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U.S. Nuclear Regulatory Commission (NRC)
Washington, D.C. 20555-0001

ATTENTION: Document Control Desk

Duke Energy Carolinas, LLC
McGuire Nuclear Station, Unit 2
Docket No. 50-370
Renewed Facility Operating License NPF-17

SUBJECT: License Amendment Request to Revise Technical Specification 4.2.2, "Control Rod Assemblies"

Pursuant to 10 CFR 50.90, enclosed is a Duke Energy Carolinas, LLC (Duke Energy) License Amendment Request (LAR) for the McGuire Nuclear Station Renewed Facility Operating License and Technical Specifications (TS). The proposed one-time LAR is being submitted on an exigent basis to allow Unit 2 to remove control rod assembly H-08 during the upcoming Unit 2 outage (M2R25) and run for one fuel cycle. The assembly would only be removed if planned efforts to repair the associated thermal sleeve during the outage are unsuccessful.

The proposed LAR affects TS 4.2.2, "Control Rod Assemblies" for McGuire Nuclear Station Unit 2.

Duke Energy requests approval of this LAR before September 25, 2018, which would align with expected outage timelines. Once approved, this amendment would be implemented prior to entering Mode 5 on unit startup.

The Enclosure provides a description of the proposed change, the technical justification, an evaluation of significant hazards consideration pursuant to 10 CFR 50.92(c), a statement of environmental consideration, and the following attachments:

- Attachment 1 provides the existing TS pages marked to show the proposed changes for the McGuire Nuclear Station.
- Attachment 2 provides a clean version of the revised TS pages.

In accordance with Duke's administrative procedures and Quality Assurance Program, this LAR has been reviewed and approved by the McGuire On-Site Review Committee.

A001
NRR

Pursuant to 10 CFR 50.91, a copy of this LAR is being sent to the designated official of the State of North Carolina.

No regulatory commitments are associated with this LAR.

If there are any questions or if additional information is needed, please contact Jeff Thomas at (980) 875-4499.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 7, 2018.

Sincerely,

Thomas D. Ray

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

Enclosure

1. Evaluation of the Proposed Change

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ENCLOSURE 1

Evaluation of the Proposed Change

Subject: License Amendment Request (LAR) to Revise Technical Specification (TS) 4.4.2, "Control Rod Assemblies"

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1. McGuire Technical Specification Page Markups
2. McGuire Technical Specification – clean version

1. SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating License NPF-17 for McGuire Nuclear Station Unit 2 by adding a footnote to McGuire Technical Specification (TS) 4.2.2, "Control Rod Assemblies" (Reference 1). The footnote would allow the Unit 2 operating cycle M2C26 core to contain 52 control rods with no control rod in core location H-08. This would be in lieu of the current requirement of 53 control rods. This proposed change would allow operation for one fuel cycle and would only be used if planned efforts to repair the associated thermal sleeve during the upcoming Unit 2 outage (M2R25) are unsuccessful. A McGuire fuel cycle is nominally 18 months.

McGuire requests approval of the proposed amendment on an exigent basis pursuant to 10 CFR 50.91(a)(6) to allow Unit 2 to resume power operation following refueling outage M2R25. Approval of the proposed amendment is requested by September 25, 2018, to support Unit 2 entry into Mode 5 and to ascend to power operation.

2. DETAILED DESCRIPTION

The proposed amendment would revise TS 4.2.2 to add a footnote permitting operation with 52 control rods during M2C26 in lieu of the nominal requirement for 53 control rods. McGuire has reviewed its TS and has determined that no additional TS changes are required.

The proposed TS footnote is as follows:

> *Unit 2 is permitted to operate with 52 control rod assemblies (with no control rod assembly installed in core location H-08) during M2C26.*

The design changes and supporting safety analyses discussed in this document are performed in accordance with the current licensing basis. As such, NRC approval is only required for the proposed change to TS 4.2.2.

Attachment 1 provides a marked-up version of the affected pages of TS 4.2.2 for the McGuire Nuclear Station showing the proposed changes. Attachment 2 provides a clean version of the TS pages.

Note that for the purposes of this submittal, the terms "control rod" and "rod cluster control assemblies" (RCCAs) are used synonymously.

<u>Circumstances Establishing Need for the Proposed Exigent Amendment</u>

While reviewing video taken during the previous Unit 2 outage (M2R24), a bright ring-shaped marking was identified on the top of the H-08 control rod guide tube (CRGT). This marking indicates that the H-08 thermal sleeve had lowered and is now in contact with the CRGT. The thermal sleeve upper flange rests inside the control rod drive mechanism (CRDM) adapter tube. Flow around this sleeve causes the sleeve's upper flange to vibrate against the adapter tube, which causes wear to both components. Excessive wear causes the thermal sleeve to lower until the funnel at the bottom comes into contact with the CRGT, which results in a visible wear ring such as the one identified on the top of this CRGT.

No rings on other Unit 2 CRGT were identified during review of the video. Additionally, video from the most recent Unit 1 outage (M1R25) was reviewed, and no Unit 1 CRGT rings were identified.

If the H-08 thermal sleeve is removed and not successfully replaced, or physically separates during the outage and is not successfully replaced, there may not be a way to insert the H-08 drive rod into the CRDM successfully without the thermal sleeve in place. If the sleeve does not separate from the head during M2R25 and is not removed, the amount of wear on the thermal sleeve upper flange could lead to debris entering the sleeve, or the sleeve separating from the CRDM adapter tube in such a way as to bind control rod H-08 and not allow it to drop freely as required on a reactor trip. For these reasons, H-08 will not be reinstalled unless the thermal sleeve is replaced because reasonable assurance of H-08 operability will not exist once the head is disturbed.

This LAR addresses removal of control rod H-08 from the upcoming Unit 2 operating cycle M2C26 in the event the H-08 thermal sleeve cannot be successfully replaced during M2R25.

Although alternate reload core design and LAR analysis work has been in progress since the thermal sleeve issue was initially identified, required technical inputs were not finalized in time to support a non-exigent LAR submittal. Additionally, submitting the LAR only upon unsuccessful removal and replacement of the thermal sleeve would require an emergency LAR. Submitting on an exigent basis ensures the use of quality technical inputs and provides the NRC staff with more time, which allows for a quality review.

3. TECHNICAL EVALUATION

3.1 System Description

Unit 2 currently contains 53 full-length control rod assemblies divided into four control banks (Control Banks A, B, C, D) and five shutdown banks (Shutdown Banks A, B, C, D, E). Of the nine banks, Control Bank D is used for short-term control during normal at-power operation. The remaining control banks are normally used for reactor startup and shutdown. The shutdown banks provide additional negative reactivity to meet SDM requirements. During Modes 1 and 2, the shutdown banks are fully withdrawn from the core in accordance with TS 3.1.5 and as specified in the Core Operating Limits Report (COLR).

Control Rod H-08 is located in Control Bank D and is located in the center of the core as shown in Figure 1.

Figure 1: Control Rod Locations

North →

180°

90° (left) / 270° (right)

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2				SA-4 GR-2		CB-4 GR-2		CC-1 GR-1		CB-1 GR-1		SA-1 GR-1			
3					SD-4 GR-1		SB-4 GR-2		SB-1 GR-1		SC-1 GR-1				
4		SA-4 GR-1		CD-2 GR-2				SE-1 GR-1				CD-1 GR-1		SA-1 GR-2	
5			SC-4 GR-1										SD-1 GR-1		
6		CB-4 GR-1				CC-4 GR-2		CA-1 GR-1		CC-1 GR-2				CB-1 GR-2	
7			SB-4 GR-1										SB-1 GR-2		
8		CC-4 GR-1		SE-4 GR-1		CA-2 GR-2		CD-3 GR-2		CA-1 GR-2		SE-2 GR-1		CC-2 GR-1	
9			SB-3 GR-2										SB-2 GR-1		
10		CB-3 GR-2				CC-3 GR-2		CA-2 GR-1		CC-2 GR-2				CB-2 GR-1	
11			SD-3 GR-1										SC-2 GR-1		
12		SA-3 GR-2		CD-2 GR-1				SE-3 GR-1				CD-1 GR-2		SA-2 GR-1	
13					SC-3 GR-1		SB-3 GR-1		SB-2 GR-2		SD-2 GR-1				
14				SA-3 GR-1		CB-3 GR-1		CC-3 GR-1		CB-2 GR-2		SA-2 GR-2			
15															

0°

XX-Y / GR-A — XX - BANK NAME; Y - RCC NO; A - GROUP NUMBER

Control Bank	Number Rods	Shutdown Bank	Number of Rods
A	4	A	8
B	8	B	8
C	8	C	4
D	5	D	4
		E	4

3.2 Current Licensing Basis

Duke Energy self-performs all reload licensing analysis except for Loss of Coolant Accident (LOCA) for McGuire Nuclear Plant and requires NRC-approved reload methodologies to license the reload core. Reload analysis methodologies reviewed are included in the cycle-specific COLR per TS 5.6.5.

The COLR reload analysis methodologies are not invalidated by the removal of control rod H-08 from the McGuire 2 Cycle 26 (M2C26) core design. There is no direct reference to Control Bank D configuration in these methodologies except a figure of Control Bank D configuration and description of 53 control rod clusters in McGuire Unit 2 included as background information. Reload methodology analyses and supporting computer codes remain applicable to model and evaluate as designed/operated configuration of the plant, and the reload methodology is not dependent upon control bank configuration. Cycle-specific reload evaluations of TS limits, Safety Analysis limits, and Operating limits without control rod H-08 for M2C26 are performed to ensure reload analysis methodology limits remain satisfied and safety analysis limits remain bounded.

As described in UFSAR Section 4.2.3.2.1, "Reactivity Control Components":

The full length rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature variations. Two criteria have been employed for selection of the control groups. First the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, in view of the fact that some of these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that the power capability is met. The control and shutdown groups provide adequate shutdown margin which is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped assuming that the highest worth assembly remains fully withdrawn and assuming no changes in xenon or boron concentration. However, with all rod cluster control assemblies verified fully inserted by two independent means, it is not necessary to account for a stuck rod cluster control assembly in the shutdown margin calculation.

As described in UFSAR Section 4.3.2.4.12, "Rod Cluster Control Assemblies":

Fifty three full length Rod Cluster Control Assemblies are employed. The full length rod cluster control assemblies are used for shut-down and control purposes to offset fast reactivity changes associated with:

1. The required shutdown margin in the hot zero power, stuck rod condition,
2. The reactivity compensation as a result of an increase in power above hot zero power (power defect including Doppler, and moderator reactivity changes),
3. Un-programmed fluctuations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits),
4. Reactivity ramp rates resulting from load changes.

The allowed control bank reactivity insertion is limited at full power to maintain shutdown capability. Because of the reduction in the magnitude of the power defect with decreasing *power, control rod reactivity requirements are also reduced and more rod insertion is allowed.*

The control bank position is monitored and the operator is notified by an alarm if the limit is approached. The determination of the insertion limit uses conservative xenon distributions and axial power shapes. In addition, the rod cluster control assembly withdrawal pattern determined from these analyses is used in determining power distribution factors and in determining the maximum worth of an inserted rod cluster control assembly ejection accident.

Power distribution, rod ejection, and rod misalignment analyses are based on the arrangement of the shutdown and control groups of the rod cluster control assemblies. In preparing for power operation, all shutdown banks are withdrawn before withdrawal of the control banks is initiated. After which, Control Banks A, B, C, D are withdrawn sequentially in 50% overlap. The limits of rod positions and further discussion on the basis for rod insertion limits are provided in the Technical Specifications.

3.3 Impact on the Safety Analysis

The removal of control rod H-08 from Control Bank D is considered a permanent plant change for M2C26 and impacts all the nuclear design and safety analysis characteristics for this reload core design. As such, the reload design process, which is used for each new fuel cycle, has been followed to determine the nuclear design changes and impact to core and fuel performance, as well as impact to the accident analyses described in UFSAR Chapter 15 for control rod H-08 removed. This involved determining the nuclear design changes associated with core operation with control rod H-08 removed and evaluating the affected nuclear design parameters against a set of bounding values contained in the Reload Design Safety Analysis Review Checklist (REDSAR).

NRC-approved reload design methods and the REDSAR process are used to determine if the change in core design adversely impacts the bounding key safety parameters assumed in the Chapter 15 safety analysis. Additionally, impacts on Departure from Nucleate Boiling (DNB) and fuel thermal limits such as centerline fuel melt (CFM) due to the change in power distribution attributable to the new core design with control rod H-08 removed are reviewed.

Evaluation of impacts to core and fuel performance, as well as the impact to the safety analyses described in UFSAR Chapter 15 and REDSAR parameters, are documented in the cycle-specific Reload Safety Evaluation calculation to confirm the acceptability of safe operation with the new core design. There were no changes in methods or safety analysis limits used to perform the core reload design change process for M2C26 with control rod H-08 removed. The M2C26 core design with control rod H-08 removed shuffled two once-burned assemblies to reduce peaking in the center of the core to mitigate the loss of control rod H-08 in Nuclear Design analyses. Results of the safety analysis impact evaluation are described below.

Since the control rod in core location H-08 is in Control Bank D and this control bank is the first inserted control bank as described in the Rod Insertion Limits in the COLR, the removal of control rod H-08 potentially impacts all UFSAR Chapter 15 rod position results and associated peaking results. The removal of control rod H-08 impacts the calculation of and subsequent comparisons to some of the parameters assumed in the UFSAR Chapter 15 analysis. These parameters are:

- Available shutdown margin and most reactive stuck rod worth;
- Boron and boron worth with RCCAs inserted;

- Rod worth of the adjacent RCCAs with RCCAs inserted;
- The trip reactivity as a function of time;
- Available control bank worth for drop/insertion/withdrawal/ejection;
- Power distribution peaking limits (i.e., DNB) and fuel thermal limits

The impact of removing control rod H-08 on representative key parameters is discussed below. The analysis supporting the evaluation of these impacted parameters was performed using NRC approved methodology described in TS 5.6.5 of the COLR. The M2C26 COLR will be submitted to the NRC 60 days after cycle startup; however, with one exception, all COLR methodology references remain unchanged as a result of control rod H-08 being removed. The exception is an added footnote describing removal of RCCA H-08 and cycle-specific calculated COLR limits.

Shutdown Margin

The proposed change impacts the available shutdown margin (SDM). TS 3.1.1 states that the required SDM shall be within the COLR limit. Maintaining the SDM within this limit ensures the safety analysis described in Chapter 15 of the UFSAR remains bounding. Section 2.2.1 of the COLR provides the limit for Modes 1 and 2. An evaluation of the impact on the reduction of SDM due to the removal of control rod H-08 has been performed, and the results are presented in Table 1. The SDM is reduced from 1.804% ΔK/K to 1.457% ΔK/K, which remains bounded by the 1.3% ΔK/K limit for Modes 1 and 2 specified in COLR Section 2.2.1. By maintaining the 1.3% ΔK/K SDM limit, the safety analysis described in Chapter 15 of the UFSAR remains bounding with regards to SDM for accidents initiated in Modes 1 and 2. In addition, the worth of the most reactive stuck rod in an N-1 configuration, when considering control rod H-08 inserted, is 0.813% ΔK/K in core location F-10. With control rod H-08 removed, the worth of the most reactive stuck rod in a N-1 configuration (still at core location F-10) is 0.708% ΔK/K.

Table 1
Comparison of Effect on End-of-Life Shutdown Margin

	M2C25 RCCA In H-08	**M2C26 RCCA In H-08**	**M2C26 NO RCCA In H-08**
Control Rod Worth, % ΔK/K			
All Rods Inserted Worst Stuck Rod (N-1) and Core Location All Rods Inserted minus Worst Stuck Rod (N-1) Less 10%	6.879 0.802 (F-10) 6.077 5.469	6.987 0.813 (F-10) 6.174 5.557	6.360 0.708 (F-10) 5.652 5.087
Control Rod Requirements, % ΔK/K Power Defect Rod Insertion Allowance	 3.303 0.341	 3.332 0.371	 3.274 0.306
Total Requirements, % ΔK/K Shutdown Margin, % ΔK/K (reduced 0.05 % ΔK/K for analyzed burnup window)	3.644 1.775	3.703 1.804	3.580 1.457
Safety Analysis Limit, % ΔK/K	1.300	1.300	1.300

COLR Sections 2.2.1 and 2.2.2 also provide the required SDM limits for Modes 3, 4, and 5. Per these sections, SDM must be at least 1.3% ΔK/K in MODES 3 and 4, and it must be at least 1.0% ΔK/K in MODE 5. These SDM limits are maintained as a function of control rod position and reactor coolant system (RCS) critical boron concentration for Modes 3, 4, and 5. These limits are based on the SDM required for the most limiting analyses to establish SDM, which are the steam line break event from hot zero power (HZP) and the chemical and volume control system (CVCS) malfunction that results in a decrease in boron concentration in the RCS.

Other accidents impacted by SDM limits are Rod Ejection and uncontrolled rod withdrawal from subcritical or lower power conditions and at-power conditions, as described in TS Bases 3.1.1. By maintaining an SDM of greater than 1.3% ΔK/K, the steam line break event remains bounding. As discussed above, the removal of control rod H-08 does not result in an SDM of less than the limit of 1.3% % ΔK/K. A key parameter for the CVCS malfunction event is SDM. An evaluation of the effect on SDM with control rod H-08 removed and the highest worth RCCA stuck out shows that the SDM limits presented in the COLR remain bounding.

Operationally, the required RCS SDM boron concentrations for Modes 3, 4, and 5 will be higher with control rod H-08 removed in order to meet the COLR SDM limits. Table 2 below provides the minimum required shutdown boron concentration with all rods in (ARI) minus the most reactive stuck rod and ARI for beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) conditions.

Table 2
Minimum Required Shutdown Boron Concentration
with ARI Minus the Most Reactive Stuck Rod and ARI Conditions

ARI Minus Most Reactive Stuck Rod	RCCA in H-08 Required Boron (ppm)			No RCCA in H-08 Required Boron (ppm)		
	68°F	350°F	557°F	68°F	350°F	557°F
BOC	1809	1786	1592	1811	1795	1645
MOC	1604	1592	1395	1609	1629	1454
EOC	761	643	239	761	665	297
ARI	RCCA in H-08 Required Boron (ppm)			No RCCA in H-08 Required Boron (ppm)		
	68°F	350°F	557°F	68°F	350°F	557°F
BOC	1674	1665	1473	1695	1695	1524
MOC	1524	1521	1296	1552	1564	1370
EOC	671	567	150	695	609	223

<u>Boron Concentration and Boron Worth</u>

The removal of control rod H-08 was evaluated for impact on boron concentration and differential boron worth as a function of boron concentration in a rodded configuration. The removal of control rod H-08 increases the boron concentration and reduces boron worth (makes it more negative) as a function of boron concentration when all RCCAs are inserted into the core. This impacts the CVCS malfunction (i.e., boron dilution accident (BDA)) that results in a change in boron concentration requirements in the RCS for Modes 1, 2, 3, 4, and 5. TS require that the SDM in the various modes be above a certain minimum value.

The difference in boron concentration, between the value at which the relevant alarm function is actuated and the value at which the reactor is just critical, determines the time available to mitigate a BDA event. Mathematically, this time is a function of the ratio of these two concentrations, where a large ratio corresponds to a longer time. During the reload safety analysis for each new core, the above concentrations are checked to ensure that the value of this ratio for each mode is larger than the corresponding ratio assumed in the accident analysis. Each mode of operation covers a range of temperatures. Therefore, within that mode, the temperature which minimizes this ratio is used for comparison with the accident analysis ratio. For accident initial conditions in which the control rods are withdrawn, it is conservatively assumed, for the purposes of calculating the critical boron concentration, that the most reactive RCCA does not fall into the core at reactor trip.

Determination of the boron concentrations for calculation of the BDA ratio is performed conservatively assuming all RCCAs are inserted or most reactive RCCA out of the core to minimize the calculated ratio. Based on this conservative assumption, removal of control rod H-08 has an impact on the boron concentration assumed in the analysis for the BDA event. Table 2 provides an example of the minimum required shutdown boron concentration with ARI minus the most reactive stuck rod and ARI for BOC, MOC, and EOC conditions and shows an increase in required shutdown boron for various modes of operation with control rod H-08 removed. Cycle-specific BDA ratio evaluations are performed, and required shutdown boron concentrations are determined to ensure that the limit assumed in the safety analysis remains bounding. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR Section 15.4.6 BDA.

Note: Post LOCA Subcriticality boron concentrations calculated to support UFSAR Section 15.6.5 LOCA are calculated at conservative ARO configuration; therefore, these results are not impacted by the removal of control rod H-08.

Trip Reactivity

The removal of control rod H-08 reduces the trip reactivity as a function of rod insertion position, which reduces the trip reactivity as a function of time after the RCCAs begin to fall. The normalized trip reactivity as a function of RCCA insertion position and normalized trip reactivity as a function of time after the RCCAs begin to fall is presented in the UFSAR. An evaluation of the effects of the removal of control rod H-08 shows that the trip reactivity as a function of RCCA insertion position and the resulting trip reactivity as a function of time after the RCCAs begin to fall remains bounding. Table 3 provides a comparison of the trip reactivity as a function of rod position for M2C26 with and without control rod H-08 inserted. Therefore, the removal of control rod H-08 does not impact the trip reactivity assumed in UFSAR Chapter 15 events.

Table 3
Trip Reactivity Values

Control Rod Position (% Inserted)	RCCA in H-08 Normalized Rod Worth (%ΔK/K)	NO RCCA in H-08 Normalized Rod Worth (%ΔK/K)	REDSAR Normalized Rod Worth (%ΔK/K)
0	0.000	0.000	0.000
11.1	0.010	0.011	0.007
22.1	0.015	0.016	0.012
33.2	0.022	0.022	0.016
44.2	0.033	0.032	0.025
55.3	0.054	0.051	0.047
66.4	0.097	0.091	0.088
77.4	0.195	0.192	0.182
88.5	0.498	0.507	0.450
99.6	0.995	0.995	0.990
100.0	1.000	1.000	1.000

Moderator Temperature Coefficient (MTC)

UFSAR Chapter 15 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions to ensure that the accident results are bounding.

The consequences of accidents that cause core overheating must be evaluated when the MTC is positive. Such accidents include the rod withdrawal transient from any power level, turbine trip, and loss of forced reactor coolant flow. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative. Such accidents include sudden feedwater flow increase and steam line break.

In order to ensure a bounding accident analysis, the MTC is assumed to be its most limiting value for the analysis conditions appropriate to each accident. The bounding value is determined by considering rodded and unrodded conditions, whether the reactor is at full or zero power, and whether it is the BOC or EOC life. The most conservative combination appropriate to the accident is then used for the analysis. The MTC is modeled in the safety analysis as a density coefficient based on the core average coolant density.

The removal of control rod H-08 only slightly impacts the moderator temperature coefficient calculated at the conservative bounding conditions determined for the UFSAR accident analyses. Moderator temperature coefficient results for M2C26 with control rod H-08 in and out are shown in Table 4 and confirm that the limit assumed in the safety analysis remains bounding. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR for the above listed events.

Table 4
MTC Limit Summary for M2C26 with and without H-08 Control Rod

Limit Description	**Limit (pcm/°F)**	**M2C26 Reload Value - RCCA in H-08 (pcm/°F)**	**M2C26 Reload Value - NO RCCA in H-08 (pcm/°F)**
Tech Spec/COLR Limit			
Most Positive Hot Full Power (HFP) All Rods Out (ARO) MTC	< 0	-6.58	-6.65
Most Negative EOC, HFP, ARO MTC Limit	> -43.00	-41.00	-40.96
Most Negative EOC, HFP, ARO, 300 ppmb Surv Limit	≥ -36.50	-35.13	-35.10
Most Negative EOC, HFP, ARO, 60 ppmb Surv Limit	≥ -41.25	-39.44	-39.40
REDSAR Limit			
Most Positive HZP ARO MTC, pcm/°F	≤ 7.0	+0.45	+0.36
Most positive EOC, HFP, pcm/°F	≤ -24	-32.04	-32.02
Most negative EOC, HFP, pcm/°F (includes uncertainties)	≥ -51	-49.00	-48.96
REDSAR Limit			
Least negative EOC, HZP, pcm/°F	≤ -15	-18.71	-18.71
REDSAR Limit			
Most Positive (MOC), HFP ARO MTC, pcm/°F	≤ -10.0	-15.12	-16.59

UFSAR Chapter 15 Accident Analyses Impacts from Removal of Control Rod H-08

Removal of control rod H-08 from M2C26 has an impact on most comparisons to UFSAR Chapter 15 accident analysis parameters routinely evaluated as part of the reload design process. In addition to SDM, MTC, trip reactivity, boron concentration, and boron worth accident analysis parameters discussed above, accident analysis impact on control rod worth, peaking limits (DNB and CFM), and other accident analysis parameters were also evaluated. The removal of control rod H-08 impacts these parameters by reactivity effects on calculated boron concentrations, control rod position reactivity worth, or power distribution effects due to different control rod pattern during rodded power maneuvers.

Cycle-specific evaluations were performed to determine if the change in core design adversely impacts the REDSAR bounding key safety parameters assumed in the UFSAR Chapter 15 safety analysis and impact on DNB and fuel thermal limits due to the change in power distribution. The REDSAR bounding key safety parameters are developed in UFSAR Chapter 15 accident analysis of record (AOR) to ensure expected reactivity parameters and peaking conditions for various accident conditions are bounded, therefore if the cycle specific evaluation meets the bounding parameters the AOR remains satisfied. Results of the cycle specific evaluations confirm that the limits assumed in the safety analysis remain bounding; therefore, the removal of control rod H-08 from M2C26 does not impact the results presented in the UFSAR Chapter 15 accident analyses. Results and discussion of the UFSAR Chapter 15 accident analyses for M2C26 with control rod H-08 removed are provided below.

HZP and HFP Steam Line Break (SLB) Accident

For HZP SLB, if the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, the current Chapter 15 analysis shows the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The increase in the core power could result in CFM and DNB.

For HFP SLB, SLB break size, break location, and MTC determine simultaneously the magnitude of the pre-trip power increase. Therefore, for a given break, the maximum pre-trip power level is determined by analyzing a range of negative MTC values which bound the current core designs. The MTC is modeled as a density coefficient based on the core average coolant density. System analyses are performed to determine a limiting CFM case and a limiting DNB case.

The removal of control rod H-08 also impacts the localized reactor core power distribution for events where a return to power or increase in power with control rods inserted can occur, such as the SLB event from zero power and HFP. The Duke Energy reload core methodology for the SLB event from zero power and HFP uses Safety Analysis and Nuclear Design methods to determine if the reference transient analysis state points (reactor power level, inlet temperature, pressure, flow, and core boron concentration) reported in the REDSAR remain bounding for the reload core. If the transient analysis state points are not bounding, the transient analysis is re-performed. A DNB analysis is then performed using the power peaking factors for the reload core.

Cycle-specific parameter evaluations for SLB accident are presented in Table 5 and confirm that the limits assumed in the safety analysis remain bounding. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR section for the SLB accident.

Table 5
Steam Line Break REDSAR Parameter Comparisons

M2C26 Reload Value Statepoint Peaking Evaluations						
SLB Limit RCCA Configuration	**Pbar**		**$F_{\Delta H}$**		**F_Q**	**Limit**
	Max Assembly Rad Peak	**Max Assy Rad Peak Loc**	**Max Pin Rad Peak (FΔH)**	**Max Pin Rad Peak Loc**	**Max Pin Total Pk (Fq)**	
HZP SLB						
Statepoint Peaking H-08	5.176	E11	5.822	E11	8.594	Pass DNB Eval
Statepoint Peaking NO H-08	4.653	E11	5.299	E11	8.134	Pass DNB Eval
Peaking Margins						
HZP SLB Statepoint	**MDNBR**		**CFM**			**MDNBR/CFM**
MDNBR/CFM H-08	1.73 (Ref. 24)		26.5%			1.3275 / >0%
MDNBR/CFM NO H-08	1.91 (Ref. 15)		30.4%			1.3275 / >0%
Axial Power Distribution						
Axial Power Distribution	**Power Fraction in Upper Third of Core**					
H-08	46.24%					≤ 68%
NO H-08	46.76 %					≤ 68%
HFP SLB Peaking Margins						
HFP SLB Statepoint	**DNBR**		**CFM**			
DNBR/CFM H-08	4.92%		4.55%			>0%
DNBR/CFM NO H-08	9.82%		4.15%			>0%

Locked Rotor Accident (LRA)

The LRA postulated is an instantaneous seizure of a reactor coolant pump rotor. The LRA is analyzed assuming offsite power maintained and offsite power lost conditions. A DNB analysis is then performed using the power peaking factors at ARO conditions and axial skewed power distribution for the reload core. The removal of control rod H-08 could impact the localized reactor core power distribution for the LRA event.

Cycle-specific parameter evaluations for LRA accident are presented in Table 6 and confirm that positive margin exists. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR section on LRA.

Table 6
LRA DNB Pin Census Results

RCCA Configuration	**DNB Limit (%)**	**OSPL Power (%)**	**DNB Margin (%)**		**OSPM Power (%)**	**DNB Margin (%)**
H-08	>0%	81.81	5.09		91.06	5.46
NO H-08	>0%	81.81	6.64		91.06	7.08

Uncontrolled Bank Withdrawal (UCBW) from Subcritical or Low Power Startup Condition

An RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs resulting in a power excursion. The maximum reactivity insertion rate analyzed in the detailed plant analysis is that occurring with the simultaneous, complete overlap withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed. The most limiting axial and radial power shapes, associated with having the two highest combined worth banks in their high worth position, are assumed in the DNB analysis. The removal of control rod H-08 will impact the localized reactor core power distribution for events where a power excursion occurs.

Cycle-specific parameter evaluations for UCBW from subcritical accident are presented in Table 7 and confirm that positive margin exists. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR section on UCBW accident from subcritical.

Table 7
UCBW from Subcritical Max Withdrawn Worth, Reactivity Insertion Rate, and Peaking Evaluation Results

RCCA Configuration	**Limit Max Withdrawn Worth, (pcm)**	**Max Calculated Value (pcm)**	**RCCA Configuration**	**Max Calculated Value (pcm)**
H-08	≤ 3250	2861	NO H-08	2599
RCCA Configuration	**Limit Reactivity Insertion Rate (pcm/sec)**	**Max Calculated Value (pcm/sec)**	**RCCA Configuration**	**Max Calculated Value (pcm/sec)**
H-08	≤ 65	32.1	NO H-08	31.1
UCBW from Subcritical Statepoint Peaking Evaluation Margins				
RCCA Configuration	Limit	DNB Margin	Limit	CFM Margin
H-08	>0%	9.76%	>0%	17.78%
NO H-08	>0%	3.32%	>0%	21.34%

Uncontrolled Bank Withdrawal (UCBW) at Power

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad the Reactor Protection System is designed to terminate any such transient before DNB occurs.

The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combinations of the two control banks having the maximum combined worth at maximum speed. Axial and radial power shapes, associated with having the rod bank maneuvers as described above are evaluated in the DNB analysis. The removal of control rod H-08 will impact the localized reactor core power distribution for events where a rod power maneuver occurs.

Cycle-specific parameter evaluations for UCBW from power accident are presented in Table 8 and confirm that positive DNB margin exists. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR section on UCBW accident at power.

Table 8
UCBW at Power Max Withdrawn Worth, Reactivity Insertion Rate, and Peaking Evaluation Results

Power & RCCA Configuration	**Limit Max Withdrawn Worth, (pcm)**	**Max Calculated Value (pcm)**	**RCCA Configuration**	**Max Calculated Value (pcm)**
100%FP H-08	≤ 575	438	100%FP NO H-08	360
50%FP H-08	≤ 2000	1450	50%FP NO H-08	1278
10%FP H-08	≤ 3050	2434	10%FP NO H-08	2202
Power & RCCA Configuration	**Limit Reactivity Insertion Rate (pcm/sec)**	**Max Calculated Value (pcm/sec)**	**RCCA Configuration**	**Max Calculated Value (pcm/sec)**
100%FP H-08	≤ 45	12.1	100%FP NO H-08	10.7
50%FP H-08	≤ 45	19.6	50%FP NO H-08	18.9
10%FP H-08	≤ 45	22.8	10%FP NO H-08	22.1
UCBW at Power Statepoint Peaking Evaluation Margins				
Power & RCCA Configuration	Limit	DNB Margin	Limit	CFM Margin
All Powers H-08	>0%	6.87%	>0%	12.36%
All Powers NO H-08	>0%	1.89%	>0%	12.34%
UCBW at Power Excore Detector Response (Rod Shadowing)				
Power & RCCA Configuration	Excore Response Limit (%)		Calculated Excore Response (%)	
50%FP H-08	≤5%		0.0	
50%FP NO H-08	≤5%		0.0	
10%FP H-08	≤12%		11.5	
10%FP NO H-08	≤12%		10.7	

Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)

RCCA misoperation accidents include:
a. One or more dropped RCCAs within the same group
b. A dropped RCCA bank
c. Statically misaligned RCCA
d. Withdrawal of a single RCCA

Withdrawal of a Single RCCA

Withdrawal of a single RCCA (SUCR) results in both a positive reactivity insertion tending to increase core power and an increase in local power density in the core area associated with the withdrawn RCCA, which could eventually result in DNB.

For SUCR, the maximum positive reactivity insertion rate is greater than that for the maximum speed withdrawal of the most reactive single Control Bank D RCCA from at or above its insertion limit, accounting for uncertainties in the indicated RCCA position. Axial and radial power shapes, associated with having a single Control Bank D rod withdrawn as described above are evaluated in the DNB analysis. The removal of control rod H-08 will impact the localized reactor core power distribution for events where a single Control Bank D rod withdrawal occurs. Note: for control rod H-08 removed in M2C26, a SUCR accident event cannot occur for this core location.

Cycle-specific parameter evaluations for SUCR accident are presented in Table 9 and confirm that positive margin exists. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR section on SUCR accident.

Note: Statically misaligned RCCA DNB accident analysis is bounded by the more conservative SUCR DNB analysis. Therefore, if no pins are in DNB from the SUCR DNB analysis and it passes accident analysis, statically misaligned RCCA accident also passes accident analysis.

Table 9
SUCR (Single Rod W/D) Max Withdrawn Worth, Reactivity Insertion Rate, and Peaking Evaluation Results

Power & RCCA Configuration	**Limit Max Withdrawn Worth, (pcm)**	**Max Calculated Value (pcm)**	**RCCA Configuration**	**Max Calculated Value (pcm)**
100%FP H-08	≤ 105	87	100%FP NO H-08	83
Power & RCCA Configuration	**Limit Reactivity Insertion Rate (pcm/sec)**	**Max Calculated Value (pcm/sec)**	**RCCA Configuration**	**Max Calculated Value (pcm/sec)**
100%FP H-08	≤ 25	1.53	100%FP NO H-08	1.5
SUCR Statepoint Peaking Evaluation Margins				
RCCA Configuration	Limit Pins in DNB	Pins in DNB and DNB Margin	Limit	CFM Margin
All Powers H-08	<0%	0% pins & 7.51%	>0%	30.07%
All Powers NO H-08	<0%	0% pins & 10.27%	>0%	32.50%
SUCR (Single Rod W/D) Minimum Ratio of 2nd Highest Excore Flux Signal				
RCCA Configuration	Min Ratio of 2nd Highest Excore Flux Signal		Minimum Calculated 2nd Highest Excore Flux Signal Ratio	
H-08	≥0.9		0.9486	
NO H-08	≥0.9		0.9498	

<u>One or more Dropped RCCAs within same Group or Dropped RCCA Bank (DRA)</u>

For the one or more RCCAs from the same group dropped which do not result in a reactor trip, power may be reestablished either by reactivity feedback or control bank withdrawal. Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. The equilibrium process without control system interaction is monotonic, thus removing power overshoot as a concern, and establishing the automatic rod control mode of operation as the limiting case. For a dropped RCCA event in the automatic rod control mode, the Rod Control System detects the drop in power and initiates control bank withdrawal. Power overshoot may occur due to this action by the automatic rod controller after which the control system will insert the control bank to restore nominal power.

DRA analysis statepoints are calculated and nuclear design models are used to obtain hot channel factors at conditions consistent or conservative with respect to the primary system conditions and reactor power. By incorporating the primary conditions from the transient and the hot channel factor from the nuclear analysis, the DNB design basis is shown to be met.

The DRA is evaluated for all the dropped rod combinations of control and shutdown bank groups as described above and a peaking evaluation performed to compare to REDSAR peaking limits to ensure DNB would not occur for DRA. The removal of control rod H-08 will impact the localized reactor core power distribution for DRA. Note: for control rod H-08 removed in M2C26, a DRA accident event cannot occur for control rod in this core location. Also, Control Bank D rod group with H-08 excluded will go from 3 to 2 control rods.

Cycle-specific parameter evaluations for DRA are presented in Table 10 and confirm that positive margin exists. Therefore, the removal of Control Rod H-08 does not impact the results presented in the UFSAR section on DRA.

Table 10
DRA Max Dropped Rod Worth, Control Bank D Withdrawn Worth, and Peaking Results

<table>
<tr><th>Burnup & RCCA Configuration</th><th>Limit Max Control Bank D Worth Available for Withdrawal, (pcm)</th><th>Max Calculated Value (pcm)</th><th>Burnup & RCCA Configuration</th><th>Max Calculated Value (pcm)</th></tr>
<tr><td>BOC, H-08</td><td>≤ 375</td><td>316</td><td>BOC, NO H-08</td><td>252</td></tr>
<tr><td>MOC, H-08</td><td>≤ 450</td><td>367</td><td>MOC, NO H-08</td><td>285</td></tr>
<tr><td>EOC, H-08</td><td>≤ 500</td><td>445</td><td>EOC, NO H-08</td><td>358</td></tr>
<tr><td colspan="5">DRA Peaking Evaluation Margins</td></tr>
<tr><td>RCCA Configuration</td><td colspan="2">Limit</td><td colspan="2">Calculated Value</td></tr>
<tr><td>Initial FΔH H-08</td><td colspan="2"><1.60</td><td colspan="2">1.5508</td></tr>
<tr><td>Initial FΔH NO H-08</td><td colspan="2"><1.60</td><td colspan="2">1.5259</td></tr>
<tr><td>FΔH as Function of Dropped Rod Worth H-08 & NO H-08</td><td colspan="2">See Figure 2</td><td colspan="2">See Figure 2</td></tr>
<tr><td>DRA Axial Power Shape vs Burnup</td><td colspan="4">Evaluated in cycle specific RSE and determined to be bounding</td></tr>
<tr><td colspan="5">DRA 2nd Highest Excore Tilt for all Dropped Rods</td></tr>
<tr><td>RCCA Configuration</td><td colspan="2">Limit</td><td colspan="2">Calculated Value</td></tr>
<tr><td>H-08</td><td colspan="2">≥0.80</td><td colspan="2">0.8581</td></tr>
<tr><td>NO H-08</td><td colspan="2">≥0.80</td><td colspan="2">0.8578</td></tr>
</table>

Rod Ejection Accident (REA)

REA is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a rod cluster control assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Certain features in the McGuire units are intended to preclude the possibility of a rod ejection accident, or to limit the consequences if the accident were to occur. These include a conservative mechanical design of the rod housings and a nuclear design, which lessens the potential ejection worth of RCCAs and minimizes the number of RCCAs inserted at high power levels.

Ejected rod worths are calculated in cycle specific evaluations using three dimensional steady state neutronics codes, which have been approved for reload design analyses. Ejected rod worth calculations are performed assuming that the control banks containing the ejected rod are inserted to the power dependent rod insertion limit, including uncertainties. For ejected rod worth calculations performed at power, no credit is taken for the reactivity feedback resulting from the increase in fuel temperature and moderator temperature during the transient. The effects of transient xenon conditions are also considered. Confirmation that rod ejection hot channel factors remain bounding for reload cores is accomplished through a series of three dimensional static calculations using steady state neutronic codes approved for reload design analysis.

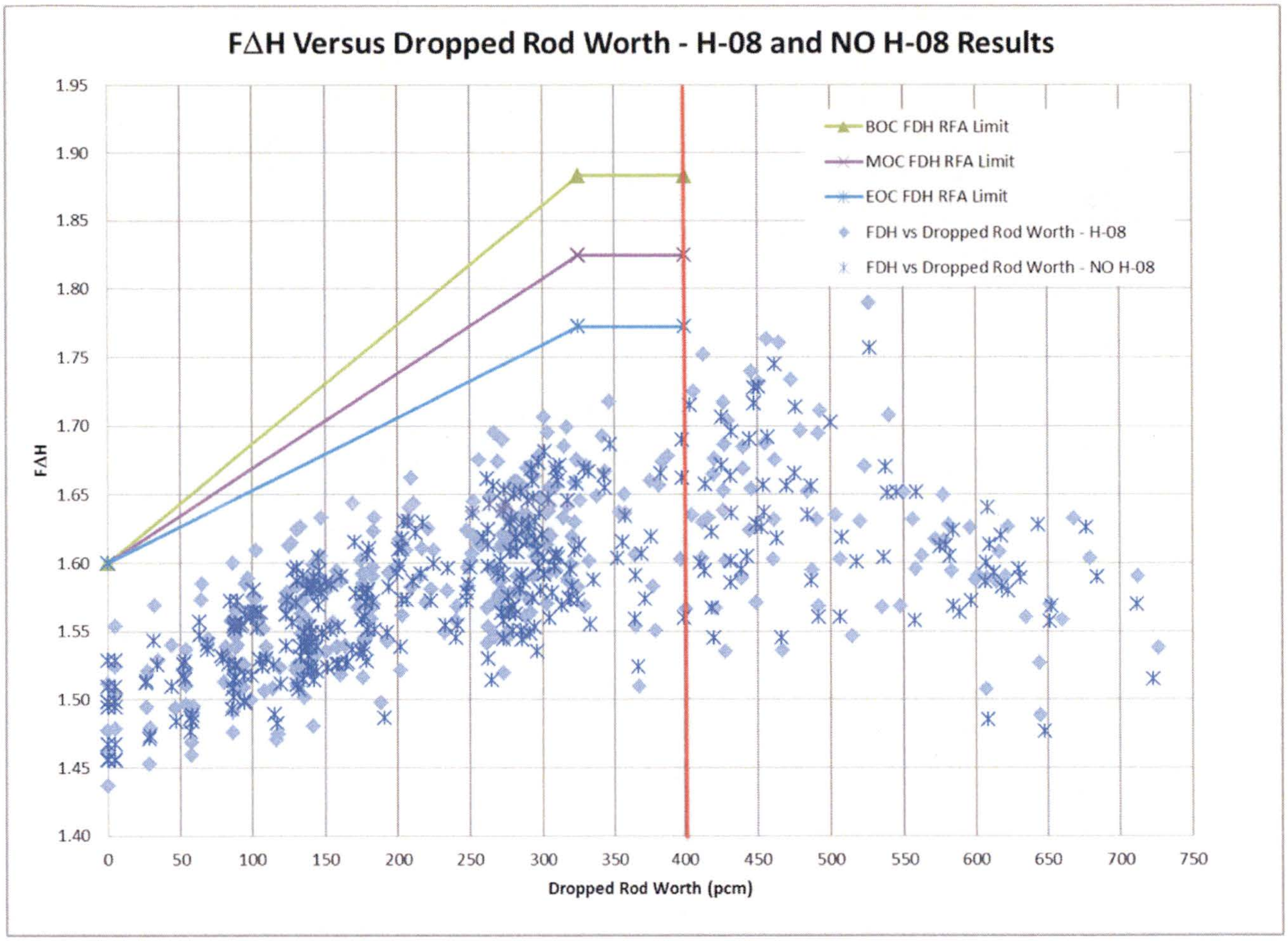
The REA is evaluated for the plant and control bank conditions described above and bounding REA REDSAR limits and a peaking evaluation performed to determine the number of pins in DNB for REA. The removal of control rod H-08 will impact the localized reactor core power distribution for REA. Note: For control rod H-08 removed in M2C26, an REA accident will not occur in the H-08 location for this configuration.

Cycle-specific parameter evaluations for REA are presented in Table 11 and confirm that the limits assumed in the safety analysis remains bounding. Therefore, the removal of control rod H-08 does not impact the results presented in the UFSAR section on REA.

Table 11
REA Max Ejected Rod Worth and Peaking Evaluation Results

Power, Burnup & RCCA Configuration	**Limit Ejected Rod Worth, ($)**	**Max Calculated Value ($)**	**RCCA Configuration**	**Max Calculated Value ($)**
HZP, BOC H-08	≤ 1.32	0.74	HZP, BOC NO H-08	0.63
HFP, BOC H-08	≤ 0.19	0.11	HFP, BOC NO H-08	0.10
HZP, EOC H-08	≤ 1.45	1.26	HZP, EOC NO H-08	1.01
HFP, EOC H-08	≤ 0.26	0.18	HFP, EOC NO H-08	0.17
REA Maximum Total Peaking Factor (Fq)				
Power, Burnup & RCCA Configuration	**Limit Total Peaking Factor (Fq)**	**Max Calculated Value (Fq)**	**RCCA Configuration**	**Max Calculated Value (Fq)**
HZP, BOC H-08	≤ 19.60	10.35	HZP, BOC NO H-08	8.66
HFP, BOC H-08	≤ 4.75	3.29	HFP, BOC NO H-08	3.19
HZP, EOC H-08	≤ 20.78	17.91	HZP, EOC NO H-08	14.97
HFP, EOC H-08	≤ 4.84	3.97	HFP, EOC NO H-08	3.79
REA Pins in DNB Census Results				
Burnup and RCCA Configuration	Pin DNB Census Limit (%)	Calculated Pin DNB Census (%)	Burnup and RCCA Configuration	Calculated Pin DNB Census (%)
HFP, BOC H-08	<22%	6.68%	HFP, BOC NO H-08	6.25%
HFP, EOC H-08	<22%	5.69%	HFP, EOC NO H-08	4.24%

Figure 2



<u>Miscellaneous Safety Analysis Limits</u>

Miscellaneous Safety Analysis limits such as delayed neutron data (beta and prompt neutron lifetime), Doppler temperature coefficients, and fuel temperatures are not significantly impacted by the unrodded configuration of the core. These parameters are more driven by the core design. Since the shuffle of two once-burned assemblies in the M2C26 core design with control rod H-08 removed did not significantly impact the core design characteristic, the miscellaneous Safety Analysis limits changes were negligible. Cycle-specific parameter evaluations of these safety analysis limits show negligible change and confirm that the limit assumed in the safety analysis remains bounding.

Safety Analysis Evaluation Summary

To summarize, the impact of the removal of control rod H-08 in M2C26 on the nuclear design and safety analysis on all UFSAR Chapter 15 events accident analyses has been evaluated using the NRC-approved methods described in TS 5.6.5. These NRC-approved reload design methods were used to determine if the change in core design adversely impacts the bounding key safety parameters assumed in the UFSAR Chapter 15 Safety Analysis and impacts on DNB and CFM due to the change in power distribution attributable to the new core design with control rod H-08 removed. Cycle-specific parameter evaluations for UFSAR Chapter 15 Safety Analysis parameters confirm that the limit assumed in the safety analysis remains bounding for all UFSAR Chapter 15 Safety Analysis accidents.

Therefore, the removal of control rod H-08 for M2C26 does not impact the results presented in UFSAR Chapter 15. Table 12 presents a summary of the impact of removal of control rod H-08 on each Chapter 15 Safety Analysis accident. Observed cycle-specific results from the UFSAR Chapter 15 Safety Analysis technical evaluation with control rod H-08 control rod removed are summarized below:

- UFSAR Chapter 15 accidents with rod worth limits show the available Control Bank D worth for drop/insertion/withdrawal will be less due to removal of H-08 control rod from the M2C26 core.
- Rod Ejection Accident (REA) ejected rod worth and peaking results were reduced versus REDSAR limit because power is anchored toward center of core with no control rod in H-08 during the REA. REA bounding initial conditions assumption for the safety analysis remain unchanged for this cycle, and without a control rod in core location H-08, an REA will not occur for this configuration.
- Dropped RCCA Bank (DRA) peaking versus maximum REDSAR limit was reduced, and available Control Bank D worth for withdrawal is reduced with control rod H-08 removed. DRA Control Bank D rod group with H-08 will go from 3 to 2 control rods.
- HZP SLB peaking results were reduced because power was anchored toward center of core with no control rod in H-08 and max stuck rod out. HFP SLB peaking results changed to reflect peaking for control D bank configuration evaluated during rod maneuvers and results show improved DNB margin.
- BDA calculated boron concentrations reflect control rod H-08 removed from control bank configuration analyzed for required operating Mode boron conditions in calculation of ratios.
- Single Rod Withdrawal peaking and Control Bank D worth for withdrawal were reduced because power was anchored toward center of core with no control rod in H-08 during accident.
- Uncontrolled Control Bank Withdrawal (UCBW) at power max withdrawn worth and reactivity insertion rate. However, minimum DNB peaking margins were reduced because heavily rodded configurations peaking increased in center of core during rod maneuvers attributed to Control Banks D, C, and B being deeply inserted, which moves power to unrodded center of core. CFM peaking was insignificantly impacted.
- UCBW from subcritical max withdrawn worth and reactivity insertion rate were reduced for control rod H-08 removed. However, minimum peaking margins were reduced because at HZP, heavily rodded configurations peaking increased in center of core during rod maneuvers attributed to Control Banks D, C, and B being deeply inserted, which moves power to unrodded center of core.
- SDM and maximum stuck rod worth were reduced due to removal of control rod in H-08 with subsequent reduction in available rod worth; however, adequate margin to the SDM limit remains.

Table 12
Impact on UFSAR Chapter 15 Accident Analyses

#	UFSAR	Description	Comments
1	15.1.1	Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature	Bounded by 15.1.2 and 15.1.3.
2	15.1.2	Feedwater System Malfunction Causing an Increase in Feedwater Flow	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
3	15.1.3	Excessive Increase in Secondary Steam Flow	MTC and trip reactivity remains bounding. DTC and other parameters remain unchanged.
4	15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	See 15.1.5 analysis.
5	15.1.5	Steam System Piping Failure	HZP Steam Line Break (SLB) SDM, MTC, boron worth, max stuck rod out peaking, trip reactivity, and DNB ratio remains bounding. HFP SLB MTC, statepoint peaking for DNB and CFM remain bounding. DTC and other parameters remain unchanged.
6	15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	Not applicable .
7	15.2.2	Loss of External Electrical Load	Bounded by 15.2.3.
8	15.2.3	Turbine Trip	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
9	15.2.4	Inadvertent Closure of Main Steam Isolation Valves	Bounded by 15.2.3.
10	15.2.5	Loss of Condenser Vacuum and Other Events Causing Turbine Trip	Bounded by 15.2.3.
11	15.2.6	Loss of Non-Emergency AC Power to the Plant Auxiliaries	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged .
12	15.2.7	Loss of Normal Feedwater Flow	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
13	15.2.8	Feedwater System Pipe Break	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
14	15.3.1	Partial Loss of Forced Reactor Coolant Flow	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
15	15.3.2	Complete Loss of Forced Reactor Coolant Flow	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
16	15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	MTC, SDM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
17	15.3.4	Reactor Coolant Pump Shaft Break	Bounded by 15.3.3.

#	UFSAR	Description	Comments
18	15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition	MTC, max withdrawn rod worth and insertion rate, statepoint peaking DNB and CFM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
19	15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	MTC, max withdrawn rod worth and insertion rate, statepoint peaking DNB and CFM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
20	15.4.3	Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)	MTC, available withdrawn rod worth and insertion rate, dropped rod worth, statepoint peaking, and DNB and CFM, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
21	15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	MTC and trip reactivity remains bounding. DTC and other parameters remain unchanged.
22	15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Required SDM limits remain bounding. Boron worth remains bounding. All other analysis parameters not impacted.
23	15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	No impact. Inadvertent loading is detected using incore instrumentation during startup testing.
24	15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	MTC, ejected rod worth, total peaking (Fq), and DNB, and trip reactivity remains bounding. DTC and other parameters remain unchanged.
25	15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	There are no relevant physics parameters. No impact.
26	15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Bounded by 15.5.1.
27	15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	There are no relevant physics parameters. No impact.
28	15.6.2	Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment	There are no relevant physics parameters. No impact.
29	15.6.3	Steam Generator Tube Failure	There are no relevant physics parameters. No impact.
30	15.6.5	Loss-of-Coolant Accident	The only relevant physics parameter for LOCA is the initial power distribution. Linear heat flux (kw/ft) limits are established as a function of core elevation and evaluated in cycle-specific Maneuvering Analysis (MA). Fq limits remain bounded.
31	15.7	Radioactive Release From a Subsystem or Component	Analysis parameters not impacted.

#	UFSAR	Description	Comments
32	15.8	ATWS	Trip reactivity remains bounding. All other analysis parameters not impacted. ARO MTC is not impacted by the removal of Control rod H-08.
Note: UFSAR Sections 15.4.5, *A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate;* and 15.6.4, *Spectrum of BWR Steam System Piping Failures Outside Containment* apply to boiling water reactors and are not applicable to McGuire.			

Impact on Operating Analysis Support

Nuclear Design reload analysis methodologies are used to perform cycle-specific calculations (MA and Startup and Operational Report (SOR)) to support TS limits and surveillance and to generate data to support the startup and operation of the M2C26 core. Removal of control rod H-08 from the M2C26 core design does not invalidate the methodologies used in the development of the Nuclear Design models for M2C26 nor in the performance of the cycle-specific reload analyses MA and SOR as described in these methodology reports.

The Nuclear Design methodology and computer code package (CASMO-4/SIMULATE-3) currently used are applicable to model and evaluate as designed/operated configuration of the plant. Cycle-specific reload evaluations of TS limits and Operating limits without control rod H-08 for M2C26 are performed to ensure reload analysis methodology limits remain satisfied. The CASMO-4/SIMULATE-3 models reactivity parameter and power distribution prediction performance would not change due to removal of control rod H-08 from M2C26 core and the methodology is not dependent upon control bank configuration. The Core Operating Limits Report Methodology is not constrained by the removal of H-08 since explicit modeling of the core is employed in the verification of thermal limits and development of monitoring factors.

SOR data and operation characteristics for M2C26 will be different with control rod H-08 removed from Control Bank D configuration (i.e., Control Bank D rod worth, integral rod worth, AFD control, etc.). However the data generation methodology is not constrained by the removal of H-08 since explicit modeling of the as design configuration of the core is employed in the generation of the SOR data. These differences will be reflected in the cycle-specific SOR and communicated to the site in the Reload Change Document to identify expected changes in core design and behavior and training needs .

Cycle-specific MA results showed acceptable analysis margin with the current COLR Axial Flux Difference (AFD) limits and Rod Insertion Limits results with control rod H-08 removed from the M2C26 core. Differences are expected in TS 3.2.1 and 3.2.2 monitoring factors due to removal of H-08 from M2C26 core, but the methodology is not constrained by the removal of H-08 since explicit modeling of the core are employed in the verification of thermal limits and development of monitoring factors.

Other licensed methodology or analyses used in cycle-specific reload analyses to license the core were reviewed and are summarized as follows:

- Dynamic Rod Worth Measurement (DRWM) Using CASMO/SIMULATE methodology used in the SOR. Direct reference to Control Bank D configuration is simply descriptive information (i.e., figure of control bank configuration), and the method remains valid based upon arguments made previously for the methodologies above.

- MNS UFSAR Chapter 4, Reactor, Chapter 7, Instrumentation and Controls, and Chapter 15, Accident Analysis. Direct references to Control Bank D configurations are inferred and acceptable when NRC-approved reload methodologies are used to model and evaluate designed/operated configuration of the plant.
- Criticality Analysis of McGuire Spent Fuel Storage Racks and McGuire / Catawba Isotopic Inventory Calculation have no direct reference to Control Bank D configuration in the calculation. REDSAR limits for these analysis are evaluated in cycle-specific analysis and verified to remain bounding.

Conclusion

M2C26 Reload Safety Evaluation for H-08 Contingency Redesign validated all cycle-specific REDSAR safety analysis limits and determined the UFSAR Chapter 15 accident analyses remain bounding with respect to the M2C26 Safety Analysis Physics Parameters (SAPP), MA, SOR, and Thermal-Hydraulic parameters with control rod H-08 removed.

3.4 Field Work Required to Remove Control Rod H-08 from Service

Control Rod H-08 will be removed from service by performing the following work items, which will be evaluated in accordance with appropriate design change procedures:

- Unlatch the control rod drive shaft from the RCCA and CRDM and completely remove the drive shaft from the reactor vessel
- Remove RCCA from the fuel assembly located in core location H-08
- Install cap on H-08 CRDM adapter
- Install a thimble plug in the fuel assembly located in core location H-08 to maintain proper reactor coolant flow through the fuel assembly
- Remove H-08 inputs to the Digital Rod Position Indication (DRPI) software
- Modify plant computer position indication and alarm points for H-08
- Remove rod control system fuses for control power to the H-08 CRDM

Modifications to the DRPI software resulting from the removal of Control Rod H-08 have no impact on the reactor protection system. DRPI is a non-safety related system independent of the rod control and reactor protection systems. By adjusting the software to remove alarms associated with Control Rod H-08, DRPI continues to function for all other control rods.

These changes are reviewed and approved by McGuire engineering using site procedures for design changes.

3.5 Evaluation of Potential Design Impacts

Thermal-hydraulic impacts

The H-08 CRDM adaptor cap provides an equivalent hydraulic flow-path to the CRDM housing as the original H-08 rodded configuration. Thus removal of the Control Rod H-08 drive shaft and RCCA has a negligible effect on thermo-siphoning inside the CRDM housing during normal operating conditions (i.e., water circulation inside the CRDM due to temperature differential between the outside and inside of the CRDM housing). Additionally, thermal transients caused by rod up and down motion, which dominate the thermal response of the CRDM, are eliminated. Since Control Rod H-08 is in a control bank, it is subject to periodic rod movements during operation at power. Therefore, the CRDM thermal stress analysis would remain valid after the removal of the H-08 control rod drive shaft and RCCA.

A bypass flow analysis was performed to determine the acceptability of not installing a flow restrictor in the un-rodded H-08 CRGT. Westinghouse determined that the existing large and small break LOCA analyses remain bounding. Furthermore, the non-LOCA UFSAR Chapter 15 accident analyses assume a bypass flow fraction that bounds the condition where the H-08 CRGT is unrestricted.

A thimble plug assembly will be installed on the fuel assembly in core location H-08. The fuel assembly flow characteristics with the thimble plug installed are hydraulically equivalent to the fuel assembly with an RCCA installed. As described in UFSAR Sections 4.2.3, the thimble plug assembly serves the following functions:

- Accommodate the differential thermal expansion between the fuel assembly and core internals.
- Maintain positive contact with the fuel assembly and the core internals.
- Limit bypass flow through unoccupied thimbles to acceptable design values.

The thermal-hydraulic reactor internal vessel evaluation is not impacted by removal of the control rod drive shaft and RCCA as long as a thimble plug is installed in the fuel assembly. The hydraulic equivalence, between the H-08 upper guide tube with and without a control rod drive shaft installed, ensures that there will be no impact on rod drop times at other core locations and that the current Technical Specification 3.1.4.3 rod drop time surveillance limits will continue to be met.

Seismic and structural impacts

The H-08 CRDM adaptor cap components are structurally adequate and meet the allowable ASME Code stress limits. The components are not required to conform to the ASME Boiler and Pressure Vessel Code requirement since they do not function as part of a core support structure. Material properties were taken from Section II, Part A of the Code. Structural analysis of the components demonstrated that all of the calculated stresses are within the ASME Code allowable limits.

The H-08 CRDM adaptor cap components are constructed of similar stainless steel material to that of the existing interfacing components (guide tube and CRDM adaptor housing). Thus, the interfacing components have similar thermal expansion properties and are compatible with RCS fluid process conditions. The material of construction is consistent with that described in the UFSAR . The installation process for the CRDM adaptor cap components includes requirements for proper pre-load/torque and has additional provisions to ensure sub-components are secure (e.g., tack-weld and crimp of cap-screw).

There is no impact on the functionality and structural integrity of the reactor vessel upper internals from removal of the control rod drive shaft and RCCA and use of a thimble plug in the fuel assembly in core location H-08. Therefore, there is no impact on the current reactor vessel internals analyses.

UFSAR Section 3.7.3.15 discusses the CRDM housing dynamic analysis (seismic and LOCA). Removal of the control rod drive shaft reduces the overall weight of the CRDM, whereby the CRDM dynamic stress evaluation would remain bounding with removal of the control rod H-08 drive shaft and RCCA.

Other Considerations

The change in RCS water volume is not appreciably impacted by removal of H-08 RCCA. Similarly, the change in the reactor vessel assembly is not appreciably impacted by removal of H-08 RCCA.

3.6 Adequate Level of Safety

The evaluations of the impact on the safety analyses have demonstrated that requirements for reactivity control provided by control rods continue to be met, even with removal of control rod H-08 during M2C26. Therefore, the assumption that control rod insertion will provide sufficient negative reactivity to shut down the reactor remains valid.

There will be a reduction in the available SDM as a result of removing control rod H-08; however, SDM will be maintained within the limits provided in the COLR and as required by TS 3.1.1. As shown in Table 1 (see Section 3.3), the required SDM is maintained, and additional margin is still present. Compliance with the TS provides reasonable assurance that the proposed change does not endanger the health and safety of the public.

3.7 Impact on Operator Actions

Per normal station process, a set of cycle-specific reactivity setpoints is generated during each refueling outage, normally during No Mode. These setpoints are fed into abnormal and emergency operating procedures. Removal of control rod H-08 does not affect this process requirement. The target numbers for boration change, but no new operator actions are expected.

Additionally, simulator runs were performed to gather data on plant performance without control rod H-08. These runs demonstrate that the plant will continue to operate within required limits and still maintain sufficient margin.

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

TS 4.2.2, "Control Rod Assemblies," describes a Design Feature required per

10 CFR 50.36(c)(4). The proposed change does not eliminate the design feature requiring control rod assemblies. Rather, it allows for a revised number of control rod assemblies. As outlined in the Technical Evaluation, all safety analysis limits are met, and the Unit 2 operating cycle M2C26 core has been evaluated with and without the H-08 control rod assembly per the methodologies set forth in TS 5.6.5, "Core Operating Limits Report (COLR)."

McGuire TS 3.1.4, "Rod Group Alignment Limits," requires all shutdown and control rods to be operable. Since the control rod in location H-08 would be removed under the proposed change, this TS requirement would not be applicable. As such, no changes to TS 3.1.4 are required.

The requirements of 10 CFR 50.62(c) applicable to McGuire continue to be met. Removal of Control Rod H-08 does not impact Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry, and changes to parameters described in the LAR do not impact the ATWS analysis. Therefore, the requirements of 10 CFR 50.62(c)(1) continue to be met. Subsection (c)(2) is not pertinent to a Westinghouse reactor such as McGuire, and subsections (c)(3) through (c)(5) are applicable only to boiling water reactors.

McGuire will maintain the ability to meet the applicable General Design Criteria (GDC) as outlined in 10 CFR 50, Appendix A. Pertinent criteria are discussed below.

Criterion 10 – Reactor Design

This criterion remains satisfied because removal of Control Rod H-08 does not impact the at-power core power distribution, SDM is maintained per the requirements of the COLR analysis, and the design and safety limits for all UFSAR Chapter 15 accidents remain satisfied.

Criterion 11 – Reactor Inherent Protection

This criterion remains satisfied because removal of Control Rod H-08 does not impact the ability to detect or control core power distribution, and the at-power nuclear reactivity feedback coefficients remain unchanged.

Criterion 12 – Suppression of Reactor Power Oscillations

As per the COLR analysis, the removal of this control rod will not result in power oscillations, which would result in conditions exceeding specified acceptable fuel design limits.

Criterion 23 – Protection System Failure Modes

The removal of Control Rod H-08 from the reactor vessel does not impact the fail-safe function of the remaining 52 control rods, which will still reliably maintain an adequate reactor protection system. The mechanical removal of the control rod drive shaft and the installation of the thimble plug in the fuel assembly do not have any mechanical impact on the function of the remaining 52 control rods. The electrical removal from service of Control Rod H-08 involves pulling fuses to remove control power to the respective stationary, lift, and movable coils. The remaining control rods are not impacted by this electrical change and will continue to meet their design function. The modification design change process ensures that the associated plant modifications involve only Control Rod H-08 and do not affect other control rods.

Thus, the requirements for Criterion 23 are met by maintaining the control rod insertion capability upon failure of the drive mechanisms or induced failure by an outside force.

Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions

A Unit 2 operating cycle M2C26 redesign reload analysis, performed according to methods referenced in TS 5.6.5, confirm that the fuel design limits are not exceeded. The reactor trip function remains fully capable of performing its function with 52 control rods, and fuel design limits are not exceeded for analyzed malfunctions of the reactivity control systems.

Criterion 26 – Reactivity Control System Redundance and Capability

Removal of Control Rod H-08 does not impact the ability of the reactivity control system to perform its function. Under normal operating conditions, including anticipated operational occurrences, acceptable fuel design limits are not exceeded. This includes appropriate margin for malfunctions, such as a single stuck rod. Rod control, reactor trip, and reactor coolant system boron addition functions will continue to perform their design and safety functions with removal of Control Rod H-08.

Criterion 27 – Combined Reactivity Control Systems Capability

This criterion is satisfied because the removal of Control Rod H-08 does not impact the ability of the reactivity control systems to reliably control reactivity changes and that adequate SDM is maintained when considering highest stuck rod worth. Evaluations of the removal of Control Rod H-08 during M2C26 demonstrate that SDM and safety analysis limits are met throughout the fuel cycle.

Criterion 28 – Reactivity Limits

This criterion is satisfied because removal of Control Rod H-08 has been evaluated to ensure trip reactivity insertion rate, SDM, and the safety analysis limits remain met for the UFSAR Chapter 15 accidents for the entire fuel cycle.

Criterion 29 – Protection against Anticipated Operational Occurrences

The removal of Control Rod H-08 does not impact the ability of the reactivity control systems to perform their safety functions. The mechanical removal of the control rod drive shaft and RCCA and the installation of the thimble plug in the fuel assembly do not have any mechanical impact on the function of the remaining 52 control rods. The electrical removal from service of Control Rod H-08 involves pulling fuses to remove control power to the respective stationary, lift, and movable coils. The remaining control rods are not impacted by this electrical change and will continue to meet their design function. The modification design change process ensures that the associated plant modifications involve only Control Rod H-08 and do not affect other control rods.

Thus, a high probability of control rod insertion continues to exist under anticipated operational occurrences, even with the removal of control rod H-08 during M2C26.

4.2 No Significant Hazards Consideration Determination

The proposed amendment to McGuire Unit 2 TS 4.4.2, "Control Rod Assemblies," will allow 52 control rod assemblies if the thermal sleeve at core location H-08 is not repaired during M2R25. This allowance would only be in effect during Unit 2 operating cycle M2C26. Currently, TS 4.2.2 requires 53 control rod assemblies in each reactor core.

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

Removal of control rod H-08 for M2C26 will be performed using approved plant processes and procedures. The change in the probability and consequence of accidents previously evaluated in the UFSAR has been evaluated and is shown to be non-significant. An evaluation of the impact on the safety analysis shows that the current safety analysis remains bounding.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

Removal of Control Rod H-08 for M2C26 does not create any new failure modes, and the design function and operation of SSCs are unchanged. No new operator actions are created. The modification to remove Control Rod H-08 ensures that Reactor Coolant System flowrate through the reactor vessel remains unchanged. Reactivity control and insertion characteristics continue to meet all design and safety functions, and plant equipment will continue to meet applicable design and safety requirements. Therefore, the proposed change does not create the possibility of a new or different kind of accident than those previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

Removal of control rod H-08 does not exceed or alter a UFSAR design basis or safety limit. The minimum SDM requirement is not changed, and analysis shows that additional margin above this limit still exists even with the control rod removed. Therefore, the proposed change does not significantly reduce a margin of safety.

Based on the above, it is concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Precedent

McGuire has identified the following precedent licensing action where operation with a removed control rod assembly was approved. Insights from this precedent licensing action have been incorporated into the proposed change as appropriate.

1. South Texas Project submitted a LAR on December 3, 2015 (Reference 2), to allow operation with one full-length control rod assembly removed. This LAR was approved by SE dated December 11, 2015 (Reference 3).

4.4 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5. ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6. REFERENCES

1. McGuire Nuclear Station Unit 2 TS 4.2.2, "Control Rod Assemblies," Amendment 267

2. South Texas Project submitted a LAR on December 3, 2015 (ADAMS Accession No. ML15343A347)

3. South Texas Project LAR was approved by SE dated December 11, 2015 (ADAMS Accession No. ML15343A128)

ATTACHMENT 1

McGuire Technical Specification Page Markups

4.0 DESIGN FEATURES

4.1 Site Location

The McGuire Nuclear Station site is located at latitude 35 degrees, 25 minutes, 59 seconds north and longitude 80 degrees, 56 minutes, 55 seconds west. The Universal Transverse Mercator Grid Coordinates are E 504, 669, 256, and N 3, 920, 870, 471. The site is in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either ZIRLO™ or Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ZIRLO™, zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies.* The control material shall be silver indium cadmium (Unit 1) silver indium cadmium and boron carbide (Unit 2) as approved by the NRC.

* Unit 2 is permitted to operate with 52 control rod assemblies (with no control rod assembly installed in core location H-08) during M2C26.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.00 weight percent;

b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. A nominal 10.4 inch center to center distance between fuel assemblies placed in Region 1 and

e. A nominal 9.125 inch center to center distance between fuel assemblies placed in Region 2.

f. Neutron absorber (Boral) installed between fuel assemblies in the Region 1 racks.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.00 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

d. A nominal 21 inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft.-7 in.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1463 fuel assemblies (286 total spaces in Region 1 and 1177 total spaces in Region 2).

ATTACHMENT 2

McGuire Technical Specification – Clean Version

4.0 DESIGN FEATURES

4.1 Site Location

The McGuire Nuclear Station site is located at latitude 35 degrees, 25 minutes, 59 seconds north and longitude 80 degrees, 56 minutes, 55 seconds west. The Universal Transverse Mercator Grid Coordinates are E 504, 669, 256, and N 3, 920, 870, 471. The site is in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina.

4.2 Reactor Core

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The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either ZIRLO™ or Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ZIRLO™, zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies.* The control material shall be silver indium cadmium (Unit 1) silver indium cadmium and boron carbide (Unit 2) as approved by the NRC.

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4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.00 weight percent;

b. k_{eff} < 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

c. $k_{eff} \leq 0.95$ if fully flooded with water borated to 800 ppm, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

d. A nominal 10.4 inch center to center distance between fuel assemblies placed in Region 1 and

e. A nominal 9.125 inch center to center distance between fuel assemblies placed in Region 2.

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4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.00 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR; and

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The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 745 ft.-7 in.

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