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

Dominion

JUN 25 2003

Docket Nos. 50-245
50-336
50-423
B18900

RE: 10 CFR 50.59(d)(2)

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

Millstone Power Station, Unit Nos. 1, 2 and 3
10 CFR 50.59 and Commitment Change Report for 2002

Pursuant to the provisions of 10 CFR 50.59(d)(2), the reports for changes made to the facility for Millstone Power Station, Unit Nos. 1, 2 and 3, are submitted via Attachments 1, 2 and 3, respectively. This report covers the period January 1, 2002, to December 31, 2002, with one exception. The one exception was for work associated with S3-EV-99-0092, which was implemented in 2001. This exception still falls within the 24 month report periodicity specified within the regulation.

Attachment 4 transmits the 2002 Commitment Change Report for Unit Nos. 1, 2 and 3. The annual Commitment Change Report is being submitted consistent with the Millstone Power Station's Regulatory Commitment Management Program.

There are no regulatory commitments contained within this submittal.

Should you have any questions regarding these reports, please contact Mr. David W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

J. Alan Price
Site Vice President - Millstone

IE47

cc: See next page

Attachments (4)

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Attachment 1

Millstone Power Station, Unit No. 1

10 CFR 50.59 Report for 2002

Index for Millstone Unit No. 1
Changes to the Facility - 2002

Millstone Unit No. 2 / Millstone Unit No. 3 (MP2/MP3) Station/Service Air Cross-Tie

S1-EV-00-0043 Rev. 0
DCR M2-99061 Rev. 0

Description

This change separated the Millstone Unit No. 1 (MP1) and Millstone Unit No. 2 (MP2) Station Air (SA) systems.

Reason

With the decommissioning of MP1, MP2 lost the capability to have backup SA supplied from the MP1 SA system via an existing MP1/MP2 cross-tie.

Summary

The modification has no direct or indirect impact on equipment important to safety. There is no change in the function of the SA system. The Margin of Safety of the protective boundaries has not been reduced and there will be no adverse impact on public health and safety.

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Attachment 2

Millstone Power Station, Unit No. 2

10 CFR 50.59 Report for 2002

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Changes to the Facility - 2002

Index for Millstone Unit No. 2
Changes to the Facility - 2002

Index for Millstone Unit No. 2
Changes to the Facility - 2002

Letdown Heat Exchanger Shell Side Relief Valve 2-RB-326

S2-EV-98-0277 Rev. 0
DCR M2-98086 Rev. 0

Description

Relief valve 2-RB-326, which protects the Letdown Heat Exchanger shell side, was increased in capacity and size. The existing relief valve connection was blank flanged and a larger connection (1.5 inches) in the shell pressure boundary was made.

The existing relief valve was replaced with a soft-seated valve. At the same time, the set pressure was changed from 165 psig to 150 psig. The change in set pressure agrees with the design pressure of the existing Letdown Heat Exchanger Shell Side and is in accordance with ASME VIII requirements.

Reason

Valve 2-RB-326 was part of another design change that changed the valve from a hard seat to a soft seat and the set pressure to 165 psig. It was noted through a corrective action that the valve was undersized. The initial sizing did not consider the fact that the tube side pressure (600 psig) is greater than the shell pressure (150 psig), and the relief valve was not sized for a tube leak or break. The required relief valve flow is 107 gpm while the existing valve could only pass 24 gpm. The new larger relief valve was attached to the 6-inch discharge line instead of the heat exchanger in order to maintain the heat exchanger's ASME VIII Code Stamp.

Summary

All existing codes and standards are invoked in the new installation. The relief valve is procured in accordance with ASME VIII, the piping in accordance with ANSI B31.1, and supports are in accordance with Seismic I requirement.

The changing of the valve to increase the valve capacity and size, and the relocation of the relief valve to the discharge pipe, does not increase the probability of occurrence or affect consequences of any previously evaluated accidents or malfunctions. It does not create a different type of accident or malfunction. This modification has no impact on the Margin of Safety as defined in the basis of any Technical Specification.

The new Letdown Heat Exchanger shell side relief valve, 2-RB-326, enhances the system's operation and reliability, and provides the required protection in the event of a tube leak or break.

Control Room, Switchgear Area and Turbine Building Built-up Roof Modification

S2-EV-99-0110 Rev. 0
MMOD M2-99031 Rev. 0

Description
This modification provides for replacement of the existing built-up roof with a new, cold application built-up roof on Control Room Roof Building No. 204, Switchgear area roof, and Turbine Building No. 203.

Non Safety Related and Non-QA built-up roofing material was applied over roof slab (Control Room and Switchgear area are on the concrete slab and Turbine Building is on metal deck) to seal the roof, contain ice, snow, and rainwater and to direct the water to the roof drains systems. The built-up roof is not relied upon to perform a safety related function during or following any design basis event.

The replacement built-up roof consisted of an application process that utilized cold application roofing products. This cold roofing material was applied to all three roofs: Control Room roof, Switchgear area roof and Turbine Building roof.

New composition flashing at the walls, parapets, pitch pockets, vent pipes, and roof drains etc., were installed in accordance with the roofing manufacturer's specification and as detailed on the Millstone Unit No. 2 (MP2) design drawings.

Reason
An Adverse Condition Report identified that the MP2 Control Room Building No. 204 was significantly saturated with excess water due to widespread failure of the originally installed roofing and flashing material. Similar conditions were discovered while performing walk-downs of Switchgear area Building No. 203 and Turbine Building No. 203 roofs.

Summary
This activity does not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety or a previously evaluated Accident. It does not create the possibility of a malfunction or an accident of a different type. The implementation of the built-up roof for all three roofs is safe, because the new roofs were installed per the manufacturer's specification, implementation plans and as detailed on the design documents. The design reviews of all applicable documents show that the pressure boundary and structural integrity of the built-up roof as well as building structural integrity remains within the design basis criteria, and are not adversely affected.

Sodium Hypochlorite Supply Injection Location for "A," "B," and "C" Service Water Pumps

S2-EV-99-0153 Rev. 0
DCN DM2-00-0877-99
DCN DM2-01-0877-99

Description

Design Change Notices (DCNs) DM2-00-0877-99 and DM2-01-0877-99 modify the Sodium Hypochlorite Injection system supply line injection locations for the "A" and "B" Service Water (SW) pumps respectively. The subject DCNs relocated the injection point from the SW suction bowl to outside of the bell, at a point located approximately 4 inches above the pump suction baffle. The hole in the pump suction bowl was closed with a threaded plug consistent with existing plugs on the pump lower column. To secure the PVC piping, an additional strap hanger was mounted to the lower pump column using the existing standard detail shown on the pump drawing.

The Facility as described in the Safety Analysis Report was modified to remove the word "directly." from the above description of the injection point.

Reason

Operating experience has shown that sodium hypochlorite crystallizes in the elbow when the SW pumps are taken out of service. The physical arrangement of the piping precludes removal of the blockage from above the pump. The only means of removing the clog is to take the associated SW pump out of service and to send a diver into the bay to manually remove the clog. This method is not cost effective.

Summary

The modification does not affect the ability of any safety system to perform its design function, nor does it affect the integrity of any of the fission product barriers. It results in a minor description change to the Final Safety Analysis Report (FSAR) by removing the word "directly." This change does not affect the ability of the plant to reach and maintain cold shutdown. The modification does not increase the probability of occurrence or consequences of a malfunction of equipment important to safety as previously evaluated, nor does it create the possibility of a malfunction of a different type. The activity does not increase the probability of occurrence or consequences of accidents previously evaluated in the FSAR, nor does it create the possibility of an accident of a different type. The Margin of Safety as defined in the basis for any Technical Specification is not reduced.

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

S2-EV-00-0016 Rev. 0
Temp. Mod. 2-00-003 Rev. 1

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

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.

Summary

The freeze seal was installed while the plant was in mode 5 with the 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 that is well suited to freeze seal information. The pipe integrity Non Destructive Examination verified suitability prior to installing the seal.

Thermal Performance Test of Unit No. 2 Reactor Coolant Closed Cooling Water (RBCCW) Heat Exchangers X18A/B

S2-EV-00-0032 Rev. 0
EN 21246 Rev. 0

Description

This procedure performs a thermal performance test on the RBCCW heat exchangers during plant cooldown, prior to refueling. This test is accomplished by stabilizing the Reactor Coolant System (RCS) temperature in a 10° Fahrenheit (F) band just below 250° F, shifting all of the decay heat removal load onto one Shutdown Cooling (SDC) heat exchanger (and thus one RBCCW heat exchanger), and measuring the heat removal capability of the individual RBCCW heat exchanger. This heat load is then shifted to another RBCCW heat exchanger and the heat removal capability is again measured. Following this test, plant cooldown using two SDC heat exchangers is resumed.

Reason

Thermal performance testing of the RBCCW heat exchangers is required by GL 89-13 in order to validate the cleaning frequency of the heat exchangers. Since the RBCCW heat exchangers are cooled by Service Water (SW), the tubes are subject to micro and macro fouling which can reduce the heat transfer capability of the heat exchanger over time. This thermal performance testing will determine the level of fouling on both an un-cleaned (one that has not been cleaned in over 3 months) and a just-cleaned heat exchanger (within last 14 days) as measured by the heat transfer capability of each heat exchanger under similar heat loads.

Summary

This test will provide heat removal capacity data for the RBCCW heat exchangers. Control of this test is maintained in the control room and RCS temperature control is conducted throughout this test using normal valve manipulations and normal plant indicators. This test places the entire core decay heat load on one SDC and one RBCCW heat exchanger. This is well within the design capability of these heat exchangers in this mode. Additionally, the other SDC and RBCCW heat exchangers remain ready to assume decay heat load with the movement of one valve which is controlled from the control room.

There is no increase in the probability of occurrence of any accident or malfunction as a result of conducting this thermal performance test. No accident initiators are affected by this test. This test does not increase the probability or consequences of any accident and does not introduce the possibility of any new accident.

Instrument Air Accumulator Check Valve Tests

S2-EV-00-0043	Rev. 0
SP2604X and	Rev. 2, Minor Rev.1
Forms 014 & 015	

Description

This activity changed the applicability requirement of the procedure to allow performance of the surveillance in plant modes where the equipment is required to be OPERABLE. The Limiting Condition of Operation (LCO) ACTION statement for the Containment Spray System will be entered.

Reason

The existing procedure required the performance of the surveillance in plant modes in which the equipment is not OPERABLE. This does not allow the facility to efficiently perform the surveillance.

Summary

This activity relies on the LCO ACTION statement for Containment Spray. The LCO ACTION statement is much longer, 72 hours, than the time required to perform the surveillance, 3 hours.

This activity does not initiate a malfunction not previously evaluated in the Safety Analysis Report. The Instrument Air System is not a safety system and the air accumulators are a back up for a loss of instrument air. As loss of air to these valves is not an initiator for the accidents evaluated, it will not increase the frequency or the consequences of those accidents.

Millstone Unit No. 2 / Millstone Unit No. 3 (MP2/MP3) Station/Service Air Cross Tie

S2-EV-00-0072 Rev. 0
DCR M2-99061 Rev. 0

Description

This change separated the Millstone Unit No. 1 (MP1) and Millstone Unit No. 2 (MP2) Station Air (SA) systems and permanently installed an existing SA cross-tie to the Millstone Unit No. 3 (MP3) SA system. A bypass around excess flow check valve 2-SAS-EFV-20 was installed for increased flow to/from the MP2/MP3 cross-tie for high demand conditions.

Reason

With the decommissioning of MP1, MP2 lost the capability to have backup SA supplied from the MP1 SA system via an existing MP1/MP2 cross-tie. In addition, the buildings that were supplied by the MP2 SA and are part of the MP1 decommissioning will be "cold and dark" which may cause active system piping/components to fail. The lines connecting the MP1 and MP2 systems and the MP2 line to the Solidification Building were cut and capped to separate from MP1.

The existing line containing a temporary spool piece that connects the MP2 and MP3 SA systems was permanently installed to enable a more rapid response to a loss of SA and for high SA demand conditions. The Excess Flow Valve Bypass was installed to allow increased flow during use of the MP2/MP3 cross-tie.

Summary

The modification has no direct or indirect impact on equipment important to safety. There is no change in the function of the SA system, including the backup system. The Margin of Safety of the protective boundaries has not been reduced and there will be no adverse impact on public health and safety.

Provide Nitrogen Supply to Sensor AE-7650B; Replace Sensors AE-7650A, AE-7650B and Analyzer AIT-7650; Add Oxygen Monitor AE-7651

S2-EV-00-0089 Rev. 0
DCN DM2-00-0585-00

Description

This change replaces obsolete online gas analyzer equipment in the Primary Sample Sink. AE-7650B, "Reactor Coolant System (RCS) Sample Hydrogen Sensor," AE-7650A, "RCS Sample Oxygen Sensor," as well as AIT-7650, "RCS Sample O2/H2 Analyzer," were replaced. The replacement hydrogen sensor AE-7650B requires a nitrogen supply to purge the instrument chamber, hence, a nitrogen purge was added to the system. Oxygen monitor AE-7651 was added to the sample room to monitor for oxygen deficient atmosphere. All functional requirements associated with the original sample equipment, to provide online oxygen and hydrogen sampling of the RCS letdown heat exchanger outlet, remain unchanged.

Reason

The replacement of the obsolete equipment requires the addition of a nitrogen supply source that is required to continuously purge the hydrogen sensor AE-7650B while this sensor is online. This new equipment was added to the Piping and Instrumentation Diagram (P&ID).

Summary

This change to the RCS letdown sample equipment does not increase the probability of occurrence of an accident, increase the probability of occurrence of a malfunction of equipment, or increase the consequences of an accident evaluated previously in the safety analysis report. Besides the online oxygen and hydrogen sensors, no other plant equipment is affected by the modification. The replacement equipment, the addition of the nitrogen supply source, the addition of the oxygen monitor, and the changes to the associated P&ID are safe.

Replacement of Millstone Unit No. 2 Emergency Diesel Generator (EDG) Underground Fuel Oil Tank

S2-EV-01-0004 Rev. 0
DCR M2-00031 Rev. 0

Description

This change replaced the existing fuel oil delivery system to the diesel oil supply tanks with a new fuel oil delivery system, increasing the available volume of fuel oil. The existing tank, associated pumps and instrumentation were abandoned in place to be removed under another design change. The new tank and equipment perform the same functions as the existing tank and stores the required fuel for the EDGs. The existing tank and equipment was non-safety related and non-seismic, with the exception of the power being supplied to the pumps from a vital source through Non QA cable. The new system also is non QA with QA pump power, but also provides for a Seismic piping connection on the exterior of the EDG building to supply diesel fuel oil to the Diesel Oil Supply Tanks under specified conditions. The new system provides the same functions as the old system. The new above ground tank is inside a dike to contain any leakage or spillage, and will have non-QA heating pads and non-QA heat tracing to maintain the fuel oil above 40 degrees Fahrenheit. The power supplied to the heating pads and heat tracing is from a non-vital source through non-QA cables.

Reason

The existing underground storage tank reached the end of life and needed to be removed/remediated to comply with Department of Environmental Protection requirements.

Summary

This change was installed to the current standards, which meet or exceed the original standards and enhance the operation to the EDGs. The availability of the existing Unit No. 2 EDGs, other than a short Limiting Condition of Operation (LCO) period to perform testing, is not affected by this change. The level of oil in the Diesel Oil Supply Tanks were not affected during the implementation of the connection to the new tank, and the level was maintained above the minimum level required, maintaining the integrity and ability of the EDG to operate.

This design change 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. It does not degrade the margin of safety as defined in the basis for any technical specification.

Reactor Building Closed Cooling Water (RBCCW) Pressure Transient Mitigation Modifications

S2-EV-01-0008 Rev. 0
DCR M2-01005 Rev. 0

Description

The modifications include the following:

- installation of one 3-inch orifice in each piping line 8"-HBD(B)-115 between the RBCCW Surge Tank discharge valves 2-RB-1A and 2-RB-1B and their associated 20" return headers,
- change RBCCW Low Suction Pressure Trip Setpoint from 15 psig decreasing to 13 psig decreasing and increase the time delay from 5 seconds to 10 seconds,
- replacement of Sample Cooler X-64/X-65 thermal relief valve 2-RB-322,
- removal of relief valve gags currently installed in the remaining 39 RBCCW thermal relief valves.

Reason

The installation of the orifice plates dampen the pressure spikes currently seen during Loss of Normal Power (LNP) type testing to completely eliminate the lifting of these thermal relief valves during these and similar such pressure transients. This also provides reasonable assurance that the relief valves would not lift during actual LNP or Loss of Coolant Accident (LOCA)/LNP scenarios. To ensure the installation of these orifices do not result in a pump trip on momentary low suction pressure, the trip setpoint is being lowered to provide additional margin for both the LNP and LOCA/LNP cases. The installation of the orifice plates will enable all relief valve gags in the RBCCW system to be removed. The upgrade of 2-RB-322, which is an obsolete Anderson-Greenwood relief valve, will improve its reliability and ensure leak tightness. These modifications eliminate significant operator burdens and resolve the root cause of relief valve lifting during pressure transients while improving overall system reliability.

Summary

The addition of orifice plates in the surge tank to pump suction lines, the decrease in the suction pressure trip setpoint and increase in time delay, as well as the upgrade of relief valve 2-RB-322 and subsequent removal of relief valve gags does not increase the probability or consequence of an accident or malfunction of equipment important to safety. In addition, it does not introduce any malfunctions or accidents of a different type than previously evaluated. The margin of safety as defined in the Technical Specifications is also not impacted.

Replacement of Check Valves 2-CH-432, 2-CH-433, 2-CH-328, 2-CH-331 and 2-CH-334

S2-EV-01-0009 Rev. 0
M2-00025 Rev. 0

Description

This change replaced the existing piston check valves 2-CH-432 and 2-CH-433 with nozzle check valves which are located on the charging lines to loops 2A and 1A of the Reactor Coolant System (RCS), respectively. It replaced the existing piston check valves 2-CH-328, 2-CH-331 and 2-CH-334 with nozzle check valves, which are located on each of the charging pump discharge lines. The existing ½ inch spool piece between the check valves and the downstream isolation valves were removed to accommodate the new check valve configuration. Temporary freeze seals downstream of 2-CH-432 and 2-CH-433 were installed to isolate RCS during the removal and replacement activities.

Reason

Internal inspection indicated wear in the check valves 2-CH-432 and 2-CH-433 due to chattering of the piston from low flow velocities. There is a potential for the pistons in these check valves to stick open. Piston check valves were oversized for this application and hence required replacement. Check valves 2-CH-328, 2-CH-331 and 2-CH-334 on the discharge line of the charging pumps were also piston check valves. There was also evidence of check valves slamming contributing to damage of the check valve as well as the upstream and downstream components. Due to the uniqueness of the design, the nozzle check valves would minimize slamming due to slower disc motion. This replacement resulted in a Piping & Instrumentation Diagram (P&ID) change due to the replacement requiring elimination of the .05" spool piece between the check valve and isolation valve on each charging pump discharge header. Valves 2-CH-432 and 2-CH-433 cannot be isolated from the Reactor Coolant System when performing the removal and replacement activities. In order to prevent the reactor coolant from spilling out the downstream side of the check valves, dual temporary freeze seals were installed prior to removal and maintained in place until the new valves were installed and satisfactorily tested.

Summary

This evaluation addresses the impact of the temporary freeze seals and the change to the P&ID.

The modified piping arrangement does not change the piping layout or the function of the piping. It minimizes the weld joints while maintaining the pressure boundary integrity. The minimum required flow rates required per the accident analysis of the Final Safety Analysis Report are not affected by this change. The temporary freeze seals were installed on the 2-inch charging lines downstream of the charging header check valves 2-CH-432 and 2-CH-433 during modes 5 and 6. The performance of the freeze seals did not impact the margin of safety as defined in the basis of any technical specification. The contingency actions ensured that the borated water flow path would be operable at all times. The margin of safety inherent in the physical protective boundaries was not adversely affected.

Modify Reactor Head/Pressurizer Vent Tubing

S2-EV-01-0015 Rev. 0
MMOD DM2-00-0752-00

Description

Sections of the existing ½ inch tubing in the Head Vent System and the Pressurizer Vent System were replaced with ¾ inch tubing.

Reason

Level measurement errors occurred during Reactor Coolant System fill and vent evolutions due to the vent tubing restricting the venting flow. The existing tubing reduces down to ½ inch in places, but a majority of the existing tubing vent paths are ¾ inch. This modification replaced the ½ inch sections of Head Vent and Pressurizer Vent tubing with ¾ inch tubing.

Summary

This change replaces the existing tubing with larger tubing for the Head Vent and Pressurizer Vent systems. All materials used were installed to the current standards, which meet or exceed the original standards and enhance the operation of the system. The tubing is adequately supported. This modification does not introduce any new malfunctions or increase the probability of occurrence or the consequence of an accident or malfunction previously evaluated. The change does not increase the risk to the public.

Emergency Diesel Generator (EDG) Lube Oil Temperature Instrument Upgrade

S2-EV-01-0017 Rev. 0
DM2-01-0085-01

Description
The scope of this evaluation is the changes covered by DM2-00-0085-01 and DM2-01-0085-01. The modification replaced the existing Allen-Bradley EDG Lube Oil Temperature Alarm Switches TS-8799 and TS-8800 with Allen Bradley Temperature Switches that have remote sensing bulbs. The thermowell was also replaced.

This modification replaced eight existing switches and four indicators with eight Omega indicating temperature switches. It changed the lube oil high temperature alarm set point from 185° Fahrenheit (F) to 195° F. The high temperature switch setpoint for EDG protective shutdown was not changed.

Reason
Since the lube oil temperature switches were mounted directly on the lube oil pipe, they were subject to the same vibration as the pipe. As a result, many vibration induce failures occurred in the past. This modification caused the switch and electrical enclosure to be mounted on a rigid steel structural member and the temperature sensing bulb to be mounted in a thermowell in the pipe. The sensing bulb is connected to the switch housing by a flexible metal capillary. Therefore, EDG reliability was improved and the switch housing with contacts and adjustments is more accessible for maintenance and calibration.

The existing installed generator bearing temperature alarm switches and indicators were obsolete and all of the spares had been used for replacements. New indicators provide full-time digital temperature indication. (Old indicators were sweephand analog indicators that had to be manually selected to display values.) The existing setpoint for the generator bearing alarm actuation of 185° F resulted in spurious alarms during the summer months. The alarm response is for the operators to monitor the temperature using hand held instrumentation, which required increased visits to the diesel room during the performance of the surveillance.

Summary
This modification changed Final Safety Analysis figure 8.3-5, "A" Diesel Generator (DG) Data Sheet OPS Form 2346A-004, "B" DG Data Sheet OPS Form 2346A-006 and ARP 2591/AB. The change to the type of switch impacted the figure, thus requiring a safety evaluation. This modification does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, as previously evaluated in the Safety Analysis Report. Additionally, this modification does not degrade the margin of safety as defined in the basis for any technical specification.

Remove Precise Control of Reactivity System (PCRS)

S2-EV-01-0021	Rev. 0
DCR M2-01007	Rev. 0

Description

This change removed the Millstone Unit No. 2 PCRS. The mechanical portion of the system was removed downstream of 2-CH-915 and upstream of 2-CH-091, and blank flanges were installed at both locations. Instrument Air supply to 2-CH-936 was disconnected downstream of 2-IA-228, and a plug installed. Electrical controls and indications at Control Room panels C04 and C01X were removed and blanks installed on the panels.

Reason

The PCRS was installed to maintain reactor power at one hundred percent by providing pure make-up water to the Charging Pump suction header. This diluted the boron concentration to compensate for fuel burn-out. This system is no longer required due to the upgrade of the Boric Acid/Primary Makeup Water (PMW) Blending system, and is a long-term system tag-out. Malfunction of this system could cause boron dilution, and therefore a reactivity mismanagement event. Additionally, this system imposed an operator burden by providing a distracting Safety Injection Actuation Signal valve indication on Control Room panel C04 and unnecessary controls and indications on Control Room panel C01X.

Summary

The PCRS, a sub-system of the Chemical and Volume Control System, is no longer needed due to the upgrade of the Boric Acid/PMW Blending System. As a conservative initiative to remove a potential boron dilution flowpath and eliminate the operator burden associated with obsolete controls and indications on Control Room panels, the PCRS is being removed. Removal of the PCRS does not adversely impact operation of any system, structure or component. Removal of the system improves plant safety. This activity does not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety as previously evaluated.

2-SI-469 Relief Valve Replacement and Orifice Installation

S2-EV-01-0022 Rev. 0
DCR M2-01008 Rev. 0

Description

The existing Class 1 relief valve 2-SI-469 was replaced. The existing relief valve discharged 54 gpm at 300° Fahrenheit (F) and 300 psig. An orifice was installed upstream of 2-SI-469 to make a class break. The orifice was designed to pass less than 40 gpm at 300° F and 300 psig, thus ensuring the make up capability of the charging pump was not compromised. The new relief valve was installed as a Class 2 valve, and discharges less than 20 gpm at 300° F and 300 psig.

Reason

The existing 2-SI-469 was an ASME Section III Class 1 Relief valve. In order to test this valve, a freeze plug needed to be installed. The test frequency of the existing valve was every outage since this was the only valve in this valve "family." This valve was old and no longer in production so a direct replacement would have been difficult to obtain if a valve failure occurred. By installing an orifice, a class break between Class 1 and Class 2 was made in accordance with ANSI/ANS 51.1. The relief valve could then be installed as a Class 2 valve and since it is similar to other valves in Unit No. 2, a larger valve family exists. The larger the valve family, the lower the testing frequency.

Summary

This change replaced the existing relief valve with a new relief valve and orifice that increases the reliability of the relief valve and limits the loss of Reactor Coolant in the case of a malfunction. The new valve and orifice is better suited to the design conditions than the existing valve. Therefore, the change does not create the possibility of a malfunction of a different type than any previously evaluated. This modification improves the pressure boundary integrity and decreases the probability of a loss of coolant accident. All material, piping and valve was installed to the current standards, which meet or exceed the original standards and enhance the operation of the system.

Adding Disconnect Switches to Appendix R Motor Operated Valves (MOVs) 2-MS-65A, 2-MS-65B, and 2-MS-202

S2-EV-01-0023 Rev. 0
DCR M2-00030 Rev. 0

Description

This change modified the Appendix R MOVs 2-MS-65A, 2-MS-65B, and 2-MS-202 to eliminate an Appendix R requirement to remove starter coils from their applicable Motor Control Center (MCC) circuit breakers and install new keyed disconnect switches.

Reason

The Appendix R requirements directed a Plant Equipment Operator (PEO) or electrician to remove starter coils from their applicable MCC circuit breakers. This modification accomplishes the same function without the requirement to remove the starter coils.

Appendix R MOVs 2-MS-65A, 2-MS-65B, and 2-MS-202 were previously identified as three valves that have a potential to be susceptible to spurious operation due to fire induced control cable shorts (hot shorts). The function of the new disconnect switches eliminates the need to remove starter coils and still complies with the Appendix R requirements. The new disconnect switches ease the burden placed on the PEOs.

Summary

The new disconnect switches provide assurance that valves 2-MS-65A and 2-MS-65B will remain closed during or after a fire, so that after the main steam isolation valves (MSIVs) are opened, the bypass valves can be closed in mode 3 or 2 while ascending into mode 1. The bypass valves have no automatic opening feature with the disconnect switches in the "off" or open position, however, if the bypass valves are open (disconnect switch in the "on" position) they will automatically close upon receipt of a main steam isolation signal.

The new disconnect switch provides assurance that valve 2-MS-202 will remain open during or after a fire, so that steam will be available to the Auxiliary Feedwater pump turbine.

Reduction of the operator burden by re-installing the MOV operating coils (open/close) eliminates the need for tools to return the MOVs to operation. The operation of the disconnect only requires the use of a key and the closing of a handle. The opening of the power circuit to the valves at the disconnects ensures that the valves will not change position (i.e. they will remain in their required safety position) due to hot shorts and thus safely supports Appendix R requirements.

Replacement of Millstone Unit No. 2 (MP2) Feedwater Pump Speed Control System

S2-EV-01-0026 Rev. 1
DCR M2-99044 Rev. 0

Description

This modification replaced the existing mechanical-hydraulic Steam Generator Feedwater Pump (SGFP) speed control system with a digital control system. This replacement involved removing or disabling most of the existing hydraulic governor components. The major change was the replacement of the speed controller that allows operation of the system in automatic rather than the current manual operation.

Reason

The existing Feedwater Pump Speed Control (FPSC) was unable to provide stable automatic speed control. Since initial plant startup, the feedwater pump speed has been controlled manually from the Millstone Unit No. 2 control room. Manual control of the feedwater pumps is an operator burden because it requires a dedicated operator during plant transients.

Summary

This change does not impact containment analysis and design basis limit for containment barrier is not exceeded or altered. Thermal/hydraulic analysis is not adversely impacted by this activity. Linear heat rate and Departure from Nucleate Boiling Ratio criteria is met. Existing assumptions on feedwater flow bound the replacement system and calculated environmental conditions in the turbine building remain valid. Fuel failure does not occur; moderate frequency event criteria is met. Reactor Coolant System pressure boundary design basis limit is not exceeded and feedwater behavior for replacement design is bounded by existing analysis assumptions. Feed pump trip on Main Steam Isolation is not changed.

Reactor Coolant Pump (RCP) Leak Detection Special Procedure and Technical Specification Basis Change Regarding Reactor Building Closed Cooling Water (RBCCW) Header Independence.

S2-EV-01-0031	Rev. 0
SPROC ENG00-2-09	Rev. 0
LBDCR 2-18-01	Rev. 0

Description

Sampling of the RBCCW System has identified the potential for an intermittent leak of reactor coolant into the RBCCW System. The Special Procedure (SPROC) assists in determining whether a Reactor Coolant System (RCS) to RBCCW leak by the Reactor Coolant Pump (RCP) seal cooler or thermal barrier exists and if so, which RCP seal cooler or thermal barrier is leaking. This is accomplished by the isolation of the RBCCW pump mini-flow lines, alignment of the radiation monitor to a single RBCCW train, and swapping of the RBCCW cooling of the RCPs from one RBCCW train to the other in a systematic manner. During the valve manipulations required for swapping of the cooling trains, the two RBCCW headers are no longer independent as required by the Technical Specifications. For these brief periods of time, the appropriate Technical Specification Action Statement is entered. Clarification of the Technical Specification Bases identifies the appropriate action statement to enter when header independence is lost.

Reason

The SPROC was developed for assistance in determining if an RCP seal cooler or thermal barrier is the source of RCS leakage and, if so, which RCP seal cooler or thermal barrier is leaking. To support this SPROC, the Technical Specification Bases for the RBCCW and Service Water Systems were clarified to identify the appropriate action statement to enter when RBCCW header independence is lost.

Summary

The proposed activity was reviewed as it impacted accidents and malfunctions previously evaluated. It was determined that the implementation of the SPROC would not adversely impact the operation of any equipment that the RBCCW supports for normal operation and accident conditions. In addition, the clarifications made to the Technical Specification Bases for the RBCCW and Service Water Systems do not change the accident analyses of record nor affect the design functions of any system, structure or component. The proposed changes do not increase the likelihood of occurrence or the consequences of an accident or malfunction.

Reload Design for Millstone Unit No. 2 (MP2) Cycle 15

S2-EV-02-0003 Rev. 0
DCR M2-01013 Rev. 0

Description

This modification reconfigured the MP2 core for Cycle 15 operation. The core management strategy for Cycle 15 is very similar to that for Cycle 14. Following the completion of Cycle 14 operation, five (5) Batch N assemblies, fifty-six (56) Batch P assemblies, and twenty (20) Batch R assemblies were discharged from the reactor core. The Cycle 15 core contains one (1) Batch N assembly, sixty-four (64) Batch R assemblies, seventy-two (72) Batch S assemblies, and eighty (80) fresh Batch T assemblies. The single Batch N assembly was reinserted. The Fresh (Batch T) fuel assemblies include the High Thermal Performance (HTP) spacer and the FUELGUARD lower tie plate. Cycle 15 is the first implementation of these design features at MP2.

Related Design Basis and Licensing Basis document changes were revised so that these documents are consistent with the Cycle 15 plant design.

Reason

This change supports the new fuel product for Cycle 15, the Cycle 15 core design, and Cycle 15 operation. This change was necessary in order to allow Cycle 15 operation and the continued production of electricity by MP2.

For Cycle 15, eighty (80) fresh Batch T fuel assemblies and one (1) twice burned Batch N fuel assemblies were added to the core and eighty-one (81) reactive (Batch N, P, and R) fuel assemblies were discharged to the Spent Fuel Pool. A new core and "cycle" of operation was necessary because the Cycle 14 core had reached a burnup value such that there was insufficient reactivity to operate efficiently.

Summary

Cycle 15 operation with a more negative Moderator Temperature Coefficient (changed from - 28.0 pcm/° Fahrenheit (F) to -32.0 pcm/° F) and an increased Linear Heat Rate (changed from 14.6 kW/ft to 15.1 kW/ft) are acceptable and do not require prior NRC approval. Additionally, the use of revised methodologies for non-Loss of Coolant Accident (LOCA) analyses, revised methodologies for Small Break LOCA analysis, and the use of the High Thermal Performance correlation for Departure from Nucleate Boiling analyses are determined to be acceptable. Also, it is concluded that there is no impact caused by introducing the FUELGUARD Lower Tie Plate on long-term coolability due to debris or insulation material entering the lower plenum of a reactor vessel.

The changes do not adversely affect the accidents and malfunctions previously evaluated. Additionally, the changes do not create a new type of event not previously evaluated and they do not adversely impact the fission product barriers as described in the Final Safety Analysis Report.

Millstone Unit No. 2 Reactor Vessel Head Nozzle Repairs

S2-EV-02-0010 Rev. 0
DCR M2-02001 Rev. 0

Description

This design change was for repairs to those reactor vessel head nozzles that exhibited flaw indications that could not be removed by grinding. None of the indications involved through wall flaws. The changes introduced by the repair of the Control Element Drive Mechanism (CEDM) nozzle included the use of ambient temperature temper bead weld, leaving a remnant of the existing J-groove weld where flaw characterization, per ASME XI, is not readily possible and flaw removal is not practical. The change resulted in exposure of low alloy steel to primary coolant in the annulus area just below the Inner Diameter Temper Bead repair weld and above the existing J-groove weld and buttering remnant, potential exposure of Reactor Vessel head manganese sulfide laminations, and change in minimum inner diameter of the CEDM nozzle and guide funnel. Relief request RR89-34 for the temper bead weld technique and for use of an evaluation for embedded flaws, was submitted to the NRC and is therefore not part of this 50.59 evaluation.

Reason

The inspections of Alloy 600 control rod drive mechanism nozzles at Oconee and Arkansas Nuclear One plants have identified unexpected circumferential cracking due to primary water stress corrosion cracking. This prompted an inspection of the Millstone Unit No. 2 reactor vessel head penetrations. The inspection revealed indications in the CEDM nozzles that were repaired.

Summary

The changes being evaluated are the result of repairs to the Millstone Unit No. 2 CEDM nozzles that had indications of flaws. The small break Loss of Coolant Accident (cold leg break) remains bounding over leaks in the CEDM nozzle area. The repair was performed in accordance with ASME Section III and Section XI, Class I requirements to assure that GDC 14 requirements are maintained. Therefore, a postulated break in the CEDM pressure housing causing a rod ejection event and the associated consequences remain bounded.

Another evaluated change is the dimension of the inner diameter in the replacement lower nozzle and guide. No new malfunctions associated with the failure of control rods to be inserted or withdrawn properly are being introduced. The assurance that the control rod will perform when required is provided by Technical Specification (TS) 3.1.3.4 and the rod drop time surveillance testing that is performed, as well as TS 3.1.3.1 that requires Control Element Assemblies to be operable and not immovable as a result of excessive friction or mechanical interference. The repaired CEDM nozzle configuration is such that the effect of the anti-ejection on the repaired CEDM nozzle will not be different from the original CEDM nozzles. This device is not credited in any of the accident analyses.

Docket Nos. 50-245
50-336
50-423
B18900

Attachment 3

Millstone Power Station, Unit No. 3

10 CFR 50.59 Report for 2002

Index for Millstone Unit No. 3
Changes to the Facility - 2002

Index for Millstone Unit No. 3
Changes to the Facility - 2002

Generic Safety Evaluation for Non-Design Basis Piping and Instrumentation Diagram (P&ID) Changes

S3-EV-97-0438 Rev. 0
M3-EV-970259 Rev. 2

Replace Auxiliary Boiler Fuel Oil Strainer 3FOA-STR2A
DCN DM3-00-0642-99

Replace Auxiliary Boiler Fuel Oil Strainer 3FOA-STR2B
DCN DM3-00-0643-99

Description

The scope of the change bounded by this Safety Evaluation (SE) is contained in Engineering Evaluation M3-EV-970259, Revision 2. The changes affect the symbology, general information and component identification depicted on the P&IDs. The changes were evaluated as group types. Each change is based upon a separate technical justification that includes a validation against original design basis documents. This SE only addresses changes to the facility as found on P&IDs provided as figures in the Final Safety Analysis Report.

Reason

The assessed changes described in Engineering Evaluation No. M3-EV-970259, Revision 2, are considered to be either administrative or Maintenance Support Engineering Evaluation (MSEE) Design Change Notices (DCNs). In all cases, the listed changes are technically justified by the original design basis, i.e. existing documentation is available that supports the conclusion of each change and does not conflict with the description of the equipment, its operation or technical requirements described elsewhere in the Safety Analysis Report.

Summary

The types of P&ID changes, evaluated as group types, are bounded by Engineering Evaluation No. M3-EV-970259, Revision 2, and do not introduce operational changes to any plant system, structure or component. They are the type of DCN changes described and bounded within the scope of the Design Control Manual that do not include design basis changes. The changes do not introduce any new malfunctions or increase the probability of occurrence or the consequence of an accident or malfunction previously evaluated. The changes do not increase the risk to the public.

Carbon Dioxide (CO_2) Discharge Change from Automatic to Manual Initiation for Cable Spreading Room

S3-EV-99-0092 Rev. 0
DCR M3-99030 Rev. 0

Description

This modification substituted human/manual intervention for automatic actuation of the CO_2 Fire Protection System for the Cable Spreading Room. The original configuration had the ball valve local to the Chemtron Panel for the Cable Spreading Room aligned in the open position. This change re-aligned the ball valve such that the valve will be closed under the system's normal configuration. The CO_2 System initiation sequence still begins only if both zones of cross-zones of detection are in alarm or one of the two manual discharge stations have been placed in "Discharge." This modification now requires the ball valve to be opened before the timing sequence begins.

Reason

The primary reason for this change is to enhance personnel safety. CO_2 can be lethal to personnel in the area of a discharge, as well as posing a threat to personnel in adjacent areas. As a result of an inadvertent CO_2 initiation, the existing CO_2 Fire Protection System was locked out from operation, with a continuous fire watch posted as a compensatory measure.

Summary

Changing the Cable Spreading Room CO_2 protection system from automatic to manual actuation does not represent an increase in the risk to the health and safety of the public. Fire safety is assured by the presence of operable fire detectors, a dedicated on-site fire brigade and the capability to activate the CO_2 system in a timely manner, when needed, in the event of an actual fire.

The Cable Spreading Room CO_2 Fire Protection System is not safety-related. This system is not required to shutdown the reactor or mitigate the consequences of postulated accidents, nor to maintain the reactor in a safe shutdown condition. This change does not increase the probability of previously evaluated malfunctions.

Separation of Millstone Unit No. 1 (MP1) Auxiliary Steam System From Millstone Unit No. 2 (MP2)

S3-EV-00-0011 Rev. 1
DCR M3-99042 Rev. 0

Description

This change modified the MP2 Auxiliary Steam System to accept steam from Millstone Unit No. 3 (MP3). It provided a heating steam/process steam supply from the MP3 Auxiliary Steam System to the MP2 Auxiliary System and performed post modification testing of the system.

The system modification entailed separating the services provided by the MP1 Auxiliary Steam System with respect to supplying a source of steam to the MP2 Auxiliary Steam System. The redesign of the system utilizes the existing MP3 Auxiliary Steam System and the two existing MP3 Auxiliary Steam System boilers to provide auxiliary steam for MP3, MP2 and to the site fire water storage tanks, for freeze protection.

Reason

The decommissioning of MP1 coupled with the retirement of the MP2 auxiliary reboiler required that the MP2 Auxiliary Steam System be modified to accept steam from MP3. The decommissioning of MP1 requires that Structures, Systems and Components (SSC) associated with MP1 yet common to MP2 and/or MP3 are separated. This allows independence from MP1 such that abandonment or demolition of any MP1 SSC will not adversely affect the safe operation of either MP2 or MP3.

Summary

The design characteristics associated with this modification do not degrade design specifications for material and construction practices, nor challenge any safety systems during normal operational transients. The Auxiliary Steam System provides heating steam to the fire water storage tanks. Connection of the Auxiliary Steam system to the fire water storage tanks is made in accordance with specifications and ASME Section IX. The steam and condensate return lines from the fire water storage tank continue to be electrically heat traced in accordance with specifications. The function of the MP2 and MP3 Auxiliary Steam System and SSC receiving heating/process steam are not altered as a result of this modification. There are no protective functions associated with the operation of the Auxiliary Steam System and no effect on equipment important to safety.

Testing of the system does not introduce a reduction in personnel safety, nor place the plant in a configuration that will jeopardize plant operation.

Replace Auxiliary Boiler Fuel Oil Strainer 3FOA-STR2A

S3-EV-00-018 Rev. 0
DCN DM3-00-0642-99

Description
This change replaced the Auxiliary Boiler fuel oil duplex strainer, 3FOA-STR2A, manufactured by Zurn, with a Hayward Industrial Products, Model 50 duplex strainer. Minor piping and instrument tubing modifications were required to suit drain line fit up and physical dimension changes of the strainer. The strainer is also provided with ¼inch chamber vent connections in lieu of vent cocks; the Zurn strainer chambers were vented using $^1/_8$ inch vent cocks which were furnished with the strainer.

Also, a field walkdown of the existing fuel oil strainer and associated piping found configuration discrepancies between the as-installed piping and the Piping and Instrumentation Diagram (P&ID). A reconciliation was performed and the P&ID updated to eliminate any inconsistencies.

Reason
Fuel Oil duplex strainer 3FOA-STR2A, manufactured by Zurn Industries, Inc., had leakage problems between strainer chambers and was not repairable. This strainer was no longer available as the Zurn strainer line was bought out by Hayward Industries.

Summary
Installing vent lines and associated valves to vent the chambers of the Hayward replacement strainer, and revising the P&ID and Final Safety Analysis Report (FSAR) figure to include these vent lines does not introduce an operational change to any plant system, structure or component. All piping and tubing modifications associated with the strainer replacement meet existing plant specifications. Corrections/enhancements made to the P&ID and FSAR figure are considered administrative. Installation of the vent lines provides the operator with a suitable method of venting the chambers of the Hayward replacement strainer as the existing $^1/_8$ inch vent cocks furnished with the Zurn strainer perform. The design of the replacement strainer does not alter any basic strainer design parameters or functions or change any strainer operation or performance requirements.

Motor Driven Feedwater (FW) Pump (3FWS-P1) Impeller Modification

S3-EV-00-0026 Rev. 0
DCR M3-00003 Rev. 0

Description

This modification installed a new impeller designed by Ingersol Dresser Pump (IDP) in the Motor Driven FW Pump. The new impeller incorporates a number of improvements such as: an increase inlet vane/fillet thickness (3/16 inch to 3/8 inch); use of the investment casting process; use of a bias wedge/leading edge hook design; an optimized inlet vane angle; and inlet vane tip diameter reduction from 10¾ to 10 inches. The new impeller uses a 0.003 - 0.006 inch impeller/shaft interference fit.

The new IDP impeller is a replacement to the original equipment manufacturer, Byron Jackson (BJ), supplied impeller. Three IDP replacement impellers were procured. The IDP impeller is intended as an acceptable replacement impeller for use in all three FW pumps (3-FWS-P1 and 3-FWS-P2A/B). Additionally, this change increased the pump trip process setpoint from 240 psig to 250 (10 psig increase) for 3CNM-PS74A.

Reason

The Motor Driven FW pump impeller failed twice in 1998 due to high cycle fatigue caused by high dynamic vane loads occurring during low flow conditions. This modification reduces hydraulic instability, vibration, and inlet vane stresses during FW pump low flow operating conditions.

Summary

This modification does not introduce undesirable system interactions/hydraulic instabilities. It has no effect on the accident initiating events, accident mitigation, protective boundaries or radiological consequences. The new impeller incorporates a number of improvements that provide better pump operation at low flow operating conditions. These changes result in a reduction in hydraulic instability, vibration and inlet vane stresses during low flow operating conditions. The changes improved FW pump reliability by making impeller failure less likely due to high cycle fatigue.

The hydraulic performance is very similar to the existing impeller, except the decrease in the inlet eye diameter results in an increased net positive suction head (NPSH) requirement at the design flow. There is no substantive change in the FW system operating pressures or flows. The increase in FW pump required NPSH and the FW pump suction pressure trip increase will have no adverse effect on FW system reliability or the probability of a loss of FW accident.

Feedwater Isolation Valve (FWIV) Accumulator Modification

S3-EV-00-0035 Rev. 0
DCR M3-00007 Rev. 0

Description

This change increased the allowable operating pressure range of the FWIV accumulators. Specifically, the valve accumulator maximum system operating pressure was increased from 5200 to 5400 psig. The existing maximum valve accumulator system pressure remains at 5400 and the low end alarm and limit remains at 4750 psig and 4650 psig, respectively.

In addition, a manifold with a relief valve with a relief setpoint of 5400 psig was added to each accumulator system. Each relief valve has a reseating pressure of approximately 5100 psig. The relief valves maintain the accumulator pressure within acceptable system pressure limits. An orifice with a small bore has been located in the relief valve flow path to the reservoir to prevent draining of the system in the event that the relief valve fails in the open position.

Reason

This change eliminates/reduces the Operator burden associated with maintenance of the FWIV accumulator system pressure and eliminate/reduces entrance into the associated Limiting Condition of Operation (LCO). Implementation of this modification also reduces the potential for a reactor trip by reducing the potential of mispositioning a FWIV during a manual FWIV hydraulic pressure reduction evolution.

Summary

The increase in the operating band and the addition of the relief valve for each accumulator system is anticipated to eliminate/reduce the entrance into the associated LCO by making the system more able to accommodate/automatically adjust to environmental temperature fluctuations.

The change improves the "availability" of the system with respect to the maintenance rule. The increase in reliability of the FWIVs has a positive impact on the health and safety of the public by reducing the probability of a loss of feedwater accident/transient.

The increase of the upper end of the hydraulic pressure operating range and installation of a manifold containing a relief valve does not increase the consequences of a malfunction of equipment important to safety, nor does it create the possibility of a malfunction of a different type than any previously evaluated. The modification maintains the closure time of the FWIVs and maintains the assumptions of accidents with respect to the FWIVs. The function of the FWIVs remains unchanged.

Millstone Unit No. 2 / Millstone Unit No. 3 (MP2/MP3) Station/Service Air Cross Tie

S3-EV-00-0050 Rev. 0
DCR M2-99061 Rev. 0

Description

This change separated Millstone Unit No. 1 (MP1) from the MP2 Station Air (SA) systems and permanently installed an existing SA cross-tie spool piece to the MP3 SA system. A bypass around excess flow check valve 2-SAS-EFV-20 was installed for increased flow to/from the MP2/MP3 cross-tie for high demand conditions.

Reason

With the decommissioning of MP1, MP2 lost the capability to have backup SA supplied from the MP1 SA system via an existing MP1/MP2 cross-tie.

The existing line containing a temporary spool piece that connects the MP2 and MP3 SA systems was permanently installed to enable a more rapid response to a loss of SA and for high SA demand conditions. The Excess Flow Valve Bypass was installed to allow increased flow during use of the MP2/MP3 cross-tie.

Summary

The modification has no direct or indirect impact on equipment important to safety. There is no change in the function of the MP3 SA system including the role as a backup system to the MP3 Instrument Air system. The Margin of Safety of the protective boundaries has not been reduced and there will be no adverse impact on public health and safety.

Fire Protection Deluge Waterspray System Deletion of Manual Initiation Capability Within Control Room (CR)

S3-EV-00-0073 Rev. 0
MMOD DM3-00-0342-00 Rev. 0

Description

This modification removed the manual initiation capability from the water deluge systems that could be manually initiated from within the Control Room via the Fire Protection Color Graphics Unit.

Reason

The primary reason for this change is to prevent recurrence of a false actuation of any of the waterspray systems that are controlled by the Color Graphics Unit. During the performance of routine surveillance testing associated with the "B" Reserve Station Service Transformer Fire Protection Water Deluge System, the "B" Main Transformer Water Deluge System falsely actuated. Trouble shooting identified the software within the Fire Protection Color Graphics unit to be the source of the problem. However, troubleshooting by in-house personnel and the supplier of the Color Graphics did not pinpoint the cause of the software problem. Therefore, confidence in the manual initiation capability associated with the Color Graphics unit was lost. The false actuation of these systems could result in an unnecessary plant shutdown or plant transient.

Summary

This change prevents false actuations of any of the affected waterspray systems as caused by the Color Graphics unit by performing both hardware changes as well as disabling the manual initiation capability of the Color Graphics screen. The change does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. It will not adversely affect any response to a fire event. The waterspray systems provide a means for limiting the consequences of a fire event through automatic as well as manual actions. The existing automatic protection capability, as well as the existing local manual initiation capability of the identified waterspray systems, are not impacted by the change.

The automatic water deluge fire protection system is not safety-related. This system is not required to shutdown the reactor or mitigate the consequences of postulated accidents, or maintain the reactor in a safe shutdown condition. Deleting the manual initiation capability from within the CR for the affected waterspray systems does not increase the consequence of a malfunction of equipment important to safety.

Removal of 3QSS-RO43

S3-EV-01-0018 Rev. 0
Temp. Mod. 3-01-006 Rev. 0

The above temporary modification, which reinstalled the piping on the Refueling Water Storage Tank (RWST) recirculation line without the orifice plate (3QSS-RO43), was previously reported as installed and has since been removed:

Establishing Maximum Letdown Using a Single Charging Pump

S3-EV-02-0011 Rev. 0
SPROC ENG02-03-001 Rev. 1

Description

This change revised Special Procedure (SPROC) ENG02-03-001 to establish a parallel path around the charging control valve, 3CHS*FCV121, by throttling open bypass valve 3CHS*VO54 such that the Reactor Coolant System (RCS) makeup flow could match the maximum letdown flow of 133 gpm. This was a one time only test performed in Mode 1 and required a single charging pump to be running with two letdown orifices (i.e., 45 gpm and 75 gpm) in service. Pressurizer level was monitored to verify level was automatically maintained within programmed level range.

Reason

A previous test performed for charging pump 3CHS*P3A (SPROC ENG02-3-001) demonstrated that pressurizer level could not be maintained at maximum letdown flow of about 133 gpm. During the test, the observed RCS makeup flow was 123.7 gpm. Review of test data concluded that pump 3CHS*P3A was performing as expected and that the flow path hydraulic resistance from the charging pump to the RCS was higher than expected. Establishing a parallel flow path around charging control valve 3CHS*FCV121 by throttling open a local bypass valve, 3CHS*V054, lowered the system's hydraulic resistance and thus increased RCS makeup capability.

If this test had been successful, operating procedures would have been revised to allow the single Charging System (CHS) pump configuration for maximum letdown. Using maximum letdown maximizes RCS purification so that activity levels are as low as possible prior to refueling outages. This test failed however, therefore, no procedures were changed.

Summary

This test was performed in Mode 1 and use of the letdown system was limited such that the sum of letdown flow, total RCS leakage, and Reactor Coolant Pump seal return did not exceed 148 gpm. This assured that the assumptions made for a postulated fire induced loss of all charging remained valid.

Makeup flow control could have required operator action during performance of the proposed SPROC, so plant operators were responsible to continuously monitor for any sign of deviation or malfunction, such that the possibility of an undetected malfunction was remote. In addition, the activity did not affect an operator's ability to mitigate pressurizer level perturbations and restore pressurizer level to the programmed level.

Divert Reactor Coolant System (RCS) Check Valve Leakage to the Primary Drains Transfer Tank (PDTT)

S3-EV-02-0019 Rev. 0
TMOD M3-02-010 Rev. 0

Description

This temporary modification diverts RCS check valve leakage from the "C" Accumulator to the PDTT using the High Pressure Safety Injection/Low Pressure Safety Injection (SIH/SIL) test piping.

Reason

RCS coolant was leaking past the "C" RCS Cold Leg Check Valve (3RCS*V107) and Accumulator Check Valve (3SIL*V19) into the "C" Accumulator. To eliminate the inleakage, a path of least resistance was established from the accumulator discharge piping to the PDTT through the SIH/SIL test piping.

Continued RCS leakage into the accumulator would ultimately reduce the accumulator boron concentration below the Technical Specification limit of 2600 ppm.

Summary

The change results in normally closed containment valves 3SIH*CV8871 and 3SIH*CV8864 being kept in the open position and normally open 3SIH-V986 being kept closed. The design basis of the plant already allows opening the containment isolation valves periodically to perform required Pressure Isolation Valve (PIV) testing in accordance with technical specifications. The change ensures that the leakage can be controlled and potential dilution of the "C" accumulator precluded.

The change does not affect the integrity or mode of failure of the Emergency Core Cooling System, the containment isolation system, or any other system or component. It does not impact the High Energy Line Break (HELB) or the Electrical Equipment Qualification (EEQ) program, or operator actions following an accident. The temporary modification is fully compliant with General Design Criteria and assures containment integrity following an accident. This change does not have an adverse effect on accidents and malfunctions evaluated; it does not have the potential to create a new type of event not previously evaluated. It does not have an adverse impact on fission product barriers or on evaluation methodologies described in the Final Safety Analysis Report.

Docket Nos. 50-245
50-336
50-423
B18900

Attachment 4

Millstone Power Station, Unit Nos. 1, 2 and 3

Annual Commitment Change Report for 2002

COMMITMENT CHANGES

Commitment Number	Original Commitment	Revised Commitment	
B17248 (COMCR 2-01-011)	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.	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. The surveillance to be performed on a refueling basis. (See letter B18435, DNC Inc., to U.S. NRC, "Millstone Nuclear Power Station, Unit No. 2, Revised Commitment Associated with Generic Letter 83-28," dated December 4, 2001.)	The Reactor Trip Circuit Breaker (RTCB) trip torque measurements have been performed at a quarterly and subsequently semi-annual basis at Millstone Unit No. 2. The RTCBs are also cycled on a monthly basis during the performance of the logic matrix testing. Millstone Unit No. 2 has been performing the RTCB trip torque test since 1984 and the data collected has shown no evidence of degrading conditions.
B17248 (COMCR 2-01-012)	Trip torque will be trended. This trending will be performed semi-annually.	Canceled.	Surveillance of the Reactor Trip Circuit Breaker (RTCB) trip torque at three month intervals was initially desirable to establish the maintenance interval for revitalizing bearing grease to attain the normal RTCB maintenance interval. Since required preventative maintenance (PM) intervals are a function of many complex factors, such as environment, temperature, obtained trip torque after maintenance, etc., a specific time interval for lubrication revitalization for all possible situations was difficult to estimate. Millstone Unit No. 2 has been performing RTCB trip torque testing since 1984 and the data collected had shown no evidence of degrading conditions.
B17248 (COMCR 2-01-013)	The on-line testing conducted for the reactor trip system includes trip bar torque testing. This surveillance will be performed semi-annually.	Perform Reactor Trip Circuit Breaker (RTCB) trip torque testing every refueling outage.	The RTCB trip torque measurements have been performed at a quarterly and subsequently semi-annual basis at Millstone Unit No. 2. The RTCBs are also cycled on a monthly basis during the performance of the logic matrix testing. Millstone Unit No. 2 has been performing the RTCB trip torque test since 1984 and the data collected has shown no evidence of degrading conditions.

COMMITMENT CHANGES

Commitment Number	Original Commitment	Revised Commitment	
B16113 (COMCR 2-02-007)	A design change to simplify maintenance activities will be implemented during the next refueling outage after 2R13, in which ESAS actuation cabinets 5 & 6 are down-powered. Due: 2R14	Closed. Transferred to the Corrective Action Program.	The design change will simplify maintenance activity during Modes 5 & 6 (i.e. when the plant is shut down). ESAS is not required to be operable in these modes. ESAS as currently configured without the proposed enhancement will perform its required safety function. The specific commitment originates from a License Event Report (LER) corrective action. LER corrective actions are not considered as Regulatory Commitments. Therefore, it is appropriate to transfer this item to the Corrective Action Program.

COMMITMENT CHANGE REFERENCES

2-01-011	B17248	Northeast Nuclear Energy Company, dated 9/30/98	Martin L. Bowling, Jr. to U.S. NRC	Millstone Nuclear Power Station, Unit No. 2, Conformance to Generic Letter 83-28 – Revised Testing Commitments
2-01-012	B17248	Northeast Nuclear Energy Company, dated 9/30/98	Martin L. Bowling, Jr. to U.S. NRC	Millstone Nuclear Power Station, Unit No. 2, Conformance to Generic Letter 83-28 – Revised Testing Commitments
2-01-013	B17248	Northeast Nuclear Energy Company, dated 9/30/98	Martin L. Bowling, Jr. to U.S. NRC	Millstone Nuclear Power Station, Unit No. 2, Conformance to Generic Letter 83-28 – Revised Testing Commitments
2-02-007	B16113	Northeast Nuclear Energy Company, dated 3/3/97	J. A. Price to U.S.NRC	LER 95-020-01, Automatic Actuation Of An Engineered Safety Feature During Maintenance