Tom Ray
Vice President
McGuire Nuclear Station

Duke Energy
MG01VP | 12700 Hagers Ferry Rd.
Huntersville, NC 28078

704.946.2305
Tom.Ray@duke-energy.com

RA-19-0424
October 8, 2020

10 CFR 50.4
10 CFR 50.71(e)
10 CFR 50.59

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

McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370,
Renewed License Nos. NPF-9 and NPF-17

Subject: Submittal of Updated Final Safety Analysis Report (UFSAR) Revision 22, Technical Specification Bases Revisions, UFSAR/Selected Licensee Commitment Changes, and 10 CFR 50.59 Evaluation Summary Report

Ladies and Gentlemen:

Pursuant to 10 CFR 50.71(e), Duke Energy Carolinas, LLC (Duke Energy) hereby submits Revision 22 to the UFSAR for the McGuire Nuclear Station (MNS), Units 1 and 2. In accordance with 10 CFR 50.71(e)(4), this UFSAR revision is being submitted within six months following the most recent refueling outage, which concluded on April 13, 2020. The MNS UFSAR is included in this submission via two CD-ROMs (Enclosures 1 and 2). Enclosure 1 provides a copy of the UFSAR that has been redacted for public use. Enclosure 2 provides a copy of the UFSAR that contains sensitive information to be withheld from public disclosure per 10 CFR 2.390(d)(1). Changes made since Revision 21 are identified by vertical lines in the margins of the pages that are indicated as Revision 22.

By Safety Evaluation Report dated 03/08/2019 (Reference 1), the NRC authorized for MNS a revision to the Reactor Vessel Internals Aging Management Program based on MRP-227-A. Revision 22 of the MNS UFSAR incorporates the revision of the Reactor Vessel Internals Aging Management Program.

In accordance with 10 CFR 50.59(d)(2), Duke Energy is providing a report summarizing the 10 CFR 50.59 evaluations of changes, tests, and experiments implemented during the period from November 1, 2018 to September 17, 2020 for MNS. This report is included in Enclosure 3.

Pursuant to 10 CFR 50.4, Duke Energy is providing the MNS Technical Specification Bases changes that were made according to the provisions of Technical Specification 5.5.14, “Technical Specifications (TS) Bases Control Program.” Enclosure 4 contains the MNS TS Bases in a CD-ROM, last revised on 5/14/20.

Additionally, in accordance with 10 CFR 50.71(e), Duke Energy is providing MNS Selected Licensee Commitments (SLC) Manual, last revised on 8/27/20 . The MNS SLC manual is provided in Enclosure 5 as a CD-ROM. The MNS SLC manual constitutes Chapter 16 of the UFSAR.

In addition, in accordance with Duke Energy's commitent management program (AD-LS-ALL-0010, Commitment Management), notification of a regulatory commitment change is provided in Enclosure 6.

There are no new regulatory commitments contained in this letter.

If you have any questions regarding this submittal, please contact Art Zaremba, Manager – Regulatory Affairs, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 9, 2020.

Sincerely,

Thomas D. Ray, P.E.
Site Vice President
McGuire Nuclear Station

References:

1. McGuire Nuclear Station, Units 1 and 2 – Staff Evaluation Related to Reactor Vessel Internals Inspection Plan Based on MRP-227-A, ADAMS Accession Number ML19058A416, March 08, 2019.

Enclosure:

1. McGuire Nuclear Station Updated Final Safety Analysis Report 2020 Update - Rev 22 Redacted Version, CD (Public Use Only)
2. McGuire Nuclear Station Updated Final Safety Analysis Report 2020 Update - Rev 22 CD (Non-Public Use)
3. McGuire Nuclear Station 10 CFR 50.59 Evaluation Summary Report
4. McGuire Nuclear Station Technical Specification (TS) Bases CD
5. McGuire Nuclear Station Selected Licensee Commitments (SLC) Manual CD
6. McGuire Nuclear Station Notification of Regulatory Commitment Change

xc:

L. Dudes, USNRC Region II – Regional Administrator
E. Miller, USNRC NRR Project Manager - MNS
A. Hutto, USNRC Senior Resident Inspector - MNS

Enclosure 1

McGuire Nuclear Station
Updated Final Safety Analysis Report 2020 Update - Rev 22 Redacted Version, CD
(Public Use Only)

Enclosure 2

McGuire Nuclear Station
Updated Final Safety Analysis Report 2020 Update - Rev 22 CD
(Non-Public Use)

Enclosure 3

McGuire Nuclear Station
10 CFR 50.59 Evaluation Summary Report

Title:
MNS Thermal Hydraulic SGTR Analysis – UFSAR Revision

Documentation Number(s):
Action Request (AR) 02233443

Brief Description:
A set of updates to McGuire Nuclear Station (MNS) UFSAR Section 15.6.3.2, Table 15-23, and Figures 15-226 - 15-232, and to Table 15-12, associated with UFSAR 15.6.3.3, has been prepared. The updates are based on changes made in two revisions to the transient thermal-hydraulics analysis of the MNS steam generator (SG) tube rupture (SGTR) completed with the computer code RETRAN. They also are based on two revisions to the Alternative Source Terms (AST) dose analysis of the MNS SGTR. The activity consists of the following changes in the revisions to the transient and AST analysis:

- Decreasing the assumed time after unit trip to stop Auxiliary Feedwater System (AFWS) flow to the ruptured SG (transient analysis)
- Increasing the assumed time after unit trip to start cooling the Reactor Coolant System (RCS) with the intact SG Power Operated Relief Valves (PORVs - transient analysis)
- Taking a smaller uncertainty in setting the initial pressurizer pressure (transient analysis).
- Taking the most recent table of target core exit temperature versus ruptured SG pressure in simulating the initial cooling of the RCS (transient analysis)
- Simulation of scrubbing of iodine from flashed SGTR break flow when the ruptured SG tubes are submerged (transient analysis)
- Accounting for releases of fission products from the intact SGs to 200 oF (AST analysis)
- Changes to the margins added to the calculated values of the post-SGTR radiation doses (AST analysis).

These changes are integrated and interconnected and as needed were evaluated as a single activity. No actual plant modification or procedure revision accompany this activity. No changes in the operation of the RCS, AFWS, or SG PORVs is imputed. Each radiation dose calculated for the MNS SGTR all equate to “not more than a minimal increase” in that dose. There were no changes to the method of either the transient analysis or put into place and the UFSAR updated without prior NRC approval.

Title:
MNS Turbine Trip and Feedwater System Pipe Break

Documentation Number(s):
Action Request (AR) 02236974

Brief Description:
The proposed activities are changes in the turbine trip event (UFSAR 15.2.3) and the feedwater system pipe break event (UFSAR 15.2.8) which result in revisions to Methodology Report DPC-NE-3002-A (UFSAR Chapter 15 System Transient Analysis Methodology).

Turbine Trip Event
High initial pressurizer pressure (i.e., positive uncertainty) and high initial reactor coolant system temperature (i.e., positive uncertainty) were being used in the turbine trip analysis. However, more limiting peak primary pressure results occur if a low initial pressurizer pressure (i.e., negative uncertainty) and low initial RCS temperature (i.e., negative uncertainty) are used. Thus, this screened in because it was adverse. It also screened in because it was a methodology change. Using high initial pressurizer pressure and high initial RCS temperature resulted in a peak primary pressure of 2,649.3 psig. Using low initial pressurizer pressure and low initial RCS temperature along with a new high pressurizer pressure trip resulted in a peak primary pressure of 2,651.6 psig. Therefore, these changes are conservative with respect to the limit of 2,735.0 psig and this was not a departure from a method of evaluation. It also did not exceed or alter the limit. It had no impact on initiating an accident or malfunction and there are no consequences associated with this change. It also did not create a new accident or a malfunction of an SSC with a different result.

Feedwater System Pipe Break Event
At one time, check valves existed on the feedwater system piping outside of containment. In the event of a feedwater system pipe break, these check valves would isolate the steam generator from the break and not result in a steam generator blowdown outside of containment. However, with the installation of the replacement steam generators at MNS and CNS Unit 1 in the 1990's, the feedwater line check valves in containment were removed. Without the check valves in containment, a break outside containment (in the doghouse) could result in complete blowdown of the steam generator in the faulted loop. Thus, the feedwater system pipe break event (UFSAR 15.2.8) was re-evaluated for a break outside of containment (in the doghouse) to see if the removal of the check valves was adverse to the feedwater system pipe break event. For a break inside containment, all trips were initiated by high containment pressure safety injection. For a break outside containment (inside the doghouse), trips are initiated by trips other than high containment pressure safety injection. Thus, this was a methodology change as well.
A break outside containment (in the doghouse) delays reactor trip, auxiliary feedwater, safety injection, and main steam line isolation which are all adverse and conservative. The acceptance criteria of concern for the feedwater system pipe break event is that there is long term core cooling which is assured by applying the more stringent requirement of no hot leg boiling. For the current analysis (the break inside containment) for MNS and CNS Unit 1, the margin to hot leg boiling is 35° F. For the break outside containment (in the doghouse), the margin to hot leg boiling is 3° F for MNS and CNS Unit 1. Thus, when the analysis is performed with the break outside containment (in the doghouse), it clearly has conservative results and is acceptable because there is no hot leg boiling. Thus, for the feedwater system pipe break changes, the methodology change was conservative and limits were still acceptable. Therefore, there was no departure from a method of evaluation and design basis limits are not exceeded or altered.

The changes to the feedwater system pipe break event could in no way initiate an accident or malfunction. Therefore, there is also no increase in the frequency of an accident or the occurrence of a malfunction of an SSC. Since there is no hot leg boiling, there is also no increase in the consequences of an accident or malfunction associated with the change to the feedwater system pipe break event. Since there is no new failure mode, the event still constitutes a feedwater system pipe break. Therefore, it does not create a possibility for an accident of a different type or a malfunction of an SSC with a different result.

In conclusion, none of the 50.59 criteria requiring prior NRC approval are met for either event, therefore no submission is necessary.

Title:
MNS Turbine Trip Event UFSAR Revision

Documentation Number(s):
Action Request (AR) 02249233

Brief Description:

The proposed activities are revisions to the MNS UFSAR for changes to the turbine trip event (UFSAR 15.2.3). Using an initial pressurizer pressure of 2,205 psig with a high pressurizer pressure reactor trip of 2,415 psig resulted in a peak primary pressure of 2,651.6 psig. Using a lower initial pressurizer pressure of 2,183 psig with a high pressurizer pressure trip of 2,425 psig resulted in a peak primary pressure of 2,652.7 psig. These results were adverse with respect to the limit of 2,735 psig. The MNS UFSAR needed to be revised for the associated changes.

The changes to the turbine trip event were purely analytical and involved no plant changes. The design function of the reactor coolant system (RCS) is to transport and contain the reactor coolant which removes heat from the fuel. The peak RCS pressure increased. Since the change adversely affects an evaluation demonstrating that intended design functions will be accomplished, these changes required a 50.59 evaluation. This change was acceptable because it did not exceed a limit or alter a limit. Thus, there were no dose consequences associated with this change. This change also did not increase the frequency of an accident or increase the likelihood of a malfunction. This change also does not create the possibility for an accident of a different type or a malfunction with a different result. Since this change involved changes to input assumption, it did not involve a change to a methodology or a departure from a method of evaluation. Thus, in conclusion, none of the 50.59 criteria requiring prior NRC approval are met for either event. Therefore, no submission is necessary.

Enclosure 4

McGuire Nuclear Station
Technical Specification (TS) Bases – CD

Enclosure 5

McGuire Nuclear Station
Selected Licensee Commitments (SLC) Manual - CD

Enclosure 6

McGuire Nuclear Station
Notification of Regulatory Commitment Change

Commitment Tracking Number: 2019-M-001

Existing Commitment:
Commitment by H. B. Barron at Enforcement Conference as part of an ongoing review; Engineering and Maintenance will check for floor movement due to heaving in the concrete in the lower ice condenser floor (reference PIP-M97-02686 and LER#370/97-03). The issue at that time was Lower Inlet Door (LID) test failure due to binding/interference with floor flashing that caused the opening force required to exceed the maximum allowable per the tech specification. The root cause of the problem was attributed to floor heaving.

A letter by H.B. Barron to NRC dated 11/06/1997 in reference to violation 50-369, 370/97-16 made a regulatory commitment to monitor the ice condenser floor movement. The following corrective actions were taken:

1. A process was established to measure and trend ice condenser floor wear slab elevations during refueling outages to assure no binding of the ice condenser lower inlet doors during the subsequent operating cycles. Surveyors have selected 6 locations in each bay and have been reading the elevation at each of those points since End of Core (EOC) 12 for McGuire Units 1 and 2 to monitor and trend for change in elevation.
2. A program was established to require inspection of the gap between the ice condenser floor wear slab and the lower inlet door survey mark and the gap between ice condenser floor and bottom edge of turning vanes for each unit at each bay prior to return to service from any cold shutdown during power operations. Reference procedure PT/0/A/4200/045.

Revised Commitment:

Change the frequency for action 1 of the commitment to every other refueling outage. Keep the frequency for action 2 of the commitment as is.

Revised commitment:
Ice condenser floor wear slab elevations will be measured on each unit during every other refueling outage and trended to assure no binding of the ice condenser lower inlet doors occurs during subsequent operating cycles. (AR 2258458)

Basis for Revision:

Since the commitment was made, several modifications were implemented to keep the LIDs free from any possible obstructions due to floor heaving and to provide additional means to monitor the floor temperature. To ensure enough clearance is maintained between the floor and the bottom of the LIDs, all existing flashing (that caused the interference when the floor heaved) was removed completely by modifications that were implemented during 2EOC13 (MGMM11942 Removal of I/C Lower Inlet Door Flashing U2) and during 1 EOC12 (MGMM10263A Removal of 1/C Lower Inlet Door Flashing U1).

The following modifications were also implemented to help the plant operators predict cyclical thermal fluctuations to limit unanticipated and undesirable floor movement:

- MGMM13708 (unit 1) was implemented during 1EOC16

- MGMM13709 (unit 2) was implemented during 2EOC16

  Those Modifications have mounted eight RTDs in each unit at the most heat load sensitive floor locations and alarms the control room upon sensing undesirable temperature that may lead to thermal fluctuations.

Engineering reviews the data collected by surveyors on elevation readings taken across the floor of the ice bed and compares the surveillance results with baseline readings that were taken during 1EOC12 and 2EOC11 refueling outages to assess for negative trending in floor movements. Collected data/charts indicate that the clearance from the top of the concrete floor to the bottom of the LIDs has unnoticeable changes and thus the original concern of a blockage to a LID movement due to floor heaving is not probable.

The measured parameters (glycol header pressure and temperature, relative humidity and temperature inside and outside I/C) that are recorded by the weekly surveillances are monitored by engineering and maintenance for any changes over extended period of time to help them recognize variation in the ice bed performance. Operations does rely on temperature readings at the chart recorder in the control room to determine performance of the ice bed. Semi-Daily surveillance items (PT/1/A/4600/003 A and PT/2/A/4600/003 A) requires Operations shift personnel to verify the ice bed temperatures every 12 hours is 27° F. Those activities are pre-indicator to factors that may lead to floor heaving, thus giving us sufficient time to correct.

Extending the surveillance frequency of floor wear slab elevation measurements to every other refueling outage to the floor elevation readings will not affect the overall performance of the ice condenser.