MAINE YANKEE ATOMIC POWER COMPANY

ATTACHMENT

MAINE YANKEE FACILITY CHANGES

1983

Pursuant: 10 CFR 50.59

2954L-SPC

PA 29-81 Auxiliary Boiler Smoke Detector Alarm

PA 29-81 provided independent control room indication of high density smoke in the auxiliary boiler system room.

Prior to this alteration, the auxiliary boiler room smoke alarm was one of several auxiliary boiler system indicators connected to a general alarm on the main control board. This plant alteration transferred the smoke alarm from the general alarm to a dedicated computer alarm. The remaining auxiliary boiler system indicators were not changed.

The auxiliary boiler system is a non-nuclear safety (NNS) system and is not relied on to prevent or mitigate the consequences of any design basis accident. This alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 1-82: Tap Line from Caustic Addition System

PA 1-82 added a tapline from the caustic addition system to the waste hold-up tank. The tapline permits manual adjustment of the pH level of the liquid waste in the waste hold-up tank.

Both the caustic addition system and the waste solidification system are NNS. The alteration does not increase the possibility of uncontrolled radioactive release or other accident, nor does it create the possibility of occurrence of a previously unreviewed accident. Therefore this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 11-82: Redundant High Level Switch For Spent Fuel Pool

PA 11-82 added a redundant high water level switch for the spent fuel pool. A second identical level switch was installed in parallel with the existing spent fuel pool level switch. Failure of a single level switch will not prohibit activation of the MCB warning alarm.

The spent fuel alarm system is non-nuclear safety. The addition of the switch enhances the warning capabilities of the present monitoring system and does not affect the operation of any other plant system. Therefore this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 14-82: Auxiliary Steam and Potable Water Supply for
New Maine Yankee (MY) Staff Building

PA 14-82 provided heating and domestic water to the new Main Yankee staff building.

Staff building heating water is supplied via the plant auxiliary steam system and condensate is returned to the auxiliary condensate system. Staff building drinking water is supplied via the existing plant domestic water system. All piping connections to the staff building exit the plant through the turbine building.

The steam, condensate and domestic water supply affected by this alteration are non-nuclear safety systems. This alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 15-82: Radio Frequency Monitoring for the Main Generator

PA 15-82 provided a radio frequency (RF) monitoring system for the main generator. The new system monitors radio frequency of the generator's stator windings for surveillance/performance purposes. Plant computer indication of RF monitoring system is provided. The computer alarms upon system shutdown and upon monitor level exceeding a preset level for greater than ten seconds.

The RF monitoring system is a non-nuclear safety system. The new system enhances the monitoring capabilities of the main generator and does not impact any plant safety system. This alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 33-82: Drumming Area Clean-Up

The waste effluent piping from the boron recovery evaporator (EV-1) and the liquid waste disposal evaporator (EV-2) were removed from the old drumming room and rerouted to the waste holdup tank. Following the removal of the piping, the drum roller and drum computer were also removed.

This alteration had no impact on the operability of boron recovery or liquid waste disposal systems. The alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 50-82: Dry Pipe Sprinkler System in Rad-Waste Compactor Building

PA 50-82 installed a dry pipe sprinkler system in the new low level rad-waste compactor building (LSA building). Water to the new system is supplied by the existing fire protection system. Air supply to charge the piping system is provided by the existing 100 psi instrument air system. A dry pipe valve separates system water supply from the air filled system piping. The valve trips open on a release of air pressure sufficient to overcome the pressure differential. Remote alarm indication is provided by the common alarm check valve on the existing wet pipe system upstream of the new dry pipe valve.

This installation does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety. Failure or inadvertent operation of this system will not cause failure of safety related components or impair the operability or design of the plant's existing fire protection system. Therefore this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 60-82: Auto-trip of RCA Monorail (MR-10)

PA 60-82 provided a means to automatically de-energize the bus bar on the RCA monorail, (MR-10) when RCA building roof hatch is opened. When moving equipment through the roof hatch, the monorail bus bar could be damaged if the equipment came into contact with it. To prohibit this occurrence, this alteration installed a limit switch mounted on the roof hatch which will deenergize the monorail bus bar upon opening the hatch.

Those systems are non-nuclear safety and this change only improves their safe operation. Therefore, this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 61-82: Smoke Detection for New Computer

PA 61-82 installed 5 smoke detectors in the new computer room with both control room and local annunciation. The activation of any of the new smoke detectors will annunciate alarms in the control room and at local alarm panels. Each panel has indication for four detectors with a light showing the location of each detector.

This change does not affect a nuclear safety classified system and improves the fire detection capability in this area. Therefore, this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 2-83: Chemical Spray Addition Tank Heater Breaker Change

PA 2-83 replaced the existing 40 amp circuit breaker for the chemical spray addition tank (TK-54) heater with 20 amp circuit breaker. The circuit cables for the tank heater are rated as 20 amp cables, therefore, the new circuit breaker is a more conservative configuration.

The tank heater is considered a non-safety related system. This alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 6-83: Uninterruptable Power Supply for Dimension 400 Phone System

PA 6-83 installed an uninterruptable power supply (UPS) cabinet and back-up battery power supply for the Dimension 400 phone system. Twenty lead oxide battery cells were installed in the #5 battery room. The uninterruptable power source supplies three phase AC power to its load. Upon interruption of the regular AC power source, the inverter is powered by the standby battery system. This switch is done without interruption of power to the load. Back-up primary power for maintenance of the UPS is provided through the manual transfer switch located in the telephone room and wired to the lighting panel LP-1 in the staff building.

The Dimension 400 phone system is classfied as a non-nuclear safety system and this change enhances its operability in the event of loss of service power. Therefore this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 7-83: Computer Monitoring of Feedwater Header

PA 7-83 provided computer monitoring of feedwater header pressure through the installation of a 2.5 ohm, .01% resistor in the feedwater header pressure instrumentation loop. The voltage across the new resister is monitored as an analog input to the plant computer. Computer alarm of decreasing feedwater header pressure will occur at 1090 psi by CRT output of a warning message.

This alteration enhances the warning capabilities of the feedwater header monitoring system, and it does not adversly affect any nuclear safety related equipment. Therefore this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 12-83: Raw Water Heat Exchanger Condensate Removal System

PA 12-83 installed a condensate handling unit on the raw water heat exchanger (E-50). A pump, motor, and receiver tank with a float switch were installed below the raw water level control valve (RW-A-14). Previously this system had been vulnerable to water hammer events. To prohibit this vulnerability, the new tank holds the raw water heat exchanger condensate until activation of the float switch causes the pumping unit to deliver the condensate to the auxiliary boiler condensate receiver tank.

The system affected by this change is non-nuclear safety related and the alteration is designed to further improve its safe operation. Therefore this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 13-83: Primary Aerated Vent Header Low Point Drain

PA 13-83 installed a drain valve and cap at the low point of the aerated vent header located just outside of the Aerated Drain Tank (ADT) cubicle. Prior to this alteration, the aerated vent header could have filled with water due to overfilling the aerated drain tank or excessive moisture in the aerated vent header. Also, the previous design did not allow gravity drain of the vent header. This alteration greatly reduced the complexity of the vent header drainage system. Fluid from the aerated vent header is directed to the aerated drain tank sump.

Because this system carries radioactive fluid, the drain connection was provided with the cap and valve to guard against single failure. The containment side of the vent is blank flanged to maintain containment integrity. This change does not increase the possibility of a radioactive release. Thus this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 15-83: Fuel Pin Transfer System

PA 15-83 installed a fuel pin transfer system for the spent fuel pool (SFP) area. The fuel pin transfer system consists of an indexing platform, fuel pin transfer tool, tool storage rack, fuel pin spacer blocks, storage tank and hydraulic system. The transfer tool is hydraulically operated and is equipped with a gripper for attachment to the fuel pins. The tool is supported by the manual indexing platform that is attached to the existing SFP work platform. The hydaulic system is free standing, operates using primary water as a hydraulic fluid and has a 200 gallon st[illegible] tank. Also included in this alteration was the fabrication and inst[illegible]ion of the fuel pin spacer blocks. These spacer blocks are placed below [illegible]ssemblies to enable access to the fuel pins by the transfer to[illegible] these spacer blocks provide adequate venting to allow passage [illegible]ter through and around the fuel assembly.

The transfer system will allow ac[illegible] fuel pins in the fuel assemblies to accomodate fuel pin co[illegible]ur reconstitution in the SFP. The gripper on the transfer tool is [illegible]o fail closed on the loss of hydraulic fluid. The indexing platform [illegible]ws the system to be manually placed over any desired fuel pin in its operating range.

All systems affected by the installation are non-nuclear safety related and safety precautions have adequately addressed hydraulic failures in the system. This evaluation addresses only the consequences of the transfer system and not the pin consolidation process. This alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

2800L-SPL

PA 19-83: Auto Trip and PAB Exhaust Fan (FN-2)

PA 19-83 provided contacts on the primary vent stack particulate and gas monitors to control FN-2. A high rad level in the stack will prevent auto-start and/or will trip FN-2 if it is operating. Prior to this alteration, FN-2 would auto-start when heating and ventilation unit #2 started and the outside temperature was above 50°F.

This alteration utilizes input from two high radiation alarms and fail-safe logic to prevent the inadvertent release of radioactivity through FN-2. This alteration reduced the possibility of an inadvertent release while maintaining the original function of the exhaust fan. Therefore this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 20-83: Installation of Isolation Valves to the Primary Water Storage Tank (PWST)

PA 20-83 installed two flanged ball valves on the inlet and outlet of the PWST. These valves allow isolation of the piping between the PWST and the thermo siphon heater (E-37) to permit maintenance/repair of the PWST heating system without draining the PWST.

The installation improves the operational control of the PWST heating system by allowing its isolation to conduct repairs. This alteration will not create the possibility of an accident or malfunction not previously evaluated and does not increase the probability or consequences of an accident previously evaluated. Therefore, this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 23-83: Uninterruptable Power Supply (UPS) for the FEMCO

PA 23-83 installed a UPS for the FEMCO through a direct feed line from the security distrubution panel (S-5), an isolating transformer, a manual transfer switch, and then to the communication distribution panel. Inverter #6, which feeds the security distribution panel, is backed by an emergency diesel generator. The transfer switch enables a choice between the present power source and the new one powered by Inverter #6. The new components were chosen to withstand a maximum system amperage of 40 amps at 120 VAC that occurs during alarm tests. The normal position of the transfer switch is toward Inverter #6.

The systems affected by this alteration are non-nuclear safety related. The change enhances the performance in the FEMCO system by improving its reliability. Therefore, this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PA 29-83: Contractor Parking Lot Lighting

PA 24-83 provided lighting for the Contractor's Parking Lot (Lot A). Two 400 watt sodium flood lights with mercury switches were mounted on wooden poles installed at the south end of the parking lot. Power is supplied from the well-house 120 VAC distribution panel.

This alteration does not affect any nuclear safety related system. Therefore, this alteration does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PDCR 2-83: Modification of the Steam Generator Feed Rings

PDCR 2-83 modified the steam generator (S.G.) feed rings by capping the present 76, 1" nozzles located on the underside of the feed ring and installing 28, 3" long radius 90° elbows to the top of each feed ring. Also, the 5 drain holes located on each feed ring were plugged. The present 1" threaded nozzles on the underside of the feed rings were capped with 3000 lbs. forged carbon steel caps. The new 3" long radius 90° elbows were installed by first cutting holes in the top of each ring. The first hole in each ring is located approximately 18" from the centerline of the feedwater nozzle. The remaining holes are equally spaced with a center-to-center distance of 15-3/4". The elbows were welded in place by a continuous 1/4" fillet weld. The outer end of each elbow is oriented radially toward the center of the S.G. Those elbows installed in the vicinity of the various can deck drains were off-set at an angle so that feedwater from the elbows is directed away from the can deck drain lines. The plugs installed in the feed ring drain holes were a tapered type plug and machined to fit the as-built size of the holes and fillet welded. The design input and analysis for these modification considered seismic loading, the additional weight and the change in weight distribution introduced to the feed ring. Flow characteristics were considered to ensure these changes will not effect the S.G. performance and integrity.

The changes were designed so as not to degrade or alter the reliability or availability of the steam generators. The occurance of water hammers is minimized through this change, thus minimizing the chance of damage to those vital systems due to this phenomenon. The desired mode for a primary plant cooldown after a plant trip or during accident conditions is via the steam generators, these design changes improve system reliability, and thereby improve plant safety. This change does not involve an increase in the probability or consequences of an accident previously evaluated or the possibility of a new or different kind of accident from any previously evaluated. Therefore this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PDCR 3-83: Steam Generator Chemical Feed Line Removal

PDCR 3-83 cut and capped the chemical feed line connected to the 14" steam generator main feed line. Chemical feed addition can now be supplied via the condensate system and separation of this line effectively reduces the amount of piping subject to system pressure, reducing the possibility of a rupture.

The chemical feed line is a 3/4" line which connects to the 14" steam generator main feed line downstream of the S.G. feedwater check valve (FW-131,231,331) on each S.G. These chemical feed lines were cut and capped and the section of feedwater line that it ties into was replaced. The support for the disconnected chemical feed line is Safety Class 2 rated since this line retains its Class 2 standing.

The systems affected by this change are Safety Class 2 and the modification adhered to these design standards. This modification does not introduce the possibility of accident occurence or malfunction of equipment, not previously addressed. Thus this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

PDCR 10-83: Dual Linear Output Isolation Of Nuclear Instrumentation Channels

PDCR 10-83 provided isolated output of the dual linear power subchannels to a patch panel in the Loose Parts Monitoring Panel (LPMP). Nuclear instrumentation channels 5 through 10 were used to provide the needed signals for noise analysis of the reactor vessel internals. The isolated patch panel provides connection to instrumentation for this analysis and yet allows access without affecting the Reactor Protection System (RPS). The alteration was accomplished by two similar approaches. First channels 5 through 8, which were previously wired for computer input, were interrupted and a TEC model 156 isolator was installed at each interruption. These isolators provide the electrical isolation required by IEEE 384-1981. Two parallel paths were wired from the isolators. One directly to the computer using the existing cable and the other to the patch panel terminal block in the LPMP. The second approach delt with the remaining channels 9 and 10. These previously did not have their subchannels wired to any terminal block. These inputs were first brought to a terminal block before being isolated. The remaining steps were the same as above except these channels were not previously wired for computer input and new cable was provided for this purpose. All isolators are powered by a Lamba 24 VDC supply mounted on the LPMP.

This design change adhered to the latest electrical separation requirements. The isolation configuration conforms to the IEEE 323-74, 344-75 and 384-1981. The design change does not adversely impact the RPS system functions. All components up to and including the isolators were designed to safety class 1E requirements. The cabling and the power supply are non-nuclear safety related. The isolation design of this modification prohibits this system from affecting the performance of other monitoring systems. Therefore, this design change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

EDCR 83-06: Mechanical Anchorage of Emergency Buses 5 thru 8

EDCR 83-06 upgraded the mechanical anchorage for plant emergency electrical buses 5 thru 8. Buses 5 and 6 were anchored to the floor using angle iron secured by bolts and lock washers.

Buses 7 and 8 were secured using steel brackets as cross-pieces over the buses and angle iron as verticle supports. The design criteria for structural steel modification was in accordance with AISC Specifications for the design, fabrication, and erection of structural steel for buildings. The steel, concrete anchors, and miscellaneous bolts, nuts and/or threaded rod were in compliance with plant engineering guidelines.

The emergency buses affected by this change are safety class systems, therefore, all material and anchorage support were chosen to fit the design criteria. This design change increases the capacity of these buses to withstand a seismic event, therefore, the possibility of any accident occuring has not been created or increased. This design change does not involve an unreviewed safety question as defined by 10 CFR 50.59.

EDCR 83-07: Modification of Feedwater Line Supports

EDCR 83-07 redesigned and installed the feedwater line supports H-13 and H-15. Supports H-13 and H-15 experienced concrete damage in the area of their main concrete embedment from an apparent water hammer that occurred during a plant trip. These supports were redesigned and installed in accordance with their safety class 2 designation.

The feedwater line affected by this EDCR is Safety Class 2, therefore, the two supports were constructed to fit Safety Class 2 requirements. This design change ensures that H-13 and H-15 are capable of supporting the required dead, thermal, and seismic loads. There is not an increase in the probability of an accident, equipment malfunction, or the occurence of an accident not previously evaluated. Therefore, this change does not involve an unreviewed safety question as defined by 10 CFR 50.59.