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October 18, 1996

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

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

Pursuant to 10CFR 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, 1995 to April 1, 1996.

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

Very truly yours,

T. C. McMeekin, Vice President
McGuire Nuclear Station

Attachment

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IE47 1/1

NSM-12096
NSM-22096

A new Loose Parts Monitoring System (LPMS) was installed for each plant unit by several modification packages over several refueling outages. The new LPMSs replaced existing LPMSs, which are outdated. Installation in parts was chosen due to outage work scope and ALARA considerations. The scope of this evaluation is limited to NSMs MG-12096/00, 12096/01, 12096/02, 22096/00, 22096/01 as well as VNs 12096/01B and 12096/01D.

No new failure modes are created by the new systems. 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.

There are no changes of safety limits, setpoints, or plant parameters resulting from these modifications. Existing containment penetrations are used. LPMS circuits passing through the containment penetrations are low energy, therefore, double fusing is not required. 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 these modifications. The basis for the Selected Licensing Commitment is not compromised by these modifications. The new LPMSs will provide sufficient capability to detect loose metallic parts in the NC system. Therefore, the margin of safety as defined in the basis for any Technical Specification is not decreased.

No Technical Specification changes are required. A change FSAR Section 7.7.1.12 is required when the new LPMS becomes operational. NO USQ exists.

NSM-12279/P6

Modifications to the Emergency Diesel Generator Starting Air (VG) system were performed for both Units 1 and 2 to increase diesel generator reliability, under partial NSMs MG-1(2)279/P1 through P7. Following these mods, VG will not be required for continuous control air while a diesel generator is running. MG-12279/P4 (Subtrain 1B1, P5 (Subtrain 1B2), P6 (Subtrain 1A2) and P7 (Subtrain 1A1) replace the piping, valves, air dryer and aftercooler on VG system subtrains between the air compressors and the air

tank inlet check valves. Safety Related Raw Water (RN) piping and valves serving the VG aftercoolers will be replaced with stainless steel piping and valves, and the existing strainers on the RN lines feeding the aftercoolers and dryers were removed. An RTD and temperature transmitter will be installed on VG subtrains 1A1, 1A2, 1B1 and 1B2 for local monitoring of aftercooler discharge temperature using local meters. These mods do not create any USQs or licensing issues. No Technical Specification changes will be required. Changes will be needed to FSAR section 9.5.6 (VG information), Table 9-8 (RN flow requirements), Figure 9-31 (RN Summary Flow Diagram), and Table 8-1 (Essential Power loads) to reflect these mods. No USQ exists.

NSM-12340
NSM-22340

Test panels and power sensing equipment were installed on the incoming feeders and safety system supply feeders for the following power distribution equipment:

7 KV Switchgear (EPB system)
4 KV Switchgear (EPC system)
600 V Load Centers (EPE system)
600 V Motor Control Centers (EPE system)
125 VDC Vital Power System (EPL system) (MG-22340 only)

The power sensing equipment is needed to verify analytical models used in nuclear safety-related auxiliary power calculations. The power sensing equipment itself has no safety-related functions.

Current transformers (CTs) and potential transformers (PTs) will be installed on the AC power distribution equipment, for power sensing. The CTs and PTs will interface with voltage, current, and watt transducers, which provide signals to indicating equipment.

Existing DC shunts are used on the 125 VDC Vital Power system for power sensing. The DC shunts were wired to added test blocks, where signals will be obtained for indicating equipment.

Equipment Added	**QA Condition**	**Location**
Instrument Transformers	1	Interfaces with AC power feeders in Auxiliary Building
AC Test Panels and Transducers	N/A	Turbine Building
DC Test Panels, and Test Blocks (MG-22340 only)	1	Auxiliary Building

The subject power systems are not accident initiators. The QA Condition 1 portions of the added equipment are seismically qualified. Applicable train separation requirements were implemented. An Appendix R review was conducted with no concerns identified. Therefore, the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The subject power systems supply power for accident mitigation equipment. The power systems will continue to perform the same power distribution functions. For the AC power systems, the added instrument transformers are designed and qualified for interface with safety related power feeders. Existing DC shunts, that interface with the DC power feeders, will be utilized for DC power sensing. The ability of the power systems to perform their safety functions is not degraded by the added power sensing equipment. No common failure modes are created. Therefore, the consequences of an accident of malfunction of equipment important to safety evaluated in the SAR are not increased.

The added power sensing equipment functions the same as existing power sensing equipment utilized for other purposes on the subject power systems. No new failure modes are created. No accidents previously thought 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.

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 analyses 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-12441
NSM-22441

Pressure switches in the Containment Air Release and Addition System (VQ) (VQPS5000 and VQPS5010) have proven to be unreliable in performing their control and alarm functions. The combination of a Barton 288A pressure switch and 351 bellows sensor have a process deadband of 7% of instrument span (span is 30 INWC). The combination of setpoint inaccuracy and deadband is so large compared to the magnitude of pressures in the operating range that the initiation and termination setpoints cannot reliably repeat within the desired operating range. This has resulted in numerous repair work orders and a persistent control room annunciator indicating Containment Air Release is necessary or in progress. In addition, due to the lack of reliability for the control switches, operations personnel currently perform the air additions and releases manually. This modification will delete the pressure switches, the re-set switches for resuming automatic control, and associated control circuits. Initiation and termination of air addition or release will be done manually upon receipt of the annunciators. The annunciators are to be driven by a current alarm added to the existing containment pressure loop.

In the event of an accident, the associated valves, 1/2VQ1, 1/2VQ2, 1/2VQ3, 1/2VQ5 and 1/2VQ6, are automatically closed by the St signal. This Engineered Safeguards Feature will not be impacted by this modification.

This modification will not result in any unreviewed safety questions. The deletion of the two control circuits will not impact the function of the containment air addition and release system. FSAR Section 9.5.12 and Figure 9-150 will need to be revised. No changes to the Technical Specifications are required. No USQ exists.

NSM-22445

A Westinghouse Technical Bulletin dated March 14, 1991 advised Duke that there may not be qualified isolation devices in certain circuits of the 7300 series Process Control System to provide electrical separation between it

and other non-safety related circuits/equipment to which it provides signals. Such a problem was identified for McGuire. The existing design of the Process Control (EIA) System has a non-safety related annunciator connected to a relay contact on the NCT (channel test) card. This non-safety related annunciator indicates that the channel is being bypassed for testing. The other outputs on the NCT card are used to provide outputs to Containment Spray actuation trains A and B. In this application a proven electrical isolation device shall be used to provide separation. The NCT card used in this circuit has not been proven by test to be a qualified isolator. The NAI annunciator interface card is a qualified isolation device.

This modification will take the output from the relay contact on the NCT card and wire it to the input of an NAI card circuit. The outputs of the NAI cards were routed to the IC Input Cabinet to activate the annunciator. The existing 120 VAC power supply to the annunicator circuit in the IC cabinet will be replaced with a 125 VDC power supply source. The 125 VDC power supply was routed to the IC cabinets. Four DC relays were installed in the circuits to replace the existing AC relays.

This modification does not increase the probability of an accident or malfunction of equipment important to safety that has been previously evaluated in the SAR. This modification does not increase the consequences of an accident or equipment malfunction that has been previously evaluated in the SAR. The EIA system is not considered to be an initiator of any accidents that have been evaluated in the SAR. The proposed modification enhances the reliability of the EOA system by providing qualified isolation devices between certain safety related and non-safety related circuits. This modification will prevent a fault from affecting any containment spray actuation circuits.

This modification will not create the possibility for an accident or malfunction of a different type than any evaluated in the SAR. The component (NAI card) to provide the qualified isolation function, is a device currently utilized for this purpose and has been appropriately tested by Westinghouse. This modification will not result in any new failure modes.

This modification will not reduce the margin of safety as described in the bases for any Technical Specification. The proposed modification provides a qualified isolation

device, where one did not previously exist. This will increase the safety margin, in that a fault in the non-safety related portion of the circuit can not adversely impact the containment spray actuation circuit.

NSM-22454
NSM-22455

A limited number of locations at McGuire utilize Thermo-Lag 330 material as an Appendix R fire barrier to protect safe shutdown equipment in the event of a fire. Concerns regarding the use of Thermo-Lag as an Appendix R fire barrier were raised by the NRC, as documented in Generic Letter 92-08 and NRC Bulletin 92-01. In response to NRC concerns, modifications affecting the Standby Shutdown System (SSS) control systems, the fire barriers in the Unit 2 Motor Driven Auxiliary Feedwater Pump Room, and the operating and abnormal procedures associated with the safe shutdown of the unit in the event of certain fires were committed to be completed during the End-Of-Cycle 11 refueling outage. These actions will ensure compliance with Appendix R regulatory requirements, will permit for the complete removal of the Thermo-Lag material and, thus, resolve the Thermo-Lag issue at McGuire. The implementation of the commitments made in response to Generic Letter 92-08 and NRC Bulletin 92-01 is provided by NSMs-MG-22454/00 and MG-22455/00.

NSM MG-22454 will allow the operator to isolate 250 VDC feeder voltage to the RN/CA cross connect valves subsequent to opening the valves. This modification entails installation of one "Enable/Disable" control switch on the SSS control panel for valves 2CA161C and 2CA162C.

NSM MG-22455 will install an eight inch check valve downstream of valve 2CA161C. In addition, a one inch vent valve will also be installed downstream of 2CA1651C, but upstream of 2CA221.

Presently, the automatic SSS function of valves 2CA161C and 2CA162C during a fire event are protected by Thermo-Lag. To support removal of the Thermo-lag material and not rely on the automatic operation associated with the opening of valves 2CA161C and 2CA162C, the SSS startup procedures are to be revised to allow the remote manual operation (from the SSS) of these valves and disconnect power after they have been opened.

NSM MG-22454 will allow the operator to isolate the 250 VDC feeder voltage to valves 2CA161C and 2CA162C, prior to opening these valves from the SSS. This modification will preclude the possibility of inadvertent valve closure in the event of a fire. The Appendix R Operational Safe Shutdown procedures will be revised to direct the operator to manually open 2CA161C and 2CA162C and to disable the valve motor via the Enable/Disable control switch. These manual actions would be performed within the allowed 10 minute window for transferring plant control and realignment from the control room to the SSS.

Once 2CA161C and 2CA162 are opened, condensate grade water from the Upper Surge Tank (UST) and Auxiliary Feedwater Condensate Storage Tank (CACST) would drain to the lake due to the elevation difference between the two sources (the UST and the CACST are at a higher elevation than the lake). NSM MG-22455 will install an eight inch check valve downstream of valve 2CA161C. The purpose of the check valve is to prevent the condensate grade water from draining to the lake. This will maximize the volume of condensate grade wear available for the Turbine Driven (CA) Pump (TDP) during SSS operation following an Appendix R safe shutdown fire event and ensure the continued long term supply of feedwater for the CA System TDP without relying on the current automatic operation of valves 2CA161C and 2CA162C.

No changes to the Technical Specifications are required to implement these modifications. No safety limits, setpoints or parameters assumed in any accident analysis are affected by these modifications. No USQ exists.

NSM-22457

This modification replaced the existing Feedwater (CF) venturis with flow nozzles. The nozzles and a portion of the upstream and downstream piping will be made of stainless steel. The new flow elements will provide highly accurate and reliable flow measurement, avoid the effects from fouling, and will not be susceptible to upstream tap erosion.

This modification will also install a monorail system to provide a means to remove the flow nozzle section for cleaning and inspection. Three monorails are required. One monorail in the upper transmitter shelter will span both steam generator feedwater lines. Two monorails will be required in the lower transmitter shelter, each monorail

spanning a steam generator feedwater line. The monorail system will consist of a steel monorail beam and a 1 1/2 ton hoist.

The existing venturi flow meters are susceptible to the effects of fouling deposits on the venturi surfaces and to pitting around the high pressure taps. The new flow element replacements are flow nozzles that will provide highly accurate and reliable flow measurement, avoid the effects from fouling experienced by the venturis and not be susceptible to erosion in the vicinity of its upstream pressure taps. The new flow elements will not impact the capacity or performance of the CF system, but the flow measurement quality will improve.

The monorail hoists will only be used during outages, when the CF system has been removed from service. Their metering section can be raised a maximum of 4 feet above the concrete floor. Engineering calculations have been performed that show that a much larger load, dropped from a greater height, will not fail, or cause penetration or spalling, of the concrete roof.

The Technical Specifications do not need to be revised as a result of the proposed modification. No changes to the FSAR are needed. No design bases or safety functions of any structure, system or component is affected by this modification. No USQ exists.

NSM-22473

This modification moves Diesel Generator Lube Oil System (LD) safety related pressure switches 2LDPS5120, 2LDPS5123, 2LDPS5130, 2LDPS5133 and non-safety pressure transmitters 2LDPT5121, 2LDPT5131. These instruments are being moved to improve response time for the trip switches. Along with this relocation, non-safety pressure switches 2LDPS5121 and 2LDPS5131 and non-safety pressure transmitters 2LDPT5120, 2LDPT5130 and pressure gauges 2LDPG5120, 2LDPG5130 will be deleted. The pressure gauges are replaced by electronic indicators 2LDP5120 and 2LDP5130 which will receive their signal from 2LDPT5121 and 2LDPT5131. The electronic indicators supply the signal to annunciator point 2AD19/C3 and the associated computer point. This replaces the function of 2LDPS5121 and 2LDPS5131. The overspeed dump valve and associated turbine and components which were abandoned in place by a previous change will be removed.

Safety related pressure switches 2LDPS5122 and 2LDPS5132 will also be deleted and their function down rated to non-safety related. These switches provided input to the before and after lube oil pump on low lube oil pressure. This function improves lube oil function but is not required to operate for the diesel to start and operate properly. This signal will now be provided by non-safety related pressure transmitters 2LDPT5121 and 2LDPT5131 and a current alarm module. Electrical separation is provided between this non-safety signal and the lube oil pump circuit by a safety relay.

The relocation of the instrumentation will result in shorter impulse lines and less instrumentation on the impulse lines. This should improve instrument response and lessen the chance of a false low lube oil pressure signal stopping the diesel. No instruments necessary for safe reliable operation of the diesel are deleted.

There are no USQs associated with this modification. The pressure switches being moved are described in FSAR section 7.6.15.1. Since the description for Unit 1 and Unit 2 will now be different, the FSAR will be updated to reflect the new Unit 2 arrangement. No Technical Specification changes are needed. No USQ exists.

NSM-29040/P2

This modification involves the installation of lugs to the inside of each of the four steam generator cavities. These lugs will be used to support scaffolding during the steam generator outage. These lugs will be welded plates which will in turn be welded to Type I embedded plates through each of the cavities.

The lug/lug plate material and mounting meet all the necessary design requirements and the steam generator cavity wall will continue to function as it did prior to the modification. Accident mitigation is unaffected by this modification since SG cavity wall will function as it did prior to this modification. All other accident mitigating equipment is unaffected by this modification. There are no changes to any of the safety limits, setpoints or operating parameters as a result of this modification. In addition no fission product barriers, which include containment, reactor coolant pressure boundary, fuel cladding and the fuel, are not degraded by this modification. No Technical Specification or FSAR changes are required. No USQ exists.

NSM-52395

Corporate Facilities built a new office/shop facility on the east side of the plant across the road for the Unit 2 Turbine Building. This NSM provides the tie-ins from the plant to the new office/shop facility. The modification will not change the function of any systems or components at McGuire. The following tie-ins are provided:

1. Public Address System
2. Fire Detection
3. Fire Protection
4. Compressed air
5. Industrial Waste (Shop Drains)
6. Security Lighting (exterior only)
7. Sanitary Sewer
8. Local Area Network
9. Electrical Power (provided by offsite supplier)
10. Drinking Water (provided by offsite supplier)
11. Telephone/Communications

Lighting is in compliance with the security plan commitments. There are no other security considerations. The affected systems (RY, VS, WC, EFA, ECP, ELN) or parts of these systems that are affected are not safety related. The systems have been reviewed and will not be functionally or adversely affected. The only part of the NSM that has a QA condition associated with it is the Fire Protection/Detection part which is QA3. Relocation of yard drainage piping and catch basins will not affect the plant or plant systems. No accident initiators or safety equipment have been affected by this modification; 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 no safety system is adversely affected and they are expected to respond as before, there is no increase in the consequences of an accident or of a malfunction of equipment important to safety evaluated in the SAR. The level of security is not decreased. No new failure modes for any safety related systems have been introduced. Based on this and the above, the possibility either 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 will be 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-3409
MM-3416

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 station will accomplish in order to comply with the Generic Letter. The minor modifications provide for the diagnostic testing for MOV 1CF0151 and 1LD0108 and constitute part of the actions necessary for compliance to NRC Generic Letter 89-10.

The actual changes involve re-setting the open and close torque switches so that the motor operator will produce the necessary torque, that will be converted by the stem nut to thrust, to fully open and/or fully close the valve disc when design basis system conditions present. The minimum required and maximum allowed thrust used as the test acceptance criteria has been determined by an engineering calculation. This engineering calculation was performed in accordance with the latest revision of the Duke Power specification which establishes the parameters and criteria used to determine the minimum required and maximum allowed thrust levels. The diagnostic test system used to facilitate thrust testing has been included in the engineering calculation. The final output thrust level achieved during the diagnostic test will be sufficient to allow valve operation at design differential pressure and system pressure without exceeding the limitations of the operator or valve components.

The MOV affected by Minor Modification 3416 is in the Diesel Generator Engine Lube Oil System. The function is to provide a bypass of the full flow lube oil filter on high differential pressure. The safety position is open and its normal position is closed. The MOV affected by Minor Modification 3409 is in the Feedwater System. The function is to provide system isolation to each auxiliary feedwater nozzle. The safety function is to close and its normal position is open. Re-setting the open and close torque

switches will not affect open and closure times. The existing stress analysis of the piping associated with these valves will not be affected by re-setting the open and close torque switches. Since these modifications will ensure the 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 USQ exists.

MM-3465
MM-3466

These modifications were implemented as a result of the 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.

These valve operators were modified to resolve concerns of not attaining sufficient thrust to completely pull the valve disc out of its seat under normal or high DP conditions. Presently on these valves, the torque switch bypass contact is located on the primary switch pack. Due to the primary switches characteristics, this bypassing action does not stay in the circuit but for a very short period of time. This is normally on the range of 5% of total valve travel. Due to system conditions with the present setup, after the bypass circuit opens, there still may be high resistance in the seating area that could cause the actuator motor to cut off due to insufficient torque. To assure that these valves will open fully, the modified torque switch bypass contact will be in the circuit for 50% of the valves travel, +/- 25%. This will ensure that the maximum motor torque will be applied to the unseating action. After this bypass circuit drops out, the torque switch will be in the circuit to deenergize the actuator should a high resistance be present after complete unseating. The torque switch bypass circuit will be moved to the AOP auxiliary switches, and the computer points will be moved to the primary switch pack. By moving the computer indication to the primary switches, this will also give the computer a more accurate stroke time.

Functionally, these valves will operate identically to their present operation. This switch modification will not affect open or closing times. With these modifications , these valves will be more reliable in obtaining the desired

positions. Other indications and interlocks will also not be affected.

The MOVs affected by these Minor Modifications are in the Diesel Generator Engine Lube Oil system. The function is to bypass the full flow lube oil filter on high differential pressure. Their safety position is open and normal position is closed. The existing seismic analysis and stress analysis of the associated piping will not be affected by the implementation of the 50% torque switch bypass modification. Since these modifications ensure the valves 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 SAR is not increased. No USQ exists.

MM-3620

This modification converted the hoist connection on the Control Rod Drive Mechanism (CRDM) Missile Shield lifting rig from a female to a male type connection. Brackets and bolts were added for securing the balance position of the lifting rig beam.

The modification allows the use of a new load cell which has two female-type connections. A load cell is used to measure the load on the rig during lifting operations.

The CRDM missile shield lifting rig is used to remove the concrete shields that cover the Control Rod Drive Mechanism. The shields are removed for reactor refueling operations. The lifting rig is used to perform this operation during plant modes 5 (cold shutdown), 6 (refueling), and "no-mode" (defueled). The lifting rig and missile shields are QA Condition 1 qualified.

The CRDM Missile Shields and lifting rig are not evaluated as accident initiators in any FSAR Chapter 15 accident analyses. The modified lifting rig and the new load cell are qualified to QA Condition 1 requirements. Therefore the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The CRDM lifting rig does not perform any accident mitigation functions. The lifting rig is QA Condition 1 qualified because of its use in lifting heavy loads over the reactor. The modification will not degrade the ability

of the lifting rig to perform this safety function. The CRDM lifting rig will perform the same functions as before the modification. Therefore the consequences of an accident or malfunction of equipment important to safty evaluated in the SAR are not increased.

The modification adds no new functions to the lifting rig. The modification enables the use of a load cell for measuring load on the lifting rig during lifting operations. The load cell is a new component added to the lifting rig system. However, there are no new failure modes created by the addition of this new component. The load cell meets specifications required for use in the intended application. 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-3721

This modification changes the diesel generator lube oil heater pump internal relief valve setting from 75 psig to 50 psig. The relief valve numbers are 1/2 LD11 and 41. The purpose of the relief valve is to provide overpressurization protection for the pump should the pump discharge valve be inadvertently closed. Changing the relief valve setpoint will not affect this design function. Since normal pump discharge pressure is 12-14 psig, there will not be a problem in lowering the pump relief valve to 50 psig. The 50 psig setpoint matches the setpoint of relief valves 1/2 LD 13 and 43 downstream of the lube oil heater. The safety related function of the lube oil heater loop is to maintain DG standby oil temperature and provide a pressure boundary to loss of lube oil during a seismic event. The relief valve setpoint change will not affect the safety function of the DG lube oil heater loop. No USQ exists.

MM-3860
MM-3866

This modification installs check valves on the valve actuators for the main steam power operated relief valves. These actuators operate to open and close valves 1/2SV-0001, 1/2SV-0007, 1/2SV-0013, and 1/2SV-0019. The valves are installed on the actuators using piping and fittings in accordance with Pipe Specification 151.4 and are not safety related. The check valves will be installed on the spare ports of the actuator and will serve to allow air into the actuator during valve closure. This will prevent a vacuum from forming behind the actuator piston as the valve is closed and improve valve closing time.

The PORVs serve to relieve pressure and dissipate heat in the Steam Generators when necessary by opening on an overpresssure signal in the associated steam line and venting steam to the 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 actuators operate using Station Air (VI) for motive power. This is not a safety source of air and thus opening of the valve is not safety related. Valve closure is a safety related function but does not depend on VI. The check valves will not impact the safety analysis of the valve operation. The stroke time of the valves 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 valve for the PORVs are closed and de-energized when the PORV fails to close. The closing function of the PORV is accomplished whether the check valves operate as designed or not. Should the check valves fail in the open position, the PORVs will go to their safe position of closed. The failure of the check valves to operate may increase the stroke time of the PORVs but they would operate in the same manner as before installation of the check valves. Failure of the valves to close will cause the PORVs to be unable to open. This is not a safety function of the valves. The safety relief function is provided by valves 1/2SV2 through 1/2SV6, 1/2SV-8 through 1/2SV-12, 1/2SV-14 through 1/2SV-18, and 1/2SV-20 through 1/2SV-24.

The addition of the check valves does not impede any safety function of the PORVs. The only safety function of the PORVs is to close for containment isolation. This function will be accomplished regardless of check valve operation. The function of the check valves is to enhance PORV valve closing but failure of a check valve either open or closed will not prevent the PORV from closing to meet its safety function in a safe time. The accident analysis may assume that a PORV fails to close. This failure is less likely with addition of the check valve for vacuum relief. Therefore, there is no increase in the probability of an accident or malfunction of equipment important to safety evaluated in the SAR. The failure of PORVs to operate is already considered in accident analysis. Therefore there is no increase in consequences of an accident evaluated in the SAR. The check valves serve to enhance operation of the PORVs but do not change the function or operation of the valves or valve actuators. Therefore, the modification does not create the possibility of an accident or malfunction of equipment different than those considered in the SAR. The failure of a check valve to close may prevent opening of the associated PORV. The opening of the PORV is not a safety function. No failure of a check valve leads to a failure of the PORV to close which is the safety function of the valve. Therefore, there is no increase in the probability or consequence of a malfunction of equipment important to safety evaluated in the FSAR. The PORVs function as a containment isolation valve is not changed. No closure time requirements are specified for these valves. Therefore no reduction to the margin of safety defined in the bases of the technical specification is created by this modification. No USQ exists.

MM-3919

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 station will accomplish in order to comply with the Generic Letter. This minor modification provides for the

diagnostic testing for MOV 1RN0086 and modification of the opening control logic. This minor modification constitutes part of the actions necessary for compliance to NRC Generic Letter 89-10.

The diagnostic portion of the minor modification to 1RN0086 involves re-setting the open and close torque switches so that the motor operator will produce the necessary torque to fully open and/or fully close the velvet disc when design basis systems conditions are present. Re-setting the torque switches is accomplished on a torque test bench. The criteria used for the bench test, the minimum required and maximum allowed torque has been determined by engineering calculation. This engineering calculation was performed in accordance with the latest revision of the Duke Power Specification which establishes the parameters and criteria used to determine the minimum required and maximum allowed torque levels for 1RN0086. The inaccuracies of the diagnostic test system used to facilitate torque testing for 1RN0086 have been included in the engineering calculation. The final output torque level achieved during the diagnostic bench test will be sufficient to allow valve operation at design differential pressure.

This minor modification involves removing the opening torque switch from the control logic for a larger portion of the opening travel. Presently for 1RN0086, the torque switch bypass does not cover the new requirement of 90 % ± 5% of the opening travel. Under the current open torque switch logic, when 1RN0086 is 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 in 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 (resistance) time, full capabilities of the motor are not available and failure to satisfy the safety function is a possibility. To assure that 1RN0086 will fully open, the modified open torque switch bypass contact will be in the opening circuit for a minimum of 90% of open travel with a set-up tolerance of ± 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 will be in the circuit to deenergize the actuator should the controlling limit switch fail.

Functionally, 1RN0086 will operate identically to its present operation. This switch modification will not affect open or closing stroked times. With this modification, 1RN0086 will be more reliable in obtaining the desired positions.

The MOV affected by this minor modification is in the Nuclear Service Water System. 1RN0086 provides isolation of RN A train supply to the KC heat exchanger. The safety function of 1RN0086 is to open and its normal position is open. 1RN0086 is considered an ACTIVE valve. Re-setting the open and close torque switches as well as installing the 90% ± 5% open torque switch bypass will not affect open and closure times. The existing stress analysis of the piping associated with 1RN00686 will not be affected by re-setting the open and close torque switches. Since the modification will ensure that 1RN006 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-4035

This modification relocated a roof access ladder from the south side of the North Personnel Access Portal (PAP) to the east side. The new location places the ladder outside the protected area fence, but within the area for vehicle inspection and access.

The ladder is being relocated to facilitate weekly inspections of equipment on the building roof outside the protected area fence. With the ladder in the existing location, a security guard is required to be posted at the roof gate during equipment maintenance. The new ladder location allows access to equipment without opening the roof gate, therefore, only security notification will be required.

The PAP is not an accident initiator in any accident analyses, not is it an accident mitigator. Therefore the probability or consequences of an accident previously evaluated in the SAR is not increased.

The modification involves no plant safety equipment or plant safety functions. The PAP is a building exterior to the plant. 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 PAP is an SSC evaluated as part of the Physical Security Plan for the station. The modification meets the requirements of this plan. No new functions 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 clading) are not degraded. No assumptions made in any accident analysis are affected by this modification. Therefore the margin of safety as defined in the basis for any Technical Specification is not decreased. No USQ exists.

MM-4039
MM-4040

The purpose of these modifications is to upgrade the non-nuclear safety related Auxiliary Feedwater system to steam generator injection flow rate instrument loops. The "as-built" instrument loops have an output range of only 0-300 gallons per minute. This span is insufficient for monitoring the combined output of two auxiliary feedwater pumps (motor-driven and turbine driven). The new revised instrument loop range will be 0-600 gallons per minute. These modifications will allow the full monitoring of the CA injection flow rate to each steam generator versus the current over-range condition experienced by the existing instrumentation.

Actual field changes involve the replacement of the four non-nuclear safety related transmitters with new Rosemount Smart transmitters capable of handling the higher process input differential pressure. The remote received gauges will be modified by replacing the existing meter face plate with a new 0-600 gallons per minute scale. The Operator Aid Computer analog points will be revised to reflect the new instrument loop span.

These minor modifications will not compromise the design or function of the CA system in any manner. These minor

modifications will not affect the operation of the nuclear safety related instrument loop sharing the same injection flowrate metering orifice. The affected instrument loops are not nuclear safety related of any QA condition. The use of non-nuclear safety related instruments in tandem with nuclear safety related instruments has been reviewed and approved under MCC-1381.05-00-0223. These modifications will not change the instrument loop configuration as noted in the calc. The new transmitters have been reviewed from a pressure boundary integrity standpoint and is approved for this use code and application. No USQ exists.

MM-4045

This modification replaces the Class C Component Cooling (KC) check valve (2KC-958) on the outlet side of the Unit 2 Post Accident Liquid Sample Panel (PALS) with a Class C manual isolation valve (2KC-974) . This valve acts as the class break boundary between the downstream Class C piping and the upstream, non-seismic, Class E piping, and serves to limit loss of component Cooling System (KC) inventory due to the failure of the Class E piping. The manual valve will perform this function as well as the existing check valve, since 2KC-974 will be administratively maintained in the closed position, and will be opened and closed to support use of the PALS panel. Note that this modification increases the number of valves which must be opened to use the PALS panel by one and this will not effect the ability of the panel to be used for its intended function. The probability for increased loss of reactor coolant inventory following a postulated seismic event will not be increased by this modification. Likewise, the quantity of previously analyzed KC inventory losses will not be increased.

The Nuclear Sampling System (NM) and the PALS panel do not serve an emergency function and are designed to be operated manually. This modification will therefore not increase the probability or consequences of an accident evaluated in the SAR. Likewise, there will be no increase in the probability of a malfunction of equipment important to safety evaluated in the SAR.

This modification will not change the function or capability of the KC and NM systems. The possibility of a malfunction or accident of a different type other than previously analyzed in the SAR is not increased.

No safety limit, setpoint, or operating parameter will be changed by this modification. Therefore no margin of safety as defined in the basis of the Technical Specifications will be reduced. No USQ exists.

MM-4097

This modification allows 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 Residual Heat Removal and Containment Spray 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 will be added directly in place of existing drain plugs in some cases. Reducers will be required where there are diameter differences between oil drain and drain valves.

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 flow in the vicinity of the equipment. Commercial Grade and seismically qualified valves will be used. Siesmically qualified attachment configurations (reducers and standoffs) will be used. Reducers and tubing will be 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 qualification and Commercial Grade qualification , the oil drain valves and attachments will be 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 failure modes 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 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 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-4109

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 will be 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 will be waterproofed and handrails added as needed for safety reasons.

The administrative controls incorporated during the implementation of this modification ensure that there will be not 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 will be jeopardized by implementation tasks. Therefore, the consequences of accidents analyzed in the FSAR are not increased by the modification.

The construction of the Turbine Building has been reviewed to ensure that aboom 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. There is no increase in the probability of equipment malfunction. Administrative controls will be used to alleviate the potential for equipment material handling accidents during the implementation of 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 system due to improved materials. The margin of safety as defined in the Technical Specification 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-5175

This minor modification modifies the mounting configuration for the removable door panels on the diesel generator battery charger cabinets (EPQ system), located in the diesel generator rooms. The modification will allow easier

access to the cabinets for routine weekly maintenance on the DG batteries. The modification is applicable to Units 1 and 2, Train A and B DG battery charger cabinets.

The EPQ system is not evaluated as an accident initiator in any FSAR Chapter 15 accident analysis. Therefore the probability of an accident previously evaluated in the SAR is not increased.

The modification was reviewed and approved to meet seismic requirements. QA Condition 1 requirements will be met for the cabinet bracing. QA Condition 4 requirements will be met by the door fasteners. The modification involves no electrical changes. No electrical separation or Appendix R requirements are applicable. Therefore, the probability of a malfunction of equipment important to safety previously evaluated in the SAR is not increased.

The diesel generators perform the plant safety function of providing power for plant safety equipment in case of loss of offsite power. The DG batteries supply power for the DGs rating. Seismic integrity of the battery charger cabinets is maintained by the proposed modification, therefore, the performance of plant safety functions are not degraded. There will be no change in EPQ system functions, and no failure modes are created in common with the other DG units/trains. Therefore the consequences of an accident or malfunction of equipment important to safety evaluated in the SAR is not increased.

There are no new functions or failure modes created by the proposed modification. 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 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-5441
MM-5442

These minor modifications installed key switches in six turbine trip solenoid valve circuits. 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 complete. This should improve turbine trip protection reliability. No new failures or consequences of a failure are created by the installation of these switches.

MM-5451
MM-5452

These modifications were implemented to resolve a recurring problem regarding the DG FD Fuel Oil Booster Pump Breaker accidentally opening while work was being performed in the DG rooms. The breaker is located in a floor mounted box, with the handle at approximately knee level. The handle is being bumped inadvertently as people pass by the box, due to the congested walk area around the box. The breaker serves a safety function, so an undesired circuit isolation is unacceptable.

These modifications will not decrease the level of nuclear safety. The cover will not prevent the automatic fault clearing function of the breaker, should this be required. It will not prevent the equipment operators from opening or closing the breaker as required. The presence of the cover will not cause the breaker to open or close, but will prevent the breaker from being accidentally operated.

This design has been reviewed by Civil Engineering for seismic impact. There is no seismic impact to this QA1 equipment due to the addition of the cover. The cover itself has minimal weight, so it will not induce a seismic related problem. The bolts which will hold the cover in place will be sealed with silicone RTV to prevent any possibility of water inleakage, which poses no safety concern. No USQ exists.

The motor control center breakers, fan motors and associated equipment modified and/or replaced under this minor modification are not nuclear safety related. The changes made under this modification are being made to achieve better breaker coordination between the motor control center feeder breakers and the branch breakers in the transformer control panels. It is very difficult to obtain perfect breaker coordination with serves molded case circuit breakers. However, this modification will achieve the best coordination available between the TED and KA breakers without sacrificing coordination with the upstream breakers. The magnetic instantaneous trip setpoint in the KA3200 (Westinghouse) feeder breakers will be set to position "4" (1500 Amperes) which is a change from "Lo" (1000 Amperes), by this modification.

There are no adverse impacts to the cabling associated with the components affected by this modification. The cables connecting the KA3200 breakers to the transformer control panels are 250 MCM, 3 conductor cables. The combination of a KA3200 trip mechanism and a 1500 amp setpoint is within the appropriate rating of the cable, as calculated by the CAPTOR program, so there will be no damage to the cable insulation during a fault condition. The cabling within the transformer control panels is still protected by the originally installed GE-TFJ 150 amp and GE-TED 15 amp breakers, so there is no adverse impact to the internal panel cables either.

The correct motor for the cooling fan motor application on the main power transformers is the 1/2 HP motor specified in the minor modification. The manufacturer's drawing originally specified a 1/2 HP motor for this application. The root cause of the excessively high motor failure rate is apparantly the undersizing of these motors. This modification should increase the motor life, since the 1/2 HP motors will be required to do the same amount of work now required of the 1/3 HP motors (but have a 50% greater HP rating). The two motors are identical in dimensions, shaft size, frame and mounting details, RPM, etc. They differ only in HP and full-load current (1.0 amps for the 1/2 HP motor vs 0.75 for the 1/3 HP motor). The information on our motor sizing discrepancy, as discussed in the body of the mod, was obtained in a telephone conversation with General Electric. There is no negative impact to plant safety due to the installation of the 1/2 HP motors, in fact; plant reliability will be increased.

The fuel-load current through the KA3200 breaker will be 85.2 amps with both cooling groups powered by one motor control center. The full-load currents through the 150 amp main breakers for cooling groups 1 and 2 will be 47 amps and 38 amps, respectively. The current through each of the TED branch breakers serving a cooler unit will be 9.4 amps (each cooler has one pump motor with 5.4 FLA and four fan motors with 1.0 FLA). There are 5 coolers in cooling group 1 and 4 coolers in cooling group 2. There is a heater circuit (0.3 amps) on each cooling group. All breakers are comfortably loaded (all less than 80% rated current) even with the new motors installed. The use of the 1/2 HP motors will not in any way degrade the transformer cooling units. For these reasons, therefore, there is no adverse impact to plant safety. No USQ exists.

MM-6164
MM-6165

These modifications allow the station to remove the unused Rod Control Cluster Assembly (RCCA)mast, hoist, tower, and other associated components from the unit 2 manipulator crane. The mast was intended to be used to shuffle rod control clusters from one fuel assembly to another inside of the reactor building. However, this shuffle is actually performed in the fuel building after the entire core has been unloaded so the mast is never used. Unless the mast is removed the crane cannot be parked at its southmost location during the SG replacement outage. The crane is currently treated as a non-QA Condition component. However, the crane is currently being evaluated to determine if the crane should be upgraded to QA Condition 4, seismically designed. Therefore, for the subject modification the crane will be conservatively treated as a QA-4 structure. This only affects the addition of the cover plate over the hole created by removing the mast and tower. No formal calculation has been performed to qualify the plate attachment (i.e. welds, bolts, and hinges), but due to the low mass of the plate the attachment is obviously adequate.

The manipulator crane, with or without the RCC mast has no significant affect on the consequences of any SAR accidents. The absence of the RCC mast will make the manipulator crane less likely to fail in a seismic event (due to the reduced mass) and therefore less likely to initiate an accident not previously evaluated in the SAR.

There are no significant affects on the consequences of any SAR equipment malfunctions. Removal of the RCC mast should in fact preclude the probability of a safety related equipment malfunction by reducing the chance of a seismic interaction of the manipulator crane with other equipment because of the reduction in mass of the crane. These modifications will result in changes to the FSAR which will note that the RCC mast has been removed. No USQ exists.

MM-6207

The use of pipe in lieu of duct in safety related air handling systems has been a source of confusion. Duke Class H piping is non safety related, but pipe functioning as duct can be safety related Class H duct as defined in Duke Nuclear Guide 1.26. This modification made editorial revisions to HVAC flow diagrams and selected supporting documentation for the purpose of clarifying the QA condition labeling. This modification does not increase the probability of an accident evaluated in the SAR. Editorial revisions to FSAR Figures 9-121 and 9-122 are required. No USQ exists.

MM-6588

This modification installs manual test isolation valves for Instrument Air (VI) containment penetrations 2M220 and 2M359. The isolation valves will be located just upstream of the outside containment isolation valves for penetrations 2M220 and 2M359 (2VI129 and 2VI160 respectively). In addition, a branch line will be relocated to upstream of the new isolation valve for each penetration. This work will allow these penetrations to be isolated and tested without securing VI flow to other equipment.

The equipment associated with the affected headers is designed to fail to the safe position and the work is to be performed during No Mode, therefore this modification should not increase the consequences of a malfunction of equipment important to safety and the possibility of an unevaluated malfunction occurring is not created by the implementation. No Technical Specification or FSAR changes are required. No USQ exists.

MM-6936
MM-6937

These modifications will replace the existing 5KVA hydrogen mitigation step-down transformer with a 7.5KVA transformer. While performing these modifications, the transformer secondary conductors to the panelboard will be increased accordingly and the circuit breaker on B train will be decreased to a 20A circuit breaker to be consistent with train A.

These modifications affect the Hydrogen Mitigation System (EMH), the 600 Volt Essential Power System (EPE) and the Emergency Diesel Generator (EQA) loading. The non-safety EHM panelboard and transformer are normally supplied by the EPE system and the EPE system is supplied by the EDG during design basis events. Since this modification is to increase the capacity of the 600/120 volt transformer, the support electrical systems must be reviewed for adequacy with the increased load. The EPE system can supply the additional 1KVA demand and decreasing the circuit breakers will not affect the proper operation of the circuit. The EDG must supply this panel during a loss of power and based on the sizing calculation there is additional 1KVA margin available. Adding 1KVA to the EDG requires FSAR Table 8-1 be updated to represent the increased load.

Since the power system can supply the addtional 1KVA to the EHM system, there is no detrimental affect on the power system and the EHM panelboard has an improved power supply. The EPE and EQA systems are still capable of providing the safety functions and design bases.

Based on this evaluation, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR is not increased. Also, the possibility for an accident or malfunction of a different type than any evaluated previously in the SAR is not created. The margin of safety as defined in the basis for any technical specification is not decreased by this change. No Technical Specification changes are required. FSAR Table 8-1 will be changed to show the increased KVA. No USQ exists.

MM-7067
MM-7068

During certain operating conditions, differential expansion of the condenser inlet and outlet tube sheets causes the

boot seal trough on the outlet side of the condenser to rise to a higher elevation than on the inlet side. This results in the existing overflow line to rise out of the water, resulting in spillage out of the trough on the inlet side and the ends of the condenser. This spillage is a housekeeping concern. Installation of another overflow line on the north end of the 1A and 2A condenser boot seal trough and on the south end of the 1C and 2C condenser boot seal trough should eliminate the spillage. These modifications do not increase the probability or consequences of an accident evaluated in the SAR, nor do they create the possibility for an accident or malfunction of equipment important to safety evaluated in the SAR. There is no reduction in the margin of safety as defined in the basis for any technical specification. There are no technical specifications that address the level or presence of water in the boot seal troughs. Plant functions are not degraded by these modifications. No fission product barriers are affected. FSAR figures 10-1 and 10-20 will be revised to show the new overflow lines form the boot seal troughs. No USQ exists.

MM-7096
MM-7125

Sight glasses are to be added to Unit 1 and Unit 2 "A" and "C" reactor coolant pump motor lower bearing cooler drain piping and reactor building KC drain header to determine drain valve seat leakage and input to KC drain tank from the reactor buildings. These sight glasses will be installed into the non safety related portion of the KC system. These modifications will not increase the probability of an accident evaluated in the SAR. Revisions to FSAR figures 9-62 and 9-64 are required. No Technical Specifications are required. No USQ exists.

MM-7130

The automatic turbine trip initiated on any one of six high exhaust hood temperature signals will be removed. This modification does not prevent the turbine from performing its intended function. The removal of this non-safety trip will prevent spurious trips associated with the malfunction of one of the temperature switches. Control room indication is still provided to the operators to allow them to make a determination to trip the turbine if the high-high setpoint is reached. No USQ exists.

MM-7757

This minor modification will replace reactor head vent solenoid valves 1NC2173 and 1NC275 with Target Rock model 79L-009BB valves. The model 79L-009BB is a 1" valve identical to the present 79L-009-001 solenoid valve being used except that instead of a screwed and seal welded body/bonnet joint, it uses a bolted body/bonnet joint. The 79L-009BB will be procured under the same specification as the 79L-009-001, with exception to section 8.7.4 which calls for the screwed and seal welded body/bonnet joint, it uses a bolted body/bonnet joint.

The reactor head vent valves are suspected as potential sources of leakage. 1NC274 and 1NC275 will be replaced at the same time as 1NC272 and 1NC273. There are two spare 79L-009-001 solenoid valves that will be used to replace 1NC272 and 1NC274 that are the same design as the currently installed seal welded body/bonnet sealing design for 1NC273 and 1NC275.

The piping arrangement and configuration will be unchanged by this modification. FSAR section 3.5.2.4, Missile Protection, Missile Selection, Valves" states that valves under 2" do not present credible missles since the bonnet is screwed into the body of the valve and forms a canopy seal.

The replacement of the Reactor Head Vent Solenoid Valves 1NC273 and 1NC275 with Target Rock model 79L-009BB valves that use a bolted body/bonnet joint does not involve an unreviewed safety question. The basic mechanical, structural and electrical behavior of the valves and the head vent system are unaltered by the modification.

The valve design is in full compliance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1977 Edition with Winter 1977 Addenda. There are no new failure mechanisms introduced and no reduction in the margin of safety. The barriers to fission product release are unaffected by the modification. No USQ exists.

MM-7948

This change allows for varying amounts of resin to be loaded into the Spent Fuel Pool (KF)demineralizers. Presently, FSAR table 9-1, Spent Fuel Cooling System Component Design Data, shows the resin volume of the Fuel

Pool Cooling Demineralizer to be 40 ft(3), with a bed depth of 2.5 ft. This change allows for resin loads of between 15 and 40 ft(3), with corresponding bed depths of 1.0 to 2.5 ft. Allowing a smaller resin load affects only the amount of time the resin load will be in service, not affecting the ability of the resin to keep the spent fuel pool water clean. No USQ exists.

MM-8024
MM-7624

These MMs were generated for the plugging and stabilization of steam generator tubes during the EOC 10 refueling outage. Station Technical Specifications require the periodic inspection of steam generator tubes to identify defective tubes. Duke Power Company removed from service defective tubes and tubes at its option by the installation of 3/4" diameter rolled plugs in the hot and cold leg side of the tube. These MMs document the installation of all new steam generator tube plugs for EOC 10. If during the eddy current inspection tubes are confirmed to have circumferential defects at the top of the tubesheet, these tubes may be stabilized by the installation of a 3/4" Tube Cable Stabilizer Assembly of 3/4" Tube Segmented Stabilizer. Circumferential defects under specific loading and environmental conditions, may continue to propagate until tube severance after the tube is removed from service. Stabilization spans the defect with a cable or segmented stabilizer and prevents the cantilevered tube form impacting adjacent tubes.

Nuclear Engineering Safety Analysis performed a thermal hydraulic evaluation to support plugging of up to 24% of the tubes in any one steam generator and/or 20% total for all four steam generators. This calculation documents that plugging up to the given percentage of steam generator tubes does not constitute an unreviewed safety question and Technical Specification changes are not required. This calculation provides written explanation for all "no" answers for this evaluation with regard to the impact that plugging has on the existing plant licensing basis, unit operation, and performance. The BWNT (Framatome) stabilizer safety evaluation discusses the design of the stabilizers and concludes that they are adequate to withstand any worst case loading that they might be subjected to during operation. Duke has reviewed and concurs with their conclusions.

Field work associated wit the installation of the plugs and stabilizers will be performed under Duke reviewed and approved Framatome QA program and procedures. Adequate QA controls exist with the installation process, procedures and plug and stabilizer manufacture to ensure that the installation will not affect steam generator material strength to increase the likelihood of tube/plug damage. The probability or consequences of an accident previously evaluated in the SAR will not be increased. The probability or consequences of a malfunction of equipment important to safety previously evaluated or different than already evaluated in the SAR will not be created. The stabilization and plugging of the affected tubes will serve to maintain the integrity of the NC system pressure boundary and will ensure that the plant design basis and steam generator operability is maintained. The number of steam generator tubes plugged will be verified not to exceed 24% for any one steam generator or 20% for all four steam generators. No USQ exists.

MM-8069

The Waste Gas System compressor packages (except relief valves 1WG-30 and 1WG-44) have been downgraded from Duke Class C (ANS Safety Class 3) to Duke Class E (ANS Safety Class NNS). The quality group classification of the system and associated equipment is based on the potential of the system to release effluent greater than accepted limits. The criterion for accidents involving radioactive releases of the Waste Gas System, is that of site dose shall be substantially below the 10 CFR 100 limits. A 0.5 rem limit has been established by Standard Review Plan 11.3 and Branch Technical Position 11-5.

Calculations have been performed to evaluate the radiological consequences of a major leak or a component failure within the Waste Gas System for the McGuire Station. The results confirm that for an unplanned release, the maximum potential off-site exposure for the worst case failure is 222 mrem. No USQ Exists.

MM-8262

This minor modification replaces valve 2NI-180 in the Safety Injection (NI) system with a new valve. This valve has been evaluated and the applicable NI system calculation have been reviewed and reconciled for the changes. No new failures are introduced or operating characteristics of the

ECCS changed by this minor modification. The control logic is unaffected, and there are no Appendix R concerns. No USQ exists. No changes to the FSAR or Technical Specifications are required.

MM-8285

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 Compnay nuclear station will accomplish in order to comply with the Generic Letter. This modification provides instructions to replace the existing actuator because of abnormally high running amperage. It also provides the diagnostic test criteria that will be used to test the replacement actuator. This minor modification constitutes part of the actions necessary for compliance to NRC Generic Letter 89-10.

The diagnostic portion of the minor modification to 2RN0042 involves re-setting 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. Re-setting the torque switches is accomplished on a torque test bench. The criteria used for the bench test, the minimum required and maximum allowed torque has been determined by engineering calculation. This engineering calculation was performed in accordance with the latest revision of the Duke Power Specification which establishes the parameters and criteria used to determine the minimum required and maximum allowed torque levels for 2RN0042. The inaccuracies of the diagnostic test system used to facilitate torque testing for 2RN0042 have been included in the engineering calculation. The final output torque level achieved during the diagnostic bench test will be sufficient to allow valve operation at design differential pressure.

The MOV affected by this modification 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. 2RN0042 is considered an ACTIVE valve. Re-setting the open and close torque switches will not affect open and closure times of 2RN0042. The existing stress analysis of the piping associated with 2RN0042 is not affected by re-setting the open and close torque switches. Since this 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 evaluation is the FSAR is not increased. No USQ exists.

Revision to NRC commitment (FSAR change) regarding the monitoring of eight Postulated Break Locations in the Cold Leg Accumulator Injection Piping

This evaluation involves a regulatory commitment change regarding the monitoring of the eight postulated break locations in the 10-inch cold leg accumulator injection piping with the acoustic leak detection system. The main differences between the current commitment and the revised commitment are: 1) the sensitivity requirement that the leakage detection equipment will have to meet, and 2) the leak rate from the postulated break locations at which an orderly shutdown of the unit would begin.

The intent of the current commitment is to provide early warning of possible degradation of the pressure boundary at the postulated break locations so that appropriate action (unit shutdown) can be taken prior to catastrophic failure of the 10 inch CLA injection line. The ALDS provides the monitoring function. The proposed changes to the current commitment will, in effect, allow the RCS leakage detection system to perform the monitoring function. Although the RCS leakage detection system is less sensitive that ALDS, the RCS leakage detection system will still be able to provide early warning of leakage from the postulated break locations. The indication of leakage from the postulated break locations by the RCS leakage detection will still be well in advance of the reactor coolant piping failing catastrophically (rupture).

Analysis has been performed to show that even with a leak rate of approximately 20 gpm, rupture of the CLA injection piping will not occur. A leak rate of 20 gpm is well within the capabilities of the RCS leakage detection system. As such, the RCS leakage detection system will

provide early warning of possible degradation of the pressure boundary so that unit shutdown prior to catastrophic failure of the piping would still be accomplished.

The proposed commitment change will not result in any unreviewed safety questions or licensing issues. No changes to the Technical Specification will be required to implement this commitment change. No design bases or safety function of any structure, system or component is adversely affected. The FSAR will be updated as a result of this commitment change.

OP/1/A/6400/05

This procedure provides the instructions for operation of the Temporary Component Cooling (TKC) System and contingencies in the event cooling is lost to the Spent Fuel Pool (KF) 1A heat exchanger and to the seal water heat exchanger. Directions for shutting down the TKC System are also delineated within the Temporary Operations (TO) Procedure. This procedure provides instruction on filling and venting of the TKC System, including the alignment of station systems and components (valves) to support filling and venting of TKC. The procedure also provides direction for isolating, draining, filling and venting the piping between 1KC-1A and 1KC-2B.

The TO procedure results in isolating the Unit 1 AB non-essential header from its heat loads for a short period of time (less than 72 hours) and the operation of the TKC System to provide cooling flow to the KF-1A heat exchanger and to the seal water heat exchanger. To replace 1KC-1A, the Unit 1 AB non-essential header will need to be isolated. This results in terminating cooling flow to several heat loads/heat exchangers. The temporary loss (up to 72 hours) of cooling flow to several of the AB non-essential heat loads is of no consequence or will be handled by other means. The loss of normal cooling flow to the KF heat exchangers and to the seal water heat exchangers for an extended period of time does impact equipment important to safety (heatup of the spent fuel pool water and heatup of seal injection water which impacts the RCP seals). As such, the TKC System as operated in accordance with the TO procedure will provide KC cooling flow to these heat exchangers while the AB non-essential is isolated. While the temporary KC cooling loop is in use, the Unit 1 KC A train will be removed from service and is considered to be inoperable, and the 72-hour Action Statement for Specification 3.7.3 is entered. No changes to the FSAR are needed. There is no need to revise the Technical Specifications as a result of the TO. No design criteria or safety functions of any structure, system or component is adversely affected by this TO. No USQ exists.

EP/1/A/5000/FR-P.1
EP/1/A/5000/FR-I.1
EP/1/A/5000/ES-1.1
EP/1/A/5000/ECA-2.1
EP/1/A/5000/ECA-0.2
EP/1/A/5000/ECA-0.1
EP/1/A/5000/E-3

EP/1/A/5000/G-1
AP/1/A/5500/35
OP/1/A/6400/05
OP/1/A/6100/10K

As documented in PIP-1-M96-0210, valve 1KC-1A could not be operated electrically, and as a result the valve was closed and power removed. With 1KC-1A closed, the B train of the Component Cooling (KC) System is supplying the cooling water to the various essential and non-essential heat loads. In addition, valves 1KC-3A, 1KC-50A and 1KC-230A will be maintained closed while the B train of the KC System is supplying the cooling water to the various essential and non-essential heat loads. The proposed changes to various procedures will provide instruction to the operators on how to re-align the KC System in the event that the B train of the KC System is lost.

Maintaining 1KC-1A, 50A and 230A closed, ensures that these valves will be in their safety position in the event of an accident. Train separation is maintained through all Modes of operation with these closed. The ability of the KC System to provide cooling water to all components is unaffected by maintaining these valves closed, in that B train can provide the required flow for all heat loads. Upon receipt of a safety injection signal (Ss) or a containment isolation phase B signal(Sp), all components will either go the to their safety related positions or they were already in their safety related positions. The flow to the ND heat exchangers and ND pump mechanical seal cooling water heat exchanger is unaffected by maintaining 1KC-1A closed. No changes to the FSAR are needed. No design criteria or safety functions of any structure, system or component is affected by the proposed changes. No USQ exists.

TN/1/A/2441/00/01I
TN/2/A/2441/00/01I
TN/2/A/2441/00/01E

These procedures provide detailed instructions for controlling the implementation of the Electrical portion of NSM MG-12441/00 and MG-22441/00. The modifications eliminate the automatic function for valves 1(2)VQ1, 1(2)VQ2, 1(2)VQ3, 1(2)VQ5 and 1(2)VQ6, by deleting containment air pressure switches 1VQPS5000 and 1VPQS5010. These instruments have proven unreliable with much maintenance required and little benefit realized. Containment air additions and releases will be done by

manual controls, with containment air pressure alert annunciator lights.

These modifications will not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The modifications will not create the possibility for an accident or malfunction of a different type than any evaluated in the SAR. Manual control of the containment air releases and additions will not adversely affect ESF function (containment isolation) or create any new failure modes or operating characteristics for the plant. The modifications will not reduce the margin of safety as described in the bases for any Technical Specification. Portions of this modification are QA Condition 1, specifically the sparing of instrument tube penetrations and removal of Train-related automatic circuits in area terminal cabinets 1ATC10 and 1ATC11. The pressure switches and associated control relays are non-safety. No USQ exists.

TN/1/A/2456/00/01E

This procedure provides detailed instructions in controlling the implementation of the Electrical portion of NSM MG-12456/00. This modification replaces Train A cables, Appendix R required, that pass through the Unit 1 Train B switchgear room with mineral insulated cables to satisfy concerns expressed by Generic Letter 85-16, Supplement 1. This modification will ensure continued compliance with Appendix R regulatory requirements, without the need to rely on Thermo-Lag as a fire barrier to protect safe shutdown equipment. The functional operation of the components utilizing these circuits will be unchanged. This modification involves no USQ's or safety concerns. No Technical Specification changes are required. No USQ exists.

TT/1/A/2456/00/1E

The purpose of this procedure is to verify that the control circuit for the pressurizer heater group 1A functions as designed following installation of mineral insulated cables added by NSM MG-12456.

This test will not increase the probability of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The test will be performed with the equipment out of service and breakers racked out with control power still on the control circuit. Active

components will be isolated from the portion of the control circuit being tested. The test involves voltage verification only. No device in the circuit will be activated. The test will be performed prior to entering the mode of operation in which the equipment is required. During the performance of this test the redundant train components will be fully functional. The affected equipment will be de-energized and isolated from other plant systems. The test will verify the proper installation of control circuit components. No USQ exists.

TT/1/B/2454/00/1E
TT/2/B/2454/00/1E

These test procedures provide detailed instructions in controlling the Functional Verification of the electrical portions of NSMs MG-12454/00 and MG-22454/00. The test procedures functionally verify the installation and operation of the "Disable/Enable" control switch and modification to the control circuit for valves 1(2)CA161C and 1(2)CA162C. These test procedures allow the operator to isolate the 125VDC and the 250 VDC feeder voltage to the RN/CA cross connect valves 1(2)CA161C and 1(2)CA162C, subsequent to opening the valves. Correct voltages are verified on any 125VDC and 250VDC circuits that were modified. In addition, each valve is cycled independently to the open position and realigned to its normal full closed position.

These test procedures do not increase the probability of an accident or malfunction of equipment important to safety that have been previously evaluated in the SAR. The test procedures will not create the possibility for an accident or malfunction of a different type than any evaluated in the SAR. The test procedures functionally verify the installation and operation of the "Disable/Enable" control switch and modification to the control circuit for valves 1(2)CA161C and 1(2)CA162C and will not adversely impact SSS functions. The test procedures will not result in any new failure modes or operating characteristics for the SSS or will not reduce the margin of safety as described in the bases for any Technical Specification. The test procedures are non-QA and SSS related. The test procedures will require changes to the McGuire Fire Protection Safe Shutdown Review Manual. No USQ exists.

TN/2/B/2454/00/01E

This procedure provides detailed instructions in controlling the implementation of the electrical portion of NSM MG-22454/00. This modification allows the operator to isolate the 125VDC and the 250 VDC feeder voltage to the RN/CA cross connect valves 2CA161C and 2CA162C, subsequent to opening the valves.

This modification will not increase the probability of an accident or malfunction of equipment important to safety that has been previously evaluated in the SAR. This modification will not create the possibility for an accident or malfunction of a different type than any evaluated in the SAR. The installation and operation of the "Disable/Enable" control switch and modification to the control circuit for valves 2CA161C and 2CA162C will not adversely impact SSS functions. This modification will not result in any new failure modes or operating characteristics for the SSS. This modification will not reduce the margin of safety as described in the bases for any Technical Specification. This modification is non-QA and is SSS related. This modification will require changes to the McGuire Fire Protection Safe Shutdown Review Manual. No USQ exists.

MP/2/A/7150/57

The purpose of this procedure is to provide a method of removal and installation of the Unit 2 Reactor Vessel Head and task guidance for a refueling outage. Changes to this procedure incorporate the results of the Dominion Engineering, Inc. design study R-3181-00-2 (DEI-456), Rev.0, "Reactor Vessel Bolting Evaluations - Rotterdam Dockyard Vessels - McGuire Unit 2 and Catawba Unit 1", dated January 1996. The design study was prepared in accordance with the quality assurance program defined in the Dominion Engineering, Inc. "Quality Assurance Manual for Safety Related Nuclear Work", DEI-002, Revision 12, effective September 26, 1995. Dominion Engineering, Inc. is on Duke Power Company's approved vendors list. The contents of the design study are QA-1. The design study is bounded by the original manufacturers design stress report of the McGuire Unit 2 Reactor Vessel and ASME Code.

The technical changes to this procedure impacting stud tensioning and plant operations with a missing or no preload closure stud does not place the NC system pressure boundary at risk or decrease the nuclear safety of the

plant. A change to the FSAR is necessary in section 9.1.4.2.2 in the "Reactor Vessel Stud Tensioner" description. There are no required changes to Technical Specifications. The procedure changes other than those from the Dominion Engineering, Inc. design study do not affect any SSC in a non-conservative manner or nuclear safety of the plant. There are no non-conservative or unanalyzed conditions that impact reactivity management.

The SAR description will be changed to reflect a maintenance technique for tensioning reactor vessel closure studs. The stud tensioning process was changed from a two step to a one step process. As a result, each of the 54 studs receives an application of hydraulic pressure to achieve the desired preload for plant operations. No USQ exists.

PT/1/A/4150/044

The purpose of this procedure is to provide guidance for placing the feedwater flow differential pressure (DP) test gauges in service, for taking data, for performing mass flow calculations, and for removing the DP test gauges from service. The DP test gauges measure the pressure drop across the high pressure taps and the pipe wall taps as well as the pressure drop across the high pressure taps and the throat taps. To perform the test, the unit is required to be in mode 1, greater than 50% power level. The intent of the test is to provide data so that a comparison can be made between the use of the wall taps instead of the throat taps as the low pressure tap for measuring final feedwater flow.

The proposed procedure will not result in any USQ or licensing issues. No changes to the Technical Specification will be required. No design bases or safety functions of any structure, system or component is adversely affected by this procedure. No FSAR revisions are required. No USQ exists.

MP/0/B/7150/121

Ice accumulates on the ice condenser wall panels during the units normal operating cycle. This is caused by the natural sublimation of ice from the ice baskets. The wall panels being a colder surface in the ice condenser attracts the vapor and forms a frost on the wall panel surface. This frost will accumulate until the ice baskets are frozen in place making ice weighing difficult and increasing the

amount of outage maintenance. If left unattended the ice would migrate into the ice bed and cause flow blockage.

This procedure gives direction for defrosting the ice condenser wall panels. It is based on the Westinghouse procedure found on MCM-1201.17-853. Administrative controls in the procedure will not allow its use during Modes 1, 2, 3 or 4 and monitors the progress of the procedure to ensure the ice bed is protected.

The wall panel defrost is discussed in the FSAR in section 6.2.2.1.1 and describes the process in some detail. The FSAR also describes a floor defrost to be implemented in conjunction with wall panel defrost. The floor defrost is not incorporated in the procedure due to concerns of water entrapment in the ice condenser floor which during cooldown would freeze causing spalling of the ice condenser floor. Omitting this section is considered a change of maintenance practice and does not increase any safety concerns in the ice condenser.

Wall panel defrost procedure is not evaluated in the SAR in any accident analysis and therefore can not increase the probability or consequences of an accident previously evaluated or the possibility for a malfunction of a different type.

The wall panel defrost procedure is not evaluated in the SAR in any accident analysis and therefore can not increase the probability or consequences of an accident previously evaluated.

The wall panel defrost procedure is completed while the unit is in mode 5, 6 or no mode. Ice condenser accident analyses are evaluated while the unit is in modes 1 through 4. Ice condenser equipment is returned to formal service prior to mode 4 and does not create the possibility for an accident of a different type.

The wall panel defrost procedure utilizes non-safety equipment during its process and is restored to normal service prior to mode 4. This procedure does not increase the probability or consequences of a malfunction of equipment important to safety evaluated in the SAR.

This procedure does not reduce the margin of safety as defined in any Technical Specification. No USQ exists.

MP/0/A/7650/72

The purpose of this procedure is to provide a method of testing check valves using air. The purpose of the test is to find any problems with the alignment of the disk to the valve body seat, and any defects in the seating surfaces. The process involves removing the bonnet, installing and inflating an air bladder downstream of the valve, securing the bladder to a structural component in the valve with a lanyard, reinstalling the bonnet, or a test bonnet, pressurizing the space between the bladder and the valve through a fitting in the bonnet, and observing the pressure decrease in the test space, over time. The bladder will be pressurized to about 25 psi, which will have no effect on the integrity of the piping, and the test space will be pressurized to an even lower pressure, to insure the bladder does not move in the pipe. Following the test, the bladder and lanyard are removed and the valve reassembled.

This procedure involves a maintenance activity on equipment which has been removed from service. It will have no effect on operating system, and will not alter the equipment being tested. There are no equipment alterations by this testing and systems are returned to the design configuration. No new accident possibilities are created, there is no increase in the probability of a malfunction of equipment of a different type than any evaluated in the SAR. No new accident will be created and no safety margins are affected. No Technical Specification or FSAR changes are required. No USQ exists.

OP/0/B/6200/109

This is a new procedure to provide instruction on operation of the makeup water polishing demineralizers that are being temporarily installed in the Demineralized Water (YM) System. The polishing demineralizers are of fiberglass design and are being provided from Ecolochem, Inc. The YM system provides high purity demineralized water to the station primary and secondary systems.

The YM system is considered as an accident initiator only in that it can be used to dilute boron concentrations and add reactivity to the reactor core in an uncontrolled manner. This modification is not associated with any part of systems or components which prevent or control this type

of reactivity addition. The YM system is not safety related and is not required in any analyzed accident mitigation. The power supply and water input to the system are not safety related and do not degrade the ability to supply water to any system or component important to safety. Therefore, the probability of an evaluated accident is not increased.

The added demineralizers and their operation do not affect any components or systems used to mitigate an accident. The YM system function and operation are not changed outside of use of the demineralizers and testing of the results of the additional processing. These activities have no impact on any analyzed accident consequences.

This equipment and procedure will be implemented on a temporary basis to evaluate the benefits of additional demineralizers on the YM system. The procedure contains sufficient controls and checks to ensure that the addition and use of the demineralizers on a temporary basis will not degrade the operation of the YM system or introduce water chemistry changes that may be detrimental to the operation of any connecting plant system. The overall function and operation of the YM system will not be changed. The YM system is considered as an accident initiator only in that it can be used to dilute boron concentrations and add reactivity to the reactor core in an uncontrolled manner. This modification will not impact any existing provisions to prevent uncontrolled dilution.

The procedure contains sufficient controls and checks to ensure that the addition and use of the demineralizers on a temporary basis will not degrade the operation of the YM system or introduce water chemistry changes that may be detrimental to the operation of any connected plant system. The YM system and components are not considered important to safety since they are not needed to mitigate any accident and are not accident initiators.

The additional demineralizers do not impact any components or systems which mitigate the consequences of an accident. No functions important to safety are provided by the added components. The changes in water chemistry will be monitored to ensure that water quality is acceptable at all times.

A malfunction of the added components would not be different than the malfunction of the current demineralizers. The added components may be bypassed

without degrading YM system performance or degrading water quality below currently acceptable standards.

No fission product barriers are impacted by this modification. No Technical Specification or FSAR changes are needed. No USQ exists.

OP/1/A/6200/04A
OP/2/A/6200/04A

These procedures align the Residual Heat Removal System (ND) to the Refueling Water Storage Tank (FWST) using the Containment Spray (NS) system recirculation line. The purpose of these procedures is to reduce the radiation levels in the ND system piping by flushing the lines to the FWST. The FWST will be placed in purification mode to keep the FWST clean. The alignment used in this procedure will result in one train of ND and the same train of NS being inoperable for a short period of time. The ND train will be inoperable due to the suction isolation valve being closed. The NS train will be inoperable during the procedure due to the manual alignment of NS to the FWST, and the closure of the suction valve. The ND and NS pump motors will be racked to disconnect to prevent pump damage in the event of an auto start signal during the procedure. A dedicated operator will be available to manually close the NS suction isolation valve from the ECCS sump in the event of a safety injection signal and a failure of remote valve control, to prevent the transfer of sump water to the FWST.

The ND and NS systems are not accident initiators in any Chapter 15 accidents. However, they do act as accident mitigators in the LOCA accidents evaluated in Chapter 15 of the FSAR. Time is considered to be adequate for system restoration for both ND and NS trains to mitigate the small break LOCA, where ND is not required until sump recirculation modes for NI and NV pump suction boost. One train of the ND and NS systems will be declared inoperable during execution of this procedure. This is allowed by the Technical Specifications for short durations (72 hour Action Statement). The procedure alignment will not degrade in any way the effectiveness of the operable train on either system. Therefore, there is no increase in the probability or consequences of an accident evaluated in the FSAR.

The procedures alignment requires that the train being used for FWST recirculation be declared inoperable and does not

create an alignment which would make the other ECCS train inoperable. Operator action is specified to isolate the NS recirculation path to the FWST by closing 2NS-1B or 2NS-18A in the event of a reactor trip or Ss signal. This can be accomplished from the control room. A dedicated operator is assigned for compensatory actions in the event that local, manual operation is necessary. These valves will isolate NS from the ECCS system and prevent transfer of water from the sump to the FWST. No other immediate actions would be required. 2ND-68A or 2ND-67B, and two manual valves can also be used to isolate the flow path to the FWST and are specified in the restoration steps. No new single failure would prevent isolation of the FWST prior to sump recirculation.

The NS pumps will receive a small amount of forced flow from the ND pumps which may cause forward direction rotation of the prime element. This has been reviewed by Ingersoll-Dress Pumps Engineering Department and found to be acceptable for the pump based on the seal design and configuration.

The opening of the ND system heat exchanger mini-flow valve will not make the opposite train inoperable, since the flow balance injection test was performed with the opposite train mini-flow valve failed open. No new failures are created as a result of this alignment since the system is not an accident initiator, and the sequence of active valve positionings would not result in a new accident flowpath. Therefore, the proposed procedure will not create any new accidents or malfunctions of a different type than any evaluated in the FSAR.

Since the ND and NS pump motors will be racked to disconnect, no damage to the inoperable pumps will occur while the suction isolation valves are closed. The ND and NS train being used to recirculate water to the FWST will be declared inoperable by this procedure, and the ACTION statement will be entered. This condition (loss of one train) is analyzed in the FSAR for the accidents using the ND and NS systems. No single failure assumptions are required to be considered while in the action statement for the Technical Specification, and thus the consequences are not worse than analyzed. This evolution is equivalent to performing maintenance on a single train of ND and NS while both ECCS subsystems are required per Technical Specifications. Therefore, the evolution, does not increase the probability or consequences of a malfunction of equipment important to safety evaluated in the FSAR.

The margin of safety as defined in the Technical Specifications is related to the confidence in the fission product barriers. Operator action is taken credit for in isolating the recirculation path from the ND and NS systems, which will prevent transfer of water from the ECCS sump to the FWST. This is justified by the procedural cautions and limitations placed on this additional operator. A single train of ND and NS is maintained operable throughout the evolution of this procedure, as analyzed in the FSAR. Therefore, the evolution does not reduce the margin of safety as defined in the Technical Specifications. Single failure of neither ND or NS is considered for the brief period that the systems are aligned for the flush. No USQ exists.

PT/1/A/4150/01A
PT/2/A/4150/01A

This purpose of this procedure is to monitor the NC System for system leakage after it has been closed following a refueling outage and/or maintenance which involved opening the primary system. This procedure change will delete the requirement to pressurize the NC system to between 2350 and 2400 psig while performing the NC system leak test. The original procedure requirements were taken from the Westinghouse Reactor Operating Instructions, section M-2 dated June 15, 1975.

In a letter dated March 27, 1995 the NRC allowed for use of ASME Code Case N-416-1. In their letter of approval the NRC stated that "the staff concludes that compliance with the Code hydrostatic testing requirements for welded repairs or replacements of Code Class 1, 2, and 3 components would result in hardships without a compensating increase in the level of quality and safety." In granting the use of ASME Code Case N-416-1, the need to overpressurize the NC system is no longer required. The ASME Code only requires that the NC system be brought up to nominal operating temperature and pressure.

This procedure revision will not increase the probability of an accident evaluated in the SAR. This revision will delete the need to overpressurize the NC System.

This revision will not increase the consequences of an accident evaluated in the SAR. Deletion of the requirement to overpressurize the NC System to perform the system leak test has been determined not to be justified. Any leakage

present in the system will be found with the system at normal operating pressure and temperature.

This revision will not create the possibility for an accident of a different type than any evaluated in the SAR. Since the leak test will be conducted at normal operating pressure and temperature, no new operating configurations will be introduced.

This revision will not reduce the margin of safety as defined in the basis for any Technical Specification. No USQ exists.