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10 CFR 50.59 (d)

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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
McGuire Nuclear Station, Units 1 and 2
Docket Numbers 50-369 and 50-370
Renewed Facility License Numbers NPF-9 and NPF-17
Summary Report of Evaluations Performed Pursuant to 10 CFR 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 (MNS) for changes occurring during the period from January 1, 2014, to December 31, 2014. These evaluations demonstrate that the associated changes do not meet the criteria for license amendments as defined by 10 CFR 50.59(c)(2).

This submittal document contains no regulatory commitments.

If there are any questions or if additional information is needed, please contact Brian Richards of Regulatory Affairs at (980) 875-5171.

Sincerely,

Steven D. Capps

Attachment

IE47
NRR

xc:

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Attachment 1

McGuire Nuclear Station (MNS)
Changes Evaluated Under 10 CFR50.59

Replace Startup Range and Intermediate Range Excore Detectors on Unit 1
Modification EC 97370
Action Request 00292297

The Westinghouse Excore Nuclear Instrumentation System (NIS) Source Range and Intermediate Range detectors (N31/N35 and N32/N36), including cabling and electronics, were replaced with Thermo Scientific (formerly Gamma-Metrics) fission chamber detectors and electronics. The cabling and other preliminary field work was accomplished previously under two (2) nuclear station modifications: NSM-12569/P1 and NSM-22569/P1. The detectors and rack-mounted electron equipment were changed under Engineering Change EC 97370 (MD201618) for MNS Unit 1. The new detectors are similar to the wide-range Gamma-Metrics detectors, channels N51/N52, installed previously in response to post-accident monitoring requirements imposed by NUREG-0737.

The changes required a revision to Technical Specifications Table 3.3-1 and supporting bases because the existing NIS Intermediate Range trip setpoints are specified in units of ion chamber amperes, whereas the replacement Thermo Scientific system is calibrated in units of percent Rated Thermal Power (RTP). These changes were submitted in a License Amendment Request (LAR) dated July 1, 2009, and approved by the NRC in August 2010. Therefore, the scope of the change made under 10CFR50.59 excludes the specified setpoint changes.

Replace Main Power Relays (Unit 1)
Modification EC 103951
Action Requests 00368781 and 00438510

The Generator Circuit Breakers (GCBs) must be capable of clearing the worst case fault conditions subjected to the Main Power System. However, as system conditions have changed over time, it has become more challenging to ensure that the asymmetrical interrupting fault duty of the GCBs is maintained. Major contributing factors include an increase in the short-circuit MVA from the offsite power source and an increase in fault current contribution from the main generator stator replacement. In order to ensure the GCBs are capable of clearing their worst case fault conditions, new protective relays were installed with the capability of delaying the opening time of the GCBs to ensure that they are not over-dutied when they are charged with clearing faults on the Main Power System.

The changes made maintain the existing protection schemes with equal or superior reliability and security. The manner in which the Main Power Protective Relaying can impact operability of the offsite power sources is unchanged. There was no impact on the Technical Specifications.

The Unit Main Power System and its associated protective relaying are non-safety systems but are designed to support accident mitigation functions. Because they are backed by the safety-related onsite emergency power system, the EPA systems are not required and are not qualified to withstand external events; however, either operation or mis-operation of components within these systems can directly or indirectly lead to a ANS Condition I load rejection, a loss of external load and a turbine trip, both of which are ANS Condition II events, Faults of Moderate Frequency, analyzed in UFSAR Sections 15.2.2 and 15.2.3, respectively. A failure or delay in the response in the primary protective relaying can also lead to partial or complete loss of non-emergency AC power to the station auxiliaries - an ANS Condition II event analyzed in UFSAR Section 15.2.6. The proposed change has been evaluated relative to each of the eight (8) evaluation questions in 10CFR50.59. Considering the probability and consequences of the various hardware and software failure modes of the proposed changes, the evaluation concluded that this change did not require prior NRC approval.

Measurement Uncertainty Uprate Flow Measurement Postponement
Modifications EC 105885 (U1) and 105886 (U2)
Action Requests 00429363, 00438255, 00444690, and 00443117

A Measurement Uncertainty Recapture (MUR) uprate in Rated Thermal Power (RTP) was requested of the NRC in an MNS License Amendment Request (LAR) dated March 5, 2012. The NRC issued the approving Safety Evaluation Report (SER) on May 16, 2013, permitting MNS Units 1 and 2 to increase RTP from 3411 MWt to 3469 MWt.

With the exception of online re-scaling of main steam (SM) and main feedwater (CF) flow loops, field implementation activities were evaluated and determined to be capable of being performed while the unit being uprated was online. Online re-scaling of SM and CF flow loop instruments was concluded to pose an unacceptable risk to occurrence of plant transient. However, without performing SM and CF flow instrument re-scaling, the Control Room readouts for these flow parameters would exhibit an approximate 1.7% offset at the MUR uprated level (i.e., at 100% RTP (3469 MWt), SM and CF flows would read 101.7%).

Since this Control Room readout configuration would conflict with the Human Factors Engineering guidance published in NUREG-0700, to which MNS has committed in its responses to NUREG-0737 for Control Room design, tuning adjustments were implemented to the Ovation Distributed Control System (DCS) and Operator Aid Computer (OAC) to correct the Control Room readouts, such that at 100% RTP (3469 MWt), SM and CF flows read 100%. Unlike re-scaling of the SM and CF flow instruments, implementing the DCS and OAC tuning adjustments could be performed with the affected MNS unit online with acceptable risk to occurrence of a plant transient. One aspect of utilizing DCS and OAC tuning adjustments in this manner is that the overall indication range of the Control Room readouts for SM and CF flows becomes limited to “0 to 117%” instead of the “0 to 120%” overall indication range for these instruments cited in UFSAR Tables associated with post-accident monitoring. These UFSAR tables were appropriately updated to reflect these changes to SM and CF flow instrument overall indicated range, which remains in compliance with Regulatory Guide 1.97, Rev. 2, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.

OBDN Resolution SNSWP Temperature Limit Increase
Modification EC 108853
Action Requests 00439117 and 00449630

A corrective action program report was written to document an Operable but Degraded/Non-Conforming (OBDN) condition in which previous guidance in Station procedures did not correspond to the assumed plant configuration for the Standby Nuclear Service Water Pond (SNSWP) thermal analysis. Operating procedures, which were consistent with the adoption of the standard Westinghouse emergency response guidelines, did not direct the operators to secure trains at any time during the accident event. As a result, the additional heat loads and flow rates challenge the ability of the SNSWP to supply Nuclear Service Water (RN) at a temperature that satisfies the 95°F design basis limit for the plant heat exchangers.

The limiting SNSWP thermal analysis load case in MCC-1150.01-00-0001 is a one-Unit Loss of Coolant Accident (LOCA), one-Unit Cooldown, concurrent with a Design Basis Seismic event that causes failure of Cowans Ford Dam, and a Loss of Offsite Power (LOOP) on both Units. The original analysis assumed that the plant would react to this event in a maximum safeguards condition for the first four hours, and a minimum safeguards condition thereafter once proper operation of the operating trains had been verified. This scenario was also reflected in the original McGuire Final Safety Analysis Report (FSAR). The new thermal analysis MCC-1150.01-00-0008 assumes maximum safeguards flow for the entire 30 day period. The terms "minimum safeguards" and "maximum safeguards" refer to the number of trains operating in response to an Engineered Safeguards signal. For the purpose of this evaluation, the minimum safeguards condition corresponds to one train of RN for each unit whereas the maximum safeguards condition assumes both RN trains are operating on each unit. Credit is being taken for monitoring the condition of the SNSWP and the performance of RN loads during the course of an event and taking actions as needed to ensure adequate RN flow is available to meet the cooling demands of the affected components. Changes to AP/1(2)/A/5500/020 have been implemented.

To resolve the OBDN condition, engineering change EC 108853 increased the RN design temperature from 95°F to 102°F for the SNSWP intake up to the plant heat exchangers' inlet. The temperature limit for the normal low level intake from Lake Norman or the RN non-essential header remained unaffected.

Thermal Analysis:
A revised thermal analysis calculation of the SNSWP has been performed and is documented in MCC-1150.01-00-0008. This revised analysis of the SNSWP ensures adequate bounding conditions for thermal loading of the SNSWP, maximum safeguards flow for the duration of the accident, bounding initial temperature conditions in the SNSWP, and bounding meteorological conditions concurrent with the operating condition of the two units. The revised design plant heat input for the SNSWP is documented in calculation MCC-1223.24-00-0130. These two new calculations supersede the original calculations for the SNSWP thermal analysis.

Civil Scope:
The civil stress of all affected RN piping and piping supports for the increased RN design temperature have been performed to demonstrate acceptability.

Mechanical Scope:
The mechanical analyses documented in MCC-1223.24-00-0126 have been performed to demonstrate the operational capability of all affected plant components to perform the required functions at the increased design temperature of 102°F (i.e., RN valves, RN pumps, RN pumps' NPSH, RN strainers, plant heat exchangers' capabilities, and RN instrumentation). The affected mechanical design documents (RN flow diagrams, RN piping drawings, RN pipe specification line listings) have been revised to reflect the increased design temperature.
The "design temperature" attribute in the Equipment Database (EDB) for all affected equipment records was changed from 95°F to 102°F under EC 108853.

The affected UFSAR sections were updated as appropriate per EC 108853.

A review of Technical Specification Bases was performed. The bases for TS 3.7.7 were revised to reflect the maximum NSW inlet temperature to 102°F.

A review of Selected Licensee Commitments was performed. No required changes were identified.

This 50.59 evaluation bounded the changes to design documents, the UFSAR, and the Technical Specification Bases document.

The evaluation concluded that this change did not require prior NRC approval. The design functions of the RN system are maintained with respect to accident response and long term core and component cooling when Operator actions taken that are consistent with NEI 96-07 guidance.

Fuel Manipulator Crane Upgrades
Modifications EC 77048 and 77051
Action Requests 00441282 and 00448032

Note: Evaluations associated with the fuel manipulator crane upgrades were previously reported on February 6, 2014. These previously reported evaluations were revised as a result of unresolved inspection item URI 2013008-01, which has since been closed. The revised evaluations are included in this current report.

This modification resolved reliability and obsolescence problems confronting the fuel manipulator cranes in the reactor containment and spent fuel building. The power and control systems were upgraded, including the position sensors, motor drives, control consoles, and wiring. The existing analog controls were replaced using a digital programmable logic controller (PLC) with a graphic user interface (GUI). The PLC can be programmed in advance with the refueling sequence and the step-wise destinations of each fuel assembly.

The PLC controls allow multi-axial travel (in the x-y dimensions) within established safe operation zones. The maximum crane bridge and trolley speeds were increased, but the existing acceleration limits were retained. The hoist slow-speed zones were reduced, but an adequate distance was retained for the safe insertion of a fuel assembly into a storage or core location.

The fuel handling system operation, as described in the UFSAR, is essentially unchanged. The mechanical travel stops were retained to provide diverse bridge, trolley, and hoist operating limits. The design functions, interlocks and other safety features of the fuel handling system were retained and were not adversely impacted by the modification. The modification incorporates improvements in the man-machine interface and increased automation by the manipulator crane. Therefore, the changes result in a reduction in human errors and consequently an improvement in safety.

The failure modes and effects analysis (FMEA) and software coping analysis performed for the replacement controls and variable speed drives concluded that the use of digital controls does not increase the probability or the consequences of any fuel handling accident currently described in the UFSAR, nor does it introduce the possibility of a new type of accident not previously considered. No more than one fuel assembly is moved at a time, in either building. Therefore, the assumptions and conclusions of the existing fuel handling accident analyses remain valid.

No reductions in the existing margins of safety were created by this modification. The existing shutdown margins during refueling operations were maintained. The existing submergence limits and radiation shielding margins were retained. The existing travel stops for the bridge, trolley, and hoist were retained. No changes to the Technical Specifications or their Bases were required. No changes were required in the Selected Licensing Commitments (SLC) Manual. The affected UFSAR sections were updated as appropriate.

B Train RN to CA suction relocation (Unit 1)
Modification EC 101080
Action Request 00441688

EC 101080 relocated the Unit 1 “B” Train Auxiliary Feedwater (CA) pump assured supply from the return side of the Emergency Diesel Heat Exchanger (KD HX) to the supply side of the KD HX. The new location allows the CA pumps to be supplied with cooler Nuclear Service Water (RN) prior to heating up through the KD and other heat exchangers, increases the pressure to the CA pump’s supply, and eliminates the concern for dissolved gases coming out of solution and potentially sweeping into the CA pumps.

With the CA pump supply relocated, two concerns exist: one with the flow balance of the Nuclear Service Water (RN) system due to an increase in the RN system demand and another with an increased differential pressure across the RN and CA system isolation valves. A test line was added to allow a flow balance of the system to be performed since an additional flow demand is added to the RN system.

The increased differential pressure requires additional opening/closing thrust for the existing valves because the additional differential pressure creates a loss of margin on the existing valves' ability to perform their safety function. The additional differential pressure requires replacement of the valves and actuators for 1RN-162B, 1CA-18B and 1CA-116B. The new valves and larger actuators have been assessed, along with all other associated components in the circuits and the existing cabling requires no changes.

A continuous bleed line was installed to bleed pressure between valve 1RN-162B and 1CA-18B/1CA-116B into the lower pressure "B" train KD HX return piping to ensure that high pressure RN supply water does not leak into lower pressure CA suction piping and contaminate the system during normal plant operation. A pressure gauge was installed to allow for monitoring of the bleed line effectiveness. Connections with valves was provided at each end of the bleed line to allow for isolation and chemical cleaning of the line as needed, and high point venting as necessary.

An interlock exists between 1CA-18B and 1RN-171B that will open 1RN-171B when 1CA-18B opens on low CA header pressure. 1RN-171B is the isolation for the “B” Train KD heat exchanger cooling water. When 1RN-171B opened in the past, it supplemented the supply to the CA system when the KD heat exchanger was not in use. The relocation of the RN assured supply to CA makes the interlock unnecessary and the interlock will be deleted. This interlock deletion will create a positive affect by preventing a flow diversion to the KD heat exchanger if the heat exchangers are not needed.

To limit the demand on RN when the Containment Spray (NS) system is placed in service, CA flow coming from the same train of RN will be limited to 900 gpm when NS is started. If CA is operating at a maximum flow in a large break loss of coolant accident (LOCA), flow to the KD heat exchanger could be diverted to just below the RN flow balance acceptance criteria if NS is running. Analysis indicates that in a LOCA steam generator level will be restored and the CA flow run back to near minimum flow conditions prior to starting NS but the limitation provides an additional backup to predicted conditions. NS will only be started for a large break LOCA.

During the hydraulic analyses performed in support of EC 101080, latent CA system performance issues were discovered that are not related to the physical modification to the plant associated with EC 101080. A corrective action program report documents latent issues related to pump run-out and protective relay settings. As a mitigating action, setpoints have been developed for inclusion into the emergency procedures to manage the concerns for hydraulic limitations and maintaining margin to overcurrent trip settings. When operating a motor driven CA pump feeding steam generators, the flow to those steam generators will be limited to 600 gpm when steam generator pressure is less than 600 psig. This limitation prevents approaching a run-out condition on the pump and maintains steady state running currents below the overcurrent trip setpoints while ensuring the minimum required CA flow for decay removal (450 gpm) is met. Given the existing curves, the motor driven pump steady state conditions can be accurately predicted. EC 101080 updated design deliverables with the setpoints as a matter of convenience, hence the inclusion in this 50.59 evaluation.

The new location of the assured supply is adverse to the flow balance of the RN system as described in the UFSAR. For this reason the 50.59 review was elevated to an evaluation versus a screen. Previous modifications to the "A" trains also had 50.59 evaluations performed for the same reason. Previous "A" train modifications went straight to 50.59 evaluations without screens and this same approach was taken for EC 101080.

The evaluation concluded that this change did not require prior NRC approval. The alteration to RN flow paths does not infringe on the minimum flow requirements of the RN system for all accident conditions. The installation of pipes, supports, valves, and valve operators in accordance with the design basis codes and standards did not introduce new failure mechanisms or adversely affect the ability of the RN and CA systems to meet their design functions.

The affected UFSAR sections were updated as appropriate.

No changes were required to the Technical Specifications.

Assured CA Suction for Extended SBO (Order EA 12-049) Unit 1
Modification EC 109071
Action Request 00442503

Engineering Change (EC) 109071 implemented the following activities:

NRC Order EA-12-049, "Order Modifying Licenses with regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events", requires licensees to implement a set of strategies to mitigate severe external events. These strategies are required to maintain or restore core cooling, containment and spent fuel pool (SFP) cooling capabilities following a beyond-design-basis external event. The McGuire Auxiliary Feedwater (CA) System cannot support the initial mitigation phase of core cooling using installed equipment required by the Order as currently configured. Severe external events (tornado, etc.) can render the normal, non-safety-related Auxiliary Feedwater Storage Tank (CAST) suction source to the CA pumps unavailable. Postulated simultaneous loss of all AC power as required by the Order renders the safety-related assured suction sources (A and B trains of the Nuclear Service Water (RN) System) unavailable. The remaining Standby Shutdown System (SSS) related suction source (captured volume in embedded Condenser Circulating Water (RC) piping) is aligned manually, and could not be aligned in time to prevent loss of the turbine-driven CA pump (sole operating pump under loss of all AC conditions).

1CA-161C and 1CA-162C are normally closed, DC motor operated gate valves that separate the embedded RC piping from the suction side of the CA pumps. They are de-energized to prevent any spurious opening that could potentially introduce air into the CA suction lines. This scope will replace 1CA-162C with an air-operated, fail-as-is (air to open, air to close, double-acting) gate valve, to be tagged 1CA-162B. 1CA-161C will be maintained open and de-energized. Because the non-modified valve will be normally-open, 1CA-162C is the preferred valve to modify (1CA-161C as the normally-open valve would be less susceptible to raw water corrosion issues than 1CA-162C).

To ensure that opening of 1CA-162B does not prevent or delay opening of the safety-related RN suction sources to CA on low pump suction pressure, the safety-related pressure switch setpoints must be raised from 3.5 to 7.0 psig and TS 3.3.2 revised accordingly. This setpoint change was not addressed further in this 50.59 evaluation as a License Amendment Request (LAR) had already been submitted for the setpoint changes.

The time delay relays associated with the safety-related RN suction sources will be increased by one second from 2.5 to 3.5 seconds to guard against inadvertent alignment of the RN source(s) during suction pressure transients at the time of pump start, and the filtered water storage tank (CACST) will be maintained normally aligned (CA-6 open) to reduce the magnitude of suction pressure transients upon pump start. This changed the response time described in the UFSAR.

Since the CACST will become a normally aligned suction source, a vent will be installed between 1CA-5 and 1CA-6 to ensure that this currently unventable portion of the CACST piping can be maintained water solid (Unit 1 issue only). Check valves will be installed on the CACST makeup lines from the hotwell to prevent CACST inventory from draining back to the hotwell on a loss of hotwell pumps. A vent valve will be installed upstream of each check valve in order to facilitate check valve testing.

The new time delay relays associated with 1CA-162B will be set 1 second longer (4.5 seconds) than the safety-related RN suction time delay relays, again to ensure that opening of 1CA-162B valve does not prevent or delay opening of the safety-related RN suction sources to CA on low pump suction pressure.

Local pressure gauges will be installed to monitor CACST and CAST tank levels, in support of required FLEX Support Guide (FSG) development, in order to allow for timely operator action to initiate tank makeup or align alternate suction sources to the TDCA pump.

This EC is dependent on completion of "B" Train RN-CA EC 101080, in order to allow the auto-open feature of CA-162 to be reinstated without re-introducing an air entrainment degraded condition (M-06-02284) and associated compensatory actions.

The affected UFSAR sections were updated as appropriate.

Revision to Selected Licensee Commitment (SLC) 16.9.7 bases, as described in the engineering change, has been performed.

Revision to Technical Specification 3.3.2 was performed via License Amendment Request submitted on September 12, 2013, and approved by the NRC by safety evaluation report dated August 27, 2014.

Implementation of an automatic swap over feature in the event of a loss of the CAST and RN reduces operator burden and will maintain stable TDCA pump operation when required. Changes to setpoints and time delays will prevent an inadvertent actuation of Extended Loss of AC Power related functions during design basis event response. The increase in response time for alignment to RN still ensures an adequate water supply to the CA system for all of the CA system design basis events. The new method of controlling valve position and preventing adverse system interactions has not increased the frequency or consequence of evaluated accidents or malfunctions. No new accidents or malfunctions will be introduced by this activity. The evaluation concluded that this change did not require prior NRC approval.

Update UFSAR Due to Reanalysis of Unfiltered Control Room Inleakage
Action Request 00442992

The activity under evaluation pursuant to 10 CFR 50.59 consists of two changes to the Alternative Source Terms (AST) analysis of the McGuire Nuclear Station (MNS) tornado missile accident (TMA) and the associated reports of the change in updates to UFSAR 15.10.3 and Tables 15-12, 15-35, 15-39, and 15-64. The changes to the AST analysis of the MNS TMA are as follows:

1. The radial peaking factor used to determine the fuel assembly fission product activity and source term for the TMA was set to 1.65 in place of taking burnup dependent factors with a maximum value of 1.575.
2. The unfiltered control room inleakage rate with a Control Room Area Ventilation System (CRAVS) Outside Air Pressurized Filter Train (OAPFT) in operation was set to 210 cfm in place of 150 crm.

In a related activity, the revision to the AST analysis of the TMA included setting the unfiltered control room inleakage rate with the CRAVS OAPFTs off to 500 cfm in place of 625 cfm. This change was found to screen from the requirements of an evaluation pursuant to 10 CFR 50.59.

The TMA source term is not an accident initiator, and the change to the fuel pin radial peaking factor does not make it one. Likewise, the control room envelope is not an accident initiator; the change to the unfiltered control room inleakage rate does not make it one. The effect produced by these changes on the calculated values of post-TMA radiation doses does not equate to a more than a minimal increase in the consequences of either an accident or malfunction of a system, structure, or component evaluated in the UFSAR. This activity impacts no fission product barrier design limit. The changes are made to input parameters, and no element of the method of AST analysis used for MNS is revised.

The evaluation concluded that this change did not require prior NRC approval.

Update OP/1/A/6100/SU-19 to Partially Open SG PORV in Mode 3
Action Request 00448608

OP/1/A/6100/SU-19 was revised to add steps in Section 3.31 of Enclosure 4.2. This one time restricted change was made to direct manually opening a Steam Generator Power Operated Relief Valve (SG PORV) approximately 5% while in Mode 3 at full temperature and pressure on Unit 1. The activity was performed with the MSIVs aligned open and the condenser dump valves in-service. This action was added to evaluate the noise level of PORV operation in the community surrounding McGuire.

The valve is intended to support steam release under the operating conditions. This test does not challenge the performance of the components outside of their design functions. The test is conservatively conducted in Mode 3. SG PORVs are addressed in Technical Specification (TS) 3.7.4, which requires that three SG PORV lines be operable in Modes 1,2 and 3, and in Mode 4 when the steam generator is relied upon for heat removal. Failure of a SG PORV is evaluated in the UFSAR, and thus the proposed activity is fully bounded by the UFSAR Safety Evaluation. No new or amended technical specifications are required. There are no new accidents or malfunctions made credible by this test, and no fission product barrier limits or evaluation methodologies are impacted.