Duke Power Company
McGuire Nuclear Generation Department
12700 Hagers Ferry Road (MG01VP)
Huntersville, NC 28078-8985

T. C. McMEEKIN
Vice President
(704)875-4800
(704)875-4809 Fax

DUKE POWER

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Subject: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370

Pursuant to 10 CFR 50.59, please find attached a summary of Nuclear Station Modifications, Minor Modifications and procedure changes made to the McGuire Nuclear Station for the period of April 1, 1994 to April 1, 1995.

Questions or problems should be directed to Kay Crane, McGuire Regulatory Compliance at (704) 875-4306.

Very truly yours,

PR Herran for

T. C. McMeekin, Vice President
McGuire Nuclear Station

Attachment

cc: Mr. Victor Nerses, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. S. D. Ebneter, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW - Suite 2900
Atlanta, Georgia 30323

Mr. George Maxwell
Senior Resident Inspector
McGuire Nuclear Station

Duke Power Company
McGuire Nuclear Station
Summary of Nuclear Station Modifications Completed Under 10CFR50.59

NSM-12164
NSM-22164

Description:

These modifications will upgrade the hydrazine addition system at McGuire to reduce drum handling by personnel. This modification will also allow more accurate metering of hydrazine, including direct and undiluted addition, into the Main Condensate (CM) system.

NSMs MG-12164 and MG-22164 will upgrade the hydrazine addition portion of the Chemical Addition (YA) system. A vendor-supplied hydrazine tote bin will be located on the 760' + 6" elevation of the Service Building. The tote bin contains approximately 300 gallons of 54.4% hydrazine solution. There will be a sight glass provided on the tote bin to allow the level to be monitored continuously. Also, a nitrogen blanket will be provided in the tote bin. The primary source of nitrogen will be the (GN) system. A tap will be provided also to allow for use of a nitrogen bottle if required.

One low volume hydrazine addition pump per unit will be added to meter the hydrazine flow at 0.4 gph through each hydrazine supply line to the CM system. The new pumps will require pulsation dampeners to smooth out the flow to the CM system. The pump will have an explosion proof motor. It will be a diaphragm metering pump with the capability to be controlled electronically; however, there will be a handwheel in place to manually control the actuator. A pressure gauge will be added to monitor the diaphragm for ieakage.

A Volume Control Receiver (VCR) will be added downstream of the tote bin to obtain measured amounts of hydrazine without interrupting the normal flow. The VCR will also have a sightglass.

There will be curbing added around the tote bin to contain the contents of a full tote bin. Curbing will also be added around the pumps.

Pressure gauges will be added at inlet and outlet of the nitrogen bottle regulator and at the regulator of the tote bin inlet as well as, at the pump discharge and diaphragm.

Safety Review and USQ Evaluation:

The YA system is a non nuclear safety related system. It is used to maintain the proper O_2 concentration in the CM system. The tote bin will supply hydrazine to the VCR, the low volume hydrazine addition pump, the condensate hydrazine supply tanks, and the S/G hydrazine supply tanks.

The new system bypasses the original system. This system will be operated by adding the hydrazine directly to the CM system. However, if the low volume hydrazine addition pump fails, the hydrazine will be added from the Tote Bin to the Condensate Hydrazine Supply Tank by nitrogen blanket pressure and gravity feed. The hydrazine in the Tote Bin is the same concentration as used previously.

The piping and the valves are Class G designed for 150 psig and 120 F. There will be a class break after the metering pump. There will be core drills; however, no fire walls will be breached. There will be no changes to the fire protection system, HVAC, and no emergency eye wash systems will be required.)

There will be curbing around the tote bin and the low volume hydrazine addition pump to contain 100% of the contents. The tote bin and the pump will be located in the Service Building. The piping will be located primarily in the Turbine Building.

These modifications will improve the safety of the personnel for the addition of hydrazine to the CM system. The addition of hydrazine can be more accurately monitored and metered. Hydrazine is a volatile toxic chemical. The maximum inventory to be stored on-site has been increased from approximately 550 gallons to 1000 gallons. The tote bin, containing approximately 345 gallons will be located in the Service Building. The impact of this increase has been evaluated per Reg. Guide 1.78. It has been judged that these NSMs will have no adverse impact on Control Room habitability. Guidelines exist for handling, controlling, monitoring, and neutralizing leaks/spills of hazardous chemicals.

There are no significant new failure modes. An Appendix R review has been performed. There were no concerns identified. There are procedures in place to monitor and maintain the appropriate concentration of hydrazine in the CM system. Dilution of hydrazine and pumping in to the Auxiliary Feedwater (CA) system will not change.

The YA system is not an accident initiator. Therefore, the probability of an accident previously evaluated in the SAR is not increased.

The YA system is a non nuclear safety related system. The YA system is not used to mitigate the consequences of a Design Basis Accident. The increase in the maximum inventory of hydrazine stored on-site will not adversely affect Control Room habitability or the ability of the control room operator to mitigate an accident. Therefore, the consequences of an accident previously evaluated in the SAR is not increased.

There are no new failure modes. The piping, valves, pump, and tote bins will meet system design criteria. An appendix R review has been performed. There were no concerns identified. Therefore, the possibility of a new accident or malfunction is not created.

The associated piping, valves, and pumps are designed to meet design specifications. There will be no seismic interaction between the YA system and safety related equipment. The YA system is a non nuclear safety related system. Therefore, the probability of a malfunction of equipment important to safety evaluated in the SAR is not increased.

These NSMs will not degrade the effectiveness of any equipment important to safety to perform its function in the event of a Design Basis Accident (DBA). Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The YA system is not evaluated in the FSAR. The increase in the amount of hydrazine to be stored on-site has been evaluated per Reg. Guide 1.78 (Position C.4) and will not adversely impact the safety of the control room operators. There are no applicable sections in the Technical Specification.

These modifications do not adversely impact the fission product barriers or plant parameters important to safety/analysis. Therefore, the margin of safety as defined in the bases to any Technical Specifications is not reduced. No USQ exists.

NSM-12289, 22289, 52289

Description:

These NSMs install test connections on the WZ (Groundwater Drainage) system, sumps A, B, and C, to facilitate pump flow tests and check valve leakage tests. The system is safety related as it performs the function of relieving hydrostatic pressure from the auxiliary and reactor buildings which are Category 1 structures. This system is discussed in FSAR sections 2.4.13, 9.5.8, and 7.6.11. Technical Specification 3/4.7.13 addresses Groundwater Level with no specific requirements on availability of sump pumps and piping. Each sump (of the 3 existing) has two 100% capacity pumps aligned to separate essential power sources (trains A and B). There are no electrical changes involved with this NSM. The mechanical scope includes necessary piping and valves to accomplish the tests. The piping being added is class C (safety related and seismic) out to an isolation valve at which a class change occurs to class G (non-safety). This design preserves the integrity of the existing lines and offers seismic protection to the functionally safety related portions of the system.

The WZ system is used in control of and mitigation of internal flooding events that may occur in the auxiliary building. Flooding events are design events and are not accidents. Mitigation of design events does not necessarily require single failure considerations unless the flood logically exists when making design basis event combination assumptions such as earthquakes and accidents. In these cases, the scenario would be considered a design basis event, not a design event (flood). The WZ system is redundant and safety related and can evacuate all flood sources to which it can be subjected within design basis assumptions.

Safety Review and USQ Evaluation:

The WZ system is not an accident initiator and no accidents previously thought incredible are made credible by this modification. Therefore, the probability of accidents previously evaluated in the FSAR is not increased and no new accidents are created. The WZ system is not an accident mitigation system and no new malfunctions are created. Mitigation of flooding events will not be affected by this NSM. Therefore, the consequences of accidents and malfunctions of equipment important to safety previously evaluated in the FSAR are not increased.

These NSMs modify the existing WZ system to facilitate pump flow and check valve leakage tests. All applicable criteria have been followed and the design creates no new failure modes.

The seismic integrity of the system is preserved with class C piping connecting to the existing piping and locked closed isolation valves at the class C to G boundary. Therefore, the probability of a malfunction of equipment important to safety evaluated in the FSAR is not increased and no new malfunctions are created.

Margin of safety is related to the confidence in the fission product barriers. These modifications are not related to the fuel, cladding, NC pressure boundary or the containment vessel. No safety

limits, setpoints or limiting safety system settings are affected by these modifications. Therefore, the margin of safety defined in the basis to the Tech. Specs is not reduced. No USQ exists.

NSM-19420

Description:

The purpose of NSM MG-19420 is to re-route the Main Feedwater (CF) system piping to interface with four new BWI Steam Generators (S/G's) to be installed in the Unit 1 Reactor Building. The CF piping requires re-routing because the CF inlet will be at a higher elevation on the new S/G's than on the existing S/G's.

NSM MG-19420 has been divided into two parts. Part 1 is associated with work to be implemented during EOC9. Part 2 is associated with EOC10 work. This calculation supports the evaluation of Part 1, but may be revised later to include Part 2.

PART 1 (EOC9):

During EOC9, a baseplate for a rupture restraint will be installed on the crane wall in each of the 4 S/G enclosures. The rupture restraints will be installed by MG-19420, Part 2 during EOC10.

Also during EOC9, two baseplates and support structure for a spring support will be installed on the crane wall in each of the 4 S/G enclosures. The spring supports will be installed during EOC10, by MG-19420, Part 2. Instrument tubing for S/G Narrow Range Level instruments will require relocation as a result of the baseplate addition. Tubing relocation is covered by NSM MG-19410.

Safety Review and USQ Evaluation:

PART 1 (EOC9):

Safety-related Systems, Components, or Structures, (SSCs) directly or indirectly involved in this modification include the following:

New baseplates and support structures (added by this modification)
Reactor Building Crane Wall
Steam Generators (existing and new)
Main Feedwater (CF) lines (existing and new)
Reactor Coolant (NC) lines
S/G Narrow Range Level instruments

The CF and NC systems are evaluated as accidents initiators in FSAR Chapter 15. However, the added baseplates and support structures are qualified in accordance with QA Condition 1 requirements, and have been designed for seismic integrity, seismic mounting, adequacy for application, and to meet required material specifications. For these reasons, interaction of the added components with nearby SSCs (not modified) is not credible. Since there are no credible interactions with nearby SCCs, the performance of plant safety functions will not be degraded and there are no common failure modes created between redundant equipment trains. The crane wall has been evaluated for the baseplate and support addition, and will not be degraded.

Therefore the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased. Also, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

No new plant functions are added by the modifications. No new failure modes are created. No accidents previously considered incredible are made credible by this NSM. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modifications. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSMs. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

NSM-12401

Description:

Main steam line drain valves 1SM83, 1SM89, 1SM95, and 1SM101 were reviewed in accordance with Generic Letter 88-14 to determine if air operated valves would perform their design function under a loss of air. These valves are currently normally open, fail open, and incident open air operated valves that are listed as Containment Isolation Valves in the McGuire FSAR, Table 6-113. Concern was raised over the ability of these valves to isolate the steam generators following a seismic event.

Results of a subsequent operability evaluation revealed that the Chapter 15 accident analyses would not be adversely affected by this steam release path. Dose consequence analyses considered this amount of steam release during the recovery period for accidents involving secondary side releases. Later safety analyses were revised to consider the phenomenon of Steam Generator tube bundle uncovery (TBU) which may occur during some accident transient scenarios. As a result of this change in assumed accident conditions, it is necessary to isolate the Main Steam line drain lines in some cases. Accident analysis results require that the valves be closed in 30 minutes from the onset of the accident, assuming worst case conditions for accidents involving TBU.

General Design Criteria (GDC)-57 states that containment isolation valves for closed systems (outside containment) shall be either automatic, or locked closed, or capable of remote manual operation. These valves fall under the category of GDC-57 valves and can be remotely operated. However, for accidents, credit can only be taken for local manual operation since the valves have air actuators with non-safety related controls which are assumed incapable of functioning during a Design Basis Accident. To better meet the intent of GDC-57, these valves are being changed from fail open to fail closed which is the more conservative safety related position.

This NSM will change the failure mode of the previously listed valves from fail open to fail closed by replacing the operator on these valves. The incident position will also be closed (i.e. loss of offsite power, Steam Generator Tube Rupture, accidents requiring Main Steam isolation, seismic events causing loss of downstream Class G piping, or loss of instrument air).

Safety Review and USQ Evaluation:

This change will help isolate the Main Steam System and containment during a design basis accident (DBA) with a loss of offsite power. These valves are located in Duke Class B piping and perform a containment isolation function and therefore are considered QA1. The valve operators are also QA1 because the valve must be capable of being closed and remain closed to isolate containment (S/G's) during DBA's. This NSM will enable each valve to be closed by locally, manually removing the air from the operator allowing the spring to close. All of the valve controls will be considered non-QA because the assured method of closing the valves does not involve the normal valve controls. It is also acceptable and possibly required for the valves to be closed for accident scenarios Since containment and Main Steam line isolation is enhanced and accident analyses are still acceptable, there is no increase in the probability or consequences of an accident evaluated in the SAR. During accidents, the closure of these valves will not damage any safety related equipment . Based on this and the above information, the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR is not increased. Normal valve operation will be essentially unchanged and there are no adverse affects on any system . Valve misalignments or failures will be determined as before and will allow corrective actions before system problems occur . Valve control is no less reliable than before. Therefore, the possibility for an accident or for malfunction of a different type than any evaluated in the SAR is not created.

No safety limit, setpoint, or operating parameter will be changed by this modification. No fission product barrier is diminished. Therefore, the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

NSM-12427

Description:

NSM MG-12427 adds a manual valve and test valve assembly to 15 of the 18 branches from the safety injection test header to air operated globe isolation valves and to the EMO Kerotest globe isolation valve, NI122B. These globe valves have a history of seat leakage problems which cause fluctuations in the Cold Leg Accumulator (CLA) levels. Tech. Spec. 3/4.5.1 addresses the specific requirements of the CLAs and problems in this area can affect unit availability. The block valves will be installed in the Class E portion of the check valve test header piping downstream of the Class B header boundary valves.

This modification is to facilitate individual valve testing to isolate potential leakage into the check valve test header that can cause ND system pressurization, Cold Leg Accumulator (CLA) leakage, and NC to NI check valve leakage masking.

Safety Review and USQ Evaluation:

This modification will affect the Safety Injection (NI) System. The NI system is an accident initiator evaluated in Chapter 15.5.1.1 of the FSAR. However, this accident is initiated by spurious electrical signal or operator error, not mechanical components. Since this NSM is adding valves with no control circuitry changes, there can be no affect on the frequency of this accident. Therefore, the probability of an accident evaluated in the SAR is not increased.

No system used to mitigate an accident is adversely affected, and there are no changes relative to other systems. The NI test header modified by this NSM is not in any ECCS flow path related to NI injection or recirculation and the test header is isolated during normal operation, therefore, no possible alignment configuration of the test/block assemblies will affect the ability of the ECCS to perform its intended function. Valves and piping classifications used are consistent with requirements. Hence, there is no increase in the probability or consequences of a malfunction of equipment important to safety or increase in consequences of an accident evaluated in the SAR. Furthermore, design review and revision of the piping stress analysis will ensure that compliance with seismic criteria is not violated.

No new credible failures are introduced or operating characteristics of the ECCS changed in this NSM. The control logic is unaffected, the globe valve function is unchanged, and no Appendix R concerns have been introduced. Therefore, no new malfunctions or accidents of a different type than evaluated in the SAR are created.

Margin of safety refers to margins associated with the integrity of the fission product barriers. This NSM does not affect the fuel, cladding, or NC pressure boundary. The containment boundary (including the containment isolation valves) is not degraded. Also, this modification will not affect any safety limits, set points, or operating parameters. Hence, the margin of safety as defined in the bases to any technical specification will not be reduced. No USQ exists.

NSM-12413
NSM-22413

Description:

The charging and letdown functions of the Chemical and Volume Control System (NV) are employed to maintain a predetermined water level in the Reactor Coolant System (NC) pressurizer, thus maintaining proper reactor coolant inventory during all phases of unit operation. This is achieved by means of a continuous feed and bleed process during which the feed rate is automatically controlled based on pressurizer water level. The bleed rate can be chosen to suit various unit operational requirements by selecting the proper combinations of letdown orifices in the letdown flow path.

Two letdown orifices are arranged in parallel to reduce the pressure of the letdown stream to a value compatible with the letdown heat exchanger design. One of the orifices is sized to pass normal letdown flow; the other can pass less than the normal letdown flow. A third letdown path is provided via a control valve. Any combination of the orifices and control valve can be utilized in order to increase letdown flow such as during reactor heatup operations and maximum purification. This arrangement also provides a full capacity for control of letdown flow. The two orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. The control valve is also controlled by remote manual operation. A low pressure letdown controller controls the pressure downstream of the letdown heat exchanger to prevent flashing of the letdown liquid.

Orifices (1, 2)NVFE6200 are sized to pass the normal flow of 75 gpm. Orifices (1, 2)NVFE6210 are sized to pass 45 gpm. The NSMs will replace these orifices with new orifices that have been redesigned to pass the same flows at the same pressure drops, but with reduced cavitation. In addition, piping upstream and downstream will be replaced, including thermal wells (1, 2)

NVTW5110. This piping will be raised off the floor to allow adequate clearance for welding. Where possible all socket welds will be replaced with butt welds. Each of the three flowpaths has thermal expansion loop. Two pieces of properly sized bent pipe will be welded together to make new thermal expansion loops and reduce the number of required welds. All piping is austenitic stainless steel and all joints and connections will be welded. The function and operation of this part of the NV System is unchanged.

Safety Review and USQ Evaluation:

Since the components are designed in accordance with applicable codes with due consideration for the design and operating conditions , the materials are either austenitic stainless steel or a material compatible with reactor coolant , and joints and connections are welded , the structural integrity of the system is maintained. The piping and orifices are still Duke Class B Support/restraints and have been evaluated. Since the structural integrity of the system is unaffected and no other piping is adversely affected, no Loss of Coolant Accident is affected. The system will function and be operated as before. Based on the above, reactor coolant inventory and boron concentration are not affected. Therefore, there is no increase in the probability or consequences of an accident evaluated in the SAR. Since NV System parameters are not changed and the integrity of the system remains intact, there is no increase in the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. No new failure modes have been identified. Based on this and the above, the possibility for an accident of a different type or a malfunction of a different type than any evaluated in the SAR is not created.

No safety limit, setpoint, or operating parameter will be changed by these modifications. Therefore the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

NSM-12423

Description:

During the McGuire Unit One end-of-cycle 8 (1EOC8) refueling outage, a Type A, Integrated Leak Rate Test (ILRT) was conducted as required by McGuire Technical Specification 4.6.1.2(a), "Surveillance Requirements for Primary Containment/Containment Integrity." After successful completion of this ILRT, testing was performed on all dual-ply bellows assemblies as specified by Technical Specification 4.6.1.2(h). The bellows assembly on penetration 1-M441 (Steam Generator 1D Main Steam Line) exhibited large leakage in comparison to other similar penetrations and in comparison to previous test results for this penetration which showed only minor leakage. Additional tests were performed to further quantify the magnitude of leakage at each of the four bellows, including how much leakage represented bypass leakage around the Annulus and Annulus Ventilation System. These tests revealed that about one third of the penetration leakage represented bypass leakage subject to Technical Specification 4.6.1.2(f), and that most of the bypass leakage was associated with one of the two bellows located outside the Annulus. The NRC was subsequently notified and permission was granted to continue operation providing that monthly tests were performed to assure that the leak does not degrade during Cycle 9. An epoxy patch was qualified and tested at Senior Flexonics and was applied to the leak area. The patch has held successfully to date. However, as part of a permanent solution, McGuire has decided to modify the bellows assembly during the next outage (1EOC9).

The bellows assembly for penetration 1-M441 is a 54" diameter dual-ply design constructed of four separate dual-ply bellows. The outer ply serves as a pressure test boundary for verifying the integrity of the inner ply, which is considered containment boundary.

For this modification, a bellows clamshell will be installed on the interior side of the containment vessel wall sleeve and attached to the SM guard pipe at penetration 1-M441. Pathway Bellows, Inc. will be contracted to design, manufacture, and install the clamshell assembly. This assembly will be designed for a 40 year plant life and meets or exceeds the design specifications for the original bellows assembly.

Safety Review and Evaluation:

The bellows assembly for penetration 1-M441 is not an accident initiator. However, it does perform an accident mitigation function. The assembly acts as a part of the primary containment boundary and is thus a fission product barrier. This modification will add a clamshell assembly to the original assembly, resulting in a unit that performs as well as the assembly for the original design. Therefore, there is no increase in the consequences of accidents previously evaluated in the FSAR which require containment integrity for accident mitigation. This modification will improve the current situation, which includes temporary Technical Specifications higher leakage limits, thereby lowering the consequences of accidents previously analyzed. Since no accident initiators are involved in this modification, there is no increase in the probability of accidents previously evaluated in the FSAR.

The modification is adding a clamshell assembly to the current bellows assembly. No new failure modes are created by this modification which have not been previously evaluated. Therefore, the modification will not create the possibility or malfunction of a different type than any evaluated in the FSAR. The bellows assembly will perform the same function and operate under the same principle as the original assembly.

The new bellows assembly will not result in a degradation of performance as compared to the original bellows assembly. It will function under the same principle as the original assembly and fall under the same testing program. The new assembly meets or exceeds the design specifications for the original assembly. No new failure modes are created by this modification and no new interactions with equipment important to safety are created. Therefore, there is no increase in the probability or consequences of a malfunction of equipment important to safety evaluation in the FSAR.

The margin of safety as defined in the Technical Specifications is related to the confidence in the fission product barriers. This modification will not degrade the containment fission product barrier in any way. The new combined bellows assembly for penetration 1-M441 is an equivalent or better assembly than the original design. The original assembly has known leak rates that result in consequences within our licensing limits. The new assembly will not result in an increase in the leak rates. Therefore, this modification does not reduce the margin of safety as defined in the basis to the Technical Specifications. No USQ exists.

NSM-12422

Description:

Modification MG-12422 adds additional redundant controls to Main Feedwater (CF) regulating valves 1CF-17, 1CF-20, 1CF-23, and 1CF-32. Dual 7300 cards and pneumatic control schemes are added for each valve. The controls are added such that the operator is allowed to select the train of controls used. The control input and logic are unchanged from the existing control scheme. The additional controls are only provided to allow swapping to an alternate control in the event the normal controls are not working properly. This should improve the reliability of the CF regulating valves. The function and operating characteristics of the CF valves will not be affected by this modification. No safety controls or functions are changed or degraded by this control addition.

MG-12422 involves replacing 4 M/A stations on the Main Control Board to accommodate the dual controls. This change does not require any cutting or seismic requalification of the control board since the replacement equipment fits the existing cutouts on the board and the weight of the replacement components is the same as the existing equipment. Four new driver cards and four new blocking diode cards are added to the non safety 7300 cabinets. Addition of dual pneumatic controls will add four new E/P converters, four new solenoid valves, four new valve positioners, and four new volume boosters. Power supply is provided from existing breakers. All additional equipment has been reviewed for Appendix R requirements.

All the components added or modified by this NSM are non-safety. The mounting of this equipment has been reviewed for seismic concerns associated with the control board and cabinets where the components are located and determined to be acceptable.

Safety Review and USQ Evaluation:

Inadvertent closure and opening of the CF regulating valves are considered in licensing basis accident analysis as feedwater malfunctions which may cause either a decrease or increase in feedwater flow. Therefore, the controls are considered important to safety. This modification does not change the function or operating characteristics of the regulating valves. The safety controls of the valves are not changed. The modification provides a redundant control mechanism which should improve the reliability of the valves. Therefore, this modification does not increase the probability of an accident as evaluated in the SAR. Operators may compensate for a failure of a component in the control system by switching to the alternate control. However should the system fail to operate correctly, the resulting regulating valve failure would not be different than currently considered. Therefore, neither the consequences of an accident evaluated in the SAR nor the consequences of a malfunction of equipment important to safety are increased. It is expected that the dual control scheme will be more reliable than the current control scheme. The dual control scheme does not change the manner in which the regulating valves are controlled. Therefore, the modification does not create the possibility for a malfunction of a different type than evaluated in the SAR or create the possibility for an accident of a different type than considered in the SAR. The new components are considered as reliable as the currently used components. No components are required to perform beyond their previously specified capability by this modification. Therefore, there is no increase in the probability of a malfunction of equipment important to safety as evaluated in the SAR.

The margin of safety as defined in the Technical Specifications is related to the confidence in fission product barriers. This modification does not create or change any interactions with fission product barriers. Therefore, the modification does not reduce the margin of safety as defined in the basis to the Technical Specifications. No USQ exists.

NSM-19010

Description:

There are platforms and other items that will interfere with removal and replacement of the steam generators (S/Gs). The purpose of this NSM was to remove any of these platforms and other items which can be removed in the outage prior to the steam generator replacement outage. This will reduce the amount of work required during the replacement outage.

The items covered by this NSM are as follows:

1. Remove platforms at each S/G secondary access manway, elevation (El.) 798' + 3 ¾".

2. Modify platforms below the S/Gs (at El. 739' + 0") to install a storage location below the grating for new manway covers. Also the platform below B & C S/Gs will be extended to provide more working area, as exists for the platform below A & C.

3. Move platform landings up 3' - 6" from the following elevations to avoid interference with the feedwater supply line (CF system) reroute: (CF reroute covered by NSM MG-19420)

786' + 3 ¾"	(A S/G)	787' + 2 ¾"	(B S/G)
788' + 0"	(C S/G)	787' + 8 ¾"	(D S/G)

4. Modify platforms at El. 783' + 5 in A and B S/G cavities to avoid interference with S/G shell cone inspection ports.

5. Various cable trays and supports will be modified as detailed in the Final Scope Document.

Safety Review and USQ Evaluation:

The platforms involved in the modification are not safety-related and do not perform any functions important to safety, however, they are seismically designed and installed (QA Condition 4) to prevent interaction with safety-related components. QA Condition 4 requirements are maintained for the added/modified platforms (items 2-4). Removal of platforms (Item 1) has no safety significance.

The cable tray, cable tray supports involved in the modification are safety-related (QA Condition 1). QA Condition 1 requirements are maintained for added or modified cable tray and cable tray supports (Item 5). No cable changes will be made except some cables are to be moved (shifted without repulling) to added electray, and some conduit will be reworked. Cable separation requirements will be maintained. Therefore the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The platforms involved in the modification do not perform any plant safety functions. The cable trays and cable tray supports perform the safety function of supporting safety-related cables. There is no change in qualification of the cable trays and cable tray supports, so the performance of plant safety functions will not be degraded, and the same functions will continue to be performed. Cable separation will be maintained, so no common failure modes will be created. Therefore the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

There are no new plant functions added by the modification, nor are any new failure modes created, since all existing criteria will be maintained for the components involved. No accidents previously considered incredible are made credible by the NSM. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSM. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

NSM-12392
NSM-22392

Description:

These modifications add an additional level of undervoltage protective relaying on the 4.16 KV Essential Power (EPC) system buses, 1ETA, 1ETB, 2ETA, 2ETB, and revise the setpoints for the existing protective relaying.

The EPC system is normally powered from the offsite power grid through the 6.9 KV Normal Auxiliary Power (EPB) system which connects to the Main Generator and McGuire Switchyard. In the event that this offsite power supply is lost, emergency power is provided to the EPC system by two redundant Emergency Diesel Generators (D/Gs) per unit. This event is known as a Loss of Offsite Power (LOOP) accident, and is evaluated in FSAR Section 15.2.6 . Existing protective relaying on the EPC system buses serve to detect loss of offsite power by detecting a loss of bus voltage. When this occurs, the buses are automatically separated from the offsite power supply. Also, the D/Gs automatically start and supply emergency power to the buses.

The purpose for the protective relaying added by this modification is to detect a degraded voltage condition which may prevent nuclear safety related plant equipment, required for safe plant shutdown and/or accident mitigation, from performing intended safety functions. The concern to be mitigated by this modification is the potential for having a persisting degraded voltage level that is above setpoint of the existing relaying, but below a level where damage to operating safety-related equipment may occur over a period of time.

A voltage sensing relay will be installed on each of the 3 power phases for each bus. If 2 out of 3 relays on a bus detects degraded voltage, 2 timing relays are started. One timing relay assures the degraded voltage is not a short duration transient. If the degraded voltage persists until after this relay is finished its timing cycle, an annunciator alarm is activated. The 2nd timer continues its timing cycle to allow a period of time for the operators to implement any possible actions to

correct the degraded condition. If the degraded condition remains present until the end of the 2nd timing cycle, separation from the offsite power grid occurs automatically. Also, at any time during the 2nd timing sequence, separation from offsite power will occur automatically in the event of a Safety Injection signal.

Timing sequence length and relaying setpoints for the existing and new protective relaying were determined by engineering evaluation . The additional protective relaying equipment is qualified for use in a nuclear safety related (QA Condition 1) application. Seismic qualification will be maintained for the enclosures in which the new relaying equipment will be installed.

The modification is intended to satisfy an NRC commitment made as a result of the EDSFI audit. The relaying scheme design satisfies NRC Branch Technical Position PSB-1 .

Safety Review and USQ Evaluation:

Separation of the EPC system from offsite power due to loss of degradation of the offsite power source is a Loss of Offsite Power (LOOP) Accident, evaluated in FSAR Section 15.2.6. These modifications provide additional automatic means for the EPC system buses to separate from offsite power. Automatic separation from offsite power (a LOOP* accident) will be more probable because the voltage setpoints for the new relaying will be higher than the settings for the existing relaying. The closer relay settings are to 100% bus voltage, the more frequently actual bus voltage can be expected to occur at or below the setpoint. Therefore, the probability of an accident previously evaluated in the SAR is increased.

The occurrence of a LOOP* presents a challenge to safety systems. More probable (e.g., more frequent) LOOPs* increase the frequency of safety system challenges, which increases the probability of malfunction of equipment important to safety. However, offsetting this probability increase is a probability decrease due to the protection of safety equipment from degraded voltage conditions, given by the added protective relaying. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The EPC System is required to provide power for equipment used for accident mitigation and safe shutdown. The ability of the EPC system to perform its required safety functions will not be degraded by the modifications. No common failure modes are created between redundant EPC system power trains. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

No new failure modes are created by the modifications. No accidents previously considered incredible are made credible. The added protective relaying is expected to be as reliable as the existing relaying. The added equipment is qualified to QA Condition 1 (nuclear safety related) requirements, and qualifications of equipment enclosures have been maintained. Thus, the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

The setpoints for the existing Loss of Power protective relays are lowered by these modifications. The new setpoints have been evaluated and will not prevent the protective relaying from performing its required safety function. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSMs, except as previously discussed for

probability of a Loss of Offsite Power (FSAR Section 15.2.6). Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

*This is not a full LOOP, an actual LOOP event is separation at the 6.9 kV bus. This represents a partial LOOP.

NSM-19710

Description:

This modification is part of the Steam Generator (S/G) Replacement Project (SGRP). The Steam Generator Blowdown Recycle System (BB) and the Steam Generator Wet Lay-up Recirculation System (BW) will require modification.

The existing SG's each have two 2" nominal outside diameter (OD) BB System nozzles. The replacement SG's (RSG's) will each be provided with two 3" nominal OD BB System nozzles. MG-19710 modifies the BB System piping to accept the 3" OD SG connections.

Also, the existing SG's each have a secondary side shell drain connection. Each SG is equipped with a 1" OD capped drain line including a drain valve (CF System). The RSG's will not be provided with secondary shell drain connections, but SG shell side draining will be provided similarly through drain lines which will be added to each SG's BB header. MG-19710 will permanently remove the existing SG shell side drain lines and reuse the drain valves to install SG secondary side drain lines on each SG's BB System header.

An instrumentation tap currently exists on each SG shell side drain line. These taps were previously utilized for BW System level instrumentation. The BW System level instrumentation was deleted by NSM MG-12217. There is currently no identified future plans for use of these instrumentation taps. Therefore, these instrument taps will be deleted with the removal of the existing SG shell drain lines.

Implementation has been divided into two parts. Part 1 will be implemented in EOC09, and Part 2 will be implemented in EOC10. This evaluation only covers Part 1 as described below.

> Part 1: Part 1 will delete 22 snubbers, 1 spring, 2 rigid restraints, and replace 4 snubbers with struts. All of these supports are on the BB System.

Piping analyses is complete on the present configuration and the new piping configuration with these support changes. These analyses take into account all normal, upset, and faulted conditions, including re-evaluation of pipe break locations. The review and evaluation includes pipe stress, pipe movements, valve accelerations, nozzle load evaluation, and other standard piping analysis review items. The Pipe Rupture Interaction and Jet Impingement Load/Location Analyses has also been evaluated. The revised analysis utilizes new variable damping response spectra, approved by the NRC, to qualify the routing changes and to delete as many snubbers as possible. The supports are all QA Condition 1. New materials are compatible with the containment environment.

Safety Review and USQ Evaluation:

The results of the piping analyses indicate that these support changes are acceptable for the present BB piping configuration and the new piping configuration. The calculated piping stresses remain within ASME Code allowables. Based on this, the new piping support system is adequate to allow the BB System (which serves no safety function) to function as designed and to not adversely affect any other systems either as the piping is configured now or as the piping configuration is modified. Therefore, the probability of an accident or of a malfunction of equipment important to safety evaluated in the SAR is not increased. Using the same reasoning as above, all systems will respond during all events or accidents essentially as before. No single failure or train separation requirements are affected. Therefore, there is no increase in the consequences of an accident or of a malfunction of equipment important to safety evaluated in the SAR. No new failure modes were identified. The QA conditions of the supports are QA 1. There are no new line break locations. Consequently, this NSM, Part 1, does not create the possibility for an accident or for a malfunction of a different type than any evaluated in the SAR.

No safety limit, setpoint, or operating parameter will be changed by Part 1 of this modification. Therefore the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

NSM-12219
NSM-22219

Description:

This modification will remove the instrument loops associated with the Reactor Coolant Pump (NCP) labyrinth seal d/p gages (1,2NVPG5000, 5010, 5020, 5030). These loops have root valves off each NCP at two taps, the high and low side of the labyrinth seal. There are impulse lines routed inside lower containment to the crane wall and through the Containment Vessel penetration into the annulus. The manifold valves, excess flow check valves, and flow gages are located inside the annulus. This modification involved the removal of the impulse lines, root valves, excess flow check valves, manifold valves and d/p gages A blank flange was used to isolate the existing flanges on each NCP. The containment penetration was capped according to existing procedures.

Safety Review and USQ Evaluation:

In the past there was a scheme that utilized the labyrinth seal d/p to set up and validate the seal flows to the NCP seals. However, the following loop instrumentation is used for monitoring and metering: 1,2NVP5300, 5310, 5320, 5330 (for seal water flow) and 1,2NVP5200, 5210, 5220, 5230 (for d/p of #1 seal). In addition, the instrumentation being deleted (1,2NVPG5000, 5010, 5020, 5030) does not serve a safety function.

The NCPs will not be affected or degraded by deletion of this instrumentation. The probability and consequences of accidents described in FSAR Section 15.3, Decrease in Reactor Coolant System Flow Rate, and Section 15.4.4, Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature will not be increased since pump operation will not be affected. The probability and consequences of NV system-related accidents referenced in FSAR 15.4.6 and 15.5.2 are not increased because these accidents involve boron concentration changes in the NC

system which are not be affected by deletion of this instrumentation. All appropriate lines will be isolated via blank flanges. There are currently various flanged connections for the NCPs that are tightened to the appropriate setting. Using blank flanges on these instrument lines that are removed will not increase the likelihood of a loss-of-coolant accident such that the creation of a different type of accident or malfunction will not be increased. The NCPs will not be degraded; therefore, there is no increase in the consequences of an accident or of a malfunction of equipment important to safety evaluated in the FSAR. No USQ exists.

NSM-52151

Description:

This NSM will upgrade the plant telephone system. The purpose, operation and function of the system is unchanged by this NSM. A 3-rack satellite distribution frame will be added in an enclosure on the east side of the Turbine Building. The soundproof enclosure is a prefabricated structure with lighting, convenience outlets and air condition for personnel comfort. The air conditioning is self contained and does not interface with any other mechanical systems. Power for the enclosure will be provided from the lighting and power panelboards in the area.

Cables used in high radiation areas are not required to be environmentally qualified as the communication system is not safety related. The same type cables will be used in these areas. All mounting brackets for added phones are mounted QA-4 to avoid any seismic interaction concerns.

Four 6 inch diameter core drills are required on the Turbine Building operating floor. This presents no flooding concern as the non-Category 1 Turbine Building does not house any equipment requiring flood protection. Floor loading has been considered and is acceptable.

Existing spare containment electrical penetrations will be utilized for telephones added to containment. Since communication cables are not capable of high valued faults, double fusing is not required. An Appendix R evaluation has been performed with no concerns identified.

Safety Review and USQ Evaluation:

The communication system is not an accident initiator. Therefore, the probability of accidents previously evaluated in the FSAR is not increased.

The communication system is not used for accident mitigation. The electrical penetrations being affected do perform a passive mitigation function ensuring a leak-tight containment barrier. They are not degraded by this NSM. Therefore, the consequences of accidents previously evaluated in the FSAR are not increased.

No new failure modes are created. No Appendix R concerns were identified. As stated earlier, the penetration assemblies are not degraded by this NSM. The aspects of this NSM located in the Turbine Building are not important to safety. Therefore, the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased. Also, no accidents or malfunctions of equipment important to safety, different than previously evaluated are created.

Since all applicable criteria have been followed for each aspect of this NSM, the probability of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

The only aspect of this NSM related to Margin of Safety is the electrical penetration assemblies. As stated earlier, existing spare penetrations are being utilized and double fusing is not required due to the inability of these communication cables to produce high valued faults. Therefore, the Margin of Safety defined in the bases to the Tech. Specs. is not reduced. No USQ exists.

NSM-12279/p2

Description:

Modifications to the Emergency Diesel Generator (D/G) Starting Air System (VG) are planned to increase D/G reliability. These modifications will be implemented by several partial NSMs: MG-12279/p1 through MG-12279/p7.

The work scope for MG-12279/p2 has been divided into 2 major parts, outage and pre-outage work, in order to reduce the outage work scope. The pre-outage work was implemented by Work Orders (WOs) 94038379 and 94038448. The WOs were previously evaluated per 10CFR 50.59.

This evaluation covers only the outage work to be implemented by MG-12279/p2, listed below:

1) The existing pressure switches which control VG compressor operation will be replaced by pressure transmitters to improve accuracy of air pressure control. The pressure transmitters are QA Condition 1 qualified (nuclear safety related).

2) In the existing configuration, the VG system is required to supply air during all phases of D/G operation (standby, running, and shutdown). When the D/G is running, air from VG is required to keep the D/G fuel racks in the open position. When the air is removed, the racks close and the engine shuts down. To increase D/G reliability, the system configuration will be changed so that air will not be required to keep the engine running.

3) The existing hydraulic/pneumatic overspeed circuit will be removed. Speed sensors and circuits will be added so that 2 out of 3 logic will be used to detect overspeed and trip the D/G. This added redundancy is expected to increase the reliability of the overspeed trip function so that failure of an individual component will not prevent D/G shutdown or cause premature shutdown. The added sensors and circuits are QA Condition 1 qualified.

4) During blackout conditions, with no Loss of Coolant Accident (LOCA) present, and with the D/G at or above 95% speed, the VG air header will align to the Instrument Air (VI) header via VG93 (Train A) and VG 94 (Train B). This provides VI with an assured source of makeup air from VG. If a significant break occurred in VI, VG will not be able to keep up with demand. A pressure switch will be added to monitor the VG headers and close VG93 (94) if VG pressure falls to 180 PSIG. The added pressure switch is QA Condition 1 qualified.

5) An additional temperature transmitter, capable of local indication, will be added on the aftercooler exhaust lines for each VG tank lineup. Temperature will be read via a strap-on RTD on each air line. These transmitters and RTDs are for monitoring only, and do

not perform a safety function. They will be mounted and will be in accordance with QA Condition 4 (seismic) requirements.

6) An additional permanent welding receptacle will be added in each D/G room. The additional receptacles will not perform any safety functions. They will be mounted in accordance with QA Condition 4 requirements.

Safety Review and USQ Evaluation:

These modifications involve the following D/G support systems:

VG --- D/G Starting Air
LD --- D/G Lube Oil
EQC - D/G Control System
VI --- Instrument Air

The D/Gs and the support systems are not accident initiators in any accident analyses. The required equipment qualifications (QA Condition 1 or 4) have been met for the added equipment. Overall D/G reliability is expected to be improved due to the modifications. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The D/Gs and support systems perform the safety function of supplying emergency power to equipment required for safe plant shutdown and accident mitigation. All of the systems involved will continue to perform the same functions. All required equipment qualifications and specifications are met, so the performance of required safety functions will not be degraded. There are no interfaces created between redundant D/G trains, so no common failure modes are created. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

There are no new failure modes created. All required qualifications and specifications are met. Therefore, no accidents previously considered incredible are made credible by this NSM. D/G reliability will not be decreased, and is expected to be increased. Thus, the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification.

The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSM. Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased.

This modification will install a pressure switch to monitor the VG header pressure, and open VG93 and VG94, Cross Connect to the Instrument Air (VI) header, with a non-LOCA event, with a Blackout signal present, and with the associated diesel at or above 95% of its rated speed, with VG header pressure greater than 180 psig. A 15 second Time Delay relay also ensures that sufficient time has elapsed for diesel start. 3 speed switches are employed which sense diesel output electrical frequency (Hz). This frequency signal is converted into diesel rotational speed (rpm). A 1 out of 3 logic is employed in the circuitry.

Cross connecting VG with a failed VI system (VI lost due to a seismic or tornado event) could cause a failure of the diesel to start. This circuitry and modification will not cause this failure. A single active failure of 1 of the 3 speed switches will not cause a failure of the diesel to start; should a signal from a failed speed switch be present concurrent with a blackout signal and no LOCA signal, a 15 second Time delay relay is also required prior to opening VG93 or 94. Should the header depressurize to 180 psig, the pressure switch would then close VG93 or 94.

This pressure switch has a normally closed contact. This contact must open on 180 psig pressure increasing, which provides the signal for VG93 or VG94 to remain open concurrent with a blackout signal. This pressure switch would close at about 180 psig VG pressure decreasing, which would isolate VG from VI. The fail position of this switch is fail close. Failure of this switch to close at 180 psig decreasing would not affect any nuclear safety systems.

A loss of power to the circuit would cause VG93 and VG94 to fail closed.

The VG system is normally isolated from the diesel at 40% of rated diesel speed. No USQ exists.

NSM-12375
NSM-22375

Description:

The subject modifications involve replacing Service Building and Warehouse (VO) System HVAC units TB1-AHS-2, TB1-AHS-6, and TB2-AHS-2. These HVAC units are non-safety related. TB1-AHS-2 serves the Unit 1 Turbine Building Battery Room, TB2-AHS-2 serves the Unit 2 Turbine Building Battery Room, and TB1-AHS-6 serves the Central Alarm Station (CAS).

The existing units are water-cooled by the Conventional Service Water (RL) System. The HVAC units are significantly degraded due to condenser fouling. The new units will be air-cooled. The HVAC units for each battery room will be replaced with two 50% units. Decreased maintenance, increased reliability and performance of the VO System are benefits expected from the modification.

The RL supply line will be capped at the main header. Most of the 2" RL piping and valves associated with the HVAC units will be removed. The new cooling units will be installed in the same location as the existing units. Some concrete pad, ductwork and wall opening changes will be necessary.

An Appendix R Review was performed for electrical power supply changes.

Safety Review and USQ Evaluation:

The VO system is not an accident initiator, and is not addressed in any FSAR Chapter 15.0 accident analyses. The system is non-safety related. There are no seismic or electrical separation requirements applicable to the system. An Appendix R review was performed with no concerns identified. The modifications are expected to increase the reliability and performance of the HVAC supply to the Turbine Building Battery rooms and the Central Alarm Station.

Temperature margins for equipment in these areas should be improved. The Turbine Building battery rooms contain batteries 1DP and 2DP. These batteries supply non-essential unit DC loads such as motors and backup lighting. The Central Alarm Station contains security alarm equipment, which is not required for reactor shutdown. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

VO system cooling units TB1-AHS-2, TB2-AHS-2, and TB1-AHS-6 are not used to mitigate accidents. The VO system will continue to perform the same functions as before the modification. The cooling units are non-safety and do not perform any safety functions, nor interact with other equipment so that the performance of a safety function is degraded. Therefore the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR is not increased. There are no new system functions added by the modifications, and no new failure modes created. No accidents previously considered incredible are made credible by these NSMs. The new cooling units are expected to be more reliable than the existing units. Seismic qualification is not required because the equipment is non-safety and located in the Turbine and Service Buildings, which are non-seismic structures. An Appendix R review was performed, with no concerns identified. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modifications. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSMs. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

NSM-52378

Description:

Water-cooled air conditioning units SB-AHS-5 and SB-AHS-6 will be replaced with roof-mounted air-cooled units. The existing conventional service water (RL) line which supplies these units will be capped at the main header. Piping, valves, and fittings associated with SB-AHS-5 and SB-AHS-6, that will no longer be used, were removed.

Air conditioning units SB-AHS-5 and SB-ASH-6 serve the Secondary Alarm Station (SAS) and Personnel Access Portal (PAP). The existing units have not cooled the SAS and PAP properly due to RL pipe and air conditioning condenser fouling. The modification is expected to decrease required maintenance, increase cooling system performance, and increase system reliability.

The existing air conditioning units are located on the floor level of the Service Building, in the areas served. The replacement units will be located on the roof of the Service Building, above the areas served. The existing ductwork will be modified as required to accommodate the new units. A new return air duct will be added. An existing fresh air intake will be closed, because the new air conditioning units incorporate fresh air intakes. New roof penetrations were added for the air conditioner tubing. A roof curb is provided for each unit. Equipment support steel is provided.

The new units were sized based on the original capacity of the existing units. The new units are non-safety related (as are the existing units). The electrical power loading will change due to the new units. A new thermostat control will be added for SB-AHS-6 and existing thermostats will be used for SB-AHS-5.

Safety Review and Evaluation:

No FSAR Chapter 15 accidents or events are initiated by the Service Building HVAC (VW) system, nor the Conventional Service Water (RL) system, therefore, the systems are not accident initiators. No seismic or separation criteria are applicable to the systems. The power supply (non-safety) was upgraded to allow for the increased power loading. An Appendix R review was performed, with no concerns identified. The modifications are expected to improve the VW system performance and reliability. The potential for the roof-mounted equipment becoming a tornado missile was considered, and determined to be no concern. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The RL and VW systems are not used to mitigate accidents, and do not perform any plant safety functions. The performance of safety functions will not be degraded by the modifications. The RL and VW systems will perform the same functions as before, except RL system cooling water will not be used to cool the new air conditioning units (they will be air-cooled). There are no common failure modes created. The new fresh air intakes are in approximately the same location as the existing fresh air intake. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

No new functions were addedand no new failure modes were created by this modification. The Service Building roof loading was review and loading calculations were updated. No accidents previously considered incredible are made credible by this NSM. The new air conditioning units are expected to be more reliable than existing units. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSM. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

NSM-19040

Description:

The purpose of this NSM is to remove or relocate steam generator (S/G) interferences and add rigging/attachments, which can be implemented during (1EOC9) the outage prior to the steam generator replacement outage (1EOC10). This will reduce the total scope of work required during the replacement outage.

Items covered by this NSM are summarized below.

1. Removal of unused conduit.
2. Relocation of electray. The electray contains cables which are relocated with the electray. The relocation is not of sufficient distance to be considered cable re-routing.
3. Relocation of light fixtures.
4. Removal or modification of piping hangers for the Safety Injection (NI) and Main Steam (SM) systems.
5. Addition of lifting lugs for SM rupture restraints.
6. Addition of lugs for scaffolding.
7. Addition of supports for skyclimber and lugs for lifelines.
8. Addition of lugs to support the Upper Lateral Supports.
9. Addition of attachments on Polar Crane for lifelines.
10. Addition of cranewall baseplates and embedded plate stiffeners.
11. Remove rupture restraints from Cross-over Leg.
12. Addition of lifting lugs to the S/G enclosure domes.
13. Prepare nuts/bolts (remove paint) and other items for removal in 1EOC10.

Safety Review and USQ Evaluation:

All added materials are compatible with the containment environment. As determined by engineering evaluation, no degradation will occur to structures, systems or components (SSCs) due to added components, or piping hanger deletion/modification. Electrical separation requirements have been maintained for relocated cabling and electray. QA qualification and seismic mounting requirements have been maintained for all SSCs involved. By current pipe break criteria, the Crossover Leg rupture restraints are no longer required and may be removed as necessary. Since all applicable qualifications and criteria (as summarized here) have been maintained for the components involved in the modification, status quo has been maintained for the SSCs involved. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

As stated above, all applicable qualifications and criteria have been maintained for the components involved in the modification, therefore, the performance of nuclear safety functions will not be degraded, and no common failure modes are created that could render redundant trains of safety-related equipment inoperable. All added components are passive civil structures, which do not change the functions of the SSCs to which they are added. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

All added components are passive civil structures which perform no nuclear safety functions. No new active components are added, therefore, no new functions are added. No new failure modes are created for the SSCs involved, since (as stated previously) all applicable qualifications and criteria have been maintained. Also for this reason, no accidents previously considered incredible are made credible by the NSM. Thus, the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of this modification, and since (as stated previously) all applicable qualifications and criteria have been maintained for the SSCs involved, the fission product barriers (RCS pressure boundary, containment, fuel

pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSMs. Therefore, the margin of safety as defined in the basis for any Technical Specifications is not decreased. No USQ exists.

NSM-19610

Description:

MG-19610 modifies the Nuclear Sampling System (NM) piping to be compatible with the McGuire Replacement Steam Generators (RSGs). The NM system provides representative samples of process fluids for laboratory analysis without requiring containment access. One of the sampled fluids is the secondary water in each steam generator. The RSGs differ from the current Steam Generators in size and nozzle locations for some connections. The NM system currently connects to the Steam Generator Blowdown System (BB) for secondary water sampling. The RSGs will have a separate NM nozzle at a different location which will require rerouting of the connecting NM piping on each generator. The new sample locations are selected to provide a good sample of the secondary water without undue influence from tube bundle turbulence or auxiliary feedwater flow during post accident testing. The piping analysis has been revised to incorporate the new pipe routing and current piping analysis methodology. This results in a reduction in the number of snubbers required. The piping material for the piping being replaced is the same as the existing piping.

The modification is divided into 2 parts to minimize outage time during the outage when the RSGs are installed. Part 1 of the modification, to be performed during EOC-9 is specifically addressed in this evaluation. During EOC-9 approximately 40 feet of piping along with a 1/2 inch valve per steam generator will be installed. The piping will be supported by a total of 26 new supports. Based on the revised piping analysis, 11 snubbers, 1 spring, 1 rigid support, and the vertical part of another support will be removed from the existing NM piping. Five snubbers will be replaced with rigid supports, 1 snubber will be replaced with shim plates and 1 spring support will be added. These modifications will leave the existing NM piping in place and qualified, and install part of the revised piping without creating any interference problems for operation or steam generator replacement. The new piping installed, will be sealed to prevent foreign material intrusion during plant operation and will be seismically qualified to ensure that it will not damage safety related components in the area by falling during a seismic event.

Safety Review and USQ Evaluation:

This modification will not functionally change the NM System. Part 1 of the modification will not change any sample points or impact operation in any way since the only changes are from revision of piping analysis and installation of piping which will not be connected to the system until implementation of part 2 of the modification. The added piping which is not connected to the system, is seismically qualified to ensure no adverse interaction impacts any safety component or system during operation. The support changes as result of piping analysis revision will leave the piping fully qualified for all design events to which the system was previously qualified. The NM system is not used in accident mitigation or analysis and is isolated on containment isolation. There is no change in the operation of the system for part 1 of the modification. Therefore, there is no increase in the probability or consequence of an accident evaluated in the SAR. Since the current piping and operation are unchanged in part 1 of the modification, no possibility of a different accident than those considered in the SAR is created. The system qualification is not degraded by the revised piping analysis and the added piping is

fully qualified for location in this area. Therefore, there is no increase in the probability of malfunction of equipment or any increase in the consequence of a malfunction as evaluated in the SAR. Since there is no change in operation, there is also no possibility of a different type of malfunction created. The NM system penetrates the containment structure which is a fission product barrier. This modification does not impact the ability of the system to be isolated in any way. All design and licensing criteria are met in the same manner as before the modification. Therefore there is no reduction in the margin of safety as defined in the basis for any technical specifications. No USQ exists.

NSM-22383

Description:

This modification provides improved HVAC service to room 1251, located at the Unit 2 upper containment entrance. The room has been converted into office/training space. The room is presently served by the Fuel Pool Ventilation System (VF). The HVAC service has proven to be inadequate since the room's conversion to office space. The new HVAC system will consist of an air-cooled split system heat pump. The existing VF system arrangement supplies fresh air to the room from a main VF duct via a supply duct. This modification installed an air handling unit in line with the supply duct, to cool or heat the air for the room as required. The air handling unit was suspended from the ceiling just outside the room. Fresh air will continue to be supplied to the room via the existing supply duct, through the air handling unit. The new system utilizes refrigerant lines routed to a roof-mounted condenser through a wall penetration less than 6" diameter. The penetration is sealed to maintain the fire rating of the wall. Thermostatic controls were added in the room to control heating or cooling based on room temperature. MEVN-2578 temporarily installed a small air-conditioning unit in the east wall of the room. This unit was removed and the hole repaired by the subject NSM.

A calculation was generated to size the new heat pump unit and refrigerant lines. A calculation was revised to account for the added roof loading due to the new HVAC unit and support pad. Electrical power load assignments were made for the new equipment and an Appendix R Review was performed for the electrical changes, with no concerns identified.

The VF system is non-QA Condition 1, however, the filter packages on the VF exhaust are QA Condition 1. The ability of the VF system to filter a portion of airborne contaminants resulting from postulated fuel handling accidents in the Fuel Building, is factored into the FSAR Chapter 15 analyses of those accidents. The VF system is evaluated in FSAR Section 9.4.2.3.

Safety Review and USQ Evaluation:

The VF System is not addressed as an accident initiator in any FSAR Chapter 15 accidents. The supply portion of VF (to which the new equipment is installed) is not safety-related (no applicable QA Condition). No seismic or separation criteria are applicable. No concerns resulted from the Appendix R review. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in SAR is not increased.

The VF system performs the function of providing an air flow differential across the Fuel Pool. This is to ensure that air from the Fuel Pool area (potentially containing airborne radioactive contaminants) will be filtered through the VF exhaust. The VF system exhaust serves to mitigate

postulated fuel handling accidents, analyzed in FSAR Chapter 15, by removing a portion of the resulting airborne radioactive contaminants. The performance of these safety functions will not be degraded by the modification because:

* The VF exhaust train is not involved in the modification, therefore, the VF filtering capability will not be affected.
* The new system is a split unit utilizing refrigerant lines instead of ductwork to penetrate the building walls. The 6" penetration created for refrigerant lines and power supply to the exterior portion of the new HVAC unit will be sealed. The penetration for the wall-mounted air condition unit to be removed will be repaired. Therefore, no new leak paths for airborne radioactivity will be created.

For the above reasons, the consequences of an accident or malfunction of equipment important safety evaluated in the SAR are not increased.

New equipment was added to the VF system to provide additional heating and cooling to Room 1251. As discussed above, this new function will not prevent the VF system from performing any functions important to safety. The wall fire rating will be maintained. No accidents previously considered incredible are made credible by this NSM. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the NSM. Therefore the margin of safety as defined in the basis for an Technical Specification is not decreased. No USQ exists.

NSM - 52315

Description:

The modification replaces the closed circuit television video (CCTV) switchers in the Central Alarm Station (CAS) and Secondary Alarm Station (SAS). The existing video switchers are obsolete, and replacement parts are unavailable. New switchers will be installed that utilize video capturing technology, which digitizes and saves frames of CCTV picture prior to and following an alarm.

Safety Review and Evaluation:

The CAS and SAS are not accident initiators in any accident or any analyses, nor are they accident mitigators. Therefore the probability or consequences of an accident previously evaluated in the SAR is not increased.

The modification involves no nuclear safety-related equipment or plant safety functions. The CAS and SAS are not located in buildings which house safety-related equipment. All of the modification work is within the CAS and SAS. No common or new failure modes are created involving any equipment or systems within the plant. No accidents previously considered incredible are made credible by the modification. Thus the possibility of an accident of a different type than evaluated in the SAR will not be created.

The CAS and SAS are SSCs evaluated as part of the McGuire Physical Security Plan (Reference FSAR Section 13.7). The modification does not decrease the effectiveness of this plan. However, revision of the Security Plan is required, pursuant to 10CFR 50.54 (p). No new functions relating to plant operation are added and no new failure modes are created. Therefore the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR are not increased. Nor will the possibility of a malfunction of equipment of a different type than evaluated in the SAR be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

NSM - 22258

Description:

Temporary scaffolding is installed in the two Unit 2 Containment Spray (NS) / Residual Heat Removal (ND) heat exchanger rooms (2A and 2B) when maintenance on the heat exchangers is required. The installation and removal of this scaffolding is dangerous, time consuming, and results in radiation exposure to the maintenance workers. This NSM will provide permanent platforms which will facilitate maintenance and improve ALARA.

The platform for the 2B heat exchanger room is designed to suit the new NS heat exchanger. The platform for the 2A heat exchanger Room is designed to suit the original NS heat exchanger, but will incorporate features that will make the platform easily modified to suit a new NS heat exchanger if a replacement heat exchanger is required in the future.

This NSM will add/modify three platform levels in each of the ND/NS heat exchanger rooms (2A and 2B). The first platform is at entrance level (EL. 750' - 0"). The existing platform will be enlarged in order to move the down-ladder from the high dose side (ND) to the low dose side (NS). An up-ladder from the high dose side (ND) to the low dose side (NS), also an up-ladder will be added to the landing level (EL. 765' - 7 3/4"). At this level, grating will be put on the NS heat exchanger seismic support structure. The main platform is at EL. 775' - 0".

The platforms will be comprised of structural steel members which will frame into the heat exchanger room walls, or support on the existing steel structures. Grating will be used as the walking surface for the platforms. Bolted connections will be used for the platform members, so that they can be easily disassembled if required. The platforms will be QA Condition 4.

Safety Review and Evaluation:

The Containment Spray (NS) and Residual Heat Removal (ND) systems are nuclear safety related systems. The NS System is used to mitigate the consequences of design basis accidents (DBA). The NS Systems safety function is to remove thermal energy in the event of a LOCA. The ND Systems safety function is to remove sensible heat and decay heat from the core and reduce the temperature of the Reactor Coolant system (NC) during plant cooldown and refueling operations. These systems are QA Condition 1.

The platforms will be installed QA 4. The platforms will not have any seismic interactions with the safety related equipment in the rooms. The platforms are not accident initiators. Therefore, there is no increase in the probability of an accident or of a malfunction of equipment important to safety evaluated in the SAR. Since the platforms are seismically designed and also will not prevent access to any safety related valves, there will be no adverse affects on any safety related equipment or systems. The capability of the safety systems to perform their safety functions is not affected. Based on this, there is no increase in the consequences of an accident or of a malfunction of equipment important to safety evaluated in the SAR. The platforms will be QA 4. Access to safety related equipment is not inhibited. There are no new failure modes identified. Therefore, the possibility for an accident or for a malfunction of a different type than any evaluated in the SAR is not created.

No safety limit, steeping, or operating parameter will be changed by this NSM. Therefore, the margin of safety as defined in the basis of the Technical Specifications will not be reduced.

NSM-12425
NSM-22425

Description:

In the late 1970s there were approximately 30 events that produced pressure excursions above the Technical Specification pressure temperature (P/T) limits while reactors were operating at low temperatures. The frequency of these overpressure events was high enough for the NRC to classify them as anticipated operational occurrences. Based on this classification PWR licensees implemented procedures to reduce the potential for overpressure events and installed equipment modifications to mitigate such events. The protection systems used to mitigate and reduce the potential for these events are termed Low Temperature Overpressure Protection (LTOP) systems.

At McGuire Nuclear Station, each train (for both Units 1 and 2) of the NC system has a 0-3000 psig pressure transmitter providing inputs to the 7300 system for the generation of the LTOP open signal to the Pressurizer PORVs. These modifications will add a QA-1 narrow range pressure transmitter (0-600 psig) to each train of the NC system, for both Units 1 and 2. The purpose of these modifications is to substantially reduce the instrumentation uncertainty associated with the LTOP function and its related effects on the Appendix G heatup and cooldown curves. These transmitters (1(2)NCPT5122 and 1(2)NCPT5142) will replace the existing wide range transmitters (1(2)NCPT5120 and 1(2)NCPT5140) function of providing inputs into the LTOP function. Note that the wide range pressure transmitters will continue to be used for other functions. A signal to the 7300 system is provided by the transmitters in order to generate a LTOP open signal. The transmitters will also send a signal to the OAC for Operations use during LTOP mode.

The setpoint for opening Pressurizer PORVs 1(2)NC32B and 1(2)NC34A (on signals from the new transmitter loops) will be conservatively reduced from 385 psig to 380 psig rising.

The current wide range transmitters are calibrated to 1%. The transmitters installed as part of this modification are calibrated to 0.25%, thus significantly reducing the level of uncertainty associated with the LTOP function.

The impulse lines of the new transmitters will tap into the impulse lines of the existing wide range transmitters. The narrow range transmitters will be mounted in close proximity to the wide range transmitters in the electrical penetration room on elevation 755' (both Units). The existing LTOP circuitry within the 7300 cabinets will be utilized. A channel test card (NCT) and a loop power supply card (NLP) will be added to each loop in the 7300 cabinets and all necessary rewiring performed. Each transmitter will require one safety cable to be routed from the transmitter to the 7300 cabinets. The new transmitters and associated equipment will be located outside of Containment (both Units); therefore, there are no containment penetration concerns associated with the modifications. Electrical has reviewed the modifications for Separation Criteria and Appendix R concerns. In addition, the new transmitters will be mounted seismically

The transmitters will normally be valved out using double isolation valves in order to prevent overpressurization. The transmitters will only be used during LTOP mode. Procedures for Unit Startup and Shutdown will be revised to reflect the necessary steps to instruct I&E to valve in and out these transmitters as necessary. The valving of the transmitters may occur between 600 psig and the time LTOP is required to be in service. Operations can monitor these transmitter signals on the OAC.

Safety Review and Evaluation:

These modifications will not result in an increase in the probability or consequences of any accidents evaluated in the SAR. These modifications will reduce the uncertainty involved in the generation of the LTOP open signal by adding the more accurate narrow range transmitters. This in effect, will assist in ensuring that the 10 CFR 50 Appendix G pressure/temperature limits are not exceeded during an LTOP event. In addition, the modification packages list specific instructions for implementation of these modifications to ensure that all Technical Specification requirements are met for operability.

These modifications do not create the possibility of a malfunction or an accident of a different type than any evaluated in the SAR. These modifications will result in an improvement to the existing LTOP function reducing the uncertainty involved in the generation of the LTOP open signal. The new pressure transmitters will operate in the same fashion as the original wide range pressure transmitters. While the potential exists for the overpressurization of the new transmitters, administrative controls will ensure that this type of malfunction does not occur.

The new pressure transmitters will not increase the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. The new QA-1 pressure transmitters will operate in the same fashion as the original wide range pressure transmitters. The rest of the LTOP function and the 7300 system will operate in the same manner as before. These modifications will only reduce the uncertainty involved in the generation of the LTOP open signal, thereby ensuring that pressure/temperature limits are not exceeded during LTOP events. The setpoint change (to 380 psig rising) conservative, and will also ensure that these limits are not exceeded during LTOP events.

There are no changes of safety limits or plant parameters as a result of these modifications. The fission product barriers (RCS pressure boundary, containment, fuel pellets and cladding) are not degraded. No assumptions made in any accident analyses are affected by these modifications. Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased.

NSM-12081

Description:

The Condenser Circulating Water (RC) System receives discharges from several sources, some of which are the Ventilation Unit Condensate Drain Tanks (VUCDTs), the Waste Monitor Tanks (WMTs), and the Turbine Building Sumps (TBSs). The rate at which the VUCDTs, WMTs, and TBSs can be drained to the RC system is limited by four inch discharge lines. This modification provides a new route for discharges from the VUCDTs, WMTs and TBSs which significantly increases discharge flow. The new route enables simultaneous WMT, VUCDT and TBS discharges. SSCs affected include the discharge piping form the VUCDTs and WMTs to the RC System, discharge piping from the TBSs to the RC System, and RC System piping. Eight inch piping was added upstream of existing four inch discharge piping on the Turbine Room Sump Pump (WP) System. This new piping connects to the RC System downstream of the recirculated Cooling Water (KR) Heat Exchangers. Each of the new TBS discharge lines contains a flow meter, totalizer and sampler on a 3/4 inch bypass line off the 8 inch TBS discharge line. A strainer and a backwash line were installed immediately upstream of each sampler. Also, the TBS Pump discharge flow restricting orifice plates was bypassed with 8 inch piping containing check valve and a butterfly valve.

New piping routes flow from existing VUCDT/WMT discharge piping to the new TBS discharge line. A check valve was provided between the TBS and the VUCDT/WMT discharge points in the new line. The new discharge piping from the VUCDT/WMTs contains a check valve and an air operated valve. The air operated valve will receive signals from the WMT and VUCDT EMFs, and from the RC pump minimum flow interlock; the valve will close if high radiation is detected from the VUCDT, the WMTs, or if RC flow is too low to achieve the required dilution.

A manual valve and an air operated valve is located on the new discharge line downstream of where the new VUCDT/WMT discharge piping joins the new TBS discharge piping. The air operated valve can be controlled from either the Waste Processing Liquid Panel or the Main Control Board, and automatically closes on the RC minimum flow interlock.

Safety Review and Evaluation:

This modification will affect the Liquid Waste Monitor and Disposal (WM) System, described in FSAR Section 1.2, the Condenser Circulating Water (RC) System, described in FSAR Section 10.4.5, and the Turbine Room Sump Pump (WP) System, which is not evaluated in the FSAR. The SSCs directly affected are the discharge lines from the WM System to the RC System, TBS pump discharge lines, and RC piping downstream of the KR Heat Exchangers. The new TBS discharge lines are Class G piping (and are connected to existing Class G piping) up to the check valve separating the TBS discharge from the VUCDT/WMT discharge. At this point, stainless steel Class E piping is used up to the manual isolation valve, and back to existing Class E WM System piping.

Discharge flow from the VUCDT and WMTs to the RC System will terminate on a high radiation signal from 1(2) EMF44 (for the VUCDTs, or from 1EMF49 (for the WMTs) to a new air operated valve. VUCDT/WMT discharge flow will currently terminate on the same signals, only to a different valve. The new TBS discharge line is downstream of 1(2) EMF31, so that on a high radiation alarm, discharge flow may be terminated from the Main Control Board.

The Steam Generator Tube Failure event is described in FSAR Section 15.6.2. The Radioactive Liquid Waste System Leak or Failure event is described in FSAR Section 15.7.2.

NSM- 22181

Description:

This modification will provide the equipment necessary to monitor temperatures and displacements of the pressurizer surge line and to monitor temperatures of the 1.5" charging lines to the cold legs of the RCS. This modification is in response to NRC Bulletins 88-11 and 88-08 and will not be permanent. The bulletins address the need to include the effects of thermal stratification in the qualification of surge line piping based on plant specific data and the effects of thermal stresses on piping connected to the RCS. MPVN-1373 relocates some existing thermocouples and adds some new thermocouples to other parts of the NI and NV system and also the ND system.

Strap on bands will contain multiple thermocouples oriented circumferentially about the pipe to provide temperature information. Displacement gages will be wired to brackets mounted on straps that will be buckled around mirrow insulation. Any insulation that is removed to install the bands will be replaced to near as-found condition. There will be a wire penetrating the insulation in some places for the displacement gage installation. Cables from this equipment will be routed to a cabinet in the incore instrumentation room and from there will feed a mainframe to be set up in the cable spreading room. This system is non-QA since it provides no safety related function. All instruments and cabinets will be mounted to QA-4 requirements as necessary to prevent seismic interaction concerns. The electrical penetration being utilized is adequate for this application. An Appendix R review has been performed. The effect of the additional weight of the instrumentation on the affected pipes has been considered and is acceptable.

Safety Review and Evaluation:

The instruments and their associated components do not perform any safety functions. Instruments and cabinets inside containment are mounted QA-4. Pipe stress effects due to additional weight are insignificant. The insulating characteristics of the system will not be degraded. The pressure boundaries of the ND, the RCS, and NI/NV systems will not be degraded (no welding). Therefore, the probability of accidents and malfunctions of equipment important to safety are not increased. Also, since all safety systems are not adversely affected, the consequences of accidents and malfunctions of equipment important to safety will not be increased. No new malfunctions of equipment are introduced because proper design considerations have been made. Since no new malfunctions of equipment are introduced there are no new accidents created by this modification. All fission product barriers are maintained by this modification. The fuel is not affected, the RCS pressure boundary is not degraded, and containment integrity is preserved. Therefore, the margin of safety as defined in the basis of the Tech. Spec. is not reduced.

NSM- 42416

Description:

The demineralized water (YM) system provides filtered demineralized water to the upper surge tanks (CM system) for makeup and to other systems throughout the plant that require high quality water. For several years, a temporary filtering system has been used to ensure the quality

of this makeup water. The purpose of this modification is to prepare the location and install a building for the permanent YM filtering system which will replace the temporary one. A permanent YM Processing Building (pre-fabricated metal building) will be located on the south side of the Auxiliary Electric Boiler Room. The building will have permanent electrical power for the building needs. The electrical supply for the Reverse Osmosis (RO) skid and the recirculation pumps will remain as temporary power until the building is erected.

Permanent power will then be supplied to the Reverse Osmosis skid. Temporary power will continue to be supplied to the recirculation pumps until permanent power is added by NSM MG-52416. The piping will remain as temporary hoses or PVC piping located near the skid.

The RO skids and filter tanks will be placed in the building, thereby eliminating the need for the temporary mobile processing trailer. This safety evaluation considers the preparatory work to be completed prior to erecting the building and building installation work.

Safety Review and USQ Evaluation:

The YM system is not an accident initiator in any FSAR Chapter 15 accident analyses. The only safety related system that the YM system interfaces with is the Control Room Area HVAC Chilled Water System. The YM system provides makeup water for this system. The modification will not affect this function. Modifications to the YD system, the electrical system, instrument air system, and the Auxiliary Electric Boiler Room do not involve any systems important to nuclear safety. The changes to the security lighting will not result in a degradation to the Security Plan. Therefore, there is no increase in the probability of any accident or malfunction of equipment important to safety previously evaluated in the SAR.

The YM system provides high quality water to various systems throughout the plant. This modification will not degrade the performance of the YM system or any system that it provides purified water to. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

No new failure modes are introduced. No accidents previously considered incredible are made credible by this modification. Therefore, the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

Since the modification does not interact with any fission product barrier or a system that could affect a fission barrier, it will not result in a degradation to such barriers (RCS pressure boundary, containment, fuel pellets, and fuel cladding). No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical Specifications is not decreased.

NSM-12266
NSM-22266

Description:

These modifications change load center (LC) and motor control center (MCC) breakers on the 4 KV power system, and upgrade several power cables. The modifications are QA Condition 1. These modifications are being made in response to an analytical model review of the 4KV power system, which was conducted to review the system because of changes made since its original

design. NSM MG-12266 modifies the Unit 1 4KV power system, and NSM MG-22266 modifies the Unit 2 4KV power system.

Summary of changes:

1. Replace trip units on incoming breakers in MCCs 1EMXH, 1EMXG, 2EMXH, and 2EMXG, with units of higher amperage.

2. Pull new cables to MCCs 1EMXH, 1EMXG, 2EMXH, and 2EMXG to upgrade for higher amperage.

3. Revise settings on breakers in LCs 1(2) ELXA, 1(2) ELXB, 1(2) ELXC, and 1(2) ELXD, that supply power to MCCs 1EMXH, 1EMXG, 2EMXH, 2EMXG.

4. Revise incoming breaker settings for LCs, 1ELXA, 1ELXB, 1ELXC, 1ELXD, 2ELXA, 2ELXB, 2ELXC, 2ELXD.

5. Revise the settings for the 32DGT and 59EGN relays on Unit 1 and 2.

6. Pull new cables to the Unit 1 and 2 backup pressurizer heater power panels to upgrade for higher amperage.

Safety Review and USQ Evaluation:

The 4KV power system is QA Condition 1, however, the system is not an accident initiator. The system is not addressed in plant accident analyses (McGuire Nuclear Station FSAR, Chapter 15). A seismic qualification review was determined to be unnecessary because breaker trip units in MCCs 1(2)EMXH and 1(2)EMXG were replaced with equivalent units having a higher amperage rating). An Appendix R review was conducted, with no concerns identified. Therefore, the probability of accidents previously addressed in the FSAR is not increased.

Cable and breaker sizing, as well as breaker coordination, for the affected LCs, MCCs, and power panels have been considered. Breaker trip units used for replacement in MCCs 1(2)EMXH and 1(2)EMXG were ordered with 10CFR50, Part 21 dedication provided by manufacturer. Thus, the probability of a malfunction of equipment important to safety previously evaluated in the SAR is not increased, and no new malfunctions are created.

No accidents previously thought incredible are made credible by this NSM, thus the possibility of an accident of a different type than evaluated in the SAR will not be created.

The 4KV power system supplies power for equipment used for accident mitigation. The modification does not change the function or operation of the 4KV power system. The NSM introduces no common failure modes. The ability of the 4KV power system to perform its safety function is not degraded. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the FSAR is not increased.

There are no changes of safety limits, setpoints, or plant parameters because of this modification. No assumptions made in any accident analysis are affected by this NSM. Therefore, the margin of safety as defined in the bases for any Technical Specification is not increased.

NSM-12434
NSM-22434

Description:

Present Control Room HVAC Panel and local panel indications of Auxiliary Building Ventilation (VA) Supply, Filtered Exhaust and Unfiltered Exhaust Fan operation is based on discharge check damper positions via limit switches. Since these indications have proven to be unreliable, these mods were initiated so that indications of fan operation will be based on pressure switches. Existing pressure switches across the Filtered and Unfiltered Exhaust fans will be used to indicate fan operation, and pressure switches will be added for the Supply Fans to provide indications. Lights and nameplates on the Control Room HVAC Panel (and local panels) will be changed to more accurately reflect fan ON/OFF indication (as opposed to damper OPEN/CLOSED indication). The pressure switches for the Supply Fans will be mounted QA-4 (seismic).

The VA Supply Fans are interlocked to not start unless the Unfiltered Exhaust Fans have started, to ensure that a negative pressure is maintained in the Aux Bldg. VA Exhaust consists of two 50% Unfiltered Exhaust Fans and two 50% Filtered Exhaust Fans, one each per train. The Unfiltered Exhaust Fans discharge air from the non-contaminated area of the Aux Bldg to the Unit Vent. The Filtered Exhaust Fans discharge air from potentially contaminated areas of the Aux Bldg and filter the air if a LOCA, blackout, or high radiation signal from EMF-41 occurs.

Safety Review and USQ Evaluation:

These modifications will not increase the probability of an accident or malfunction of equipment important to safety evaluated in the SAR. The likelihood of an accident evaluated in the SAR will not be impacted. The VA System is not an accident initiator. The design functions of the VA System will not be impacted. These mods will improve VA Filtered and Unfiltered Exhaust, and Supply Fan status indications.

These modifications will not increase the consequences of an accident or equipment malfunction evaluated in the SAR. The post-accident response of VA components will be the same as before. The operation of existing pressure switches across the Filtered and Unfiltered Exhaust Fans will not be impacted by these mods. The reliability of VA indications will be improved by using the pressure switches for fan status. The likelihood of discharge check damper failures will not be increased.

These mods will result in the loss of discharge damper position indications for the affected fans. Reliance on these indications to verify proper VA alignment will no longer be possible, and any procedures relying on these indications will need to be revised. These dampers may need to change position to perform their functions, however, the likelihood of one of these check dampers (or another check damper downstream of these dampers) failing to change position when desired is not considered to be high enough to require remote position indications. These dampers are counterbalanced to swing open on flow. The Filtered Exhaust bypass dampers, 1(2)ABF-d-3, will still have Control Room position indication. Based on the above, the ability to verify proper post-accident VA alignment will be maintained.

These mods will not create the possibility for an accident or malfunction of a different type than any evaluated in the SAR. No new failure modes are created and no accident mitigation

equipment will be impacted. ECCS pump room exhaust fans will not be affected and the operating characteristics of VA components will not be changed.

These mods will not reduce the margin of safety as described in the bases for any Tech Specs. The margin of safety as defined in the bases for Tech Specs 3/4.7.7 (Auxiliary Building Filtered Ventilation exhaust System and 3/4.7.12 (Area Temperature Monitoring) will not be reduced. VA operability ensures that radioactive materials leaking from ECCS equipment within the Aux Bldg following a LOCA are filtered prior to reaching the environment. Area temperature limits ensure that safety-related equipment will not be subjected to temperatures in excess of environmental qualification temperatures. The ability of the VA System to perform these functions will not be degraded.

NSM- 29410

Description:

No Unreviewed Safety Question (USQ) concerns, technical specification changes, or licensing issues have been identified. This modification does not change the functions of any systems, structures or components (SSCs) as described in the FSAR. The deletion of the wet layup (BW) loops from the Main Steam (SM) lines, as shown on the Unit 1 equivalent FSAR drawings does represent a change to the FSAR. The function of the BW system was deleted per NSM MG-22217. The facility will still perform all design, operation and accident mitigation functions as presently described in the SAR, including other SSCs which could be affected by this modification.

This modification does not change methods of operation, or alter a test or experiment as described in the SAR. The BW System was previously disabled, so the removal of the BW instrumentation has no affect on function. The only procedures which could be changed would be those requiring pipe cap verification, if needed.

The removal of interferences described in this modification does not appear significant enough to require inclusion in the FSAR, SLCs, or other SAR documents. The activities involved in this modification do not appear to be of similar importance to those described in the FSAR.

This modification should not adversely affect any SSC that is necessary to operate the facility in accordance with the SAR. Parameters which are being considered to ensure that other SSCs are not affected include cable separation in cable reroutes, seismic considerations in cutting cable trays, and pipe class in the piping changes (addition of tees and pipe caps).

This modification does not involve any tests or experiments not described in the SAR. Post modification testing would only involve proper operation of existing equipment (such as Reactor Coolant Pump 2B) or the leak tightness of pipe runs where tees were installed.

It does not appear that this modification could increase the probability or consequences of an accident evaluated in the SAR. This modification does not create the possibility for an accident of a different type than any already evaluated in the SAR, or the possibility of a malfunction of a different type than any evaluated in the SAR. This modification should not increase the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. No safety limit, setpoint, or operating parameter will be changed by this NSM. The margin of safety as defined in the technical specification bases should not be reduced by this

modification. Rerouted components meet all the present criteria, as do all new components. All new materials are suitable for use in post accident containment. No new interaction consequences have been identified, nor any other adverse affects on SSCs. All separation, independence, and single failure requirements appear to be maintained. No functions will be added or deleted by this modification and no new failure modes will be introduced.

Safety Review and USQ Evaluation:

The Steam Generator Wet Layup Recirculation (BW) System instrumentation and tubing previously disabled under NSM MG-22217 will be removed from the S/G cavities under NSM MG-29410. The taps for this instrumentation on the Main Steam (SM) piping will be removed and capped in accordance with Class B pipe requirements. The BW System isolation and root valves, reservoir, and piping associated with the SM and CF connections will be removed. The BW instruments include transmitters, pressure gauges, pneumatic and electric receiver meters, and pneumatic to electric converters. Associated cables will also be removed involving panel wiring changes, electrical penetration work, and removal of field routed conduits/electray. No panels or penetrations will be degraded by this cable deletion.

Lights to be relocated are part of the normal lighting system and areas will remain sufficiently lit. No-functional cable tray, pie supports, and BW tubing supports at the top of the S/Gs will be removed. Additional non-functional steel attached to the crane wall and empty cable tray running along the crane wall will be removed. The removal of these components will not affect any Structure, system, or Component (SSC) since they are not currently in use.

Some cable tray sections will be cut back. The structural integrity of these cable trays is not affected by the change. The criteria for safety related cables are maintained. Cables routed through cut out sections of tray will be supported by unistruts along the crane wall. The structural integrity of the crane wall is not degraded by the unistrut addition. Rerouted and repulled cables will meet present separation, independence, and Appendix R requirements.

Reactor Coolant (NC) Pump 2B power cables will be replaced with new, rerouted cables. The new cables meet all present criteria for materials, class, separation, independence, seismic, and Appendix R requirements. New and existing NC pump cable trays and supports meet all the present criteria. Removal of existing NC pump cable trays that are not needed will not affect any SSCs.

It is anticipated that subsequent steam generator replacement work will require the rerouting of field routed instrumentation tubing in the S/G cavities. This work will not be performed under this modification, as opposed to the corresponding Unit 1 modification NSM MG-19410, due to a different work plan for Unit 2. If the field routed tubing to be rerouted is safety-related and serves CF instrument loops related to Engineered Safety Features (ESF), the Reactor Trip System (RTS), Process Control Cabinets, S/G level monitoring, and Post Accident Monitoring, it will be necessary to ensure that the reroutes maintain channel independence and separation criteria, and are seismically qualified where required, thus meeting single failure requirements. It will also be necessary to ensure that the response, accuracy, and functional capabilities of the instrumentation are not adversely affected. If such field routed tubing is rerouted below the S/G cavities, a pipe rupture analysis will need to be performed in conjunction with tubing installation and before startup. The satisfactory completion of this analysis will be reflected in the 50.59 evaluation for the reroute. The 50.59 evaluation for this reroute would also need to reflect that the tubing

reroute reserves the abilities of the instruments to perform their safety functions, and that any tubing insulation installed meets design criteria and is securely installed.

The rerouted Makeup Demineralized Water (YM) System piping is the two inch class H line that supplies the decontamination sinks in Containment. The non-safety piping is less than four inches nominal diameter and is therefore not subject to non-seismic interaction review. this reroute does not adversely affect any SSC.

Duke Power Company
McGuire Nuclear Station
Summary of Minor Modifications Completed Under 10CFR50.59

Minor Modification	Valve
3962	1KC0305
3963	1KC0315
3960	1KC0228
3958	1KC0003
3968	1FW0032
3967	1FW0001
3964	1NC0056
3959	1KC0018
3485	2LD0108
3486	2LD0113

DESCRIPTION:

These modifications are being implemented as a result of findings in NRC Bulletin 85-03 "MOV Common Mode Failures During Plant Transients Due to Improper switch Settings", and Duke Power's response to NRC Generic Letter 89-10.

The valve operators are being modified to resolve concerns of not attaining sufficient thrust to completely unseat the valve disc under normal or abnormal differential pressure conditions. Presently, the torque switch bypass contacts are located on the primary switch pack. Due to the switches characteristics, this bypassing action does not stay in the circuit but for a very short period of time, i.e. normally 5% of the total open travel. Due to system conditions with the present open torque switch setup, after the bypass circuit opens, there still may be high enough resistance in the seating area that could cause the valve operator motors to cut off due to insufficient torque. To assure that the valve discs will fully open, the modified torque switch bypass contact has been in the circuit for 50% of the total open valve travel, $\pm$ 25%. This will ensure that maximum motor torque is available for the valve disc unseating action. After the bypass circuit drops out, the torque switch has been in the circuit to deenergize the operator should a high resistance be present after complete unseating. The torque switch bypass circuit has been moved to the add-on-pack auxiliary switches, and the computer points has been moved to the primary switch pack. By moving the computer indication to the primary switches, the computer will provide a more accurate stroke time.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3962 is in the Component Cooling Water System. 1KC0305 serves as a containment isolation valve in the KC supply to the NV excess letdown heat exchanger. The safety function of 1KC0305 is to close and its normal position is closed. 1KC0305 is considered an active valve. Installation of a 50% bypass to the open torque switch

circuit will not affect closure times of 1KC0305 and the existing stress analysis of the piping associated with 1KC0305 will not be affected. Since this Minor Modification will ensure that 1KC0305 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3963 is in the Component Cooling Water System. 1KC0315 serves as a containment isolation valve in the KC supply to the NV excess letdown heat exchanger. The safety function of 1KC0315 is to close and its normal position is closed. 1KC0315 is considered an active valve. Installation of a 50% bypass to the open torque switch circuit will not affect closure times of 1KC0315 and the existing stress analysis of the piping associated with 1KC0315 will not be affected. Since this Minor Modification will ensure that 1KC0315 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3960 is in the Component Cooling Water System. 1KC0228 isolates the Reactor Building non-essential headers and provides component cooling train separation. The safety function of 1KC0228 is to close and its normal position is closed. 1KC0228 is considered an active valve. Installation of a 50% bypass to the open torque switch circuit will not affect closure times of 1KC0228 and the existing stress analysis of the piping associated with 1KC0228 will not be affected. Since this Minor Modification will ensure that 1KC0228 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3958 is in the Component Cooling Water System. 1KC0003 isolates the Reactor Building nonessential headers and provides component cooling train separation. The safety function of 1KC0003 is to close and its normal position is open. 1KC0003 is considered an active valve. Installation of a 50% bypass to the open torque switch circuit will not affect closure times of 1KC0003 and the existing stress analysis of the piping associated with 1KC0003 will not be affected. Since this Minor Modification will ensure that 1KC0003 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3968 is in the Refueling water System. 1FW0032 isolates the flow path from the RWST to the Refueling water pump and isolates gravity flow from the RWST to the Refueling Cavity. The safety function of 1FW0032 is to close and its normal position is closed. 1FW0032 is considered an active valve. Installation of a 50% bypass to the open torque switch circuit will not affect closure times of 1FW0032 and the existing stress analysis of the piping associated with 1FW0032 will not be affected. Since this Minor Modification will ensure that 1FW0032 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3967 is in the Refueling Water System. 1FW0001 isolates the flow path from the RWST to the Refueling Water pump and isolates gravity flow from the RWST to the Refueling Cavity. The safety function of 1FW0001 is to close and its normal position is closed. 1FW0001 is considered an active valve. Installation of a 50% bypass

to the open torque switch circuit will not affect closure times of 1FW0001 and the existing stress analysis of the piping associated with 1FW0001 will not be affected. Since this Minor Modification will ensure that 1FW0001 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3964 is in the Reactor Coolant System. 1NC0056 serves as a containment isolation valve from the reactor makeup water pumps and header to the PRT. The safety function of 1NC0056 is to close and its normal position is open. NC0056 is considered an active valve. Installation of a 50% bypass to the open torque switch circuit will not affect closure times of 1NC0056 and the existing stress analysis of the piping associated with 1NC0056 will not be affected. Since this Minor Modification will ensure that 1NC0056 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3959 is in the Component Cooling Water System. 1KC0018 isolates the Reactor Building non-essential headers and provides component cooling train separation. The safety function of 1KC0018 is to close and its normal position is closed. 1KC0018 is considered an active valve. Installation of a 50% bypass to the open torque switch circuit will not affect closure times of 1KC0018 and the existing stress analysis of the piping associated with 1KC0018 will not be affected. Since this Minor Modification will ensure that 1KC0018 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3485 is in the Diesel Generator Engine Lub Oil System. The function of 2LD0108 is to bypass the Full Flow Lube Oil Filter on high differential pressure. The safety position of 2LD0108 is open and its normal position is closed. The existing seismic analysis of 2LD0108 and the stress analysis of the associated piping will not be affected by the implementation of the 50% torque switch bypass modification. Since this modification will ensure that 2LD0108 will perform as required for design basis system and emergency conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3486 is in the Diesel Generator Engine Lub Oil System. The function of 2LD0113 is to bypass the Full Flow Lube Oil Filter on high differential pressure. The safety position of 2LD0113 is open and its normal position is closed. The existing seismic analysis of 2LD0113 and the stress analysis of the associated piping will not be affected by the implementation of the 50% torque switch bypass modification. Since this modification will ensure that 2LD0113 will perform as required for design basis system and emergency conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

MINOR MODIFICATION	**VALVE**
4158	1RN0042
4156	1RN0040
4152	2RN0040
3956	1RN0018
3930	1RN0299

MINOR MODIFICATION	VALVE
3984	2RN0086
4159	1RN0043
4161	1RN0063
3936	0RN0013
3914	1KC0002
4126	2RN0279
4127	2RN0296
4130	2RN0299
4154	2RN0042
6310	1WL0321, 1WL0322
4153	2RN0041
4160	2RN0187
4155	2RN0043
4157	1RN0041
3929	1RN0297
3924	1RN0187
3927	1RN0279
3954	1KC0050, 1KC0053, 1KC0338, 1KC0424, 1KC0425

DESCRIPTION:

The diagnostic portion of these Minor Modifications involve resetting the open and close torque switches so that the motor operator will produce the necessary torque to fully open and/or fully close the valve disc when design basis systems conditions are present. Resetting the torque switches is accomplished on a torque test bench and the criteria used for the bench test, the minimum required and maximum allowed torque have been determined by Engineering Calculation MCC-1205.19-00-002 and are provided by controlled document MCM 1205.19-0039-001. This Engineering Calculation was performed in accordance with the latest revision of Duke Power Specification DPS-1205.19-00-0002 which establishes the parameters and criteria used to determine the minimum required and maximum allowed torque levels. The inaccuracies of the diagnostic test system used to facilitate torque testing have been included in the Engineering Calculation. The final output torque level achieved during the diagnostic bench test has been sufficient to allow valve operation at design differential pressure.

TORQUE SWITCH BYPASS:

This portion of the Minor Modifications involve removing the opening torque switch from the control logic for a larger portion of the opening travel. Presently, the torque switch bypass does not cover the new requirement of 90 % $\pm$ 5% of the opening travel. Under the current open torque switch logic, the valves are required to perform the safety function and after the open torque switch bypass opens, there may still be high enough resistance at the valve disc to cause the actuator motor to deenergize leaving the valve disc in mid-stroke. This high resistance can occur at unseating or at some mid-stroke position. The high resistance is basically a torque requirement that is due to unseating and/or hydrodynamic factors. With the open torque switch in the circuit during this high torque time, full capabilities of the motor are not available and failure to satisfy the safety function is a possibility. To assure the valves will fully open, the modified open torque switch bypass contact has been in the opening circuit for a minimum of

90% of open travel with a set-up tolerance of $\pm$ 5%. This will ensure maximum motor torque is available to be applied to the opening stroke. After the bypass circuit drops out, the open torque switch has been in the circuit to deenergize the actuator should the controlling limit switch fail.

Functionally, the valves will operate identical to its present operation. This switch modification will not affect open or closing stroke times. With this modification, the valves has been more reliable in obtaining the desired positions.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 4158 is in the Nuclear Service Water System. 1RN0042 provides isolation of RN Train supply to the Auxiliary Building non-essential header. The safety function of 1RN0042 is to close and its normal position is open. 1RN0042 is considered an active valve. Resetting the open and close torque switches as well as installing the 90% $\pm$ 5% open torque switch bypass will not affect open and closure times of 1RN0042. The existing stress analysis of the piping associated with 1RN0042 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0042 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4156 is in the Nuclear Service Water System. 1RN0040 provides isolation of RN Train supply to the Auxiliary Building non-essential header. The safety function of 1RN0040 is to close and its normal position is open. 1RN0040 is considered an active valve. Resetting the open and close torque witches as well as installing the 90% $\pm$ 5% open torque switch bypass will not affect open and closure times of 1RN0040. The existing stress analysis of the piping associated with 1RN0040 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0040 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4152 is in the Nuclear Service Water System. 2RN0040 provides isolation of RN Train supply to the Auxiliary Building non-essential header. The safety function of 2RN0040 is to close and its normal position is open. 2RN0040 is considered an active valve. Resetting the open and close torque witches as well as installing the 90% $\pm$ 5% open torque switch bypass will not affect open and closure times of 2RN0040. The existing stress analysis of the piping associated with 2RN0040 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0040 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3956 is in the Nuclear Service Water System. 1RN0018 provides isolation of RN Train supply to the RN pump. The safety function of 1RN0018 is to open and its normal position is open. 1RN008 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % $\pm$ 5 % open torque switch bypass will not affect open and closure times of 1RN0018. The existing stress analysis of the piping associated with 1RN0018 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0018 will perform as required for

design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3930 is in the Nuclear Service Water System. 1RN0299 provides isolation of Unit 1 RV Auxiliary Building Vent units discharge to RN system. The safety function of 1R0299 is to close and its normal position is open. 1RN0299 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1RN0299. The existing stress analysis of the piping associated with 1RN0299 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0299 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3984 is in the Nuclear Service Water System. 2RN0086 provides isolation of RN A Train supply to the KC heat exchanger. The safety function of 2RN0086 is to open and its normal position is open. 2RN086 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 2RN0086. The existing stress analysis of the piping associated with 2RN0086 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0086 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4159 is in the Nuclear Service Water System. 1RN0043 provides isolation of RN B Train supply to the non-essential header. The safety function of 1RN0043 is to close and its normal position is open. 1RN0043 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 %± 5 % open torque switch bypass will not affect open and closure times of 1RN0043. The existing stress analysis of the piping associated with 1RN003 will not be affected by resetting the open and close torque switches. Since this minor Modification will ensure that 1RN0043 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOVs affected by Minor Modification 4161 is in the Nuclear Service water System. 1RN0063 provides isolation of the Auxiliary Building RN non-essential header. The safety function of 1RN0063 is to close and its normal position is open. 1RN0063 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1RN0063. The existing stress analysis of the piping associated with 1RN0063 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0063 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3936 is in the Nuclear Service Water System. 0RN0013 provides isolation of Low Level Intake supply to RN A train. The safety function of

0RN0013 is to open and its normal position is open. 0RN0013 is considered an active valve. Resetting the close torque switch as well as disabling the open torque switch completely from the opening travel will not affect open and closure times of 0RN0013. The existing stress analysis of the piping associated with 0RN0013 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 0RN0013 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor modification 3914 is in the Component Cooling System. 1KC0002 provides isolation to the auxiliary Building non-essential headers and provide component cooling train separation. The safety function of 1KC0002 is to close and its normal position is closed. 1KC0002 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1KC0002. The existing stress analysis of the piping associated with 1KC0002 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1KC0002 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4126 is in the Nuclear Service Water System. 2RN0279 provides isolation of Unit 2 Auxiliary Building Vent units discharge to RN system. The safety function of 2RN0279 is to close and its normal position is open. 2RN0279 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure items of 2RN0279. The existing stress analysis of the piping associated with 2RN0279 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0279 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4127 is in the Nuclear Service Water System. 2RN0296 provides isolation of RN A Train essential header discharge. The safety function of 2RN0296 is to open and its normal position is open. 2RN0296 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % + 5 % open torque switch bypass will not affect open and closure times of 2RN0296. The existing stress analysis of the piping associated with 2RN0296 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0296 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4130 is in the Nuclear Service Water System. 2RN0299 provides isolation of Unit 2 Auxiliary Building Vent units discharge to RN system. The safety function of 2RN0299 is to close and its normal position is open. 2RN0299 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 2RN0299. The existing stress analysis of the piping associated with 2RN0299 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0299 will perform as required for design basis system conditions, the probability or consequences of

an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4154 is in the Nuclear Service Water System. 2RN0042 provides isolation of RN Train supply to the Auxiliary Building non-essential header. The safety function of 2RN0042 is to close and its normal position is open. 2RN042 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 2RN0042. The existing stress analysis of the piping associated with 2RN0042 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0042 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOVs affected by Minor Modification 6310 are in the Liquid Waste Recycle System. 1WL0321 and 1WL0322 are containment isolation valves for the Ventilation Unit Condensate Drain Tank inputs. The safety function of 1WL0321 and 1WL0322 is to close and their normal position is open. 1WL0321 and 1WL0322 are considered active valves. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1WL0321 and 1WL0322. The existing stress analysis of the piping associated with 1WL0321 and 1WL0322 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1WL0321 and 1WL0322 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4153 is in the nuclear Service Water System. 2RN0041 provides isolation of RN B Train supply to the non-essential header. The safety function of 2RN0041 is to close and its normal position is open. 2RN0041 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 2RN0041. The existing stress analysis of the piping associated with 2RN0041 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0041 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4160 is in the Nuclear Service Water System. 2RN0187 provides isolation of RN B Train supply to the KC Heat Exchanger. The safety function of 2RN0187 is to open and its normal position is open. 2RN0187 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 2RN0187. The existing stress analysis of the piping associated with 2RN0187 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0187 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4155 is in the Nuclear Service Water System. 2RN0043 provides isolation of RN B Train supply to the non-essential header. The safety

function of 2RN0043 is to close and its normal position is open. 2RN0043 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 2RN0043. The existing stress analysis of the piping associated with 2RN0043 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0043 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 4157 is in the Nuclear Service water System. 1RN0041 provides isolation of RN B Train supply to the non-essential header. The safety function of 1RN0041 is to close and it's normal position is open. 1RN0041 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1RN0041. The existing stress analysis of the piping associated with 1RN0041 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0041 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3929 is in the Nuclear Service Water System. 1RN0297 provides isolation of RN B Train essential header discharge. The safety function of 1RN0297 is to open and its normal position is open. 1RN0297 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1RN0297. The existing stress analysis of the piping associated with 1RN0297 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0297 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3924 is in the Nuclear Service Water System. 1RN0187 provides isolation of RN B Train supply to the KC Heat Exchanger. The safety function of 1RN0187 is to open and its normal position is open. 1RN0187 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1RN0187. The existing stress analysis of the piping associated with 1RN0187 will not be affected by resetting the open and closure torque switches. Since this Minor Modification will ensure that 1RN0187 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOV affected by Minor Modification 3927 is in the Nuclear Service Water System. 1RN0279 provides isolation of Unit 1 Auxiliary Building Vent units discharge to RN system. The safety function of 1RN0279 is to close and its normal position is open. 1RN0279 is considered an active valve. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1RN0279. The existing stress analysis of the piping associated with 1RN0279 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1RN0279 will perform as required for design basis system conditions, the probability or consequences of

an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

The MOVs affected by Minor Modification 3954 are in the Component Cooling System. 1KC0059 and 1KC0053 isolate the Aux Building non-essential headers and provide component cooling equipment train separation. 1KC0338, 1KC0424, and 1KC0425, Containment isolation valves in the KC supply to the Reactor Vessel Support Coolers. These valves are considered to be active valves. Resetting the open and close torque switches as well as installing the 90 % ± 5 % open torque switch bypass will not affect open and closure times of 1KC0050, 1KC0053, 1KC0338, 1KC0424, and 1KC0425. The existing stress analysis of the piping associated with these valves will not be affected by resetting the open closure torque witches as well as installing the 90 % ± 5 % open torque switch bypass. Since this minor Modification will ensure that these valves will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQs exists.

MM-5033

Description:

The purpose of MEVN-5033 is to replace the existing DG 2A and 2B Fuel Oil Storage Tank (FOST) Level Indication Instrumentation. The past configuration was based on a sensed pressure using a filled capillary gauge with switches. The new configuration uses a Monitor and Gauging System, which is based on sensing product level using a probe/float arrangement. Technical Specifications require a minimum usable volume of Fuel Oil in each storage tank of 39,500 gallons as a limiting condition for operation for the corresponding diesel. The new instrumentation will provide a low level alarm at a level corresponding to usable tank volume of approximately 41,000 gallons. The proposed instrumentation has accuracy and repeatability that exceed the existing instrumentation. The proposed instrumentation is used extensively in the chemical and petroleum industry. This instrumentation does not perform a safety function, and in the proposed application, does not increase the probability of a malfunction of equipment important to safety.

Safety Review and USQ Evaluation:

The accident discussed in the FSAR concerning the FOST considers a tank rupture. The removal of the tank cover degrades the protection of the tank from t0RNado generated missiles. However, the provisions that has been taken as discussed on the attached page will prevent the probability and thus the consequences from being increased. There are no other possibilities of an accident created, as the accident that is addressed, is one in the same. Since the level inventory has been monitored and appropriate precautions are being taken to monitor weather, the probability of a malfunction of the D/G is not increased, and therefore the consequences are not increased as well. The malfunction of a different type than has already been evaluated in the FSAR does not exist while performing this modification. The ability to start redundant D/Gs or to cross-tie FOSTs will still exist. No USQ exists.

MM-3833

Description:

The purpose of this Minor Modification is to document the use of the VL System as the mechanism for cooling the Reactor Vessel Support Coolers and associated concrete. In addition, since the KC System will no longer perform this function, low KC Flow Rate alarms has been permanently silenced.

Safety Review and USQ Evaluation:

The possibility, probability, and consequences of an accident or malfunction of equipment is not increased as a result of the implementation of this Minor Modification. A LOCA of the NC System due to concrete strength degradation from long term exposure to temperatures greater than 150°F would not be expected since design documentation exists indicating that the VL System is capable of maintaining acceptable concrete temperatures. The Reactor and its supports are not systems used to mitigate accident consequences. The safety function of the concrete and vessel supports will not be compromised as shown by analysis. No common failure modes are created because redundant trains of safety equipment are not involved. The VL System is capable of maintaining the Reactor Vessel Head region at a temperature of 135°F or less.

Silencing the Operator Aid Computer "low KC Flow Rate to RV Support Coolers" alarms has no significant effect on the design or function of the KC System. System performance will not be compromised. The margin of safety defined in the basis for any Technical Specification will not be reduced because no Technical Specifications has been changed and no operating plant parameters (i.e., concrete temperatures, VL & KC System operating characteristics) has been significantly affected

Justification for performing this Minor Modification is provided by calculation MC-1223.23-00-0019 and in the Operability Evaluation performed for PIP 2-M93-0902. No USQ exists.

MM-5476

Description:

The molded case switches has been installed in DC panelboards 1EVDA, 1EVDB, 1EVDC and 1EVDD. These panelboards are part of the 125VDC Vital Instrumentation and Control Power System.

Safety Review and USQ Evaluation:

The switches look like a molded case circuit breaker, but they do not have overcurrent trip capability. They has been protected by thermal magnetic circuit breakers in the distribution center from long time and instantaneous overcurrents. The available fault current is less than the switch rating.

The addition of these devices will not increase the possibility or the consequences of an accident evaluated in the SAR. Since the construction is identical to the circuit breakers, the probability

of a malfunction would not be increased and any consequences of a malfunction are already included in the existing analysis.
The installation of this switch will not reduce the margin of safety at the plant. No USQ exists.

MM-5453, 5454, 5455, 5456, 5457, 5458, 5459, and 5460

Description:

The main steam system is designed to remove heat generated by the nuclear fuel in the Reactor Coolant System. Main steam isolation valves (MSIVs) are provided in each steam generator line immediately downstream of the code safeties to isolate each individual steam generator and prevent reverse flow in the event of a steam line rupture. These four valves close on a high-high Containment pressure signal and/or on a high steam line pressure rate of change signal or a low steam line pressure signal. The intent of the low steam line pressure signal closure was to isolate the generators following a rupture of the main steam line between the steam generators and the turbine steam stop valves. However, in the event of a LOOP, these valves may actuate due to low steam line pressure resulting from steam blowing down through the steam drain lines downstream of the MSIVs.

There are sixteen (16) drain valves located downstream of the MSIVs. These valves are placed in low points in the steam lines upstream of the turbine steam stop valves and are designed to keep water out of the turbines. Currently, these valves fail open upon a loss of offsite power (LOOP). This modification will replace the normally closed solenoids on the 16 drain valves with normally open solenoids so that they will fail closed in the event of a LOOP. Air for these solenoids is supplied from the VI common header, a non-safety component that supplies air to both Unit One and Unit Two. In the event of a dual unit LOOP, the secondary steam system may remain pressurized an additional five to eight minutes, providing the Control Room Operators with additional time to throttle CA flow, thereby potentially averting a low steam pressure alarm and subsequent main steam isolation. For a Unit-specific LOOP, approximately one hour minimum could be gained for secondary side pressurization, depending on the air compressor lineup feeding the common VI header. Closing these drain valves will minimize the likelihood of receiving a secondary side low steam pressure alarm, which could result in a safety injection (SI) and steamline isolation.

Safety Review and USQ Evaluation:

The subject drain valves do not act as accident initiators or mitigators for any accidents analyzed in Chapter 15 of the McGuire FSAR. The purpose of this modification is to provide the operator more time in responding to a LOOP event so that a low steamline pressure alarm and subsequent safety injection signal might be averted. No credit is taken for these valves in any accident analyzed in Chapter 15 of the McGuire FSAR. For a LOOP event, they will provide an additional five to eight minutes of pressure in the secondary system prior to the valves reopening due to VI header pressure bleeding off. This additional time may provide the operators enough time to throttle CA in order to avert a safety injection and secondary side isolation. Therefore, this modification does not involve an increase in the probability or consequences of an accident previously evaluated in the SAR.

The drain valves are not nuclear safety related components. The replacement solenoids that will fail closed are not nuclear safety related components and are equivalent replacements for the

original solenoids. The fail-closed solenoids are being installed to provide additional time for operators to respond to a LOOP event and avert a potential overcooling event. The fail-closed solenoids do not create the possibility for an accident or a malfunction of a different type than any evaluated in the SAR.

There is no increase in the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. As discussed above, this modification is only replacing the existing fail-open solenoids with equivalent fail-closed solenoids. Failing these valves closed on a LOOP can potentially allow water to enter the turbines, which could result in damage to the turbines. This is seen as an economic risk only. Since the turbine trips following a LOOP, then it cannot act as an accident initiator during this scenario. Any water entering the turbines following the LOOP could only cause an economic impact on the operation of the plant. No changes are being made that will affect any equipment in a non-conservative manner.

There are no changes of safety limits, setpoints, or plant parameters as a result of this modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

MM-5483 and 5484

Description:

This modification adds analog computer points for the level in each Containment Floor & Equipment Sump. This modification will improve the capability of detecting a 1 gpm leak into these sumps as outlined in Reg. Guide 1.45, Tech Spec. 3.4.6.1 and FSAR Section 5.2.7. The OAC will use these sump level values to calculate leakage rate and alert operators when a 1 gpm or greater leak is present cumulative within 1 hour .

Safety Review and USQ Evaluation:

The OAC is not an accident initiator; modification does not increase the probability or increase the consequences of an accident evaluated in the SAR. It does not create the possibility of an accident of a different type than any evaluated in the SAR. The OAC does not affect the way any equipment operates; therefore the modification does not increase the probability or increase the consequences of a malfunction of equipment important to safety evaluated in the SAR. It does not create the possibility for an equipment malfunction of a different type than already evaluated in the SAR. The modification does not reduce the margin of safety as defined in the basis for any technical specification. Containment Floor and Equipment Sump Level is not safety related and the addition of an OAC point in these loops does not affect any safety-related systems. No Unreviewed Safety Questions are introduced by this modification.

This modification will improve the ability to detect unidentified reactor coolant leakage. The modification does not change the operation of any equipment or cause any new action to be taken. The addition of these OAC points should improve the ability to see failure of fission product barrier (RCS) and thus improve the margin of safety. No USQs exist.

MM-5480

Description:

The EMF44 loss of flow alarm has been interlocked with the existing flow instrumentation WLFE5900. This instrumentation monitors the flow in the common discharge line just downstream of the recirculation line for the Ventilation Unit Condensate Drain Tank. Therefore, any discharge from the Ventilation Unit Condensate Drain Tank passes through this flow element.

WLFE5900 produces a 4-20 mA signal that corresponds to a flow of 0-150 GPM. This signal has been passed through an alarm module which will activate the EMF44 loss of flow alarm at a point very near the bottom of the scale (≈4mA) on WLFE5900. The controlling procedure for Ventilation Unit Condensate Drain Tank releases requires a release in the range of 30 to 60 GPM. Therefore, the "actuation point" for the EMF44 loss of sample flow alarm has been at a very conservative position.

Safety Review and USQ Evaluation:

This modification has no affect at all on any of the accident scenarios evaluated in Ch. 15 of McGuire's FSAR. This equipment is not important from a Nuclear Safety perspective. Nor does this modification negatively impact the operation of EMF44 as described in section 11.4.2.1.8 of FSAR. With the conservative setpoint discussed in the preceding paragraph, no unmonitored release has been possible. Therefore, no unresolved safety question exists.

MM-5446

Description:

This MM revises PLS to allow Power Range Lower and Upper deviation alarms to be set based on actual plant operating variations. MNS has experienced intermittent nuisance alarms in the past so this change will allow setpoint changes more appropriate to operating conditions. Later revisions of the PLS by Westinghouse at newer plants include this provision.

Safety Review and USQ Evaluation:

Tech. Spec. 4.2.4.1 addresses QPTR surveillance requirements and speaks to alarm operability. The OAC Nuclear 06 program provides this alarm function. The NIS system, specifically the Power Range Lower and Upper Deviation alarms (annunciators in Control Room) are used ONLY for supplemental information and not to meet any Tech. Spec. surveillances. Associated annunciator response requires operators to check the OAC alarm/deviation status. Therefore, revision of setpoints has no effect on Tech. Specs. Since no credit is taken for these alarms, no FSAR accident scenarios are affected or introduced. Additionally, FSAR equipment is not affected. No USQ exists.

MM-5432

Description:

McGuire Unit One incore flux thimble location D03 is bent and inaccessible to a neutron detector or eddy current probe. This thimble was damaged during an inspection in which the incore flux detector was removed. Since the thimble is inaccessible, a decision has been made to cap the thimble location to prevent any future damage and potential leakage from the reactor coolant system. While not a concern at the present, thimble wall thinning has been identified as a generic industry concern in all Westinghouse PWRs that utilize bottom mounted instrumentation. In general, thimble tubes can experience thinning as a result of flow-induced vibrations at locations associated with geometric discontinuities or area changes along the flow path (such as areas near the lower core plate, the core support forging, the lower tie plate, the upper tie plate, and the vessel penetration).

The isolation valves associated with each thimble location are not nuclear safety-related components. A breach in a damaged thimble can be considered as a Small Break LOCA (SB-LOCA). Therefore, in the event of a breach of the damaged thimble, the isolation valve at the seal table would become essentially a part of the reactor coolant pressure boundary. Not being a nuclear safety-related component, this valve would not be qualified to fulfill this function. Therefore, the decision has been made to remove the tube and fittings between thimble D-3 and its isolation valve, close off its isolation valve, and cap the thimble at the seal table with a QA-1 qualified cap fitting. This seal would serve as an acceptable component of the reactor coolant pressure boundary in the event of failure of the thimble.

In addition to capping thimble D-3, this safety review will generically discuss plugging of damaged thimbles. There are 58 incore guide thimbles in McGuire Unit One. Each thimble is manufactured from cold-worked stainless steel with a 0.201 inch internal diameter and a 0.049 inch wall thickness. Technical Specification 3.3.3.2 states the limiting conditions for operation for the movable incore detection system. In general, 75% of the detector thimbles must be operable at all times with a minimum of two detector thimbles per quadrant operable. Given the total of 58 incore guide thimbles, a total of 44 guide thimbles must be operable at all times. Therefore, a total of 14 guide thimbles can be placed out-of-service at any one time. The maximum number of guide thimbles that can be capped in the various quadrants are as follows:

Quadrant Description	Total Thimbles	Maximum # of Capped Thimbles
Quadrant One Interior Thimbles	11	9
Quadrant Two Interior Thimbles	10	8
Quadrant Three Interior Thimbles	11	8
Quadrant Four Interior Thimbles	11	9
Axis Thimbles on Q1-Q2 Boundary	4	2
Axis Thimbles on Q2-Q3 Boundary	4	2
Axis Thimbles on Q3-Q4 Boundary	3	1
Axis Thimbles on Q4-Q1 Boundary	4	2

Note that these totals are for any one quadrant. Furthermore, the maximum number of capped thimbles listed above is very conservative for any one quadrant since the axis thimbles can be counted with the interior thimbles, providing more than two thimbles for any quadrant, as

required by Technical Specification 3.3.3.2. No more than 14 guide thimbles total from all four quadrants may be placed out-of-service at any one time in order to maintain operability per Technical Specification 3.3.3.2.

Safety Review and USQ Evaluation:

The permanent capping of damaged incore guide thimbles with QA-1 qualified cap fittings does not involve an increase in the probability or consequences of an accident previously evaluated in the SAR. The movable incore flux detection system is used only to provide confirmatory information on the neutron flux distribution and is not required for the day-to-day operation of the plant. The system has been evaluated as a SB-LOCA for the unlikely scenario in which a breach in a thimble wall occurs. However, the postulated breach would only allow reactor coolant into the thimble and not past the seal table, where the QA-1 qualified cap fitting would maintain the reactor coolant pressure boundary. Therefore, a postulated SB-LOCA transient within the movable incore flux detection system is bounded by the current Chapter 15 SB-LOCA analysis.

The isolation valves supplied for each of the 58 incore detector flux thimbles are not nuclear safety related components. Therefore, in the event of a breach of a thimble due to damage or wall thinning, the isolation valve is not qualified to perform its function of reactor coolant pressure boundary. The addition of the QA-1 qualified cap fittings does not constitute the possibility for an accident or a malfunction of a different kind than any evaluated in the SAR. The cap fittings are installed to provide additional assurance that the reactor coolant pressure boundary will remain intact in the unlikely event of a breach of a thimble wall.

There is no increase in the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. As discussed above, this modification is only adding QA-1 qualified cap fittings to ensure that the reactor coolant pressure boundary integrity is not compromised during the unlikely event of a breach of a thimble wall. No changes are being made as a result of this modification that will affect any equipment in a non-conservative manner. The QA-1 qualified cap fittings act as a passive safety system to provide additional levels of assurance that the reactor coolant pressure boundary has been maintained at the seal table.

There are no changes of safety limits, setpoints, or plant parameters a result of this modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical Specifications is not decreased. No USQ exists.

MM-5360

Description:

This minor modification installs pneumatic snubbers on the impulse lines for flow transmitters 1NDFT5181 and 5191, which provide Train A and B ND Flow signal to the process control system to provide flow indication and low flow alarm. The snubbers will filter out process noise during drain down conditions, which results in erroneous low flow alarms.

Safety Review and USQ Evaluation:

The snubbers and the piston selected are sized so that they will not affect the integrity of the flow signal. Therefore, the snubbers will not affect the ability of the instruments to perform their intended function, nor affect/result in an additional accident scenario. These instruments do not provide a safety function. The impulse lines, however, are classified as safety related since they are tapped off the ND lines. This minor mod does not affect the integrity of the impulse lines.

This minor mod has no effects on any accident equipment function addressed in the SAR nor does it affect any safety margins. No USQ exists.

MM-5354

Description:

The cover installed on the motor starter for EMF52 is for safety reasons. As covered in PIP 0-M93-0854, the starter electrical contacts were accidentally contacted with a flexible hose. Installation of the electrical box, MMIS id # 0343-1016, will eliminate the possibility of accidental contact with the electrical contacts.

Replacement of the FNA-5A fuse with FNQ-10A is to allow for proper operation of the sample pump motor. This motor draws approximately 27 amps starting current. During testing, the FNA-5A fuse was replaced by FNQ-5A (FNA type fuses no longer used). The FNQ-5A fuse repeatedly blew when starting the motor. The FNQ-10A is suitable for this application (hook-up wire used in this application is 16 AWG).

Safety Review and USQ Evaluation:

These changes do not increase the probability or consequences of an accident evaluated in SAR since they do not affect the normal operation of the EMF52 monitor. EMF52 is only used for monitoring purposes. It is not an accident initiator. No USQ exists.

MM-4116

Description:

The Feedwater Control Valves are the final control elements used by the Feedwater Control system to modulate the feedwater flow to the steam generators and hence control the steam generator water level. On a feedwater isolation signal, the valves are closed to provide back-up isolation for the Feedwater Isolation Valves.

The actuators on the Feedwater Control Valves are reverse acting air to open, spring to close air operators provided by Copes-Vulcan. Air is applied to the top of the actuator to open the valve, while a spring provides the motive force to close the valve on loss of air.

The design of the actuator permits the yoke to move within the stationary frame, loosely guided by four spacers. This modification will position four guide bushings in the diaphragm housing that will tighten the guidance provided by the spacers.

Safety Review and USQ Evaluation:

This modification will not increase the probability or consequences of an accident since it does not change the operating characteristics of the feedwater control valves.

The possibility for an accident of a different type is not created since the Feedwater Control Valve is not the primary device to isolate the feedwater on the feedwater isolation signal. This function is covered by the Feedwater Isolation Valves with backup being provided by the Feedwater Control Valves and/or tripping the Feedwater Pump Turbine.

This modification will not increase the probability or consequences of a malfunction of equipment important to safety since the Feedwater Control Valve is not the primary feedwater isolation device.

This modification does not create any new type of malfunction that did not already exist. No USQ exists.

MM-4109

Description:

Roof repairs to the Unit 1 Turbine Building roof are necessary due to the failure of several mechanical fasteners caused by high wind loads. The scope of work for this modification is to replace the roof with a new roofing system. The existing roof has been removed to the deck and the deck repaired as necessary. The new roof assembly shall consist of a mechanically attached layer of isocyanurate insulation with a wood fiber layer adhered in cold adhesive and a cold adhered, modified bituminous membrane. The surface will then be white granule surfacing. The parapet wall has been waterproofed and handrails added as needed for safety reasons. Details of the actual changes to the roof structure are in McGuire Nuclear Station, Roof Repair Specifications for Turbine Building #1, No. 7461, May 1994.

The repairs to the Unit One Turbine Building roof has been made in accordance with specifications and any site safety requirements.

The Turbine Building is not a safety related structure; however, this building does house structures, systems, or components important to safety. A 50 ton crane has been used to lift materials to and from the roof area. This size crane is needed for its boom length, not for lifting excessively heavy loads. The crane has been set up on the west side of the Unit One Turbine Building (near the roll up doors at the truck bay). The center pin location has been 40 feet down and 50 feet out from the southwest cORNer of the building. The crane location and footing have been analyzed to ensure that no damage to underground components and structures (specifically, the Condenser Cooling Water intake pipes) will occur and that all required loads can be moved without compromising the safe operation of the plant. For this modification, the crane is the primary concern for loading. The actual material to be lifted is much lighter. Material has been lifted to and from a predetermined location on the roof. If the crane is to be unused for any extended period, the boom will either be secured or retracted to prevent the potential for damage due to seismic or wind activity. The civil group has reviewed the crane setup configuration for potential failure and subsequent damage to plant equipment. The 6900/5000V 1ATC transformer is under the potential boom swing path on the 760 elevation. Due to the construction of the Turbine Building, along with the operational characteristics of the hydraulic crane, the

concern for crane failure and subsequent damage to equipment in the building is alleviated. In addition, the crane has been analyzed for failing in the direction of the 230kV transmission station on the south end of the Turbine Building. However, since the crane is being shored up under each footing, the concern for tipping is alleviated.

Because of the large amount of material to be removed from the roof, a temporary trash chute has been installed on the roof during this modification work. The trash chute has been self supporting from a ramp temporarily installed on the roof to facilitate dumping into the chute. The chute has been secured to the turbine building wall to prevent excessive movement during high winds. The roof loading from this equipment had been reviewed and determined to be acceptable.

Part of the modification will involve the removal of the coping that covers the parapet wall around the circumference of the Turbine Building roof. The standard piece of coping is 9'11" long and is made of aluminum. As the Turbine Building roof is subject to substantial winds, there is a concern that the coping material could be dropped over the edge of the roof and interact with the 230kV lines located on the south end of the Turbine Building. Administrative controls has been implemented to ensure that no materials fall over the edge of the roof. The 230kV lines are approximately 20 feet apart. While a piece of coping could interact with one line, there is no potential for arcing between lines and subsequent damage to plant equipment.

Safety Review and USQ Evaluation:

The administrative controls incorporated during the implementation of this modification ensure that there has been no potential for a mishandling event which could lead to a turbine trip event as discussed in Section 15.2.3 of the FSAR. Therefore, there is no increase in the probability for this accident or any others discussed in the FSAR. No systems or components which are required to be operable during the implementation of this modification has been jeopardized by implementation tasks. Therefore, the consequences of accidents analyzed in the FSAR are not increased by the proposed modification.

The construction of the Turbine Building has been reviewed to ensure that a boom failure would not penetrate the building and damage any equipment important to safety. Therefore, the implementation of this modification does not create the possibility of a new accident due to movement of equipment or materials. The modification of the roofing system will improve the overall performance of the roofing system on the Turbine Building. Therefore, the modification will not create the possibility of an accident of a different type than previously considered.

Implementation of the modification will not cause any interaction with safety related equipment needed for the safe operation of the plant or for accident mitigation. The new roofing system will improve the performance of the Turbine Building roofing system. Therefore, the probability of equipment malfunction is not increased for this modification.

Administrative controls has been used to alleviate the potential for equipment and material handling accidents during the implementation of this modification. The new roofing system will perform its function in the same manner as the original system. Therefore, the consequences of a malfunction of equipment are not increased by this modification.

Neither the installation or the operation of the new roofing system will create the possibility of a new malfunction of equipment different than that previously considered. The operation of the

new roofing system is not different from the original and the expected performance of the new roofing system is expected to be better than the current roofing system due to improved materials.

The margin of safety as defined in the Tech Specs is related to the confidence in the fission product barriers. This modification does not cause any interactions with any fission product barriers. Therefore, the modification does not reduce the margin of safety as defined in the basis to the Technical Specifications. No USQ exists.

MM-4106

Description:

The modification deletes the existing double packed packing set with lantern ring and leak off connection and replaces it with a five ring "live loaded" packing set for main steam power operated relief valve (PORV) 1SV0007. The purpose of modifying the packing configuration from a double packed to a single packed arrangement is to 1) reduce packing drag in order to decrease friction and improve valve stroke time; 2) improve packing seal to diminish potential for development of stem packing leakage; and 3) compensate for in-service consolidation of the packing by means of the "live loading." The superior performance of a five ring packing configuration with "live loading" as compared to the traditional double packed arrangement with leak off has been well researched and tested as documented in EPRI Report NP-5697 (Valve Stem Packing Improvements).

Deletion of the leak off connection will consist of uncoupling the packing leak off connection from the leak off drain line and installing cap per Pipe Specification PS900.2 for line listing SV01. The leak off connection is 1/2 inch schedule 80 carbon steel pipe and is more than adequate for any pressure stresses imposed:

Stress = (pressure x diameter)/(2 x thickness) = 3386 psig
where pressure = 1185 psig (per valve design pressure)
diameter = 0.840" (per Crane Technical Paper 410)
thickness = 0.147" (per Crane Technical Paper 410)

Currently, the stem packing leakoff lines for the S/G PORVs are piped to a header that opens to floor drains in the Interior/Exterior Dog Houses. The floor drains are piped to the WZ sumps. This routing of the leakoff is desirable in the event of a primary packing leak when a primary system to secondary system leak exists, because it provides for better containment of the potentially contaminated leak-off (condensed steam). However, this modification will provide for a better valve packing arrangement, which decreases the potential for development of stem packing leakage.

Safety Review and USQ Evaluation:

The PORVS serve to relieve pressure and dissipate heat in the Steam Generators when necessary by opening on an over pressure signal in the associated steam line and venting steam to atmosphere. The valves close to provide containment isolation in conjunction with the main steam isolation valves per GDC-57. The PORVs operation is considered in accident analysis. Accident analysis assumes that these valves fail to close or open in an accident depending on the severity of the failure to the specific accident. The stroke time of the PORVs is maintained

below 60 seconds by procedure but failure of the valve to close in this amount of time does not render the valve inoperable. Complete failure of the PORV to close is a violation of Technical Specification 3.6.3. Block valves for the PORVs are closed and de-energized when the PORV is inoperable. From an accident standpoint, credit is taken for manually closing and de-energizing the PORV (block) isolation valves. No USQ exists.

MM-4085

Description:

This MM will remove a safety hazard by re-routing the WL Drain Header for AHU condensate drains to a higher elevation across the walkway on elevation 750'. The elevation for the pipe across the walkway is 755'-4" and this MM will re-route it to 757'-6". This WL drain header delivers drainage to the VUCDT from the following AHUs:

VC AHUs (CR-AHU-2, CR-AHU-1, CRA-AHU-1, and CRA-AHU-2),
VA AHUs (Count Room Air Cond. Unit, and Aux, Bld.
Supply AHUs 1A, 2A, 1B, and 2B),
VA AHUs North of QQ (Radwaste Area Vent. Supply Units
3, 47, 108 and 109, CHM-AHU-1, and the Vent
Stack Drain for the Radwaste Release Point)
VF AHUs (Fuel Handling Area Supply Air Fans 1 and 2).

Safety Review and USQ Evaluation:

This change to the WL drain header will not affect the function of the system. The piping reroute will still be maintained for drainage into the Liquid Radwaste system. Also, normally the activity from these AHUs are well below permissible levels for discharge. There is no direct impact on safety functions that are described in the design basis events.

Therefore, no safety systems has been degraded or altered as previously evaluated in the FSAR/Tech. Specs. This MM will not increase the probability of a failure of safety related equipment and no unreviewed safety question is created as a result of this MM.

MM-4018

Description:

The purpose of this modification is to install flow instrumentation and a flow indicator in the discharge piping of the "D" VI compressor.

Safety Review and USQ Evaluation:

The compressor instrumentation has been used to measure the discharge flow rate of the compressor for the purpose of trending compressor performance and troubleshooting problems with the compressor. The remaining two centrifugal compressors and the three reciprocating compressors has been available to supply air to loads served by the system.

Based on the above scope, the probability of a previously evaluated or unevaluated accident will not be created or increased. In addition, the consequences of a previously evaluated accident will

not be increased. Also, the probability for malfunction of equipment important to safety (both previously evaluated or unevaluated) will not be created or increased. Finally, the margin of safety as defined in any Tech Spec bases will not be affected. No USQ exists.

MM-4080

Description:

This modification is only to give specific instructions for field terminations of the VC compressor motor. This will allow proper hook-up of the motor. The motor has been tested after installation of the terminals.

Safety Review and USQ Evaluation:

The FSAR addresses the VC/YC system in section 6.4.2 and is also addressed in 3/4.7.6 of the Technical Specifications. The FSAR and T.S. do not address the terminations of conductors for the compressor motor. Therefore, proper terminations of the motor leads will not affect the FSAR or Tech Specs. No USQ exists.

MM-3892

Description:

This MM will add 4 new lifting lugs to each Reactor Coolant Pump (RCP) concrete hatch. The concrete hatch has existing lifting lugs that are bolted to an embedded coupling nut. The coupling nut has corroded due to exposure to water, and the threads have deteriorated.

New base plates has been anchored to the top of the hatches. In the center of each baseplate has been a threaded hole for the lifting lug to bolt into. Silicone caulk has been applied around the perimeter of the plate between the plate and the concrete.

Safety Review and USQ Evaluation:

The new lugs will allow the RCP concrete hatches to be safely lifted. The hatches are QA 1 but the lifting mechanism is Non-QA. Hatches are only moved along safe load paths and will not damage safety related equipment needed for decay heat removal or plant shutdown. The RCP hatches are only removed in mode 5, 6, or no-mode. Seismic induced failure or interaction will not reduce the functioning of a Nuclear Safety System to an unacceptable safety level. Since the hatches will not be degraded, their function as a barrier between the lower and upper compartments inside Containment is not adversely affected. All materials are compatible with the Containment environment.

Since the hatches are only removed in modes as discussed above, the lugs are adequately designed for expected loads, and load paths prevent damage to safety related equipment, there is no increase in the probability of an accident or a malfunction of equipment important to safety evaluated in the SAR. No safety equipment required in modes 5, 6, or no-mode has been affected. The pressure boundary between upper and lower containment is not degraded. Therefore, the consequences of an accident or of a malfunction of equipment important to safety evaluated in the SAR is not increased. The new lugs are adequate for potential loads and the

load paths have not changed. Based on this, the possibility for an accident or for a malfunction of a different type than any evaluated in the SAR is not created.

No safety limit, setpoint, or operating parameter has been changed by this MM. Therefore the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

MM-3870

Description:

The NRC Severe Accident Policy Statement and Generic Letter 88-20 led to the development of IPEEE for McGuire Nuclear Station which requires that certain equipment interaction be reviewed for a .3g ground response earthquake versus the .15g ground response Safe Shutdown Earthquake (SSE) (ATTACHMENT 1). This review determined 600 V Motor Control Centers (MCC) 1EMXB and 1EMXB-1 are in contact with one another. The same is true of MCC's 2EMXB and 2EMXB-1. An angle brace has been bolted to the top of each set of cabinets presently in contact.

Safety Review and USQ Evaluation:

Each unit has two redundant and independent 4160 Volt (V) Essential Auxiliary Power Systems which normally receive power from the 6900 V Normal Auxiliary Power System which furnishes power to all the large station auxiliary loads.

During a blackout condition, power to each of the redundant 4160 V Essential Auxiliary Power Systems is provided by a completely independent diesel-electric generating unit. On each unit, all engineered safety equipment is assigned to the two 4160 V Essential Auxiliary Power Systems with capacities and quantities such that the failure of components in one of the two systems does not affect the other system. With this arrangement of diesel-electric generating power sources, distribution system, and loads, complete redundancy of the entire 4160 Essential Auxiliary Power System is provided. The two systems are not electrically tied together at any time.

The 4160 V Essential Auxiliary Power Systems feed the 600 VAC Essential Auxiliary Power Systems. Each of two of these systems per unit include two load centers that supply power to large loads such as heater loads and 600 volt Motor Control Centers (MCC's) which are located in load concentration areas in the station. Connected to the MCC's are all the 600 volt loads which require power during blackout or accident conditions. Complete redundancy of these loads is provided in order to assure proper operation of safety features in the event of the failure of any single component in the 600 VAC Essential Auxiliary Power Systems.

The 600 V Essential Motor Control Center System for each unit is made up of two completely redundant and independent networks, designated Train A and Train B. Train A and Train B feed identical loads that perform the same function. Either Train A or Train B can power all of the necessary essential loads. 1EMXB and 1EMXB-1 are part of Train B on Unit 1. The feed loads located in the Auxiliary Building. 2EMXB and 2EMXB-1 are part of Train B on Unit 2. They also feed loads located in the Auxiliary Building.

The modification will not adversely affect either the seismic qualification of the equipment or the capability of the MCC's to perform their design functions. Therefore, there is no increase in the probability of an accident or of a malfunction of equipment important to safety evaluated in the SAR. The changes are Unit and Train related, maintaining single failure criteria. The trains on each Unit remain independent and redundant. Based on this, there is no increase in the consequences of an accident or of a malfunction of equipment important to safety evaluated in the SAR. Since seismic and electrical integrity is maintained, the possibility for an accident of a different type or for a malfunction of a different type than any evaluated in the SAR is not created.

No safety limit, setpoint, or operating parameter has been changed by this modification. Therefore, the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

MM-3867

Description:

A ladder storage device is needed in Diesel Generator (D/G) rooms 1A and 2A. ASTM A-36 angle irons has been welded to the existing door frame channels as described in the MM package. A 14 foot ladder has been stored in the device.

Safety Review and USQ Evaluation:

Each D/G unit is housed separately in a Category 1 structure which is part of the Auxiliary Building. The device design is QA Condition 4. Potential for missile generation (ladder or device) has been considered and is acceptable. The devices do not degrade the present structure. Therefore, the diesel units remain fully independent and redundant for each Nuclear Unit.

Since the D/G's are not accident initiators, the probability of an accident evaluated in the SAR is not increased. There is no increase in the consequences of an accident or of a malfunction of equipment important to safety evaluated in the SAR because the MM does not affect the independence or redundancy of the diesels and the single failure criterion is met. The structure is not degraded and the missile potential has been evaluated; no new failure modes have been identified. Therefore, the possibility for an accident or for a malfunction of a different type than any evaluated in the SAR is not created. Likewise, there is no increase in the probability of a malfunction of equipment important to safety evaluated in the SAR.

No safety limit, setpoint, or operating parameter has been changed by this modification. Therefore, the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

MM-3811

Description:

This minor modification will document the modification on 2A Component Cooling Heat Exchanger. 2 tubes were plugged following a 100% Eddy Current Inspection. 1 tube was plugged due to increasing pit depth in the tube. The other tube was plugged due to an obstruction which could not be removed. It has been determined that this obstruction is a fairly

large dent. There were 19 tubes previously plugged in the heat exchanger prior to implementation of this MM. There are two component cooling heat exchangers per unit. One heat exchanger is assigned to each train of KC equipment. One heat exchanger is required during normal plant operation and is adequate to provide minimum engineered safeguards heat transfer requirements. The heat exchangers are of horizontal, straight tube, single pass design with nuclear service water circulating through the tubes. Each heat exchanger has 4100 tubes of which 10% or 410 tubes can be plugged without affecting the required heat transfer capability of the heat exchanger.

Safety Review and USQ Evaluation:

Plugging of the damaged/obstructed tubes in the heat exchanger will prevent possible mixing of RN and KC waters. RN water may have high chloride content which is detrimental to the stainless steel components requiring KC water. Implementation of the MM will reduce possibility of tube break in the heat exchanger without affecting its required heat transfer capability. The component cooling heat exchanger is a QA-1 Safety Related component. No USQ exists.

MM-3672

Description:

The subject modification has been used to replace the existing Standby Nuclear Service Water trash racks with new racks using stainless steel materials and fasteners.

An inspection of the Standby Nuclear Service Water Intake structure revealed deterioration of the trash racks and fasteners. The existing racks and fasteners were galvanized steel.

Safety Review and USQ Evaluation:

The Standby Nuclear Service Water Pond (SNSWP) provides the ultimate heat sink for shutdown of McGuire Nuclear Station in the event that normal cooling is not available from Lake Norman (due to an earthquake). The Standby Nuclear Service Water Intake structure is a Seismic Category 1 structure. The new trash racks perform the function of screening large debris (tree limbs, etc.) from the Standby Nuclear Service Water Intakes. The new trash racks will perform the same function as efficiently as the existing racks because of similarity in mesh size. The new trash racks have been evaluated as a commercial grade item approved for use in this QA Condition 1 application. The new trash racks were reviewed to ensure structural integrity equal to or greater than the original racks.

The Standby Nuclear Service Water Pond, dam, intake structure, or trash racks are not accident initiators in any accident analyses. The new trash racks are qualified to meet the seismic and QA requirements for use in the intended application. Therefore the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The Standby Nuclear Service Water System serves as the ultimate heat sink for any accident scenario. The performance of this safety function will not be degraded by the modification. The new trash racks will function in the same manner as the existing racks. There are no common failure modes created by this modification. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

There are no new functions added by the modification. No new failure modes created. No accidents previously considered incredible are made credible by the modification. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

MM-3503

Description:

This modification will remove unnecessary piping support 2McA-S-ZD-500-1-D. This will clear the area where NSM 22279 would conflict with ZD system supports. Calculation MC 1206.16-30-3100 can be referenced to show the supporting analysis for the removal of this support.

Safety Review and USQ Evaluation:

This modification does not adversely impact the ZD system or ZD system support for 2B diesel generator. Since there is no adverse impact on any safety system the probability and consequences of an accident or equipment malfunction previously evaluated in the FSAR is not increased. The possibility of an accident or equipment malfunction important to safety not previously evaluated in the FSAR is not created. No USQ exists.

MM-3493

Description:

This modification will remove unnecessary piping supports of the ZD system. This will clear areas where NSM 22279 would conflict with ZD system supports. Calculation MCC 1206.16-30-3100 can be referenced to show supporting analysis for the removal of these piping supports or the modification of the support.

Safety Review and USQ Evaluation:

This modification does not adversely impact the ZD system or ZD system support for 2A diesel generator. Since no adverse impact on the safety function of any system exists, the probability and consequences of an accident or equipment malfunction previously evaluated in the FSAR is not increased. The possibility of an accident or equipment malfunction important to safety not previously evaluated in the FSAR is not created. No USQ exists.

MM-3223

Description:

The sole function of the pressure switches to be removed is to alarm upon sensing pressure below setpoint. Each normal and emergency fan is monitored by one of these switches. Upon sensing low pressure, the "Diesel Building Ventilation Malfunction" annunciator in the D/G control panel will locally alarm. Also, the "Diesel Panel Trouble" annunciator in the control room will alarm. Given the nature of the control room alarm, an operator is forced to go to the D/G room to see which local alarm has been activated. These pressure switches have a history of high maintenance/replacement and frequent calibration and frequently alarm even though fans are functioning normally. Fan on/off lights have been provided on the ventilation control panels in each D/G room.

Safety Review and USQ Evaluation:

Neither the probability nor consequences of an accident previously evaluated in the FSAR will increase since these switches are non-safety related and are incapable of causing an accident or contributing to its consequences.

The possibility of creating an accident other than those evaluated in the FSAR does not exist since the D/G and its associated equipment are used in accident mitigation and are not normally in operation.

The function of these switches is not important to safety even though they monitor some fans. Malfunctions of these switches have been a common occurrence, but their failure is incapable of keeping the emergency fans from operating. Therefore, the probability of a malfunction of equipment important to safety as evaluated in the FSAR is not increased by switch removal.

Since other means have been provided (on/off lights) locally and operators are already forced to go to the D/G rooms during emergency situations, a malfunction of the emergency fans has been identified almost immediately. In the event an operator is not present in the D/G room during an emergency, OAC alarms sound at 115°F and local high temperature thermostats trigger both the local and control room alarms which will bring an operator to the room. Therefore, removal of these switches will not prevent detection of an emergency fan malfunction any sooner than is currently possible with switches in place and consequences of the malfunctions are unaffected.

Since these switches are related only to the ventilation fans, their removal has been incapable of creating malfunctions in other safety equipment not previously evaluated in the FSAR.

The margin of safety as defined in the Tech. Spec. bases will not be reduced since removal of these switches will not prevent any safety-related equipment from performing its accident mitigation purpose. No USQ exists.

MM-2821

Description:

Inspections have revealed coating degradation and corrosion on the steel containment vessel (SCV inside containment behind the cork expansion joint at the concrete/steel interface at floor elevation 766+8-1/2 under the ice condenser). Therefore, the cork expansion joint has been permanently removed and the steel vessel repainted in accordance with McGuire Coating Specification 120-I. Due to problems accessing the vessel in some areas for surface preparations, coating application, and inspection, the coatings in these areas will not be considered to meet Service Level I requirements. In addition, the existing seal below the ice condenser has been replaced with a new seal. This will allow condensate that could initiate corrosion to escape through drain holes in the seal. The SCV and miscellaneous steel used in attaching the replacement pressure seal to the concrete and SCV has been coated in accordance with Maintenance Coating Specification 120-I also, except where accessibility prevents this as discussed above. The joint between the pressure seal and SCV has been caulked and has been considered functionally non-QA Condition.

The Exempt Change adds additional areas of the SCV to be coated in accordance with Service Level I Coating Specification 120-I. These areas are behind the VX Fan Pit Floor slab at elevation 766+8-1/2 and 2) behind all other floor slabs at elevation 738+3. Parts of these areas may be inaccessible resulting in coatings unable to meet Service Level I requirements.

Areas of the SCV surface which are not accessible to meet the QA requirements of Coating Specification 120-I has been documented on a Duke Power Service Level I Coating Deviation Record Form. The coatings used on these areas are considered "unqualified" and are presently considered as part of the maximum allowable limit of 20,000 square feet of containment unqualified coatings as established below.

Safety Review and USQ Evaluation:

The SCV, Ice Condenser, and the Containment Recirculation Sump are the Structures, Systems, and Components (SSCs) affected by this Exempt Change.

The purpose of the Containment system is to provide a barrier confining the potential releases of radioactivity from severe accidents for the protection of the public health and safety. The Containment may be considered as the Containment Vessel, the Containment Isolation System, and the Reactor Building.

The Ice Condenser is designed to limit the Containment peak pressure, to the design pressure, for all pipe breaks including complete severance of the largest Reactor Coolant System (NC) pipe. Consideration is given to subcompartment pressurization and different pressures resulting from these pipe breaks. The seal below the Ice Condenser floor is designed to separate upper and lower containment following an accident. The seal is also designed to prevent air exchange between the Ice Condenser and lower containment during normal operation. Cork was used during construction and as an expansion joint and currently serves no function. The Containment Recirculation Sump is used to provide a flow path to the Residual Heat Removal (ND) and Containment Spray (NS) pumps and heat exchangers following an accident which drains the Refueling Water Storage Tank (RWST). In addition, the sump provides flow to the Safety Injection System (NI) and the Chemical and Volume Control System (NV).

Removal of the cork expansion joint and repainting the containment vessel will remedy the existing containment corrosion concerns at elevation 766+8-1/2. Long term corrosion concerns in this area are addressed by the new seal as described above. This seal is designed to allow passage of moisture away from the SCV surface through drainage orifices. The addition of the drain holes does not affect ice condenser performance or integrity. The lower to upper containment bypass flow increase is comparatively insignificant. Neither will the ice condenser lower inlet door operation be affected. Repainting the other areas described above on elevations 766+8-1/2 and 738+3 will inhibit corrosion in these areas.

The use of unqualified coatings in the above areas is acceptable based on technical justification (Ref. 6) and review by the NRC. Additionally, the areas covered by this MEVN are not in the vicinity of high energy lines and are therefore not susceptible to the high energy line direct impingement effects. Basically, all areas in Containment where unqualified paint is used has been documented as either physically unable to reach the containment sump if failure occurs, or is within our total allowable of 20,000 sq. ft. of unqualified coatings which have the potential to fail and be transported to the sump.

The SCV, Containment Recirculation Sump, and Ice Condenser, are not accident initiators. Therefore, the probability of accidents previously evaluated in the SAR are not increased.

The SCV, Containment Recirculation Sump, and Ice Condenser are accident mitigators. Except for unqualified paint flaking off the SCV, no new failures are introduced. The NRC staff agrees that a 20,000 square feet limit for unqualified coatings is unlikely to result in either significant sump screen blockage, or degraded Emergency Core Cooling Systems (ECCS) or NS System performance. The amount has been documented as acceptable and will not adversely affect the capability of any SSCs to perform as required. The Ice Condenser performance will not be degraded with respect to lower inlet door operation or bypass flow. Since the functions of SSCs are not degraded, the consequences of accidents or malfunctions of equipment important to safety previously evaluated in the SAR are not increased. Likewise, there is no increase in the probability of a malfunction of equipment important to safety evaluated in the SAR.

No accidents previously thought incredible are made credible by this change. The operations of the Recirculation Sump, the ECCS, and the NS System are not degraded or altered by some small portion of flaked paint reaching them. The sump will perform its intended function. Therefore, the possibility for an accident or for a malfunction of a different type than any evaluated in the SAR is not created.

The containment shell will not be degraded by the application of the coating to the corroded areas. The performance of the ECCS and NS System will not be degraded by partial blockage of the sump. It is recognized by this evaluation that the Reactor Coolant pressure boundary (a primary fission product barrier) will have been breached if the recirculation sump is required. Since no fission product barriers are degraded by this change and no safety limits, setpoints or acceptance limits are affected, the margin of safety defined in the basis to the Technical Specifications is not reduced. No USQ exists.

MM-1588

Description:

The purpose of MEVN 1588, the installation of a time delay relay in the high voltage sensing circuits of diesel generator battery chargers 1EDGA & 1EDGB, is to improve their reliability and performance by eliminating battery charger shunt trips due to momentary voltage surges.

Safety Review and USQ Evaluation:

This exempt change is Q.A. 1. This change will not degrade the reliability, safety, or performance of the unit. No unresolved safety question has been introduced as a result of this exempt change.

MM-3287

Description:

This MM will replace Fuel Transfer Isolation Valve 1KF122 with a new knife gate valve, and add/modify the support restraints for the new valve. The purpose of the modification is to eliminate stem and drive system galling problems which caused the valve to fail. Valve replacement requires modification of two lateral supports on the transfer canal liner, and the addition of two spring hangers supported from the operating floor. The fuel transfer isolation valve is non-QA, however the supports are QA Condition 4 to ensure seismic interaction of the valve and the fuel transfer tube does not create excessive loads on the transfer tube. Installation requires removal of the existing limit switch and connection of the vendor supplied Namco EA-170 limit switch for open indication. The existing interlock with the fuel transfer system are not changed.

Safety Review and USQ Evaluation:

The Spent Fuel Cooling System (KF) is designed to remove heat from the spent fuel pool and maintain the purity and optical clarity of the pool water during fuel handling operations. The purification loop is the primary means for removing impurities from either the refueling cavity, transfer canal water during refueling, or the refueling water storage tank water following refueling.

The fuel transfer tube connects the refueling canal (inside the reactor Containment) and the spent fuel pool (outside the Containment). The fuel transfer tube is closed on the refueling canal side by a blind flange at all times except during refueling operation. The fuel transfer isolation valve located on the spent fuel pool side of the fuel transfer tube serves to isolate the fuel transfer tube during refueling to allow draining of the tube, and provide a low pressure (temporary) containment isolation during refueling modes. The fuel transfer valve is normally open to provide a flow path of water from the spent fuel pool to the fuel transfer tube for suction to the standby makeup water pump for SSF.

The Fuel Transfer Isolation Valve (1KF122) is not considered a containment isolation valve, neither does it have a QA1 function. Its function is to isolate the fuel pool. Valve leakage requirements have not changed. Reference to the valve being a containment isolation valve has been removed from the FSAR. The reason for this is that the McGuire FSAR currently reflects

in part the original design of the KR System requiring valve KF122 to be locked closed. However, when the Standby Shutdown Facility (SSF) was fully implemented and declared operable for compliance with 10CFR50 Appendix R, valve KF122 was then required to be locked open during plant operation in order to provide a suction source of borated water for the Standby Makeup Pump to be used as an alternate Reactor Coolant Pump seal injection system. Since the Standby Makeup Pump takes suction from the spent fuel pool via the fuel transfer tube, for Standby Shutdown System (SSS) operation, this valve must remain locked open in Modes 1-4. Environmental conditions and material requirements have been considered. The fuel transfer isolation valve is non-seismic . Failure of the valve will not cause fuel in the spent fuel pool to be uncovered due to elevation and building structural arrangement. The fuel transfer tube to which the isolation valve is attached is part of the containment boundary and is QA 1 . The deadweight and seismic loads transferred to the end of the fuel transfer tube are less than the original design; therefore, the original design basis and associated calculations are considered bounding .

The valve is not considered in any accident scenarios. Therefore, the probability or consequences of an accident evaluated in the SAR is not increased. Since the valve will function as before, will not adversely interact with the transfer tube, and is materially compatible with the system, there is no increase in the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. No new failure modes were found and no new valve functions are required. For these reasons and the ones discussed above, the possibility for an accident or for a malfunction of a different type than any evaluated in the SAR is not created.

No safety limit, setpoint, or operating parameter has been changed by this MM. Therefore, the margin of safety as defined in the basis of the Technical Specification will not be reduced. No USQ exists.

MM-3679
MM-3680

Description:

This modification reduces the Diesel Generator (DG) cooling water and lube oil system keep warm temperature from 150 F to 120F. The reduced keep warm temperature will decrease the likelihood of baking a varnish on the cylinder liners which could lead to operational problems due to cylinder blowby or scuffing. Oil tends to oxidize more rapidly as its temperature increases. The oxidation rate doubles for every 10 degree centigrade temperature rise. Reducing the lube oil temperature by 30 F will significantly reduce oil oxidation which means that the oil will retain its lubrication characteristics much longer on the cylinder liner surface. Another benefit is the reduced heat load for the DG rooms.

Safety Review and USQ Evaluation:

This change will not affect the DG's ability to meet operational requirements of fast starting and loading. The McGuire diesel manufacturer (Nordberg) conducted a series of 300 fast starts and loadings on identical engines to ours for Carolina Power and Light's Brunswick Steam Electric Plant. These starts were documented by Nordberg Test Report 438. During these tests the keep warm temperatures were 120 to 130 F with at least one start below 100 F. No problems or failures were encountered during this testing. This modification was discussed with four former Nordberg engineers (including two former chief engineers) as an owner's group meeting agenda

item on June 2, 1993. All four agreed that this change would cause no long term or emergency operating problems. They recommended that we make this change to improve cylinder liner lubrication because of the way nuclear stations operate their DGs with long periods of being in standby readiness. No new accident scenarios or consequences has been introduced by this modification.

The current DG low temperature operability limit of 110 F is not justified based on our discussion with the Nordberg engineers. The fact is that the Nordberg engines will meet their emergency mission without regard to lube oil or cooling water temperature. Their recommendation was to not assign a lower temperature limit for DG Operability and to not be concerned about heating the oil and water to 120 F before starting the break-in runs after outage work. The oil and water will heat up gradually as the break-in runs are being performed.

Also, the Chromalox thermostat model AR219LT (which replaces AR2529LT) has been evaluated as a commercial grade item for use on both the cooling water and lube oil heaters under CGD-33010.11-05-0001. No USQ exists.

MM-3763

Description:

This modification will allow the replacement of oil drain plugs on various safety related motors, pumps, and gear reducers with angle drain valves. The modification involves equipment in the Auxiliary Feedwater (CA), Chemical Volume and Control (NV), Safety Injection (NI), Component Cooling (KC), Fuel Pool Cooling (KF), and Nuclear Service Water (RN) systems.

The manual oil drain valves are expected to reduce unavailability of the components to which they are added, because periodic oil sampling may be done without removing the equipment from service.

The drain valves has been added directly in place of existing drain plugs in some cases. Reducers has been required where there are diameter differences between oil drain and drain valve. Short tubing has been required in other cases to stand the valves off from the oil reservoir, allowing clearance for operating the valve.

Safety Review and USQ Evaluation:

This modification affects equipment in several systems which are accident initiators. Loss of oil from the equipment could cause equipment failure due to loss of lubrication or fire in the vicinity of the equipment. Commercial Grade and seismically qualified valves has been used. Seismically qualified attachment configurations (reducers and standoffs) has been used. Reducers and tubing has been QA Condition 1. The existing oil drain plugs do not serve as an ASME pressure boundary, therefore, there are no pressure boundary requirements applicable to the new oil drain valves. Based on the seismic and Commercial Grade qualification, the oil drain valves and attachments has been at least as reliable as existing oil drain plugs, serving the purpose of preventing oil loss from the oil reservoirs. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

This modification affects systems and equipment which perform accident mitigation and/or plant safety functions. The functions of the affected equipment and systems will not be changed by the modification. The added oil drain valves will serve the same oil retention function as the existing oil drain plugs. No common mode failures are created which could render redundant safety-related equipment inoperable. The performance of safety functions is not degraded because, as stated previously, the reliability of the oil drain valves to prevent oil loss from the equipment is equivalent to that of the existing oil drain plugs. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased. The ability to manually open the oil drain to obtain an oil sample without removing the equipment from service is a new function provided by the oil drain valves, that was not feasible with the existing oil drain plugs. Procedural controls are necessary to prevent this new function from introducing a potential for equipment failure due to operator error during the sampling operation (the oil sampling procedure is not evaluated herein). Operator error allowing loss of oil is a potential single failure. Combination of this single failure with another single failure is not credible, even if oil samples from redundant equipment are being obtained simultaneously. For this reason, and the valve reliability as previously discussed, no accidents previously considered incredible are made credible by this modification. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

MM-4086

Description:

This modification will create a small recirculation loop from the main feedwater (CF) and return the water to the suction side (CM) of the feedwater pumps. This loop will run through a sampling device that will evaluate the potential of our feedwater to cause electrochemical corrosion of the secondary side components, including the steam generators. This data will help determine optimum secondary side chemistry.

Safety Review and USQ Evaluation:

There is an existing root valve on the CF header that has been utilized, therefore, no new taps has been required on the CF header. The water has been returned to the CM system through an existing high point vent on the discharge of the 1C2 feedwater heater. No new taps has been required on the CM system. The consequences or probability of a failure in the new loop is no greater than the failure of existing instrument lines on the CF or CM systems. Therefore, we have not increased the possibility or consequences of an accident evaluated in the SAR or created the possibility for an accident of a different type than any evaluated in the SAR. The feedwater system is not required to mitigate the consequences of an accident; therefore, we do not increase the probability or consequences of a malfunction of equipment important to safety. Loss of Feedwater is evaluated in the SAR; therefore, we do not create the possibility of a malfunction of a different type than evaluated in the SAR. No bases for the Plant Systems TS's reference the Feedwater System. No USQ exists.

MM-5227

Description:

This minor modification removes the existing ingress capability of the North and South PAP exit turnstiles. It also adds additional circuitry to enhance the directional rotation alarms. Details of this modification are classified as safeguards information and must be controlled as such.

Safety Review and USQ Evaluation:

Changes will not affect QA condition systems or components and due to location will not introduce seismic concerns. It will not degrade nuclear security to any extent, but will offer improvements to existing intrusion detection capabilities. No USQ exists.

MM-5230
MM-5231

Description:

The setpoint for the Auxiliary Feedwater (CA) suction pressure switches listed below has been changed to 5 psig.

1CAPS5002	Motor-driven CA Pump 1A
1CAPS5350	Motor-driven CA Pump 1A
1CAPS5360	Motor-driven CA Pump 1B
1CAPS5012	Motor-driven CA Pump 1B
2CAPS5360	Motor-driven CA Pump 2B
2CAPS5012	Motor-driven CA Pump 2B
1CAPS5042	Turbine-driven CA Pump 1
1CAPS5370	Turbine-driven CA Pump 1
1CAPS5390	Turbine-driven CA Pump 1
1CAPS5381	Turbine-driven CA Pump 1
2CAPS5042	Turbine-driven CA Pump 2
2CAPS5370	Turbine-driven CA Pump 2
2CAPS5390	Turbine-driven CA Pump 2
2CAPS5381	Turbine-driven CA Pump 2
1CAPS5044	Standby Shutdown System
2CAPS5044	Standby Shutdown System
1CAPS5380	Standby Shutdown System
2CAPS5380	Standby Shutdown System

The setpoint for the CA suction pressure switches listed below has been changed to 6 psig. The zero reference elevation for these switches is at a lower elevation than for the other switches.

2CAPS5002	Motor-driven CA Pump 2A
2CAPS5350	Motor-driven CA Pump 2A

The CA suction pressure switches initiate automatic swap-over to RN on decreasing CA pump suction pressure for their respective CA pumps.

This change increases the suction pressure at which the CA suction swaps to the Nuclear Service Water (RN) Assured Supply upon decreasing suction pressure. The new pressure switch settings facilitate RN swap-over above a minimum static water elevation of 730'+10" for Unit 1 and 731'+7-3/8' for Unit 2. The setting change increases the probability that swap-over will occur before it is no longer feasible to obtain water from the normal suction sources (Upper Surge Tank, CA Condensate Storage Tank, or the Condenser Hotwell). The pressure switch setpoints are being raised an additional 2 psi, because recent test results show some of the settings are changing more than 1 psi in a short period of time. Raising the settings will ensure the switches actuate above the minimum value for safe swap-over.

The CA system is safety-related, and is used for normal plant shutdown, accident mitigation, and can be used during startup. The RN Assured Supply to CA system is safety-related.

Safety Review and USQ Evaluation:

The CA system is not an accident initiator addressed in any FSAR Chapter 15 accident analyses. The reliability of the RN system Assured Supply to the CA system is increased by the setpoint change. Therefore the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The CA system is an accident mitigator for any emergency or normal shutdown operation that requires decay heat or reactor coolant heat dissipation through the Steam Generators. The performance of this safety function is not degraded by the setpoint change. There is no change of system functions and no common failure modes are created between CA system trains. Therefore the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR is not increased.

The modification adds no new CA or RN system functions. No new failure modes are created. No accidents previously considered incredible are made credible by these modifications. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

Technical Specification Table 3.3-4 gives the allowable and trip setpoints for CA Suction Pressure. The trip setpoint is greater than or equal to 2 psig. The change of pressure switch setpoints to 5 psig or 6 psig is in the conservative direction. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are adversely affected by the modifications. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

MM-5249
MM-5250

Description:

The purpose of this minor modification is to revise Unit 1 and Unit 2 diesel generator sync-check relay setpoints.

Safety Review and USQ Evaluation:

The probability of an accident and/or a malfunction previously evaluated in the FSAR will not be increased by this minor modification; the D/G Protective Relay System is not classified as an accident initiator and therefore is not included in the FSAR accident evaluation.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety will not be increased as a result of this modification. Changing the setpoints of the D/G sync-check relays will ensure that the maximum angle between the line and bus voltage is not exceeded when the diesel generator is paralleled to the 4160 volt bus. It will also ensure that the time delay after which the relay will operate is set such that the required maximum allowable slip frequency of 0.1Hz is established. These changes will enhance the reliability of and prevent damage to the diesel generators by allowing the sync-check relays to operate according to correct setpoints; currently, the relays are set for a +/-20 degree phase angle difference (+/- 10 degrees is correct) and a 0.037 Hz slip frequency (0.1 Hz is correct). It is necessary to change the settings not only to prevent damage to the diesel generators but to ensure that essential loads, especially motors are not subjected to extreme torque conditions if voltage phase or angle differences exceed 10 degrees during paralleling.

The probability of an accident and/or possibility of a malfunction other than that evaluated in the FSAR will not be increased. Changing the relay setpoints does not introduce a new function or a new failure mode of common failure modes that could render both diesel trains inoperable.

The safety related functions of the diesel generators will not be impacted due to the setpoint changes of the sync-check relays. The sync-check relays, although classified as 1E, do not perform a 1E function; they are blocked during an emergency start of the diesel because the diesel is the single source powering the 4160 volt bus at that time. The sync-check relays are used when the diesels are paralleled (during normal surveillance tests) to the 4160 volt bus. They are also used when offsite power to a 4160 volt bus has been restored to allow the diesel generator to be taken off line without interrupting power to the 4160 volt bus.

The safety related functions of the 125V DC Vital I&C power system, which provides control power to the relays, will not be impacted due to the setpoint changes of the sync-check relays. The setpoint changes have no effect on the relay control power. The setpoint changes do not change or affect the 125V DC bus loadings. The relays have a 30A fuse between the relay and the breaker; the fuse should isolate the relay from the 125V DC bus should a fault in the relay circuit occur. The setpoint changes have no effect on the fuse size or on fuse/breaker coordination.

No unreviewed safety question is introduced by these setpoint changes. The setpoint changes do not impact the margin of safety as defined in the bases to the Technical Specifications. No changes to the FSAR or Tech Specs has been required.

MM-5294

Description:

This modification will add 2 seismically mounted (QA Condition 4) cabinets in the Electrical Penetration Room, Unit 1 side, at elevation 767'. The cabinets will contain radio repeater equipment to be used for emergency communications. The new equipment is in addition to

existing communications equipment installed by a previous modification in the same room. Existing antennas has been used for broadcasting multiplexed signals of different frequencies from the existing and new radio repeaters. Power for the new equipment has been obtained from normal lighting panelboard 1L17, located in the same room.

Safety Review and USQ Evaluation:

No FSAR Chapter 15 accidents or events are initiated by the systems, components, or structures involved in this modification. The new cabinets has been mounted to comply with the requirements of QA Condition 4 (seismic mounting). New cabling has been installed per applicable electrical separation criteria. An Appendix R review was conducted with no concerns identified. Use of existing antennas, located at sufficient distance away from the Control Room, precludes an increased probability of an unplanned reactor trip. Therefore the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The equipment affected by this modification performs no accident mitigation functions. The equipment performs no plant safety functions. The performance of safety functions are not degraded by the addition of the new equipment. The new equipment performs the same functions as existing radio repeater equipment. There are no common failure modes created involving redundant trains of plant safety equipment. Therefore the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

No new functions are added and no new failure modes are introduced. No accidents previously considered incredible are made credible by this modification. Thus the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created. There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

MM-5461

Description:

The purpose of this editorial Minor Modification is to allow the use of test adapters in lieu of the existing 10-32 machine screws used to secure electrical conductors on the Solid State Protection System terminal blocks. The test adapters are Pomona Electronics Model 5763-0 Terminal Strip Banana Jack Adapters. They are designed to provide both a fastening function and circuit test point function. Their use has been reviewed and analyzed under PIP 0-M93-1295. The results of this analysis indicated that the terminal strip banana jack adapters are functionally equivalent to the existing terminal block conductor fastening screws. The banana plug adapters has been used in a mild environment, are fabricated from a compatible material, and will not violate electrical separation requirements. Implementation of this editorial Minor Modification will have no adverse effect on the operation or function of the Solid State Protection System.

Safety Review and USQ Evaluation:

This editorial Minor Modification will enhance the safety and reliability of routine Solid State Protection System testing performed by both the Operations Test Group and the Instrument and Electrical Section. This Minor Modification will eliminate the requirement for the use of mechanical "alligator" type connectors used during periodic testing. These type of electrical connectors have caused past problems during testing and carry a higher risk regarding test safety and Unit reliability. The use of the Pomona Test Adapters will significantly reduce this risk.

The components affected are QA Condition 1 and nuclear safety related. This Minor Modification will not mandate the specific test adapter locations be indicated on the affected design documents. No USQ exists.

MM-5463

Description:

The purpose of this minor modification is to revise the manual controls for the containment isolation valves 1CF26, 1CF28, 1CF30 & 1CF35.

Safety Review and USQ Evaluation:

These valves have a self-contained electro-hydraulic actuator. When the pump/motor is operating, the pressures, directions (open or close) and functions are controlled by pressure switches, solenoids and relief valves. When the motor/pump drives the piston operator to the fully closed position, the pump generates the necessary pressure for the required closing force. The pressure increases until it reaches the close position pressure switch (PS1) setting which opens and turns off the motor starter. The same operation applies to the open direction except that the open position pressure switch turns off the motor starter. In addition to the valve hydraulic controls, the electrical circuit provides manual controls to the operator via pushbuttons located on the Main Control board. The manual controls are wired such that the close position pressure switch (PS1) contacts are required to be closed so that the operator can assume control. The close position pressure switches (PS1) of the valve hydraulic system, due to internal hydraulic fluid leakage or temperature changes in the trapped fluid, have satisfied their setpoints when the valves are in the open position. This condition will not allow the operator to energize the motor starter circuit to close the valves.

This modification has been made to the non-safety portion of the controls for the Main Feedwater Containment Isolation valves. By revising the close circuit such that the close position pressure switch (PS1) contacts are bypassed, the operator has been able to close these valves in case of an inaccurate switch reading. In addition a set of relay contacts from the open portion of the circuit has been wired into the close portion of this circuit. This relay is an isolation relay, the coil is in a class 1E circuit, but the contacts are wired in the non-safety portion of the circuit. This change revises the circuit such that when the operator energizes the open portion of the circuit, the manual close portion of the circuit is blocked from energizing, provided that the manual pushbuttons are not depressed simultaneously. Actual field changes involve wiring changes in several ATC's using existing spare conductors and relays

This circuit modification will not cause a failure such that CF could not be isolated or cause a CF isolation when one was not wanted. The safety controls for CF isolation were not defeated by

this change. The safety controls are separate from the manual controls being modified. These changes will assure that the operator can manually control these valves during normal operation thereby making operation more reliable. For these reasons, this circuit is not an accident initiator as described in chapter 15 of the SAR. This minor modification does not change any accident assumptions or create an accident not previously identified. This change will not reduce the margin of safety as defined in the Technical Specifications, nor will it compromise the design or function of the CF system in any manner. No USQ exists.

MM-5478

Description:

The purpose of this Minor Modification is to delete EMF50 Flow alarm.

Safety Review and USQ Evaluation:

FSAR section 11.4.2 states that "Control Room alarms are incorporated ... for annunciating loss of sample flow to detectors where off-stream monitoring is incorporated ..." EMF50 does <u>NOT</u> incorporate such off-stream sampling. All gases exiting the waste gas system pass through EMF50. The flow rate of this sample stream is effectively monitored by the instrument loop consisting of 0WGFE6140, 0WGFT6140, and 0WCR6140. This instrument loop is in series with the flow alarm instrumentation on EMF50. Since both measure the same process, only one is actually needed. 0WGXX6140 serves the purposes of SLC Table 16.11-5, item 1b.

Chemistry personnel are in charge of releases from the waste gas system. They are capable of making these releases using the 0WGXX6140 loop instrumentation since that instrumentation is available to them at the waste gas panel on elevation 716'. Notification of release start/stop has been made verbally to Operations.

The present configuration of the EMF50 alarm is confusing to Operations at times because the alarm is often invalid. Because of the duplication of this instrumentation with the 0WGXX6140 loop, the best solution is to remove the EMF50 alarm altogether. NO USQ exists.

MM-5479

Description:

The purpose of this minor mod is the addition of interlock to radiation monitor EMF49 loss of flow alarm with waste monitor tank discharge flow loops 0WMLP5130 and 0WMLP5140.

Safety Review and USQ Evaluation:

As shown in drawing MC-1565-4.1, radiation monitor EMF49 monitors the combined output from waste monitor tank pumps A and B. EMF49 actually draws a sample from the combined 4" line (across 1WM245). Presently, if no release is being made, there is no flow through the monitor and the flow alarm is "active" in the control room. This alarm is not applicable in this condition as there is no flow in the 4" line that EMF49 is sampling.

This modification will clear the alarm in the control room if there is no flow from either waste monitor tank. If flow is present from either tank, the EMF49 flow alarm has been 'armed' so that

it will activate should flow stop in the EMF49 sample line. This ensures no unmonitored releases from the waste monitor tanks.

Releases are made from the waste monitor tanks at a flow rate of approximately 90GPM. The setpoints for the interlock circuitry (pressure switches) installed by this modification has been approximately 5 GPM to ensure that the flow alarm is armed properly when a release is in progress. This modification has no effect on the accidents described in Section 15 of FSAR (specifically, section 15.7.2). This modification has no effect on the high radiation alarms and corresponding control action of EMF49. No USQ exists.

MM-5481

Description:

The purpose of this minor mod is to interlock EMF44 flow alarm with WLFE5900.

Safety Review and USQ Evaluation:

The EMF44 loss of flow alarm has been interlocked with the existing flow instrumentation WLFE5900. This instrumentation monitors the flow in the common discharge line just downstream of the recirculation line for the CVUCDT. Therefore, any discharge from the CVUCDT passes through this flow element.

WLFE5900 produces a 4-20 mA signal that corresponds to a flow of 0-150 GPM. This signal has been passed through an alarm module which will activate the EMF44 loss of flow alarm at a point very near the bottom of the scale (≈4mA) on WLFE5900. The controlling procedure for CVUCDT releases requires a release in the range of 30 to 60 GPM. Therefore, the "actuation point" for the EMF44 loss of sample flow alarm has been at a very conservative position.

This modification has no affect at all on any of the accident scenarios evaluated in Ch. 15 of McGuire's FSAR. This equipment is not important from a Nuclear Safety perspective. Nor does this modification negatively impact the operation of EMF44 as described in section 11.4.2.1.8 of FSAR. With the conservative setpoint discussed in the preceding paragraph, no unmonitored release has been possible. Therefore, no unresolved safety question exists.

MM-6003, 6005

Description:

The modification will disable the NIS Negative Rate Trip Function by removing the Solid State Protection System (SSPS) logic ground. All associated annunciators has been revised. The trip function has been functionally disabled within the SSPS, however the circuit will remain in the NIS Power Range circuitry. The associated NIS bistable has been adjusted to prevent spurious actuations (Reactor Trips). Redundant MCM drawings has been deleted by this Minor Mod. There is an FSAR revision in support of this Minor Mod.

Safety Review and USQ Evaluation:

An analysis has been performed justifying removal of this trip function from the SSPS. Technical Specifications were revised (Facility Operating License Amendments 128 (Unit 1) and 110 (Unit 2) dated November 27, 1991 removed the requirement for this trip function on Unit 1 and Amendments 130 (Unit 1) and 112 (Unit 2) dated March 6, 1992 removed the requirement for this trip function on Unit 2). No accident analyses assumes actuation of the Power Range Negative Rate Trip Function. The Trip function is being disabled to avoid spurious reactor trips. The function has not been required to be operable since the above License Amendments were issued, however, the trip functions have not yet been removed from the plant.

The modification does involve the SSPS, which is a structure, system, or component (SSC) which is described in the FSAR, and it does more than replace components with equivalent components. The modification will disable the Negative Rate trip function as described in the FSAR (Section 7.2) by the removal of the logic ground for this function and adjustment of the bistable. The actual circuit for the trip will remain in the NIS Power Range circuit, though it has been non-functional. This is being done to prevent spurious reactor trip signals.

The modification will not degrade the effectiveness of an SSC important to safety in any design basis accident or event. FSAR Chapter 15 analyses do not take credit for the power range negative rate trip. The NRC has approved removal of the negative rate trip function in the issuance of the referenced Technical Specification changes.

As part of this modification, the references to the trip function in the FSAR are being deleted or modified to describe the fact the function is disabled.

The modification will not increase the possibility of an accident evaluated in the FSAR. The purpose of this reactor trip logic was to shutdown the reactor on RCCA misalignment and is not involved in the initiation of any accident. The removal of this trip, previously approved by the NRC, will reduce the chance of spurious reactor trips and any transients associated with a trip.

The modification will not increase the consequences of an accident evaluated in the FSAR. This trip function has been analyzed by Duke and found to be unnecessary. Other reactor trip functions will automatically shut down the reactor in the event that a reactor trip is required. NRC review of the trip function has also found it to be unnecessary and supports its removal.

This modification will not create the possibility for an accident of a different type than any evaluated in the FSAR. No new equipment is being added to the plant that may malfunction to initiate a new or different type of accident. The removal from service of this equipment and its function has been reviewed and approved by the NRC.

The modification will not increase the probability of a malfunction of equipment important to safety evaluated in the FSAR. The removal of this trip function from service will not directly impact the function of any other equipment in the plant. The removal of this trip function will decrease the probability of a spurious reactor trip and the resulting transient and demands on equipment important to safety. The prevention of spurious or unnecessary challenges of equipment important to safety will reduce the probability of malfunction, as the equipment is cycled less, and thus wear is slower.

The modification will not increase the consequences of a malfunction of equipment important to safety evaluated in the FSAR. The consequences of removing this trip function have been reviewed and approved by the NRC. In the event of an equipment malfunction that would have called for a negative rate reactor trip, it has been determined that either a reactor trip is not necessary or a different trip function will trip the reactor. Thus the consequences of any equipment malfunction has been evaluated and appropriate actuations will take place.

The modification will not create the possibility for a malfunction of a different type than already evaluated in the FSAR. In removing this trip function from service, no new equipment configurations has been created that could create a new type of malfunction than previously existed and has been evaluated in the FSAR.

The modification will not reduce the margin of safety as defined in the basis for any Technical Specification. The requirement for this trip function to be operable has been removed from the Technical Specifications with the issuance of the referenced license amendments. Analysis by Duke and approved by the NRC demonstrates that this trip function is unnecessary and no margin of safety is reduced by its removal.

This modification disables a Reactor Trip function which has been determined to be unnecessary. The trip function acts only to trip the rods and does not affect boron concentration, reactor coolant system temperature, or any other reactivity parameter. Since the modification has been performed during a refueling outage, an inadvertent actuation of this trip function during implementation has been of no consequence. This modification is to prevent an inadvertent or spurious reactor trip and the resulting transient and challenge to plant safety systems. The NRC has previously review and approved deletion of this particular reactor trip function. This modification is conservative and poses no reactivity management concerns.

MM-6007
MM-6008

Description:

MM-6007 and 6008 change the fuse size for the circuits supplying FW Sump Pump Motors A and B. The fuse size has been changed from 10 amps to 15 amps. The purpose of the FW Sump Pumps is to remove rain water from the pit around the Refueling Water Storage Tank and the Reactor Makeup Storage Tank pipe trench.

Safety Review and USQ Evaluation:

Problem Investigation Form serial number 2-M94-0573 was written when it was discovered that water, approximately 4 feet deep, was in the pit around the Refueling Water Storage Tank. The fuse feeding Unit 2 FW Sump Pump Motor A had blown. The sump pump motors are rated at 1/2 horsepower. Full load current for a 1/2 HP, 115 VAC motor is 9.8 amps. The size of the fuse protecting the motors has been increased to avoid nuisance power interruptions. 15 amps is the next highest standard fuse rating above 10 amps, this value also agrees with the fuse rating allowed by the National Electrical Code.

The FW Sump Motors are not Safety Related. These sump pumps are not required for accident mitigation; therefore, the consequences or probability of an accident evaluated in the SAR will not be increased. No new circuit controls has been added by this modification; therefore, the

possibility for an accident different than any evaluated in the SAR will not be created. Increasing the size of the fuses will make the pumps more reliable; therefore, the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR will not be increased. No new equipment has been added by this modification; therefore, the possibility for a malfunction of a different type than any evaluated in the SAR will not be created. The sump pumps are not required to be operable by the Technical Specifications; therefore, the margin of safety as defined in the basis for any technical specification will not be reduced. No USQs exist.

MM - 3523

Description:

This MM will replace eight (8) of the snubbers in the Diesel Generator D/G) Engine Cooling Water System (KD) with adjustable rigid struts manufactured by Anchor/Darling. The snubbers being replaced are those expected to have small movements (less than 1/8 inch). The replacement is a drop-in (pin to pin) replacement. The replacement will also allow deletion of 3 spring supports.

The purpose of this modification is to reduce snubber maintenance as required by the Technical Specifications, Section 3/4.7.8. the subject snubbers are located on Duke Class C lines. The piping design is still in compliance with Design Conditions, Load Combinations, and Code Criteria as listed in FSAR Table 3-47. Likewise, the Stress Criteria for supports, Restraints, and Anchors, as shown in FSAR Table 3-50 are adhered to. the same analysis methodology as before is used.

Safety Review and USQ Evaluation:

The KD System is a closed cooling system designed to maintain the temperature of the D/G engine within a safe operating range. Auxiliary purposes are supplying cooling water for the lube oil cooler and the intercooler which cools the air leaving the turbocharger. The KD System for each diesel unit is a Duke Class C System. Each diesel unit is housed separately in Category I structures which are part of the Auxiliary Building. Since the diesel units themselves are fully independent and redundant for each nuclear unit, they meet the single failure criterion.

The piping system and support/restraints replaced were done so in consideration of loads, load combinations, and allowable stress criteria in conjunction with applicable codes as discussed in the FSAR. The replacement of the snubbers with the adjustable rigid struts will not create any new line break locations or adversely affect previously evaluated break locations. Therefore, the probabilities of accidents or malfunctions of equipment important to safety previously evaluated in the FSAR are not increased. No redundancy or separation criteria are violated by this MM so there is no increase in the consequences of accidents or malfunctions of equipment important to safety previously evaluated in the FSAR. Since the system has been re-analyzed and found acceptable, no new failure modes are manifested as a result of this modification; therefore, no possibility of accidents or malfunctions of equipment important to safety different from any already evaluated in the FSAR is created.

Since re-analysis indicates the piping support system for KD is adequate, this change will not decrease the margin of safety during seismic events. No safety limit, setpoint, or operating parameter has been changed by this modification. Therefore, the margin of safety as defined in the basis of the Technical Specifications will not be reduced. No USQ exists.

MM-5441
MM-5442

Description:

Minor Mods to install key switches in six turbine trip solenoid valve circuit.

Safety Review and USQ Evaluation:

The Main Turbine trip solenoid valves are redundant, allowing one or more to fail without defeating the overspeed trip protection. With the new requirements for testing of these valves, having permanent switches installed will reduce the risk of solenoid valve failure by allowing them to be properly tested and ensuring that the circuit is restored to normal after the test is completed. This should improve turbine trip protection reliability. No new failures or consequences of a failure are created by the installation of these switches.

New key switches are non-safety, in circuit powered by IE power. This is in agreement with how other components in the circuit are wired. No USQ exists.

MM - 3834

Description:

Safety Review and USQ Evaluation:

The purpose of this Minor Modification is to document the use of the VL System as the mechanism for cooling the Reactor Vessel Support Coolers and associated concrete. In addition, since the KC System will no longer perform this function, low KC Flow Rate alarms has been permanently silenced.

The possibility, probability, and consequences of an accident or malfunction of equipment is not increased as a result of the implementation of this MM. A LOCA of the NC System due to concrete strength degradation from long term exposure to temperatures greater than 150 degrees F. would not be expected since design documentation exists indicating that the VL system is capable of maintaining acceptable concrete temperatures. The Reactor and its supports are not systems used to mitigate accident consequences. The safety function of the concrete and vessel supports will not be compromised as shown by analysis. No common failure modes are created because redundant trains of safety equipment are not involved. The VL System is capable of maintaining the Reactor Vessel Head region at a temperature of 135 degrees F. or less.

Silencing the Operator Aid Computer "low KC Flow Rate to RV Support Coolers" alarms has no significant effect on the design of function of the KC System. System performance will not be compromised. The margin of safety defined in the basis for any Technical specification will not be reduced because no Technical Specifications has been changed and no operating plant parameters (i.e., concrete temperatures, VL & KC System operating characteristics) has been significantly affected.

MM - 3755

Description:

2FW-27 Packing Leak Repair by packing sealant injection through leak off line.

Safety Review and USQ Evaluation:

2FW-27 has a packing leak. Leakage has been observed through the seal leak-off line. 2FW-27 can not be isolated to perform maintenance. It is not practical or cost effective to drain the FWST to repack the valve at this time. Leak repair (packing component will be injected through a seal leak-off line fitting to stop the leakage.

2FW-27 is a Duke Class B component wh ch serves as the FWST Isolation Valve to the ND Pump. This valve is normally open to provide a flow path to the ND pump suction for injection phase during an accident. 2FW-27 closes to isolate the FWST from the ND pump suction after switchover to the recirculation phases of an accident. 2FW-27 has no automatic function.

This type repair is routine and controlled by station procedures, and directives such that the plant systems and components will not be affected in a significant manner. No new failures are introduced b this modification. The margin of safety is not affected by this repair. The FSAR and Technical Specifications are not compromised. No USQs exist.

MM - 5433

Description:

McGuire Unit 2 incore flux thimble locations J-10 and F-8 are bent and inaccessible to a neutron detector or eddy current probe. These thimbles were damaged during the inspection in which the incore flux detectors were removed. Since the thimbles are inaccessible, a decision has been made to cap the thimble locations to prevent any future damage and potential leakage from the reactor coolant system. While not a concern at the present, thimble wall thinning has been identified as a generic industry concern in all Westinghouse PWRs that utilize bottom mounted instrumentation. In general, thimble tubes can experience thinning as a result of flow-induced vibrations at locations associated with geometric discontinuities or are changes along the flow path (such as areas near the lower core plate, the core support forging, the lower tie plate, the upper tie plate, and the vessel penetration).

The isolation valves associated with each thimble location are not nuclear safety-related components. A breach in a damaged thimble can be considered as a Small Break LOCA (SB-LOCA). Therefore, in the event of a breach of the damaged thimble, the isolation valve at the seal table would become essentially a part of the reactor coolant pressure boundary. Not being a nuclear safety-related component, this valve would not be qualified to fulfill this function. Therefore, the decision has been made to remove the tube and fittings between thimbles J-10 and F-8 and their isolation valves, close off their isolation valves, and cap the thimbles at the seal table with a QA-1 qualified cap fitting. These seals would serve as an acceptable component of the reactor coolant pressure boundary in the event of failure of the thimble.

In addition to capping thimbles J-10 and F-8, this safety review will generically discuss plugging of damaged thimbles. There are 58 incore guide thimbles in McGuire Unit 1. Each thimble is

manufactured from cold-worked stainless steel with a 0.201 inch internal diameter and a 0.049 inch wall thickness. Technical Specification 3.3.3.2 states the limiting conditions for operation of the movable incore detection system. in general, 75% of the detector thimbles must be operable at all times. Therefore, a total of 14 guide thimbles can be placed out-of-service at any one time. The maximum number of guide thimbles that can be capped in the various quadrants are as follows:

Qudrant Description	Total Thimbles	Max # of Capped Thimbles
Quadrant One Interior Thimbles	11	9
Quadrant Two Interior Thimbles	10	8
Quadrant Three Interior Thimbles	11	8
Quadrant Four Interior Thimbles	11	9
Axis Thimbles on Q1-Q2 Boundary	4	2
Axis Thimbles on Q2-Q3 Boundary	4	2
Axis Thimbles on Q3-Q4 Boundary	3	1
Axis Thimbles on Q4-Q1 Boundary	4	2

Note that these totals are for any one quadrant. Furthermore, the maximum number of capped thimbles listed above is very conservative for any one quadrant since the axis thimbles can be counted with the interior thimbles, providing more than two thimbles for any quadrant, as required by Tech Spec 3.3.3.2. No more than 14 guide thimbles total form all four quadrants may be placed out-of-service at any one time in order to maintain operability per Tech Spec 3.3.3.2.

Safety Review and USQ Evaluation:

The permanent capping of damaged incore guide thimbles with QA- qualified cap fittings does not involve an increase in the probability or consequences of an accident previously evaluated in the SAR. The movable incore flux detection system is used only to provide confirmatory information on the neutron flux distribution and is not required for the day-to-day operation of the plant. The system has been evaluated as a SB-LOCA for the unlikely scenario in which a breach in a thimble wall occurs. However, the postulated breach would only allow reactor coolant into the thimble and not past the seal table, where the QA-1 qualified cap fitting would maintain the reactor coolant pressure boundary. Therefore, a postulated SB-LOCA transient within the movable incore flux detection system is bounded by the current Chapter 15 SB-LOCA analysis.

The isolation valves supplied for each of the 58 incore detector flux thimbles are not nuclear safety related components. Therefore, in the event of a breach of a thimble due to damage or wall thinning, the isolation valve is not qualified to perform its function of reactor coolant pressure boundary. The addition of the QA-1 qualified cap fittings does not constitute the possibility for an accident or a malfunction of a different kind than any evaluated in the SAR. The cap fittings are installed to provide additional assurance that the reactor coolant pressure boundary will remain intact in the unlikely event of a breach of a thimble wall.

There is no increase in the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR. As discussed above, this modification is only adding QA-1

qualified cap fittings to ensure that the reactor coolant pressure boundary integrity is not compromised during the unlikely event of a breach of a thimble wall. No changes are being made as a result of this modification that will affect any equipment in a non-conservative manner. The QA-1 qualified cap fittings act as a passive safety system to provide additional levels of assurance that the reactor coolant pressure boundary has been maintained a the seal table.

There are no changes of safety limits, setpoints, or plant parameters as result of this modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical Specifications is not decreased. No USQ exists.

MM - 5477

Description:

Safety Review and USQ Evaluation:

The molded case switches have been installed in DC panelboards 2EVDA, 2EVDB, 2EVDC and 2EVDD. These panelboards are part of the 125VDC Vital Instrumentation and Control Power System.

The switches look like a molded case circuit breaker, but the do not have overcurrent trip capability. They will be protected by thermal magnetic circuit breakers in the distribution center from long time and instantaneous overcurrent. The available fault current is less than the switch rating.

The addition of these devices will not increase the possibility or the consequences of an accident evaluated in the SAR. Since the construction is identical to the circuit breakers, the probability of a malfunction would not be increased and any consequences of a malfunction are already included in the existing analysis.

The installation of these switches will not reduce the margin of safety at the plant. No USQs exist.

MM - 6409

Description:

This modification is being implemented as a result of the differential pressure tests performed for 1CA0042 and 1CA046. After review and analysis of the differential pressure data, it was determined that neither valve would completely close and isolate flow under design basis conditions. This review and analysis expanded to 1CA0058 and 1CA062 and the decision was made to install the 95% close torque switch bypass modification.

The valve operators were modified to resolve concerns of not attaining sufficient thrust to completely seat the valve disc against design basis systems conditions. To assure that the valve disc will fully close, the modified torque switch bypass contact has been in the circuit for 95% of the total closed valve travel. this will ensure that maximum motor torque is available for the

valve disc seating ı. After the bypass circuit drops out, the torque switch has been in the circuit to deenergi. operator after hard seat contact. The differential pressure tests performed for 1CA0042 and 1CA0046 indicate that the highest closing thrust requirements occur before hard seating contact. This is due to the large differential pressure expected across the valve seat during a design basis event. The present thrust set-up with the close torque switch in the circuit will seat the valve completely once the high thrust requirements associated with differential pressure have been overcome. The close torque switch bypass circuit has been moved to the primary switch pack, and the open torque switch bypass circuit will remain on the add-on-pack (AOP) switches. The computer point has been moved to the add-on-pack.

Functionally, valves 1CA0042, 1CA0046, 1CA0058, and 1CA0062 will operate identically to its present operation. This switch modification will not affect open or closing stroke times of 1CA0042, 1CA0046, 1CA0058, and 1CA0062. With the implementation of Minor Modification 6409, 1CA0042, 1CA0046, 1CA0058, and 1CA0062 has been more reliable in obtaining the desired position(s). Other indications and interlocks will not be affected.

Safety Review and USQ Evaluation:

The MOVs affected by Minor Modification 6409 are in the Auxiliary Feedwater System. 1CA0042, 1CA0046, 1CA0058, and 1CA0062 provides isolation from the Motor Driven Pumps in the event of a faulted Steam Generator. These valves also provide a flowpath for CA in the event a Steam Generator is not faulted. The safety function of 1CA0042, 1CA0046, 1CA0058, and 1CA0062 is to open and close and their normal position is open. 1CA0042, 1CA0046, 1CA0058, and 1CA0062 are considered ACTIVE valves. Installation of a 95% bypass to the close torque switch circuit will not affect closure times of 1CA0042, 1CA0046, 1CA0058, and 1CA0062 and the existing stress analysis of the piping associated with 1CA0042, 1CA0046, 1CA0058, and 1CA0062 will not be affected. Since this Minor Modification will ensure that 1CA0042, CA0046, 1CA0058, and 1CA0062 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. The appropriate documents requiring revision due to this Minor Modification have been identified and are listed on the Minor Modification form.

As described in the SAR, the facility has not been changed by implementation of this modification. The modification does not alter the component. Signals which initiate valve operability are not affected by the modification. They do not affect any of the Chapter 15 analyses. No USQ exists.

MM - 6338

Description:

NRC Generic Letter 89-10, issued on June 28, 1989, instructs nuclear power stations to develop a program to provide for the testing, inspection, and maintenance of motor operated valves (MOVs) so as to provide the necessary assurance that they will function when subjected to design basis system conditions. The level of testing, inspection, and maintenance performed for MOVs meeting the selection criteria established by the Generic Letter is much greater than that previously performed by Duke Power Company nuclear stations. As required by NRC Generic Letter 89-10, Duke Power Company has developed a comprehensive program plan that describes

the actions that Duke Power Company nuclear stations will accomplish in order to comply with the Generic Letter.

The actual changes involve resetting the open and close torque switches so that the motor operator will produce the necessary torque that has been converted by the stem nut to thrust to fully open and/or fully close the valve disc when design basis systems conditions are present. The minimum required and maximum allowed thrust used as the test acceptance criteria has been determined by Engineering Calculation MCC-1205.19-00-0003 and is provided by controlled document MCM 1205.19-0039. This engineering Calculation was performed in accordance with the latest revision of Duke Power Specification DPS-1205.19-00-0002 which establishes the parameters and criteria used to determine the minimum required and maximum allowed thrust levels.

The inaccuracies associated with the diagnostic test system used to facilitate thrust testing for 1CA0046 have been included in the engineering calculation. The final output thrust level achieved during the diagnostic test has been sufficient to allow valve operation at design differential pressure and system pressure without exceeding the limitations of the operator or valve components.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 6338 is in the Auxiliary Feedwater System. The function of 1CA0046 is to provide isolation for S/G "C" from the Motor Driven Pump B in the event of a faulted S/G. 1CA0046 also is required to open to provide a flowpath from S/G "C" from the Motor Driven Pump B if the S/G is not faulted. 1CA0046 is normally open and is an ACTIVE valve. Resetting the open and close torque switches will not affect open and closure times of 1CA0046. The existing stress analysis of the piping associated with 1CA0046 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1CA0046 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased.

MM - 5464

Description:

Modify manual controls for containment isolation valves.

Safety Review and USQ Evaluation:

The purpose of this minor modification is to revise the manual controls for the containment isolation valves 2CF26, 2CF28, 2CF30 & 2CF35. These valves have a self-contained electrohydraulic actuator. When the pump/motor is operating, the pressures, directions (open or close), and functions are controlled by pressure switches, solenoids and relief valves. When the motor/pump drives the piston operator to the fully closed position, the pump generates the necessary pressure for the required closing force. The pressure increases until it reaches the close position pressure switch (PS1) setting which opens and turns off the motor starter. The same operation applies to the open direction except that the open position pressure switch turns off the motor starter. In addition to the valve hydraulic controls, the electrical circuit provides manual controls to the operator via pushbuttons located on the Main Control board. The manual

controls are wired such that the close position pressure switch (PS1) contacts are required to be closed so that the operator can assume control. The close position pressure switches (PS1) of the valve hydraulic system, due to internal hydraulic fluid leakage or temperature changes I the trapped fluid, have satisfied their setpoints when the valves are in the open position. This condition will not allow the operator to energize the motor starter circuit to close the valves.

This modification has been made to the non-safety portion of the controls for the Main Feedwater Containment Isolation valves. By revising the closed circuit such that the close position pressure switch (PS1) contacts are bypassed, the operator has been able to close these valves in case of an inaccurate switch reading. In addition, a set of relay contacts from the open portion of the circuit has been wired into the close portion of this circuit. This relay is an isolation relay, the coil is in a Class 1E circuit but the contacts are wired in the non-safety portion of the circuit. This change revises the circuit such that when the operator energizes the open portion of the circuit, the manual close portion of the circuit is blocked from energizing, provided that the manual pushbuttons are not depressed simultaneously. Actual field changes involve wiring changes using existing spare conductors and relays.

This circuit modification will not cause a failure such that CF could not be isolated or cause a CF isolation when one was not wanted. The safety controls for CF isolation were not defeated by this change. The safety controls are separate from the manual controls being modified. These changes will assure that the operator can manually control these valves during normal operation thereby making operation more reliable. For these reasons, this circuit is not an accident initiator as described in Chapter 15 of the SAR. This minor modification does not change an accident assumptions or create an accident not previously identified. This change will not reduce the margin of safety as defined in the Technical Specifications, nor will it compromise the design or function of the CF system in any manner. No USQs exist.

MM - 5265

Description:

At present, selective coordination does not exist between the load center ground trip units and the MCC feeder breakers. For certain postulated ground faults, the ground trip unit may actuate and open a load center breaker before the breaker at the motor control center level trips. Thus, the existing configuration may not minimize the amount of equipment isolated from the system for adverse conditions such as a fault. To achieve selective coordination the ground fault trip units should be defeated.

Safety Review and USQ Evaluation:

The probability of an accident and/or a malfunction previously evaluated in the FSAR will not be increased by implementing this modification; the 600 volt essential power system is not classified as an accident initiator and therefore is not included in the FSAR accident evaluation.

Defeating the ground trip units on the load centers applies only to load centers that are fed from solidly grounded transformers, namely, 2ELXC and 2ELXD. The other load centers are fed from transformers which are high resistance grounded; ground faults in this case have been limited to less than one amp rendering the ground trip setting virtually irrelevant since the trip units detect ground faults of 100A or greater.

The consequences of an accident and/or the malfunction of the 600 volt essential power system equipment will not be compromised as a result of defeating the ground trip units. Rather, with selective coordination achieved, the reliability of the 600 volt essential system has been increased. Duke Power's ground fault detection and protection philosophy was recently a subject of consideration for the Power Consistency Team . It is assumed that should a ground fault occur within a motor control center or load center, one would expect it to rapidly develop into a high level fault that would be cleared by upstream protective circuit breakers. Should a low level fault develop and persist, existing ionization/smoke detectors in the area should alert operators to the problem. The most probable location of a ground fault is within the load device itself. Protection there is typically provided by thermal overload devices in combination with an instantaneous magnetic or thermal magnetic molded case circuit breaker. Thus, defeating the ground fault trip units will not prevent the faults from being protected/detected and therefore will not compromise the safety related functions of the 600V essential system. Also, the margin of safety as defined in the bases to the Technical Specifications will not be reduced.

The probability of an accident and/or possibility of a malfunction other than that evaluated in the FSAR will not be increased. Defeating the ground trip units does not introduce a new function or a new failure mode of common failure modes that could render both essential trains inoperable. No USQ exists.

MM - 4019

Description:

Deletion of Unit 1 Containment Compressor

Safety Review and USQ Evaluation:

The existing containment compressor can be started from outside containment to provide a supply of air to VI loads inside containment when needed. This modification will remove the Unit 1 Containment Compressor from service and provide manually operated, normally closed, cross-connect valves inside containment to allow one or more containment VI header(s) to supply loads off the others. The compressor is being abandoned because it produces air with dewpoints exceeding the requirements of Generic Letter 88-14 and SOER 88-10. In addition, problems with the compressors oil/air separator has resulted in oil being discharged into downstream piping. Attempts to correct these problems have been unsuccessful. The high cost of installing a new compressor skid which would satisfy the GL/SOER requirements and correct the oil problem is not warranted given the relatively minor benefits of operating the compressor. No USQ exists.

MM - 4090

Description:

This modification is to install a type of mini-flow for the YM pumps. These pumps go into a recirculation mode when the Reverse Osmosis Package is taken out of service. This mode can last for several hours. During this time, the water in the loop heats up beyond the system design parameters. Once the RO package returns to service, the system temperature drops rapidly, causing thermal stresses in the aluminum pipe. These stresses exceed recommended levels and have been attributed to several weld cracks in the last few years.

This modification will tap off of the recirculation loop with a 3/4" line that contains a valve that is sensitive to temperature. The chosen valve is an Ogontz Model 3/4FR110BRS w/o the continuous leak port. The valve has been set up to open as temperature approaches the design temperature and release a small amount of the hot fluid to the Water Treatment Room Sump. This will allow cooler water from the pumps suction source into the loop to mix with the cool the fluid in the recirculation loop. This will keep temperatures from exceeding the design limits and reduce the effect of the thermally induced stresses by limiting their magnitude and differential temperatures.

By only relieving as the temperature approaches the design temperature, this operation will limit wastage versus a continuous bleed type of temperature control for minimal cost difference.

The coupling that will allow installation of the valve has been added under WO93071222 Task 2 in order to take advantage of the system being out of service for replacement of a section of the aluminum pipe. This will eliminate the necessity of disassembling the system again for addition of the valve connection to the recirculation header.

Safety Review and USQ Evaluation:

This modification will create a type of mini-flow line for the YM pumps. The line will only come into service when the pumps go into recirculation long enough to heat the fluid until it approaches the design temperature of the system piping.

At that time the self contained temperature actuated valve has beengin to open, discharging some of the warm water to the water treatment room sump. This will allow cooler water from the pumps suction source to enter the line and cool the remaining volume of water in recirculation. The purpose has been to prevent exceeding the design temperature of 120 degrees F. Therefore, the water discharged to the sump will not be hot enough to create a personnel safety hazard.

Although the YM system is necessary in order to keep the units online, it is not required to mitigate the consequences of any accidents. Although we are adding a component and it could be perceived as another item that could fail. We are also eliminating a repetitive source of system failures that should lead to increased system availability and reliability.

The YM system is not safety related, does not provide any accident or post-accident mitigation function. Therefore, modification to the YM system does not increase the probability or consequences of a malfunction of equipment important to safety.

If the mini-flow failed in the closed direction, we would be left with the current configuration. Therefore, we have not increased the possibility or consequences of an accident evaluated in the FSAR. Although YM is non-safety and assumed to be lost during accident conditions, the "mini-flow" line is small and would not significantly degrade the makeup capability of the YM system if it failed in the open direction. Thus we have not created the possibility of an accident or malfunction of a different type than evaluated in the FSAR. No USQ exists.

MM - 3895

Description:

This modification extended the Auxiliary Diesel Fuel Oil Storage Tank vent from the height of approximately 3" from the ground to a height of 3' above the ground.

Safety Review and USQ Evaluation:

This will not increase the probability of an evaluated accident nor create a new type of accident. It will reduce the probability of the vent becoming clogged by debris or snow and reduce the probability of a fuel oil spill. The Auxiliary Diesel is not required to function following t0RNado events; therefore, the vent is not required to function following a t0RNado missile strike (as are the EDG vents).

The proposed design allows pre-fabrication of the vent extension. The new extension can then be bolted to the existing flange, allowing establishment of the new vent path before capping the current path. This design requires no welding in the area of the vent. This minimizes the possibility of a fuel oil fire due to combustible vapors in the vent path.

Piping size, class, and material will remain the same. The addition of 2' of 2" pipe will not adversely affect the nozzle loading of the tank. The configuration is such that the moment on the nozzle has been negligible and has support from 2' of soil. The additional vent height would increase the internal pressure in the tank by approximately 1 psi during overflow conditions. This is considered negligible because it would be countered by the soil loading on the buried tank. The increase in length by 2' of the overflow line will have insignificant effect on the resistance to flow. No USQ exists.

MM - 3366

Description:

Installation of new air supply for fuel pool gate seals.

Safety Review and USQ Evaluation:

The Fuel Pool gates are used to maintain level in the fuel pool when operational considerations require that the fuel transfer canal and/or the cask storage area be drained. The gate seals use Instrument Air (VI) to help provide a leakage seal around the gates. At present, air is supplied to the Fuel Pool gate seals via a rubber hose. One end of the hose is attached to a VI quick disconnect while the remainder is routed across the crane tracks and connected to the inlet connection on the seal(s). Any movement of the crane requires that the hose be moved in order to prevent the crane from cutting the hose. If the hose were to be cut, air pressure in the seals would be lost and operator action would be required to maintain fuel pool level.

This modification provides a means of supplying VI to the seals without routing the rubber hose over the crane tracks. Instrument tubing connected to the VI source has been rerouted between the crane tracks and the edge of the fuel pool at an elevation that will allow the crane to pass over without touching the tubing or rubber hoses. This configuration will lessen the chances of a rupture of the supply hose and consequently lessen the chances that fuel pool level has been lost.

Note that quick disconnects will now be located at the edge of the pool. It is possible that foreign material in the VI lines could exit the system when the hoses are removed from the quick disconnects. In order to prevent this material from entering the pool, this modification provides instructions to orient the quick disconnects in a manner which would direct expelled foreign material away from the pool. Note that the consequences of a tubing or hose rupture are not increased by this modification since only the VI routing to the seals is being changed. This rerouting introduces no new hazards or consequences that were not present before.

As a result of the above, the implementation of this modification will not increase the probability nor the consequences of an accident previously evaluated in the FSAR. In addition, it will not create the possibility of an accident which has not been previously evaluated in the FSAR. The possibility or consequences of FSAR evaluated equipment malfunctions will not be increased. Also, the possibility of unevaluated equipment malfunctions occurring will not be created. A review of Technical Specifications associated with systems that could be effected by this modification revealed that there should be no reduction in the margins of safety defined by the bases to there Technical Specifications. No USQs exist.

MM - 4119

Description:

This modification added a drain line to an existing port on valves 1/2BB92. This port was not utilitized for its intended purpose.

Safety Review and USQ Evaluation:

Recently it has been observed that 1BB92 has begun to accumulate water on the discharge side of the valve. This water could alter the relieving pressure or become a water hammer slug when/if the valve lifted. The body drain was opened and water drained shortly after the initial detection. Based on temperature profiles it is believed that water has again accumulated in this pipe. Draining the water is very awkward and does pose some personnel hazard. Therefore, a permanent drain is desired.

This part of the BB system is non-safety and is not required for accident mitigation or recovery. This modification should increase the valve reliability and increase the probability that the valve has been able to perform its intended function. This modification will not change the system function nor is it significant enough to include in the FSAR.

Neither the valve nor the drain will perform a function that will create a different type of accident than already evaluated in the FSAR. It is non-safety and will not interact with any safety related component; therefore, it does not increase the probability or consequences of a malfunction of equipment important to safety, nor does it increase the probability or consequences of an accident evaluated in the FSAR.

This part of the BB system does not have TS requirements associated with it and this modification will increase the reliability of 1/2BB92; therefore, it does not reduce the margin of safety as defined in the basis for any TS. No USQs exist.

MM - 3826

Description:

Change out the suction and discharge piping of the RF chlorination pump from carbon steel to stainless steel.

Safety Review and USQ Evaluation:

Currently, the suction and discharge piping of the RF chlorination pump is primarily carbon steel. While investigating low pump discharge pressure, it was discovered that the CS pipe was severely restricted by corrosion products. This modification replaced the carbon steel with stainless steel. This should prevent the corrosion product buildup and allow proper chlorination of the RF system. Proper chlorination is needed to assure Asiatic clam larvae are eliminated from the RF system.

Replacing the Carbon Steel with Stainless Steel will increase the reliability/availability of the RF system.
No USQ exists.

MM - 3789

Description:

Removal of the internals of check valve 2RN113.

Safety Review and USQ Evaluation:

The purpose of this Minor Modification is to allow the removal of the internals of check valve 2RN113. This check valve is located in the RN Train 2A assured Makeup Header serving the Unit 2 Spent Fuel Pool. The only safety function of this valve is to allow makeup water flow in the forward direction. However, it is also intended to prevent the siphoning of KF water from the Spent Fuel Pool should the RN piping upstream of 2RN113 rupture. Removing the internals from 2RN113 will have no effect on the intended safety of this check valve. The formation of a potential siphon is prevented by removing the internals of high point vent valve 2KF136. The modification to the high point vent valve was performed under a Minor Modification during 2EOC8 Refueling Outage which modified the internals of the RN Train 2B Assured Makeup check valve 2RN214.

This Minor Modification does not inhibit the "open" safety function of the RN Assured Makeup header to the KF System nor does it affect the "close" function since the continuous vacuum break has been previously established. The FSAR and Technical Specifications are not affected in any significant manner due to the implementation of this Minor Modification. The component affected by this modification is QA Condition 1. The implementation of this modification will reduce both short term and long term maintenance costs. No USQs exist.

MM - 4065

Description:

The purpose of this Minor Modification is to allow the removal of RN expansion joint H-6178 and replacement with a flanged piping section.

Safety Review and USQ Evaluation:

This existing expansion joint is installed in the "back flush" supply piping for RN Strainer 2A. This piping is 4" schedule 40 carbon steel and is Duke Class C piping. This expansion joint was discovered to have experienced an internal leak on the downstream bellows (closest to the 2A strainer nozzle connection). The problem was identified on PIP 2-M93-1158 by Mechanical Systems Engineering and resolved by Mechanical Equipment Engineering. Civil Engineering supported the proposed resolution for this PIP and recommended that the 4" expansion joints be replaced on an as needed basis with a flanged piping section. Analysis and review of the Piping stress and support/Restraint calculations (MCC-1206.12-18-2115 and MCC-1206.02-84-0018) for this portion of RN piping (Math Model RN 378) indicated the affected expansion joint could be removed and replaced with a flanged piping section without any significant effect on the piping stress. This Minor Modification only affects the 4" expansion joint on RN Strainer 2A. The other three (3) expansion joints incorporated into the other three (3) RN Strainers will not be replaced or modified by this project. They has been replaced whenever defects or malfunctions are experienced. The new flanged piping section has been fabricated using pipe and flanges qualified for use on RN Class C piping. The length of the new piping section has been identical to the length of the removed 4" expansion joint.

This Minor Modification affects QA Condition 1 and nuclear safety related piping. The new flanged piping section will not compromise the design or function of RN Strainer 2A in any manner. The RN System will not be affected in any detrimental aspect due to the replacement of this malfunctioning expansion joint. The new flanged piping section has been installed during the upcoming Unit 2 EOC9 refueling outage. The affected piping section cannot be adequately isolated with Unit 2 online with out incurring significant system unavailability. No USQs exist.

MM - 5412

Description:

This modification redesigns the non-safety in-core thermocouples cables to improve reliability and maintenance capabilities.

Safety Review and USQ Evaluation:

This modification does not challenge the safety of the plant in any way or effect thermocouple function or performance. The addition of the junction box will reduce maintenance time and protect the system during outage against unintentional abuse/wear and tear. This modification is not significant enough to require inclusion into the FSAR. No unanswered safety questions result form this modification. No USQs exist.

MM - 5130

Description:

The subject modification updates design drawings of the Aux Feedwater (CA) Pump Control Panels, to show fuse and terminal block information that was not shown on the original drawings. The existing fuses within the CA panels have been changed out to FNQ-type to be consistent with present fusing standards, and fuse blocks has been changed out as necessary.

The panel inspection, by which the drawing discrepancies were found, was mandated by a previous NRC commitment to inspect all panels for proper fusing.

Safety Review and USQ Evaluation:

There are no FSAR Chapter 15 accidents or events initiated by the CA system, therefore, the system is not an accident initiator. The CA Pump Control Panels are QA Condition 1. The fuses and fuse blocks meet QA Condition 1 and 4 (seismic mounting) specifications. An Appendix R review was completed with no concerns identified. The FNQ-type fuses to be substituted were sized according to established fusing criteria. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The CA system is used for accident mitigation and performs plant safety functions. The fusing equipment will perform the same functions as prior to the modification. No common failure modes are created by the fuse changeout to FNQ-type. The fuse changeout does not degrade the safety functions of the CA system, because equivalent or better QA Condition 1 qualified fuses has been used. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR is not increased.

No new functions are added and no new failure modes are created by the modification. No accidents previously considered incredible are made credible by the modification. The FNQ-type fuses are at least as reliable as the existing fuses. Thus, the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of the modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) are not degraded. No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical specification is not decreased. No USQs exist.

MM - 5244A

Description:

This modification replaces the Honeywell controllers in the Waste Gas System A and B train gas analyzer racks with Moore manual loaders A&D RIS modules.

Safety Review and USQ Evaluation:

The loaders will allow manual control, as specified by Chemistry, of oxygen feed valves which regulate oxygen inlet to the Waste Gas System recombiners. The RIS modules will provide

actuation of High hydrogen alarms on the recombiner panels. Although this control portion of the gas analyzer rack is part of the waste gas system which is evaluated in Tech Spec and FSAR, it is not a safety related part of the system. Other interlocks and safeguards not affected by this mod are in place to insure specified safety limits are satisfied. Therefore, there is no increase of probability, consequences, or possibility of n accident or equipment malfunction. Likewise, a implementation of this mod does not reduce the margin of safety as defined in the basis for Tech. Spec. No USQs exist.

MM - 2792

Description:

Purpose of MEVN-2792 is to show significance of pressure switches 1,2NDPG5040 and 1,2NDPG5050 in ND flow diagrams, and change labeling on these instruments so they are identified as Pressure switches.

Safety Review and USQ Evaluation:

Pressure switches 1,2NDPG5040 and 1,2NDPG5050 control ND pump mini-flow valves 1,2ND68A and 1,2ND67B respectively. These valves open at low and close at high differential pressure setpoints. Currently, ND flow diagram does not show the relationship between the pressure switches and the mini-flow valves. This Exempt Change will revise the ND flow diagram to show this relationship. Also, pressure switches 1,2NDPG5040 and 1,2NDPG5050 are incorrectly labeled as pressure gauges. Affected documents will also be revised under this MEVN to change instrument label of 1,2NDPG5040 to 1,2NDPS5040 and 1,2NDPG5050 to 1,2NDPS5050.

This change will increase personnel awareness of the significance of pressure switches 1,2NDPG5040 and 1,2NDPG5050 and will eliminate possible confusion as to their instrument type. Documents affected by the relabeling has been revised and affected plant procedures have been updated as required. This exempt change did not require any physical work other than changing the instrument labels. Safety of ND system was not degraded, and FSAR and Technical Specifications were not affected. No USQs exist.

MM - 3361

Description:

The purpose of this Minor Mod is to revise MCM 12-5.19-0037 001 (EMO Torque switch Setting List) to reflect new thrust values for 1NI0122 and 2NI0122.

Safety Review and USQ Evaluation:

These new thrust values are a result of applying a new method for accounting for test equipment accuracy and undervoltage concerns. These factors will in fact create a smaller and more conservative window for future actuator set up. The thrust windows are necessary to ensure that structural limits for both the actuator and the valve are not exceeded and still provide sufficient thrust to close the valve under worst case conditions.

No new failure mode has been introduced as result of this modification. Additionally the probability, possibility, or consequences of a malfunction will not be increased by this modification. No FSAR or Technical Specifications are affected. No USQs exist.

MM - 6182, 6183, 6184, 6185

Description:

The Safety Injection (NI) system, Chemical and Volume Control (NV) system, Residual Heat Removal (NS) system, and Refueling Water (FW) system comprise the Emergency Core Cooling system (ECCS) at McGuire. The primary function of the ECCS is to transfer heat from the core following the loss of a large amount of reactor coolant. This is accomplished during the injection and recirculation (cold-leg and hot-log) phases of ECCS operation. The NV system inject to the NC system cold-legs when NC pressure is lower than the shutoff head of the NV Centrifugal Chargine Pumps (CCPs). During the injection phase, the two NC CCPs deliver flow to the reactor vessel through a common header, then through four 1-1/2 inch branch lines into each NC cold leg. Throttle valves in each of the 1-1/2 inch lines are adjusted to prevent CCP runout, to provide proper flow split between injection points, and to ensure that adequate flows are provided to the injection points. Two parallel, normally closed motor-operated gate valves (NI9A and NI10) isolate the CCPs from the NC system during normal operation. These valves will automatically open following receipt of a Safety Injection signal, providing an injection path. Downstream of these valves, on the four 1-1/2 inch branch lines, are the throttle valves (NI480, NI481, NI482, and NI483). The four throttle valves are locked into position following adjustment, and no action is required during accident mitigation. Downstream of these valves are located four manual isolation valves (NI14, NI16, NI18, and NI20). The purpose of these valves is to isolate the injection lines for maintenance and testing purposes. Check valves downstream of the manual isolation valves are provided for isolation from the NC system.

These temporary modifications removed the internals from the four Unit 2 manual isolation valves (2NI14, 2NI16, 2NI18, and 2NI20) during unit operation. Seat damage to at least three of these valves has occurred, which could possibly affect the NV coldleg injection flow balance. These valves, or their internals, were replaced during refueling outage, 2EOC8.

Safety Review and USQ Evaluation:

The removal of the valve internals of 2NI14, 2NI16, 2NI1, and 2NI20 will allow more flow to be delivered from the CCPs to the NC coldlegs. Following removal, it has been necessary to adjust the throttle valves and perform a flow balance prior to Unit 2 startup.

FSAR Chapter 15 events for which NV would provide injeciton are any accident resulting in a Safety Injection signal, including Small and Large Break LOCAs (FSAR Section 15.6.4), a Steam System Piping Failure (FSAR Section 15.1.5), the Inadvertent Opening of a Steam Genreator Relief or Safety Valve (FSAR Section 15.1.4), a Feedwater SystemPipe Break (FAR Section 15.2.8), a Rod Ejection Accident (FSAR Section 15.4.8), the Inadvertent Opening of a Pressurizer Safety or Relief Valve 9FSAR Section 15.6.1), or a Steam Generator Tube Failure (FSAR Section 15.6.2).

The SSCs affected by this modification are the NV CCPs and the four discharge flow paths to the NC cold-legs. This modification increases the flow through these flow paths. The three functions of these lines are to prevent CCP runout, properly split flow, and provide adequate

flows to the cold-legs. Since the removal of the internals will increase flow, provision of adequate flow will not be degraded. The prevention of CCP runout has been ensured by adjusting the throttle valves. Properly split flow is ensured by the performance of an NV cold-leg injection flow balance. Therefore, this modification does not adversely affect the operating charateristics of the SSCs involved. The design bases of the ECCS will not be adversely affected because measures have been taken to ensure that NV cold-leg injection has not been degraded. No USQs exist.

MM - 5493

Description:

The purpose of this modification is to install a barrier to separate a Safety Train A (red) cable tray section from a non safety related (black) cable tray section between cable tray junction points 7519 and 7521. The subject cable tray is located in Unit 2 Auxiliary Building Elevation 716' + 0", near Column Line HH-55.

The modification is corrective action for PIP item # 0-M94-0351. The PIP concerns a case where Train A cables were to exit a cable tray section designated Train A (at junction point 7521), but during cable installation (through to be at the time of plant construction) the cables were pulled past the Train A junction point into a cable tray section designated non-safety (at junction point 7519).. The Train A cable tray section ends at a T-intersection with the non-safety tray section. Although there is no physical barrier separating the tray sections, the relative position of the Train A versus non-safety designated junction points (7521 and 7519) was considered to provide for adequate separation. The operability evaluation determined that the cables involved in the PIP are operable as installed; therefore, they do not require relocation. However, it was determined desirable to add a physical barrier to more clearly define the boundary between the Train A and non-safety designated tray sections. The boundary consists of a 5" high metal wall to be installed across the T-section of cable tray. The effect of installing the boundary is to redesignate a portion of the formerly non-safety designated tray section as Train A. No distance separation is provided in addition to the metal barrier.

Safety Review and USQ Evaluation:

FSAR Section 8.3.1.2.7.5 and Installation Specification MCS-1390.01-00-0036 define the cable separation criteria applicable to McGuire Nuclear Station. Distance separation, with and without a barrier, is specified for use between redundant trains of safety cables. There is no barrier or distance separation specified for use between safety and non-safety cables. The only specification established for separation of safety and non-safety cables is that a non-safety cable that has "run with" safety train cables for part of its route shall not "run with" the redundant train of safety cables for another part of it's route (termed "channel-hopping"). The common definition of "running with" applied at McGuire is a non-safety cable routed within a safety designated cable tray for some portion of its route. In that case, the non-safety cable has likely "touched" one or more safety cables within the safest tray. Non-safety cables in close proximity to, but not actually touching, safety cables are not considered to have "run with" the safety cables. The common practice has been not allowing safety cables to touch non-safety cables (except where the non-safety cable is routed in a safety designated tray). The Train A and non-safety cables involved in the subject modification, as installed, are not considered to be in violation of established separation criteria (either documented or common practice). The barrier is not required to establish proper separation in the case of existing cables, but serves to prevent

violation of established separation criteria for any future installed cables. The added barrier is installed in accordance with QA Condition 4 (seismic) criteria. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The cables and cable tray involved in the PIP and modification will perform the same functions , except a portion of non-safety designated cable tray has been designated safety. As standard practice, cable trays at McGuire are purchased and installed to meet QA Condition 1 (nuclear safety-related) criteria. There are no qualifications, material, or installation differences between non-safety and safety designated cable trays. The only difference is assigned designation, to prevent undesired mixing of safety related and non-safety cables. Since adequate cable separation has been determined to exist with or without the added barrier, and the barrier has been installed to meet seismic criteria, the performance of plant safety functions will not be degraded and no common failure modes are created because of the modification. Therefore, the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR are not increased.

There are no new plant functions added by the modification. Since seismic and separation criteria are met, no new failure modes are created and no accidents previously considered incredible are made credible by the modification. Thus, the possibility of an accident or malfunction of equipment of a different type than evaluated in the SAR will not be created.

There are no changes of safety limits, setpoints, or plant parameters because of this modification. The fission product barriers (RCS pressure boundary, containment, fuel pellets, and cladding) will not be degraded. No assumptions made in any accident analysis are affected by the modification. Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQs exist.

Safety Review:

The MOV affected by Minor Modification 3368 is in the Feedwater system. 2CF0134 serves to provide main feedwater system isolation during the reverse purge process. The safety function of 2CF0134 is to close and it's normal position is closed. 2CF0134 is open only during the reverse purge process for controlling main feedwater valve tempering flow. Resetting the open and close torque switches will not affect open and closure times of 2CF0134. The existing stress analysis of the piping associated with 2CF0134 will not be affected by resetting the open and close torque switches. Since this Minor Modification ensures that 2CF0134 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

MM-3950, 3946, 3949, 3347, 3837, 3675, 3784

Description:

NRC Generic Letter 89-10, issued on June 28, 1989, instructs nuclear power stations to develop a program to provide for the testing, inspection, and maintenance of motor operated valves (MOVs) so as to provide the necessary assurance that they will function when subjected to design basis system conditions. The level of testing, inspection, and maintenance performed for MOVs meeting the selection criteria established by the Generic Letter is much greater than that

previously performed by Duke Power Company nuclear stations. As required by NRC Generic Letter 89-10, Duke Power Company has developed a comprehensive program plan that describes the actions that Duke Power Company nuclear stations will accomplish in order to comply with the Generic Letter. This Minor Modification provides for the diagnostic testing for MOVs and constitutes part of the actions necessary for compliance to NRC Generic Letter 89-10.

The actual changes involve resetting the open and closed torque switches so that the motor operator will produce the necessary torque to fully open and/or fully close the valve disc when design basis systems conditions are present. The control logic for the motor operator will not change. The valves will open and close under the control of the limit switch with the torque switch connected to the circuit. resetting the torque switches is accomplished on a torque test bench and the criteria used for the bench test, the minimum required and maximum allowed torque has been determined by Engineering Calculation MC-1205.19-00-0012 and is provided by controlled document MCM 1205.19-0039. This Engineering Calculation was performed in accordance with the latest revision of Duke Power Specification DPS-1205.19-00-0002 which establishes the parameters and criteria used to determine the minimum required and maximum allowed torque levels.

The inaccuracies of the diagnostic test system used to facilitate torque testing have been included in the Engineering Calculation. The final output torque level achieved during the diagnostic bench test has been sufficient to allow valve operation at design differential pressure and system pressure without exceeding the limitations of the operator or valve components.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3950 is i[illegible] uclear Service Water System. 0RN0015 provides isolation of RN Supply Train Cro[illegible]er. The safety function of 0RN0015 is to close and its normal position is closed. 0RN0015 is considered an active valve. Resetting the open and close torque switches will not affect open and closure times of 0RN0015. The existing stress analysis of the piping associated with 0RN0015 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 0RN0015 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3946 is in the Nuclear Service Water System. 0RN0003 provides isolation of RC Crossover Supply to RN A train. The safety function of 0RN0003 is to close and it's normal position is close. 0RN0003 is considered an active valve. Resetting the open and close torque switches will not affect open and closure times of 0RN0003. The existing stress analysis of the piping associated with 0RN0003 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 0RN0003 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3949 is in the Nuclear Service Water System. 0RN0014 provides isolation of RN System Train Crossover. The safety function of 0RN0014 is to close and it's normal position is closed. 0RN0014 is considered an active valve. Resetting the open and close torque switches will not affect open and closure times of 0RN0014. The existing stress analysis of the piping associated with 0RN0014 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 0RN0014 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction will ensure that 0RN0014 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3347 is in the Diesel Generator Engine Lube Oil System. The function of 2LD0108 is to provide a bypass of the Full Flow Lube Oil Filter on high differential pressure. The safety position of 2LD0108 is open and it's normal position is closed. Resetting the open and close torque switches will not affect open and closure times of 2LD0108. The existing stress analysis of the piping associated with 2LD0108 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2LD0108 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3837 is in the Component Cooling Water System. The function of 1KC0018 is to isolate the Reactor Building non-essential headers and provide component cooling equipment train separation. The safety position of 1KC0018 is closed and it's normal position is closed. Resetting the open and close torque switches will not affect open and closure times of 1KC0018. The existing stress analysis of the piping associated with 1KC0018 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1KC0018 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3675 is in the Component Cooling Water System. The function of 1KC0230 is to isolate the reactor building non-essential headers and provide component cooling equipment train separation. The safety function of 1KC0230 is to close and it's normal position is open. Resetting the open and close torque switches will not affect open and closure times of 1KC0230. The existing stress analysis of the piping associated with 1KC0230 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1KC0230 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3784 is in the Safety Injection System. The function of 2NI0100 is to open during the cold leg injection phase of ECCS operation to allow flow from the RWST to the NI pumps. 2NI0100 is also required to close upon operator action to isolate the RWST from NI pumps and RHR suction during the sump recirculation phase of ECCS operation. Power is normally removed from this valve to prevent inadvertent actuation. Resetting the open and close torque switches will not affect open and closure times of 2NI0100. The existing stress analysis of the piping associated with 2NI0100 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2NI0100 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3948 is in the Nuclear Service Water System. 0RN005 provides isolation of RC Crossover Supply to RN B Train. The safety function of 0RN0005 is to close and it's normal position is close. 0RN005 is considered an ACTIVE valve. Resetting the open and close torque switches will not affect open and closure times of 0RN0005. The existing stress analysis of the piping associated with 0RN0005 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 0RN0005 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3934 is in the Nuclear Service Water System. 0RN0011 provided isolation of Low Level Intake Supply to RN B Train. The safety function of 0RN0011 is to close and it's normal position is open. 0RN0011 is considered an ACTIVE valve. Resetting the open and close torque switches will not affect open and closure times of 0RN0011. The existing stress analysis of the piping associated with 0RN0011 will not be affected by resetting the open and close torque switches. Since the Minor Modification will ensure that 0RN0011 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 3412 is in the feedwater system. The function of 1CF0157 is to provide system isolation to each auxiliary feedwater nozzle. The safety function of 1CF0157 is to close and it's normal position is open. Resetting the open and close torque switches will not affect open and closure times of 1CF0157. The existing stress analysis of the piping associated with 1CF0157 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1CF0157 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of

equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOV affected by Minor Modification 6439 is in the Nuclear Service water System. The function of 2RN0162 is open to supply RN to the CA pumps. 2RN0162 is normally closed in order to isolate RN B Train supply. 2RN0162 is an ACTIVE valve. resetting the open and close torque switches will not affect open and closure times of 2RN0162. The existing stress analysis of the piping associated with 2RN0162 sill not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2RN0162 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

The MOV affected by Minor Modification 3410 is in the feedwater system. The function of 1CF0153 is to provide system isolation to each auxiliary feedwater nozzle. The safety function of 1CF0153 is to close and it's normal position is open. Resetting the open and close torque switches will not affect open and closure times of 1CF0153. The existing stress analysis of the piping associated with 1CF0153 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1CF0153 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ F aluation:

The MOV affected by Minor Modification 3411 is in the Feedwater system. The function of 1CF0155 is to provide system isolation to each auxiliary feedwater nozzle. The safety function of 1CF0155 is to close and it's normal position is open. resetting the open and close torque switches will not affect open and closure times of 1CF0155. The existing stress analysis of the piping associated with 1CF0155 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 1CF0155 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Safety Review and USQ Evaluation:

The MOVs affected by Minor Modification 7098 is in the Containment Spray System. The safety function of valve 2NS0003 is to close to realign the spray pumps suction from the RWST to the containment sump. During modes 1-4, 2NS0003 is open to provide a flowpath from the RWST to the containment spray sumps. 2NS0003 is an active valve. Resetting the open and close torque switches will not affect open and closure times of 2NS0003. The existing stress analysis of the piping associated with 2NS0003 will not be affected by resetting the open and close torque switches. Since this Minor Modification will ensure that 2NS0003 will perform as required for design basis system conditions, the probability or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. No USQ exists.

Duke Power Company
McGuire Nuclear Station
Summary of Procedure Changes Completed Under 10CFR50.59

TT/1/A/9100/457

MM-4019 deleted the Unit 1 containment compressor and provided the capability of cross-connecting Instrument Air (VI) headers inside containment. The purpose of this test is to verify that each of the four main containment VI headers is capable of supplying all VI loads inside containment via the cross-connect piping.

Safety Review and USQ Evaluation:

During implementation of this test, the normal VI supply to as many as three of the four containment VI headers are interrupted at one time or another. At least one header's normal supply will remain open at all times to provide air to its loads and the loads of the isolated headers via the new cross-connect piping. Provisions are made to unisolate the normal VI supply to all the containment VI headers should any of the containment VI headers experience pressure problems while cross-connected. In order to ensure that pressure problems are detected, this procedure provides for monitoring of the containment header's after valve manipulations which could have negative effects on the header pressures.

While the containment VI headers are cross-connected, there will be no VI train separation. Since no credit is taken for VI separation in the FSAR, the lack of separation poses no negative safety related consequences. If the implementation of this procedure were to cause problems that resulted in low containment VI header pressures, the components served by the affected headers are designed to fail to their failsafe position.

This procedure provides the option of inducing heavy VI loads on the containment headers by cycling NC PORV's. Instructions are provided to N/A this cycling if plant conditions do not allow the safe cycling of these valves.

Based on this discussion, the probability or consequences of any new or existing accident or equipment malfunction should not be increased or created by implementation of this procedure. In addition, the margin of safety as defined in the basis for the Technical Specifications are not reduced.

EP/2/A/5000/01	**EP/1/A/5000/01**
EP/1/A/5000/08	**EP/1/A/5000/09**
EP/2/A/5000/10	**EP/1/A/5000/10**
EP/1/A/5000/04	**EP/2/A/5000/04**
EP/1/A/5000/02	**EP/2/A/5000/02**
EP/2/A/5000/03	**EP/1/A/5000/03**
EP/2/A/5000/05	**EP/1/A/5000/05**
EP/2/A/5000/06	**EP/1/A/5000/06**
EP/2/A/5000/08	**EP/2/A/5000/11**
EP/1/A/5000/11	**EP/2/A/5000/12**
EP/1/A/5000/12	**EP/1/A/5000/13**

EP/2/A/5000/13	**EP/2/A/5000/14**
EP/1/A/5000/14	**EP/2/A/5000/15**
EP/1/A/5000/15	**EP/2/A/5000/16**
EP/1/A/5000/16	**EP/2/A/5000/09**

Safety Review and USQ Evaluation:

These procedures were deleted based on a commitment to the NRC to upgrade McGuire emergency procedures. This is being done in order to more closely conform to the guidelines of the Westinghouse Owners Group Emergency Response Guidelines (ERGs).

The SER supplements contain deviations from the ERGs that were previously approved by the NRC for McGuire Nuclear Station. Several of these deviations were deleted or modified as a result of this rewrite to minimize the deviations from the generic guidelines. Since these items are identified in the SER, the following evaluation is provided to determine if an unreviewed safety question is generated:

The probability of an accident previously evaluated in the FSAR is not increased. The emergency procedures are used to respond to accidents already in progress and are not accident initiators.

The consequences of an accident previously evaluated in the FSAR is not increased. By eliminating the previously identified deviations, the emergency procedures are less complex. By reducing the complexity of the procedures, the potential for operator error is reduced and the likelihood of successful mitigation of an accident is increased.

The possibility of an accident which is different than any already evaluated in the FSAR is not created. The emergency procedures are used to respond to accidents already in progress and where specific procedural requirements are stated in the analysis, they will be carried forward in the upgraded emergency procedures.

The probability of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased. The emergency procedures initially ensure that all the required safety systems are in the alignments required as a result of the applicable safety signal(s). Procedure direction is provided to align systems required for safety in the event of an equipment malfunction consistent with their safety functions and design parameters.

The consequences of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased. The emergency procedures provide contingencies to the operator in the event of equipment malfunction and provide a procedural method to respond to equipment malfunctions during an accident.

The possibility of malfunctions of equipment important to safety than already evaluated in the FSAR are not created. The procedures provide operator guidance in response to equipment malfunctions during an accident. The direction to operate various equipment is in accordance with their intended safety functions.

The margin of safety as defined in the bases to any Technical Specification is not reduced. Tech Specs require emergency procedures to implement the requirements of NUREG-0737 and

Supplement No. 1 to NUREG-0737. By removing or modifying the stated deviations, the emergency procedures used at McGuire will more closely follow the generic guidelines which were based on the NUREG requirements.

Based on this review, the assumptions stated in the accident analysis are still intact and the deletion of this procedure does not create an unreviewed safety question.

TT/2/A/9100/427	**TT/1/A/9100/430**
TT/2/A/9100/425	**TT/1/A/9100/427**
TT/1/A/9100/425	**TT/1/A/9100/429**
TT/1/A/9100/431	**TT/1/A/9100/432**
TT/1/A/9100/435	**TT/1/A/9100/436**
TT/2/A/9100/436	**TT/2/A/9100/432**
TT/2/A/9100/435	**TT/2/A/9100/431**
TT/2/A/9100/430	**TT/2/A/9100/429**

The purpose of these procedures is to setup and maintain the Nuclear Service Water System operation for sufficient duration to ensure satisfactory differential pressure testing of certain valves as mandated by NRC Generic Letter 89-10.

Safety Review and USQ Evaluation:

Performance of these procedures do not place the RN system in any unanalyzed configuration. There is no potential for degrading or compromising the design or function of the system. These procedures will dynamically VOTES test the affected valves to ensure they are capable of their safety related function under design basis accident flow rate conditions. Performance of these procedures has no impact on the operability or availability of the Nuclear Service Water System.

TT/2/A/9100/442

This temporary procedure was developed to troubleshoot suspected valve leakage causing momentary pressurization in the safety injection pump suction piping immediately following operation of the 2B NI Pump. It consists of necessary steps to install, align system pressure to, obtain data from, isolate system pressure from and remove a pressure gauge on valve 2NI-212. This will allow the gauge to be left in place for the period of time necessary to troubleshoot the source of NI suction pressurization.

Safety Review and USQ Evaluation:

This pressure gauge will be installed on a temporary basis and will be removed in a short term. It is capable of withstanding the operating parameters to which it could be exposed. The installation is entirely within the auxiliary building and therefore, any leakage of NI system water resulting from this modification will be contained and separated from the environment. The PG specified for use is 0-2500 psig, well above any expected pressures in the line. The gauge will only be valved in while operators are at the gauge taking data. Since this gauge will not be left inservice and unattended, any failure of it would be immediately recognized by the operator, and steps taken to close its isolation valve, 2NI-212. Therefore, there are no instances where such failure would go undetected resulting in either a loss of system inventory or a contamination spread.

For the reasons mentioned above, it is determined that the installation, operation, and possible failure of the gauge installed by this procedure will not result in the compromise of any safety system. No additional failure modes and no unreviewed safety questions not previously considered in the FSAR are created. No USQ exists.

OP/1/A/6200/09

Temporary Mod 6403 was installed to allow NC leakage into the NI System Check Valve Test Header (CVTH) to be diverted from and thus relieve boron dilution of the CLAs. Due to changing conditions the success of the Temp Mod has been limited by the need to valve in the pressure gauge in order to monitor the CVTH pressure. Also the drain header isolation valve seats have been shown to hold better than the isolation valves upstream of the throttle valve which has resulted in pressurizing the clear plastic tubing downstream of the needle valve. This drain header isolation valve will now remain open for the duration of the Temp Mod installation.

This change opens 1NI-97 to align the pressure gauge installed by Temp Mod 6403 to the CVTH whenever in the alignment throttling with the needle valve downstream of 1NI-187 also installed by Temp Mod 6403. It also opens the drain header isolation valve downstream of the clear plastic tubing installed by Temp Mod 6403 for the duration of the Temp Mod installation. These changes are being made to allow continual monitoring of the CVTH (NI Check Valve Test Header) pressure and to reduce the risk of overpressurizing the plastic tubing.

Safety Review and USQ Evaluation:

If the valve, pressure gauge or associated tubing should fail during operation, the system can be returned to a normal alignment. Diversion of flow from ECCS is prevented by administrative closure of 1NI-96B whenever 1NI-120B is opened. Diversion of flow from the CLAs is prevented by automatic actuation of the inside and outside containment isolation valves 1NI-95A and 1NI-96B. Surveillance of the system installed by Temp Mod 6403 will be conducted twice per shift to verify changing conditions have not adversely affected the installation.

Since this test will be performed within design conditions using capable system flowpaths, the probability or consequences of a previously evaluated accident are not increased and there is no possibility that any unevaluated accidents could occur. Likewise, the probability or consequences of a previously evaluated malfunction of equipment should not be increased and there is no possibility of creating an equipment malfunction not previously evaluated. Existing Technical Specifications will be satisfied during the test and the margins of safety as described in the technical specification bases shall not be reduced. No USQ exists.

TT/1/A/9700/126

This procedure provides an operational test of the circuitry of the Diesel Generator 1A following the re-installation of the diesel generator voltage regulator. The removal of the regulator was required after the regulator failure that occurred on August 21, 1994. The test involved testing diesel generator operation. This test procedure was used only during 1EOC9.

Safety Review and USQ Evaluation:

Post installation tests for Diesel Generator 1A voltage Regulator are being performed by numerous test procedures. Among these are the subject test and TT/1/A/9700/125. TT/1/A/9700/125 will verify that the local voltage regulator controls and the remote voltage regulator controls perform the same control actions within the voltage regulator. TT/1/A/9700/126 will demonstrate that these control actions are correct. For example, one test will demonstrate that the local and control room raise voltage buttons yield the same response of the regulator motor driven potentiometer and the second test will demonstrate that the voltage is actually increased.

The Diesel Generator and its associated auxiliaries are not accident initiators and this test will not increase the probability of an accident nor create the possibility of an accident which has not been evaluated.

These tests were conducted during no-mode with Unit 1 ETA supplied from busline 2A via SATA and the 1A Diesel Generator out of service. The power supply configuration for Unit 1 has been established to minimize shutdown risk and is in accordance with Technical Specification requirements. The consequences of an accident evaluated in the SAR will not be increased.

These tests were performed to demonstrate that the Diesel generator Voltage Regulator will perform in a manner to accomplish its design function as described in the SAR and verified by pre-operational testing. Since the purpose is to demonstrate that the equipment performance characteristics are maintained there will not be an increase in the probability of malfunction nor an increase in the consequences of malfunction of this equipment. Similarly, an unanalyzed malfunction will not be created.

During the conduct of this test, the 1A Diesel Generator is not relied upon for compliance with any Technical Specification requirement and no margin of safety will be reduced. Normal protective relaying will act to protect the aligned power sources in the event that a malfunction occurs on the 1A Diesel Generator while it is connected to 1 ETA for testing.

Guidance is provided for exiting the procedure in the event that an unexpected response is obtained and terminating the test is considered the appropriate course of action. Although it is recognized that the subject test is infrequently performed it is concluded that appropriate precautions have been included and protective measures in place so that nuclear safety will not be degraded. No USQ exists.

TT/0/A/9100/458

This test collects flow and other data on the "D" Instrument Air (VI) compressor. This information will be used to determine how to modify the compressor's impeller so that the compressor can be restored to full capacity.

Safety Review and USQ Evaluation:

The operating procedure for the VI system states that the centrifugal VI compressors will be operated in the following configuration:

Lead Compressor > Set @ 105 psig
1st Backup > @ 100 psig

2nd Backup > Set @ 95 psig

This test will operate the centrifugal compressors in various combinations within the 95-110 psig range during which time all three of the compressors will remain operable supplying VI loads. However, one compressor will always be set at a pressure 5 psig below the setpoint of the other two compressors. This "backup" compressor will not be placed in any unusual alignment that might cause it to experience problems. Consequently, should the alignment/setpoint of the other two centrifugal compressors cause them to trip, this "backup" compressor should be unaffected and should remain available to help maintain VI pressures. In addition, this procedure requires that at least two of the reciprocating VI compressors be operable to assist in maintaining acceptable VI pressures if needed.

This procedure will operate the affected instrument air compressors within their design pressure range and below the design pressure rating of the VI system. The only design parameter that will be exceeded during implementation of the test is the motor amps drawn by the motor for the "D" compressor. The maximum amps that the motor can draw is designated as the current limit high (CLH) for the compressor. Controls in the compressor circuitry prevent motor amps from exceeding this CLH value (set at 405-410 amps). If necessary, this procedure will temporarily adjust the CLH for the compressor to values above the current CLH so that needed compressor data can be obtained. The vendor for the compressor motor has indicated that the motor can be safety operated as high as 477 amps for 16 minutes. Steps in this procedure will prevent the CLH from being adjusted higher than 430 amps (8 amps below the trip setpoint for the motor overload heaters.)

If the implementation of this procedure were to cause compressor problems that resulted in low VI pressures, the components served by the VI system are designed to fail to their failsafe position. Based on this and the above discussion, the probability or consequences of any new or existing accident or equipment malfunction should not be increased or created by implementation of this procedure. In addition, the margin of safety as defined in the basis for the Technical Specifications shall not be reduced. No USQ exists.