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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.

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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RGC File

Nuclear Station Modifications Completed Under 10CFR50.59

<u>MG-12472</u>
<u>MG-22472</u>

These Nuclear Station Modifications upgraded the fuel transfer cart conveyor drive control system with a new control system that allows McGuire to vary the cart speed up to a maximum speed of about 40 feet/minute. The modifications also replaced the hydraulic load cells and mechanical geared limit switches on the upenders with new electronic components that perform the equivalent function.

The Fuel Handling (FC) system consists of the equipment needed for the refueling operation. Basically, the equipment is comprised of cranes, handling equipment and a fuel transfer system. A fuel transfer tube connects the refueling canal and the spent fuel pool. This tube is fitted with a blind flange on the canal end and a gate valve on the spent fuel pool end. The blind flange is in place except during refueling to ensure containment integrity. Fuel is carried through the tube on an underwater transfer car. The upender at either end of the fuel transfer tube is used to pivot a fuel assembly. Before entering the transfer tube the upender pivots a fuel assembly to the horizontal position for passage through the transfer tube. After the transfer car transports the fuel assembly through the transfer tube, the upender at that end of the tube pivots the assembly to a vertical position so that it can be lifted out of the fuel container.

The fuel transfer system is used to transport fuel assemblies between the refueling canal and the spent fuel pool (transfer canal), through containment. Fuel transfer system components include the conveyer cart, fuel assembly container, and the upenders (lifting frames). The fuel transfer system includes an above-water electric-motor driven transfer cart that runs on tracks extending from the refueling canal through the transfer tube and into the spent fuel pool and an upender lifting frame at each end of the transfer tube. The upender in the refueling canal receives a fuel assembly in the vertical position from the reactor manipulator crane. The fuel assembly is then lowered to a horizontal position for passage through the transfer tube and raised to a vertical position after passage through the transfer tube by the upender in the spent fuel pool. The fuel pool manipulator crane takes the fuel assembly to a position in the spent fuel storage racks.

These modifications involve a system (the fuel transfer drive components and controls) that is normally not in operation and does not interface with any other mechanical systems. This system is used for the movement of fuel between the reactor vessel (in containment) and the fuel pool in the fuel building. The Fuel Transfer System (FTS) is not an accident mitigation system and is not designed as a QA-1 system.

These modifications improve the reliability of the FTS system. No safety related structures, systems or components (SSCs) are impacted. No new failure modes are introduced. The FTS is not an accident mitigation system. The FTS is not considered to be an initiator of any previously evaluated accidents and these modifications do not result in the FTS causing any previously evaluated accidents. Integrity of the fission product barriers is not impacted. No Technical Specification changes are required. Prior NRC approval is not required.

MG-12518/P2
MG-22518/P1

These modifications installed non-safety related Auxiliary Feedwater Storage Tanks (CASTs) (one per unit) and isolated all other sources . The tanks are located near the southwest and east side of each units turbine building. The new tanks are multi-legged, elevated storage tanks that are 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 CASTs are aligned to the auxiliary feedwater (CA) pumps at all times. They are used as the primary suction source. The upper storage tanks (UST) and the auxiliary feedwater condensate storage tanks (CACST) will remain tied to the CA system but will be normally isolated from the CA pumps. The recirculation flow from the CA pumps will be returned to the CAST. The new tanks are used as the suction source for all pump testing. During outages, the new tanks will be used to fill the steam generators for wet lay-up. The new CASTs will be the primary suction source for the CA pumps during transients. If the tanks become 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 CASTs 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 CASTs do not need to be isolated, 1/2CA2 will not need to be closed. Maintaining 1/2CA2 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 these modifications. The safety grade indication of auxiliary flow to each steam generator will not be impacted. 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. The modifications increase the nominal capacity of the preferred condensate-quality water to approximately 300,000 gallons. The new elevated storage tanks are non-safety related, not seismically qualified and will not be protected from tornado-generated missiles. The modifications are 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 Technical Specification changes are required. Prior NRC approval is not required.

MG-42481

This modification involves the design and construction of an Independent Spent Fuel Storage Installation (ISFSI). The ISFSI site is located on the west side of the plant site inside the station security. 10 CFR 72.212 requires specific evaluations, including a 10 CFR 50.59 evaluation, to be performed prior to use of the RN-32A cask system. The scope of this evaluation was to evaluate those whether activities related to the storage of spent fuel under the general license issued for the TN-32A dry storage cask. The activities that are associated with the TN-32A cask system for the storage of spent fuel can be categorized into the following general areas: 1) loading of a cask, 2) transport of a loaded cask to/from the ISFSI, 3) cask storage operations and 4) unloading of a cask.

This evaluation examines the effect that dry fuel storage related activities, performed with equipment and programs authorized under 10 CFR 50, has on the McGuire licensing basis. The equipment (cask design, storage pads, etc.) and other activities that are performed outside the authorization of the license for McGuire are subjected to the requirements of 10 CFR 72, Subpart K and are not addressed within this evaluation.

The use of the TN-32A cask system requires several heavy load lifts. The heavy loads lifts that are performed in order to store spent fuel in a TN-32A cask system are similar to those involved with casks used for shipping spent fuel as discussed in Section 9.1 of the UFSAR. The lifts to be performed by the turbine bay crane will be in accordance with the Duke Power Company Lifting Program Manual. The manual ensures that this crane complies with applicable government regulations and industry standards. The fuel building crane will perform the next major lift of the cask. The postulated cask drop accident was updated to include the TN-32A cask. The analysis shows that TN-32A cask can not interact with fuel assemblies that are stored in the spent fuel pool. This is contingent upon following a prescribed load path while handling the cask. Further, the analysis also shows that the liner plate will not fail and no leakage from the spent fuel pool will occur. As such, in the event of a cask drop, the liner plate will not be damaged, no leakage will occur and the spent fuel will remain flooded.

During cask movement of the TN-32A cask within the fuel building, an impact limiting devices will not be employed. As such, a calculation of the radiological impact of the potential drop of the cask was performed. The analysis assumes that all fuel pins fail and that a failure of the cask confinement system also occurs in order to maximize the dose consequences for this event (TN-32A cask drop in the truck corridor during cask movement). The analysis shows that for this event, the doses are well within the acceptance criteria and guidelines associated with the SRP for this event, in that the resultant doses are a small fraction of the 10 CFR part 100 limits.

The spent fuel bridge is used to remove fuel assemblies from the storage racks and place the assemblies into the cask. These fuel movement operations will be performed similarly to routine fuel placement/shuffles that are normally performed. The TN-32A cask is transferred to the ISFSI by the cask transporter over a prescribed haul path. The cask transporter limits the lift height of the cask to less than 12 inches above the ground. An evaluation of this prescribed haul path was performed. This evaluation shows that the movement of the cask to the cask storage location (ISFSI) loading will not adversely impact underground utilities.

Other facility equipment that will be utilized in support of activities associated with the TN-32A cask system are: Station Air (VS), Instrument Air (VI), Makeup Demineralized Water (YM) systems, and station electrical power. The equipment will be used in a similar manner, as they are currently utilized in support of activities for the operation of McGuire. Modifications of the equipment or changes to procedures are not necessary in order to support dry fuel storage activities.

Based on the above determination, the dry fuel storage activities associated with spent fuel storage in a TN-32A cask are safe and will not adversely affect the safety function of the spent fuel pool nor its structural integrity. Also, these dry fuel storage activities will have not adverse impact on any other structure, system or component at McGuire. Accordingly, this evaluation has not identified any safety concerns. No Technical Specification changes are required. Prior NRC approval is not required.

MG-15227

This modification removed the existing metal reflective mirror insulation and support structure from the Unit 1 reactor coolant pumps. This modification also installed flexible blanket insulation on all four Unit 1 reactor coolant pumps. The flexible insulation blanket system consists of 4 inch thick Nukon insulation blankets what will be wrapped around the reactor coolant sumps, and 22 gauge stainless steel jacketing that is designed to enclose all of the insulation. In addition, a drip shield will be provided to help protect the insulation from oil and water.

The existing reactor coolant system and equipment design bases are unchanged by this modification. The insulation materials comply with Regulatory Guide 1.82 and Regulatory Guide 1.36. By meeting the requirements of Regulatory Guide 1.36, the flexible blanket insulation does not involve any corrosion concerns. In addition, assurance is also provided that the insulation has the appropriate fire resistance characteristics, is capable of withstanding design basis seismic forces intact, and will not become a health hazard due to neutron activation over the life of the plant. Further, the blanket insulation will not adversely impact to the seismic qualification of the reactor coolant pumps due to the weight difference between the mirror insulation and the blanket insulation. The primary safety issue with the use of flexible blanket insulation on the four Unit 1 reactor coolant pumps involves the potential for the generation of debris as a result of a LOCA. Evaluations performed show that there will be no adverse impact to the performance of the Emergency Core Cooling System (ECCS) pumps in mitigating the consequences of an accident.

The evaluation of this modification successfully satisfied the criteria of 10 CFR 50.59. No Technical Specification changes are required. Prior NRC approval is not required.

MG-52521

This modification will terminate spare inverter SKX to busses KXA and 1KU (via disconnect switch multiple lugged outputs). This modification also provides 125 VDC power source from distribution center DCA by connecting cable to the hot DCA buss. An alternate power supply from regulated power distribution center MKA (via distribution center MKA-1) will be connected as an input to the spare inverter's manual bypass switch. This modification also installs modified doors and pans for distribution center DCA.

Procedure TN/0/B/2521/P3/01E provides instruction for implementing NSM MG-52521/P3. This part of the modification connects the 125VDC/120VAC non-vital inverter SKX to the

125VDC distribution center DCA, 240/120VAC distribution center MKA-1 and 120 VAC panelboard KXA, 120VAC panelboard 1KU. 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 not formal EQ requirements for the 240/120V AC Auxiliary Control Power System (EPF) system, however; the new spare inverter is rated for operation in an environment with ambient temperatures of 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.

The evaluation of this modification and the procedure to implement the modification successfully satisfies the criteria of 10 CFR 50.59. No Technical Specification changes are required. Prior NRC approval is not required.

Procedure Changes Completed Under 10CFR 50.59

OP/1/A/6250/006
OP/2/A/6250/006

The purpose of these procedures is to define the operation of the Main Steam (SM) system. During unit heatup valve alignment of certain valves are checked against Enclosure 4.4, Valve Checklist to verify that the valves are in their correct position. The changes to Enclosure 4.4 concern the Atmospheric Dump Isolation Valves (ADIVs) (valves 1&2SV45 through 1&2SV52).

As Atmospheric Dump Valves (ADVs) are isolated by their respective ADIVs, the steam relief capacity of the remaining ADVs in service will decrease. The anticipated relief capacity in the event that only one ADV is in service will be approximately 6 percent of full load. Although the steam relief capacity, as described in the UFSAR is decreased, this relief path via the ADVs is still functional. The mission of the AVDs to dissipate reactor coolant system heat following a reactor or turbine trip in a controlled manner is still accomplished, even with just one ADV in service.

With only one ADV in service, it is likely that a turbine generator full load rejection event will result in a turbine trip, reactor trip and lifting of one or more main steam safety valves. In this configuration, this impact appears inconsistent with UFSAR discussions. Notwithstanding, analysis of the turbine generator full load rejection event, as discussed in UFSAR 15.2, takes no credit for the ADVs. The analysis of this event only considers the function (lifting) of the main steam safety valves to dissipate the reactor coolant system heat. Accordingly, this procedure change results in a plant configuration that is consistent with the licensing basis analysis described in the UFSAR. No Technical Specification changes are required. Prior NRC approval is not required.

AP/0/5500/045

This is a new procedure that provides guidance to mitigate the effects of a fire that has the potential of causing loss of control of safe shutdown systems during Modes 1-3.
This abnormal procedure will be entered when there is an active fire in one of the specified Appendix R fire areas. This procedure provides guidance to mitigate the effects of a fire that has the potential of causing loss of control of safe shutdown systems during Modes 1-3. This procedure is not concerned with fires in the service or turbine building, or while the unit is in Modes 4, 5, 6, or no-mode. In general, the actions of each enclosure are intended to maintain the functionality of the designated safe shutdown train. The actions taken per this procedure are precautionary in nature in order to preserve the functionality of the safe shutdown train. The resultant configuration will not place the unit outside the plants licensing design bases nor inconsistent with analyses or descriptions described in the UFSAR. No Technical Specification changes are required. Prior NRC approval is not required.

OP/2/A/6200/119

This new procedure converts chemistry procedure (CP/2/A/8700/018) into a revised format. An evaluation was conducted to allow a change in corrosion inhibitors in the Emergency Diesel Generator Jacket Water Cooling (KD) system from sodium nitrite to sodium molybdate, specifically Nalco/Calgon Power Group product LCS-1200. This change also includes deletion of the enclosures associated with feed and bleed of the individual trains which are duplicated in another operating procedure and a change to the Chemistry Manual since the use of corrosion inhibitors in the KD system are described in the manual.

The KD system is designed to maintain the temperature of the diesel generator engine within an optimum range. Auxiliary purposes are to supply cooling water to the lube oil cooler and the intercooler, which cools the air leaving the turbo-charger.

The KD system is a closed cooling system that circulates water through the diesel generator engine. The water is heated as it passes through the engine, the intercooler and the lube oil cooler, and gives off this heat to the Nuclear Service Water System. The jacket water pump and the intercooler water pump circulate water through the system during normal operation. The jacket water heater circulating pump operates continuously, while the jacket water heater cycles on and off as required when the engine is shut down, to maintain standby, keep warm temperatures.

The KD system for each diesel unit is a Duke Class C System. Each diesel unit is housed separately in Category I structure which is part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in a separate room. Since the diesel units themselves are fully independent and redundant for each nuclear unit, they meet the single failure criterion.

The type of corrosion inhibitor used in the KD system is not addressed in the UFSAR. Changing to Sodium Molybdate does not affect the specific gravity of the coolant. The heat transfer rate remains unchanged, and the heat exchanger sizing calculations are not affected. From this, it is concluded that the safety related function of the KD system, maintaining the Emergency Diesel Generator's operating temperature, remains unchanged as a result of the corrosion inhibitor change. No Technical Specification changes are required. Prior NRC approval is not required.

TO/1/B/9600/126

The change to this procedure, "Auxiliary Feedwater Storage Tank (CAST) Pre-operational Cleanup and Preparation" involves providing procedural guidance for purification of the CAST water from the tank through a steam generator blowdown demineralizer to the hotwell for the initial fill of the hotwell prior to startup of the condensate system. In addition, a change was made to the CAST nitrogen sparge delivery pressure from 25-100 psig to 10-50 psig since adequate sparge is expected at the lower pressure range.

Per UFSAR Section 10.4.10, the Auxiliary Feedwater System (CA) assures required feedwater flow to the steam generators to remove decay heat while maintaining steam generator water levels adequately to prevent undue thermal cycling of the steam generators. These activities are being performed in conjunction with normal condensate fill and preparation for startup. The CA system standby readiness to fulfill its design function is not impaired by this activity. The CAST level will be maintained greater than 15 feet and the availability of the assured CA supply source (Nuclear Service Water System) is not affected.

The change in nitrogen sparge pressure does not alter the design function of the sparge system and does not affect the ability of the CAST to carryout its aforementioned purpose.

Per UFSAR Section 10.4.8, the Steam Generator Blowdown System (BB) is designed to operate manually and on a continuous basis as required to maintain acceptable steam generator secondary side water chemistry. The BB system is designed to prohibit radioactive discharge to the environment from the blowdown liquid. The system serves no safety function.

The BB system is normally isolated and in standby during initial condensate fill operations. The use of the BB demineralizers for the purification of hotwell fill water from the CAST is not addressed in the SAR. The use of the specified demineralizer for this purpose will not affect the ability of the system to support subsequent startup activities and meet its design requirements to maintain steam generator secondary-side water chemistry and remove radioactivity from blowdown liquid.

Pressure, temperature, and calculated flowrates through the demineralizer during this described activity are well below design maximums. The discharge flow path to the hotwell is per normal alignment with the resulting increase in hotwell level limited to a maximum of 6 feet to preclude overfilling.

Affected components are not operated outside original design parameters and no systems of components important to safety are impacted in a manner that would increase the probability or the consequences of accidents or component malfunctions. No Technical Specification changes are required. Prior NRC approval is not required.

OP/0/A/6550/11
PT/0/A/4150/37

Due to the degradation of the Boraflex panels, a new set of burnup versus enrichment limits for Region 1 are established to ensure K_{eff} 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 McGuire Nuclear Station Technical Specifications. These new limits are to be implemented by changes to Procedures OP/0/A/6550/11, Internal Transfer of Fuel Assemblies and PT/0/A/4150/37, Total Core Unloading.

The function of the spent fuel 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 K_{eff} 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 K_{eff} remains less than or equal to 0.95 with the anticipated degradation of the Boraflex panels and the spent fuel pool filled with unborated water. The compensatory actions and procedure changes affect no design criteria or safety functions of any structure, system or component. The UFSAR is not affected as a result of the new limits. No Technical Specification changes are required. Prior NRC approval is not required.

TO/2/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 KG system. The purpose of this procedure is to give instructions in performing a chemical flush of the KG system while the unit is on-line. This procedure will also stipulate what actions to take in response to unexpected system conditions of when monitored parameters exceed specified limits. The actions that the procedure will specify to be performed with 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 circulated high purity water through the stator coil hallow 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 exit 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 I505, 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_2EDTA 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 reactor with the chemical Na_2EDTA to form a soluble complex compound, and is not an immediate, bulk stripping of the oxides from the copper surface. No Technical Specification changes are required. Prior NRC approval is not required.

TO/2/9600/291

Procedure TO/2/9600/291, SKX Inverter Test, is a Temporary Operating (TO) procedure utilized to properly align and remove the SKX static inverter to the 2KU and KXB panelboards. This procedure provides detailed steps and instructions for the alignment and removal of the spare non-vital inverter (SKX) to panelboard KXB and panelboard 2KU.

Nuclear Station Modification MG-52521/P1 and P2 installed a new spare inverter and associated equipment (static switch, manual bypass switch, breaker alignment panel, breakers and various cables) for the **EPF** system. The new inverter is an addition to the existing four inverters (1KU, KXA, 2KU, and KXB) that comprise the EPF system. The new inverter and associated equipment are not nuclear safety related. The modification will not affect the design, function, or operation of the EPF system. The new spare inverter is rated 35 KVA output with 125 VDC input. The spare inverter, SKX, is capable of replacing either the KXB inverter or the 2KU inverter. The new spare inverter will operate similar to the current inverters (KXB and 2KU).

In support of post modification tests, a temporary operating procedure (TO/2/A/9600/291) was developed to control the alignment and removal of SKX to Panelboards 2KU and KXB, while receiving power from either Distribution Center DCB or from its alternate power source, Distribution Center MKB. The basic sequence for aligning SKX to inverter KXB or 2KU (only one at a time) is defined within the final scope document for Nuclear Station Modification MG-52521/P2. Procedure TO/2/A/9600/291 is consistent with the basic sequence outlined in the FSD assuring proper plant configuration needed for functional verification testing of the modification to install a spare non-vital inverter, SKX. No Technical Specification changes are required. Prior NRC approval is not required.

PT/2/A/4206/022

The purpose of this procedure is to satisfy full open verification requirements of the Cold Leg Accumulator (CLA) check valves using CLA dumps to the reactor coolant system cold legs. Acoustic monitoring is used to verify backstop hits. The test is done in No Mode with the reactor vessel head removed, which ensures that no reactivity concerns, thermal transient concerns, or impacts to core cooling via residual heat removal will be present. Full canal and spent fuel pool will be maintained. A very low CLA pressure and level will be used to ensure no overcooling of the CLA occurs and that nitrogen is not injected into the reactor coolant system (nor residual heat removal pump suction). The CLA will be injection one at a time and canal level checked between each dump: adjustment will be made if canal level is too high to accommodate another dump. Communication will be established with a non-licensed operator at the motor operated CLA block valve so that it can be manually closed if it fails to close from the control room.

This test is being performed under conditions which ensure no safety nor equipment impacts occur. CLA health is ensured by the low nitrogen pressure and water level (this limits the nitrogen expansion which will occur, thus limiting overcooling to a final nitrogen temperature of 40 degrees F). Nitrogen will not be injected as a result of this low initial pressure, so that degradation of ND pump suction is not a concern. The no mode conditions ensure no thermal transients (CLAs should be about 80 to 90 degrees F) with reactor coolant (NC) at the same temperature. There are no overpressurization concerns with the head off the vessel. There are no canal overfill concerns with ample margin being maintained throughout the test for total-possible CLA dump.

This procedure does not increase the probability of an accident or malfunction, increase the consequences of 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 Technical Specification changes are required. Prior NRC approval is not required.

PT/2/A/4206/15A
PT/2/A/4206/15B

These revisions to the 2A and 2B Safety Injection Pump Head Curve Procedure includes the following: 1) verification of cold leg check valve full-open strokes using header flow and pressure, and comparing against baseline flow and dp, without relying on flow balance in accordance with NUREG-1482, 2) verifying closure of miniflow isolation valve 2NI-147A (to verify it does not backleak to the FWST), 3) verification of 2NI 148 and 2NI-116 being seated using flow diversion versus pump rotation (lack of rotation of the 2A and 2B NI pump), and 4) procedural enhancements, editoral changes, data collection simplifications.

These tests are completed in modes 5, 6 or no mode when Emergency Core Cooling System (ECCS) accident alignments are not required. Sufficient procedural control is provided to maintain reactor coolant system cooldown limits. This revision does not change test alignments. None of the alignments in this procedure would initiate a Chapter 15 accident. The option to verify full movement-to-open for the NI to cold leg secondary checks via change in flow and dp from baseline conditions is within the scope of OM-10. A conservative acceptance margin (flow and pressure) is used, which is based on the lowest-flow check. The full stroke verification (where FWST water is injected to NC cold legs) enclosure now ensures no reactivity changes will occur during the injection. No accidents evaluated in the SAR will be more probable as result of the changes to this procedure.

This procedure does not increase the probability of an accident or malfunction, increase the consequences of 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 Technical Specification changes are required. Prior NRC approval is not required.

TT/2/A/9100/537
TT/1/A/9100/546

These procedures accommodate flushing of the volume control tank (VCT) relief header utilizing inlet flow to the seal water return heat-exchanger. The intent of the flush is to eliminate hotspots within the VCT relief header, which collect over time due to primary sample return flow(s). The procedure is intended to be performed during crudburst clean-up. The flush requires a charging pump to be in operation with a normal VCT suction, miniflow and charging alignment established.

The Chemical and Volume Control (NV) System is not credited nor required to perform an ECCS functions during the mode allowed for procedure performance. The NV system is required to provide a single boration flowpath for reactivity control during the allowed mode of performance. Establishment of adequate shutdown margin is required during the mode allowed for performance and the NV system provides the requisite operable boration flowpath. The procedure maintains adequate NV pump discharge alignments to ensure there is no potential for NV pump deadheading or loss of suction, thus operability of requisite boration flowpath is assured. The UFSAR analysis postulates an at power line break in the normal letdown line downstream of the outboard containment isolation; however, no failure is postulated for NV seal return piping. No Technical Specification changes are required. Prior NRC approval is not required.

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

Nuclear Station Modification (NSM) MG-12507/00 replaces 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. Relays in the auxiliary relay panels in the switchyard that are associated with the bus lines will be replaced because inspection of the relays in these panels revealed cracks in some of the relay's housings. The NSM will be implemented while Unit 1 is shutdown during refueling. Work in the switchyard will be controlled by this procedure. This TN will direct all switchyard isolation/removal and installation/restoration activities. Wiring changes will be performed in miscellaneous cabinets at the switchyard.

This procedure met the criteria for screening except for steps which block the 67GI/GT relays. For Bus Line 1B the relays are blocked temporarily while some circuit wiring is isolated and then relocated to support the removal of the Unit 1 relay panels for Bus Line 1A. 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. Although operation with some primary relaying blocked is somewhat different than described in the UFSAR, this operation will not require a change since these are temporary alignments that are returned to full agreement with documentation prior to procedure completion. This alignment is considered to be not required in the future since it is to support the implementation of a specific modification and the new replacement relaying does not require this circuit. No Technical Specification changes are required. Prior NRC approval is not required.

CP/2/B/8400/051

This is a new procedure that allows for detection of small amounts of inleakage using tracer gases injection. The gases used are Sulfur Hexafluoride (SF_6) and Helium. SF_6 is heavier than air and readily mixes with water while helium is both lighter and than air and inert. Both are easily detected in a gaseous flow stream using specialized analytical equipment.

For condenser raw water leakage, tracer gas (SF_6 or Helium) is injected into a condenser water box via the Condenser Cleaning System (RA). The gas mixes with raw water where it can be drawn into the condensate system via hotwell vacuum through a defective tube. The tracer gas then comes out of solution and combines with normal offgas which is swept from the system via the condensate steam air ejectors and monitored for the presence of tracer gas using sensitive analytical instrumentation.

In the case of air in-leakage, gas is sprayed into or around components suspected of leaking. For components associated with the secondary cycle, tracer gas is again drawn off in the hotwell and detected as previously discussed. For other system components, tracer gas must be separated in a vapor space of a tank or similar component or have the gas stripped from liquid sample prior to analysis.

In addition to their normal function during plant power operation, the main condensers are used to remove residual heat from the reactor coolant system during the initial cooling period after unit shutdown when the main steam is bypassed to the condensers by the steam dump system. The condensers are also used to condense the main steam bypassed to the condenser in the event of sudden load rejection by the turbine-generator or a turbine trip. In the event of generator load rejection, the condensers condense 40 percent of full load main steam flow from the steam dump system, and the power-operated relief valves will discharge the remaining main steam flow to atmosphere to effect safe reactor shutdown and to protect the main steam system from overpressure. If the main condensers are not available during normal unit shutdown, or sudden load rejection on a turbine trip, the power-operated relief valves and the spring-loaded code safety valves can discharge full main steam flow tot he atmosphere and effect safe reactor shutdown. Non-availability of the main condensers considered here includes failure of the circulating water pumps to supply cooling water, or loss of condenser vacuum for any reason.

The condenser cleaning system is designed to provide cleaning of the main and feedwater pump condensers by circulation of slightly abrasive balls through the raw water (tube) side of the heat exchangers to remove mud and biological growth and serves non safety-related function.

The main condenser evacuation system is not assigned a safety class as it serves no plant safety function. It can be used during reactor cooldown or following a turbine-generator or reactor trip when main steam is bypassed to the condenser; but is not necessary to have the condenser available, as discussed in UFSAR 10.4.1, "Main Condensers" to have a safe reactor shutdown under these conditions. No Technical Specification changes are required. Prior NRC approval is not required.

TO/1/A/9600/127

This activity supports higher auxiliary letdown flow through a parallel demineralizer alignment. The higher flowrate is intended to be used infrequently and for a short duration (start-up and shutdown) in order to reduce crudburst clean-up duration and/or to reduce general area dose rates while on aux-letdown.

The auxiliary letdown and NV system purification/filtration functions are not safety related. The Technical Specifications do not address components within the auxiliary letdown or NV purification flowpath, thus margin of safety as defined by the Technical Specifications is unchanged. A parallel demin alignment is required to support the higher aux-letdown flows, to assure adequate purification. Establishment of the parallel demin alignment and ensuring proper boron equilibrium prior to placing in service is supported by other approved station procedures. The higher purification flowrates were evaluated and determined to be acceptable for piping/components within the auxiliary letdown flow path. The higher flowrates will permit a more expedient clean-up of the reactor coolant system during crudburst clean-up, thereby saving critical path time. The flowrate was previously limited by "nominal" steady-state design flow values documented in the UFSAR/Design Basis Document for various components within the auxiliary letdown flowpath. The components/piping within the auxiliary Residual Heat Removal (ND) letdown flow path was evaluated and determined capable of withstanding the higher purification flowrate for infrequent operation (start-up, shutdown evolutions of short duration). The evaluation further determined that the higher flowrates did not present any new component failure modes and that no component degradation would result (e.g. erosion, waterhammer, etc.) The aux-letdown piping is seismically designed, and sufficiently rugged to support the higher flowrates.

The UFSAR analysis postulates a line break in the normal letdown line downstream of the outboard containment isolation; however, this accident analyses applies only to normal operation and is not applicable to aux-letdown operation. The higher flowrates are only permitted for auxiliary letdown operation below Mode 4, thus the analysis of UFSAR 15.6.2 is unchanged. Failures of components within with aux-letdown piping and the low pressure purification loop are not postulated by the UFSAR. If a break were to occur in the aux-letdown piping the consequences of failure and mitigation strategy would be unchanged as a result of this modification. The scope of this activity will not degrade any piping nor components, such that the probability nor consequences of any previously analyzed event (accident or malfunction) is not increased. No Technical Specification changes are required. Prior NRC approval is not required.

Minor Modifications Completed Under 10CFR 50.59

MM-12794

This modification installed anchors into a limited number of guide tubes (4 to 6) of a Westinghouse fuel assembly that was manufactured prior to 1985. The anchors were installed into a fuel assembly prior to lifting. The anchors may be removed after the fuel assembly has been moved. This modification will add a note to applicable fuel assembly drawings to allow for the installation and removal of the anchors on an as-needed basis. New drawings will be created showing the anchors and the removal/installation tool details.

The intent of this modification is to repair the top nozzle connection to meet or exceed the original design requirements. To verify, a conservative static load analysis was performed to demonstrate that an acceptable safety margin exists. Assuming only four of the six anchors carry three times the dry weight of one fuel assembly derived this static load. The installation of the anchors into the guide thimbles will not cause any increase in reactivity because the anchors displace water, which reduces reactivity. The only accident of concern is the fuel handling accident in the fuel building. This modification repairs the top nozzle in such a manner that the fuel assembly can be moved utilizing the existing spent fuel handling equipment. In addition, this modification repairs the top nozzle connection to meet or exceed original design requirements. Accordingly, there will be no impact on the fuel handling accident in the fuel building event. The design change to install anchors into guide tubes will not introduce any new failure modes for handling fuel assemblies in the fuel building. The anchors are classified as safety related. No Technical Specification changes are required. Prior NRC approval is not required.

MGMM-12309

This modification revises the Updated Final Safety Analysis Report (UFSAR) Section 6.2.4.2 to specify requirements for a closed system to be consistent with General Design Criteria (GDC) 57 of 10CFR50, Appendix A. This modification also revises the McGuire Technical Specification Bases Section B3.6.3 to delete reference to Standard Review Plan 6.2.4 and instead, reference UFSAR Section 6.2.4.2.

There is no fieldwork associated with this modification. The purpose of the modification is to clarify and to provide consistency amongst various documents regarding the requirements of a closed system for the containment isolation system. The change to the UFSAR will make the discussion consistent with that provided by GDC 57 of 10CFR50, Appendix A. The Technical Specification Bases is revised to reference UFSAR Section 6.2.4.2 instead of the Standard Review Plan 6.2.4 for the requirements of a closed system. The changes to these documents will not require any modification to the facility nor how the plant is operated. The minor modification should reduce any confusion that may have existed by providing a consistent definition for a closed system throughout all applicable documents. No Technical Specification changes are required. Prior NRC approval is not required.

MGTM-0183

This Temporary Modification (TM) removes the existing 40μ Spent Fuel Pool Cooling System (KF) prefilter and replace it with a 70μ filter during 1EOC14, prior to removing the fuel from the reactor vessel. The temporary change will remain in effect through two filter change evolutions. This is an attempt to evaluate the impact of a higher micron rated filter on the filter change-out frequency during a time of expected high particulate entrainment. The KF prefilter is the first filter in the KF System purification loop. This TM will facilitate removal of 100% of particulate $\geq$70μ. There is a 30μ net increase in particulate size passed through the prefilter to the demineralizer. The KF resin trap (next inline restriction) will pass particulate sized at 177μ; a 107μ difference. This difference will not significantly increase the potential for resin trap blockage.

This TM affects the KF system purification loop prefilter which is a non-QA Condition part contained within a QA Condition 2 (Class E, non-safety housing) housing. No credit is taken for this component in the UFSAR Chapter 15 Accident Analyses. The implementation of this TM does not involve any unusual system alignments or operating parameters outside the design basis.

The KF purification loop is designed to maintain water clarity in the refueling water storage tank (FWST), spent fuel pool (SFP), and refueling cavity. Also, this segment of the system maintains the radioactivity concentration in the SFP water such that the dose rate at the water surface is low enough to allow unrestricted area access for station personnel. Final filtration of the purification loop is unchanged, at approximately 2μ with the post filter. There are no postulated accidents that rely on the purification loop for mitigation. No new failure scenarios, mechanisms, or modes are introduced. This TM plays no role in maintaining the SFP Technical Specification minimum water level or boron concentration. The purification loop does not aid in removal of decay heat from the spent fuel assemblies, which is the most risk significant design basis for the KF system. No Technical Specification changes are required. Prior NRC approval is not required.

MGMM-11682

This modification is designed to minimize identified piping movement that has occurred within the Steam Generator Blowdown (SG BB) System following unit transients. To mitigate this action, existing SG BB control valves that are located upstream from the blowdown tank will receive a signal from the auxiliary feedwater (CA) autostart circuitry and close at the same time as the SG BB containment isolation valves. An additional function that will be performed by the SG BB control valves will be to isolate upon receipt of a high radiation signal from 1 EMF-34, Steam Generator Sample Monitor.

These changes are being implemented by installing solenoids to route signals to the SG BB flow control valves. These solenoids will be energized to close the valves when a signal is received from either CA autostart logic or 1EMF-34. All controls will be removed from the SG BB auto-isolation valves and these valves will be placed opened and abandoned. The control switches for these valves will be removed from the main control board 1MC1 and the conventional sampling panel.

Rerouting the control output signal from 1EMF-34, steam generator sample monitor, from closing the existing SG BB auto-isolation valves to the SG BB flow control valves will not adversely impact any system that is important to plant safety. The valves that close presently on the high radiation signal from 1EMF-34 are located in the auxiliary building within piping that has been classified as seismic. The BB piping in the auxiliary building retains the seismic classification to prevent potential flooding by safety related equipment. The valves that are in the turbine building are in non-seismic piping, however no impact on offsite dose exists, as the containment isolation valves are also available for containing any primary to secondary contamination.

The SG BB system transports condensate grade water to the turbine building that has been drained from the steam generators from cleanup through the BB demineralizers. This water is typically not contaminated by primary side radioactivity. 1EMF-34 monitors the activity level and is designed to isolate BB flow prior to reaching levels that could affect off-site dose. This function is not safety related, however it is regulatory required to secure potential radioactive releases from affecting public safety. This action can be performed effectively by the SG BB control valves.

Isolating the SG BB system upon receipt of the CA autostart signal is part of the logic to maintain secondary side inventory during transients that require the utilization of the CA system. Closing the SG BB control valves concurrently by use of this signal is intended to minimize piping movement of this system following the isolation signal. This action permits the high pressure and temperature of the fluid to be relieved in a controlled fashion during system recovery. This change is intended to prevent damage to non-safety related equipment and will not interfere with any equipment important to plant safety. No Technical Specification changes are required. Prior NRC approval is not required.

Miscellaneous Changes Completed Under 10CFR 50.59

Selected Licensee Commitment 16.9.7

Updated Final Safety Analysis Report (UFSAR) Chapter 16, Selected Licensee Commitment (SLC) 16.9.7 provides a regulatory commitment addressing operability and testing requirements for the Standby Shutdown System (SSS). This changes does not alter the configuration of any structure, system or component (SSC), but clarifies the license basis for the SSS instrumentation.

UFSAR Chapter 16 was reformatted in revision 0, dated December 14, 1999. The revision 0 reformat of this SLC inadvertently omitted information regarding SSS instrumentation (a) readout locations, and (b) minimum channels operable. This activity revised this SLC to restore the previously omitted information into the newly formatted commitment. No technical specification changes are required. NRC approval is not required to implement this change.

UFSAR Change

A discrepancy was identified with regard to the design bases of the Fire Protection System. This change clarified locations of fixed fire suppression systems as related to the Standby Shutdown System. Section 9.5.1.1, Item 2 was revised to read, "Provide fire extinguishment by fixed water sprinkler system or Halon 1301 extinguishing systems in selected areas of the plant as identified in the Fire Hazards Analysis section of the Plant Fire Protection Program DBD MCS 1465.00-00-0008."

This section of the UFSAR implies that there is fixed fire suppression in all areas of the plant that contain equipment associated with Train A, Train B, or the Standby Shutdown System. The specific location of the fire suppression system in the plant was determined by the fire hazard analysis. The fire hazard analysis was submitted to the NRC for review as part of the plant licensing process. The NRC reviewed the location and documented acceptance in the plant Safety Evaluation Reports.

The approved fire protection program is maintained and the change does not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. NRC approval is not required to implement this change.

Technical Specification Bases Change

The McGuire spent fuel pool storage racks contain Boraflex neutron-absorbing panels that ensure that reactivity of the stored fuel assemblies is maintained within required limits. It has been observed that after Boraflex receives a high gamma dose from the stored irradiated fuel it can begin to degrade and dissolve in the wet spent fuel pool environment. To address Boraflex degradation, revisions to McGuire Technical Specifications (TS) 3.7.15 – Spent Fuel Assembly Storage and TS 4.3 – Fuel Storage were submitted and subsequently approved by the NRC. These changes provided revised fuel enrichment and burnup requirements which take credit for soluble boron in maintaining acceptable margins of sub-criticality in the spent fuel storage pools. In addition, the changes provided revised criteria for acceptable levels of sub-criticality in the McGuire spent fuel storage pools. The changes were necessary to offset the loss of some boron in the spent fuel storage cell Boraflex panels at McGuire.

The evaluation of this change successfully satisfied the criteria of 10 CFR 50.59. The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained was not affected by these changes. Similarly, no probabilities or consequences of equipment malfunctions were impacted. The possibility was not created for any new type of accident or equipment malfunction. No safety margins were reduced by the proposed changes. NRC approval to implement this change is not required.

Technical Specification Bases Change

This change revises the Catawba and McGuire Technical Specification Bases associated with the ice bed portion of the ice condensers. These changes are associated with a proposed Technical Specification amendment, which includes changes to two Technical Specification surveillance requirements (SRs). The first is a revision of the ice bed chemical analysis and sampling SR. The second is a revision to the ice bed flow area verification SR.

The ice bed chemical analyses and sampling change affects the current Catawba Technical Specification Bases for surveillance requirement SR 3.6.12.3, and McGuire 3.6.12.5. The changes involve clarifying the methodology for the chemical analyses of the ice condenser ice bed (stored ice). Also, this change includes clarification of a new TS SR, added by the proposed TS amendment, regarding sampling requirements for ice additions to the ice bed.

The ice bed flow area verification changes affect the current Catawba SR 3.6.12.1 and McGuire 3.6.12.4, which require a visual inspection of the air/steam flow area within the ice condensers. This TS Bases change clarifies that the visual surveillance program is to provide a 95 percent confidence level that flow blockage does not exceed the 15 percent ITS acceptance criteria. Whereas, the 0.38 inch program required inspection of as few as two flow channels per ice condenser bay, the new program will require at least 33 percent of the flow area per bay to be inspected.

The TS Bases clarifies which structures are to be inspected. The revision limits the structures to be inspected to only include "between ice baskets" and "past lattice frames and wall panels." The TS Bases revision also is expanded to explain why other structures within the ice condenser are not inspected per the SR. Also, the Westinghouse definitions for frost and ice have been added to the TS Bases to explain why frost is not an impediment to air/steam flow through the ice condenser.

The TS Bases changes do not involve any physical changes to the ice condenser, any physical or chemical changes to the ice contained therein, or make any changes in the operational or maintenance aspects of the ice condenser as required by the TSs. NRC approval to implement this change is not required.

Technical Specification Bases Change

Prior to implementation of the Improved Technical Specifications (ITS) at McGuire in November 1998, old McGuire Technical Specification 3.3.2 – ESFAS Instrumentation, provided an option to allow continued operation of the affected Unit until performance of the next channel operational test if an inoperable RWST level channel was placed in the trip condition. The applicable operational test had a monthly surveillance interval. McGuire ITS 3.3.2 provides an option to place an inoperable RWST level channel in the trip condition within 1 hour or be in Mode 3 within 7 hours and Mode 4 within 13 hours. However, unlike the old TS 3.3.2, ITS 3.3.2 allows operation in the trip condition for an indefinite period of time. With no time restrictions, a channel placed in the trip condition cannot be considered the design basis single failure assumed during a design basis accident. Therefore, with one channel of RWST level instrumentation in the trip condition, a single failure of another channel coincident with a design basis LOCA could result in premature automatic swapover of the low head residual heat removal (ND) pumps to the containment sump. For a failure leading to early swapover, plant analyses assume operators do not have sufficient time to resolve the problem prior to pump damage. Consequently, as a result of this premature swapover, both trains of the ND pumps could fail due to insufficient containment sump water level. This could prevent these pumps from performing their post-LOCA cooling function. In addition, since they are dependent on the residual heat removal (ND) pumps for a water supply during the post-LOCA recirculation phase, the safety injection (NI) and chemical and volume control (NV) pumps may not be available to assist in this recirculation cooling function.

On March 6, 2000, McGuire submitted Licensee Event Report (LER) 369/00-02, Revision 0 which identified this Technical Specification deficiency. Duke committed to submit a Technical Specification amendment as a planned corrective action in that LER. This amendment was approved by the NRV via an SER issued April 12, 2001. These changes provide a limitation on the amount of time an inoperable RWST level channel can be in a trip condition. This will minimize the time a Unit is exposed to the possibility of a premature swapover of the ECCS pumps to the containment sump.

This TS Bases change was evaluated in conjunction with the requirements of 10 CFR 50.59. The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained is not affected by this change. Similarly, no probabilities or consequences of equipment malfunctions were impacted. The possibility is not created of any new type of accident or equipment malfunction. NRC approval to implement this change is not required.

Selected Licensee Commitment Change

This Selected Licensee Commitment (SLC) contains the requirements for the fire suppression water system. Conditions C, D and E specify the required actions for an inoperable fire suppression pumps primary and secondary automatic starting functions or jockey pumps unable to maintain system header pressure. In brief, these required actions allow either the primary auto start function, the secondary auto start function, or one fire suppression pump in continuous operation to satisfy the auto start requirement for the required fire suppression pump(s). There are threee fire suppression pumps – A, B and C. This SLC requires fire suppression pump C and one other fire suppression pump to be operable. Conditions and Required Actions C, D, and E are revised to reflect that they are required for the required pumps only. This is accomplished by inserting the word "required" into these conditions and required actions. Some re-arrangement of the existing words are also made for clarity. The associated Bases is also revised to clarify that the primary auto start circuit will start fire pumps at higher pressure setpoints than those associated with the primary and secondary auto start functions and the continuous operation of one fire pump into this SLC. These editorial changes to not pose any safety concerns. NRC approval to implement this change is not required.

New Selected Licensee Commitment

The McGuire spent fuel pool storage racks contain Boraflex neutron-absorbing panels that ensure that reactivity of the stored fuel assemblies is maintained within required limits. It has been observed that after Boraflex receives a high gamma dose from the stored irradiated fuel it can begin to degrade and dissolve in the wet spent fuel pool environment. To address Boraflex degradation , the spent fuel racks have been analyzed taking credit for soluble boron and reduced credit for the degraded Boraflex neutron absorber panels. This new SLC16.9.24 provides for periodic monitoring of future actions to ensure the reactivity of the stored fuel assemblies is maintained within required limits.

These changes successfully satisfied the criteria of 10 CFR 50.59. The manner in which the plant and its accident mitigating equipment are designed, operated, and maintained was not affected by these changes. Similarly, no probabilities or consequences of equipment malfunctions were impacted. The possibility is not created for any new type of accident or equipment malfunction. NRC approval to implement this change is not required.

Selected Licensee Commitment Change

SLC 16.9.7 contains the requirements for the Standby Shutdown System (SSS). Table 16.9.7-1 lists the SSS components that are required to be operable in Modes 1, 2 and 3. One of these components is the SSS 250/125V battery and battery charger. There is currently inadequate information in the SLC to determine what is required for the SSS 250/125V batteries and battery chargers. The SLC Bases is revised to clarify that only two out of three battery-battery chargers pairs are required to be operable by the SLC. No plant structure, systems or components are modified or caused to operate in a different manner from current operation. No regulatory commitments are being eliminated or reduced by this change. The change goes not pose any safety concerns. NRC approval to implement this change is not required.

UFSAR Change

The criteria for Residual Heat Removal (ND) swapover to the auxiliary reactor building spray header is to delay the swapover until latent and decay heat levels in the reactor coolant 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) prior to ice bed meltout to prevent a containment pressure spike. At 50 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 ECCS water temperature returning to the reactor vessel at 50 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 50 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 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. The use of the phrase "at approximately 50 minutes" allows for the variation in operator times when performing the EOP steps for aligning ND to NS. The difference between the heat loads at 45 and 50 minutes after the postulated LOCA is less than 4% and therefore, the statements in section 3.3.3.9 are applicable at 45 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 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. NRC approval to implement this change is not required.

Selected Licensee Commitments Change

Selected Licensee Commitments 16.5 and 16.5-4, Reactor Coolant System (RCS) Instrumentation – Reduced Inventory Operation, refer to the RCS level instrumentation consisting of wide range reactor vessel level instrumentation system (RVLIS) and narrow range RVLIS. This nomenclature is misleading due to the number of instrumentation and the time period during reduced inventory that additional instruments is available or not available for level indication. The SLC should have been updated when additional instrumentation for level indication was added to the system. The performance and operational characteristics of the RCS level, as described in the UFSAR, are not affected by this change.

The RCS level instrumentation is used for accident mitigation and is not an accident initiator. Adding additional instrumentation will not increase the probability of occurrence of an accident previously evaluated in the safety analysis report. Additional instrumentation will not affect the probability of a failure and therefore, the probability of malfunction of equipment important to safety is not increased, but decreased. The consequences of an accident, as previously evaluated will not be affected. The equipment will be capable of operation as described in the UFSAR. The consequences of a malfunction are not increased beyond that addressed in the UFSAR.

This change does not change the operating characteristics or methods for reduced inventory. This change does not change the performance parameters of the equipment and thus does not affect the design parameters or the capability of the equipment. NRC approval to implement this change is not required.

Reactor Core Power Distribution Software Change

This change involves an upgrade to software used for offline core power distribution monitoring. The current software, COMETO2 is being replaced by COMETO3.

COMETO3 is an improved version of COMET02. The new software incorporates various enhancements that do not affect the fundamental methodologies used to infer power distribution or calculate Technical Specification margin.

COMETO3 was certified per Duke Power's directive for software certification and verified to yield the same results as COMETO2, except for the new modifications. The modifications were verified and, as applicable, are in compliance with the Improved Technical Specifications or submitted revisions. COMETO3 is, therefore, considered equivalent to COMETO2.

The change involves no material changes to the plant. The COMET software and resident workstation are not part of any SSC important to safety and does not directly affect any SSCs. The three systems indirectly associated, the Movable Incore Detector System, the excore power range detectors, and the Reactor Protection System, are all unaffected by this change. The only safety significant function performed by, or involved with, this system is to generate data to evaluate ITSs 3.2.1, 3.2.2 and 3.2.4, and to periodically calibrate the power range AFD indications, are required per SR 3.3.1.6 (Table 3.3.1-1). The new software is functionally equivalent to the replaced software and yields the same analytical results. Therefore, assurance of the fuel integrity limits associated with the ITSs and the OPΔT and OTΔT reactor trips (AFD parameter input) are not compromised. This change does not impact any plant parameters, safety limits or setpoints that potentially affect the fission product barriers. NRC approval to implement this change is not required.

Nuclear Physics Parameters and Power Distributions for Core Designs Software Change

This change involves the use of an upgraded version of the reactor physics code used for reload design. SIMULATE-3P is an advanced two-group nodal code based on the QPANDA neutronics model. This code is used for the calculation of steady state and transient power distributions, control rod worths, integral reactivity worths, reactivity coefficients, and kinetics data. TABLES-3 is the data processing code that links CASMO-3 cross section data to SIMULATE-3. The CASMO-3/SIMULATE-3P codes package is the NRC approved methodology used by Duke Power for calculating nuclear physics parameters per topical report DPC-NE-1004A. SIMULATE-3P, Version 6 and TABLES-3, Version 5 were certified in accordance with Duke Power's directive for software certification and verified to yield the same results as SIMULATE-3P Version 4 and TABLES-3 Version 4 except for the modifications. The modifications were verified in compliance with the Technical Specifications and approved methods.

The SIMULATE-3P software, and workstations where is resides, are not part of any SSC important to safety and do not directly affect any SSC. The software is not installed at the plant, but rather on workstations in the general office. Regulatory commitments associated with this software have been reviewed and determined acceptable. A reactivity management review was performed with no concerns identified. This change does not adversely affect any design bases, safety functions, safety limits, safety margins, setpoints, or core parameters. The capability to shutdown the plant and maintain it in a safe condition is unaffected. NRC approval to implement this change is not required.

Change to McGuire Unit 2, Cycle 13, Core Operating Limits Report (COLR)

This evaluation involved a change to the McGuire Unit 2, Cycle 13, Core Operating Limits Report to lower the minimum required refueling boron concentration for the Reactor Coolant System (NC) only. The refueling operations specification is applicable for the reactor coolant system, refueling canal and the refueling cavity for Mode 6 conditions (TS 3.9.1). The minimum boron concentration limit and plant refueling procedures ensure that the Keff of the core will remain within the mode 6 reactivity requirement of Keff<=0.95.

The methodologies used to determine the require minimum boron concentration are consistent with the methodologies described in TS 5.6.5 which have been approved by the NRC.

McGuire 2 is currently shutdown for refueling and preparing to enter Mode 6 for core offload. The plant is experiencing some difficulties borating the reactor coolant system up to the current limit of 2675 ppm, the option of reducing the required boron concentration for mode 6 is being evaluated.

The current COLR refueling boron concentration limit of 2675 ppmb is very conservative relative to the cycle specific value and is acceptable to reduce the NC system refueling boron concentration from 2675 to 2500 ppmb during core offload.

The only analyses potentially impacted by this change are the boron dilution and post LOCA boron analyses. The reduction in required boron concentration from 2675 to 2500 ppmb does not impact either of these analyses evaluated in the M2C13 SAPP since these events are limiting at BOC and the M2C13 core burnup is at approximately 479 EFPD. Current boron concentration limit assumed in the McGuire criticality analysis and the SFP boron dilution accidents is 2475 ppmb. A conditional limit for the NC system boron concentration of 2500 ppbm coupled with site actions during refueling activities precludes the SFP boron concentration from being reduced below its 2675 ppmb limit. NRC approval to implement this change is not required.

Data Processing Code for Calculating Design Margin for Fuel

This change involves an upgrade to the Super-MARGINS (SMARGINS) software. The current software, SMARGINS, version 7 is being replaced by version 8.

SMARGINS is a data processing code which compares results from nuclear design calculations using SIMULATE-3 to fuel design limits, and thereby verifies acceptability of the applicable limits for the core design. SMARGINS also edits a variety of data based on SIMULATE-3 results. SMARGO08 was certified in accordance with Duke Power's directive for software certification and verified to yield the same results except for the modifications.

The SMARG08 software, and workstations where it resides, are not part of any SSC important to safety and do not directly affect any SSCs. The software is not installed at the plants, but rather on workstations in the general office. There are no regulatory commitments associated with this software. A reactivity management review was performed with no concerns identified. No new methods are introduced by this code revision. This change does not adversely affect any design bases, safety functions, safety limits, safety margins, setpoints, or core parameters. The capability to shutdown the plant and maintain it in a safe condition is unaffected. NRC approval to implement this change is not required.