5K" Operation	S2-EV-00-0069
Temp. Mod. 2-00-016	Freeze Seal 4"-JBD-32 for M2-99068 Installation	S2-EV-00-0083
Temp. Mod. 2-99-007	Containment Camera Installation	S2-EV-99-0084
Temp. Mod. 2-99-014	Temporary Modification to Redirect Reactor Coolant Pump (RCP) Vapor Stage Leak-off from the Primary Drain Tank (PDT) to the Containment Sump	S2-EV-99-0090
Temp. Mod. 2-99-030	Temporary Contractor Water Treatment Facility	S2-EV-99-0144

Temporary Modification No.	Title	Safety Evaluation No.
Temp. Mod. 2-99-031	Removal of Loop 1 Hot Leg Resistance Temperature Detector (RTD), TE-112HD, from Reactor Protection System (RPS) Channel D Input - Substitution with Loop 2 Hot Leg RTD, TE-122HD	S2-EV-99-0145

Calculation No.	Title	Safety Evaluation No.
Calculation 00-ENG-02975-C2	Containment Liner Minimum Wall Thickness Evaluation	S2-EV-00-0034

S2-EV-97-0036
DCR M2-97016

Reactor Building Closed Cooling Water (RBCCW) System - Relief Valve Replacement - Phase 2

Description

Twenty-five (25) RBCCW system thermal relief valves, all with hard seats, were replaced with new soft seated relief valves maintained at a set pressure of 150 psig. The soft seated relief valves provide a more reliable and tighter shutoff, thereby minimizing seat leakage. These relief valves, along with three (3) valves previously replaced per a prior closed out design package (twenty-eight [28] total), were set to a set pressure of 150 psig.

Reason for the Activity

These change outs were a response to RBCCW system leakage increase from approximately 4.5 gallons per day to greater than 100 gallons per day after a RBCCW surge tank level was lowered. Initial response was to require fourteen (14) relief valves in the RBCCW system to be replaced to preclude leakage. Further investigation determined that there were an additional twenty-five (25) valves that needed to be replaced to preclude leakage.

Safety Evaluation Summary

The replacement of the existing relief valves with new relief valves with soft seat material, while maintaining a set pressure at system design pressure, 150 psig, did not increase the probability of occurrence or affect the consequences of an accident or malfunction previously evaluated.

The change in relief valve design to incorporate the soft seat material (Ethylene Propylene DM), while maintaining original valve set pressure along with extensive testing results, is a more reliable valve design/function/operation when compared to the existing relief valves. Additionally, the soft seat material (Ethylene Propylene DM) will not deteriorate, thus a sticking relief valve is no more likely to occur than for the existing relief valves.

It did not create a different type of accident or malfunction. The replacement component is of the same size, installed in the same location, and is manufactured in accordance with the same and/or later standards of the component being replaced. The replacement relief valves were specifically designed to be compatible for installation in the existing piping system and/or equipment. Maintenance of the set pressure at 150 psig is per the ASME Section VIII requirements.

Shop testing and periodic field bench testing of the relief valves demonstrated that the valves provide the proper thermal relief function for the equipment that it protects from over pressure during all operating modes.

S2-EV-97-0221
DCR M2-97053

Screen Wash Pumps (P-8A, B) Lubrication Water Modification

Description

Screen wash pumps P-8A and P-8B supply water for screen washing. The upper bearing/packing of the Screen Wash pumps were lubricated and cooled by domestic water, supplied through the grease fitting on the packing box. The domestic water discharged through the bearing into the Intake Bay. The cooling water was supplied continuously, even when the pumps were not operating. This modification provided an alternate lubrication/cooling design by modifying the Millstone Unit No. 2 Screen Wash Pumps P-8A and P-8B and the on-site spare. The existing chlorinated domestic water lubrication/cooling flow was replaced with pump discharge sea water and only when the pump is operating. This was accomplished by creating a flow path in the packing box such that the pumped fluid (non-chlorinated sea water) can flow upward through the flushing grooves of the bushing to the packing lantern ring and exit the bearing housing via the existing grease cup fitting and tubed to discharge to the drain, located in the lower part of the pump shaft packing housing. A needle valve was installed in the discharge tubing to limit the cooling/lubrication flow, as required. The existing domestic water piping and valves were removed downstream of 2-WW-298 to the pumps. Final Safety Analysis Report (FSAR) Figure NO 9.10-02 was updated to reflect the change. The modification to provide an alternate lubrication/cooling design was performed on the existing pumps in 1999. The modification has now been completed on the spare Screen Wash Pump.

Reason for the Activity

This modification was implemented based on the Connecticut Department of Environmental Protection (DEP) taking issue with the discharge of bearing lubrication water for the Screen Wash Pumps into the Intake Bay, and in particular, the use of domestic city water as cooling water due to the fact that it contains chlorine. This was a non-permitted discharge and considered a compliance issue by the state DEP. To eliminate unlicensed chlorinated water discharge, the use of domestic water for bearing lubrication/cooling was discontinued.

Safety Evaluation Summary

The change provided an equivalent method of lubrication using the pump discharge water for upper bearing cooling. The FSAR Figure 9.10-2 was updated to reflect the removal of the applicable domestic water piping. The function of the screen wash pumps is to provide a means of removing debris from the traveling water screens thus ensuring adequate inventory for the Circulating Water and Service Water pumps. However, the function is not safety related. The Screen Wash Pumps and the Domestic Water System are not required to perform any safety related function. The change does not affect the probability of occurrence of any malfunctions or accidents, does not affect the consequences of malfunctions or accidents, and does not create new malfunction or accidents.

S2-EV-99-0033
FSARCR 99-MP2-38
DCR M2-99004

Safety Injection Tank Nitrogen System Modification

Description

This modification upgraded the entire Safety Injection Tank (SIT) vent Air Operated Valve (AOV) assemblies, including all valve accessories such as solenoid valve, limit switch, regulator, air actuator, and associated control tubing, to a QA Category 1 qualification level. Also, the electrical service to the solenoids associated with AOVs (2-SI-613, 2-SI-623, 2-SI-633, and 2-SI-643) was upgraded to satisfy Class 1E requirements and on a different electrical train other than the one that their associated SIT isolation Motor Operated Valve (MOV) utilize. This allows remote nitrogen venting of the safety injection tanks if the SIT isolation MOV can not be closed due to loss of emergency power. This activity also permanently removes the vent caps installed downstream of the AOVs.

Reason for the Activity

Venting of the SITs in the existing condition required the removal of the caps installed downstream of the AOVs (2-SI-613, 2-SI-623, 2-SI-633, and 2-SI-643) and an operator to open the AOVs remotely. The new design eliminates the need to remove those caps and allows remote operation of the AOVs to vent the SITs during a Small Break Loss of Coolant Accident (SBLOCA), preventing nitrogen intrusion into the Reactor Coolant System (RCS).

Safety Evaluation Summary

The modification of the SIT prevents nitrogen intrusion into the RCS during a SBLOCA. The modification was performed in a quality controlled manner, using quality components, and required verification ensuring a quality installation. The modification did not result in any increase in the probability or consequences of any malfunctions or accidents involving equipment important to safety. No new active failure of the AOV is credible. The activity ensures that the post-accident operation of required systems, as described in the basis of the Technical Specifications, is protected. The SIT vent valve assemblies are not accident initiators. All components are seismically and environmentally qualified. This change did not result in the creation of a malfunction or accident of a different type than previously evaluated in the Safety Analysis Report.

S2-EV-99-0052
MMOD M2-99017

Steam Generator Blowdown Flow Instrumentation

Description

This modification installed ultrasonic flowmeters to monitor steam generator blowdown flow. Steam Generator Blowdown flow rate is controlled by manipulation of manual valves in the blowdown system. The blowdown flow rate is determined by using a set of operating procedure curves which correlates valve position to blowdown flow.

Reason for the Activity

The flowmeters were used to support validation of the operating procedure curves. This was done to resolve concerns with respect to the quality of the curves and ensure that the curves support inaccuracy assumptions contained in the effluent report.

Safety Evaluation Summary

The addition of non-intrusive ultrasonic flow instruments (transducers and electronics) to the steam generator blowdown system and associated conduit, wiring and supports does not adversely affect the operation of any plant system. The existing logic for isolation of blowdown on high radiation was unaffected by this change. The validated operating procedure curves will continue to be used, as required in the plant Technical Specifications, to determine blowdown flow. The instrumentation will be available to revalidate the flow curves whenever operations is required to adjust blowdown

Based upon the results of this safety evaluation: the addition of the flow instrumentation, electronics, transducers, conduit, wiring, and supports will not affect the probability of occurrence or consequences of any accident or malfunction previously evaluated, does not result in any new malfunctions or accidents not previously evaluated, and does not reduce the margin of safety as defined in any Technical Specification.

S2-EV-99-0084
Temp. Mod. 2-99-007

Containment Camera Installation

Description

This temporarily installed nine (9) video cameras, associated control boxes, wiring, and supports within containment. Additionally, two (2) cables exited containment via a spare module and were routed to the control room for video monitoring and camera control. This temporary modification has been removed.

Reason for the Activity

During reactor power operations, the containment is sealed and generally not entered due to the radiological environment. Consequently, cameras were installed in various locations to monitor sensitive areas within containment. The cameras allowed Operations Department personnel to visually monitor containment without a containment entry.

Safety Evaluation Summary

The cameras were installed as a non-intrusive containment monitoring device. The cameras were not relied upon for any safety related purpose and were placed so as not to impact the operation of any safety related equipment. The additional mass added to containment was insignificant and did not alter any accident mitigation strategies. Power was supplied by non-emergency lighting panels, cable separation was provided, and cable runs were not placed in any cable trays. This temporary modification did not affect malfunctions or accidents evaluated, or not evaluated, in the Safety Analysis Report. It did not impact the margin of safety specified in the basis of the Technical Specifications.

S2-EV-99-0090
Temp. Mod. 2-99-014

Temporary Modification to Redirect Reactor Coolant Pump (RCP) Vapor Stage Leak-off from the Primary Drain Tank (PDT) to the Containment Sump

Description

This temporary modification redirected the Reactor Coolant Pump (RCP) seal vapor stage leak-off flow from the Primary Drain Tank (PDT) to the containment sump. The temporary modification to each of four reactor coolant pumps consisted of:

1. Removing flanged piping spool in vapor seal leakoff drain line,
2. Installing blind flange on the PDT side of drain line,
3. Installing flange, pipe nipple, and flexible drain hose from RCP to containment sump drain trench.

This temporary modification has been removed.

Reason for the Activity

Boric acid accumulation on top of the seal packages of all four reactor coolant pumps indicated that the vapor stage leak-off/drain to the PDT was not working properly. The vapor stage leakoff is supposed to gravity drain to the PDT. Nitrogen overpressure in the PDT and the drain piping layout precluded the proper drainage of this seal leakage. This resulted in the leak-off water backing up and flowing out the top of the seal housing. Dispersion by the rotating pump shaft and evaporation caused a boric acid buildup around the area inside the driver mount (seal package, seal cooler, vibration probes, etc.).

Safety Evaluation Summary

This modification did not alter the basic purpose of the vapor stage leak-off line. The modification involved only non-safety related piping and had no effect on the performance of any safety related equipment. The design and materials of construction for this modification were reviewed and determined to be suitable for the intended service and capable of handling the expected pressures, temperatures, and flow rates.

S2-EV-99-0130
MMOD M2-99-043

Nitrogen Supply Line Modifications to Quench Tank and Primary Drain Tank (PDT)

Description

1. Reroute the nitrogen supply piping by disconnecting from the common supply piping (nitrogen and primary makeup water) and connect directly to the quench tank vent piping at a higher elevation (elevation -12' 2").
2. Add a drain trap at the lowest elevation of the quench tank vent piping leading to the waste gas header downstream of AOV 2-RC-400 to discharge any water present in the vent piping due to overfilling of the quench tank or due to condensation. Route the drain line to the containment sump drains.
3. Replace the existing relief valve, 2-LRR-296, on the PDT with a larger capacity relief valve.
4. Remove the internal springs in check valves 2-RC-430 and 2-LRR-51 on the nitrogen supply line to the quench tank and the PDT.

Reason for the Activity

1. This modification eliminates back pressure on the check valve, 2-RC-430 on line ¾"-HCD-31, to provide nitrogen flow to the quench tank.
2. When filling the quench tank with water, inadvertent overfilling could lead to water collecting in piping upstream of 2-RC-400 and subsequently being discharged into the waste gas header. This water column has the potential to cause the relief valve 2-RC-242 to lift or prevent the discharge of the waste gases into the waste gas header from the quench tank during establishment of the nitrogen blanket. Provision of a branch drain line to the containment sump on the quench tank vent drain line upstream of 2-GR-100 with a drain trap will automatically drain any water collected in the lowest elevation when 2-RC-400 is opened for waste gas discharge to the waste gas header. A ¾ inch globe valve will be added upstream of the drain trap to serve as a boundary test valve for local leak rate testing of containment isolation valves 2-GR-11.1 and 2-GR-11.2.
3. The code relief valve, 2-LRR-296, is not adequately sized to ensure protection of the PDT if the pressure regulator, 2-LRR-52.1, failed open during the use of high pressure nitrogen for makeup to the Safety Injection Tanks.
4. This improved the flow to these tanks without affecting the function of the check valves.

Safety Evaluation Summary

These modifications are safe and will not impact any other equipment or systems important to safety. Neither the nitrogen supply line nor the quench tank to which the nitrogen is supplied is safety related equipment. Failure of the drain trap to be leak tight will not cause a breach in waste gas header boundary since remote isolation valves 2-RC-400 and 2-GR-11.1 are normally closed. The nitrogen piping is designed for a much higher pressure than that of the PDT or the Quench tank. Therefore, failure of the check valve will not cause the nitrogen piping to fail. These modifications will not cause an unsafe condition.

S2-EV-99-0133
MMOD M2-99048

Reactor Coolant Pump (RCP) Seal Controlled Bleed-off Relief Valve Replacement

Description

The change replaced the RCP controlled bleed-off relief valve, 2-CH-199, with a hard faced seat, flanged design, relief valve. The original valve was manufactured by the Lonergan Valve Company. In several situations, the relief valve made by the Lonergan Company have a history of leaking slightly following pressure transients. In place of the Lonergan valve, a hard seated relief valve made by the Crosby Valve Company was installed. The new valve incorporated a hardened seat that would make leakage less likely and a flanged connection that would enhance valve installation and minimize radiation exposure to the mechanics.

Also as part of MMOD M2-99048, flow switch FS-2038, previously abandoned in place, was removed.

Reason for the Activity

While shifting RCP seal controlled bleed off flow from the Volume Control Tank to the Equipment Drain Sump Tank, and vice versa, valve 2-CH-199 lifted in response to the pressure transient, and did not completely reseat. The existing valve had a history of leakage past its seat, which is input to the primary drains tank as TS 3/4.4.6.2 identified RCS leakage. This resulted in multiple overhauls, which dated back to plant start-up. The valve was located within a Tech Spec High Radiation area, making frequent removal to relap/repair the valve seat an ALARA concern and a maintenance burden.

Safety Evaluation Summary

The piping affected by this modification is designed to the requirements of ASME Section III Class 2 and Class 3 requirements. Meeting the code requirements ensures a high degree of assurance that the pressure boundary will not fail. The change does not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety or a previously evaluated accident. No fission product boundaries are adversely affected by this change, therefore, there is no increase in dose consequences as a result of this change. The activity does not create the possibility of a malfunction or an accident of a different type. The margin of safety as defined in the Technical Specifications is not reduced. The activity meets the original design basis requirements and does not cause any increase in the risk to the public.

S2-EV-99-0135
FSARCR 99-MP2-96
DCN DM2-02-0825-99
MMOD M2-99047

Reactor Coolant Pump (RCP) Vapor Stage Leak-off Re-Route to the Containment Sump

Description

This plant change redirected the four RCP vapor stage leak-off drains to the Containment Sump via the Containment Drain Trench in lieu of the Primary Drain Tank (PDT) by making the following modifications:

a. Removed a section of flanged pipe spool piece and associated check valves on each RCP vapor stage leak-off drain line.
b. Removed the blind flange on PDT side of each vapor stage leak-off line.
c. Installed flange and pipe nipple to interface pipe to tubing.
d. Installed 1-inch Stainless Steel tubing and associated tubing supports along the bio-shield between the pipe nipple described in item #1.c above at each RCP and the containment drain trench. Flushing connections and union connections were provided for each new drain line.
e. Removed the existing four drains and associated check valves located between the blind flanges described in item #1.b and the PDT. Installed pipe caps on the existing pipe connections to the PDT header.

Reason for the Activity

Boric acid accumulation on top of the seal packages of all four RCPs indicated that the vapor stage leak-off/drain to the PDT was not working properly. The vapor stage leak-off is supposed to gravity drain to the PDT. Nitrogen overpressure in the PDT and the drain piping configuration precluded the proper drainage of this seal leakage. This resulted in the leak-off water backing up and flowing out the top of the seal housing. Dispersion by the rotating pump shaft and evaporation caused a boric acid buildup around the area inside the driver mount (seal package, seal cooler, vibration probes, etc.).

Safety Evaluation Summary

Based on the minor seal leakage flow involved, the change in RCP vapor stage leak-off destination from the primary drain tank to the containment sump has been determined to be consistent with existing Technical Specification leakage criteria. This modification does not alter the basic purpose of the vapor stage leak-off line. The modification involved only non-safety related piping and had no effect on the performance of any safety related equipment. The design and material of construction for this modification were reviewed and determined to be suitable for the intended service and capable of handling the pressures, temperatures, and flow rates expected. The modification had no signification impact on the Millstone Unit No. 2 Containment Mass Inventory Tracking Program.

S2-EV-99-0137
DCR M2-99052

Containment Dew Point Temperature Monitoring Instrumentation Removal

Description

DCR M2-99052 implemented design basis modifications to the plant and removed the dew point temperature monitoring system from service. This modification consisted of removing the dew point temperature indicator, selector switch, and associated wiring from main control board C01, and the removal of Foxboro Instrumentation in C01R. Field instruments, ME-8064, 9772, 9773, and 9774, and cable located inside containment were retired in place. Dew cell power supplies located in the auxiliary building were removed. Field cable was spared and retired in place.

Reason for the Activity

Plant Incident Report 2-96-038, dated January 19, 1995, reported that inside containment dew cells require more frequent maintenance (6 month cycle) than currently being performed to ensure operability. In addition, control room panel deficiency reports identified that containment air dew point indicator on control board C01 failed low and was continuously out of service. The dew cells were not reliable, were inaccurate, and required frequent maintenance to maintain operability, which could not be performed on line. Containment entries to perform this maintenance is not practical. EWR M2-95040 requested that this instrumentation be removed from service.

Safety Evaluation Summary

Dew point temperature monitoring instrumentation is not an input to the Reactor Coolant System (RCS) leakage mass balance calculation and is not required to be operable per Technical Specifications for RCS leakage detection systems. This instrumentation does not perform accident mitigating functions and is not required for post accident monitoring. Removal of the dew point temperature monitoring instrumentation does not impact the operation or function of any safety related system or component, or RCS leak detection capability and does not degrade safety barriers. The removal of the dew point temperature monitoring instrumentation does not impact licensing basis and system conformance with Licensing Basis requirements of RCS leakage detection per Regulatory Guide 1.45 and General Design Criteria (GDC) 30 of Appendix A to 10 CFR 50 and Leak Before Break Methodology GDC 4.

S2-EV-99-0142
DCR M2-99050

Safety Parameter Display (SPDS) Upgrade - Hardware

Description

This change implemented hardware changes to the SPDS system to install a dedicated and continuous SPDS monitor within the Unit 2 Control Room and add additional points. The additional SPDS computer points were derived from Hydrogen Purge Isolation Valves 2-EB-91 and 2-EB-100.

An additional monitor and Central Processing Unit (CPU), dedicated to SPDS, were installed within the Millstone Unit No. 2 Control Room and located within the Shift Manager's console. The existing CPU has SPDS continuously available but not continuously displayed.

Reason for the Activity

Hardware changes were necessary because containment isolation valves 2-EB-91 and 2-EB-100 did not provide close position indication into the plant process computer.

The installation of the dedicated monitor within the Control Room for SPDS is required to bring the plant into compliance with NUREG 0737. NUREG 0737 requires SPDS to be continuously displayed within the Control Room.

Safety Evaluation Summary

Providing close position indication for valves 2-EB-91 and 2-EB-100 into the Plant Process Computer for use by SPDS as well as the addition of a dedicated SPDS monitor within the Millstone Unit No. 2 Control Room does not represent an increase in the risk to the health and safety of the public.

Providing a dedicated SPDS monitor for the Control Room ensures Millstone Unit No. 2 is in compliance with the requirements of NUREG 0737 with regards to SPDS. The modifications do not degrade the ability of any safety related system from performing its intended safety function.

S2-EV-99-0144
Temp. Mod. 2-99-030

Temporary Contractor Water Treatment Facility

Description

This temporary modification relocates part of the current water treatment vendor (ECOLOCHEM) outside of building 215 and installs a temporary water treatment facility (IONICS) partly inside and outside building 215.

Reason for the Activity

This temporary modification is in place to maintain a continuous supply of make-up water for Millstone Unit Nos. 1 and 2 while the new vendor's permanent system is being implemented. This configuration facilitates the ability to run either vendors' system and also allows Millstone Unit No. 2 to cease operating the Millstone Unit No. 2 ECOLOCHEM system.

Safety Evaluation Summary

This change does not increase the probability or consequences of a malfunction of equipment important to safety since the water treatment system function is not changed by this temporary modification and the temporary modification does not introduce any new system interactions. The required quality of the temporary water treatment facility product water is consistent with the quality recommendations of the Steam Generator's Owners Group for both Millstone Unit Nos. 2 and 3 and equivalent to the quality produced by the current vendor.

There are no accidents associated with the installation of this temporary modification. The water treatment system does not initiate, prevent or mitigate any of the accidents evaluated in the Safety Analysis Report. It does not interact with safety related equipment in such a way as to introduce an unanalyzed accident.

S2-EV-99-0145
Temp. Mod. 2-99-031

Removal of Loop 1 Hot Leg Resistance Temperature Detector (RTD), TE-112HD, from Reactor Protection System (RPS) Channel D Input - Substitution with Loop 2 Hot Leg RTD, TE-122HD

Description of Change

This temporary modification removed temperature input TE-112HD from RPS Channel D and substituted Loop #2 hot leg temperature signal from TE-122HD. This modification has been removed.

Reason for the Activity

RPS Channel D Loop #1 hot leg temperature input from temperature element TE-112HD was intermittently spiking. The problem caused the RPS Channel D Variable High Power, Local Power Density, and Thermal Margin Low Pressure trips to occur. Troubleshooting identified the problem to be within containment but did not specifically identify the source of the problem.

Safety Evaluation Summary

Substituting temperature element TE-112HD with TE-122HD within RC30D hot leg loop averaging circuitry allowed the RPS to perform its intended safety function but be in a condition outside of the Final Safety Analysis Report. Analysis performed by Siemens Power Corporation in support of this temporary change demonstrated the acceptability of the modification and determined the change to be safe and bounded by the new analysis.

The change was not a risk to the health and safety of the public. The RPS was capable of performing its intended safety function as well as its intended safety function as delineated within Technical Specifications.

S2-EV-99-0149
DCR M2-99065
FSARCR 00-MP2-4
FSARCR 00-MP3-3

Redesign of Existing Severe Line Outage Detector (SLOD) System and Master Supervisory Panel

Description

The Severe Line Outage Detection (SLOD) system is designed to prevent instability and loss of all generation at Millstone Station. Besides avoiding unit instability, a distribution system casualty with generation above 1300-1400 MW at Millstone Station could have severe, adverse consequences on Pennsylvania and/or New York grid reactive and thermal operating conditions. The SLOD system is continuously armed and avoids instability and loss of all generation at Millstone by tripping only pre-selected units when certain conditions exist. The tripping logic associated with the SLOD system was modified to remove all trips associated with Millstone Unit No. 1. The Double Line and Breaker Failure Detection Unit Rejection Special Protection System (DBURS), two more Special Protection Systems (SPS) used to trip pre-selected units at Millstone, were deleted and removed since their functions were no longer needed due to the loss of Millstone Unit No. 1 generation. CRP-909 was connected to the master supervisory panel in the 345-kV switchyard via a new fiber optic cable. Switches on CRP-909 for control of Millstone Unit No. 1 switchyard circuit breakers and motor operated disconnects were removed.

Reason for the Activity

The decision to decommission Millstone Unit No. 1 requires the re-powering or relocation of certain systems needed for continued operation of Millstone Unit Nos. 2 and 3. The existing SLOD system was located in the Millstone Unit No. 1 control room and received power from Millstone Unit No. 1 power sources. Since this capability is still required after the decommissioning of Millstone Unit No. 1, the system required re-powering from Millstone Unit No. 2 to protect the off-site electrical grid during operation of Millstone Unit Nos. 2 and 3.

Safety Evaluation Summary

The changes made to the SLOD system do not affect any Design Basis Accidents or the consequences of these accidents. The changes to the SLOD system do not create new accidents beyond those analyzed in the Millstone Unit Nos. 2 and 3 FSARs. It does not increase the probability of an accident or malfunction, increase the consequences of an accident or affect accident mitigation. The mechanism by which SLOD trips the switchyard circuit breakers for Millstone Unit No. 3 was not affected by any of the modifications to CRP-909 or the switchyard Master Supervisory Panel. These two panels provide indication, control and alarm functions, but have no impact on the protection SLOD provides to both Millstone Station and the off-site electrical grid.

S2-EV-99-0150
FSARCR 00-MP2-3
DCR M2-99060

Hydrogen Bulk Storage Skid Relocation

Description

This change installed a new Hydrogen Storage Facility and piping system to the west side of the Millstone Unit No. 2 Turbine Building and abandoned the shared Millstone Unit No. 1 Facility. To protect adjacent buildings from possible fires, an L-shaped fire wall was installed between the buildings and the hydrogen storage unit. Excess flow check valve 2-GAH-271, and bypass valve 2-GAH-272, were relocated to the supply line outside the Turbine Building behind the fire wall, and a spool piece was installed with purge valves on both the upstream and downstream side to facilitate purging of the bulk storage facility or the hydrogen supply line. The supply line runs from the west side of the turbine building to the east side, tying in upstream of test connection 2-GAH-41 and the one inch line connecting to the outer sleeve of the encapsulated pipe going to the Volume Control Tank (VCT). The hydrogen supply piping runs through well-ventilated areas and is located at safe distances from equipment that present a fire hazard to hydrogen. Guarded pipe vented to the outside is used for the hydrogen piping to protect it from external damage and provide an additional vent path as a safeguard for leaks. The guarded pipe runs outside of the Turbine Building to a one inch vent line.

Reason for the Activity

The decommissioning of Millstone Unit No. 1 required removing the pressured hydrogen gas line within the Millstone Unit No. 1 facility. Therefore, an alternate location for the bulk hydrogen storage facility and rerouting of the hydrogen supply piping for Millstone Unit No. 2 was required. Installation of the new bulk hydrogen storage facility provides Millstone Unit No. 2 with the ability to meet all its hydrogen gas requirements at identical operating and design conditions. A required Final Safety Analysis Report (FSAR) change identified the new location of the bulk hydrogen storage facility (Section 10.2.5) and delete the Hydrogen Storage Facility as a shared system (Section 1.2.13).

Safety Evaluation Summary

Design Features required by the Fire Protection Program and FSAR Section 10.2.5 have been met. The hydrogen system provides no nuclear safety function. Hydrogen is supplied to the turbine generator rotor and the VCT at the same operating and design conditions as the original design. Therefore, there is no impact on the turbine generator or VCT. The change did not affect any protective boundary. The relocation of the hydrogen storage facility and the new hydrogen piping run in the Turbine Building do not impact systems or components required to mitigate the consequences of a malfunction of equipment important to safety. The change did not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety, nor did it create the possibility of a malfunction of a different type. The change did not increase the probability of occurrence or consequences of an accident previously evaluated nor did it create the possibility of an accident of a different type.

S2-EV-99-0151
MMOD M2-99057

Modification of Main Condenser Steam Dump Valves and Turbine Bypass Valve, (2-MS-206, 207, 208 and 209)

Description

This change modified the control air system for the main condenser steam dump valves, 2-MS-206, 207, 208 and 209. This modification included the elimination of an air booster relay to improve valve stability and installation of a second solenoid to support the quick open requirements of the valves.

Reason for the Activity

The activities associated with this modification are non-QA, however, section 10.3 of the Final Safety Analysis Report (FSAR) states that the main condenser steam dump valves, 2-MS-206, 207, 208 and 209 will quick open in less than 3 seconds following a reactor/turbine trip. This change modified the air supply piping to the valve actuators to provide sufficient air supply to allow the valves to open within 3 seconds. The quick opening function of the main condenser steam dump valves following a reactor/turbine trip is needed to prevent lifting of the Main Steam Safety Valves (MSSVs) following a reactor/turbine trip. The elimination of the booster relay was done to minimize control valve instability when modulating valve position during control of steam generator pressure during plant startup and shutdown.

Safety Evaluation Summary

This change does not increase the probability of occurrence of a malfunction of equipment important to safety or accidents previously evaluated. The purpose of the modification is to prevent lifting of the MSSVs following an uncomplicated reactor/turbine trip and operation of the main condenser steam dump valves and turbine bypass valve to eliminate the possibility that a MSSV will stick open, causing an uncontrolled cooldown. The change does not increase the consequence of a malfunction of equipment important to safety or of accidents previously evaluated in the Safety Analysis Report. The main condenser steam dump valves and the turbine bypass valve are not used for accident mitigation. The change does not create the possibility of a malfunction or an accident of a different type than any previously evaluated. The change to the air supply piping does not change the logic, the function, or the operation of the main condenser steam dump valves and the turbine bypass valve.

It does not reduce the margin of safety as defined in the basis for any technical specification. The modification is to simply revise the air supply piping for the Main Condenser Steam Dump valves to allow the valves to open within three seconds following a reactor/turbine trip as defined in the Millstone Unit No. 2 FSAR. The modification of the air supply piping required a revision of the Millstone Unit No. 2 FSAR figure 10.03-01, sheet 01. The change to the valve air supply piping for the main condenser steam dump valves is a non-QA modification. This modification has no adverse effect on the performance of any safety related equipment.

S2-EV-00-0001
FSARCR 00-MP2-7
DCR M2-99066

Removal of Reactor Protection System Calibration and Indication Panel (RPSCIP) Flow Dependent Setpoint Selector Switch (FDSSS)

Description

The FDSSS function was permanently removed from the RPSCIP circuitry.

Reason for the Activity

The design of the Millstone Unit No. 2 Reactor Protection System (RPS), includes the provision to ensure adequate protection when operating the plant at reduced power with one or two Reactor Coolant Pumps (RCPs) taken out of service. To operate with one or two RCPs taken out of service, a FDSSS for each RPS channel would be taken out of the four pump position and placed in the corresponding position (three or two pump) for the RCP configuration. This would result in the reactor coolant low flow, thermal margin/low pressure, and the variable high power trip setpoints being simultaneously changed to the established values for the selected pump configuration.

Although provisions are made in the RPS to permit operation of the reactor at reduced power with one or two RCPs taken out of service, Millstone Unit No. 2 is not licensed for this mode of operation.

Safety Evaluation Summary

The removal of the FDSSS, including the re-wiring of the RPSCIP circuitry for four pump operation only, and the revisions to the plant drawings, the Final Safety Analysis Report Sections and Figures, and the Bases for Technical Specifications do not adversely affect the operation of any safety or non-safety related structures, systems or components.

Additionally, this change does not result in any new malfunctions or increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated.

S2-EV-00-0002
FSARCR 00-MP2-6
FSARCR 00-MP2-27
DCR M2-00007

Millstone Unit No. 2 Spent Fuel Pool Cooling Analysis for 2R13 & 2C14

Description

DCR M2-00007 revised the design basis for the Spent Fuel Pool (SFP) and SFP Cooling System. There were no physical plant modifications. This Safety Evaluation is written to support a revised analyses for the Millstone Unit No. 2 SFP cooling system for the refueling outage at the end of operating cycle 13, and for operating cycle 14. The change presents the decay heat loads imposed on the SFP by refueling operations, as well as normal operation or emergency conditions, and presents the thermal-hydraulic analysis of the SFP under a variety of conditions employing the SFP cooling system, the shutdown cooling system, or combinations of the two.

Reason for the Change

Plant Condition Reports and Graded System Reviews identified that portions of the SFP cooling analysis were inconsistent with existing operating procedures. Due to these inconsistencies, new calculations were developed to correct the identified problems.

Safety Evaluation Summary

There are no physical plant changes associated with this design change. The activity serves to correct inconsistencies in the previous design basis. Since there are no physical changes to the plant, nor any change in how plant systems are normally operated, there is no increase in the probability or consequences of existing analyzed malfunctions or accidents, nor is there any possibility of creating a new malfunction/accident. The margin of safety is not reduced by the revised analysis for End of Cycle 13 refueling and Cycle 14 operation, since: (1) the SFP bulk temperature is maintained less than 127°F during MODE 1, 2, 3 and 4 and 140°F for MODE 5 and 6 in accordance to proposed Technical Requirements Manual limits, (2) the SFP heat loads are not increased, (3) the time to boiling is conservatively increased following a complete loss of SFP cooling, (4) the emergency core offload analysis for Cycle 14 is not changed from that credited for Cycle 13, and (5) the ability of the SFP cooling system to perform its intended function post-Loss of Coolant Accident is reasonably assured because the structural integrity of the system is maintained.

S2-EV-00-0003
DCN DM2-00-0293-98

Auxiliary Steam and Condensate System Modification

Description

The Auxiliary Steam and Condensate System was modified by adding a steam powered pump-trap unit with related accessories to coil X-41 per DCN DM2-00-0293-98.

Reason for the Activity

The Auxiliary Steam System was modified to improve the condensate return alleviating coil freeze-up. Since this modification affects the Piping & Instrumentation Drawings (P&IDs) which are also figures in the Final Safety Analysis Report (FSAR), a safety evaluation was required.

Safety Evaluation Summary

The Auxiliary Steam System modification was reflected on P&ID 25203-26026, sheets 3 and 4 and FSAR Figure 09.13-01, sheets 03 and 04. The modification was determined to have no effect, directly or indirectly on any safety related function. The change was reviewed in accordance with 10 CFR 50.59 guidance and was determined to be safe, since the system did not perform a safety function nor impact any safety related system, structures or components.

S2-EV-00-0004
DCR M2-99068

Millstone Unit No. 2 Fire Water Connections Cross-Tie Project

Description

This modification installed new fire water piping to the Millstone Unit Nos. 1 and 2 common fire hose stations, supplied from a Millstone Unit No. 2 source. The existing common fire hose stations remained essentially unchanged, except for their supply source piping which was upgraded to satisfy Millstone Unit No. 2 licensing commitments.

Reason for the Activity

The decommissioning of Millstone Unit No. 1 created a situation where the existing supplied fire suppression systems utilized by both Millstone Unit Nos. 1 and 2 , supplied from a Millstone Unit No. 1 source as common equipment, will no longer be available to Millstone Unit No. 2 unless supplied by a Millstone Unit No. 2 source. The addition of new fire suppression piping to the existing common fire hose stations from a Millstone Unit No. 2 source satisfied the fire suppression requirements to the common fire hose stations and permits their use for Millstone Unit No. 2.

Safety Evaluation Summary

These changes did not adversely affect any design basis accident or its consequences. This change provides an acceptable source of fire water from Millstone Unit No. 2 since Millstone Unit No. 1 is being decommissioned and has been separated from Millstone Unit No. 2.

The installation of the new fire water supply piping and its supports to the existing common fire hose stations is in accordance with NFPA 14 and satisfies the intent of the Millstone Unit No. 2 requirements for these fire hose stations. This installation meets the requirements of the original installation and satisfies seismic II over I requirements. This change does not impact the ability to safely shutdown.

All penetrations are re-sealed in accordance with the applicable seal drawing details. Furthermore, the addition of these Millstone Unit Nos. 1 and 2 common fire hose stations to the existing Millstone Unit No. 2 fire water header is within the fire water system hydraulic envelope and all system pressure and flow rates are unchanged. There is no effect on the main fire water loop, and no impact on the ability to extinguish an Appendix R fire. The Millstone Unit No. 2 combustible loading calculation has been reviewed and the Millstone Unit No. 1 Heat Load Classification remains low. There are no additional combustibles to be considered.

This change is safe and does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report and does not degrade the margin of safety as defined in the basis for any technical specification. No Technical Specifications or bases are affected by this change for any mode of operation.

S2-EV-00-0006
Technical Evaluation M2-EV-00-0001

Auxiliary, Turbine & Enclosure Buildings Auxiliary Steam Heater Replacements

Description

Replacements of self-contained steam unit heaters and ventilation air handler steam coils servicing the Auxiliary Building, the Turbine Building and the Enclosure Building require a revision to Final Safety Analysis Report (FSAR) figures. The replacement of ventilation air handler steam coils and self-contained steam unit heaters by components of different capacity, either smaller or larger, was evaluated and determined to have no impact on safety related equipment.

Reason for the Activity

Replacement of a steam coil or a steam unit heater is required due to material degradation.

Safety Evaluation Summary

The replacement of ventilation air handler steam coils and self-contained steam unit heaters which are not a "one-for one" replacement are reflected in FSAR Figures 09.13-01 sheets 03. and 04. This type of modification has been determined to have no effect, directly or indirectly, on any safety related function.

S2-EV-00-0007
SP2605P.3A
Temp. Mod. 2-00-002

Installation of Freeze Seal to Support Leak Test of 2-SI-460

Description

This temporary modification installed a freeze seal on line 6" GCB-10 upstream of vent valve 2-SI-037 to perform Inservice Testing leak testing on Low Pressure Safety Injection/Containment Spray Test Header Stop Valve 2-SI-460. This temporary modification has been removed.

Reason for the Activity

This activity was required to provide a control test boundary for leak testing 2-SI-460 without affecting the Shutdown Cooling System (SDC) operation.

Safety Evaluation Summary

The freeze seal was installed while the plant was in Mode 6 with SDC operating. The test volume between the seal and 2-SI-460 did not affect SDC operation and did not increase the potential for a boron dilution event. The requirements of the technical requirements were met at all times during this activity.

Design Engineering performed an assessment of the freeze seal. The assessment concluded that the additional mass added by the freeze seal was insignificant in the 6" line and no additional supports were required. The piping configuration was reviewed and found to have sufficient flexibility to absorb the contraction of the pipe due to the freeze seal at the specified location. The probability of a freeze seal failure was low due to the piping material being stainless steel which is well suited to freeze seal formation. The pipe integrity Non Destructive Examination testing verified suitability prior to installing the seal.

S2-EV-00-0016
Temp. Mod. 2-00-003

Installation of Freeze Seal to Support Relief Valve Setpoint Testing of 2-SI-469

Description

This temporary modification installed a freeze seal on line 1"-CCA-10 upstream of relief valve 2-SI-469 to perform relief setpoint bench testing on Shutdown Cooling (SDC) suction line relief 2-SI-469. This temporary modification has been removed.

Reason for the Activity

Testing the class one relief valve was required as part of Technical Specification 4.0.5. The valve could not be tested in place and no isolation valve exists. Therefore, a freeze seal had to be established to provide an adequate SDC system boundary.

Safety Evaluation Summary

The freeze seal was installed while the plant was in Mode 5 with Shutdown Cooling System operating. The freeze seal did not affect SDC operation and temporarily removing the relief for bench testing was acceptable since the function of the relief is to provide SDC piping overpressure protection for Reactor Coolant System pressure and provide a thermal relief function when at power with 2-SI-651 and 2-SI-652 isolated. In addition, while the valve was removed, a Category I blank flange was installed as a SDC relief line secondary boundary. This allowed SDC to remain operable.

Design Engineering performed an assessment of the freeze seal. The assessment concluded that the additional mass added by the freeze seal was insignificant in the 1" line and no additional supports were required. The piping configuration was reviewed and found to have sufficient flexibility to absorb the contraction of the pipe due to the freeze seal at the specified location. The probability of a freeze seal failure was low due to the piping material being stainless steel which is well suited to freeze seal formation. The pipe integrity Non Destructive Examination verified suitability prior to installing the seal.

S2-EV-00-0017
Temp. Mod. 2-00-006

Containment (CTMT) Closure During R13 for Steam Generator (SG) Activities

Description

This temporary modification utilized containment penetrations to allow access into containment during times of containment closure. Containment closure is defined as the containment condition where at least one integral barrier, to the release of radioactive material, is provided. The temporary modification installed a special device to be utilized as a containment closure boundary, in accordance with the requirements of Technical Specification 3.9.4. The special device consisted of a temporary spool piece, manufactured from a 6 inch pipe flange welded to the 14 inch section of 6 inch pipe. Cables, hoses, and/or wires were passed through the temporary spool piece and sealed in place. All hoses were equipped with a valve located outside containment for closure. The hoses and equipment in the sludge lance trailer and air compressor become a closed system outside containment. The sludge lancing equipment transported fluid into or out of containment and the air line pressurized.

Reason for the Activity

During refueling outages, SG Eddy Current Testing (ECT) and Secondary Side Cleaning are performed concurrent with fuel movement. To ensure timely completion of either task, direct access into the containment for the ECT/Sludge Lancing equipment (e. g., cables, fiber optics, and hoses) is necessary during periods when a containment boundary is required. Additionally, insufficient services within containment (e. g., Station air, Temporary Electrical power, containment cameras, computer cables,) resulted in other departments requesting support to access containment. Access was provided through the special device.

Safety Evaluation Summary

With the installation of this temporary modification, there were no penetrations which communicated or provided direct access from the containment atmosphere to the outside environment. Once in place, the special device became a passive component. In the event of a Fuel Handling Accident, the subject modification would become a functional barrier to prevent fission products, released into the containment atmosphere at atmospheric pressure, from migrating to the outside environment. Since the temporary modification was a passive component and no open penetrations were exposed to containment atmosphere, there was no loss of function, or additional potential leakage path.

In the unlikely event of a Loss of Decay Heat Removal, the subject modification was a functional barrier to contain the fission products released into the containment atmosphere. Prior to core boiling, which is required to raise containment pressure, the penetration isolation valves would have been closed as directed by procedure. Consequently, this accident would have had no adverse impact on containment boundary. The change did not introduce any new malfunctions or impact the consequences of any previously evaluated malfunction and did not change or impact the accident analysis.

S2-EV-00-0024
FSARCR 00-MP2-19
DCR M2-99059

Addition of Braided Expansion Joints to the Spent Fuel Pool (SFP) Cooling System

Description

This modification installed braided expansion joints at the SFP cooling pumps' suction flanges in addition to pipe support modifications on the suction and discharge of the SFP cooling piping. These modifications lowered the suction and discharge loads on the pumps' nozzles to within the vendor's allowable limits.

In addition, in order to reduce a local thermal expansion stress condition in the SFP cooling pump suction piping, the normally isolated SFP cooling pump lower suction line was opened to allow the three suction lines to be maintained at a constant temperature. The existing closed isolation valve, 2-RW-2, did not allow the piping to thermally expand at the same rate as an adjacent SFP suction line during the heat up of the SFP. Therefore, valves 2-RW-2 and 2-RW-18 were "LOCKED OPEN" to reduce the local thermal stress levels.

Reason for the Activity

During a review of Bechtel calculation, "Stress Problem 164," it was discovered that the SFP cooling pump nozzle loads were not below the design basis nozzle allowables. In addition, high thermal stresses existed in the suction line piping manifold due to valve 2-RW-2 being closed. The closed valve did not allow all the suction piping to thermally expand at the same rate during the heat up of the SFP. Opening valve 2-RW-2 allowed the piping to expand uniformly and reduced the local stress caused by thermal expansion.

Safety Evaluation Summary

The addition of the braided expansion joints, the piping support modifications, and the opening of valve 2-RW-2 does not affect the ability of the SFP cooling system to maintain the SFP at or below 140°F as required by the Technical Requirements Manual. The new SFP cooling piping and braided expansion joints are fabricated to the original pipe class (HCC), and therefore the original piping design requirements are met. The temporary configuration of the piping and pipe supports during implementation maintained one train of SFP cooling operable at all times while the modifications to the opposite train of the SFP cooling system were implemented. The suction line piping and component design does not increase the probability of a SFP piping rupture.

The installation of the new braided expansion joints does not affect the ability of the SFP cooling system to isolate pipe ruptures, and does not decrease the siphon protection of the SFP. It does not significantly affect the Net Positive Suction Head of the flow rates of the SFP cooling pumps. It does not affect train separation or the ability to isolate the opposite train of the SFP cooling system in the event of a pipe rupture. No margin of safety is reduced by this change.

S2-EV-00-0027
DCN DM2-00-0084-00
FSARCR 00-MP2-24

Setpoint Change for PS-7985A and PS-7985C "A" Emergency Diesel Generator (EDG) AC and DC Starting Air Compressor Pressure Switch

Description

The upper limit setpoint of the EDG AC and DC starting air compressor pressure switches were lowered from 240 to 230 psig. The lower limit setpoint of the EDG AC air compressor switch was changed from 210 to 200 psig. The lower limit setpoint of the EDG DC air compressor was changed from 190 to 180 psig. The affected switch numbers are PS 7985A, "A" EDG AC starting air compressor F10A, and PS 7985C, "A" EDG DC starting air compressor F10C.

Reason for the Activity

This change was made to prevent inadvertent lifting of EDG starting tank relief valves. Due to tolerances for the relief valve setpoint as well as the compressor start/stop setpoints, there could be overlap. When switch and valve inaccuracies and component calibration inaccuracies are considered, there can be interaction between the two, resulting in an undesirable situation. In the past the relief valve had lifted and the starting air for the EDG was bleeding off with one or both the starting air compressors continuously running. There had been numerous condition reports written over the years documenting this undesirable condition.

Safety Evaluation Summary

The EDG air compressors do not directly provide a plant safety function. The compressors' function is to replenish the air in the starting air tank after a EDG start and to maintain the pressure in the tank above a minimum pressure. This is considered a support function. Starting energy for the EDGs is provided by air stored in the starting air tanks. As long as the stored energy remains above a minimum amount, the safety function is not affected. This modification does not change this minimum amount of stored starting air, therefore the design basis remains unchanged. The modification changes the upper pressure limit setpoint of air in the starting tanks. While this lowers the energy available for starting the EDG, testing has proven the remaining quantity of air is more than sufficient for providing three (3) starts from each starting air tank. This modification is safe.

S2-EV-00-0028
DCN DM2-00-0086-00
FSARCR 00-MP2-24

Setpoint Change for PS-7985B and PS-7985D "B" Emergency Diesel Generator (EDG) AC and DC Starting Air Compressor Pressure Switch

Description

The upper limit setpoint of the EDG AC and DC starting air compressor pressure switches were lowered from 240 to 230 psig. The lower limit setpoint of the EDG AC air compressor switch was changed from 210 to 200 psig. The lower limit setpoint of the EDG DC air compressor was changed from 190 to 180 psig. The affected switch numbers are PS 7985B, "B" EDG AC starting air compressor F10B, and PS 7985D, "B" EDG DC starting air compressor F10D.

Reason for the Activity

This change was made to prevent inadvertent lifting of EDG starting tank relief valves. Due to tolerances for the relief valve setpoint as well as the compressor start/stop setpoints, there could be overlap. When switch and valve inaccuracies and component calibration inaccuracies are considered, there can be interaction between the two, resulting in an undesirable situation. In the past the relief valve had lifted and the starting air for the EDG was bleeding off with one or both the starting air compressors continuously running. There had been numerous condition reports written over the years documenting this undesirable condition.

Safety Evaluation Summary

The EDG air compressors do not directly provide a plant safety function. The compressors' function is to replenish the air in the starting air tank after a EDG start and to maintain the pressure in the tank above a minimum pressure. This is considered a support function. Starting energy for the EDGs is provided by air stored in the starting air tanks. As long as the stored energy remains above a minimum amount, the safety function is not affected. This modification does not change this minimum amount of stored starting air, therefore the design basis remains unchanged. The modification changes the upper pressure limit setpoint of air in the starting tanks. While this lowers the energy available for starting the EDG, testing has proven the remaining quantity of air is more than sufficient for providing three (3) starts from each starting air tank. This modification is safe.

E2-EV-00-0033
FSARCR 00-MP2-23
DCN DM2-00-0303-00
DCR M2-00006

Reload Design for Millstone Unit No. 2 Cycle 14

Description

This change reconfigured the Millstone Unit No. 2 Core for Cycle 14 operation.

Reason for the Activity

The reconfiguration of the core (i.e., load the Cycle 14 core) was necessary in order to load fresh fuel into the core to allow Cycle 14 operation and the continued production of electricity by Millstone Unit No. 2. A new core and "cycle" of operation was necessary because the Cycle 13 core had reached a burnup value such that there was insufficient reactivity to operate efficiently. The related Final Safety Analysis Report (FSAR) changes to Chapters 3 and 14 were necessary so that the licensing document and the plant design was consistent.

For Cycle 13 operation, the Thermal Margin/Low Pressure (TM/LP) pre-trip alarm had been problematic. In order to correct this problem, a relaxation of the applied uncertainty in the TM/LP trip setpoint was necessary. DCN DM2-00-0303-00 documented this relaxation and detailed the results of the re-analyzed events being acceptable.

Safety Evaluation Summary

DCN DM2-00-0303-00 documented the relaxation of the applied uncertainty in the TM/LP trip setpoint and pretrip alarm setting with new analyses by Siemens Power Corporation using NRC approved methodology. The Technical Specification bases change was made to eliminate the specific numeric values stated for measurement uncertainties and other allowances contained in the safety margin to the trip setpoints. The numeric values were subject to change on a cycle by cycle basis. The change was made to facilitate future accident analysis revisions, and was consistent with the improved Standard Technical Specifications for Combustion Engineering Plants provided in NUREG-1432, Revision 1. The TS P_{VAR} equation was not impacted. Therefore, the Technical Specification bases changes and the relaxation of the TM/LP trip setpoint and pretrip alarm setting were safe.

Based on this Integrated Safety Evaluation and the documentation provided in the References - Section 5 of the evaluation, the operation of the Cycle 14 core was determined to be acceptable and safe.

S2-EV-00-0034
Calculation 00-ENG-02975-C2

Containment Liner Minimum Wall Thickness Evaluation

Description

The minimum thickness of the containment liner was determined in engineering calculation 00-ENG-02975-C2, and this allowable thickness was less than the nominal stated value of ¼ inch from Section 5.2.2 of the Millstone Unit No. 2 Safety Analysis Report (SAR).

Reason for the Activity

10 CFR 50.55a required that the examinations of ASME Section XI, Subsection IWE, be performed for Millstone Unit No. 2 during refueling outage 2R13. The scope of the exams consisted of a general visual examination of the containment liner, penetrations, bolted assemblies, and the moisture barrier at the basemat to the liner junction at elevation -22'.

The acceptance criteria of IWE states that any wall loss greater than 10% of the nominal plate thickness requires further engineering evaluation. The engineering evaluation in calculation 00-ENG-02975-C2 demonstrated that the requirements of the Design Specification and the SAR were met with the allowable liner thickness values given in this calculation, which are less than the nominal value of ¼ inch as discussed in Section 5.2.2 of the SAR.

Safety Evaluation Summary

The minimum liner thickness was evaluated in this calculation and determined that the design basis of the liner is maintained in this condition. The liner acts to provide the primary leakage barrier for the containment, and does not provide any load carrying capability to the containment structure. The malfunctions and consequences of a malfunction associated with this activity have not been increased. The liner serves to mitigate the consequences of the accidents listed in Chapter 14 of the SAR, and the ability of the liner to perform the intended function in mitigating these accidents has not been reduced by the establishment of a minimum liner thickness. There are no new malfunctions or accidents that need to be postulated as a result of this activity.

S2-EV-00-0050
DCN DM2-00-0285-00

Replacement of "A" Service Water Inspection Tee Cover/Vent Assembly with a Blind Flange

Description

The 24" supply piping from the Service Water pumps includes a tee with a cover which can be removed during refueling outages for internal piping inspections. The existing cover had a ¾" vent valve assembly attached to the top. This tee is located in the Service Water supply piping which runs through the Turbine Building Pipe Tunnel en route to the safety related heat exchanger located in the Auxiliary Building. The vent valve was normally closed and was not operated by any plant procedure. This Design Change Notice replaced this tee cover with an ANSI rated 150# Plasticor coated carbon steel flange. Because the vent valve was not required for operation of the Service Water System, it was not installed on the replacement blank flange.

Reason for the Activity

During refueling outage 2RF13, the vent valve connection to the tee cover was damaged. The scheduled rework of the existing piping had the potential to delay Train swap and impact the refueling outage schedule. As a result, this minor piping modification was implemented to allow the return to operation of the Service Water "A" train to support the refueling outage schedule.

Safety Evaluation Summary

The removal of a manual vent valve from the Service Water supply line does not affect the operation or performance of the Ultimate Heat Sink. The change implemented by this design change does not affect Service Water flowrates to any heat exchangers cooled by the system. The ability of the safety related heat exchangers to remove the required heat loads following a design basis accident is unaffected.

S2-EV-00-0052
DCN DM2-05-0586-99

Design Change for Main Generator Temperature Monitoring Recorder TR-4510

Description

This Design Change Notice modified the nameplate data for recorder TR-4510, located in the Millstone Unit No. 2 Control Room on control panel C07.

Reason for the Activity

A previous design change removed main generator temperature input signals from recorder TR-4510 in order to install the new Foxboro I/A Series System. The new Foxboro system provides a replacement method of monitoring/displaying the main generator temperature indications from a number of resistance temperature devices and thermocouples. This modified the associated nameplate for the recorder by removing the panel nomenclature related to eight (8) input points that were removed.

Safety Evaluation Summary

The nameplate modification and installation was performed using an approved plant specification and work order. The change does not adversely affect the health and safety of the public and does not affect the reliability of any safety system, structure or component to perform its intended function. The activity was evaluated to ensure existing standards for Control Room design were met by the modification of the nameplate. The change does not reduce the margin of safety as defined in the basis for any technical specification or degrade any fission product barriers.

S2-EV-00-0057
DCN DM2-04-0346-97

Gag Relief Valve 2-RB-330

Description

This modification temporarily gagged the thermal relief valve 2-RB-330 from lifting due to Reactor Building Component Cooling Water (RBCCW) system pressure surges during pump start. The thermal relief valve 2-RB-330 located on the RBCCW system header, upstream of Containment Air Recirculation (CAR) Unit X-35C serves as the component overpressure protection when isolated. The modification temporarily gagged the thermal relief valve until a permanent fix was put in place. The necessary procedures were revised to maintain at least one isolation valve, 2-RB-28.1C or 2-RB-28.2C/2-RB-28.3C on X-35C open to provide a flow path to either surge tank or the relief valve (2-RB-329) on CAR cooler (X-35A) to prevent over-pressurization in the event of an external heat source.

Reason for the Activity

During the Loss of Normal Power test and a subsequent retest, the thermal relief valve, 2-RB-330, lifted due to a spike in the RBCCW system pressure in excess of the set pressure of the relief valve. The relief valve failed to reseat until the RBCCW system pressure decreased more than approximately 25%. Failing to reseat could cause the associated RBCCW header to be inoperable during a Loss of Coolant Accident. In addition, leakage of RBCCW could cause boron dilution in the containment sump. Therefore, by maintaining either the outlet or inlet isolation valves open, a flow path is established to prevent over-pressurization due to expanding fluids from containment heat source. This valve is gagged closed.

Safety Evaluation Summary

The relief valve is located on the RBCCW header just upstream of the CAR Cooler Unit X-35C to provide thermal overpressure protection. The CAR cooler has isolation valves which are normally open and they fail open upon a loss of instrument air or normal power. The outlet isolation valve 2-RB-28.3C receives a signal to open on Safety Injection Actuation Signal. Procedurally maintaining either one of these valves open provides relief due to any pressure surges to the RBCCW surge tank.

S2-EV-00-0061
OP2335A, Rev. 16, Chg. 5
OP2338D, Rev. 0, Chg. 3
Temp. Mod. 2-00-013, Rev. 1

Clean Radwaste Sea Water Removal

Description

Water containing high chlorides (~500 ppm) from salt water intrusion in the "B" Coolant Waste Monitor Tank (CWMT-B), e.g. T15B, was processed through a temporary demineralizer to reduce and remove chlorides and radioactivity prior to, and in addition to, utilizing the installed secondary demineralizer (T23A) before discharging the water to the environment.

Water containing high chlorides (~10,000 ppm) from salt water intrusion in the "A" Coolant Waste Receiver Tank (CWRT-A), e.g. T14A, was transferred to 5,000 gallon tanker trucks for shipment to an off-site waste processor in lieu of utilizing the installed primary demineralizers (T22A and/or T22B). The transfer procedure required the bypass of the low level pump cutout to minimize the amount of water that remained in the CWRT-A. As an alternative, this water could be pumped to the cask wash pit for storage prior to being transferred to the 5,000 gallon tanker truck.

Reason for the Activity

The activity delineated by Temp. Mod. 2-00-013 was a result of saltwater intrusion into the Clean Liquid Radwaste System. The Temp. Mod. permitted alternate treatment, storage and/or off-site processing of the high chloride liquid.

Safety Evaluation Summary

Neither the probability of occurrence, nor the consequences of a malfunction of equipment important to safety was increased since the original assumptions of the Final Safety Analysis Report (FSAR) malfunction remained unchanged. Similarly, a different type of malfunction was not created because the original FSAR malfunction remained bounding and the off-site shipment of the high chloride liquid contents was governed by shipping regulations. Since the change did not introduce new source terms or failure modes, the change did not affect previous accidents evaluated in the FSAR. The change did not create any new accidents. The margin of safety was similarly unchanged since the Technical Specifications requirements for Radioactive Effluent discharges were not changed.

S2-EV-00-0062
DCN DM2-00-0352-00

Gag Relief Valves 2-RB-308 and 2-RB-309

Description

The thermal relief valves 2-RB-308 and 2-RB-309 are located on the Reactor Building Closed Cooling Water (RBCCW) system header, upstream of High Pressure Safety Injection (HPSI) pump seal cooler P41A (X-216B) and Low Pressure Safety Injection (LPSI) pump seal cooler P42A (X-215A) respectively. These thermal relief valves serve as the component overpressure protection when the seal coolers are isolated. This modification temporarily gagged the thermal relief valves until a permanent fix is in place. The necessary procedures were revised to maintain at least one isolation valve on both seal coolers open to provide a flow path to either surge tank or the relief valve (2-RB-310) on Containment Sump (CS) Pump 43A seal cooler (X-214A) to prevent overpressurization in the event of an external heat source.

Reason for the Activity

During the performance of testing, thermal relief valves 2-RB-308 and 309 lifted due to a spike in the RBCCW system pressure in excess of the set pressure of the relief valve. The relief valves failed to reseat, leaking approximately one (1) gpm combined, even after the RBCCW system pressure reached normal system pressure. Failing to reseat could cause the associated RBCCW header to be inoperable during a Loss of Coolant Accident (LOCA) due to loss of RBCCW inventory. Therefore, by maintaining either the outlet or inlet isolation valves open, a flow path can be established to prevent overpressurization due to expanding fluids from Engineered Safety Features room heat up during sump recirculation following a LOCA.

The Millstone Unit No. 2 Final Safety Analysis Report (FSAR) section 9.4.2.1 states that components and heat exchangers served by the RBCCW system which can be isolated are equipped with self actuated, spring loaded relief valves for overpressure protection. Since the thermal relief valves are temporarily gagged until such time as a permanent fix is in place, an FSAR change request was not required.

Safety Evaluation Summary

The modification gagged the thermal relief valves closed to prevent the valves from lifting due to unexpected pressure spikes in the RBCCW system pressure during pump restarts. This change does not increase the probability of occurrence of an accident or increase the consequences of an accident which would result in radiological dose to the public. Gagging the thermal relief valves will not cause a different type of malfunction to the seal coolers or the RBCCW system header.

S2-EV-00-0063
DCN DM2-00-0355-00

Gag Reactor Building Closed Cooling Water (RBCCW) Relief Valves

Description

Thirty-seven thermal relief valves (2-RB-303A/B/C, 2-RB-304,- 305, 306, 307, 2-RB-310, 311, 312, 313, 314, 315, 316, 2-RB-318, 2-RB-320, 321, 322, 2-RB-324, 325, 2-RB-327, 328, 329, 331, 332, 333, 334, 335, 336, 337, 338, 339, 340, 341, 342, 343, 344) are located on the "A" and "B" RBCCW system headers, and serve as overpressure protection when their associated components are isolated. The modification gagged these thermal relief valves for the duration of Cycle 14, or until a permanent fix is in place.

Reason for the Activity

During the performance of testing, certain thermal relief valves lifted due to a spike in the RBCCW system pressure in excess of the set pressure of the relief valve. The setpoint for valves 2-RB-304 through 314 is 165 psig, and 150 psig for the remainder. Three relief valves failed to reseat, leaking even after the RBCCW system pressure reached normal system pressure. Although no other relief valves failed to reseat in several tests, other relief valves did lift. This presented the possibility, although very small, of a valve failing to reseat. Failing to reseat could cause the associated RBCCW header to be inoperable during a Loss of Coolant Accident due to loss of RBCCW inventory.

The Millstone Unit No. 2 Final Safety Analysis Report (FSAR) Section 9.4.2.1 states that components and heat exchangers served by the RBCCW system, which can be isolated, are equipped with self actuated, spring loaded relief valves for overpressure protection. Since the thermal relief valves are temporarily gagged until such time as a permanent fix is in place, an FSAR change request was not required.

Procedural controls are in place to ensure the relief valves are ungagged prior to completely isolating a safety-related component, to prevent thermal overpressurization.

Safety Evaluation Summary

The modification gagged identified thermal relief valves to prevent them from lifting due to RBCCW system pressure surges during pump restarts, with the exception of valves 2-RB-326/413/414/415/416. The relief valves which are providing thermal overpressure protection, are located throughout the RBCCW system, encompassing both "A" and "B" headers. Each cooler has isolation valves which are normally open. Procedurally maintaining (by locking) these valves open provide relief due to any pressure surges to the RBCCW surge tank.

S2-EV-00-0065
Temp. Mod. M2-00-013, Rev. 0
Clean Liquid Radwaste Monitor Tank Recirculation

Description

This temporary modification installed a jumper to defeat the interlock between valves 2-LRR-27.1A and 2-LRR-29.1A for the "A" coolant waste monitor tank (T-15A) or the interlock between valves 2-LRR-27.1B and 2-LRR-29.1B for the "B" coolant waste monitor tank. This enabled both inlet and outlet valves to be opened simultaneously allowing a selected monitor tank to be recirculated without cross-contaminating the parallel monitor tank or receiver tanks. Inadvertent radwaste discharges to the environment were precluded while the jumpers were installed by applying red tags to the discharge flow path. The temporary modification has been removed.

Reason for the Activity

The activity delineated by Temp. Mod. M2-00-013 was a result of saltwater intrusion into the Clean Liquid Radwaste System. The temporary modification permitted treatment of clean waste liquid without cross-contaminating other portions of the system.

Safety Evaluation Summary

Neither the probability of occurrence, nor the consequences of a malfunction of equipment important to safety was increased since the original design conditions (pressure, coolant waste tank liquid radioactivity, and volume) of the plant remained unchanged and all existing components were used. Similarly, a different type of malfunction was not created because no new failure modes were introduced. Since the change did not introduce new source terms or failure modes, the change did not affect previous accidents evaluated in the Final Safety Analysis Report or create any new accidents. The margin of safety was similarly unchanged. Since discharges were prevented during the temporary evolutions, the Technical Specifications requirements for radioactive effluent discharges were therefore not impacted.

S2-EV-00-0066
OP2335A, Rev. 16, Chg. 7
OP2338A, Rev. 14
Temp. Mod. 2-00-0014

Temporary Filtration System for Clean Waste Monitor Tank Discharge

Description

A temporary filtration system was connected to valves 2-LRR-21 and 2-LRR-22 in the Recovered Boric Acid Storage Tank Room on elevation (-)5'-0" of the Auxiliary Building. The temporary filtration system was used in conjunction with installed plant equipment to process Coolant Waste liquid radwaste before discharge to the Circulation Water System. Two skid mounted trains, each consisting of three (3) filter units valved in series, were installed. Flexible hoses were used to make the temporary connections to the plant equipment and interconnections between filter units. This temporary modification has been removed.

Reason for the Activity

An adequate supply of filter cartridges for the Coolant Waste Final Filter L-15 were unavailable. The radioactivity levels in the Coolant Waste Monitor Tanks T-15A and T-15B were "high." The temporary modification permitted additional filtration of the "high activity" liquid.

Safety Evaluation Summary

The temporary change did not increase the probability of occurrence nor the consequences of a malfunction of equipment important to safety since the original assumptions of the Final Safety Analysis Report (FSAR) malfunction remained unchanged. Similarly, it did not create a different type of malfunction because the original FSAR malfunction remained bounding. The change did not introduce new source terms or failure modes, therefore it did not affect previous accidents evaluated in the FSAR. The "high activity" liquid did not create a different type of accident because there was actually less radioactive source material during the operating conditions at the time and no new failure modes.

S2-EV-00-0069
Temp. Mod. 2-00-015

Monitor "A" Motor Generator Output Voltage, Exciter Field Current, and Control Relay "5K" Operation

Description

This modification connected a chart recorder/isolation module to provide a means of recording the identified voltage and current parameters of the "A" Motor Generator during operation.

Reason for the Activity

The voltage traces taken on the Control Rod Drive Mechanisms on selected rods showed that the voltage was of concern. The traces taken by this temporary modification documented the voltage output at the generators at power with and without rod movement. Monitoring the exciter field current and associated operation of control relay "5K" assisted Engineering to verify that the recent trips of the "A" Motor Generator were due to spurious operation of the exciter current ammeter relay and not the result of an actual spike in the exciter current.

Safety Evaluation Summary

The test meter was added to a circuit with an existing (non-recording) test meter of lesser quality and through the use of non-intrusive current transformers. The installation of the test meter was done by qualified, trained technicians using approved procedures. The failure of the test meter could have caused the M-G Set(s) to trip resulting in the control rods dropping into their fail safe position. The test meter had no impact on the fuel cladding, reactor vessel wall, or containment barrier, thus the margin of safety was not impacted.

S2-EV-00-0073
SPROC ENG99-2-11

Withdrawal/Insertion of Boraflex Poison Box

Description

The Millstone Unit No. 2 spent fuel storage racks contain boraflex as a neutron poison. The boraflex is located in poison boxes which are located inside each fuel storage location. The poison boxes are designed to be removable. This SPROC removed a neutron poison box assembly from Millstone Unit No. 2 spent fuel pool location J-5, and replaced it with a new poison box. A special tool designed for poison box removal was used. The tool weighs about 250 pounds. The poison box weighs about 200 pounds. The poison box was removed from the fuel storage cell using the spent fuel pool platform crane and the poison box handling tool, and then transported under water to the cask laydown pit. The storage box was temporarily placed in a temporary poison box holder assembly, which was also installed as part of this procedure. The temporary poison box holder assembly was hung off the 3x3 rack in the cask laydown pit. The poison box was then re-grappled in the cask laydown pit using the platform crane, and lifted to a point where the monorail crane lifted the poison box the rest of the way out of the water. The reverse process was used for insertion of the new poison box.

Reason for the Activity

The old poison box was destructively tested as part of the Boraflex surveillance program. The insertion of the new poison box to replace the removed poison box restores the spent fuel pool to the normal condition. The new poison box is manufactured to the same specifications as the old poison box.

Safety Evaluation Summary

The replacement of a poison box with an equivalent replacement box does not alter the performance characteristics of the spent fuel storage racks. No new malfunctions are created nor are any existing analyzed malfunctions affected by the poison box withdrawal or insertion process. No new accidents are created. The existing analyzed fuel drop accident is still bounding should a drop of the poison box and/or tool occur over the storage racks. The margin to safety, i.e. K-effective < 0.95, is not affected since spent fuel storage location J-5 and surrounding locations are empty of fuel during this process. The weights of the poison box and tooling are far less than the weight of a fuel assembly, and the height of movement above the racks are approximately the same as when fuel is moved. Movements of the old and new poison box in and out of the spent fuel pool, and the insertion/removal of the temporary poison box holder, take place over the cask laydown area, where there is no fuel. A potential poison box/tool drop event (or the even lesser weight of the temporary poison box holder) in the cask laydown area would be bounded by the already analyzed cask drop accident.

S2-EV-00-0083
Temp. Mod. M2-00-016

Freeze Seal 4"-JBD-32 for M2-99068 Installation

Description

This temporary modification installed a freeze seal on line 4"-JBD-32, just off the header piping and isolated HS 227. Line 4"-JBD-32 was cut and a "tee" installed to allow installation of additional hose stations per design change M2-99068. This temporary modification has been removed.

Reason for the Activity

The freeze seal was installed to allow the header line 10"-JBD-28 to remain in service, minimizing the impact to the fire protection system during the implementation of design change M2-99068.

Safety Evaluation Summary

This safety evaluation supplements the safety evaluation for Design Change M2-99068 to address the use of a freeze seal as a boundary for installation of the design change. Installation of the freeze seal did not affect any design basis accidents or their consequences. It did not create any malfunctions or contribute to any new accidents or new malfunctions. The freeze seal served as a maintenance boundary to allow additional hose stations off line 4"-JBD-32 per Design Change M2-99068. The freeze seal could be isolated in a contingency or emergency. Installation of the design change with the freeze seal allowed the header to remain in service and minimized the affect on fire protection.

Docket Nos. 50-336
50-423
B18442

Enclosure 2

Millstone Nuclear Power Station, Unit No. 3

10 CFR 50.59 Report for 2000

Millstone Unit No. 3
Plant Design Changes

Design Change Notice No.	Title	Safety Evaluation No.
DCN DM3-00-0040-99	Replacement Valves for Valves 3SWT-V928 through 3-SWT-V933	S3-EV-99-0111
DCN DM3-00-0184-00	Revise Reactor Coolant System (RCS) Low Flow Reactor Trip Setpoint	S3-EV-00-0047
DCN DM3-00-0258-99	Addition of Sample Connections for Radiation Monitor 3HVQ-49B	S3-EV-99-0073
DCN DM3-00-1046-98	Installation of True Test in Place (TTIP) Hardware to Reactor Coolant System Hydraulic Snubbers (RFO6)	S3-EV-00-0058
DCN DM3-00-1144-97	Relocation of Sodium Hypochlorite (NaOCl) Injection for Service Water Piping (SWP)	S3-EV-97-0303
DCN DM3-01-0264-98	Installation of Polypropylene (Class 133 Material)	S3-EV-99-0113

Design Change Record No.	Title	Safety Evaluation No.
DCR M3-97024	Low Level Waste Drain Tank Filter Assembly	S3-EV-98-0107
DCR M3-97062	Relocation of Sodium Hypochlorite (NaOCl) Injection for Service Water Piping (SWP)	S3-EV-97-0303
DCR M3-98016	Permanent Piping to a Contractor Water Treatment Facility	S3-EV-98-0102
DCR M3-98038	Ericsson Digital, Cordless, Cellular Telephone System DCT1900 Addition to Millstone Unit No. 3	S3-EV-98-0208
DCR M3-98043	Auxiliary Boiler Hydrazine Addition	S3-EV-99-0010
DCR M3-99009	Westinghouse Loose Parts Monitoring (LPM) System Replacement/FSARCR 99-MP3-020	S3-EV-99-0031
DCR M3-99021	Spent Fuel Pool Cooling & Purification Piping Removal	S3-EV-99-0084
DCR M3-99021	Spent Fuel Pool Gates Supplemental Rigging	S3-EV-99-0086
DCR M3-99021	Rack Handling Scenarios	S3-EV-99-0102

Millstone Unit No. 3
Plant Design Changes

FSARCR No.	Title	Safety Evaluation No.
FSARCR 99-MP3-35	Throttle Position of Service Water Control Building Booster Pump Discharge Valves	S3-EV-99-0083
FSARCR 99-MP3-46	Spent Fuel Pool Gates Supplemental Rigging	S3-EV-99-0086
FSARCR 00-MP3-3	Redesign of Existing Severe Line Outage Detector (SLOD) System and Master Supervisory Panel	S2-EV-99-0149
FSARCR 99-MP3-020	Westinghouse Loose Parts Monitoring (LPM) System Replacement/FSARCR 99-MP3-020	S3-EV-99-0031
FSARCR 99-MP3-22	Auxiliary Boiler Hydrazine Addition	S3-EV-99-0010

Minor Modification No.	Title	Safety Evaluation No.
MMOD DCN DM3-00-0258-00	Dimensional Tolerance Range for Reactor Plant Aerated Drain System Sump Pumps	S3-EV-00-0052
MMOD DCN DM3-00-0352-00	Automatic CO2 Pressure Relief Dampers 3FPL-DMPR4 and 5 Replacement with Manual Pressure Relief Dampers	S3-EV-00-0075
MMOD DCN DM3-00-0353-00	Automatic CO2 Pressure Relief Dampers 3FPL-DMPR4 and 5 Replacement with Manual Pressure Relief Dampers	S3-EV-00-0075
MMOD DCN DM3-01-0117-00	Erosion/Corrosion Replacement for 3-SVH-006-031-4	S3-EV-00-0086
MMOD DM3-00-0026-00	Turbine Exhaust Hood High Temperature Trip Logic Change/Installation of Temperature Indicators SST-TI1000 & 1001	S3-EV-00-0007
MMOD DM3-00-0141-00	Modify Turbine Plant Miscellaneous Drain (DTM) Drain Valve Configuration to Reduce High Pressure Drip Piping Vibration Effect	S3-EV-00-0034
MMOD DM3-00-0407-00	Steam Generator Blowdown Flow Transmitters	S3-EV-00-0081
MMOD DM3-00-0840-99	Turbine Exhaust Hood High Temperature Trip Logic Change/Installation of Temperature Indicators SST-TI1000 & 1001	S3-EV-00-0007
MMOD M3-00010	Remove Control of 3CND-MOV23 & 27 Valves	S3-EV-00-0010
MMOD M3-00020	Duress Panel Relocation	S3-EV-00-0029

Millstone Unit No. 3
Plant Design Changes

Minor Modification No.	Title	Safety Evaluation No.
MMOD M3-99028	Air Dryer Skid for 3SAS-C2A/B Portable Air Compressors	S3-EV-99-0079
MMOD M3-99029	Throttle Position of Service Water Control Building Booster Pump Discharge Valves	S3-EV-99-0083

Procedure No.	Title	Safety Evaluation No.
SPROC EN 98-3-06	3HVK*CHL 1A Condenser Thermal Performance Test	S3-EV-00-0067
SPROC EN 99-3-9	3HVK*CHL 1B Condenser Thermal Performance Test	S3-EV-00-0067
SPROC EN 99-3-15	Fan Pressure Test for Cable Spreading Area	S3-EV-00-0077
SPROC ENG 00-3-02	Flow and Leakage Test of Recirculation Spray System (RSS) Cubicle Floor	S3-EV-00-0039
SPROC ENG 00-3-03	Millstone 3 Rerack Installation	S3-EV-00-0066
MP 3709C	Freeze Sealing	S3-EV-00-0062
MP 3709C-001	Freeze Seal Evaluation Sheet	S3-EV-00-0062

Program & Specifications	Title	Safety Evaluation No.
EEQ Spec. SP-M3-EE-0353	Electrical Equipment Qualification Masterlist (EQML) & Component Replacement Schedules (CRSs) with Equipment Qualification Records (EQRs)	S3-EV-99-0124
MP3 EEQ Program	Deletion of the Westinghouse Power Range Detectors for the Master List	S3-EV-00-0061

Temporary Modification No.	Title	Safety Evaluation No.
Temp. Mod. 3-00-010	Temporary Connection of Instrumentation to Monitor Performance of "B" Hydrogen Recombiner	S3-EV-00-0071
Temp. Mod. 3-00-001	Freeze Seal for Repair of 3SWP*V49	S3-EV-00-0022
Temp. Mod. 3-00-002	Temporary Modification to Supplementary Leakage Collection and Release System (SLCRS) Effluent Discharge	S3-EV-00-0040

Millstone Unit No. 3
Plant Design Changes

Temporary Modification No.	Title	Safety Evaluation No.
Temp. Mod. 3-00-004	Discharge of Traps 3DTM-TRP16A/B/D to Drain Rather Than 3CNA-TK2	S3-EV-00-0065
Temp. Mod. 3-00-008	Freeze Seal to Support Valve 3SSR*CTV19D Replacement	S3-EV-00-0062
Temp. Mod. 3-86-118	Cross-tie Between Unit 3 Vendor Supplied Water Treatment System and Unit 2 Vendor Supplied Water Treatment System	S3-EV-98-0038

S3-EV-97-0303
DCN DM3-00-1144-97
DCR M3-97062

Relocation of Sodium Hypochlorite (NaOCl) Injection for Service Water Piping (SWP)

Description

The modifications associated with this change re-routed the NaOCl injection piping to the abandoned pump lubrication water connection. The existing injection piping which connected to the Service Water pump suction bells is isolated. The new injection location uses screen wash and/or domestic water to inject NaOCL.

Revision 2 of this safety evaluation redefined the ASME class break requirement from a pair of Monel positive action check valves to a single positive action Titanium check valve. A single Titanium manual valve was also provided downstream of the check valve at each injection point for maintenance isolation.

Reason for the Activity

This change was made to resolve the existing Service Water Strainer configuration problem which resulted in an unmonitored sodium hypochlorite discharge. The SWP system is provided with NaOCl for control of biological fouling. The existing NaOCl system continuously injected NaOCl near the suction bell of each of the four Service Water pumps. The blowdown of the strainer discharges directly into Niantic Bay. The strainer blowdown, which includes NaOCl, was identified as an unmonitored chemical discharge point and the NaOCl concentration in the Service Water blowdown exceeded National Pollutant Discharge Elimination System limits.

Safety Evaluation Summary

This change does not cause an increase to the risk to public health or safety. Raising the Domestic Water System pressure to 75 psig is within the design of the Chlorine System (Hypochlorite) and SWP systems. Crediting one check valve at the Chlorine System (Hypochlorite)/SWP interface is acceptable per technical procedures and meets the Final Safety Analysis Report Table 1.8-1 licensing requirements for degree of compliance with NRC Regulatory Guide 1.26. Existing SWP piping, which was isolated from use, is not detrimental to the SWP and eliminates a potential Service Water break path.

This change does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report. It does not increase the consequences of a malfunction of equipment important to safety previously evaluated and does not create the possibility of a malfunction of a different type than any previously evaluated.

S3-EV-98-0038
Temp. Mod. 3-86-118

Cross-tie Between Unit 3 Vendor Supplied Water Treatment System and Unit 2 Vendor Supplied Water Treatment System

Description

This modification provided a cross-tie to supply makeup water to Millstone Unit No. 2 or Millstone Unit No. 3 and a cross-tie to transfer makeup water between Millstone Unit Nos. 2 and 3. The modification provided redundant means of water treatment for Millstone Unit Nos. 1, 2, and 3 via the Condensate Polishing Facility.

Reason for the Activity

Installation of the water treatment system cross-tie between the Millstone Unit No. 3 vendor supplied water treatment system and the Millstone Unit No. 2 vendor supplied water treatment system provided an additional source of demineralized water to support the increased need for water during plant startups.

Safety Evaluation Summary

The installation, operation and/or potential failure of this modification had no impact on safety related systems or components. The installation does not result in the plant being operated in an unsafe condition, decrease the assumed available safety margins, nor adversely impact the consequences of an accident. It did not increase the probability of the failure of mitigating equipment nor introduces of any new accidents or equipment malfunctions.

S3-EV-98-0102
DCR M3-98016

Permanent Piping to a Contractor Water Treatment Facility

Description

This modification provided permanent piping to the Millstone Unit No. 3 contractor water treatment facility (CWTF). Unused piping associated with the Water Treatment System (WTS) was reconfigured to provide a permanent flow path to and from the CWTF. The discharge from the Makeup Demineralizer Supply Pumps, 3WTS-P15A and 3WTS-P15B, was routed through the unused recirculation line from the demineralizers to provide a flow path from the ultra-filtration skid to the CWTF. The original discharge piping from the pumps was then used to provide a flow path from the CWTF to the Millstone Unit Nos. 1 and 2 cross-tie piping.

Reason for the Activity

The Millstone Unit No. 3 WTS demineralizers had been unused since 1986. A cost benefit analysis performed at that time determined that using the Unit 3 WTS demineralizers was more costly than using a contractor to provide demineralized water. This activity made changes to the facility in order to maintain the performance of these systems without utilizing temporary modifications. It allowed the temporary modifications to be removed and closed.

Safety Evaluation Summary

The WTS and the CWTF have no safety-related functions. They are not contributors to, nor required to mitigate the consequences of any accidents. The activity does not introduce any new conditions or malfunctions that could increase the probability or consequence of existing malfunctions which have been analyzed. The piping and components added by this change meet the same standards and specifications of other components installed in the WTS and CWTF system and are therefore no more likely to fail.

S3-EV-98-0107
DCR M3-97024

Low Level Waste Drain Tank Filter Assembly

Description

This modification installed permanent in-line filters, one for each tank 3LWS-TK4A & 3LWS-TK4B, to replace the current slip stream filtration system employed by a temporary jumper. The new filtration system consists of in-line filter housings located on discharge lines downstream of pumps 3LWS-P7A & 3LWS-P7B. The new filtration systems have the ability to re-circulate flow back to their respective tanks when filtration is desired or bypassed when not desired. This enables the entire contents within the tanks to be filtered with less cycling. Therefore, pump (3LWS-P7A or 3LWS-P7B) seal wear is reduced. The filter housings have pressure differential indicating switches for detection of filter loading and a bypass path for maintenance and operation purposes. The new pressure differential indicating switches initiate an alarm at the Liquid Waste System (LWS) Panel and the Main Control Board. The modifications were previously completed on Train A. Train B of the modification, 3LWS-P7B, removal of filter elements from 3LWS-FLT2A/B and removal of 3LWS-PDIS69 have been completed.

Reason for the Activity

The existing system design required the contents of the Low Level Waste Drain Tanks to be recycled through their respective pump, 3LWS-P7A or 3LWS-PP7B through a temporary filter. The implementation of the design change enables the entire contents within the Low Level Waste Drain Tanks (3LWS-TK 4A & 3LWS-TK 4B) to be filtered so the total suspended solid (TSS) levels within the tanks are below the required National Pollutant Discharge Elimination System permit levels.

Safety Evaluation Summary

There were no safety related equipment interfaces with the installation, nor was there any safety related equipment introduced. The low level liquid radwaste system contains no potentially explosive mixtures. The addition of the in-line filtration system into the existing Low Level Radioactive Waste System did not increase the existing quantity of radio-nuclides present within the system. The modification did not change any monitoring locations, measurements of effluent volume, rates of release for specific radio-nuclides previously established within the Radiological Effluents Monitoring/Offsite Dose Calculation Monitoring. The materiel components either meet or exceed the quality control, design, procurement and fabrication requirements of the originally installed system components. The new piping and filter housings were designed and installed to fulfill the requirements of Regulatory Guide 1.143.

S3-EV-98-0208
DCR M3-98038

Ericsson Digital, Cordless, Cellular Telephone System DCT1900 Addition to Millstone Unit No. 3

Description

This change upgraded the Millstone Unit No. 3 telephone system. The change added an Ericsson Inc. Freeser® DCT 1900 Business Personal Communications System. The Freeser® System is a digital, wireless telephone system that was connected directly into the Millstone facility's existing Private Branch Exchange via a radio exchange. Freeser® base stations were installed in various locations to provide wireless coverage for Millstone Unit No. 3. Additional telephone cables were required to facilitate this telephone system upgrade. The Ericsson DCT 1900 telephone system was provided 4-hour battery backed, non-safety-related power via an uninterruptible power supply (UPS).

Reason for the Activity

This change provided readily accessible telephone communications for personnel authorized to carry an Ericsson DCT 1900 handset or where a handset is accessible. Several areas in the Millstone Unit No. 3 power block lack telephones. An injured person, or a person working alone and in a low traffic area can not summon assistance. Readily accessible communications will improve response to urgent situations. The UPS addition provides 4-hours back-up power to ensure continued telephone system operation upon a loss of power.

Safety Evaluation Summary

The operation of the new 1900 Mhz telephone system is consistent with the operation of the 800 MHz Carrier Frequency Trunked Radio System at Millstone Unit No. 3. Administrative procedural controls are similar to existing administrative procedural controls in effect for the 800 MHz radio system. These administrative procedural controls include applicable handset restrictions.

Site specific electromagnetic compatibility testing confirmed that the DCT1900 emission levels are 70% below the theoretical EPRI calculated field strengths and are consistent with the empirical IEEE calculated field strengths.

The Ericsson Inc. DCR1900 telephone system can not cause the failure of equipment credited for accident mitigation. The telephone system is non-safety related and non-seismic qualified. The equipment and components are located so as not to affect equipment important to safety. Administrative controls preclude the telephone system from initiating the malfunction of equipment.

S3-EV-99-0010
FSAR 99-MP3-22
DCR M3-98043

Auxiliary Boiler Hydrazine Addition

Description

Batch hydrazine addition process equipment was replaced with continuous hydrazine addition equipment. This change installed Chemical Feed Pump 3ABF-P4 so hydrazine is injected on a continuous basis. The existing Chemical Feed Tanks and associated piping were removed in order to provide room for the new equipment. The Chemical Feed Pump was skid mounted with a new Chemical Feed Tank, 3ABF-TK2. Since the volume of hydrazine being injected is small, ¼ inch tubing was used to inject the hydrazine into the feedwater lines downstream of the feedwater control valves 3ABF-FV39A & B. Existing test connections were used as injection points. The Hydrazine Analyzer, 3ABF-AE58, uses the existing sample lines to provide boiler water to the analyzer.

Reason for the Activity

Implementation of this change enables hydrazine to be continuously added to the auxiliary boilers providing a constant concentration of hydrazine. The Hydrazine Analyzer eliminates the need for grab samples, except to verify monitor calibration.

Safety Evaluation Summary

There is no safety related equipment or functional interface associated with the installation, nor is there any safety-related equipment or function introduced. The addition of a chemical feed pump and on-line hydrazine analyzer does not impact the operating parameters of the auxiliary boilers. The materiel components either meet or exceed the quality control, design, procurement, and fabrication requirements of the original design. All new tubing was installed and tested to ensure that the tubing maintains system pressure.

S3-EV-99-0031
FSARCR 99-MP3-020
DCR M3-99009

Westinghouse Loose Parts Monitoring (LPM) System Replacement/FSARCR 99-MP3-020

Description

This change removed the Digital Metal Impact Monitoring System (DMIMS) electronics installed in panel 3CES-PNLLPM in the instrument rack room and installed the DMIMS-DX electronics in its place. The DMIMS electronics were installed as a loaner since the manufacturer, Westinghouse, could not deliver the DMIMS-DX hardware prior to the end of the RFO6 refueling outage. The installation of these new electronics did not require a complete system calibration since the frequency sensitive components (accelerometers, charge preamplifiers, and accelerometer isolation kits) were not replaced by this revision. The baseline data and setpoints obtained earlier are valid for the new electronics for the same reason. Final Safety Analysis Report Change Request FSARCR-99-MP3-020, Table 1.8-1 and Section 4.4.6.4 were incorporated.

Reason for the Activity

The Westinghouse Loose Parts Monitoring System meets the requirements of Regulatory Guide 1.133, Revision 1, dated May 1981, and is Y2K compliant The replacement also addressed a commitment made to the NRC in LER 98-042. The commitment stated, "The LPM System will be restored to OPERABLE status, no later than 90 days after startup from RFO6, following completion of system testing at 100 percent power." The installation of the Westinghouse Loose Parts Monitoring System made the system OPERABLE before the end of RFO6 outage.

Safety Evaluation Summary

The project replaced the method of mounting the accelerometers to the steam generators, reactor vessel lifting lug, and reactor vessel lower. New holes were drilled and tapped along with the enlargement of existing holes for the steam generator and reactor vessel upper channels. An evaluation of drilling holes in the steam generator and reactor vessel head lifting lugs was performed by Westinghouse and concluded the drilling and tapping met the ASME code requirements. The evaluation also verified that the seismic support assembly for the reactor vessel lifting rig assembly and the Control Rod Drive Mechanism seismic support assembly remained structurally adequate. The modification installed new accelerometers and new amplifiers closer to the sensor location. The Instrument Rack Room hardware was replaced with a Westinghouse design that has a patented program which differentiates metal to metal impacts from background noise, thus reducing false alarms. The calibration of the system and its response to known weights at a known distance and a known force is demonstrated and recorded. A baseline report documents the system response to these weights and formalizes the bases for the setpoints to detect a loose part in accordance with Regulatory Guide 1.133. These items improve the system's ability to detect a loose part in the Reactor Coolant System or secondary side of the steam generators. Therefore, this change has no impact on the margin of safety.

S3-EV-99-0073
DM3-00-0258-99

Addition of Sample Connections for Radiation Monitor 3HVQ-49B

Description

This change modified the Engineered Safety Features building ventilation exhaust radiation monitor, HVQ-49. The existing sample connections, which were used to collect Technical Specification (TS) required gas, particulate, and iodine samples when the radiation monitor is inoperable, were removed and plugs were installed in their place. New sample connections were added. A tee fitting was installed in both the sample inlet and sample return lines. The branch line from each tee runs down the wall to a valve. The sample lines are properly supported at the valve. The TS required samples are collected at these new sample connections. These sample connections are outside the skid isolation valves such that the skid isolation valves can be shut to allow maintenance to be performed on the skid without affecting collection of the temporary sample. Sample connection is accomplished using Parker tubing fittings as is done for radiation monitor HVR*10B rather than the existing quick disconnect fittings. The replacement sample connections are functionally equivalent to the original connections. The original fixed filter particulate and iodine sample collector was replaced with an improved, more reliable collector. The replacement collector uses flexible tubing and an improved seal. The replacement collector is functionally equivalent to the original collector.

Reason for the Activity

This change was necessary to provide isolation between the sample skid and the temporary sample such that work can be performed on the sample skid without preventing collection of the TS required temporary sample. This change deleted the non-standard quick disconnect fittings in favor of the Parker tubing fittings which are used on HVR*10B, and reduced the sample temperature at the temporary sampling connections by moving the temporary sample connections away from the discharge of the skid sample pump. High temperature was the subject of a Condition Report which stated that the elevated sample temperature was responsible for frequent failure of the former sample fitting. This change replaced the skid mounted iodine and particulate sample collector with an improved version which is easier to operate and features improved seal capability.

Safety Evaluation Summary

The change provides improved, functionally equivalent hardware which simplifies collection of the TS required temporary sample when the radiation monitor is out of service. Additionally, the change provides improved, functionally equivalent hardware to collect the normal particulate and iodine samples required by TS. The new valves and tubing are consistent with Specification SP-EE-212. The new normal particulate and iodine collector is designed for QA Cat 1 service. The affected equipment is isolated from safety significant systems. The new temporary sample connections will be identical to those already installed for HVR*10B, reducing the chance of misoperation. The replacement particulate and iodine sample collector is designed similar to collectors on qualified skids by the same vendor and made of the same materials.

S3-EV-99-0079
MMOD M3-99028

Air Dryer Skid for 3SAS-C2A/B Portable Air Compressors

Description

Three pre-staged air dryer skids were added for use with the 3DAS*P15A/B portable air compressors (3SAS-C2A/B) to provide high assurance that at least one air dryer skid is available for use. 3DAS*P15A/B are safety related air driven sump pumps which remove ground water collected from Recirculation Spray System Cubicle Sump, 3DAS*SUMP7A/B. The 3DAS*P15A/B water suction inlet debris screen was removed to address calcium carbonate buildup which may block the debris screen and threaten pump operation during the required 1 year of post-accident operation. The 3DAS-P8A/B water suction inlet debris screens were removed and the 3DAS*V803 and 3DAS*V810 check valve discs were removed and the cover gaskets changed from Teflon to Graphfoil (or equivalent).

Reason for the Activity

The modifications improve the reliability of the system and help eliminate moisture in the air supply line, which has contributed to motor failure.

Safety Evaluation Summary

The skid is located in a habitable post-Loss of Coolant Accident location and accessible for repair or replacement, if required. The compressor/dryer is not required to be operated until greater than 10 hours post accident and is only required to operate intermittently to remove collected groundwater inleakage. Operator response time is available to use the skid. The dryer skid is defined as non-nuclear safety related/non-seismic consistent with the original 3DAS*P15A/B sump pump design for the 3SAS-C2A/B air compressors. Providing Class 1E power through two (2) breakers in series to non-Class 1E dryers is safe. Any failures in these non-safety-related circuits would have no impact on equipment important to safety because they have been electrically isolated. Consistent with the compressor design, redundant and reliable power is provided, which, since it is derived from safety related electrical buses, is available given a loss of offsite power. The original pump manufacturer air motor debris screen is reinstated to help prevent bearing failure due to debris entering the open bearing. This is safe because there is an upstream 60 mesh filter which prevents the accumulation of significant debris in the air motor screen. 3DAS-P8A/B suction debris screen removal is safe because the pump can perform its intended non-safety related function as assumed in the plant design. These pumps perform no support function for credited accident mitigation equipment. 3DAS*V803 and 3DAS*V810 check valves disc removal does not adversely affect the pump qualification, does not create an adverse piping system fluid transient, adversely affect sump fill and pump schedule calculations, or adversely affect the Supplementary Leak Collection and Release System boundary. Check valve cover gasket replacement provides a gasket material which is suitable to the post-accident radiation environment.

S3-EV-99-0083
FSARCR 99-MP3-35
MMOD M3-99029

Throttle Position of Service Water Control Building Booster Pump Discharge Valves

Description

This design change changed the throttle positions of several manual butterfly valves in the discharge of the Service Water Control Building booster pump discharge. As a result, the minimum required service water flow for the Control Building Water chillers was revised.

Reason for the Activity

During thermal performance testing of the “A” Train Control Building chiller, the chiller unit tripped on low Service Water flow. Evaluation of the data concluded that during operation of the chiller with warm Service Water temperatures when there is no chiller recirculation flow, adequate Net Positive Suction Head (NPSH) may not be available for the booster pump when supplied with minimum required Service Water flow. This change was implemented in July 1999 during refueling outage 3R06.

Safety Evaluation Summary

Throttling the pump and chiller discharge valves to increase system resistance will ensure that adequate NPSH is available for the booster pump during all Service Water system alignments. The revised Service Water minimum required flow was calculated to remove the maximum Control Building heat load and maintain the area temperatures below their Technical Specification limits using 75°F Service Water. The valve throttle positions have been incorporated into the Millstone Unit No. 3 Service Water system flow model and all minimum required post accident flowrates are shown to be supplied. Throttling these valves does not affect or introduce any new malfunctions and all Service Water cooled heat exchanger minimum required accident flowrates are exceeded with these system adjustments.

S3-EV-99-0084
DCR M3-99021

Spent Fuel Pool Cooling & Purification Piping Removal

Description

Portions of the safety-related spent fuel pool (SFP) cooling piping and the non safety-related purification piping submerged in the spent fuel pool were permanently removed.

Reason for the Activity

The affected portions of the SFP cooling and purification piping would interfere with the new spent fuel racks that were installed close to the SFP North wall and along its entire length.

Safety Evaluation Summary

The design change was evaluated for its impact on the structural and seismic adequacy of the modified SFP cooling and purification piping configurations and determined to be safe and does not reduce the margin of safety of these systems. The modified piping configurations were shown to not affect the safety function capability of the SFP cooling system nor the structural integrity of the SFP purification system. The affected piping systems, including supports, were evaluated for the design basis loadings and were shown to meet design basis code limits. A postulated drop on the liner during the implementation phase of this plant change would not have resulted in a pool draindown. There is no increase in the probability of occurrence or consequences of an accident or a malfunction of equipment.

S3-EV-99-0086
FSARCR 99-MP3-46
DCR M3-99021

Spent Fuel Pool Gates Supplemental Rigging

Description

Supplemental rigging was added between the safety related spent fuel pool gates and the spent fuel bridge crane to prevent the gates from toppling onto the spent fuel storage racks during a postulated gate drop. The supplemental rigging will not prevent the gate from impacting the fuel racks, but it will prevent the gate from toppling over and landing horizontally on the racks after impact.

Reason for the Activity

Due to the pool geometry, a dropped gate on the current spent fuel rack configuration cannot land horizontally on top of the racks. The effect on the racks is limited to physical deformation of the cells which has been evaluated. The installation of additional spent fuel rack modules in the middle pool area as part of the spent fuel pool rerack introduces the possibility of a dropped gate coming to rest across the top of the racks and thereby impeding cooling water circulation through the cells. The modification will prevent this from happening by maintaining the gate in an approximately vertical orientation following impact.

Safety Evaluation Summary

The plant change to add supplemental rigging to the spent fuel bridge crane to prevent the gates from toppling onto the spent fuel storage racks during a postulated gate drop accident is determined to be safe and does not reduce the margin of safety of these structures and systems. The addition of the supplemental rigging was shown to not affect the safety function capability nor the structural integrity of the spent fuel bridge crane or the spent fuel pool gates. The new resultant stresses on these structures remain well below design allowables. The supplemental rigging and appurtenances will not interfere with fuel handling activities. There is no increase in the probability of occurrence or consequences of an accident or a malfunction of equipment, nor an impact on established margins of safety.

S3-EV-99-0102
DCR M3-99021

Rack Handling Scenarios

Description

Fourteen (14) free-standing spent fuel rack modules, previously designed and fabricated, were installed in the Millstone Unit No. 3 Spent Fuel Pool to augment the existing storage capacity. Existing mobile cranes and/or forklifts were used to offload the spent fuel rack modules upon receipt at the job site and for movement of the racks to and from an interim storage location and to the fuel building. The new fuel receiving crane offloaded each rack. An upending device was used, in conjunction with the new fuel receiving crane and the new fuel handling crane, to position the racks for placement into the spent fuel pool. The racks were then lifted from the truck bay floor to the refueling floor and placed into the pool using the new fuel handling crane, a specially designed lift rig, and an intermediate hoist, to prevent the new fuel handling crane hook from being immersed in the pool water.

Reason for the Activity

It was necessary to consider the possibility of a rack handling malfunction resulting in a load drop because the new fuel receiving crane, the new fuel handling crane, the spreader frame and the intermediate hoist are not single failure proof.

Safety Evaluation Summary

The installation was accomplished without disturbing the existing racks in the pool or their stored fuel assemblies in any way. The activity was reviewed from the perspective of load drops which may occur during the handling and movement of the racks, in terms of their potential interactions with the existing racks and with other safety-related structures, systems and components in the rack travel path and determined to be safe. The handling of the racks in the spent fuel pool during their installation was controlled to ensure that no loads would be transported over irradiated fuel assemblies and, in the unlikely event of a rack drop, there would be no interaction with the existing racks in the pool containing stored fuel. This activity did not involve the use of any unreviewed methodology, nor did it require the relaxation of an established acceptance limit.

S3-EV-99-0111
DCN DM3-00-0040-99

Replacement Valves for Valves 3SWT-V928 through 3-SWT-V933

Description

This change replaced the Screenwash System Fish Spray Header isolation valves.

Reason for the Activity

The valve style and material were changed to improve the system's overall materiel condition.

Safety Evaluation Summary

This change involved replacement of several non-safety related, maintenance (manually operated) isolation valves. The use of the replacement valves is acceptable per the Millstone Unit No. 3 Piping Specification, SP-ME-572. This evaluation was required due to a minor change in the system diagram shown as Final Safety Analysis Report Figure 10.04-04.

S3-EV-99-0113
DCN DM3-01-0264-98

Installation of Polypropylene (Class 133 Material)

Description

This Design Change Notice supplement was written to allow the use of Polypropylene lined piping (piping Class 133) and fittings in lieu of the originally specified saran line piping and fittings (piping Class 132) in the Water Treatment System (WTS).

Reason for the Activity

The WTS piping was modified by DCR M3-98016 which made existing temporary connections to the Contractor Water Treatment System permanent. The system drawings called for Class 132 piping, which is a saran lined steel piping, and was installed throughout the system. Some sections of piping which were modified required the procurement of new fittings (elbows, tee, etc.) and piping. Saran lined piping in accordance with Class 132 of specification SP-ME-572 was not available and polypropylene lined piping, in accordance with Class 133, was procured for this purpose.

Safety Evaluation Summary

The WTS and the Contractor Water Treatment Facility have no safety-related functions. The activity does not introduce any new conditions or malfunctions that could increase the probability or consequence of existing malfunctions which have been analyzed. The installation of polypropylene lined piping and fittings, in lieu of the saran lined piping, does not affect the probability that any Design Basis Accident (DBA) would occur or alter the consequences of any DBA. It does not contribute to any new accidents beyond those already analyzed. Polypropylene is considered completely acceptable for use in domestic and pure water systems, and there is no compromise to the strength of this piping.

S3-EV-99-0124
EEQ Spec. SP-M3-EE-0353

Electrical Equipment Qualification Masterlist (EQML) & Component Replacement Schedules (CRSs) with Equipment Qualification Records (EQRs)

Description

This activity removed flow transmitter 3CCP*FT178A through 178D from the EQML. The EQML is a compilation of safety related equipment that have a required safety function in a harsh environment caused by a Design Basis Accident (DBA). This activity is an Electrical Equipment Qualification Program change only. No other aspects, e.g., seismic qualification or other program attributes, were affected.

Reason for the Activity

A condition report identified that Rosemount flow transmitters, 3CCP*FT178A through 178D, were installed with Litton Veam connectors. Litton Veam connectors, as a result of an NRC Environmental Qualification audit, were considered not qualified for use in containment harsh environments if they were to be used as instrumentation circuits. These transmitter were located in containment. During a DBA, the transmitters would have been exposed to a harsh environment. The corrective action plan was to re-evaluate the function and harsh environment requirements required for operability, per 10 CFR 50.49.

These transmitters were evaluated as not required per 10 CFR 50.49 and it was concluded that the transmitters will be removed from the EQML.

Safety Evaluation Summary

This activity did not involve any physical modifications to hardware to the plant. It does not affect the way the Reactor Plant Closed Cooling Water system functions, nor does it affect the manual or automatic accident mitigating functions of the plant. This change does not degrade the protective boundaries nor does it increase the probability of occurrence of already analyzed accidents or malfunctions.

S3-EV-00-0007
MMOD DM3-00-0026-00
MMOD DM3-00-0840-99

Turbine Exhaust Hood High Temperature Trip Logic Change/Installation of Temperature Indicators SST-TI1000 & 1001

Description

This activity was associated with two design change packages. DM3-00-0026-00 modified the Turbine Exhaust Hood High Temperature Trip Logic from a "one out of three" logic configuration to a "two out of three" logic configuration as well as created a new exhaust hood high temperature priority alarm. DM3-00-0840-99 justified the undocumented installation of two temperature gauges in the turbine plant sampling system. No field work was involved with this second design change package.

Reason for the Activity

Modification to the Turbine Exhaust Hood High Temperature Trip logic from "one out of three" logic configuration to a "two out of three" logic configuration was performed in order to help reduce unnecessary turbine trips as requested by the Plant Trip Reduction Committee.

Temperature gauges 3SST-TI1000 and 3SST-TI1001 were not depicted on the associated Piping and Instrumentation Drawing (P&ID) and therefore the P&ID required update. The P&ID is considered part of the Safety Analysis Report and therefore must be evaluated.

The two activities were evaluated together due to the fact that both activities are associated with the turbine as well as to alleviate the need to create another safety evaluation for the temperature gauges.

Safety Evaluation Summary

The modification to the Turbine Exhaust Hood High Temperature Trip Logic coupled with the creation of a new priority alarm prevents a single switch malfunction from causing a plant trip while maintaining a level of automatic protection. Automatic protection will be maintained with the new 2/3 logic configuration. It also provides Operations with a high exhaust hood temperature alarm which is independent of the temperature switch high temperature alarms. Operations will be capable of manually responding to an event which elevates the temperature of only a single exhaust hood by use of the new priority alarm, thereby ensuring a satisfactory level of equipment protection.

The previous installation of temperature gauges within the turbine plant sampling system provides a means for monitoring specific line sample temperatures. The turbine plant sampling system is not safety related and therefore, installation of these gauges does not impact the operation of any safety system. The new temperature gauges are rated for their intended service and are considered acceptable for their intended use.

S3-EV-00-0010
M3-00010

Remove Automatic Control of 3CND-MOV23 & 27 Valves

Description

This modification removed the automatic control portion of the motor operated valve (MOV) control circuits and leaves the manual control only by operating the associated close/open push buttons. The auto-control relay contacts were disconnected from the MOV control circuits. This modification was performed for the demineralizer 1A through 1H isolation MOVs.

Reason for the Activity

Condensate Polishing Demineralizer G isolation MOVs 3CND-MOV23G and 3CND-MOV37G closed for an unknown reason resulting in exceeding maximum flow through the remaining 6 in-service demineralizers. This change eliminates the potential inadvertent isolation of the demineralizer due to bumped Mode Selector Switches or failed relay contacts. The automatic control feature of the MOVs was never used and is not required per operating procedure OP 3319C.

Safety Evaluation Summary

The automatic control function of the MOVs was unnecessary because it was never used. The current operating procedure does not utilize the automatic control function of the MOVs. These isolation MOVs are manually opened by a pushbutton operation when the demineralizer is to be placed in service and manually closed when removed from service. Therefore, removal of the automatic control function of the MOV has no impact on operation of the Condensate Polishing Demineralizer System. No safety related components or systems are affected by this modification. The consequence of a malfunction due to a bumped mode selector switch or failure of the relay contacts was eliminated.

S3-EV-00-0022
Temp. Mod. 3-00-001

Freeze Seal for Repair of 3SWP*V49

Description

A freeze seal was installed on line 3SWP-010-40-3, "B" Diesel Service Water supply isolation, to allow replacement of 3SWP*V049.

Reason for the Activity

3SWP*V049 leaked by in sufficient capacity that isolation of 3EGS*E1B/E2B could not be accomplished to allow required diesel inspections and maintenance.

Safety Evaluation Summary

Freeze seal installation and any piping repairs were performed within existing Technical Specification (TS) Limiting Condition for Operation (LCOs) for the Diesel Generator and Service Water (SW) system. Design Engineering reviewed the seal installation and determined structural and seismic integrity were maintained. The alternate (Alpha) train Diesel and both SW trains were available during performance of this activity. In the unlikely event of SW piping failure, it was reasonable to expect repairs to be completed within the applicable TS LCO. Contingency plans had equipment and materials available and clear direction on actions to be taken in the event of piping failure.

S3-EV-00-0029
M3-00020

Duress Panel Relocation

Description

This change relocated the Duress Panel, which included the abandonment of the old station, installation and testing of the new station, and associated Plant Security Plan & security procedure changes.

Reason for the Activity

Decommissioning of Millstone Unit No. 1 required the movement of the Duress Panel from Millstone Unit No. 1 to one of the operating units.

Safety Evaluation

This change does not have any adverse impact on any systems, structures or components important to safety. The changes do not affect the function or operation of any safety related equipment or equipment important to safety.

The changes do not increase the probability of occurrence or affect consequences of any previously evaluated accidents or malfunctions, nor does it create a different type of accident or malfunction. Therefore, there is no impact to the health and safety of the public. The change has no impact on the Margin of Safety as defined in the basis of any Technical Specification.

S3-EV-00-0034
MMOD DM3-00-0141-00

Modify Turbine Plant Miscellaneous Drain (DTM) Drain Valve Configuration to Reduce High Pressure Drip Piping Vibration Effect

Description

An alternative drain valve assembly was utilized by removal of secondary drain valves 3DTM-V345, -V347, -V349, -V351 for 3DTM-TRP8A-D and drain valves 3DTM-V367, - V371, -V375 and -V379 for 3DTM-LS51A-D respectively, and provided a threaded cap closure in place of the removed valves.

In addition to valve removals, fillet weld reinforcements to increase the weld leg length were installed at the 1" sockolet branch connection to lines 3DTM-003-189-4, 003-193-4, 003-197-4, 003-201-4, 003-190-4, 003-194-4, 003-198-4, 003-202-4, interface of 3DTM-TRP8A-D and 1/2" drain valve pipe section, and interface of 3DTM-LS51A-D and 1/2" drain valve pipe section.

Reason for the Activity

A technical evaluation recommended modifications to reduce the vibration stress effect on the piping based on vibration measurements taken at various component locations indicated in a condition report.

Safety Evaluation Summary

The reduction in pipe length and removal of the secondary drain valves of drain valve assemblies associated with 3DTM-TRP8A-D and 3DTM-LS51A-D and weld reinforcements does not increase the risk of a malfunction or the probability of an occurrence of an accident for any safety related system or component. The subject DTM piping in the Turbine Building is a non-safety related system. Eliminating the secondary drain valves and providing weld reinforcements to increase fatigue life does not increase the risk to the public.

The DTM piping system will operate within its design limits. Use of a previously acceptable drain valve configuration and application of weld reinforcements do not affect any systems described in the Technical Specifications.

S3-EV-00-0039
SPROC ENG00-3-02, Rev. 0

Flow and Leakage Test of Recirculation Spray System (RSS) Cubicle Floor

Description

This procedure was performed to verify that the modification, described in DCR M3-0004 to modify sumps 3DAS*SUMP7A and 7B in the Engineered Safety Features (ESF) building at Elevation -34'9", would provide a credible means of continuous removal of ground water from the 9" porous concrete layer between the containment mat and the membrane as required. The test determined if there was enough unimpeded flow through the porous concrete layer to achieve the minimum required flow rate to the proposed standpipes.

Reason for the Activity

The Millstone Unit No. 3 Containment Building and ESF Building were originally designed with a seepage diversion system for protecting structures from the effects of groundwater and the containment steel liner from external hydrostatic pressure. However, there is a steady inleakage of groundwater to the porous concrete drainage system. A reliable system is required to remove this ground water. A new sump was installed to collect the ground water and the method for assuring separation of ground water and potentially contaminated system fluids had to be demonstrated. This test demonstrated that the cubicles/sumps remain leak tight from ground water inleakage, and demonstrated the ability to passively transfer the ground water inleakage to the new sump, 3SRW*SUMP6.

Safety Evaluation Summary

This test determines that there is enough unimpeded flow through the porous concrete layer to achieve the minimum required flow rate to the existing standpipes and demonstrates the leak tightness of the cubicles. This activity requires that each of the ground water inlets to both sumps be plugged. This test does not affect the operation of the RSS pumps. The work scope of this test and the test location cannot create the possibility of an accident of a different type than previously evaluated. A review determined that this test would not result in the probability of malfunction of equipment important to safety any different than previously evaluated. There is no increase in the consequences of previously evaluated malfunctions of equipment important to safety or possibility of a different type of malfunction of equipment important to safety than that previously evaluated. There is no change in radiological doses to the public.

S3-EV-00-0040
Temp. Mod. M3-00-002

Temporary Modification to Supplementary Leakage Collection and Release System (SLCRS) Effluent Discharge

Description

A 20 foot long by 24 inch diameter, 18 gauge metal shroud was installed and enclosed the SLCRS effluent release pipe located inside the Millstone Unit No. 1 stack. The open base of the shroud was attached to the Q-deck at elevation 26 feet. The top of the shroud was open to the atmosphere and in effect extended the point of effluent release inside the stack to elevation 46 feet. The new shroud was guy-wired to the composite floor deck for added stability. The shroud was not physically attached to any part of the SLCRS duct work.

Reason for the Activity

This activity removed the hazard associated with the impact of the SLCRS effluent, released during normal or accident conditions, upon the individuals involved with implementing DCN DM2-00-0154-00, "EBFS Pipe Modification to Facilitate Installation of Isokinetic Probe."

Safety Evaluation Summary

The design characteristics associated with this modification did not degrade design specifications for material and construction practices. The modification did not affect the safe operation of Millstone Unit No. 3 as licensed and analyzed. It would not result in any new malfunctions or increase the probability of occurrence or the consequences of an accident or malfunction previously evaluated. No changes to the operating license or Technical Specifications were required. Implementation of this modification did not adversely affect the operation of the SLCRS. The addition of a 20 foot long shroud section measuring 24 inches in diameter or greater had no significant effect on SLCRS pressure loss. It did not affect the safe operation of Millstone Unit No. 2 as licensed and analyzed due to the design considerations that promoted structural integrity of the shroud.

S3-EV-00-0047
DCN DM3-00-0184-00

Revise Reactor Coolant System (RCS) Low Flow Reactor Trip Setpoint

Description

This change revised the setpoint for the RCS Low Flow Reactor Trip from 93% to 90% loop design flow commensurate with the present Technical Specification value. The change was accomplished through rescaling of the instrument loop bistables during a regularly scheduled surveillance activity.

Reason for the Activity

Early in 1988, the setpoint for the RCS Low Flow Reactor Trip was increased to 92% loop design flow. This was due to concerns as to the accuracy with which RCS flow could be determined, and observed problems with the Rosemount transmitters measuring the RCS flow. An additional 1% margin was added later during the one cycle change out of the Millstone Reactor Coolant Pumps (RCPs) with the Seabrook RCPs. This resulted in an additional 3% margin between the trip setting and the value specified in the plant Technical Specifications. Since then, plant surveillances, calibrations and operating experience have shown that we are able to accurately measure this parameter, the transmitter failure problems have been eliminated, and the original RCPs have been reinstalled. Therefore, this trip setpoint was reset to its Technical Specification value to recover operating margin for this parameter.

Safety Evaluation Summary

The modification involved reducing the RCS low flow reactor trip setpoint to 90% loop design flow. The limiting trip point assumed in the accident analysis for this parameter is 85%. After including the associated instrument loop uncertainties there exists an additional margin (owner controlled) of 1.3% flow between the accident analysis value and the proposed 90% trip setpoint. Plant Technical Specifications list the nominal trip setpoint for this parameter as 90% loop design flow. Hence, the change to the trip setpoint is bounded by the plant accident analysis and license.

This Safety Evaluation has determined that the change does not result in any increase in either the probability of occurrence or consequences of a malfunction of equipment or accidents previously evaluated in the Safety Analysis Report, does not result in the possibility of a malfunction or accident of a different type than previously evaluated in the SAR, and does not reduce the margin of safety as defined in the basis for any Technical Specification.

S3-EV-00-0052
MMOD/DCN DM3-00-0258-00

Dimensional Tolerance Range for Reactor Plant Aerated Drain System Sump Pumps

Description

This DCN updated the Vendor Technical Manual for the Reactor Plant Aerated Drain System (DAS) Sump Pumps to include a dimensional tolerance range for new Hylum blades for the air motors of 3DAS*P15 A, B and spare. The new range revised the acceptable Hylum blade length from 2.980-2.990 inches to 2.960-2.970 inches.

Reason for the Activity

This activity incorporated the results of laboratory testing that validated a dimensional tolerance band pertaining to the new Hylum blades. This tolerance band enhanced pump reliability by allowing for blade growth without pump failure, and did not impact pump performance. This information was used to provide guidance to the field for future installation and use in the air motors of 3DAS*P15A/B and spare should the need for replacement blades arise.

Safety Evaluation Summary

This modification supports the use of Hylum blades with a length ranging between 2.960-2.970 inches, in lieu of the previous tolerance band. The basis of this change is material specific laboratory testing in water/oil baths, and pump performance testing.

This activity enhances the reliability of the DAS Sump Pumps to ensure that they are able to perform their safety related function of removing ground water from under Containment, and does not increase the probability of a malfunction. Malfunctions evaluated include the inability of the pump to generate sufficient start-up torque due to employing a shorter blade in the air motor housing, inability of the revised blade length tolerance band to prevent a blade growth related failure, and the inability of the DAS Sump Pumps to meet surveillance test flowrate requirements due to a blade related loss of performance.

S3-EV-00-0058
DCN DM3-00-1046-98

Installation of True Test in Place (TTIP) Hardware to Reactor Coolant System Hydraulic Snubbers (RFO6)

Description

Lateral support for the Steam Generators and Reactor Coolant Pumps are provided by large bore hydraulic snubbers which allow unrestrained thermal movement, but restrict dynamics motions, such as those occurring during a seismic event. In order to assure proper functioning of these snubbers, periodic inspection and testing is required per Technical Specifications. TTIP was developed as a testing method.

This DCN installed TTIP hardware that added a stainless steel test manifold (two per snubber) between the snubber body and control valves. The TTIP hardware package includes replacement tubing and attachments pre-formed to accommodate the test manifolds.

Reason for the Activity

Technical Specifications requires periodic testing of snubbers. The change continues to install TTIP hardware, as approved in PDCR MP3-87-051, to allow the safe testing of hydraulic snubbers without either removing the snubber or clevis pin. This greatly reduces maintenance time and radiological exposure.

Safety Evaluation Summary

Installation of the TTIP hardware is required to enable the functional testing requirements to be ascertained with a minimal amount of effort. The modification adds less than thirty pounds of material, in comparison to the total weight of approximately 4000 pounds. The TTIP hardware has a negligible effect on the seismic adequacy of the existing snubbers and its introduction does not present any new failure mode.

The installation of the TTIP hardware does not impact the probability of occurrence nor the consequences of any evaluated accident and does not create any new accidents. In addition, no new failure modes are introduced. There is no impact on radiological dose consequences, and the overall safety is improved.

S3-EV-00-0061
MP3 EEQ Program Deletion of the Westinghouse Power Range Detectors for the Master List

Description

This activity removed the MP3 Nuclear Instrumentation System (NIS) Power Range Neutron Detectors (PRND), 3NMP*DET41A/B, 3NMP*DET42A/B, 3NMP*DET43A/B and 3NMP*DET44A/B, from the Equipment Qualification Master List (EQML).

This activity was an EQ program classification change only and had no effect on plant hardware.

Reason for the Activity

In the 1980's, many Class 1E electrical equipment were conservatively included in the EQML in anticipation of the commercial licensing of the plant. This conservative approach was utilized in lieu of performing analysis to justify exclusion from the list. A result of the conservative approach is the undue burden placed on the plant in the form of potentially unnecessary Environmental Qualification (EQ) related maintenance activities. Removal of the PRND from the EQML relaxes and replaces the 10-year power operation EQ replacement interval with a replacement/maintenance program that is based on the actual condition of the detectors.

Safety Evaluation Summary

This activity does not affect the way the Reactor Protection System functions, nor does it affect the accident mitigating functions of the plant. This change does not degrade the protective boundaries nor does it increase the probability of occurrence of already analyzed accidents or malfunctions.

The basis for the deletion is the fulfillment of the NUREG 0588 Appendix E section 2C criteria. This is consistent with Federal Regulation 10 CFR 50.49 which requires that only safety-related equipment that must function during a Design Basis Accident, which exposed to the harsh environment, be included in the EQML.

S3-EV-00-0062
MP 3709C, Revision 004-01
MP 3709C-001, Revision 004-01
Temp. Mod. 3-00-008

Freeze Seal to Support Valve 3SSR*CTV19D Replacement

Description

This temporary modification installed a freeze seal on the 'D' steam generator sample line. The freeze seal was installed on 3SSR-750-075-02, upstream of the outboard containment isolation valve to allow replacement of 3SSR*CTV19D. The maintenance procedure for installation of a freeze seal, MP3709C and the associated form MP3709-C-001, required one time changes to allow the freeze seal to be installed on a line with a pressure greater than 400 psig. The existing procedure limited the use of a freeze seal to line pressures less than 400 psig. During the duration of the activity, 3SSR*V2019 was used as the containment isolation boundary.

Reason for the Activity

3SSR*CTV19D required replacement to restore operability of the valve. The freeze seal was necessary to allow welding the replacement valve due to the slight leakage past the upstream manual isolation valves (3SSR*V705 and 3SSR*V2019). The leakage past these valves could result in a pressure at the freeze seal equivalent to that of the secondary side of the steam generators. This pressure is greater than the existing allowable pressure in the freeze seal procedure, and therefore a one time change was necessary to allow installing a freeze seal at the higher pressure.

Safety Evaluation Summary

Installation of the freeze seal and associated procedure changes did not affect any Design Basis Accident or their consequences. It did not create any malfunctions or contribute to any new accidents or new malfunctions.

The freeze seal served as a maintenance boundary during this evolution and the containment isolation was maintained by closing the upstream manual valve (3SSR*V2019). This valve was designed to isolate 3SSR*CTV19D to perform maintenance at power. As specified in the TRM, the valve meets the requirements for containment isolation. The line pressure during the freeze seal was below the design pressure of the line and the valve downstream of the freeze seal (3SSR*V705) was open to ensure there was an adequate number of pipe diameters downstream of the freeze seal to prevent pressure buildup. This ensured the design requirements were maintained and no failure of the piping would occur due to installation of the freeze seal.

S3-EV-00-0065
Temp. Mod. M3-00-004

Discharge of Traps 3DTM-TRP16A/B/D to Drain Rather Than 3CNA-TK2

Description

This temporary modification provided an alternate means of collecting and processing the condensate removed by steam traps 3DTM-TRP16A/B/D from the turbine driven auxiliary feedwater pump steam supply lines. It installed a quenching barrel to collect the condensate from the steam trap common discharge header drain valves, 3DTM-V916 and 917 and threaded pipe and fittings at the discharge of drain valve 3DTM-V916. It opened drain valves 3DTM-V916 and 917 and closed the Turbine Driven Auxiliary Feedwater (TDAFW) pump steam supply drain header throttle valve 3DTM-V901, resulting in the diversion of condensate to Sump 8 rather than the auxiliary condensate flash tank 3CNA-TK2. Red tagging closed the trap bypass valves 3DTM-AOV65A/B/D ensuring the safety of Operations personnel monitoring the quenching barrel.

Reason for the Activity

The modification was needed to maintain operability and availability of the turbine driven auxiliary feedwater pump 3FWA*P1 while performing on-line maintenance on the auxiliary condensate systems.

Safety Evaluation Summary

This modification adequately drained condensate from the TDAFW pump steam supply lines and therefore ensured the TDAFW pump performed its safety functions. It did not alter the function of any systems, structures, or components, and did not affect any other equipment important to safety. There were no high energy piping systems subject to High Energy Line Break postulation located in the fuel building in the area of the temporary modification that were exposed to the discharge of the temporary drain piping. Flooding was bounded by Calculation PR-1038, and there were no spray sensitive essential components located in this area. The change did not affect the capability of Sump 8 or the radwaste system.

S3-EV-00-0066
SPROC ENG 00-3-03, Rev. 001-05

Millstone 3 Rerack Installation

Description

This proceduralized temporary modification installed a temporary new fuel handling crane trolley position interlock switch approximately 14 feet east of the existing switch. The existing switch was disconnected from the control circuit and the temporary switch was connected in its place. This change allowed new spent fuel rack installation to be completed while prohibiting the crane hook from traversing over stored fuel. The new fuel handling crane bridge position interlock was previously relocated to allow crane travel over the northern 10 feet of the spent fuel pool.

Reason for the Activity

Relocating the new fuel handling crane trolley position interlock was required to install the final 6 new spent fuel racks prior to receiving a license amendment that specifically allows bypassing the new fuel handling crane interlocks for rack installation.

Safety Evaluation Summary

Relocating the new fuel handling crane trolley position interlock allowed moving the crane hook over an area of the spent fuel pool that does not contain fuel while preventing movement of loads over fuel assemblies stored in the spent fuel pool. The interlock provided assurance that a load could not be carried over fuel stored in spent fuel racks.

S3-EV-00-0067
SPROC EN 98-3-06, Rev. 2
SPROC EN 99-3-9, Rev. 1

3HVK*CHL 1A Condenser Thermal Performance Test
3HVK*CHL 1B Condenser Thermal Performance Test

Description

These procedures directed the conduct of a thermal performance test on the condensers for the Control Building Chillers 3HVK*CHL 1A (SPROC EN98-3-06) and 3HVK*CHL 1B (SPROC EN99-3-9). The test determined the heat removal capability of this safety related heat exchanger.

Reason for the Activity

These tests were required to satisfy the requirements of Nuclear Regulatory Commission Generic Letter 89-13. They provide a method to measure parameters used in documenting the heat exchanger thermal performance characteristics for the Control Building chiller condensers.

Safety Evaluation

The Control Building chiller, Control Building Ventilation System and the Service Water System were all operated within their design parameters. Nothing about this test changed the function of any of these systems. Test equipment was installed on the Service Water System, Chilled Water System and on the chiller itself in order to measure appropriate parameters during the test. Portable heaters were installed in the east and west switchgear rooms of the Control Building in order to place additional heat load on the chillers and test them near their design capacity. Parameters on the chillers and the Service Water System were monitored during the test to ensure no limits were exceeded during the test. Installation of the portable heaters in the switchgear room was evaluated for ignition source issues.

All the steps within these procedures are consistent with normal operating procedures and the functions of the Ventilation System and Service Water System were not changed by the performance of these tests.

S3-EV-00-0071
Temp. Mod. 3-00-010

Temporary Connection of Instrumentation to Monitor Performance of "B" Hydrogen Recombiner

Description

This temporary modification installed test instrumentation on the "B" Hydrogen Recombiner to record detailed pressure and differential pressure measurements during performance of the recombiner blower surveillance test. A pressure sensor was installed at the inlet of the positive displacement blower and differential pressure sensors were installed across the blower, the flow meter, and across the recombiner system to allow for analysis and evaluation of system and equipment resistances to flow. The test instruments were connected to the "test tee" fitting connections of the existing recombiner instruments. The tubing and fitting materials were in accordance with SP-EE-212, Standard Specification for Instrumentation Installation, Piping and Tubing, requirements for safety related QA applications. The test instruments were standard Instrument and Controls Measuring and Test Equipment (M&TE). These instruments and tubing were temporary and were not seismically restrained or supported. This temporary modification has been removed.

Reason for the Activity

The "B" Hydrogen Recombiner failed to meet Technical Specification refueling interval surveillance requirement for blower flow rates. This surveillance utilizes the installed permanent plant instrumentation to record the various data points utilized to verify the surveillance requirements. Installation of the test equipment allowed varied and more accurate measurements to be taken to facilitate a more detailed evaluation of the flow test results.

Safety Evaluation Summary

The temporary modification installed test instrumentation on the "B" Hydrogen Recombiner to record detailed pressure and differential pressure measurements during the surveillance test for the recombiner blower flow verification. The test instruments were standard I&C M&TE equipment and were connected to existing test fitting connections of the permanent recombiner instruments. The tubing and fitting materials were in accordance with SP-EE-212, Standard Specification for Instrumentation Installation, Piping and Tubing requirements for safety related QA applications. Although these instruments, the tubing, and fittings were temporary and not seismically restrained or supported, the probability of a seismic event occurring during that period concurrent with a Loss Of Coolant Accident was extremely small and was not considered to be a credible challenge to the integrity and the design basis function of the containment pressure boundary.

This safety evaluation determined that the temporary modification would not result in any increase in either the probability of occurrence or consequences of a malfunction of equipment or accidents previously evaluated in the Safety Analysis Report (SAR), would not result in the possibility of a malfunction of accident of a different type than previously evaluated in the SAR, and would not reduce the margin of safety as defined in the basis for any Technical Specification.

S3-EV-00-0075
MMOD DCN DM3-00-0352-00
MMOD DCN DM3-00-0353-00

Automatic CO2 Pressure Relief Dampers 3FPL-DMPR4 and 5 Replacement with Manual Pressure Relief Dampers

Description

This MMOD changed self-actuating pressure relief dampers 3FPL-DMPR4 and 3FPL-DMPR5 to manually operated dampers. It added an instruction to procedure OP3341C and the MP3 Fire Fighting Strategies to open the affected switchgear area damper corresponding to the fire as part of the initial operator or fire brigade member response to a fire detection signal.

Reason for the Activity

The damper replacement is in response to an inadvertent actuation of CO2 in the Cable Spreading Room and the subsequent CO2 migration into the East and West Switchgear Rooms. The migration into the East and West Switchgear Areas (ESA, WSA) exceeded CO2 concentration limits for these areas and was partially attributed to leakage past the closed self-actuating pressure relief dampers 3FPL-DMPR4 and 3FPL-DMPR5.

The primary reason for the design change is to ensure the ESA and WSA remain habitable by providing a superior leak-tight damper design. The new manual dampers are designed with a significantly lower allowable leakage past the closed damper compared to the existing automatic pressure relief damper.

Safety Evaluation Summary

The non-safety related, manually actuated Fire Protection CO2 relief ventilation system in the ESA and WSA is not credited with performing any accident mitigation function and failure of this system does not initiate any postulated events analyzed by the Safety Analysis Report.

Creating an additional manual operator action to the existing manual CO2 system does not increase the probability a fire will occur. The new manual pressure relief dampers will not cause an over-pressurization because they will be required to be opened (by procedure OP3341C) prior to performing the manual CO2 discharge actuation. The Millstone Unit No. 3 Fire Fighting Strategies also describe the necessary actions. The occurrence of an inadvertent CO2 discharge actuation in the ESA or WSA is not increased by this modification. Separation criteria and 3 hour fire barriers between the ESA and WSA remain intact. Therefore, safe shutdown equipment and requirements for these areas during a CO2 discharge and/or fire remain available. Personnel safety for the affected areas is enhanced by the superior leak tightness of the manual dampers.

S3-EV-00-0077
SPROC EN 99-3-15

Fan Pressure Test for Cable Spreading Area

Description

A portable fan was used to pressurize the Cable Spreading Area to less than 10 inches water gauge (wg). The Cable Spreading Area northwest stairway door (C-24-3) was opened to accommodate installation of a duct connection on the inside frame. Ducting was run from the connection at door C-24-3 through the northwest stairway outside door (C-24-1/1A) to the portable fan located just beyond C-24-1/1A. In order to achieve sufficient pressure in the area, 3FPL-DMPR9&10 was blocked closed. The SPROC had 3 sections. The first section verified that the fan arrangement could not exceed 10" w.g. pressure and that the room could be depressurized in less than 45 seconds. The second section pressurized the room and verified that repairs made as a result of leakage found during the performance of Rev. 0 of this SPROC, during refueling outage RFO6 were adequately repaired. The final section used a tracer gas Sulfur Hexaflouride (SF6), injected into the room, and monitored the adjacent spaces for SF6 to determine the leakage rate into the adjacent areas of interest. Portable fans were required in the Cable Spreading area and possibly in the Control Room Boundary to mix the SF6.

Reason for the Activity

To determine leakage paths and quantities down to the East and West Switchgear Rooms, up toward the Control Room/Instrument Rack Room and in the Service Building West hallway. Leak rates were identified by injecting SF6 and sampling adjacent areas of interest to detect the presence of the gas.

Safety Evaluation Summary

The temporary equipment required for the SPROC includes a portable fan, ducting, and a specially constructed duct connection that was placed within the inside frame of blocked open door C-24-3. Outside door C-24-1/1A was also blocked open to allow ducting to be run from the duct connection at door C-24-3 to the portable fan located outside. Operation of the portable fan was monitored to assure Cable Spreading Area pressure did not exceed 10" wg. The fan shutoff point was less than 10" wg,. A fire watch and security watch was posted near door C-24-3. Therefore, any problems (including a fire) would be readily detected so action would be taken to minimize damage. The special duct connection at the frame of door C-24-3 was made of fire retardant material, so it was unlikely that any fire would spread to the Cable Spreading Area. The temporary equipment could be readily removed so both C-24-3 (a 3-hour fire-rated security door) and C-24-1/1A (1A is a tornado door) could be closed if required. A dedicated operator was in constant communication with the fan operator while the Cable Spreading area was pressurized. In the event of a Control Building Isolation or any plant transient, the fan would be shut off and the room depressurized. The time to depressurize would be verified to be less than 45 seconds by this SPROC.

S3-EV-00-0081
MMOD DM3-00-0407-00

Steam Generator Blowdown Flow Transmitters

Description

MMOD DM3-00-0407-00 documents replacement of the "D" steam generator blowdown flow transmitters (3BDG-FT46D, 47D) due to recurrent calibration problems and equipment obsolescence. The modification replaced both the low and high range transmitters with a single transmitter for the D steam generator blowdown line. A single transmitter was utilized since system limitations on flow including blowdown control valve size allowed changing the flow range from 0-400 gpm to 0-200 gpm on each of the four steam generator blowdown lines. System operation was not altered by this modification. MMOD DM3-00-0407-00 subsequently replaced the "A", "B", and "C" steam generator blowdown flow transmitters with a similar transmitter.

Reason for the Activity

This modification provided a more reliable and readily available transmitter to monitor system flows. This change provided a simplified method in providing flow indication to minimize corrective maintenance on the existing antiquated system by replacing the low and high range flow transmitters with a single full range transmitter. This change increased equipment reliability without changing or compromising system operation.

Safety Evaluation Summary

The change allowed modification of existing obsolete equipment with simplified and more reliable flow instrumentation. This change had no effect on system operation or system response. It did not alter system/component qualifications or affect the function of the blowdown system. This change provided more reliable steam generator blowdown flow indication. This safety evaluation addressed replacement of each of the four steam generator flow transmitters.

S3-EV-00-0086
MMOD DCN DM3-01-0117-00

Erosion/Corrosion Replacement for 3-SVH-006-031-4

Description

This activity replaced a portion of carbon steel class 151 line, 3SVH-006-031-04, which had been degraded by Flow Accelerated Corrosion, (FAC), with stainless steel class 302 piping, fittings, and valves. This material is less susceptible to FAC than the existing carbon steel and will last until end of plant life with no further replacements or inspections required. This change impacts drawing P&ID EM-124A, section M-9, which is figure 10-04-03, sheet 1, in the Final Safety Analysis Report and thus requires a Safety Evaluation.

Reason for the Activity

The existing piping was projected to erode to below its minimum allowable wall thickness during the next operating cycle. In order to assure continued plant operation and compliance with design codes this piping needed to be replaced. The choice of stainless steel insures that replacement of the piping will not be required for the remainder of Millstone Unit No. 3 plant life and removes this line from the requirements of continued wall thinning inspections.

Safety Evaluation Summary

The replacement piping and components comply with all design codes, specifications and standards for Millstone Unit No. 3 and does not degrade plant system performance in any way. The use of stainless steel in secondary side environments eliminates the potential for wall thinning of the piping due to FAC. This ensures continued functionality of the system for the remaining life of the plant without need for continued piping inspections to ensure compliance with minimum wall thickness requirements. The differences in material strength and coefficient of expansion between the original and new design have been dispositioned in the MMOD and do not impact the functionality of the system. The piping was procured and installed to the same codes and standards as the original piping. The piping system follows the same routing as the original system and the pipe size in unchanged. The Feed Water Heater Vents and Drains system is not a system which mitigates or initiates any design basis accidents. Its failure or malfunction will not result in any dose impact to the plant personnel or to the public. There is no increase in consequences of accidents previously evaluated.

S2-EV-99-0149
DCR M2-99065
FSARCR 00-MP2-4
FSARCR 00-MP3-3

Redesign of Existing Severe Line Outage Detector (SLOD) System and Master Supervisory Panel

Description

The Severe Line Outage Detection (SLOD) system is designed to prevent instability and loss of all generation at Millstone Station. Besides avoiding unit instability, a distribution system casualty with generation above 1300-1400 MW at Millstone Station could have severe, adverse consequences on Pennsylvania and/or New York grid reactive and thermal operating conditions. The SLOD system is continuously armed and avoids instability and loss of all generation at Millstone by tripping only pre-selected units when certain conditions exist. The tripping logic associated with the SLOD system was modified to remove all trips associated with Millstone Unit No. 1. The Double Line and Breaker Failure Detection Unit Rejection Special Protection System (DBURS), two more Special Protection Systems (SPS) used to trip pre-selected units at Millstone, were deleted and removed since their functions were no longer needed due to the loss of Millstone Unit No. 1 generation. CRP-909 was connected to the master supervisory panel in the 345-kV switchyard via a new fiber optic cable. Switches on CRP-909 for control of Millstone Unit No. 1 switchyard circuit breakers and motor operated disconnects were removed.

Reason for the Activity

The decision to decommission Millstone Unit No. 1 requires the re-powering or relocation of certain systems needed for continued operation of Millstone Unit Nos. 2 and 3. The existing SLOD system was located in the Millstone Unit No. 1 control room and received power from Millstone Unit No. 1 power sources. Since this capability is still required after the decommissioning of Millstone Unit No. 1, the system required re-powering from Millstone Unit No. 2 to protect the off-site electrical grid during operation of Millstone Unit Nos. 2 and 3.

Safety Evaluation Summary

The changes made to the SLOD system do not affect any Design Basis Accidents or the consequences of these accidents. The changes to the SLOD system do not create new accidents beyond those analyzed in the Millstone Unit Nos. 2 and 3 FSARs. It does not increase the probability of an accident or malfunction, increase the consequences of an accident or affect accident mitigation. The mechanism by which SLOD trips the switchyard circuit breakers for Millstone Unit No. 3 was not affected by any of the modifications to CRP-909 or the switchyard Master Supervisory Panel. These two panels provide indication, control and alarm functions, but have no impact on the protection SLOD provides to both Millstone Station and the off-site electrical grid.

Docket Nos. 50-336
50-423
B18442

Enclosure 3

Millstone Nuclear Power Station, Unit Nos. 2 and 3

Annual Commitment Change Report for 2000

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
A04541-05	Each refueling outage, a General Electric (GE) service engineer is on site to review maintenance practices and assist in breaker maintenance. Any recommended changes to maintenance procedures to reflect the most current experiences of the vendor and as-found breaker conditions would be incorporated as necessary. This has been the practice in the past and Millstone Unit No. 2 intends to continue it in the future.	Each refueling outage, maintenance is conducted for Reactor Trip Circuit Breakers. A GE Service Engineer may be consulted, as needed, to review Reactor Trip Circuit Breaker maintenance practices and assist in breaker maintenance. Maintenance procedures relating to Reactor Trip Circuit Breakers shall be periodically reviewed and updated based on the most current experiences of the vendor and as-found breaker conditions.	Procedure reviews for technical content and compliance to regulatory commitments are conducted biennially. Workers are required to be trained and qualified on this equipment in accordance with the Millstone Nuclear Training Manual and INPO ACAD 91-105, "Objectives and Criteria for Accreditation of Training in the Nuclear Power Industry." Vendor Service Information Letters are reviewed to determine if changes are required to maintenance practices.
A04541-08	The General Electric instruction applicable to Millstone Unit No. 2 Reactor Trip Switchgear is GEI-50299A. This manual recommends yearly breaker inspection. The average operating cycle at Millstone Unit No. 2 is approximately thirteen (13) months. Therefore, the breakers are only in service approximately one month longer than the recommended maintenance interval.	Inservice Monitoring of Reactor Trip Switchgear tripping performance is conducted to establish optimal maintenance intervals. Maintenance activities for the Reactor Trip Switchgear consistent with the vendors yearly breaker inspection guidelines is performed as a minimum on a refueling outage frequency.	Response time is monitored monthly during the Reactor Protection System Matrix Logic and Trip Path Relay Test; trip shaft torque testing is conducted quarterly to assess the performance of Reactor Trip Circuit Breakers (TCBs). Historical Records show no Reactor TCBs have failed to trip or to meet vendor acceptance criteria for trip shaft torque.
B15804-01	The inside of the affected service water piping will be inspected during each refueling outage and during any planned or unplanned outage of greater than six weeks duration. The inspections will be performed using submersible robotic video cameras. The results from previously recorded inspections including the areas that were repaired and historic condition of adjacent piping will be used in evaluating the results of each inspection.	The inside of the affected service water piping will be inspected during each refueling outage and during any planned or unplanned outage of greater than six weeks duration. The inspections will be performed using submersible robotic video cameras. The results from previously recorded inspections including the areas that were repaired and historic condition of adjacent piping will be used in evaluating the results of each inspection. The piping has been further protected from degradation by the installation of an EPDM rubber sleeve and mechanical seals over the epoxy coated area. Inspection in each outage will be performed to assure that the seal remains in place and the exposed piping surrounding the seal is not degraded.	This change does not revise the committed action, but provides the details of further repairs which have been made to this piping in an effort to further maintain the piping integrity. Installation of the rubber sleeve and seals was done to provide additional protection of the degraded piping area. Results of the robotic inspection of this coating performed during Refueling Outage 6 showed that it remains intact and in excellent condition.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
A07834-06	In response to Generic Letter 88-17, the licensee committed to maintain two channels of reactor vessel level monitor systems (RVLMS) in operation during mid-loop operations [to provide Core Temperature Measurement].	In response to GL 88-17, the licensee committed to maintain two channels of RVLMS in operation during mid-loop operations (to provide Core Temperature Measurement). One channel at a time may be temporarily disconnected to facilitate reactor cavity pit seal installation or removal, provided that no change in Reactor Coolant System (RCS) inventory is performed while only one channel is operable.	The actual processes of vessel disassembly/ assembly and pit seal installation/removal necessitate the temporary disconnection of one Heater Junction Thermal Couple cabinet at a time for a period of about two hours for each facility. The intent of the original commitment was to ensure that we would have reliable temperature monitoring capability representative of core exit conditions. It is our intent to have two channels of RVLMS in operation during reduced inventory operations with the reactor vessel head installed except as necessary to facilitate installation or removal of the reactor cavity pit seal. Additionally, although the Core Exit Thermocouples are below their calibrated range (<200 Degrees), they can be used to provide backup information representative of core exit temperature.
B16113-02	A design change to simplify maintenance activities will be implemented during the next refueling outage in which Engineered Safeguards Actuation System (ESAS) actuation cabinets 5 & 6 are downpowered.	A design change to simplify maintenance activities will be implemented during the next refueling outage after Millstone Unit No. 2 Refueling Cycle 13 in which ESAS actuation cabinets 5 & 6 are downpowered.	Millstone Unit No. 2 unexpectedly downpowered ESAS actuation cabinets during Refueling Outage 13 to support installing new under-voltage actuation modules in response to an intermittent problem observed by Operations. The design change that supports this commitment requires additional time to develop/complete. It is anticipated that the due date of Refueling Outage 14 will be met, during an anticipated down-powering of the cabinets, as originally scheduled.
B17225-14	[RAC 13, "Organizational Changes"] This procedure describes the process to evaluate and implement organizational changes to ensure changes are made in compliance with all regulatory requirements. The use of this procedure will avoid implementing organizational changes that are contrary to the Licensing Basis.	No change in description. Change to "one-time only" rather than an active commitment.	The license basis organizational description was moved to the Quality Assurance Program (QAP) and removed from the other references. Millstone Staffing & Organization Management Procedure adequately addresses that the QAP needs to be reviewed when changes to the organization are made. Therefore, based on the above, maintaining this commitment as an active commitment is no longer necessary.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
B17248-01	Verify that the trip torque required on the trip shaft is less than 1.5 pound-inches, as specified in General Electric (GE) Service Advice 175-9.3S, item #S4; "As-found" torque values will be recorded. This surveillance will be performed quarterly.	Verify that the trip torque required on the trip shaft is less than 1.5 pound-inches, as specified in GE Service Advice 175-9.3S, item #S4; "As-found" torque values will be recorded. This surveillance will be performed semi-annually.	Northeast Utilities provided the response to Generic Letter 83-28 in letter A04541, dated June 25, 1985. This letter forwarded Millstone Unit No. 2 information with respect to periodic maintenance and trending programs of the reactor trip breakers. Millstone Unit No. 2 committed to perform the check of the trip torque on a six month basis. In the Safety Evaluation Report enclosed in letter A05625, dated March 7, 1986, the staff evaluated and determined the maintenance interval of 6 months for performing this trip torque verification to be acceptable.
B17248-05	Trip torque will be trended. This trending will be performed quarterly.	Trip torque will be trended. This trending will be performed semi-annually.	Northeast Utilities provided the response to Generic Letter 83-28 in letter A04541, dated June 25, 1985. This letter forwarded Millstone Unit No. 2 information with respect to periodic maintenance and trending programs of the reactor trip breakers. Millstone Unit No. 2 committed to trending trip torque on a six month basis. In the Safety Evaluation Report enclosed in letter A05625, dated March 7, 1986, the staff evaluated and determined the trending of the trip torque on a six month basis to be acceptable.
B17248-08	The on-line testing conducted for the reactor trip system includes trip bar torque testing. This surveillance will be performed quarterly.	The on-line testing conducted for the reactor trip system includes trip bar torque testing. This surveillance will be performed semi-annually.	NRC letter A05625, dated March 7, 1986, provides the Safety Evaluation Report (SER) for Northeast Utilities response to Generic Letter 83-28 for Millstone Unit No. 2 with respect to periodic maintenance and trending programs of the reactor trip breakers (RTBs). Millstone Unit No. 2 committed to perform the check of the trip bar torque on a six month basis in letter A04541, dated June 25, 1985. The SER determined this maintenance interval to be acceptable. Performing this testing more frequently as part of on line testing is not required based on Generic Letter 83-28, section 4.5.1, and the Safety Evaluation, letter A05638, dated March 18, 1986.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
12/31/79-177	An independent review of Plant Incident Reports and Licensee Event Reports (LERs) is conducted by the Northeast Utilities Service Company Nuclear Operations Department. The Nuclear Operations Engineer is responsible for this review; it is performed by degreed engineers and technical personnel. The Nuclear Review Board (NRB) also reviews operational experiences. The NRB has the responsibility for reviewing significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety, and reportable occurrences requiring twenty-four hour notice to the Nuclear Regulatory Commission. In addition, the NRB will assess the operating experience at plants of like design based on a review and evaluation of LERs for like units compiled in the NRC's LER Monthly Report. The NRB will meet monthly to evaluate the safety significance of like plant LERs and will assign appropriate follow-up action on those LERs determined to be potentially significant to plant safety. The NRB will periodically issue a report to [plant management] summarizing their review and evaluations with appropriate recommendations. The results from each of the operational assessment functions described above will be forwarded to the plant Shift Technical Advisors (STA) to ensure close coupling of the operating experience assessment function and the STA function.	Cancel this commitment.	The Operating Experience (OE) program is an established program within the industry. It is integrated into the INPO assessment standards and included in routine INPO evaluations. The OE program is included in the Millstone Station Oversight and Information Technologies plan and is performed approximately once every two years. The results are forwarded to the Nuclear Safety Advisory Board. The operating experience assessment function was established after Three Mile Island in response to NUREG 0578. The OE function was further delineated and described in NUREG 0660 and NUREG 0737. Millstone is committed to the implementation of NUREG 0737.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
A01024-25	I.C.5, Licensee Dissemination of Operating Experiences: Northeast Utilities Energy Company will review and revise operating procedures as necessary to assure that operating information pertinent to plant safety originating both within and outside the utility organization is periodically supplied to operators and other personnel and is incorporated into training and retraining programs. The guidelines found in Attachment I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff," will be used in the development of this program. The review and modification to procedures deemed appropriate by the review is scheduled to be accomplished on the time schedule mandated by the Nuclear Regulatory Commission.	Cancel this commitment.	The Operating Experience (OE) program is an established program within the industry. It is integrated into the INPO assessment standards and included in routine INPO evaluations. The OE program is included in the Millstone Station Oversight and Information Technologies plan and is performed approximately once every two years. The results are forwarded to the Nuclear Safety Advisory Board. The operating experience assessment function was established after Three Mile Island in response to NUREG 0578. The OE function was further delineated and described in NUREG 0660 and NUREG 0737. Millstone is committed to the implementation of NUREG 0737.
A02281-02	Item No. I.C.5, Procedure for Feedback of Operating Experience to Plant Staff: You have established a Nuclear Analysis Section within the Corporate Nuclear Engineering and Operations Group. That section is responsible for screening operating experience information, conducting an analysis of the screened information and reporting that information to engineering and operations management. Your Nuclear Engineering and Operations Procedure NEO 5.08, "Operating Experience Assessment and Utilization" directs this program. You have committed to an audit verification of the program by the Nuclear Review Board. Based on our review of the above information, we find your procedures for feedback of operating experience to the plant staff to be acceptable.	Cancel this commitment.	The Operating Experience (OE) program is an established program within the industry. It is integrated into the INPO assessment standards and included in routine INPO evaluations. The OE program is included in the Millstone Station Oversight and Information Technologies plan and is performed approximately once every two years. The results are forwarded to the Nuclear Safety Advisory Board. The operating experience assessment function was established after Three Mile Island in response to NUREG 0578. The OE function was further delineated and described in NUREG 0660 and NUREG 0737. Millstone is committed to the implementation of NUREG 0737.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
B16996-70	Operating Experience (OE) information is reviewed, assessed, and properly utilized to enhance plant safety in accordance with NOQP 3.04, "Independent Safety Engineering Group and Operating Experience Assessment."	Cancel this commitment.	The Operating Experience (OE) program is an established program within the industry. It is integrated into the INPO assessment standards and included in routine INPO evaluations. The OE program is included in the Millstone Station Oversight and Information Technologies plan and is performed approximately once every two years. The results are forwarded to the Nuclear Safety Advisory Board. The operating experience assessment function was established after Three Mile Island in response to NUREG 0578. The OE function was further delineated and described in NUREG 0660 and NUREG 0737. Millstone is committed to the implementation of NUREG 0737.
A12018-30	Review vendor information within 90 days (with allowance for extension when necessary to ensure complete evaluation and resolution).	Cancel this commitment.	The Operating Experience (OE) program is an established program within the industry. It is integrated into the INPO assessment standards and included in routine INPO evaluations. The OE program is included in the Millstone Station Oversight and Information Technologies plan and is performed approximately once every two years. The results are forwarded to the Nuclear Safety Advisory Board. The operating experience assessment function was established after Three Mile Island in response to NUREG 0578. The OE function was further delineated and described in NUREG 0660 and NUREG 0737. Millstone is committed to the implementation of NUREG 0737.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
B14703-82	Corrective steps that will be taken to avoid further Violations (Violation C). Two additional engineers/senior engineers will be added to the Independent Safety Engineering Group (ISEG) organization, such that the ISEG function will be extended beyond the technical specification requirements for Millstone Unit No. 3 and expand the voluntary ISEG functions at Millstone Unit Nos. 1 and 2 as well. This action was scheduled for completion by March 1, 1994.	Cancel this commitment.	The above commitment was made to strengthen the focus of ISEG on Millstone Unit Nos. 1 & 2. At that time, the Technical Specification requirement was for four ISEG engineers and this commitment added two. Currently Millstone Unit No. 1 is permanently defueled, therefore, there is no longer a need for ISEG at Millstone Unit No. 1. The requirement for the ISEG function to be performed by five engineers/senior engineers remains for Millstone Units Nos. 2 & 3 in the Quality Assurance Program. Based on this, there is no need to retain this commitment as the ISEG function for the operating units will continue to be performed and any changes to the ISEG function are covered under 10CFR50.54(a).
B18151-02	Enhancements will be made to applicable sections of the station work control procedure to address this condition.	Enhancements will be incorporated into applicable sections of the U2WC1 procedure to address this condition.	Enhancements will be incorporated into Unit No. 2 Work Control Procedure U2WC1 in lieu of station procedures.
B18004-01	Obtain station approval and implement design changes to align the existing design basis lower limit for service water with the historically lower temperatures observed prior to the onset of next winter's sub-33 degree Fahrenheit water temperatures. Due: December 15, 2000	Obtain station approval and implement design changes to align the existing design basis lower limit for service water with the historically lower temperatures observed prior to the onset of next winter's sub-33 degree Fahrenheit water temperatures. Due: January 4, 2001	Date change only. Historical Plant Process Computer data for Millstone Unit No. 3 indicates that the 33-degree Fahrenheit low design basis temperature is not approached until mid to late January.
B13144	All qualifications are now conducted outdoors in varying weather conditions and flashing lights are utilized during night familiarization…	All qualifications, including night familiarization, are now conducted outdoors in varying weather conditions…	Flashing lights provide no added value to the training program.
B17934-01	Refueling procedures will require that one train of Spent Fuel Pool cooling, with sufficient backups, be available at the commencement of a full core off-load. Due: Prior to implementation of license amendment.	Refueling procedures will require that one train of Spent Fuel Pool cooling, with sufficient backups, be available at the commencement of a full core off-load. Due: Prior to implementation of license amendment.	Date change only. Commitment due date was originally set for December 9, 2000. Implementation of the license amendment was delayed, and commitment due date was subsequently changed to January 26, 2001, which coincided with the Corrective Actions process for Condition Report M3-01-0015. Associated Action Request Assignment 00007539-01 was completed on January 10, 2001.

COMMITMENT CHANGES

Commitment Number *	Original Commitment	Revised Commitment	Remarks
B17934-02	Compensatory measures for restoring Spent Fuel Pool (SFP) Cooling will be described in an operating procedure and will include use of a dedicated temporary power cable for the SFP cooling pumps. Due: Prior to implementation of license amendment.	Compensatory measures for restoring Spent Fuel Pool (SFP) Cooling will be described in an operating procedure and will include use of a dedicated temporary power cable for the SFP cooling pumps. Due: Prior to implementation of license amendment.	Date change only. Commitment due date was originally set for December 9, 2000. Implementation of the license amendment was delayed, and commitment due date was subsequently changed to January 26, 2001, which coincided with the Corrective Actions process for Condition Report M3-01-0015. Associated Action Request Assignment 00007539-02 was completed on January 3, 2001.

* References

(Reference Order Corresponds to Commitment Change Order)

Letter No.	Information	Subject
A04541	NU letter dated June 25, 1985 J. F. Opeka to Director of Nuclear Reactor Regulation Attn: Mr. Edward J. Butcher	Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information Generic Letter 83-28, Generic Implications of Salem ATWS Events
A04541	NU letter dated June 25, 1985 J. F. Opeka to Director of Nuclear Reactor Regulation Attn: Mr. Edward J. Butcher	Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information Generic Letter 83-28, Generic Implications of Salem ATWS Events
B15804	NU letter dated July 23, 1996 T. C. Feigenbaum to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 3 Request to Use Alternative to ASME Code Section III
A07834	NRC Letter, dated February 13, 1989 Lee H. Bettenhausen to Edward J. Mroczka	Millstone 2 Routine Inspection 50-336/88-28 (11/24/88 - 1/10/89)
B16113	NNECO. letter, dated March 3, 1997 J. A. Price to U.S. Nuclear Regulatory Commission	Licensee Event Report 95-20-01 Automatic Actuation of an Engineered Safety Feature During Maintenance
B17225	NNECO. letter, dated July 15, 1998 Martin L. Bowling, Jr. to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 3 NRC 40500 Inspection Report No. 50-423/97-82, Reply to a Notice of Violation
B17248	NU letter dated September 30, 1998 Martin L. Bowling, Jr. to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 2 Conformance to Generic Letter 83-28 - Revised Testing Commitments
B17248	NU letter dated September 30, 1998 Martin L. Bowling, Jr. to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 2 Conformance to Generic Letter 83-28 - Revised Testing Commitments
B17248	NU letter dated September 30, 1998 Martin L. Bowling, Jr. to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 2 Conformance to Generic Letter 83-28 - Revised Testing Commitments
12/31/79	NU letter dated December 31, 1979 W. G. Counsil to Office of Nuclear Reactor Regulation Att: H. R. Denton, Director	Haddam Neck Plant, Millstone Nuclear Power Station, Unit Nos. 1 and 2 TMI-2 Short Term Lessons-Learned Implementation
A01024	NU letter dated June 10, 1980 W. G. Counsil to Darrell G. Eisenhut	Haddam Neck Plant, Millstone Nuclear Power Station, Unit Nos. 1 and 2 Five Additional TMI-2 Related Requirements to Operating Reactors
A02281	NRC letter dated February 8, 1982 Mr. Dennis M. Crutchfield to W. G. Counsil	TMI Action Plan Items I.A.1.3, I.C.5, and I.C.6 as Described in NUREG-0737
B16996	NNECO letter dated March 2, 1998 M. L. Bowling, Jr. to Director, Office of Enforcement	Millstone Nuclear Power Station, Unit Nos. 1, 2 and 3 Reply to a Notice of Violation and Proposed Imposition of Civil Penalties, (NRC Inspection Report Nos. 50-245/50-336/50-423: 95-44; 95-82; 96-01; 96-03; 96-04; 96-05; 96-06; 96-08; 96-09; 96-201)
A12018	NRC letter dated December 16, 1994 Jacque P. Durr to John F. Opeka	Notice of Violation (NRC Combined Inspection 50-245/94-31;50-336/94-30; 423/94-28)

* References

(Reference Order Corresponds to Commitment Change Order)

Letter No.	Information	Subject
B14703	NU letter dated January 5, 1994 J. F. Opeka to Director, Office of Enforcement	Millstone Nuclear Power Station, Unit No. 2 Reply to a Notice of Violation, NRC Inspection Report No. 50-336/93-18
B18151	NNECO letter dated June 30, 2000 C.J. Schwarz to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 2 Licensee Event Report 2000-009-00, Entry Into an Operational Mode While in the LCO 3.6.5.2 Action Statement Is a Violation of Technical Specification 3.0.4
B18004	NNECO letter dated February 17, 2000 C. J. Schwarz to U.S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 3 Licensee Event Report 2000-001-00, Ultimate Heat Sink Below Minimum Design Temperature
B13144	NU letter dated February 27, 1989 E. J. Mroczka to U. S. Nuclear Regulatory Commission	Millstone Nuclear Power Station Unit Nos. 1, 2, and 3 "Regulatory Effectiveness Review"
B17934	NNECO letter dated December 21, 1999 R. P. Necci to U. S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 3, Request for Additional Information on Full Core Off-Load, License Amendment (TAC No. MA4586)
B17934	NNECO letter dated December 21, 1999 R. P. Necci to U. S. Nuclear Regulatory Commission	Millstone Nuclear Power Station, Unit No. 3, Request for Additional Information on Full Core Off-Load, License Amendment (TAC No. MA4586)