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Docket Nos. 50-369, 370

Attached is a summary report per 10 CFR 50.59 (b)(2) of Nuclear Station Modifications, Minor Modifications, procedure changes and other miscellaneous changes made at McGuire Nuclear Station under 10 CFR 50.59 for this reporting period. No Unreviewed Safety Questions were identified.

Questions regarding this submittal should be directed to Kay Crane, McGuire Regulatory Compliance at (704) 875-4306.

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Attachment

IE47

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Minor Modifications

MGTM-0158

This modification reduced the available indicated range of the Unit 1 B train wide range containment sump level (NILP5270) from 0-20 feet to 0-17 feet. The modification bypasses the level transmitter (1NILT5274) which monitors the 17-20 ft region of the 0-20 ft span. Procedure IP/0/A/3009/004, Containment Level Calibration, was revised due to this modification. The procedure change eliminated the requirement to calibrate the Unit 1, train B, wide range containment sump level in the 17 to 20 feet region.

In accordance with NUREG-0737, Item II.F.1(5), the containment water level monitor is to provide a continuous indication of containment water level in the control room. A narrow range instrument is to be provided to cover the range from the bottom to the top of the containment sump. A wide range instrument is to be provided to cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. The modified operating range of 0-17 feet provided by the B train instrument would represent a span from Elevation 725 feet (bottom of containment) to Elevation 742 feet (>900,000 gallons). As such, the modified B train instrument will still comply with the NUREG-0737 range requirement for this instrument. The modification on the B train loop is independent of A train indication; thus, failure of the temporary modification will not affect A train indication. Neither loop is associated with any automatic controls. The calculation for the instrument uncertainty for this instrument is not affected by this modification. Accordingly, the B train instrument, as modified, will continue to meet all of the performance, design and qualification criteria as specified in Regulatory Guide 1.97 for this instrument. No USQ exists.

MGMM-10654

This modification involves an update to the defensive strategy for McGuire Nuclear Station to install the following items in the Unit 1 Turbine Building: 1) Four portable prefabricated bullet resistant defense positions, 2) Six prefabricated bullet resistant gun ports defensive positions in walls, 3) One security station (prefab building) on elevation 786+0 near column lines 1B/20-21. The Turbine Building contains the turbine generator and associated equipment. The Turbine Building is non-QA and is not designed for tornado or seismic loads. The building qualification is not changed by this modification. No equipment important to safety is compromised by this modification. No fire protection concerns were identified within the Turbine Building. No USQ exists.

MGMM-11869
MGMM-10091

These modifications allow use of varied size filtration media for the reactor coolant pump seal injection and standby make-up pump filters, provided they do not exceed the nominal 5 micron original design. The modifications also provide for use of a non-IEM filter cartridge, which is suitable in form, fit and function.

The Chemical and Volume Control (NV) System provides the following normal operational functions: 1) maintains programmed pressurizer level and provides for reactor coolant pump seal injection, 2) maintains the reactor coolant system chemistry, activity and soluble chemical neutron absorber concentration and make-up. The NV system also supports hi-head safety injection and recirculation as part of the Emergency Core Cooling (ECCS) System. The hi-head safety injection function is credited for mitigation of the spectrum of break sizes for LOCA and also for main steam line break. The reactor

coolant pumps are non-safety related and thus are not credited for mitigation of design bases accidents. The standby make-up pump is non-safety related and is utilized as a back-up make-up/seal-injection source for various fire and sabotage events.

The original reactor coolant pump and standby make-up pump seal injection filter design provided for "nominal" 5 micron filtration. This activity allows for use of varied micron media less than "nominal" 5 micron. Allowing use of smaller micron media will serve to enhance reactor coolant pump seal life, and will further provide a secondary benefit by reducing source term and personnel dose. Finer micron filtration will more efficiently remove iron oxide and cobalt elements. In addition to allowing varied micron options, this activity allows for use of a non-OEM filter, which is comparable in form, fit and function, and is considered to be a like-for-like replacement.

Use of small micron filtration will reduce the service life of the filter because it is changed-out based on differential pressure. Small micron filtration will also result in higher dp when initially placed in service. Providing options for varied micron sized media will increase operational flexibility for varying system conditions.

This activity does not degrade any structure, system or component and is beneficial to reactor coolant pump seal health, such that the probability or consequences or any previously analyzed event (accident or equipment malfunction) is not increased. No USQ exists.

MGMM-10794

The non-safety related Ice Condenser (NF) System chillers and condensing units currently use R-502 as the working refrigerant. R-502 was banned from production in 1995 by the Environmental Protection Agency (EPA) due to its detrimental affect on ozone. A McGuire Engineering Evaluation Team recommended that all NF system chillers be retrofitted with an EPA approved refrigerant (R-507). This modification performs the retrofit which also includes replacing the chiller lubricant and filter-dryer cores.

UFSAR Section 6.2.2.6.1 states the following design basis: "The Refrigeration System serves to cool down the ice condenser from ambient conditions of the reactor containment and to maintain the desired equilibrium temperature in the ice compartment. It also provides the coolant supply for the ice machines during ice loading. The Refrigeration System additionally includes a defrost capability for critical surfaces within the ice compartment.

During a postulated loss-of-coolant accident the Refrigeration System is not required to provide any heat removal function. However, the Refrigeration System components which are physically located within the Containment must be structurally secured (not become missiles) and the component materials must be compatible with the post-LOCA environment."

The equipment modified by this modification is located in the Auxiliary Building. The equipment does not perform a nuclear safety-related function, cannot increase the consequences of an accident, have any affect on accident initiation, nor cause a malfunction of such equipment evaluated in the SAR. Chiller performance will be not affected and proper system operation will be verified after completion of the retrofit. No USQ exists.

MGMM-10796

This minor modification addresses a problem of not fully loading the non-safety related Ice Condenser Refrigerator System (NF) chillers. The amount of load on a chiller is not critical to its normal operation; however, not fully loading the chillers is detrimental to the long-term reliability of the equipment. This modification will increase the size of the glycol pump impellers to achieve about 13% greater flow. Each motor will be upgraded from 25 HP to 30 HP to support the additional load. The 150 psig relief valves (1NF42 and 1NF94) will be replaced with new valves set for 180 psig. An editorial change will also be made to MCFD-1558-01.01 to delete piping downstream of 1NF3. This piping does not exist in the field. All equipment mentioned above is non-safety related class G.

The UFSAR states the following design basis: "The Refrigeration System serves to cool down the ice condenser from ambient conditions of the reactor containment and to maintain the desired equilibrium temperature in the ice compartment. It also provides the coolant supply for the ice machines during ice loading. The Refrigeration System additionally includes a defrost capability for critical surfaces within the ice compartment.

During a postulated loss-of-coolant accident the Refrigeration System is not required to provide any heat removal function. However, the Refrigeration System components which are physically located within the Containment must be structurally secured (not become missiles) and the component materials must be compatible with the post-LOCA environment."

The pumps, motors and relief valves being modified are located in the Auxiliary Building and do not perform a safety function. The new glycol pump impeller is a standard replacement part from the manufacturer with defined performance data. The motor upgrade will provide longer service life and support the improved pump performance. The glycol header relief valves protect the system from pressure increase due to glycol thermal growth during periods when the Auxiliary Building glycol header is isolated. They also provide pump protection for short duration operation under a dead headed condition that could occur from component failure or misalignment. The relief valve setting increase is necessary for normal glycol system operation because at shutoff the pump head would be approximately 153 psig (Unit 2). The 180 psig setting provides margin to prevent unnecessary lifting of the relief valves. Components within the glycol loop are capable of handling the incremental pressure increase. Lifting of the relief valves would not affect emergency operation of any equipment described in the SAR.

The NF system Auxiliary Building refrigeration components to be modified are not nuclear safety-related and cannot have any affect on accident initiation. Accident probability will not be increased. These changes will improve component reliability. None of the components being modified can affect equipment important to safety, increase the consequences of an accident nor initiate an accident. The changes have no affect on operation of the safety-related portion of the NF system. No USQ exists.

MGMM-11018

This modification regrades portions of the cask transporter haul route. This is the planned path for transport of spent fuel dry storage canisters from the fuel buildings to the storage pads. This minor modification regrades portions of the designated haul path to ensure that a loaded cask transporter can safely transverse this area. Storm drain system catch basins CB-190 and CB-181 will be affected along with the valve box VB-4 which houses unit 2 valves ZP-75, ZP-78, ZP-82 and ZP-89.

The design commitment for the storm drain system to handle 4 inches of rain per hour and the requirement for the probable maximum precipitation will not endanger any safety related structures,

systems or components is maintained. The design details which support these commitments will be provided in more general terms to allow minor variations in the yard layout which do not impact the design requirements to exist without creating a deviation from the SAR. The changes made by this modification do not significantly impact the station design for normal watershed or response to design basis conditions. No safety related structures, systems or components are modified or impacted by the modifications to yard grading, or the 6 inch change in elevation of catch basins 190 and 191. The repair of valve box 4 is a maintenance activity and does not impact the design or function of the non-safety structure. No USQ exists.

MGMM-10655

This modification involves an upgrade of McGuire's current defensive strategy. The following items are to be installed in the Unit 2 Turbine Building:

1) Four portable prefabricated bullet resistant defense positions
2) Six prefabricated bullet resistant gun ports defensive positions in walls
3) One security station (prefab building) on El. 786+0 near column lines 2B/20-21.

The Turbine Building contains the turbine generator and associated equipment. The Turbine Building is non-QA and is not designed for tornado or seismic loads. The building qualification is not changed by this modification. No equipment which is important to safety is compromised by this modification. A fire protection review identified no concerns in the Turbine Building. No USQ exists.

MG-10897

This corrective minor modification deletes four non-safety related, non QA Condition 1, Recirculation Cooling Water (KR) flow switches. The original manufacturer design option allowed the installation of a condenser cooling water flow permissive in the chiller compressor start circuit. A flow switch, DP switch or condenser pump starter auxiliary contract could also be provided to perform the permissive function. This function is not required for safe operation of the chillers rather, it is an option that is redundant to the manufacturer provided condenser refrigerant high pressure (HP) switch. HP will trip the chiller upon loss of condenser cooling water flow since the condenser refrigerant pressure will increase. The redundant flow switches were piped into the KR flow line and added to the flow drawings but were never electrically tied to the chiller control circuit. The SAR doesn't discuss the function of or a design requirement for these flow switches. Rather, the equipment identification was added to the summary flow diagrams only because the flow switches are physically connected to process and not because they performed a desired design function. Thus, by removing this unused component from the process, this change can not increase the probability of a malfunction that does not already exist. The change will improve the plant by: 1) Eliminating components not required for continued plant operation; and 2) Preventing a potential foreign material event that could occur with a failure of these components internal parts intruding into the process because they are not maintained by the plant. This modification does not affect the safety or the operation of the plant. No USQ exists.

MGMM-11127

This modification addresses a problem of high flow induced vibration in the piping downstream of the Chemical and Volume Control (NV) System 75 gpm letdown flow orifice. Service time is limited on the fixed orifices requiring NDE inspection after only 9 months of operation. The current operational practice is to rely primarily on the variable letdown control valve which limits operational flexibility and system reliability. The resolution is to perform this modification which will replace the existing 75 gpm fixed orifice with a custom designed DRAG® trim design valve from Control Components Incorporated

(CCI). The flow orifice will be cutout and the new valve butt-welded to the reducing tee. The new valve is designed to provide a trim which can accurately control letdown flow at low flow rates (less than 10 gpm) and provide cavitation control.

UFSAR Section 9.3.4.2 describes the letdown function of the 75 gpm orifice to provide a large pressure reduction to the reactor coolant prior to it entering the mixed bed demineralizers. The resulting flow from the orifice is 75 gpm to the tube side of the letdown heat exchanger. A second pressure reduction is performed by the low pressure letdown valve (to prevent flashing downstream of the flow orifice).

Two letdown orifices (75 gpm and 45 gpm) are arranged in parallel. The 75 gpm orifice is designed to pass normal flow. Also, in parallel with the flow orifice is a third letdown path via control valve 2NV459. Any combination of flow paths can be utilized in order to control letdown flow such as during reactor heatup and maximum purification operations. The downstream relief capacity provided by 2NV6 is adequate since it can accommodate the maximum flow through all three flow paths.

The internals of 2NV459 have been previously modified to accept a CCI DRAG® trim kit. This valve has performed very well since this modification with downstream piping vibration greatly reduced. The new custom designed CCI valve (2NV454) is a Y-globe design with a trim kit very similar to that in 2NV459. The similar trim kits of the two valves give a high level of confidence that the installation of 2NV454 will also greatly reduce downstream piping vibrations. Another usable letdown flowpath provided by the new valve will give flexibility of operation and improve long-term reliability.

2NV454 meets the design requirement for its service condition. It has a stuffing box containing two sets of packing and an intermediate leakoff connection. The basic material of construction is stainless steel which handles radioactive liquid and boric acid solutions. The CCI valve will be procured per MCS-1205.06-00-0011. It is a manual throttle valve designed to meet the stringent requirements for temperature, pressure and flow for primary system application.

Valve 2NV454 will provide manual variable flow control. This valve is safety-related from the standpoint of pressure boundary but is not an active valve. The containment isolation function is provided by 2NV458. No credit it taken in the McGuire SAR for operation of the 75 gpm orifice and none will be taken for operation of 2NV454.

Based on the design requirements for primary system valves, procedural and QA controls on valve installation and post-modification testing; accident probability will not be increased. The new valve is not considered important to safety except as a reactor coolant pressure boundary. This modification will improve the pressure boundary reliability of this letdown path by reducing piping vibration. This will not increase the probability of an equipment malfunction. The new valve is a direct replacement of the orifice in the system with the same piping design requirements. This does not increase the consequences of an accident relative to the original flow orifice. The new valve has no function that could affect equipment important to safety. The capability to isolate the 2NV454 letdown line is available to the control room by closing the downstream isolation valve should any flow anomaly occur. A malfunction of equipment important to safety cannot occur. There is no function or potential failure of these valves that could lead to an accident different than previously evaluated and no reduction in safety margin. No USQ exists.

MG-11184

This modification incorporates changes to plant documents, which enhance the fire protection program for configuration control of the fire barrier penetration seals, as follows:

- Existing penetration seal specifications MCS 1206.03-01-0001 and MCS 1390.01-00-0056 will be superceded by a new corporate specification DPS 1435.00-00-0002,
- A comprehensive corporate design basis calculation DPC 1435.00-00-0006 is available to provide penetration seal configuration qualification,
- All known penetration seals will be added as certified to the McGuire Equipment Database to enhance the record retention of repairs and configuration control. Implementing each specific penetration seal into the process will ensure that each time a penetration seal is either modified or repaired that a recorded entry be made into a plant retrievable database, thus configuration management is ensured,
- The revised penetration seal configuration will be issued on approved plant drawings to enhance the ability of plant personnel to retrieve this information.

The enhancements mentioned above, in the activity description section, are to improve the documentation and traceability of the penetration seal program documents. This modification also adds each NRC committed fire barrier numbered penetration seal to the Equipment Database Program to ensure a tracking mechanism of configuration control.

The McGuire station is required to provide penetration seals in fire, flood and pressure (HVAC) boundaries. No special penetration seal designs are installed to provide radiation-shielding protection. The specific licensing acceptance criteria for fire barrier penetration seals is:

Electrical Penetration Seals:

These designs are subjected to a standard exposure fire that follows the ASTM E-119 Standard Time-Temperature Curve.

The acceptance criteria from IEEE 634 (which is identical to that of IEEE P634/D4) shall be used to evaluate the acceptability of test assemblies. The IEEE 634 acceptance limit for unexposed side electrical penetration temperature is 700°F. This temperature was selected because the IEEE-634 acceptance criteria is the auto-ignition temperature of the outer cable covering and material in contact with the cable penetration fire stop, or 700°F, whichever is lower. The ignition temperature of outer cable jacketing has been determined to be 735°F.

Mechanical Penetration Seals:

Mechanical penetration seal designs shall be subjected to a standard exposure fire that follows the ASTM E-119 Standard Time-Temperature Curve.

The acceptance criteria from IEEE-634 shall be used to evaluate the acceptability of test assemblies. The IEEE-634 acceptance limit for Duke Power mechanical penetration seals is 680°F. This temperature was selected because if represents the lowest auto ignition temperature of any material normally found in close proximity to piping penetrations.

The enhancements being made by this modification clarify the McGuire fire barrier penetration seal license requirements. McGuire assures the license requirements by establishing its fire barrier penetration seals acceptance criteria to configurations which have been successfully furnace and hose stream tested or by documenting the specific configuration with an equivalency evaluation in format based upon NRC Generic Letter 86-10.

There are no safety concerns with the implementation of this modification. The engineering design and licensing requirements are not changed; only their implementation to an as-built construction is verified and documented. No USQ exists.

MGMM-11652 and VN-11652A

This activity changes the setpoints of various pressure switches and allows for proper operation of loops 0YLMP5330 and 0YMLP5340. This modification also adds test tees and plugs to the process lines of 0YMLT5330 and 0YMLT5340 to allow for ease of calibration.

The treated water systems are shared systems that perform no safety functions and are not Technical Specification related. The function of the filtered and demineralized water systems is to remove suspended solids from lake water to produce water of acceptable quality for plant use. Water from the conventional low pressure service water system is treated with chlorine and poly-electrolyte within the reaction tank before being sent out to the various systems, which include filters. The function of loops 0YMLP5330 and 0YMLP5340 is to monitor the level within the vacuum de-aerators. The various instrumentation give associated alarms depending on the level within the de-aerator tanks and provide protection against low-low levels.

The current setpoints of the associated instrumentation to loops 0YMLP5330 and 0YMLP5340 are not proper setpoints. The preferred method of operation is intended to give operators or technicians adequate time to respond to alarms before actuation of the pump trips in which this current configuration does not provide.

Shared Indicator 0YMP5330 is being removed from service as it provides no useable information for the process control and is being abandoned.

A variation to this activity is being performed to provide an editorial update to all documents affected by the original modification and change actuators to valves 1YM0172 and 1YM0156 to be failed closed. Valves 1YM0172 and 1YM0156 are being changed to fail closed actuators to reflect original design intent.

The abandonment of the instrument has no impact on any equipment important to plant safety nor is there any reduction in safety margins. The remaining activities of the minor modification and VN correct known instrumentation control problems and change two valve actuators to original design intent. None of this work as any negative impact on any safety related equipment. No USQ exists.

MGMM-11820

This modification changes the failed position of 1CM420 (Condensate System) on MCFD-1590-01.05 and UFFSAR Figure 10-42) from "Fail Open" to "Fail Closed". This is consistent with vendor information, field setup and that of the corresponding Unit 2 valve. This modification does not perform a physical change to the plant and is editorial in nature.

Valve 1CM420 is the load rejection bypass valve which is to open on a load rejection signal to provide adequate flow from the Hotwell Pumps to the Condensate Booster Pumps. This flow ensures that the required flow is provided to the steam generators to compensate for the level shrinking in the generators. This is a not a nuclear safety related function.

Accident probability will not be increased by this drawing correction. This modification cannot create the possibility of an accident, increase the consequences of an accident or cause a malfunction of safety-related equipment. This is an editorial change. No USQ exists.

Nuclear Station Modifications

NSM-12495
NSM-22495

These modifications deleted the low steam line pressure signal input into Safety Injection (SI) actuation circuitry internal to the Solid State Protection System (SSPS). The low steam line pressure signals are generated in the 7300 Process Control System. The signals are then sent to both A and B trains of the SSPS. The signals are sent from there to four 2/3 Universal Logic Boards in the logic bay of the SSPS. The output of these cards generates main steam isolation and safety signals. These modifications removed the wiring between the logic cards and the SI/Reactor trip circuits. This deletes SI on low steam line pressure. The steam line isolation signal will remain. Control board labels were modified to remove reference to this safety injection function also. Approved Technical Specification amendment 182/164 addressed changes related to these modifications. No USQ exists.

NSM-12504
NSM-22504

These modifications replaced the Unit 1 and Unit 2 train B diesel generator load sequencer timers. The Agastat pneumatic and Cutler-Hammer D87 on-delay timers needed to be replaced because of aging and degrading timing tolerances. The replacement timers are NST Tempo 812 series solid state timers which have been tested and demonstrate better operating characteristics. The replacement timers are qualified QA-1 and located in the same cabinets as the existing timers.

The replacement timers perform the same function as the currently installed timers. Heat loads have been reviewed to assure that these components will be in an appropriate environment and will not adversely impact other equipment in the cabinets. Mounting and seismic qualification of the replacement timers has been performed to assure that all QA requirements are satisfied. All changes are contained within the electrical cabinets such that no new Appendix R concerns are generated. The modifications were implemented on a train basis by procedure. Each affected train is covered under an independent but related nuclear station modification. The modifications are implemented by implementing procedures which control the work and are reviewed independent of this evaluation. The load sequence as shown in UFSAR Figure 8-1 will continue to be met as before. The replacement timers are expected to be more consistent in performance than the current components. The replacement timers are solid state electronic components whereas the old timers are electro-mechanical devices. The electronic devices have demonstrated greater reliability and consistency in performance. They are not subject to mechanical failure as are the currently installed timers. Both the old and new timers are subject to electrical failure. No USQ exists.

NSM-12507

This modification replaces the protective relaying for the Unit 1 230 kv buslines that run from the station to the McGuire switchyard. All four channels of line sensing and ground fault detection relays will be replaced. The modification also replaced the communications portion of the buslines which involves both supervisory and transfer trip functions. This modification is being implemented to replace and update old equipment which is near the end of its service life and difficult to maintain.

The modification does not change the function of the busline protective relaying system. The equipment is updated and will operate faster than the current system to isolate the station from the busline in the event of a fault on the busline but will not change the criteria for separation from the busline.

The Offsite Power System is the preferred source of power for station operation but is not safety related. Loss of Offsite Power is considered a design basis event and the station is designed and operated such that the effects of this event can be safely mitigated. The replacement of the busline protective relays will not increase the likelihood of loss of offsite power or cause the station to remain connected to the offsite grid when abnormal conditions justify separation from the system. The criteria for operation of the busline protective relays will remain unchanged. The new equipment will increase the reliability and speed of the system in responding to problems and will be easier to maintain due to the ready availability of parts. The replacement components meet the criteria of the Maintenance Rule in that they increase reliability and reduce the risk of maintenance preventable failure. No USQ exists.

NSM-12515
NSM-22515

These modifications replaced the discharge check valves for each Auxiliary Feedwater System (CA) pump with an Automatic Recirculation Control (ARC) valve. These valves provide an assured minimum flow path for the CA pumps. The ARC valve operates as a combined check valve and glove valve with the globe valve port providing an assured flow path to prevent deadhead conditions on the pump. The check valve port is the main discharge flow path and is operated in the same manner as the previous check valve. The ARC valve is designed such that the glove port closes as the check port opens assuring a continuous discharge path for the pump and automatically eliminating recirculation flow as sufficient normal discharge flow is established. The installation of this recirculation flow path will result in removal from service of the current recirculation flow paths for the CA pumps. This piping will now be used only for pump flow testing. The current recirculation valves will have automatic controls removed and will only be operated manually.

In addition, a continuous discharge flowpath will be added between each ARC valve and the feedwater pressure boundary valves to assure that this piping is not pressurized due to valve leakage. These flowpaths will have a flow control valve in line to limit flow to 5 gpm at 1700 psid.

Post modification testing will be performed on the revised system to ensure correct flow balance and operation of the discharge and recirculation flowpaths. The operation of automatic functions on a system start signal will be verified using a simulated signal. This testing ensures that the revised system will respond as designed. Operation of all revised components will be tested and data collected to allow future determination of system health.

These modifications improve the reliability of the recirculation function for the CA pumps. The current arrangement is subject to failure due to a loss of valve control instrument power. The revised flowpath is controlled by mechanical action of the valve such that either the normal discharge flowpath or the recirculation flowpath is always available. The ARC valves are considered as reliable as the current discharge check valves and recirculation control valves for valve malfunction in other respects. The use of the ARC valves along with the provision of a flow path to prevent pressurization due to valve leakage maintains all the design operation functions being provided by the current system design. The diversion of 5 gpm of CA supply to the feedwater system during operation does not significantly impact the ability of the CA system to meet its design requirements. No new functions are added by the installation and use of the ARC valves. The testing of the systems will utilize the old recirculation flow path since the

automatic action the ARC valve prevents it being used for testing. The ARC valves and new recirculation meet the same QA standards as the associated piping and components. No USQ exists.

NSM-12517

This modification replaced the upper internal guide tube support pins, which are known as split pins. The currently installed split pins are fabricated from Inconel X-750 Revision A and each pin has the appearance of a "clothes pin." A split pin has two separate legs that have an interference fit with a hole in the upper core plate. The split pins are attached to the bottom flange of a guide tube assembly via its shank, nut and locking device. The pins are captured with a washer welded to the bottom of the guide tube support. The material for the replacement split pin is SA-316CW which is qualified and approved for this application. The modifications to the upper internals include the following activities: 1) replace all split pins with cold worked type 316 stainless steel material, 2) remove all guide tube/flow column hold down bolts and install approximately 50% of the hold down bolts fabricated from cold worked type 316 stainless steel material, except for the guide tubes adjacent to the outlet nozzles. The guide tubes located adjacent to the outlet nozzles will have 8 replacement bolts installed, 3) Remove all 15X15 guide tube flexures and removable inserts and replace them with mechanical plugs and 4) Remove all orifice plates from 17X17 plutonium recycle guide tubes. An evaluation was performed (WCAP-15252, "Duke Energy Corporation McGuire and Catawba Units 1 and 2 Replacement CW316 Support Pin Design Qualification Report") to document the acceptability of the replacement CW316 support pin material and pin design. The engineering evaluation documents the acceptability of the pin and nut assembly material and assembly. In so doing, the evaluation documents the functional equivalency of the associated replacement hardware such as the guide tube hold down cap screws, guide tube cover plate hardware (which are essentially like-in-kind replacements), and the contingent tandem locking caps. No USQ exists.

NSM-12518/P2

This modification installed a non-safety related, elevated, Auxiliary Feedwater Storage Tank (CAST) and isolated all other sources. The location of the tank is near the southwest corner of the Turbine Building. The new tank is a multi-legged elevated storage tank that is vented to the atmosphere. The tank design is a torospherical type and meets the AWWA Standard for Welded Steel Tanks for Water Storage, ANSI/AWWA D100-96. The new CAST will be aligned to the Auxiliary Feedwater (CA) pumps at all times. It will be used as the primary suction source. The Upper Storage Tank (UST) and the Auxiliary Feedwater Condensate Storage Tank (CACST) will remain tied to the CA system but normally isolated from the CA pumps. The recirculation flow from the CA pumps will be returned to the CAST. The new tank will be used as the suction source for all pump testing. During outages, the new tank will be used to fill the steam generators for wet lay-up. The new CAST will be the primary suction source for the CA pumps during transients. If the tank becomes unavailable as the result of an event, the suction will be automatically aligned to the assured suction source, the Nuclear Service Water System (RN). The CAST will maintain a volume of 300,000 gallons, which is sufficient to meet the short-term needs of the CA system. Following swap over to the assured source of water (RN), the CAST does not need to be isolated. 1CA2 will not need to be closed. Maintaining 1CA2 open after swap over to RN will have no impact on the availability of the assured source of water for the CA pumps. The automatic realignment to the RN system will remain the assured suction source for the CA system and will be unaffected by this modification. The automatic initiation of CA will not be affected by this modification. The safety grade indication of auxiliary flow to each steam generator will not be impacted by this modification. This modification will replace the three-condensate storage tanks with a single tank. The current configuration normally has a capacity of about 280,000 gallons, at best, with only 130,000 gallons of condensate quality water readily available. This modification will increase the nominal capacity of the preferred condensate-

quality water to approximately 300,000 gallons. The new elevated storage tank is non-safety related, is not seismically qualified and is not be protected from tornado-generated missiles. This modification is considered to be an equivalent replacement, which will improve the reliability and quantity of the preferred source of condensate-quality water. The operator actions implemented to maintain the availability of the preferred sources of water throughout all conditions and circumstances will be deleted, thus simplifying operation of the CA system and reducing operator burden. No USQ exists.

MG-52521/P2

This modification terminates spare inverter SKX to busses KXB and 2KU (via disconnect switch multiple lugged outputs). This modification also installs and connects a new breaker in distribution center DCB-1D. Further, this modification installs and connects a new breaker in regulated power distribution center MKB-2C. These connections will be terminated on hot busses. This modification will also modify and install doors and pans to accommodate the large breakers for DCB-1D and MKB-2C. The new spare inverter (SKX) will be located in the Auxiliary Building Battery Room, elevation 733'.

Procedure TN/0/B/2521/P2/01E provides instruction for implementing this modification. This part of the modification will connect the 125VDC/120VAC Non-Vital Inverter SKX to the 125VDC Distribution Center DCB, 240/120VAC Distribution Center MKB and 120 VAC Panelboard KXB, 120VAC Panelboard 2KU. This is a new procedure.

The new inverter and associated equipment are not nuclear safety related. This modification and the electrical implementing procedure will not affect the design, function, or operation of the EPF system as described in the SAR. The new inverter and associated equipment will be mounted per QA-4 requirements to prevent damage to other QA-1 equipment in the Battery Room. The spare inverter and breaker alignment panel will be located near the existing inverters, within the protected spray shield wall, and will be a similar size. The equipment is designed to operate for a 45-year life. However, this equipment does not have equipment qualification requirements. There are no formal environmental qualification (EQ) requirements for the 240/120 V AC Auxiliary Control Power System (EPF), however; the new spare inverter is rated for operation in an environment with ambient temperatures or 0 to 43° C minimum, with a relative humidity of 0% to 95% and an accident radiation dose of 1000 Rads. This is adequate for the intended location of the spare inverter. The heat load impact due to the addition of the spare inverter is negligible. Normally, only four out of the five inverters are expected to be operating at a given time, so this should be no different than the current state with respect to heat loading. No USQ exists.

Nuclear Station Modifications

NSM-12495
NSM-22495

These modifications deleted the low steam line pressure signal input into Safety Injection (SI) actuation circuitry internal to the Solid State Protection System (SSPS). The low steam line pressure signals are generated in the 7300 Process Control System. The signals are then sent to both A and B trains of the SSPS. The signals are sent from there to four 2/3 Universal Logic Boards in the logic bay of the SSPS. The output of these cards generates main steam isolation and safety signals. These modifications removed the wiring between the logic cards and the SI/Reactor trip circuits. This deletes SI on low steam line pressure. The steam line isolation signal will remain. Control board labels were modified to remove reference to this safety injection function also. Approved Technical Specification amendment 182/164 addressed changes related to these modifications. No USQ exists.

NSM-12504
NSM-22504

These modifications replaced the Unit 1 and Unit 2 train B diesel generator load sequencer timers. The Agastat pneumatic and Cutler-Hammer D87 on-delay timers needed to be replaced because of aging and degrading timing tolerances. The replacement timers are NST Tempo 812 series solid state timers which have been tested and demonstrate better operating characteristics. The replacement timers are qualified QA-1 and located in the same cabinets as the existing timers.

The replacement timers perform the same function as the currently installed timers. Heat loads have been reviewed to assure that these components will be in an appropriate environment and will not adversely impact other equipment in the cabinets. Mounting and seismic qualification of the replacement timers has been performed to assure that all QA requirements are satisfied. All changes are contained within the electrical cabinets such that no new Appendix R concerns are generated. The modifications were implemented on a train basis by procedure. Each affected train is covered under an independent but related nuclear station modification. The modifications are implemented by implementing procedures which control the work and are reviewed independent of this evaluation. The load sequence as shown in UFSAR Figure 8-1 will continue to be met as before. The replacement timers are expected to be more consistent in performance than the current components. The replacement timers are solid state electronic components whereas the old timers are electro-mechanical devices. The electronic devices have demonstrated greater reliability and consistency in performance. They are not subject to mechanical failure as are the currently installed timers. Both the old and new timers are subject to electrical failure. No USQ exists.

NSM-12507

This modification replaces the protective relaying for the Unit 1 230 kv buslines that run from the station to the McGuire switchyard. All four channels of line sensing and ground fault detection relays will be replaced. The modification also replaced the communications portion of the buslines which involves both supervisory and transfer trip functions. This modification is being implemented to replace and update old equipment which is near the end of its service life and difficult to maintain.

The modification does not change the function of the busline protective relaying system. The equipment is updated and will operate faster than the current system to isolate the station from the busline in the event of a fault on the busline but will not change the criteria for separation from the busline.

The Offsite Power System is the preferred source of power for station operation but is not safety related. Loss of Offsite Power is considered a design basis event and the station is designed and operated such that the effects of this event can be safely mitigated. The replacement of the busline protective relays will not increase the likelihood of loss of offsite power or cause the station to remain connected to the offsite grid when abnormal conditions justify separation from the system. The criteria for operation of the busline protective relays will remain unchanged. The new equipment will increase the reliability and speed of the system in responding to problems and will be easier to maintain due to the ready availability of parts. The replacement components meet the criteria of the Maintenance Rule in that they increase reliability and reduce the risk of maintenance preventable failure. No USQ exists.

NSM-12515
NSM-22515

These modifications replaced the discharge check valves for each Auxiliary Feedwater System (CA) pump with an Automatic Recirculation Control (ARC) valve. These valves provide an assured minimum flow path for the CA pumps. The ARC valve operates as a combined check valve and glove valve with the globe valve port providing an assured flow path to prevent deadhead conditions on the pump. The check valve port is the main discharge flow path and is operated in the same manner as the previous check valve. The ARC valve is designed such that the glove port closes as the check port opens assuring a continuous discharge path for the pump and automatically eliminating recirculation flow as sufficient normal discharge flow is established. The installation of this recirculation flow path will result in removal from service of the current recirculation flow paths for the CA pumps. This piping will now be used only for pump flow testing. The current recirculation valves will have automatic controls removed and will only be operated manually.

In addition, a continuous discharge flowpath will be added between each ARC valve and the feedwater pressure boundary valves to assure that this piping is not pressurized due to valve leakage. These flowpaths will have a flow control valve in line to limit flow to 5 gpm at 1700 psid.

Post modification testing will be performed on the revised system to ensure correct flow balance and operation of the discharge and recirculation flowpaths. The operation of automatic functions on a system start signal will be verified using a simulated signal. This testing ensures that the revised system will respond as designed. Operation of all revised components will be tested and data collected to allow future determination of system health.

These modifications improve the reliability of the recirculation function for the CA pumps. The current arrangement is subject to failure due to a loss of valve control instrument power. The revised flowpath is controlled by mechanical action of the valve such that either the normal discharge flowpath or the recirculation flowpath is always available. The ARC valves are considered as reliable as the current discharge check valves and recirculation control valves for valve malfunction in other respects. The use of the ARC valves along with the provision of a flow path to prevent pressurization due to valve leakage maintains all the design operation functions being provided by the current system design. The diversion of 5 gpm of CA supply to the feedwater system during operation does not significantly impact the ability of the CA system to meet its design requirements. No new functions are added by the installation and use of the ARC valves. The testing of the systems will utilize the old recirculation flow path since the

automatic action the ARC valve prevents it being used for testing. The ARC valves and new recirculation meet the same QA standards as the associated piping and components. No USQ exists.

NSM-12517

This modification replaced the upper internal guide tube support pins, which are known as split pins. The currently installed split pins are fabricated from Inconel X-750 Revision A and each pin has the appearance of a "clothes pin." A split pin has two separate legs that have an interference fit with a hole in the upper core plate. The split pins are attached to the bottom flange of a guide tube assembly via its shank, nut and locking device. The pins are captured with a washer welded to the bottom of the guide tube support. The material for the replacement split pin is SA-316CW which is qualified and approved for this application. The modifications to the upper internals include the following activities: 1) replace all split pins with cold worked type 316 stainless steel material, 2) remove all guide tube/flow column hold down bolts and install approximately 50% of the hold down bolts fabricated from cold worked type 316 stainless steel material, except for the guide tubes adjacent to the outlet nozzles. The guide tubes located adjacent to the outlet nozzles will have 8 replacement bolts installed, 3) Remove all 15X15 guide tube flexures and removable inserts and replace them with mechanical plugs and 4) Remove all orifice plates from 17X17 plutonium recycle guide tubes. An evaluation was performed (WCAP-15252, "Duke Energy Corporation McGuire and Catawba Units 1 and 2 Replacement CW316 Support Pin Design Qualification Report") to document the acceptability of the replacement CW316 support pin material and pin design. The engineering evaluation documents the acceptability of the pin and nut assembly material and assembly. In so doing, the evaluation documents the functional equivalency of the associated replacement hardware such as the guide tube hold down cap screws, guide tube cover plate hardware (which are essentially like-in-kind replacements), and the contingent tandem locking caps. No USQ exists.

NSM-12518/P2

This modification installed a non-safety related, elevated, Auxiliary Feedwater Storage Tank (CAST) and isolated all other sources. The location of the tank is near the southwest corner of the Turbine Building. The new tank is a multi-legged elevated storage tank that is vented to the atmosphere. The tank design is a torospherical type and meets the AWWA Standard for Welded Steel Tanks for Water Storage, ANSI/AWWA D100-96. The new CAST will be aligned to the Auxiliary Feedwater (CA) pumps at all times. It will be used as the primary suction source. The Upper Storage Tank (UST) and the Auxiliary Feedwater Condensate Storage Tank (CACST) will remain tied to the CA system but normally isolated from the CA pumps. The recirculation flow from the CA pumps will be returned to the CAST. The new tank will be used as the suction source for all pump testing. During outages, the new tank will be used to fill the steam generators for wet lay-up. The new CAST will be the primary suction source for the CA pumps during transients. If the tank becomes unavailable as the result of an event, the suction will be automatically aligned to the assured suction source, the Nuclear Service Water System (RN). The CAST will maintain a volume of 300,000 gallons, which is sufficient to meet the short-term needs of the CA system. Following swap over to the assured source of water (RN), the CAST does not need to be isolated. 1CA2 will not need to be closed. Maintaining 1CA2 open after swap over to RN will have no impact on the availability of the assured source of water for the CA pumps. The automatic realignment to the RN system will remain the assured suction source for the CA system and will be unaffected by this modification. The automatic initiation of CA will not be affected by this modification. The safety grade indication of auxiliary flow to each steam generator will not be impacted by this modification. This modification will replace the three-condensate storage tanks with a single tank. The current configuration normally has a capacity of about 280,000 gallons, at best, with only 130,000 gallons of condensate quality water readily available. This modification will increase the nominal capacity of the preferred condensate-

quality water to approximately 300,000 gallons. The new elevated storage tank is non-safety related, is not seismically qualified and is not be protected from tornado-generated missiles. This modification is considered to be an equivalent replacement, which will improve the reliability and quantity of the preferred source of condensate-quality water. The operator actions implemented to maintain the availability of the preferred sources of water throughout all conditions and circumstances will be deleted, thus simplifying operation of the CA system and reducing operator burden. No USQ exists.

MG-52521/P2

This modification terminates spare inverter SKX to busses KXB and 2KU (via disconnect switch multiple lugged outputs). This modification also installs and connects a new breaker in distribution center DCB-1D. Further, this modification installs and connects a new breaker in regulated power distribution center MKB-2C. These connections will be terminated on hot busses. This modification will also modify and install doors and pans to accommodate the large breakers for DCB-1D and MKB-2C. The new spare inverter (SKX) will be located in the Auxiliary Building Battery Room, elevation 733'.

Procedure TN/0/B/2521/P2/01E provides instruction for implementing this modification. This part of the modification will connect the 125VDC/120VAC Non-Vital Inverter SKX to the 125VDC Distribution Center DCB, 240/120VAC Distribution Center MKB and 120 VAC Panelboard KXB, 120VAC Panelboard 2KU. This is a new procedure.

The new inverter and associated equipment are not nuclear safety related. This modification and the electrical implementing procedure will not affect the design, function, or operation of the EPF system as described in the SAR. The new inverter and associated equipment will be mounted per QA-4 requirements to prevent damage to other QA-1 equipment in the Battery Room. The spare inverter and breaker alignment panel will be located near the existing inverters, within the protected spray shield wall, and will be a similar size. The equipment is designed to operate for a 45-year life. However, this equipment does not have equipment qualification requirements. There are no formal environmental qualification (EQ) requirements for the 240/120 V AC Auxiliary Control Power System (EPF), however; the new spare inverter is rated for operation in an environment with ambient temperatures or 0 to 43° C minimum, with a relative humidity of 0% to 95% and an accident radiation dose of 1000 Rads. This is adequate for the intended location of the spare inverter. The heat load impact due to the addition of the spare inverter is negligible. Normally, only four out of the five inverters are expected to be operating at a given time, so this should be no different than the current state with respect to heat loading. No USQ exists.

Miscellaneous Changes

UFSAR Change

This change corrected statements regarding chlorine gas in Sections 2.2.3, 6.4.3 and Table 9-28. The statements address the quantity of chlorine cylinders on site, the location of the cylinders, the quantity of chlorine released in the event of a postulated cylinder failure, their use to support unit operation, and the likely migration path for released chlorine.

The change is a result of the UFSAR accuracy review project. As such, there are no changes to the current design and configuration of the plant. Rather, the changes are being made to reflect the current plant design, configuration, and operation within the UFSAR. The change does not affect the operation, design basis or function of any structure, system or component. No safety or licensing issues and no revisions to regulatory commitments are involved. No USQ exists.

UFSAR Change

This change adds a clarifying note to Table 5-5. The table was changed to add a double asterisk beside the column heading term "Active," and added the following to the last line of the table on each page: "Note: ** Active here means the valve must change position to perform a safety or shutdown function and is not intended to designate that the valve is postulated to fail."

This revision does not change the current operation, design bases, or function of any structure, system or component. The change does not constitute a physical change to the current design, configuration or operation of the plant. The change is supported by current plant design and licensing documents and does not affect the Technical Specifications.

The addition of the clarifying note to differentiate the term active as used in this table from active failures which are discussed in UFSAR Chapter 3, does not result in an unreviewed safety question or any licensing issues. The change does not affect plant licensing or design bases nor will it affect the operation or safety function of any structure, system or component. No USQ exists.

UFSAR Change/Procedure Changes

This change updates UFSAR Table 6-17, 6-125, 6-127, procedures EP/1(2)/A5000/ES-1.3 and procedures AP/1(2)/5500/34. The criteria for initiating the swapover for the Containment Spray (NS) System from the Refueling Water Storage Tank (RWST) to the sump is to complete swapover prior to ice bed meltout to prevent a containment pressure spike. The added step in Table 6-125 will allow the operator to quickly move to the steps associated with aligning B train NS pump to the sump when there is a failure of the A NS pump. This change will eliminate steps which would not be of any use if there is a failure of the A train NS pump. This will allow swapover to be completed in a timely manner. There are not safety implications from this change. All actions will be performed as described and only steps that would not lead to positive results in relation to NS pump operation will be eliminated. The change to Table 6-127 will reference the new step added to Table 6-125 and ensure the correct steps are referenced. The change to Table 6-125 will not change the information given in this table.

The addition of steps to wait until NS20A and NS3B are closed ensures the permissive to allow the opening of the NS sump suction valves is received. Operators cannot proceed until the valves are open, so this does not delay any time critical action.

The change of the note in Table 6-17 maintains the 180 second delay time for the start of spray flow but is more general for the times that go into the 180 seconds. This will allow for operational flexibility while not increasing the delay times. Unlike Table 6-125, Table 6-17 does not give all the steps required to go from the injection mode to cold leg recirculation. This change will not alter the intent of this table.

The procedures listed were revised to reflect the added step in Table 6-125. This revision will only allow the operator to move through the steps more quickly in the event of the failure of the A train NS pump. No other changes to the procedures are made.

These changes increase the operator's ability to perform all required actions and have been validated on the simulator. The change only improve the operational flexibility in mitigating consequences of SAR analyzed accidents and have no impact on the probability of an accident. The changes do not change the way the NS system is operated. They do not cause any additional equipment to be operated or any equipment operated in an unusual manner. Since no additional equipment is used in any way not previously analyzed, no increase in the probability of an occurrence of a malfunction of equipment important to safety previously evaluated in the SAR is created. These changes do not cause the plant to be operated in any unanalyzed manner and do not increase the consequences of any accident previously evaluated in the SAR. All actions taken to mitigate the consequences of an accident are taken in the same manner as before and no changes to design criteria, setpoints or sequence of mitigation actions are made. No USQ exists.

UFSAR Change

This USQ evaluation supports changes to Chapter 6 to replace the "at 50 minutes" and "no earlier than 50 minutes" statements with "approximately 50 minutes" in Sections 6.2.1.3.2.6, 6.2.1.3.2.7, 6.3.2.2.2, 6.3.3.9, Table 6-19 and Table 6-128 and changing "minimum" to "approximate" in the second paragraph of 6.3.3.9 and deleting "minimum" in the last paragraph of 6.3.3.9. The changes provide consistency between sections and allow for operational flexibility.

The criteria for Residual Heat Removal (ND) System swapover to the Auxiliary Reactor Building Spray Header is to delay the swapover until latent and decay heat levels in the Reactor Coolant (NC) System are low enough to safety allow reduction in coolant flow, assure that ND suction is from the containment sump and to complete swapover to the Containment Spray (NS) System prior to ice bed meltout to prevent a containment pressure spike. At 45 minutes, the decay heat load is less than 1.5% of full power per DPC-1552.08-00-0141. From DPC-1552.08-00-0161, the Emergency Core Cooling System (ECCS) water temperature returning to the reactor vessel at 45 minutes is <150F. The ECCS flowrate to make up for the boiloff at 45 minutes is 342 gpm. From DPC-1552.08-00-0136, 730 gpm is available from one ND plus one Safety Injection (NI) System pump during cold leg recirculation mode. Since 730 is greater than 342 gpm, core cooling is assured if ND auxiliary spray flow is aligned as early as 45 minutes after the limiting LOCA. In fact, any ECCS flow in excess of 342 gpm will only be spilled on the floor. It is therefore obvious that stopping ND flow at this time has no significant effect on the accident analysis. The UFSAR description of the swapover step in procedures does not clearly indicate any tolerance for taking this action prior to 50 minutes but does indicate that the action could occur later than 50 minutes. There is sufficient conservatism in the assumptions for ice bed melt out to assure that a delay beyond 50 minutes does not adversely impact containment pressure control. In addition, plant conditions which increase ice melt out will cause the swapover function to occur sooner and plant conditions which tend to delay swapover also result in slower ice bed melting. The change in description of the swapover action from "at 50 minutes" or "after 50 minutes" to "approximately 50 minutes" is consistent with the consideration of this action in safety analysis and more clearly describes the action use of the procedures

in practice. There are no changes to safety analysis margins, setpoints or acceptance criteria associated with this change. No USQ exists.

UFSAR Change

This USQ evaluation supports changes to Section 5.2.1.5 and 6.3.3.2, Emergency Core Cooling System (ECCS) Performance. These changes are to support previous changes to the method, in which the plant is operated, which were oversights when those changes were made. Current operating procedures maintain a 75F/hr cooldown limit, and these changes are to support those limits.

Two changes are made, the removal of the paragraph describing procedural limitations imposed in section 5.2.1.5, and deletion of a statement in section 6.3.3.2.

The statement in 6.3.3.2 was added during a revision to be an example of a way of meeting the requirements of the discussion regarding isolation of the cold leg accumulators. It states that the analysis conducted assumed that the reactor core had been shutdown for 2.5 hours and the temperature was 425F. The 50 F/hr is just one of the methods that could be employed to reach this condition and at that time was the cooldown limit by procedure, it indicated that the requirements of that section would be met with no changes to current operating procedures. The paragraph in section 5.2.1.5 indicated an intention of the plant to operate at 50F/hr. Further information indicated that 75F/hr is an acceptable administrative operating limit, which is still significantly below the Technical Specification of 100F/hr.

These changes do not affect the way any operations are performed. The requirement of 2.5 hours and 425F is maintained and this requirement will continue to be followed. Heatup/Cooldown rate is deleted to reflect current plant procedures. These changes will not affect any equipment in the plant, nor the way it is operated. The changes more accurately reflect current plant operating procedures. No physical changes are made to the plant. No USQ exists.

UFSAR Change

General Design Criteria 62 in Section 3.1 states, "During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shutdown the core at a Keff =0.90. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. Borated water is used to fill the fuel pool at a concentration equal to that used in the reactor cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure Keff=0.95 even if unborated water is used to fill the pool." The "Keff=0.90" in the above paragraph is being changed to "Keff=0.95," consistent with the Technical Specifications and other UFSAR sections. No design bases or safety function of any structure, system or component is adversely affected by the change. No USQ exists.

UFSAR Change

This changes are textural revisions to Sections 6.3.2.11 and 6.5.1 to correctly describe instrumentation, annunciators and OAC alarms available for detection of abnormal inputs to the Residual Heat Removal (ND) and the Containment Spray (NS) System sump. The revision corrects previous errors and describes OAC enhancements provided under Minor Modification MM-11615.

There are no Technical Specification requirements associated with ND/NS sump level detection. The additional OAC alarms will enhance the ability to detect abnormal inputs to the sump. There are no

adverse effects on equipment important to safety and has no physical effect on plant equipment or systems. No USQ exists.

UFSAR Change

This change provides an accurate description of how the Turbine Building Sump (WP) can be aligned to the Condenser Circulating Water System (RC) discharge piping and monitored for releases. This alignment is the preferred method for providing discharge when unwatering the condenser water boxes or circulating water pipes. It also can be used when the Turbine Building sump could be potentially contaminated during a primary to secondary leak and the release requires mixing the RC System. There is no impact on any safety related system responses and the change is editorial in nature. The editorial changes have no impact on any accident analysis. There are no physical changes to the WP system. No USQ exists.

UFSAR Change

Section 10.4.7.2.1 was changed to:

- Include a statement addressing the use of alternate approved reducing agents for oxygen control
- Provide editorial clarification on the use of alkaline amines as pH control agents for corrosion control
- Clarify the maximum inventory requirements for ammonium hydroxide and hydrazine that provides allowances for different vendor containers and solution concentrations
- Clarify references to the source documents from which the station secondary chemistry program is derived and document the manner in which changes and exceptions are evaluated and approved
- Update the discussion on condensate polisher operation to reflect the use of up to four cells for normal operation and change the spent resin volume to reflect the maximum precoat allowances
- Remove a reference to Table 9-27 and editorial changes not affecting content

The Electric Power Research Institute (EPRI) PWR Secondary Water Chemistry Guideline, Revision 4 allows for the use of alternate oxygen scavengers provided they are approved and qualified for use and have control levels commensurate with those provided for hydrazine. Currently, carbohydrazide has been qualified for use at McGuire. Appropriate concentration limits and corrective actions have been incorporated into the station chemistry program. The use of carbohydrazide has received a separate safety evaluation as part of the inclusion within the secondary chemistry program requirements outlined in Section 3.2, Secondary Systems Analytical Requirements and Corrective Actions of the McGuire Station Chemistry Manual.

The changes to the discussion on the use of alkaline amines for pH control represents clarification and does not constitute a change to the program or methodology currently employed.

The change to the hydrazine and ammonium hydroxide maximum storage levels does not represent an increase in the actual functional product that would be allowed on site. Rather, this change allows flexibility in container size and solution concentrations as provided by approved vendors under applicable purchase specifications.

This section of the UFSAR describes the required components of a secondary chemistry monitoring program designed to inhibit corrosion. The original reference to just the PWR Secondary Water Chemistry Guidelines as a source document is not complete since numerous other EPRI documents go into the development of a comprehensive secondary chemistry program.

Changes to the EPRI recommendations stemming from technological advances and new industry experiences may require revision to the secondary chemistry program in order to maintain compliance and ensure protection for system components. The change outlining the general process employed in developing and maintaining the secondary chemistry program, including safety-related evaluations where appropriate, ensures that adequate review and scrutiny on proposed changes are performed and potential impacts to nuclear safety are considered to prevent an unreviewed safety question from existing.

The paragraph dealing with condensate polisher operation was rewritten to better reflect current polisher operation. The use of all four of the polishers is routinely required in order to maximize the amount of condensate flow polished while minimizing overall system pressure losses. The change in coal volume from 17 to 20 cubic represents the inclusion of the maximum allowed by procedure and the polisher equipment vendor.

These changes involve outlining the general process by which the secondary chemistry program is developed and maintained in compliance with current vendor and EPRI documents such that corrosion protection is maintained and no safety-related concerns exist.

The use of alternate oxygen scavengers and the storage of hydrazine and ammonium hydroxide as described by this activity are in accordance with appropriate EPRI guidelines and consistent with established industry practices. Actual quantities of potentially hazardous chemical components are unchanged from their previously approved levels.

The allowance of four condensate polisher operations does not constitute a change outside the original design purpose.

No systems or components important to safety are impacted in a manner that would increase the probability or the consequences of accidents or component malfunctions. No USQ exists.

UFSAR Change

This change involves correcting inaccurate information or the enhancement/clarification of information contained in Chapter 3, Section 3.1. The corrections or changes are:

Criterion 3 - Fire Protection, paragraph 2 will be changed to read: "Also provided are reliable supplies of water and halon to appropriate parts of the station." This change does not involve any changes to the operation, design basis or function of any structure, system or component. The change involves correcting the statement that carbon dioxide and freon are used for fire protection when in fact halon is used. This correction agrees with Section 9.5.1.4. No physical changes or procedure changes are required.

Criterion 13 - Instrumentation and Control, paragraph 1, number 5 will be changed to read: "Reactor axial power balance (manual by rod motion)." This change does not involve any changes to the operation, design basis or function of any structure, system or component. The change involves clarifying the statement: "Reactor axial power balance (manual by part length rod control.) This power balance can be achieved by manually stepping the full length control rods. This correction agrees with Section 4.3.2.2.4 and 7.7.1.2.1. No physical changes or procedure changes are required. No USQ exists.

UFSAR Change

This change involves correcting inaccurate technical information contained in Section 4.3, and associated Tables and Figures related to the nuclear design of McGuire's reactor cores. Corrections are limited to those that, when incorporated, are supported by either high-level design or lower level (i.e., supporting) design and/or operation-related documents.

These changes have no affect on the operation, design basis, function, nor initiate any modifications to any structure, system or component. The changes only provide information to more accurately describe the nuclear design. The changes do not affect the Technical Specifications or Bases, or any plant procedures. No USQ exists.

UFSAR Change

The UFSAR accuracy project team found no basis for a statement in Section 6.3.2.11: "If a single failure is assumed to totally destroy the gland retaining ring and the valve backseat and to blow out the packing, gland and lantern ring, a maximum leak rate of 17.5 gpm past the shaft would occur." This statement applies to containment sump isolation valves only. The 17.5 leak rate is correct based on the original SER Section 6.2.5 statement, assuming worst case stem-packing clearances: "Seal leakage from the bonnet seals and packing glands would not exceed 17.5 gallons per minute and would be detected within the auxiliary building and the leak isolated by closing the valve." A single failure, active or passive, will not increase this leakage; failure of the backseat, gland retaining ring is still intact; failure of gland retaining ring, valve would be open against the backseat. The sentence is being corrected to read: "Leakage past the shaft (through the packing) would not exceed 17.5 gpm, assuming a single failure." A single failure that (credible) fails the valve opening, the backseat, and the gland retaining ring bolts can not occur. Flange gasket leaks and leak detection are discussed elsewhere in Section 6.3.2.11.

The UFSAR was revised to reflect a single failure for the containment sump isolation valves at the appropriate accident pressure. This change only corrects errors and does not increase the change or consequences of an accident/malfunction, or create the possibility of any new accidents. No USQ exists.

UFSAR Change

Section 9.1.2.3.2, "Cask Drop Evaluation", was revised to include the TN-32 Cask in the list of spent fuel casks evaluated and to include the updated methodology for evaluating cask drop accidents. Selected Licensee Commitments (SLC) Section 16.15-3.9.7 was revised to include the following statement in the Bases: "The requirement for following the load path shown in Figure 3.9-1 assures that the cask can not fall into the spent fuel pool." The phrases "truck cask" found within this SLC section was changed to "spent fuel cask". In addition, errors made when Specification 3.9.7 related to the SLC manual are corrected.

In support of the ISFSI project the calculation for the postulated cask-drop accident was updated to include the TN-32 cask. This update required this section to be revised to include the TN-32 cask in the list of casks evaluated. The cask drop evaluation concluded that should the cask break free as postulated during cask handling, the cask would not enter or fall into the spent fuel pool, when traveling along a prescribed path. The prescribed path assures that the cask's center of gravity will be located over the spent fuel cask pit such that any tipping of a dropped cask would be within the confines of the cask pit. The prescribed path also precludes the cask from passing over or near safety related equipment and restricts the cask to areas designed to accommodate a dropped cask with only negligible damage to the

structural concrete. The changes do not directly affect cask handling equipment or procedures. No USQ exists.

UFSAR Change

The UFSAR does not mention the 1.5 hour self contained battery lights provided in the Standby Shutdown Facility (SSF) for hot standby. Appendix R typically requires 8 hour self contained battery lights under this condition. This revision clarifies that 1.5 hour lights, not 8 hour lights are used in the SSF. This change does not modify any existing plant equipment. It is a clarification of what is already installed and approved by the NRC (reference letter from the NRC dated April 16, 1984 to H. B. Tucker, Subject: Amendment 31 to Facility Operating License NPF-9 and Amendment 12, NPF-17). This is a documentation change only and does not affect the actual configuration of the plant. No USQ exists.

UFSAR Change

This change to the UFSAR deletes a sentence and provides additional information regarding the radiation monitoring system. UFSAR Table 11-28 is updated to provide information regarding the steam generator radiation monitors (1,2EMF71, 72, 73, & 74). A sentence in Section 11.4.2 was deleted. UFSAR Section 12.4.1 is updated to identify certain exceptions regarding alarms and displays for certain radiation monitors.

These changes are being made to more accurately reflect the current plant configuration. These changes will not result in any changes to the hardware in the plant or to how the radiation monitors are operated. The design objectives and the purpose for the EMF related monitoring system are not affected by the changes to the UFSAR. The changes are considered administrative in nature. No USQ exists.

UFSAR Change

Sections 7.6.6.2.7 and 7.6.18.1.4 were revised because of the erroneous statement that no interlocks exist in the control scheme for the Electric Hydrogen Recombiners and the Hydrogen Mitigation System. Interlocks from the Diesel Generator Load Sequencer are provided to assure that these loads are applied in a controlled manner. The interlocks were part of the original system design and the evaluation shows that performance and operational characteristics of the Electric Hydrogen Recombiners and Hydrogen Mitigation System as described in the UFSAR are not affected by the interlocks. The Electric Hydrogen Recombiners and Hydrogen Mitigation System are used for accident mitigation and are not accident initiators. No USQ exists.

UFSAR Change

This change affects Table 9-8, "Nuclear Service Water Flow Requirements". The specific change involves a reduction in essential service water flow rate to the Control Area Chilled Water (YC) system chiller condensers. The actual revision to Table 9-8 involves changes in the values for these service water supplied components under the "LOCA Signal" and "LOCA P Signal" columns. This current listing for service water flow rate is stated as 775 gpm under these columns. The revised listing will indicate a service water flow rate of 698 gpm. The total service water flow rates were revised to reflect this slight reduction in overall system flow rate. Justification for decreasing the service water flow rate requirement for the YC Chiller Condensers is contained within Calculation MCC-1211.00-33-0010 "YC Chiller Condenser Cooling Water Flow Rate Requirements." Neither the Nuclear Service Water System nor the Control Room Ventilation System are impacted in any detrimental manner as a result of this change. No USQ exists.

UFSAR Change

These changes affect Section 4.2 and associated tables and figures. These corrections or changes are a result of the UFSAR accuracy review project.

Various editorial changes were made throughout the section to correct punctuation, spelling or grammatical errors. These changes do no alter the intent and meet the requirements of a non-technical editorial change. These changes are covered by an editorial USQ evaluation.

The technical changes to the description of the mechanical design of reactor components does not change the current operation, design bases, or function of any structure, system or component. The changes were made to reflect current reactor component design, configuration and operation. The changes are supported by plant design documents, reports and procedures. The changes do not involve any physical changes to any systems, structures, or components in the plant. The changes do no affect the results of any accident analysis nor will they initiate an accident previously evaluated. The probability or consequence of any postulated accident will not be increased. There are no adverse affects on design requirements or operation of any equipment important to safety. As a result, the probability or consequence of a malfunction of equipment important to safety is not increased nor is the possibility for a different type of SSC malfunction created. Since no Technical Specifications are associated with or affected by this change, the margin of safety is not reduced. No USQ exists.

UFSAR Change

This change corrected typographical and numerical errors in Section 4.4. These corrections/changes are a result of the UFSAR accuracy review project.. As such, they do not constitute a change in the design, configuration, or operation of the plant. Rather, these changes are being made so that the UFSAR will reflect the current plant configuration. The changes to the thermal and hydraulic design of the reactor do not change the current operation, design bases, or function of any structure, system or component. These changes are supported by plant design documents, calculations, approved vendor reports, and plant procedures. No USQ exists.

UFSAR Change

UFSAR Chapter 13.1, "Organization" was revised to change administrative information to update the current organizational structure of Duke Energy Corporation's McGuire Nuclear Station. These changes do not affect the licensing or design bases nor do they affect the operation or safety function of any structure, system or component. There are no effects on plant operation and no physical changes being made to the plant. No USQ exists.

UFSAR Change

The change evaluated involves changes to Sections 8.3.1.11.2, "24,000 Volt Unit Main Power System", 8.3.1.1.4, "4160 Volt Essential Auxiliary Power System", Figures 8-17 and 8-31, and Table 8-7. Corrections or changes are limited to those that: 1) correct/clarify technical information by updating information to more clearly reflect current practices, 2) are supported by design, licensing, and/or operations-related documents, 3) do not affect the Technical Specifications, 4) do not affect the licensing or design bases nor will they affect the operation or safety function of any structure, system or component.

An evaluation of these changes indicated that these changes do not constitute a USQ. The changes are limited to correction/clarification of technical and editorial information which does not affect the licensing or design bases nor the operation of safety function of any structure, system or component. No USQ exists.

UFSAR Change

The change evaluated involves changes to UFSAR Chapter 3, Table 3-4, "Summary of Criteria - Mechanical System Components" to correct technical and editorial information. Corrections or changes are limited to those that: 1) have no effect on the operation, design bases, safety function, or regulatory commitments of any structure, system or component (SSC) for Units 1 and 2.

The changes are a result of the UFSAR accuracy review project and therefore do no constitute any change to the current design, configuration, or operation of the plant. The changes reflect the current design, configuration and operating practices of the plant and switchyard. No USQ exists.

Selected Licensee Commitment Change

This change added Section 16.9.22, Switchgear Room Ventilation System (SWGRVS) and Section 16.9.23, Control Room Area and Battery Room Ventilation System (CRABRVS) to the Selected Licensee Commitments (SLC) Manual. This revision provides the commitment provisos, applicability statement, remedial actions and testing requirements for the subsystems within the Control Room Area Ventilation System (CRAVS) that are not governed by Technical Specification 3.7.9.

The Technical Specification sections regarding the CRAVS pertains only to the Control Room portion of the system that provides a protected environment from which licensed operators can control the unit following an accident. Other subsystems of the Control Room Ventilation (VC) systems serve the Control Room Area, Switchgear Rooms and Battery Rooms.

Specification 3.7.9 has routinely been invoked unnecessarily when components of the SWGRVS or the CRABRVS systems become inoperable because these components have a "VC System" designation. To address this issue, the SLC manual was modified to include guidance for non-CRAVS components of the VC system. The guidance provided within the SLC manual has no impact to the CRAVS portion of the VC system. The update to the SLC manual only addresses the non-safety-related portion of the VC system. The function, operation or design basis of the CRAVS system will not be affected. The changes do not involve any safety or licensing issues associated with the CRAVS systems. There will be no impact to any regulatory commitments that have been made regarding the CRAVS portion of the VC system. This update to the SLC manual will not require any modifications or other changes to the VC system. No USQ exists.

Selected Licensee Commitment Change

These changes to Selected Licensee Commitment (SLC) 16.9.15 clarify appropriate use of the snubber required actions, and provide cautions for future revisions concerning maintaining the approved code requirement relief.

SLC 16.9.15 provides a regulatory commitment addressing operability and inspection requirements for snubbers. This commitment was previously a Technical Specification (TS) and was approved for relocation to the UFSAR. The changes evaluated resolve two issues that are a result of the relocation from TS to the UFSAR.

The first issue is resolved by a revision to the Bases section of SLC 16.9.15 to acknowledge that the snubber testing requirements are the bases for NRC granted relief to be performed as an alternative to ASME code testing requirements. The NRC previously approved a relief request from the snubber visual examination and functional testing requirements of ASME Section XI, 1989 edition, Article IWF-5000, Subsection IWF-5300. This relief was approved by a Safety Evaluation Report (SER) dated May 27, 1998. The relief was approved prior to the relocation of the snubber inspection requirements from TSs. The existing snubber inspection requirements were approved "based on a finding that the proposed alternative provides an acceptable level of quality and safety." Since the snubber inspection requirements have not been altered from that reviewed by the NRC staff, the relief remains valid. However, the issue being resolved is a concern that now the snubber inspection requirements are subject to revision per 10 CFR 50.59 without a required NRC staff review. Therefore, this UFSAR change adds documentation and reference of the granted relief to facilitate consideration for any warranted NRC staff review as part of any future revisions.

The second issue resolved rewording the required action of SLC 16.9.15 to accurately reflect the reduction in authority from a TS requirement to a UFSAR commitment. Specifically, the required action previously directed that upon a condition of "one or more snubbers inoperable", a completion time of "72 hours" was allowed prior to requiring the action to "declare the attached system inoperable." This required action was applicable to the limited condition of operability of TS systems. This was acceptable as a required action permitted by TSs; however, since the snubber requirements were relocated to the UFSAR, the actions do not have the authority to delay or postpone the immediate evaluation of TS required operability and applicable actions. Therefore, the wording was revised for the required actions of SLC 16.9.15 to be consistent with requiring immediate determination of operability and compliance with all TS required actions upon discover of a condition adverse to quality.

The changes to Chapter 16, SLC 16.9.15 do not alter the configuration of any structure, system, and/or component (SSC), but re-enforce compliance with regulations and requirements. No USQ exists.

Selected Licensee Commitment Change

Selected Licensee Commitment 16.9-1, "Fire Suppression Water System" Remedial Action and Testing Requirements was changed to provide Operations the flexibility to place at least one main fire pump in continuous operation OR rely upon one of the two main fire pump automatic starting function circuits to ensure the Fire Protection System remains fully capable of performing its design basis function. Additionally, this revision provides Operations the flexibility to place at least one main fire pump in continuous operation to ensure system standby readiness when the RF jockey pumps are unable to maintain header pressure.

The change to the "Testing Requirement" will ensure both the primary and secondary main fire pump auto-start features (control circuits) are tested at the specified frequency. This SLC revision does not require the revision or deletion of automatic/manual main fire pump start functions. Although to some extent, implicit with regards to the Fire Protection System design, this revision explicitly allows the previously mentioned necessary operational flexibility for the Fire Protection System. The Fire Protection System is non-nuclear safety related and does not mitigate the consequences of any design basis accident.

The purpose of this change is to provide desirable system flexibility for ensuring maximum standby readiness condition for the Fire Protection System. No unwanted system interaction will result from this

activity. Based on the RF/RY system design, the effects of this activity are judged to be acceptable. This change does not effect the licensing, design basis or safety function of any SSC. No USQ exists.

Selected Licensee Commitment Change

SLC testing requirement 16.8.3.3 requires that at least once every 18 months a verification of DG cell plates show no visual indication of physical damage or abnormal deterioration. Although visual battery cell inspection is a recognized inspection practice for lead-acid batteries because the typical lead-acid battery jar design is of a clear plastic material, the DG Nickel Cadmium battery cell container design is a non-transparent white color such that a visual inspection of the "cell plates" cannot be performed. Periodic visual surveillance and electrical testing are performed on each DG battery to ensure battery functionality and operability. The Unit Main Power (EPA) System will continue to perform the same function. This revision does not change any plant SSC or equipment. No equipment or system safety functions or design basis is changed. There are no adverse impacts to nuclear safety or any SSC due to this change. Therefore, the SLC testing requirement to verify DC cell plates show no visual indication of physical damage or abnormal deterioration per SLC 16.8.3.3 is being deleted.

The DG batteries are components in the 125 VDC D Control Power System (EPQ), which provide power to the Class 1E diesel generator fuel oil booster pump motor, generator field flashing, and the control loads necessary for proper diesel starting operation during a blackout and/or LOCA conditions. Each battery and its respective charger are housed in an enclosed, QA Condition 1, metal cabinet locating in its respective diesel generator room. The battery is located on a seismically qualified battery rack that is bolted to the floor with the cabinet mounted to the floor around the battery/rack combination. No USQ exists.

Selected Licensee Commitment Change

SLC chapter 16.9 identifies the portions of the Fire Protection (RF) System that are required to be operable and provides the testing requirements for these systems. The original Testing Requirements stated in Section 16.9-4(a)(i) requires fire hose station inspections to be conducted at least once per 31 days. The SLC revision changes this frequency from a prescribed monthly (31 days) inspection to an inspection frequency determined by performance based results. The performance based frequency is dependent on maintaining 99% fire hose station reliability over a 3 year rolling period. This approach allows fire hose station inspections to be conducted at monthly, quarterly, semiannual or annual frequencies. These frequencies will be determined by the site fire protection engineer based on annual review of completed inspection results.

Inspection of fire hose stations is discussed in UFSAR Section 9.5.1.4. There are no specific inspection or testing intervals provided in this section. This portion of the UFSAR states that "all hose stations, receive inspection and testing to ensure their proper operation." This SLC change continues to provide inspections at intervals commensurate with performance to ensure proper operation; therefore, the fire hose station inspection frequency does not deviate from the UFSAR requirements. No USQ exists.

Selected Licensee Commitment Change

The following statement was added to the Bases section for SLC 16.9-3:

Testing Requirement [a.i] requires that valves in the flow path for the required halon systems be verified to be in their correct position. Although the selector valves and the cylinder valves are in the flow path, these valves are excluded from this testing requirement for the following reasons:

1. There is no visible means of determining valve position,
2. The valves are spring loaded piston actuators which fail closed and require halon discharge header pressure to open (Selector Valves Only),
3. There is no credible means to mis-position these valves other than actual actuation of the halon system,
4. These valves are an integral component of the actuation circuitry for the halon system, which is tested per testing requirements [a.iii], and
5. This exclusion is consistent with fire protection industry practices.

The required Halon 1301 fire protection systems that are identified in SLC 16.9-3 are:

1. Auxiliary feedwater pump turbines
2. Emergency diesel generators and related oil systems within the Diesel Building

The SLC testing requirements provide assurance that the minimum Operability requirements of the Halon Fire Suppression systems are met. SLC Testing Requirement [a.i] requires verification that each valve in the flow path (manual, power-operated, or automatic) is in its correct position at least once per 31 days. The change to the Bases clarifies the intent of this testing requirement. The selector valves and the cylinder valves are in the halon system flow path as addressed by SLC 16.9-3. However, the original intent of the SLC surveillance was to exclude these valves from the fire protection system "testing requirements" provided by SLC 16.9-3, Item [a.i].

The current interpretation of the testing requirements is that valves in the flow path that need to be verified for correct position are those valves which their position is readily detectable. The selector valves and the cylinder valves are not included within this test classification as their position is not easily (visually) determined. Only by testing can the actual position of these valves be determined. There is no credible means for these valves to be mispositioned from their normal position, other than actual actuation of the halon systems which is detectable. In addition, the design of the system is of such a nature that these valves are an integral component of the actuation circuitry for the halon system. As such, SLC 16.9-3, Item [a.iii] is the testing requirement for these valves. The correct position of these valves cannot be changed unless the system is actuated by detector or manual means. Restoration of the system would include returning the valves to the closed position. It is clear that these valves were never intended to be include in the valves to be tested once every 31 days. Furthermore, in discussion with other utilities, the interpretation of the SLC working at McGuire is consistent with the industry practices. No USQ exists.

Selected Licensee Commitment Change

This change to SLC 16.9-7, Standby Shutdown System imposed a lower allowable limit for "total accumulative leakage (seal leakoff, unidentified leakage and identified)". The Remedial Action change in the total accumulative leakage limit from 26 gpm to 20 gpm accommodates the predicted increase in seal leakoff rate due to heatup of the reactor coolant pump seal injection water supply temperature following the SSF event (due to SFP heatup), per the evaluation completed by Westinghouse. An administrative change was also implemented to refer to the Inservice Testing Program for the standby makeup pump testing requirements. This change is consistent with the Technical Specification convention of referring to this program rather than citing specific head and flow values in the Technical Specification or the SLC. No USQ exists.

Selected Licensee Commitment Change

This change revised the commitment to perform a pressure test of those portions of the Fuel Oil System (FD) at a test pressure of 110% of the system design pressure to performing the pressure test at system pressure as allowed by ASME code case 498-1.

In a June 22, 1994 letter to the NRC, Duke requested approval to implement ASME code case 498-1 that allows alternate testing methods using VT-2 visual examinations on class III systems at nominal operating pressure. The NRC approved the use of code case 498-1 for use at McGuire in a Safety Evaluation Report dated February 13, 1995. This approval was not subsequently followed by a technical specification change. The requirement for the 110% pressure was removed from the technical specifications as a part of the Improved Technical Specifications program. Because the NRC has previously evaluated and approved the use of code case 498-1 for McGuire, the Selected License Commitment may be changed to reflect this approval and substitute the alternate test method for the 10-year in-service inspection.

Selected Licensee Commitment Change

Selected Licensee Commitment 16.9-1, "Fire Suppression Water System" Bases was revised to provide additional information concerning main fire pump testing requirements. The Bases change documented McGuire's licensing commitment to the 1978 National Fire Protection Code with regards to Main Fire Pump testing. This revision also included reference to additional testing conducted under the scope of SLC testing requirement 16.9.1.6 to gather supplementary data for the purpose of main fire pump performance trending and monitoring.

This change has no effect upon system or component operation. The change is designed to document main fire pump testing requirements along with discussion of additional testing beyond NFPA Code mandates which will be utilized to trend and monitor pump performance. The Fire Protection System is designed to meet the requirements of the 1978 National Fire Protection Association (NFPA) codes and later years where practical and applicable.

This SLC change has no effect on "Remedial Actions" or "Testing Requirements" currently contained within SLC 16.9-1. This revision only clarifies NFPA Code commitments with regard to main fire pump testing. No USQ exists.

Selected Licensee Commitment Change

SLC 16.9.2.3.2 , "Cask Drop Evaluation", was revised to include the TN-32 Cask in the list of spent fuel casks evaluated and to include the updated methodology for evaluating cask drop accidents. SLC 16.15-3.9.7 was revised to include the following statement in the Bases: "The requirement for following the load path shown in Figure 3.9-1 assures that the cask cannot fall into the spent fuel pool." The phrases "truck cask" found within this SLC section were changed to "spent fuel cask". The revision also corrects errors made during Technical Specification 3.9.7 relocation to the SLC.

In support of the ISFSI project, the calculation for the postulated cask-drop accident was updated to include the RN-32 cask. The cask drop evaluation concluded that should the cask break free as postulated during cask handling, the cask would not enter or fall into the spent fuel pool, when traveling along a prescribed path. The prescribed path assures that the cask's center of gravity will be located over the spent fuel cask pit such that any tipping of a dropped cask would be within the confines of the cask pit.

The prescribed path also precludes the cask from passing over or near safety related equipment and restricts the cask to areas designed to accommodate a dropped cask with only negligible damage to the structural concrete. The changes do not directly affect cask handling equipment or procedures. No USQ exists.

Selected Licensee Commitment Change

This USQ evaluation documented changes to SLC Sections 16.9.9, 16.9.10, 16.9.11, 16.9-12 and 16.9.13 which pertain to operating and shutdown boration capabilities. These SLC sections were moved from the McGuire Technical Specifications to the SLC following conversion to the Improved Technical Specifications (ITS).

SLC 16.9.9 and 16.9.10 (Boration Systems - Operating) clarify the constituents of an operable boration flow path in SLC 16.9.9. The commitment is an enhancement intended to prevent crediting the same pump twice. The specific value of shutdown margin in Remedial Action B.2 was replaced with a reference to Technical Specification Section 3.1.1. This change is editorial in nature. The deletion of SLC 16.9.10 is inconsequential since the requirements already exist in SLC 16.9.9, Technical Specification 3.1.1 and Technical Specification 5.5.8. The revised requirements of SLC 16.9.9 are equivalent to those previously specified in SLCs 16.9.9 and 16.9.10. These changes do not affect the functional requirements for boration capability and do not constitute an unreviewed safety question.

SLC 16.9.11 changed the specific value of shutdown margin in Remedial Action B.2 with a reference to Technical Specification 3.1.1. This change is editorial in nature.

Changes to SLC 16.9.12 and 16.9.13 (Boration Systems - Shutdown) made provisions in SLC 16.9.12 to permit substitution of a safety injection pump for a charging pump in Modes 5 and 6 when seal injection is not required. The rate of boration from the refueling water storage tank (RWST) utilizing a safety injection pump will be higher than that of a centrifugal charging pump over the range of system pressures allowed in Modes 5 and 6 per Technical Specification 3.4.3. Use of the Safety Injection (NI) System for boration is equally as reliable as the Chemical and Volume Control (NV) system as it is designed, constructed and tested to comparable requirements and is supplied power from the emergency diesel generator. The safety injection pump must be aligned to the reactor coolant system cold legs to be considered as a credible and equivalent substitute (that is, without performing detailed analysis).

SLC 16.9.13 commitments were deleted and incorporated into the boration flow path requirements of SLC 16.9.12. Deletion of this SLC is inconsequential since the requirements already exist in SLC 16.9.12, Technical Specifications 3.1.1 and Technical Specification 5.5.8. The Bases in SLC 16.9.13 regarding the range of reactor coolant system temperature variance which constitutes a positive reactivity change was incorporated into SLC 16.9.12 Bases . The revised requirements of SLC 16.9.12 are equivalent to those previously specified in SLCs 16.9.12 and 16.9.13. These changes to not constitute an unreviewed safety question because the functional requirements for boration capability have not been changed. No USQ exists.

Calculation MCC-1503.13-00-0734

Duke Energy has been approved to perform reload analysis on McGuire using the methodology of BAW-10186-A for Mark BW 17 fuel design. Along with this methodology change Duke is changing fuel vendors from Frametome-Cogema to Westinghouse and will use the Westinghouse Robust Fuel Assembly (RFA) in future core designs.

The use of BAW-10186P-A methodology has been reviewed and approved by the NRC. No 10 CFR 50.59 review is performed on this aspect of the changes. Core design changes as a result of this methodology is reflected in reload design documents along with changes as a result of using Westinghouse RFA fuel. The affect of these changes on dose consequence is reviewed in this evaluation.

The dimensional changes noted in the reload document which incorporates RFA assemblies and increased burnup does not significantly impact dose consequence analysis except as incorporated in the safety analysis outputs. The final results are not changed as a result of these changes. The changes do not increase the amount of fuel failure assumed as a consequence of any accident analyzed in the UFSAR beyond the current bounding values assumed. The maximum rod burn up increased from 60 GWD/Mtu to 62 GWD/Mtu. No USQ exists.

Calculation MCC-1503.13-00-0672 Revision 3

Due to the degradation of the Boraflex panels, a new set of burnup versus enrichment limits for Region 1 are established to ensure Keff remains less than or equal to 0.95. These new limits are more restrictive than the current limits specified in Tables 3.7.15-1 and 3.7.15-2 of the Technical Specifications. These new limits are to be implemented by changes to Procedures OP/0/A/6550, Internal Transfer of Fuel Assemblies and PT/0/A/4150/37, Total Core Unloading.

The function of the spent fuel pool storage racks is to provide for safe storage of spent fuel assemblies in a flooded pool, while preventing criticality. To accomplish this safety function, the Keff in the Spent Fuel Pool, including all uncertainties, is to be less than or equal to 0.95, even if unborated water is used to fill the pool. The new limits will continue to ensure that Keff remains less than or equal to 0.95 with the anticipated degradation of the Boraflex panels and the SFP filled with unborated water. The compensatory actions and procedure changes affect no design criteria or safety functions of any SSC. The compensatory actions and procedure changes affect no design criteria or safety functions or any SSC. No USQ exists.

Calculation MCC-1553.12-00-0015

This calculation documents the USQ evaluation for the use of the (Westinghouse Robust Fuel Assemblies) RG+FA fuel design and considers only the criticality implications of the new fuel design outside of the reactor.

The Westinghouse RFA fuel design is a new product that boasts several advanced features, including annular axial blankets, Integral Burnable Absorbers, ZIRLO cladding, and Intermediate Flow Mixing Grids. From a criticality modeling standpoint, the only difference between the RFA and the Westinghouse Standard "STD" design is the fuel density: the RFA is modeled with 95.5% TD fuel, while the STD design is at 95% TD.

The only accidents related to the criticality analysis outside the reactor are misloading a fuel assembly in the spent fuel pool and loss of spent fuel pool cooling. The RFA fuel will be identified in the same manner as other fuel designs and its movement will be controlled by the same procedures as previous fuel designs. The loss of spent fuel pool cooling is not affected by the type of fuel stored in the pool. The criticality analysis has no bearing on the likelihood of occurrence of one of these previously analyzed accidents. Therefore, these changes do not increase the probability of occurrence of an accident previously evaluated in the SAR.

The use of RFA fuel in areas outside the reactor will not affect any scenarios that could lead to a malfunction of equipment important to safety. The RFA design is designed to reduce or eliminate the possibility of certain types of fuel failures. The fuel handling interfaces for RFA fuel are designed to be compatible with current designs and existing equipment, so the probability of a malfunction of any equipment related to fuel handling will not increase. The design changes being implemented for the new RFA fuel design were developed to reduce the possibility of known failures (e.g. high burnup fuel skeleton for incomplete rod insertion). The probability of other malfunctions not addressed by any of the design changes for RFA fuel remains the same. The use of RFA fuel, as it relates to criticality control outside the reactor, does not increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the SAR.

This safety evaluation is limited to the impacts on the criticality analyses with RFA fuel outside the reactor. The accident scenarios related to this design change that could have an impact on the criticality outside the reactor are misloading a fuel assembly in the spent fuel pool and loss of spent fuel pool cooling, and the consequences of these accidents do not change with the introduction of RFA fuel. A heavy load drop onto the spent fuel pool could rupture fuel rods and lead to dose consequences. However, the dose consequences of such an accident are outside the scope of this evaluation. Therefore, the consequences of the accident scenarios previously evaluated in the SAR, as related to fuel criticality, are not affected by the use of RFA fuel at McGuire.

No new credible failure modes are introduced as a result of implementing the RFA fuel design. Once the fuel is outside the reactor, the only failure modes that could result in release of fission products are via direct contact with the fuel assembly that results in rupture of the fuel cladding. The only scenarios that could lead to such fuel damage are dropping or projecting something on or into the assembly. These accidents are already evaluated in the SAR under the fuel handling accident and tornado missile impact evaluations for the spent fuel pool. Therefore, because no new failure modes are introduced by the use of RFA fuel in areas outside the reactor, these changes do not increase the consequences of a malfunction of equipment important to safety evaluated in the SAR.

The implementation of RFA fuel does not introduce any new failure modes as compared with previous fuel designs. Examples of new failure modes that might affect the criticality are loss of geometry of the fuel assembly or damage to the fuel racks caused by the new fuel design. The RFA assembly is actually less susceptible to significant grid damage that could lead to loss of geometry control through the use of several design features aimed at reducing fretting failures. Given that the RFA assembly is evaluated to be compatible with the reactor internals, it would not increase the probability of damaging the fuel storage racks, given the tighter tolerances within which the fuel must operator in the reactor. Significant, geometry-altering damage to the fuel storage racks is not considered credible. Implementing the RFA design at McGuire does not create the possibility for an accident of a different type from any previously evaluated in the SAR.

As discussed above, possible new failure modes are related to damage of components that would alter the geometry of the fuel in the spent fuel pool. Such accidents are still not considered credible due to the capability of the grids to maintain the fuel rods in the proper geometry and the compatibility of the new fuel with existing handling and storage equipment. Because no new credible failure modes are introduced, use of the RFA fuel design does not create the possibility for a different type of malfunction of equipment important to safety from any evaluated previously in the SAR. No USQ exists.

MCC-1553.26-00-0278

This calculation documents the USQ evaluation of the Mk-BW lumped burnable poison (LBP) rod design changes applicable to McGuire and Catawba. This evaluation includes the following design changes:

1. change from the solid ziroconium spacer which is used to position the absorber in the correct location to a tubular zirconium spacer, and
2. the new drill out depths of the upper and lower end caps to facilitate welding and x-ray inspection on the fuel rod production line

The design and text changes evaluated are for BP rods with new tubular spacers and the addition of end caps that have been drilled out. These changes only impact the interior of the BP rods and their descriptions in the UFSAR. In terms of function and compatibility, the BPs that incorporate these design changes are essentially identical and are considered an equivalent component to the old BPs. No functional characteristics have been adversely affected as the BPs will continue to perform as they are described in the SAR. There are two accident scenarios that involve the BPs, the core misloading and the reactivity insertion by BP ejection. These accident scenarios are not adversely affected by either the design or text changes. No USQ exists.

MCC-1553.26-00-0288

This calculation documents the analysis and safety review for the update of UFSAR Sections 4.1, 4.2, and 4.4 to incorporate RFA design not justified by an approved topical or by NEI 98-03. The parameters changes are consistent with those used in the generic mechanical and thermal-hydraulic analyses performed per NRC approved methods (i.e., DPC-NE-2009P-A). Therefore, these changes do no adversely affect the mechanical and thermal performance of the fuel or its reliability.

In addition, the proposed changes do not adversely impact the reactivity, doppler coefficients, moderator temperature coefficients evaluated by reload analyses using SIMULATE-3, an NRC approved code. Furthermore, these parameters are controlled by the individual mechanical and thermal-hydraulic analyses. The license basis values are listed in the Improved Technical Specification, Table 3.4.1-1. Therefore, changes to this table do not adversely affect the mechanical or thermal performance of the fuel and or its reliability.

Calculation MCC-1553.26-00-0288 also evaluates the following changes:

1. The fuel rod length for the M2C14 RFA fuel is slightly longer than used for C2C11 evaluated in calculation CNC-1553.26-00-0254, Rev. 1. This change in fuel rod length has been incorporated into the mechanical analyses performed using NRC approved methods discussed in DPC-NE-2009P-A. This change has no affect on the thermal-hydraulic analyses. Therefore, this change does not adversely affect the mechanical or thermal-hydraulic performance of the fuel.

2. The top nozzle screw material was changed from Inconel 600 to Inconel 718 to improve the reliability and performance of these screws. This change improves the material strength and reliability of the screw. Therefore, there are no adverse affects on the reliability or the mechanical performance of the screws or the robust fuel assembly. The new screw is fully compatible with McGuire's in-reactor operating conditions. The physical size and installed configuration in the top nozzle of the new screw is identical to the previous design. There is no adverse impact on fuel handling operations or physical compatibility. Therefore, this design change does not introduce new failure mechanisms/equipment

malfunctions or increase the probability of an accident previously evaluated in the SAR (i.e., fuel handling accident).

3. The increase in the core bypass flow is accounted for in the reload cycle specific thermal-hydraulic analyses. These ensure DNB does not occur. Therefore, there is no increase in the probability of fuel rod failure due DNB. Similarly, there is no increase in the consequences of an accident previously evaluated in the SAR.

4. The WABA finger connection via a crimp nut is equivalent to the lock-welded nut in performance and reliability. There is no adverse impact to the capability of the WABAs to perform their safety-related function. The use of a crimp nut is not an accident initiator and does not introduce any new failure modes.

These updates do not adversely affect the fuel or reactivity control component reliability or performance or the operation of the plant. Therefore, the MNS UFSAR updates do not increase the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR. The updates specified in this calculation reflect the robust fuel rod and assembly design or assumptions that are consistent with the licensing basis specified in the MNS UFSAR and Improved Technical Specifications.

The MNS UFSAR updates do not adversely affect the mechanical or thermal performance of the fuel and or its reliability. Therefore, the margin of safety is not reduced. No USQ exists.

DPC-1553.26-00-0093, Rev. 2

This calculation contains a generic 10CFR 50.59 analysis and safety review for the coarse and fine mesh bottom filter nozzles at McGuire and Catawba.

Normal Operation

During normal operation, the particulate size transported from the Reactor Coolant (NC) system filters to the core is smaller than then bottom nozzle holes per FCF calculation 32-5003654-00, Mk-BW Fine Mesh Fuel Filter LOCA Impact Evaluation. Therefore, blockage of the bottom nozzles is not expected. Therefore, no new failure modes have been introduced and the probability of an accident that is different than previously evaluated in the SAR is not increased.

Post LOCA Conditions

Blockage of fuel assembly channels is not credible for a cold leg break LOCA per the FCF (32-5003654-00) and Westinghouse document, Effect of Debris During Post Accident Conditions On Core with Debris Resistant Bottom Nozzles. The FCF calculation states that flow velocities are insufficient to entrain or lift debris into the core inlet. Thus, there will be no significant debris transport to the core inlet following a cold leg break LOCA. Only small amounts of any semi-neutral density debris, if present, could be carried to the assembly entrances following a cold break LOCA. There are insufficient quantities of buoyant material in containment that could be transported to the core inlet following a cold break LOCA. Therefore, core blockage following a cold leg LOCA is not credible.

In addition, the Westinghouse document independently verified that there are insufficient velocities to transport debris into the core during cold leg break LOCA conditions. Thus, the normal flow path of coolant up through the fuel assemblies following a LOCA is undisturbed. Therefore, the operability and

the capability of the Emergency Core Cooling System (ECCS) to perform its safety related function (i.e., provide adequate core cooling following a LOCA) are not adversely impacted.

The FCF calculation verified that the adequate core cooling can be provided by flow redistribution following a cold break LOCA even if there were limited blockage by debris such as paint chips. Therefore, the capability of the ECCS to provide adequate core cooling following a cold leg break LOCA is not adversely impacted by the modification to the bottom nozzle.

The FCF calculation and Westinghouse document state that for the hot leg break LOCA, the fluid velocities in the lower plenum are only marginally adequate to transport and sustain debris at the core inlet. The FCF calculation indicates that debris greater than 7.2 mils thick will not be transported to the core inlet. Based on coating specifications, debris sources smaller than 11 mils (laminated) and 5-8 mils delaminated) are not credible. Therefore, it is unlikely that any significant blockage of the bottom nozzle would occur during hot leg break conditions and significant blockage would be self-limiting. The FCF calculation shows that, in the unlikely event that the core inlet were completely blocked, more than adequate flow is always available via alternate flow paths through the core basket.

The FCF calculation demonstrates that the operability of the Emergency Core Cooling (ECCS) System is not adversely impacted. The calculation verifies that adequate core cooling can be provided via alternate flow paths following a hot leg break LOCA even if the core were completely blocked by debris from the containment sump. Therefore, the capability of the ECCS to perform its safety related function is not adversely affected.

In addition, the FCF calculation states that the grid/fuel pin tolerances, as small as zero, are limiting. Normal spacer grids have zero gap at all grid bumper locations. These tolerances existed in previous and current fuel designs at McGuire and Catawba before the introduction of the fine mesh bottom filter nozzle. Therefore, the fine mesh bottom filter nozzle does not increase the likelihood of fuel assembly blockage.

Since there is no potential for loss of cooling capability during hot and cold leg break LOCA conditions and the likelihood of fuel assembly blockage is not increased, there are no new failure modes introduced. The fuel assembly fine mesh bottom filter size if not limiting in regards to particulate passage tolerance. Grid/fuel pin tolerances, as small as zero, for McGuire and Catawba units, were limiting before the introduction of the fine mesh fuel filter. Therefore, the grid/fuel pin tolerances remain limiting. The FCF calculation concluded that the Mark-BW fuel assemblies are no more or no less subject to trapping debris as a result of incorporating the debris filter bottom nozzle design described in revision 0. Therefore, there is no LOCA impact.

There is no increase in the probability of a malfunction of equipment important to safety previously evaluated in the SAR. Similarly, there is no increase in the probability of a malfunction different than previously described in the SAR. It is concluded that this modification does not adversely affect the capability of the ECCS to mitigate the consequences of a design basis accident. There is no increase in the consequences from an accident previously evaluated in the SAR.

The FCF calculation verifies that the operability and capability of the ECCS system to perform its safety function (i.e. provide adequate core cooling following a LOCA) is not adversely impacted by this modification. Therefore, the capabilities of the ECCS system to mitigate the consequences of a LOCA are not adversely impacted. Adequate core cooling is provided by the ECCS system following a LOCA.

By ensuring adequate core cooling via the ECCS system 1) the number of failed rods is minimized and 2) all 50.46 criteria is met. Therefore, this modification does not increase the dose consequences from an accident previously evaluated in the SAR.

Margin of Safety

ITS Section 3.5.2 and B 3.5.2, Emergency Core Cooling System and Bases: These specifications specify that two trains of ECCS must be operable. As discussed above, this modification does not adversely affect the operability of the ECCS or its capability to perform its safety related function. Therefore, the margin of safety is not reduced.

ITS Section 3.4.16 and B 3.4.16, RCS Specific Activity & Bases: These specifications specify the acceptable limits for the specific activity in the RCS during normal operation. This modification does not adversely affect the mechanical or thermal performance of the fuel or its reliability during normal operation. Therefore, the margin of safety is not reduced.

ITS Section 3.4.1 & B 3.4.1, Pressure, Temperature and Flow Departure from Nucleate Boiling Limits & Bases: This specification specifies the DNB parameters that ensure that the fuel does not undergo departure from nucleate boiling. Adherence to these parameters ensures the appropriate thermal performance of fuel operation. As discussed above, this modification does not adversely affect the mechanical or thermal performance of the fuel or its reliability. The margin of safety is not reduced. No USQ exists.

Technical Specification Bases Change

This Improved Technical Specification (ITS) Bases change adds additional details regarding the accepted operational method to assure compliance with ITS LCO 3.0.3. This information, previously contained in a Technical Specification interpretation, was deleted during ITS implementation. The change to Bases page B 3.0.3 was designed by Operations personnel during ITS training in order to assure compliance with ITS 3.0.3 requirements.

LCO 3.0.3 establishes the actions that must be taken when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical

generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

The proposed change provides additional guidelines for complying with the shutdown requirements of LCO 3.0.3. The information emphasizes that power reduction must begin within one hour or less from entering LCO 3.0.3. Operational guidelines are also provided regarding the rate at which reactor power is to be reduced. This additional information will ensure that the unit shutdown will be accomplished within the time limits of LCO 3.0.3 and at the cooldown rate that is well within unit capability and will not compromise plant safety.

LCO 3.0.3 is reserved to address those unit conditions which are unanalyzed and which occur rarely during operation. The load reductions proposed by this change are well within those listed in UFSAR Chapter 15, Condition I-3.c (ramp changes up to +/- 5%). Therefore, this change does not increase the probability of an accident or create the possibility for an accident of a different type previously evaluated in the SAR.

As stated above, In LCO 3.0.3 cases, the malfunction of equipment has already occurred and an unanalyzed unit condition has been identified, i.e. an applicable ITS condition does not exist. The additional LCO 3.0.3 details provided by this change will assure that the units is brought to a mode where the LCO does not apply or that the unit is shutdown completely. Unit shutdown will typically increase operating margins on equipment and will result in safer operation. Thus, an increase in the probability of a malfunction of equipment important to safety does not exist.

The subject LCO 3.0.3 required actions and the additional details provided by this change are designed to mitigate conditions which are unanalyzed, i.e. applicable TS conditions do not exist. Plant shutdown in an orderly manner is an operational option in pursuit of a safer operating mode. These operation shutdown details are not initiators of equipment malfunctions. Thus, the consequences of a malfunction of equipment important to safety or the possibility for a different type of malfunction previously analyzed within the SAR does not exist.

At lower operating modes, additional safety and operating margin is available since the reactor coolant system contains less available energy at lower pressure and temperature. Thus, the margin of safety is increased by returning the unit to an analyzed condition and to the safety margins as described in the SAR. No USQ exists.

Technical Specification Bases Change (1)

The Improved Technical Specification (ITS) Bases seem to imply that an Auxiliary Building Filtered Ventilation Exhaust System (ABFVES) consists of exhaust fans 1A and 1B, or exhaust fans 2A and 2B. This implication seems to be inconsistent with other Technical Specifications where it is normally required that two trains of a system are operable, i.e. train A and train B. Based on this implication, both ABFVES systems would have to be declared inoperable and Condition B of ITS 3.7.11 would be entered for both Units when a diesel generator that is aligned to these fans becomes inoperable because this diesel supplies one fan from Unit 1 and one fan from Unit 2. Condition B requires one ABFVES to be returned

to OPERABLE status within 24 hours. ITS 3.8.1, A.C. Power Sources, requires the inoperable diesel generator to be returned to operable status within 72 hours. However, because of the ABFVES implication described above, the diesel generator would need to be returned to operable status within 24 hours to avoid both Units shutdown as required by ITS 3.7.11, Condition C if Condition B is not met.

ITS 3.7.11 LCO Bases were changed to specify that an ABFVES is considered OPERABLE when its associated two fans (1A and 1B, or 2A and 2B, or 1A and 2A, or 1B and 2B, or any two fan combination provided that SR 3.7.1.4 has been performed for that combination prior to taking credit for that combination), filters, ductwork, valves, and dampers are operable. ITS SR 3.7.11.4 Bases are revised to specify that this SR is required to be performed for each fan combination described in the LCO Bases. With this change, an inoperable diesel generator that is aligned to these fans would make only one ABFVES inoperable and Condition A of ITS 3.7.11 would be entered which requires the inoperable ABFVES to be returned to operable status within 7 days. In this case, the inoperable diesel generator would have to be returned to operable status within 72 hours.

This change does not alter the intent of the Technical Specifications. No USQ exists.

Technical Specification Bases Change (3)

This change revised the bases to add a clarification for increasing hydrogen recombiner power during testing. The hydrogen recombiner functional test ensures the recombiners are operational and can attain and sustain the temperature necessary for hydrogen recombination. Verification that the recombiner operates at a power level $\geq$60 kW is part of the acceptance criteria for the test. This editorial change provides instructions to increase the recombiner power to maximum power while "not exceeding the maximum rated power." This change clarified that maximum power is not to exceed manufacturer limits. Addition of this clarification does not impede the ability to satisfy nor effect the acceptance criteria for the functional test of the recombiner. No USQ exists.

Technical Specification Bases Change (5)

Bases 3.4.14 were changed to read "The PIVs are listed in the UFSAR, Table 5-50 (Ref. 6)". The reactor coolant system pressure isolation valves which were listed in McGuire previous Technical Specifications were not in the Improved Technical Specifications (ITS), and were listed in UFSAR Table 5-50. This is the same group of valves that was in previous Technical Specification Table 3.4-1. Table 5-5 is "Active and Inactive Valves in the Reactor Coolant System Boundary", and includes several valves that are not in the scope of leak rate tested reactor coolant system pressure isolation valves. The scope of ITS Bases 3.4.14 applies to the Table 5-50 list of valves ("any two normally closed valves in series with the reactor coolant pressure boundary (RCPB), which separate the high pressure reactor coolant system form an attached low pressure system"). This is only an editorial change to the ITS Bases. No USQ exists.

Technical Specification Bases Change (6)

This change deleted Technical Specification Bases 3.3.2 Function 8.a, Item 4, ESFAS Interlock P-4, Reactor Trip. The subject ITS Bases lists the functions of the P-4 interlock to include:

1. Trip the main turbine,
2. Isolate MFW with coincident low Tavg,
3. Prevent reactuation of SI after a manual reset of SI,
4. Transfer the steam dump from the load rejection controller to the unit trip controller, and
5. Prevent opening of the MFW isolation valves if they were closed on SI or SG water level-high high.

The steam dump P-4 contact is unique from the turbine trip, SI and MFW isolation P-4 contacts in that the auxiliary switch on the RTB picks up auxiliary relays in the RTB cabinets. These auxiliary relays have contacts which, in turn, provide the P-4 signal to the steam dump circuitry. The old Technical Specification Bases for the P-4 interlock included all functions listed above with the exception of function 4. The Improved Technical Specifications (ITS) Bases for the P-4 interlock specified that none of the listed functions serves a mitigation function in the unit licensing basis safety analyses and only the turbine trip function is explicitly assumed since it is an immediate consequence of the reactor trip function. The ITS Bases also specifies that neither turbine trip nor any of the other four P-4 functions is required to show that the unit licensing basis safety analysis acceptance criteria are not exceeded. At McGuire, the turbine trip and Safety Injection (SI) reset functions have been tested as a result of recommendations from Westinghouse. Other P-4 functions have been tested as a result of good engineering practice. In these tests, all P-4 auxiliary switch contacts are measured and verified changing state accordingly. This change does not exclude the steam dump P-4 contact from being tested and reflects the intended conversion from the old Technical Specifications to the Improved Technical Specifications. This change does not involve any safety or licensing issues, and does not revise any regulatory commitments. No USQ exists.

Technical Specification Bases Change

The activity evaluated involves a change to Technical Specification Bases 3.5.2, "ECCS - Operating." TS 3.5.2 Bases, Background section, specifies that after approximately 7 hours, part of the ECCS flow is shifted to the hot leg recirculation phase to provide a backflush which, for a cold leg break, would reduce the boiling in the top of the core and prevent excessive boron concentration. The "7 hours" in the subject Bases was changed to "6 hours". This change was made to the McGuire UFSAR, Section 6.3.3.10, "Boron Precipitation Evaluation," in the 1998 update, at which time the Improved Technical Specifications (ITS) had not been implemented. As a result, the ITS Bases could not be changed in conjunction with the UFSAR change. This change is considered administrative in nature since it reflects a previous change to the UFSAR which was completed via a 10CFR50.59 evaluation. No USQ exists.

Technical Specification Bases Change

The activity evaluated involves a change to Technical Specification Bases 3.6.5, "Containment Air Temperature." TS 3.6.5 Bases, Applicable Safety Analysis section, specifies that the limiting DBA for establishing the maximum peak containment internal pressure is a LOCA, and that the temperature lower limits of 70 degrees F for the upper compartment and 100 degrees F for the lower compartment are assumed in this analysis. The "70 degrees F" in the subject Bases was changed to "75 degrees F." This change was completed to be consistent with UFSAR Section 6.2.1.1.3.1, LOCA. TS 3.6.5 references UFSAR Section 6.2. Item 8 states that the initial conditions in containment area temperature of 100 degrees F in the lower and dead-ended volumes and a temperature of 75 degrees F in the upper volume. The "75 degrees F" in this UFSAR section has been the same since Improved Technical Specification (ITS) development, and is consistent with the assumption in the current calculations for Long Term Containment Pressure and peak Containment Pressure Transient Analyses, as referenced in the Safety Analysis Inputs Manual. This change is considered editorial in nature since it reflects the current UFSAR and Safety Analysis calculations. No USQ exists.

Technical Specification Bases Change

This change clarified the BACKGROUND, APPLICABLE SAFETY ANALYSIS, LCO and APPLICABILITY areas of the BASES for the Spent Fuel Pool Ventilation System (FHVES) by adding the phrase "in filtered mode" to these areas. The change provides additional emphasis to the fact that the

FHVES must be "in filtered mode" in order to be OPERABLE as required during movement of irradiation fuel.

The FHVES filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident. The FHVES, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FHVES is composed of both a supply and exhaust section. The supply portion consists of a 100% capacity air handling unit containment water cooling coils, hot water heating coils, roughing filters, and associated ductwork and dampers. The exhaust portion consists of a 100% capacity filter train, two 50% capacity exhaust fans, and associated ductwork and dampers. The exhaust fans were originally each 100% capacity but have been modified to 50% capacity fans in order to meet the required intake and exhaust flowrate. The filter train contains a prefilter, high efficiency particulate air (HEPA) filter, and carbon filters of the gasketless design. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and carbon absorbers. The system is required to be in operation IN FILTERED MODE any time irradiated fuel is being moved in the fuel handling building.

The FHVES is discussed in UFSAR, Section 9.4 and 15.7 and MNS DBD MCS-1577.VF-00-0001. FHVES may be used for normal, as well as post accident, atmospheric cleanup functions. Credit is taken in the offsite dose calculation for the FHVES system being operable during postulated fuel handling accident and maintaining offsite dose within the limits of 10CFR20 and 10CFR100.

The subject change provides additional clarification as to the operational requirements of the FHVES. The additional information emphasizes that FHVES operation in the filtered mode during fuel movement is required to meet the safety analysis assumptions. This change does not alter the design function of the FHVES. No USQ exists.

SDQA Plan for SMARGINS, Version 7 (SDQA-30188-NGO.R00)

The software addressed in this evaluation calculates design margin to thermal limits for LOCA, F_Q, DNB and CFM analyses. This change involves an upgrade to the Super-MARGINS (SMARGINS) software. The current software, SMARGINS version 6 is being replaced by version 7. The methodology for calculating margin to thermal limits has not changed. The new software incorporates modification to support analyses outlined in DPC-NE-2009A for McGuire and Catawba, to conditionally reduce the Oconee LOCA limit based on axial power shapes, and other minor enhancements. Part of the modification for DPC-NE-2009A is to calculate the penalty factor used in Improved Technical Specification surveillance requirement 3.2.1.3 and 3.2.2.2.

SMARG07 was certified per Duke Power's directive for software certification (Nuclear System Directive-800) and verified to yield the same results as SMARG06, excepting the new modifications. The modifications were verified and are in compliance with Technical Specifications and approved methods.

This change involves no material changes to the plant. The SMARGINS software and resident workstation are not part of any SSC important to safety and do not directly affect any SSCs. The SMARGINS software is not installed at the plant, but rather on workstations in the Nuclear General Office. A safety significant function performed by this system is to generate data to evaluate Improved Technical Specifications 3.2.1 and 3.2.2. As indicated, the new software produces the same analytical results as the replaced software. Per DPC-NE-2009PA, the new penalty factors used in SR 3.2.1.3 and

SR 3.2.2.2 accurately reflects the expected change in margin over the surveillance window and will be greater than or equal to the historic value of 1.02.

The assurance of fuel integrity limits associated with the referenced Improved Technical Specifications are not compromised. This change does not impact any plant parameters, safety limits or setpoints that potentially affect the fission product barriers. No USQ exists.

Procedures

AP/1/A/5500/13
AP/2/A/5500/13

These procedures were deleted and incorporated into the existing Abnormal Procedure (AP) for Emergency Boration. The normal boration paths are utilized during normal operation for periodic system make-up and acid shots. These boration flow paths are further relied upon for mitigation of various dilution/rod control events, inadequate shutdown margin, uncontrolled steam/Tavg changes, etc.

The overall intent of the procedures are unchanged, they were merely combined within the Emergency Boration AP as a result of many identical/similar steps. The overall scope of each procedure is effectively captured by the new revision to the Emergency Boration procedure. The combined procedures further eliminate confusion regarding which procedure is applicable (similar entry symptoms). There are no impacts to design basis, safety or operation of the plant. No USQ exists.

EP/1/A/5000/ES-1.3
EP/2/A/5000/ES-1.3
EP/1/A/5000/FR-Z.1
EP/2/A/5000/FR-Z.1

These procedures, Transfer to Cold Leg Recirculation and Response to High Containment Pressure, which are the procedures for establishing Residual Heat Removal (ND) flow to the auxiliary reactor building spray to align the ND pumps at 45 minutes versus 50 minutes. The criteria for ND swapover to the auxiliary reactor building spray header is to delay the swapover until latent and decay heat levels in the Reactor Coolant (NC) system are low enough to safety allow reduction in coolant flow, assure that ND suction is from the containment sump and to complete swapover to Containment Spray (NS) prior to Ice bed meltout to prevent a containment pressure spike. At 45 minutes, the decay heat load is less than 1.5% of full power per DPC-1552.08-00-0141. From DPC-1552.08-00-0161, the Emergency Core Cooling System (ECCS) water temperature returning to the reactor vessel at 45 minutes is <150 degrees F. The ECCs flowrate to make up for the core boiloff at 45 minutes is 342 gpm. From DPC-1552.08-00-0136, 730 gpm are available from one Chemical and Volume Control (NV) plus one Safety Injection (NI) pump during cold leg recirculation mode. Since 730 gpm is greater than 342 gpm, core cooling is assured if ND auxiliary spray flow is aligned as early as 45 minutes after the limiting LOCA. In fact, any ECCS flow in excess of 342 gpm will only be spilled on the floor. It is therefore obvious that stopping ND flow at this time has no significant effect on the accident analysis. Based on this evaluation, the referenced procedures were revised to require the operator to begin aligning ND spray flow after 45 minutes. This guidance supports the licensing basis analysis. The UFSAR description of the swapover step in procedures does not clearly indicate any tolerance for taking this action prior to 50 minutes but does indicate that the action could occur later than 50 minutes. There is sufficient conservatism in the assumptions for ice bed melt out to assure that a delay beyond 50 minutes does not adversely impact containment pressure control. In addition, plant conditions which increase ice melt out will cause the swapover function to occur sooner and plant conditions which tend to delay swapover also result in slower ice bed melting. The change in description of the swapover action from "at 50 minutes" or "after 50 minutes" to "approximately 50 minutes" is consistent with the consideration of this action in safety analysis and more clearly describes the actual use of the procedures in practice. There are no changes to safety analysis margins, setpoints or acceptance criteria associated with this change. The procedure changes provide more operator latitude in taking appropriate actions during accident mitigation and do not adversely impact any part of the accident mitigation described in the UFSAR. No USQ exists.

MP/0/A/7650/031

This procedure, "Controlling Procedure for Troubleshooting and Corrective Maintenance," will be used to re-roll a single tube end in the sub-cooler section of the B YC Chiller Condenser. This work does not affect the heat transfer capability of the subject tube, nor will does it affect the pressure retaining capability of the tube/tubesheet interface pressure boundary. This activity does not change any actions described in the SAR and does not alter any assumptions made in evaluating the radiological consequences of postulated accidents. No new failure modes or operating characteristics for equipment/ systems important to safety are affected. No USQ exists.

MP/1/A/7650/060
MP/2/A/7650/116

These procedures, Operation of Polar Crane in Unit 1 and Unit 2 Upper Containment were changed to: 1) delete the need to write within the procedure whom and when Radiation Protection was notified, 2) clarify the means by which lift supervisor and crane operator stay in contact with each other and 3) commitment change to allow for the handling of heavy loads inside containment during Modes 1 and 2.

The first two changes identified do not affect the handling and control of heavy loads or the operation of the polar crane. The changes are administrative in nature, in that they involve minor revisions in certain communication processes. In 1982, the use of the polar crane to move heavy loads in containment during Modes 1 through 4 was not anticipated. The regulatory provisions, criteria and guidelines for the handling and control of heavy loads do not prohibit heavy load lifts in the containment during power operation. For the movement of heavy loads within the containment, one of the following three criteria need to be satisfied: 1) use of a single failure proof crane, or 2) rapid containment isolation, or 3) heavy load drop analysis performed. A calculation was performed to ensure that a drop of the pressurizer hatch plug on the pressurizer enclosure roof or operating floor and a drop of the polar crane load block on to the operating floor would not damage any equipment, components, or system necessary for safe shutdown. The methodology employed by this calculation is in accordance with the analytical process described in the December 20, 1980 NRC Letter and NUREG-0612. Based on this calculation the operating floor and the pressurizer enclosure roof will not allow the drop of the pressurizer hatch plug or the polar crane load block to damage safe shutdown equipment.

With the removal of the pressurizer hatch plug during Modes 1 through 4, Condition A of LCO 3.6.14 is entered. This condition requires that the plug be re-instated within one hour, or unit shutdown is to commence. No USQ exists.

OP/1/A/6100/003

This procedure is used for power reduction and plant shutdown. McGuire Unit 1 has identified control circuit failures that prevent complete manual insertion of control rods without a reactor trip. This problem has been reviewed and determined not to create a safety concern. The Technical Specification requirements to test rod movement on the rod banks affected by the control failure have been modified to allow operation through the end of the current cycle without further testing. The NRC has reviewed and approved this change. In order to bring the reactor to full shutdown, it is now necessary to open the reactor trip breakers and allow the rods to fall into the reactor instead of the manual insertion method usually used during plant shutdown. This restricted procedure change details the steps taken to shutdown in this manner.

This procedure change will incorporate a reactor trip low power as part of the unit shutdown evolution. The reactor will not be operated outside the normal parameters for shutdown. Therefore, the trip is considered a planned actuation of a reactor protection system and is not a reportable event. All plant systems are operated within their normal design operating parameters during this evolution. No USQ exists.

OP/1/A/6100/SD-9
OP/2/A/6100/SD-9

These are new procedures using TO/2/A/9600/105, Bypass of P12 Interlock for Residual Heat Removal (ND) Initiation Delay as a model. TO/2/A/9600/105 provides the method for bypass of the P-12 interlock and provides a method to use additional condenser steam dump valves for unit cooldown while in procedures OP/1&2/A/6100/SD-4. The P-12 interlock is bypassed in the 7300 cabinets to disable the interlock when appropriate pressure and temperature conditions are met during unit cooldown in mode 4. The condenser steam dumps are controlled using the Steam Pressure Controller before and after the P-12 interlock is bypassed. This procedure reduces the amount of time the ND system is needed to operate during unit cooldown and allows an acceptable cooldown rate using the condenser dump valves at lower temperatures.

The P-12 interlock provides automatic blocking of six of the nine condenser block valves when Reactor Coolant Temperature is below 553 °F and arms the steam dump system on high t-ave when the unit is in modes 1, 2 or 3. This interlock prevents large overcooling transients when load rejection conditions occur. Technical Specification 3.3.2 requires that the interlock be operable during modes 1, 2, and 3. This interlock may be bypassed when the unit is in mode 4 since it is no longer required by Technical Specifications. There is insufficient energy in the secondary system at this point to cause an overcooling accident. The effectiveness of the three remaining condenser dump valves for unit cooldown decreases as reactor coolant system temperature and pressure decrease. This procedure provides for bypassing the P-12 interlock during cooldown below 300 °F. This makes use of the ND system for cooldown in this temperature range less necessary and should reduce dose rates from the ND system during plant outages. The ND system will be operable and ready to respond to any failure of the steam dumps to operate as desired. No USQ exists.

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

These procedure revisions are limited to minor administrative/typo corrections, changing the venting practice for Enclosure 4.1 and adding Enclosure 4.8 to accommodate flush of the letdown header with charging flow during Modes 5,6 or No-Mode. Changes to Enclosure 4.1 serve to ensure adequate venting is performed prior to running a Chemical and Volume Control System (NV) pump. The new flush enclosure, auxiliary letdown and normal charging flows will be reduced/isolated as required to support the flush performance. The flush procedure entails two sequences, the first isolates normal charging loop isolation valve (train for credited boration flow path) an aligns the charging supply to the downstream side of the regenerative heat exchanger to establish reverse flush flow to the "C" crossover. The second flush segment isolates this flow path and aligns a flush flow path through the normal letdown header.

The scope of these procedure changes are minor, except for the new flush segment. The changes to Enclosure 4.1 will serve to ensure adequate venting is performed prior to running an NV pump. The new enclosure accommodates flushing of the normal letdown header during Modes 5, 6 or No-mode, both back through the regenerative heat exchanger to "C" crossover and through the normal letdown flow path. The flush source is the purified charging supply upstream of the regenerative heat exchanger. Charging

and letdown is controlled to maintain adequate pressurizer/volume control tank (VCT) level. The flush alignments maintain acceptable process design conditions and further ensure adequate NV pump suction/mini-flow paths, whereby the probability of safety related equipment malfunction is not increased. The scope of this flush does not create the possibility of an event beyond those previously analyzed. The scope of the flush will not degrade the NV system capability to mitigate a previously analyzed event; therefore, consequences of previously analyzed events would not be increased. The flush is allowed in modes for which the Emergency Core Cooling System (ECCS) operability is not required. Boration flow path operability LTOP restrictions will be adhered to during performance of this flush. No USQ exists.

OP/0/A/6450/011

The Control Area Ventilation/Chilled Water System procedure (Enclosure 4.15) was revised and Enclosure 4.17 was added. The revision to Enclosure 4.15 adds a conditional step which directs Operations to the new Enclosure 4.17, if pump suction and compression tank level conditions required cannot be achieved by draining water from the compression tank. As it's title implies, Enclosure 4.17 direct Operators to vent air from the compression tank, which will allow the required pump suction and compression tank level conditions to be established. The new enclosure also provides guidance for venting the system following system maintenance, as well as adding air to the compression tank, if water level is high.

The Control Room Area Ventilation and Chilled Water System (VC/YC) are designed to maintain the environment within the Control Room, Control Room Area and Switchgear rooms within defined temperature limits to ensure acceptable operation of unit controls, and uninterrupted safe occupancy of the control room during post accident shutdown. The YC system fulfills this design by virtue of redundant trains of chilled water, of which both have elevated compression tanks which collect air circulating within the systems, and protect the system from thermal transients. The compression tanks are located near the suction of the YC pumps, to provide a source of net positive suction head (NPSH).

Each train also has auto-make-up capability, via regulating valves. Failure of the regulating valves to perform their intended function typically results in excessive make-up to the system, which causes high YC pump suction pressure, and eventually relief valve operation. Increased sump suction pressure is of no consequence to system operation. Relief valve operation creates material condition concerns, and dilutes system chemistry.

Regulating valve setpoint adjustments require the excess system fluid to be drained, resulting in reduced compression tank level. However, if excess air has also accumulated in the compression tank, the water level in the tank can drop below the low level alarm setpoint, before the required amount of fluid can be drained.

Accordingly, Enclosure 4.15 for adjusting the regulating valve setpoint needs a conditional step, which provides a branch to procedural guidance for removing air from the tank.

The new enclosure provides guidance to ensure that the compression in the tank level is not lost. The NPSH and thermal surge function of the tank are therefore maintained throughout the activity. Note that to add air to the tank, the auto-make-up function is isolated, as it is post safety injection. Adjusting compression tank level therefore does not create any new operating characteristics for the YC system, nor does it create any new failure modes for the components affected. No USQ exists.

OP/1/A/6800/11
OP/2/A/6800/11
OP/1/A/6800/012

These Nuclear Service Water (RN) Block Tagout Procedures, are intended to align and drain the RN System to support planned maintenance activities during outages. This evolution drains a substantial volume of water into the Groundwater Drainage System (WZ) in a controlled manner.
The flow of RN water into the WZ System is controlled by procedure steps, such that only one sump pump will be required to operate to move this flow out of the Auxiliary Building. A prerequisite step requires a minimum of 5 WZ sump pumps to be operable during the draining evolution.

The purpose of the procedures is to provide a controlled drain path through the WZ in order to perform outage maintenance on the RN system. These activities are required to ensure the continued availability of this vital system during plant operations during the next operating cycle. Use of the WZ system for this purpose does not present any challenges to any safety equipment needed to protect the health and safety of the public by having an adverse affect on dose at the site boundary. Based on the limitations placed within the procedure in relationship to Auxiliary Building Flooding Accidents, no USQ exists.

PT/1/A/4200/001 U
PT/2/A/4200/001 U

These procedures align main steam isolation valves (MSIVs) so that thermal testing can be performed to qualify seat leakage on the main steam drain isolation valves and the main steam isolation inlet drain valves. The main steam line drain isolation valves (MSDIVs) and MSIV inlet drain valves are containment isolation valves that act as the containment boundary for certain design bases accidents to minimize loss of reactor coolant inventory. These valves are not required to be leak tested, per Appendix J requirements, but should be low leakage to perform their intended containment boundary function. These procedures will only allow aligning valves to that thermal monitoring can be performed to qualify the leakage past the seat.

Opening one MSDIV and downstream drain valve has a negligible effect on reactor power. The procedure only allows opening one steam drain at a time. This alignment could occur automatically, during normal system operation, any time a high level is sensed by the level switch in the line. This alignment is not outside the normal operation and design for the drain lines. There is a one hour limit on the amount of time that a MSDIV can be closed. This ensures that no appreciable moisture will build up in the lines while the valve is isolated. Since the MSDIV is normally open, any cycling during this procedure will not impact the containment isolation function of the valves. These valves are assumed to fail closed or be closed by an operator during applicable design basis events. The procedures do not allow opening the MSIV inlet drain valve, only the downstream class G valve, so the containment isolation function of the MSIV drain inlet drain valve will be maintained at all times. No USQ exists.

PT/1/A/4206/15A
PT/1/A/4206/15B

Change 14 to the 1A and change 18 to the 1B Safety Injection Pump Head Curve procedures include the following: 1) changes to show the Improved Technical Specifications (ITS) nomenclature, 2) simplification of data collection and correction for instrument error, 3) providing for stroking the hot leg checks in loop pairs, (B&C or A&D) rather than one loop at a time (there is no design or accident analysis required flow rate through individual hot legs,) 4) verification of cold leg check strokes using header flow and pressure and comparing against a known baseline, without relying on only a flow balance and total

flow rate in accordance with NUREG-1482 (credit is not taken for flow balance in this method), and conservatively applying a worst case flow scenario.

These procedure revisions do not increase the probability of an accident or malfunction, increase the consequences or an accident or malfunction, or create the possibility of any new type accidents or malfunctions. The margin of safety as defined in the Technical Specifications is not reduced. No USQ exists.

PT/0/A/4458/008

This procedure, YC Chiller Refrigerant Sensing Line Metering Valve Throttling is intended to allow adjustments to the sensing line metering valves associated with the chiller condenser control valves, 1RN-445 and 1RN-460. The current position of the respective metering valves was established during post modification testing for NSM-52493. Control valve maintenance, chiller maintenance and changes in operating conditions can all affect the stability of control valve and chiller operation, and therefore the degree of dampening required to achieve stable condenser head pressure. Re-throttling of the metering valves is intended to restore stable chiller operation. This procedure also confirms that the chiller will perform as required after the sensing signal is dampened using the metering valve. Dampening of the refrigerant signal to these valves will not change any actions described in the SAR and does not alter any assumptions made in evaluating the radiological consequences of postulated accidents. This activity creates no new failure modes or operating characteristics for equipment or systems important to safety or plant operation. No USQ exists.

TN/1/B/2507/00/01E
TN/2/B/2507/00/01E

Nuclear Station Modifications MG-12507/00 and MG-22507/00 replace the protective relaying for the bus lines that run from the station to the McGuire switchyard. This involves replacement of all four channels of line sensing and ground fault detection relays as well as the communications portion of the bus lines, which involves both the supervisory and transfer-trip schemes. In addition, relays in the auxiliary relay panels in the switchyard that are associated with the bus lines are replaced because inspection of the relays in these panels revealed cracks in some of the relay housings. These nuclear station modifications are to be implemented while the units are shutdown during refueling. These implementing procedures will control work in the switchyard and will direct all isolation/removal and installation/restoration activities. Wiring changes will be performed in miscellaneous cabinets at the switchyard.

A part of the switchyard procedures block the 67GI/GT relays for Bus Lines 2B, 1A, and 1B (MG-22507) and Bus Line 1B (MG-12507) temporarily while some circuit wiring is isolated and then relocated to support the removal of the Unit 1/Unit 2 relay panels for Bus Line 1A/2A. For the Bus Line 2B work, the 67GI relays for Bus Lines 1A and 1B are blocked temporarily while some circuit wiring is isolated and then relocated to support the removal of the Unit 2 relay panels for Bus Line 2B. The new relaying for the directional ground protection does not use this polarization circuit wiring. Blocking this relaying does not affect the operability of the offsite power system.

For MG-22507, both Unit 1 and Unit 2 off site power sources are aligned for the Unit 1 bus lines (Bus Line 1A and Bus Line 1B), for MG-12507, both Unit 1 and Unit 2 off site power sources are aligned for the Unit 2 bus lines (Bus Line 2A and Bus Line 2B) during work performance by these procedures

(during applicable outage). Therefore, any work activities impacting Bus Line 1A or 1B (MG-22507) or Bus Line 2A or 2B (MG-12507) could impact either the Unit 1 or Unit 2 SAR basis and requirements.

As stated in the SAR documents, the 67GI/GT relays are presently in a 2 of 3 scheme since the channel 3 is blocked. This relaying provides the bus line directional ground over current protection, 67GI for instantaneous protection and 67GT for time delayed operation.

The other bus line protection is provided by directional distance phase relaying (21L), instantaneous over current phase relaying (50), instantaneous over current ground relaying (50G), time delay over current phase relaying (51), and time delay over current ground relaying (51G). Bus line protective relaying normally uses 2 of 4 logic.

The primary phase fault protection is provided by the directional distance phase relaying (21L). Back up phase fault protection is provided by the instantaneous and time delayed over current relaying (50 and 51).

The primary ground fault protection is provided by the directional over current ground relaying (67GI/GT). The 67GI relays have a high current setting and operate instantly for high current ground faults. The 67GT settings are about 20% of the 67GI and are time delayed to allow for momentary unbalances present during transformer charging. These relays provide the primary protection for faults between the switchyard PCBs and the main step up transformer.

Back up ground fault protection is provided by the instantaneous and time delayed over current relaying (50G and 51G). The instantaneous (50G) relays are not directional. However, even though they are set high they will operate only for ground faults between the switchyard PCBs and the main step-up transformer since the fault contribution from the unit is much greater than the switchyard system (grid) contribution. They trip both the switchyard PCBs and transfer trip the plant breakers to isolate the associated bus line. The 51G (time delay) relays will detect faults both on the bus line and on the system due to their lower setting. These relays will only trip the switchyard PCBs. Other protective relaying for the bus lines would be required to trip the plant breakers to isolate the bus line.

These procedures provide detailed instructions on the isolation of affected equipment and the restoration of the switchyard portion of the bus lines. All work is performed while the applicable unit is shutdown during refueling and all essential loads are transferred to the operating unit's bus lines. Blocking the 67 GI/GT relays does not affect the operation of the offsite power system. Activities are to be scheduled using standard work control schedules to minimize risk to the offsite power system. The 67 GI/GT provides the primary directional ground protection for the bus lines. Backup protection is provided using 2 of 4 logic and will provide necessary protection for a ground fault on any of the bus lines. Performing the blocking, wiring changes, and unblocking activities does not change the probability of occurrence of an accident previously evaluated. Action steps associated with the blocking and unblocking of the relaying are similar to those activities performed during normal maintenance (such as relay case or test switch cover removal and replacement) and are considered in the probability determinations based upon maintenance data. Verification will be performed to ensure that the proper components are isolated, modified and restored. Activities involved in this procedure do not impact the probability of an equipment failure or malfunction for the bus line equipment such as lines, towers, PCBs or main step up transformers. The relaying provides a supervisory function to monitor and detect conditions that represent equipment failures or malfunctions and initiates actions to isolate the suspect equipment. No USQ exists.

TO/1/A/9600/115

This procedure, Chemical Flush of the Generator Stator Cooling Water (KG) System is a temporary procedure to control the evolution for chemically cleaning the system. The purpose of this procedure is to give instructions in performing a chemical flush while the unit is on-line. This procedure also stipulates what actions to take in response to unexpected system conditions or when monitored parameters exceed specified limits. The actions that the procedure will specify to be performed will either be; 1) terminate chemical addition; or 2) reduce power; or 3) trip the reactor.

The KG system is a closed loop; non-safety related system that circulates high purity water through the stator coil hollow conductors for removal of the heat in order to maintain the generator within a safe temperature range when operating. The KG system works in conjunction with the Generator Hydrogen system to accomplish this function. The flow passages through the hollow strands of copper that make up part of the windings of the stator may be reduced in size by the deposit of corrosion products thus restricting flow. A reduction of flow through these hollow strands can result in an increase of the stator coil exist temperature. One method to restore full flow is to chemically clean the hollow copper passages by adding the following chemicals: Aqueous solution of disodium salt of EDTA; 30% H_2O_2; CUPROLEX Inhibitor 1505, activator A505 and Na_2SO_4. This involves injecting PPM-level Na_2EDTA, hydrogen peroxide, and possibly traces of a reaction activator or inhibitor into the KG system. The Na_2-EDTA dissolves the copper oxide on the surface of the copper stator winding and forms a soluble ion, which will eventually be removed by the KG system demineralizer. The peroxide acts to oxidize Cu_2O to CuO so the EDTA and the activator/inhibitor can dissolve it are used to control reaction times.

The main generator is not safety related and does not perform any safety function. Malfunction of the generator resulting from this cleaning activity may cause a turbine trip. The occurrence of a malfunction of the main generator as a result of the chemical cleaning activity is not likely due to 1) Strict operational limit on the conductivity level during the cleaning; 2) The chemicals used react preferentially with undesirable copper oxides and do not attack the bare copper metal in the stator winding or materials used in the KG system; and 3) The mechanism of removing the copper oxide from the stator winding surface is a controlled reaction with the chemical Na_2EDTA to form a soluble complex compound, and is not an immediate, bulk striping of the oxides from the copper surface. No USQ exists.

TT/0/A/2493/004

This procedure, A&B YC Condenser Head Pressure Control Valve Stability Testing, will vary YC chiller condenser head pressure in order to achieve low service water flow demand, to determine if both 1RN-445 and 1RN-460 are inherently unstable at low flows. This will be accomplished by aligning the train of Nuclear Service Water (RN) supplying cooling water to the condenser, to the Standby Nuclear Service Water Pont (SNSWP), to ensure that the flow demand is responding to the coldest available cooling water. Condenser head pressure may then be adjusted by increasing/decreasing the power spring tension on the respective control valves, to further decrease flow demand, until RN flow is less than 200 GPM. The condenser head pressure may be decreased to determine if valve stability is a function of power spring tension. Condenser head pressure, sensing line signal pressure and RN flow to the condenser will then be monitored throughout the test to determine if the valve instability has its origins in the valve(s) or refrigerant filled sensing lines. Temporarily changing condenser head pressure will not change any actions described in the SAR and will not alter any assumptions made in evaluating the radiological consequences of postulated accidents. This activity creates no new failure modes or operating characteristics for equipment or systems important to safety or plant operation. No USQ exists.

TT/0/A/2493/005

The Nuclear Service Water (RN) control valves, which regulate the cooling water flow rate through the YC chiller, have been cycling during periods of low flow demand. This results in the condenser head pressure fluctuating approximately 15-20 psig. The valve manufacturer has recommended modifications to the internals of the valves to significantly reduce or eliminate this cycling. The recommended internal modifications affect only the controls internals and would have no effect on the RN System flow balance. This procedure, B YC Condenser Head Pressure Control Valve Stability Testing, installs prototypical internal components and test the stability of valve 1RN460 with these different internals installed in the valve. Once installed, the crack open pressure will be set while the chiller is idle. After the chiller is started, the condenser head pressure will be varied in order to achieve low service water flow demand to determine if 1RN-460 remains unstable at low flows. This will be accomplished by aligning the train of RN supplying cooling water to the condenser, to the standby nuclear service water pond (SNSWP), to ensure that the flow demand is responding to the coldest available cooling water. Condenser head pressure will then be adjusted by increasing/decreasing the power spring tension on the respective control valves, to further decrease flow demand, until RN flow is less than 200 gpm. Condenser head pressure, sensing line signal pressure and RN flow to the condenser will then be monitored throughout the test to determine if the valve instability has been eliminated. The YC chiller will be logged inoperable during the entire testing evolution. Temporary changing condenser head pressure will not change any actions described in the SAR and will not alter any assumptions made in evaluating the radiological consequences of postulated accidents. This activity creates no new failure modes or operating characteristics for equipment or systems important to safety or plant operation. No USQ exists.

TT/1/A/9100/115

The purpose of this procedure is to perform a functional verification for the #1 Turbine Driven Auxiliary Feedwater Pump following the installation of automatic recirculation control valve under Nuclear Station Modification MG-12515/00. This modification replaces the discharge check valves for each Auxiliary Feedwater system (CA) pump with an Automatic Recirculation Control (ARC) valve. These valves will provide an assured minimum flow path for the CA pumps. The ARC valve operates as a combined check valve and globe valve with the globe valve port providing an assured flow path to prevent deadhead conditions on the pump. The check valve port is the main discharge flow path and operates in the same manner as the previous discharge check valve. The ARC valve is designed such that the glove port closes as the check port opens assuring a continuous discharge path for the pump and automatically eliminating recirculation flow as sufficient normal discharge flow is established. The installation of this recirculation flow path will result in removal from service the current recirculation flow paths for the CA pumps. This piping will now be used only for pump flow testing. The current recirculation valves will have automatic controls removed and will only be operated manually.

In addition, a continuous discharge flowpath will be added between each ARC valve and the feedwater pressure boundary valves to assure that this piping is not pressurized due to valve leakage. These flowpaths will have a flow control valve in line to limit flow to 5 gpm at 1700 psid.

Post modification testing will be performed on the revised system to ensure correct flow balance and operation of the discharge and recirculation flowpaths. The operation of automatic functions on a system start signal will be verified using a simulated signal. This testing ensures that the revised system responds as designed. Operation of all revised components will be tested and data collected to allow future determination of system health.

This modification improves the reliability of the recirculation function for the CA pumps. The current arrangement is subject to failure due to a loss of valve control instrument power. The revised flowpath is controlled by mechanical action of the valve such that either the normal discharge flowpath or the recirculation flowpath is always available. The ARC valves are considered as reliable as the current discharge check valves and recirculation control valves for valve malfunction in other respects. The use of the ARC valves along with the provision of a flow path to prevent pressurization due to valve leakage maintains all the design operation functions being provided by the current system design. The diversion of 5 gpm of CA supply to the feedwater system during operation does not significantly impact the ability of the CA system to meet its design requirements. No new functions are added by the installation and use of the ARC valves. The testing of the system will still utilize the old recirculation flow path since the automatic action of the ARC valve prevents it being used for testing. The ARC valves and new recirculation meet the same QA standards ad the associated piping and components. No USQ exists.

TT/1/A/9100/535

This procedure accommodates post-modification testing of the hi-head safety injection pump isolation valves to the reactor cold leg nozzles. The testing will demonstrate full flow capability through each valve independently to ensure required functional capability. Demonstration of system flow balance is not required, since the isolation valves provide little contribution to over-all system resistance (by comparison to the cold leg throttle valves). The test is intended to demonstrate the isolation valves are effectively full open and that the new valves do not degrade safety injection capability. The alignments utilized for the test are accepted/intended system alignments, for which suitable procedural steps/prerequisites/sequences ensure adequate Chemical and Volume Control (NV) system pump suction, miniflow paths and runout limits are maintained. Prerequisites are provided to ensure adequate Reactor Coolant System (NC) and Volume Control Tank (VCT) surge capacity is available to accommodate the in-flow from the Refueling Water Storage Tank (FWST). Performance of this test complies with Technical Specification requirements for low temperature over protection (LTOP) and Selected Licensee Commitment requirements for boration flowpaths (as applicable for the mode of performance). Therefore the probability of occurrence of any previously analyzed event is not increased. Adequate pump suction, miniflow and runout provisions are ensured by the test alignments/sequence at all times. The test alignments are consistent with the intended/allowed system alignments and resultant process/equipment conditions created are within design limits. The test adequately demonstrates full flow capability through the hi-head injection isolation valves, thereby ensuring proper functionality for credited events. The test does not post any challenge to overall plant safety or operation, nor increase the consequence/probability of any previously analyzed events. No USQ exists.

TT/0/A/2493/006

This procedure, Stability Testing of Prototype Metrex Valve Plug, is the controlling procedure for installation and stability testing of a new plug for 1RN460 (Nuclear Service Water System). The procedure will vary YC chiller condenser head pressure in order to determine if the new plug design eliminates valve instability at low flow demands. Temporarily changing condenser head pressure will not change any actions described in the SAR and will not alter any assumptions made in evaluating the radiological consequences of postulates accidents. This activity creates no new failure modes or operating characteristics for equipment or systems important to safety or plant operation. No safety issues, Technical Specification changes or USQ is created.

TT/2/A/9700/300

This temporary procedure, Supplemental Response Time Testing for 2ENB9440 and Channel IV OPDI and OTDT, provides a means of performing response time testing of the channel 4 functions for Power Range Monitoring (N-44), Overpower Delta Temperature (OPDT), Overtemperature Delta Temperature (OTDT) while the reactor is operating in Mode 1. This surveillance testing was required to be performed during 2EOC11 and was missed due to model work order errors. The use of the temporary procedure allows testing to verify that the functions are performing as required. The surveillance testing of the trip response time for each of the functions will be performed by putting that function in the test (tripped) condition, re-energizing the tripped instrumentation loop with test equipment to normal operating condition, and using test equipment to measure the time for an input signal which should generate a trip condition to be processed by the loop and trip condition requirements to be satisfied. Each function is tested for all combinations of input signals that will produce trip signal conditions.

The reactor protection system is designed to allow individual channel testing and this type of testing is performed online as required. Response time testing is normally performed offline due to the greater ease and straightforward nature of testing during shutdown conditions. Actual measured trip times determined by these tests are compared to times assumed in safety evaluations. The measured times must be less than the trip times assumed in the safety analyses. The temporary procedures assure that the instrumentation is tested to meet all applicable requirements and that the testing does not create a trip hazard beyond the normal potential for trip when a channel is in the test condition. No instrument channel or train independence is violated by these tests. Reactor trip functions are maintained as provided for in the technical specifications. The reactor trip logic is not changed from the allowed test logic for surveillance testing. No USQ exists.

TT/2/A/9700/301

This temporary procedure provides a means of performing response time testing of the reactor coolant pump power monitoring undervoltage and underfrequency while the reactor is operating in Mode 1, 2, or 3 with all reactor coolant pumps running. This surveillance testing was required to be performed during 2EOC11 and was missed due to model work orders errors. The surveillance testing of the trip response time for each of the functions will be performed by putting that function in the test (tripped) condition while using test equipment to measure the time for an input signal which should generate a trip condition to be processed by the loop and trip condition requirements to be satisfied.

Response time testing is described in UFSAR section 7.2.2.2.3. The response time testing is designed to be performed offline and therefore, the SAR description of the response time testing must be changed to perform the tests online. Since the temporary procedure is intended to only satisfy the missed surveillance requirement and not permanently change the normal testing of these functions the actual SAR description will not be changed. The reactor protection system is designed to allow individual channel testing and this type of testing is performed online as required. Response time testing is normally performed offline due to the greater ease and straightforward nature of testing during no mode conditions. Actual measured trip times determined by these tests are compared to times assumed in safety evaluations. The measured times must be less than the trip times assumed in safety analyses. No reactor trip functions or signals are defeated by these tests. The reactor trip logic is not changed from the normal test logic for this surveillance testing. No USQ exists.