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Subject: Duke Energy Carolinas, LLC
McGuire Nuclear Station
Docket Nos. 50-369, 370
Summary Report of Evaluations Performed Pursuant to 10CFR 50.59, Changes, Tests, and Experiments

Pursuant to 10 CFR 50.59(d)(2) attached is a summary report of evaluations performed at McGuire Nuclear Station for the period ending August 31, 2008. These evaluations demonstrate that the changes do not meet the criteria for a license amendment as defined by 10 CFR 50.59(c)(2).

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

for

Bruce H. Hamilton

Attachment

IE47
NRR

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McGuire Nuclear Station (MNS)
Changes Completed via 10 CFR 50.59

MD-200375

Phase 1 of the new Containment Recirculation Sump Strainer Assembly is installed in containment on the 725 ft elevation for McGuire Unit 2. The strainer assemblies replace the existing sump screen and trash rack structure and are located in the existing sump location in the pipe chase outside the crane wall and on the inside of the crane wall at the 180° azimuth. The strainer surface area for Phase 1 is approximately 1100 ft^2 (gross).

The existing Containment Recirculation Sump Strainer Assembly is ¼ inch wire mesh screens protected by steel grating trash racks and a steel plate. It is replaced by a strainer design with multi-tube "Top-Hat" modules, including a bypass eliminator feature. The Top-Hat cylinders are fabricated from perforated plate and mounted in a horizontal orientation within the screen area and extending outward. The strainer tube modules have two layers of perforated plate for straining debris from the water. The Top-Hat module has an outer diameter of approximately 8 inches and an inner diameter of approximately 6 inches with a length varying from approximately 2 feet to 4 feet depending on specific existing interferences. The bypass eliminator feature consists of a wire mesh filter element installed inside the Top-Hat between the inner and outer diameter perforated plates. The bypass eliminator is used to capture the fibrous debris passing through the strainer screen and enhance the performance of the strainer. The new strainer elevation is 727 feet 10 inches. The sump level switches used to determine suitability for recirculation initiation are set at 727 feet 11 inches to account for instrument uncertainty. Calculated minimum water level for a small break LOCA is 728 feet. Water enters the Top-Hats through the perforated plate screens and flows through each Top-Hat. Upon exiting the Top-Hats, water will flow into the water box plenum. There is a cross over plenum that attaches to each water box. A divider barrier in each crossover plenum provides separation between the intake trains.

The strainer design in the pipe chase (outside the crane wall) includes an assembly installed horizontally directly above the Top-Hat assemblies and vertically on the sides of the strainer assembly. The horizontal assembly is constructed of floor grating and installed directly over the Top-Hat strainer assemblies located outside the crane wall. Since the pipe chase outside the crane wall is used for the passage of personnel and materials, this grating is designed to prevent damage to the Top-Hat assembly. The floor grating above the Top-Hat assemblies also serves as a vortex suppressor. The vertical assembly is constructed in a grid pattern along side the Top-Hat strainer assemblies located outside the crane wall with approximately 6" x 6" openings.

The strainer assemblies located inside the crane wall are completely enclosed to maintain the remote sump feature. This inside the crane wall enclosure is designed with perforated sides and a solid top such that these modules are vented. The enclosure is removable to permit access to the Top-Hats for inspections, etc.

The new sump structure is Duke QA-1 seismically designed and qualified to all design environmental conditions in the sump. All parts of components in contact with the sump solution during recirculation are fabricated of austenitic stainless steel or other qualified material. The new strainer does not impact any existing plant system chemistry requirements or the ability to sample water. The sump enclosure inside containment is designed for the sub-compartment differential pressures associated with a pipe break.

The revised Emergency Core Cooling System (ECCS) sump is a passive structure as before. It is not an accident initiator. The revised sump is designed with increased screen area and smaller screen mesh along with a secondary debris bypass eliminator. The likelihood of an equipment malfunction is reduced by the improved capacity of the sump. There is no change to the consequence of a malfunction of equipment. The sump is designed to ensure no air entrainment or vortex issues exist when water is at minimum flood levels for initiation of ECCS recirculation. The revised sump is qualified for environmental conditions such that the passive structure is not more likely to malfunction. The containment sump is not considered as a passive failure in the Updated Final Safety Analysis Report (UFSAR). The Residual Heat Removal and Containment Spray systems operate as before and there is no degradation of ECCS system performance. There is no change to accident consequence as a result of the sump change. The containment analysis is not adversely affected by this modification. There is no effect on the fuel or reactor coolant boundary. The sump is fully qualified using the current licensing evaluation methodology.

UFSAR Tables 15-12 and 15-33

This activity evaluated a change to the Updated Final Safety Analysis Report (UFSAR) environmental consequences for the Loss of Coolant Accident (LOCA) located in Chapter 15 Section 15.6.5, in particular the input and results presented in Tables 15-12 and 15-33.

The analysis supporting the LOCA dose consequences was revised to model a reduction in the Containment Spray (NS) flow from 3400 to 3325 gpm. As a result, the iodine spray removal time constraints were reduced and an increase in the predicted dose calculated. The dose increase did not satisfy the more than minimal criteria detailed in NEI 96-07, "Guidelines for 10 CFR 50.59 Implementation", necessary to require a license amendment request.

MD200996

The activity evaluated is a change to allow use of HFD type breakers in the place of obsolete HFB breakers for the 2EMXD breakers listed in Selected Licensee Commitments (SLC) Table 16.8.1-2. The continuous rating for three breakers being replaced is being increased to the next respective size. This increase will not affect the ability to provide electrical protection for the circuit. Breaker coordination will still be maintained with these increases. The curves between the HFD and HFB breakers are not identical. As a result, the thermal tripping times are different. SLC Table 16.8.1-2 contains response time for breakers (in seconds) at 300% of continuous rating. The response time for the HFD series breaker is generally greater than that of the HFB series and the table requires revision to reflect the increased response time. The basis for the response time is to test the breakers to manufacturer specified criteria and verify proper operation. Even though the table revision will reflect the maximum response time, whether it is that of the HFB or HFD, actual testing will be procedurally controlled to assure the breakers are tested to their respective test values. The increase in time has been evaluated and found that it will not challenge the mechanical integrity of its associated electrical penetration.

MD200998

The McGuire corrective action program documented that the Unit 2 Train B containment sump isolation valve, 2NI-184B, was leaking approximately 1.5 gallons per day of borated water for the duration of Cycle 14. With the addition of the new Emergency Core Cooling System (ECCS) containment sump structure in outage 2EOC17, the leakage could potentially plug some of the new sump screening (through boron crystallization).

MD 200998 adds double isolation connections to the inlet piping for the containment sump to the containment isolation valves 2NI-184B and 2NI-185A. These valves are containment isolation valves (CIVs) and will be locked closed and capped to ensure containment integrity during normal operation.

The connections are added to the ECCS suction piping. The design function of the ECCS, in the injection and recirculation modes, and the proper functioning of CIVs 1/2NI-184B and 185A are unaffected by the installation of the engineering change. The installation of this change does not change the procedures or the sequence of operator actions by which the ECCS is operated in either the injection or recirculation mode. The change will not result in a change to the procedure by which containment spray may be supplied.

The integrity of the portion of the containment sump line outside containment to valves 2NI184B and 2NI185A (including installation of the change) is consistent with USNRC Standard Review Plan (NUREG-0800), Section 6.2.4, paragraph II(d).

There are no credible failure mechanisms introduced which change the outcome of previously performed accident analyses or present new mechanisms which need to be evaluated.

MD100999

MD100999 adds double isolation connections to the inlet piping from the containment sump to the containment isolation valves (CIVs) 1NI-184B and 1NI-185A. These connections are vents and drains. MD100999 converts the vents and drains to isolation valves for a new drain system to allow manual draining of the accumulated leakage. This protects the new sump structure by adding the capability to drain the borated water leaking through 1NI-184B and 1NI-185A before the level increases to the point that it flows to the containment sump.

The drain system consists of a (ASME Section III Class 2, Duke Class B) reservoir constructed from standard twelve inch stainless steel pipe with welded end caps. The reservoir volume is approximately 4.8 cubic feet (36 gallons). The reservoir was seismically analyzed and mounted on the auxiliary building wall. The reservoir has QA-1, Duke Class B, double isolation; locked-closed, manual ball valves connected for vent and drain purposes. The reservoir valve assemblies are further isolated with a quick-connect instrument fitting with a cap. These manual ball valves are CIVs because they maintain the pressure boundary between containment and the auxiliary building whenever the sump piping to reservoir isolation valves are opened. The reservoir vent and drain valves are locked closed to ensure containment integrity. This change ensures containment integrity during normal operation.

The connections are added to the ECCS suction piping. The design function of the ECCS, in the injection and recirculation modes, and the proper functioning of CIVs 1NI-184B and 185A are unaffected by the installation of the change. The installation of the change will not change the procedures or the sequence of operator actions by which the ECCS is operated in either the injection, or recirculation mode. This engineering change will not result in a change to the procedure by which containment spray may be supplied.

The design of this change is such that a line break does not have to be postulated for the change or its attachment to the ECCS suction piping. The integrity of the portion of the containment sump line outside containment to valves 1NI184b and 1NI185A (including installation of proposed engineering change) is consistent with USNRC Standard Review Plan (NUREG-0800), Section 6.2.4, paragraph II(d).

There are no credible failure mechanisms, introduced by the change, which change the outcome of previously performed accident analyses or present new mechanisms which need to be evaluated.

MD500719
MD100848

The 5 year fault current models indicate that by 2008 projected fault currents would exceed the ratings for the McGuire 230 kV switchyard breakers. In order to reduce the available fault current in the McGuire switchyard, the Dutchman White and Dutchman Black lines to the Lincoln Combustion Turbine (CT) plant and the Schoonover White and Schoonover Black lines to the Riverbend coal plant were removed from the Unit 1 switchyard. The Dutchman and Schoonover transmission lines were tied together outside the McGuire switchyard, creating a direct double-circuit tie between Riverbend and Lincoln CT. A tap from the White line was connected to the McGuire switchyard, through power circuit breaker (PCB) 30, to the Yellow 230 kV bus. PCB 30 will be maintained in the OPEN position, except when needed for a "blackstart" event.

The design of the offsite power connection to the plant distribution system is dictated in part by GDC-17, which requires the onsite electric distribution system to be supplied electric power from the transmission network by two physically independent circuits. Each of the two offsite circuits is designed to be available to assure that specified fuel and reactor coolant system pressure boundary design limits are not exceeded in the event of abnormal occurrences, and the core-cooling and containment integrity are maintained in the event of a design basis accident.

Operational restrictions on the offsite power system, which are based upon the GDC 17 design requirements, are stipulated in Technical Specification (TS) 3.8.1 (Operating) and 3.8.2 (Shutdown). TS Action 3.8.1.A.3 requires that with one offsite circuit inoperable, the remaining circuit be restored to operable status within 72 hours and TS Action 3.8.1.C.2 requires that with two offsite circuits inoperable, one of the offsite circuits be restored to operable status within 24 hours.

The adequacy, and thus operability of the offsite power system is dependent in part upon the ability of the transmission system to be of sufficient voltage and capacity to power safety-related loads under defined accident conditions. If the grid is unable to supply the required voltage to the 4160-volt safety busses, degraded voltage relaying will isolate the safety busses from the offsite source, and initiate the emergency load sequencer to supply the required loads from the emergency diesel generators. Thus, degraded grid conditions can result in unwanted challenges to the safety-related onsite power system.

An evaluation of the proposed activity considering impact on the physical configuration changes, voltage adequacy of the offsite power source, system stability and protective functions has demonstrated that the proposed activity is within the current licensing bases and can be implemented via the 10CFR50.59 process.

MD-500739
MD-500740

The control room HVAC chiller (YC) controls are upgraded using a commercial off-the-shelf (COTS) digital controller. The selected device is a Trane™ Tracer CH531 controller, a mature product line that has a substantial user community and a good reliability record. The controller and supporting items have been qualified for use in nuclear safety systems by a third-part qualifier following published industry guidelines endorsed by the NRC.

No changes to the design basis accident mitigation functions of the control room ventilation chilled water system (VC/YC) are being made in connection with this modification. No changes to the VC system air handlers, heat exchangers, filters, damper controls, or thermostatically controlled three-way bypass valves are included as part of this modification. The control room continues to meet the habitability requirements in GDC-19 following this modification.

The modification changes the manner in which the YC chillers are controlled, including changes to the human machine interface (HMI) at the controller, changes to maintenance and abnormal operating procedures, and changes to the associated training for licensed and non-licensed operators. However, only minor changes are required in the control room, and the HMI changes at the YC chillers improve the ability of maintenance technicians to correctly identify and diagnose chiller problems. The impact of changes in the HMI is managed by careful design of procedures and training, incorporating human factors engineering principles endorsed by the NRC.

The modification introduces some potential new failure modes that do not exist in the legacy analog control system being replaced. However, these have been addressed by analysis and by testing, and demonstrated to be non-credible or manageable. The reliability of the replacement system has been studied using both FMEA and FTA methods. It has been concluded that new controls system gives a substantial improvement in reliability, even when the new potential failure modes are taken into account, thereby offering a new safety benefit.

The results of the review demonstrate that the proposed modification does not affect the current licensing basis adversely, and does not create more than a minimal increase in the likelihood or consequences of a failure of the control room HVAC system.

McGuire Unit 2 Cycle 18 Reload Core Design

This evaluation was performed for the McGuire Nuclear Station Unit 2, Cycle 18 (M2C18) core reload.

The M2C18 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Nuclear Engineering Division workplace procedure NE-102, "Workplace Procedure for Nuclear Fuel Management", and the M2C18 Reload Safety Evaluation confirm the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses remain bounding with respect to the M2C18 safety analysis reactor physics parameters. The safety analysis reactor physics parameters method is described in topical report DPC-NE-3001-PA.

The M2C18 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. The M2C18 Core Operating Limits Report was prepared in accordance with Technical Specification 5.6.5 and submitted to the NRC in accordance with 10 CFR 50.4. Additionally, applicable sections of Technical Specifications and the UFSAR have been reviewed, and no changes to the Technical Specifications or UFSAR are necessary for M2C18.

Technical Specification (TS) Bases 3.9.3 and UFSAR 15.4.6

This activity involved changing the Technical Specification Bases 3.9.3, "Nuclear Instrumentation" and UFSAR 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant". Other sections of the UFSAR that are updated to reflect the revised operability requirements for the source range nuclear instrumentation during Mode 6 can be summarized as follows:

1. Continuous visible count rate indication in the control room

2. Audible count rate and high flux alarm in the control room with the alarm setpoint at 0.5 decade above the steady state count rate

3. Audible count rate indication in containment and an automatic high flux containment evacuation alarm

The changes to the TS Bases and UFSAR redefine the operability requirement for the source range nuclear instrumentation during Mode 6. In short, the change deletes operability requirement Items 2 and 3 above and operability requirement Item 1 is modified to state that a visible indication be provided in the control room.

NUREG-1431, Volume 2, Revision 3.0 (Standard TS Westinghouse Plant Bases), provides TS Bases information and regulatory guidance for the nuclear instrumentation in Mode 6. As stated in the NUREG, for plants that isolate boron dilution paths, the operability requirement for the source range nuclear instrumentation includes only a visual monitoring function. Further, the possibility of an inadvertent boron dilution event during refueling operations is precluded by requiring that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the Reactor Coolant System (RCS). By isolating unborated water sources, an uncontrolled boron dilution accident is not feasible. This change aligns McGuire with the regulatory criteria and guidance provided by NUREG-1431.

This change also deletes the requirement for automatic containment evacuation alarm and the audible count rate function in containment provided by the source range nuclear instrumentation. The evacuation alarm provided by the Reactor Building Refueling Bridge Radiation Monitor, as well as manual initiation of the containment evacuation alarm will remain unaffected. By isolating unborated sources of water to the RCS, an uncontrolled boron dilution accident is not possible. With isolation of all sources of unborated water per TS 3.9.2, there is no need to include an audible count rate in the reactor building of an automatic containment evacuation alarm function as an operability requirement for the source range detectors while in Mode 6.

The source range detectors only monitor neutron flux and they do not perform any function that could result in unborated water to be injected into the RCS. The deletion of the audible count rate and high flux at shutdown alarm, the audible count rate in containment and the automatic containment evacuation alarm as an operability requirement will have no affect on the source range nuclear instrumentation systems ability to continuously monitor the reactor core for reactivity changes and to provide a visible indication of the count rate in the control room. The source range detectors will continue to monitor the neutron flux from the reactor core since that portion of the nuclear instrumentation circuitry remains unaffected. The revised operability requirements do not create any new accident or failure mode, impact the consequences or make any previously evaluated accident more likely.

MD100165
MD200166

The scope of these modifications is to improve the reliability, security and monitoring capabilities of the protective relaying for the main generators by replacing the existing protective relays with a set of multifunction microprocessor-based relays. In addition to replacing the protective functions for the existing relays, the new microprocessor-based relays will include protective functions that are not provided by the existing relaying scheme.

An evaluation of the proposed protective relaying modification has demonstrated that the change will have no adverse impact on any structure, system or component (SSC) with accident mitigation functions, and that the ANS Condition

II transients that may be initiated by the protective relaying will remain bounded by the present UFSAR analysis.

New potential failure modes and increased susceptibility to existing failures have been evaluated. These include common mode software failures, failures resulting from changing the coincident trip logic, and EMI/RFI induced failures. When considering the features of redundant trains, independent channels, diversity, self-diagnostics, and improved human-machine interfaces, the proposed relaying scheme proves to be as reliable as and more secure than the existing system.

The proposed protective relaying scheme contains protective functions that are not provided with the existing scheme. These include inadvertent energization, generator neutral ground third-harmonic differential and out-of-step protection. Each of the new protective features has been evaluated to ensure the proposed changes will not lead to additional challenges or otherwise adversely impact SSCs that serve to mitigate transients initiated by protective relay action.

The evaluation demonstrates that the proposed main generator protective relaying modification will preserve the current licensing basis. The activity will not create more than a minimal increase in the frequency or consequences of accidents or malfunctions of SSCs important to safety. The proposed activity will not create the potential for a new type of unanalyzed event, has no impact on the fission product barriers and does not affect evaluation methodology.

EP/1/A/5000ES-0.1
RP/0/A/5700/007

The current licensing basis for McGuire requires a minimum level of Lake Norman at 745.0' MSL, which is to provide sufficient net positive suction head (NPSH) for the nuclear service water (RN) system pumps. However, compensatory measures restricting lake level to not less than 752' have been in place since 2004 in order to prevent introduction of air-entrained water to the "A" train auxiliary feed water (CA) pump from "A" train RN, following a safe shutdown seismic event (SSE). Modifications to eliminate the air entrainment problem and this temporary restriction have been completed.

However, new potential air entrainment concerns have been identified recently, related to a feature in the Standby Shutdown Facility (SSF). The SSF is not credited for any function following a SSE, but it has features designed to provide a source of water to the Auxiliary Feedwater (CA) pumps during other events. The additional concerns require new compensatory actions to be taken, including imposing a new temporary restriction on minimum lake level of at least 750 ft. MSL, in order to provide an assured source of feedwater, free of entrained air.

To address these concerns, enhanced procedural guidance is added to the emergency and abnormal operating procedures for post-seismic reactor trip and seismic event response. An evaluation was performed in accordance with site directives governing assessments of operability and functionality and the application of compensatory actions. The results of the evaluation documented that these procedure changes are consistent with existing operator training, and can be accomplished entirely within the control room and in the time required.

SLCs 16.9.9 & 16.9.12 and Bases

The boration subsystem of the Chemical and Volume Control (NV) system provides the means to meet one of the functional requirements of the NV system, i.e., to control the chemical neutron absorber (boron) concentration in the reactor coolant system (RCS) and to help maintain the shutdown margin.

The requirements for boration flow paths were relocated form the McGuire Technical Specifications to the Selected Licensee Commitments (SLC) requirements in UFSAR Chapter 16 in 1998, as part of the Improved Technical Specifications (ITS) changes authorized in License Amendments 184/166.

The SLC specified two boration flow paths be functional when the plant is operated at elevated temperature and pressure, but relaxes the requirement to one flow path when reactor coolant system (RCS) temperature is below 300 degrees F, which is midway through MODE 4.

Technical Specification 3.4.12 governing low temperature over-pressure protection (LTOP) requires the minimum LTOP enable temperature be set to 300 degrees F, which conflicts with the SLC requirements for minimum boration flow paths in MODE 4. At 300 degrees F and below, the Technical Specifications require that only one centrifugal charging pump (CCP) be capable of injecting to the RCS, in order to limit the effect of a potential pressurized thermal shock (PTS) event.

Therefore, the SLC requirement for two boration flow paths has been eliminated from MODE 4 in order to relieve this conflict with the Technical Specifications. This change was evaluated using guidance published in NEI 96-07, Revision 1. The evaluation considered the positions outlined by the NRC in its safety evaluation report related to License Amendments 184/166, and it was determined that the change is consistent with those positions.

OP/1/A/6400/006, Rev 190
OP/2/A/6400/006, Rev 146

Under certain Nuclear Service Water (RN) system alignments with high RN pump flows, the pressure in the RN pump strainer can experience a pressure drop to sub-atmospheric. If the pressure in the strainer becomes sub-atmospheric, while RN strainer backwash operations to the Auxiliary Building (WZ) sump are in progress, this alignment can introduce a potential hazard in that air might be drawn into the backwash discharge line, and entrained in the RN system flow.

Air entrainment can be deleterious if the RN assured supply connection to the auxiliary feedwater (CA) system is opened, and air migrates to the suction header of the CA pumps.

To prevent this hazard, the RN system operating procedures were revised to require checking for sub-atmospheric conditions, visual monitoring the discharge flow into the WZ sump, and to manually discontinue backwash operations to the sump in the event sub-atmospheric conditions are detected.

MD100461

The Nuclear Service Water (RN) assured makeup to the Auxiliary Feedwater (CA) system is a standby portion of the RN system that is provided as the seismically qualified source of auxiliary feedwater to supply the steam generators following an accident. The RN system is designed to withstand a safe shutdown earthquake and to prevent any single failure from curtailing normal station operation or limiting the ability of the engineered safety features to perform their functions. This engineering change has no significant effect on the outcome of previously existing accident analyses and presents no new failure mechanisms that need to be evaluated.

The Unit 1 Train A CA system supply is being relocated to the supply side of the Train A Emergency Diesel Generator Heat Exchanger (KD HX) with a locked open isolation valve. The new Unit 1 Train A RN supply will allow the CA system to be fed with RN water prior to heating up and will increase pressure to the assured CA system supply. The cooler higher pressure supply will eliminate the concern for dissolved gases coming out of solution and being swept into the pumps.

The relocated supply creates two additional concerns, one with the flow balance of the RN system due to an increase in the RN system demand and another with an increased differential pressure (dp) across the RN and CA system isolation valves. A test line is being added to allow a balance of the system to be performed with an additional flow demand to the RN system. The test piping is incorporated into the discharge piping for the Unit 1 and 2 Train A RN strainer backwashes such that the flow testing will discharge to Lake Norman. The RN strainer backwashes are automatic functions which will remain operable during flow balance testing and share the same flow path to the lake. The increased dp will require that valve 1RN-69A be replaced with a valve having a larger actuator and valve 1CA-86A will have a new larger actuator.

MD201017

The Unit 2 Train B containment sump isolation valve, 2NI-184B, was discovered leaking approximately 1.5 gallons per day of borated water during cycle 14. With the addition of the new Emergency Core Cooling System (ECCS) containment sump structure in outage 2EOC17, the leakage can potentially plug some of the new sump screening (through boron crystallization).

Two engineering changes were created to resolve the problem. The first change added two double isolation connections to the inlet piping from the containment sump to the containment isolation valves (CIVs) 2NI-184B and 2NI-185A. These MD200998 connections are vents and drains. Valves 2NI-184B and 2NI-185A are installed in a vertical orientation in the auxiliary building. The drain connections are approximately 1 foot above 2NI-184B and 2NI-185A. The vent connections are approximately 3 feet above 2NI-184B and 2NI-185A on the elbows to the horizontal piping runs. The second engineering change (MD201017) converts the vents and drains to isolation valves for a new drain system to allow manual draining of the accumulated leakage. This protects the new sump structure by adding the capability to drain the borated water leaking through 2NI-184B and 2NI-185A before the level increases to the point that it flows to the containment sump.

This evaluation examined the second engineering change (MD201017) which adds a new drain system to the previously installed double isolation connections. The new drain system consists of a (ASME Section III Class 2, Duke Class B) reservoir constructed from standard twelve inch stainless steel pipe with welded end caps. The reservoir volume is approximately 4.8 cubic feet (36 gallons). The reservoir is seismically analyzed and mounted on the auxiliary building wall. The new reservoir will have QA-1, Duke Class B, double isolation, locked-closed, manual ball valves connected for vent and drain purposes. The reservoir valve assemblies are further isolated with threaded pipe caps. These manual ball valves are CIVs because they maintain the pressure boundary between containment and the auxiliary building whenever the sump piping to reservoir isolation valves are opened. The reservoir vent and drain valves are locked closed to ensure containment integrity. This design ensures containment integrity during normal operation.

The connections are added to the Emergency Core Cooling System (ECCS) suction piping. The design function of the ECCS, in the injection and recirculation modes, and the proper functioning of CIVs 2NI-184B and 185A are unaffected by the installation of the engineering change. The installation of this engineering change will not change the procedures or the sequence of operator actions by which the ECCS is operated in either the injection, or recirculation mode. This engineering change will not result in a change to the procedure by which containment spray may be supplied.

The design of this engineering change is such that a line break does not have to be postulated for the engineering change or its attachment to the ECCS suction piping. The integrity of the portion of the containment sump line outside containment to valves 2NI-184B and 2NI-185A (including installation of proposed engineering change) is consistent with the Standard Review Plan (NUREG-0800), Section 6.2.4, paragraph II (d).

There are no credible failure mechanisms, introduced by the engineering change, which change the outcome of previously performed accident analyses or present new mechanisms which need to be evaluated.

<u>M1C19 Reload Safety Evaluation</u>

The M1C19 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Nuclear Engineering Division workplace procedure NE-102, "Workplace Procedure for Nuclear Fuel Management", and the M1C19 Reload Safety Evaluation, confirm the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses remain bounding with respect to the M1C19 safety analysis reactor physics parameters. The safety analysis reactor physics parameters method is described in topical report DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology".

The M1C19 core reload is similar to past cycle core designs, with a design generated using NRC approved methods. The M1C19 Core Operating Limits Report is prepared in accordance with Technical Specification 5.6.5 and submitted to the NRC in accordance with 10 CFR 50.4. No Technical Specification or UFSAR changes are necessary to support operation of M1C19.

MD200007
MD200008
<u>MD100006, Rev.0</u>

The Emergency Diesel Generator (EDG) protective relaying (used for testing) is being upgraded using a Commercial-off-the-shelf (COTS) digital multifunction relay. The selected device is an ABB REM 544N digital relay. The relay and supporting software have been qualified for use in nuclear safety systems by third-party qualifiers following industry guidelines published by EPRI that have been endorsed by the NRC.

The proposed modification will replace three analog protective relays (50DGT, 59DGN, 32DGT) with one ABB REM544N digital multifunction relay that will incorporate these functions. This modification will also add loss of excitation/field protection (40DGT), negative-sequence protection (46Q), definite time overcurrent protection (51DGT), and over excitation protection (24DGT). The EDG protective relaying that is being modified is only used when the diesel is in test mode and connected to the bus in parallel operation with the normal supply power.

No changes to the design basis accident mitigation functions of the EDG system are being made in connection with this modification. The modification changes the manner in which the EDG protective relaying is implemented by going from separate analog relays to a single digital relay unit that encompasses multiple relay functions.

The modification introduces some potential new failure modes that do not exist in the legacy analog protective relays being replaced. These failures have been addressed by analysis and by testing (done by third-party qualifiers), and demonstrated to have minimal effects on the safety related functions. The reliability of the replacement relay has been studied using the FMEA method. It has been concluded that the new relay gives a substantial improvement in reliability, even when the new potential failure modes are taken into account.

The results of the review demonstrate that the proposed modification does not affect the current licensing basis adversely, and does not create more than a minimal increase in the likelihood or consequences of a failure of the EDG protective relaying.

MD101182
<u>MD201245</u>

The associated design basis functions of the Containment Spray (NS) system are to (1) remove heat to maintain containment pressure below the maximum design limit of 15 psig in response to a high energy line break, (2) prevent depressurization of containment below the minimum design pressure of negative 1.5 psig, (3) remove fission product iodine from the containment atmosphere following a loss of coolant accident, and (4) provide containment isolation of the NS system piping when containment spray is not needed.

Valves 1/2NS-12B, 1/2NS-15B, 1/2NS-29A and 1/2NS-32A are motor operated valves (MOVs) located outside containment that are normally closed as a containment isolation function. These valves isolate each of the four containment spray ring headers that are supplied by the NS pumps. These valves are interlocked with the NS pumps such that at least one of the two valves for each pump must provide a full open signal in order to allow the respective NS pump to start. Upon closure of these valves such that there is no longer a full open signal for a t least one of the two valves, the respective NS pump is tripped.

Operation of the NS pumps is actuated by an Engineered Safety Feature signal (Sp) initiated either manually from the Control Room, or on coincidence or two of four high-high containment pressure signals. The high-high containment pressure set point is 3 psig. This signal will actuate the NS pumps by opening valves 1/2NS-12B, 1/2NS-15B, 1/2NS-29A and 1/2NS-32A, which will permit the NS pumps to start.

The Containment Pressure Control System (CPCS) maintains containment pressure above the minimum design pressure by preventing NS pump operation below a setpoint of 0.35 psig. When containment pressure decreases below 0.35 psig, the NS pumps are secured by the automatic closing of valves 1/2NS-12B, 1/2NS-15B, 1/2NS-29A, and 1/2NS-32A, which trips the NS pumps upon loss of a one of two full open signals form each pumps respective discharge valves. If an Sp signal is present, upon containment pressure increasing above 0.35 psig, valves 1/2NS-12B, 1/2NS-15B, 1/2NS-29A and 1/2NS-32A automatically open. The NS pumps can be manually started any time one of each pump's two discharge valves have a full open signal. Provided that an Sp signal is present, the NS pumps will automatically start upon containment pressure increasing to 0.80 psig.

Electrical power is provided to valves 1/2NS-12B, 1/2NS-15B, 1/2NS-29A and 1/2NS32A through the Essential Auxiliary Power System by sequencer load group 1, which occurs within 11 seconds of ESF actuation. Similarly, electrical power is provided to the NS pumps by sequencer load group 4, which occurs within 25 seconds of the ESF actuation. If an NS pump has not started before powering sequencer load group 5, which occurs within 30 seconds of the ESF actuation, power to this NS pump is blocked until sequencer load group 7, which occurs within 40 seconds of the ESF actuation.

After an NS pump has started, containment spray through each of the spray nozzles is delayed while the piping becomes water filled. Fully atomized spray flow is credited for both heat removal and fission product iodine removal. Typically, the NS system piping downstream of valves 1/2NS-12B, 1/2NS-15B, 1/2NS-29A, and 1/2NS-32A is only partially water filled. Typically this piping is filled to the static elevation head of the refueling water storage tank (FWST). Based on the volume of empty piping, it would require approximately 25 to 40 seconds after a NS pump starts to fully fill the NS spray rings to have fully atomized spray flow from all nozzles. However, a total of 120 seconds has been used to create margin. The 120 seconds is the assumed delay of achieving fully developed containment spray after receiving the containment high-high pressure signal. This input assumption has been incorporated into the radiological dose analyses and containment pressure response analyses.

M2C19 Reload Design

The M2C19 Reload Design Safety Analysis Review (REDSAR), performed in accordance with Nuclear Engineering Division workplace procedure (NE-102), "Workplace Procedure for Nuclear Fuel Management," and the M2C19 Reload Safety Evaluation, confirm the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis remain bounding with respect to the M2C19 safety analysis reactor physics parameters. The safety analysis reactor physics parameters method is described in topical report DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology."

The M2C19 core reload is similar to past cycle designs, with a design generated using NRC approved methods. The M2C19 Core Operating Limits Report is prepared in accordance with Technical Specification 5.6.5 and submitted tot he NRC in accordance with 10 CFR 50.4. No Technical Specification or UFSAR changes are necessary to support operation of M2C19.

EP/1/A/5000/ES-1.3 Rev. 20
EP/2/A/5000/ES-1.3 Rev. 20

These procedures were revised to provide guidance to ensure adequate sump level for small break loss of coolant accidents (SBLOCAs). The Emergency Core Cooling System (ECCS) pumps will be swapped to the sump when adequate level is ensured. Waiting for a lower refueling water storage tank (FSWT) level to complete the manual part of a swappover sequence does not pose more than a minimal increase in risk since this change will only occur for a SBLOCA that very slowly depletes the FWST and when sufficient time exists for operators to complete manual actions. Securing the Residual Heat Removal (ND) pumps during SBLOCAs to protect the pumps is already part of the mitigation strategy for SBLOCA when Reactor Coolant (NC) system pressure stabilized above ND shutoff head and it is acceptable to credit this action. Extending this action for slightly larger SBLOCAs (where NC pressure has not stabilized above ND shutoff head) for initial conditions in lower Mode 3 temperature is similar to strategies used in current emergency procedures for SBLOCA. No undue challenge to nuclear safety is posed by this change.

Manually operating Containment Spray (NS) and defeating NS auto start as part of long term recovery following a SBLOCA where NS operation is not required is an acceptable means to control spray. At his point, hours into the event, NS would have already performed its design function to maintain containment pressure. Any subsequent desire to run NS is to address slow moving containment parameters and the Technical Support Center would have time to evaluate plant conditions as part of long term recovery. The controlled operation of NS also supports operator actions to align RHR as part of long term recovery.