DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

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1.0 SUMMARY DESCRIPTION

Ameren Missouri (Union Electric Company) is proposing to amend Operating License NPF-30 for Callaway Plant ("Callaway"). The proposed amendment would add a note to Technical Specification (TS) 4.2.2, “Control Rod Assemblies,” to permit the Cycle 28 core to contain 52 control rods (i.e., with no control rod in core location H-08) in lieu of the current requirement for 53 control rods. A Callaway operating cycle is nominally 18 months.

The results of control rod drop time testing performed during recent refueling outages indicate a slowing rod drop time for the H-08 control rod. The drop time is still within the TS allowable limits (per TS Surveillance Requirement 3.1.4.3), but troubleshooting is planned for the upcoming outage, Refuel 27. The troubleshooting activity could indicate the need for a repair that would not be able to be performed during the outage. If so, the H-08 control rod would be removed with the intent of leaving it removed from the H-08 core location throughout Cycle 28 such that the H-08 control rod drive mechanism would be repaired or replaced during Refuel 28. Accordingly, Ameren Missouri requests approval of this contingency license amendment request (LAR) to allow removal of the rod cluster control assembly (RCCA) associated with the H-08 position on a one-time basis (i.e., throughout Cycle 28) if removal is determined to be necessary.

For the purposes of this submittal, the terms “control rod” and “rod cluster control assembly” (RCCA) are used synonymously. Consistent with that, the RCCA associated with the H-08 core location may be referred to as the “H-08 control rod.”

2.0 DETAILED DESCRIPTION

2.1 Proposed Changes

The proposed amendment would revise TS 4.2.2 to add a note permitting operation with 52 control rods during Cycle 28, in lieu of the requirement for 53 control rods. Ameren has reviewed the Callaway TSs and has determined that no additional TS changes are required.

The current TS 4.2.2 states:

> The reactor core shall contain 53 control rod assemblies. The control rod material shall be silver indium cadmium, hafnium metal, or a mixture of both types, as approved by the NRC.

The proposed TS note (to be attached to TS 4.2.2) is as follows:

> Operation with 52 control rod assemblies (i.e., with no control rod assembly installed in core location H-08) is permitted during Cycle 28.

The impact and acceptability of operating the plant without the H-08 control rod assembly in the core during Cycle 28 is explained and evaluated in Section 3.0 of this Enclosure.

Attachment 1 provides a marked-up version of the affected Callaway TS page containing TS 4.2.2 in order to show the proposed change. Attachment 2 provides a clean version of the TS page.

3.0 TECHNICAL EVALUATION

3.1 System Description

The Callaway reactor normally contains 53 control rod assemblies divided into four control banks (Control Banks A, B, C, D) and five shutdown banks (Shutdown Banks SA, SB, SC, SD, and SE). Of the nine banks, Control Bank D is used for reactivity 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 shutdown margin (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).

The H-08 control rod is part of Control Bank D and is located in the center of the core as shown in Figure 1. With the removal of the H-08 control rod, the core during Cycle 28 will contain 52 control rod assemblies. Table 1 shows the number of rods in each bank.

Each control rod is moved by a control rod drive mechanism (CRDM) consisting of a stationary gripper, movable gripper, and a lift pole. Three coils are installed external to the CRDMs to electromechanically manipulate the CRDM components to produce rod motion. The CRDMs are magnetic jacking type mechanisms that move the control rods within the reactor core by sequencing power to the three coils of each mechanism to produce a stepping rod motion. Rod position is achieved through a timed sequence of stationary, movable, and lift coil current. At each point in time during rod positioning, the control rod is being held by either the stationary gripper or movable grippers. Tripping can occur during any part of the power cycler sequencing if electrical power to the coils is interrupted. The primary function of the CRDM is to insert or withdraw rod cluster control assemblies (RCCAs) within the core to control average core temperature and to shut down the reactor. Mechanically, each control rod location includes a guide tube, which is an assembly that sheathes and guides the control rod drive shafts and control rods.

Figure 1 – Control Rod Locations

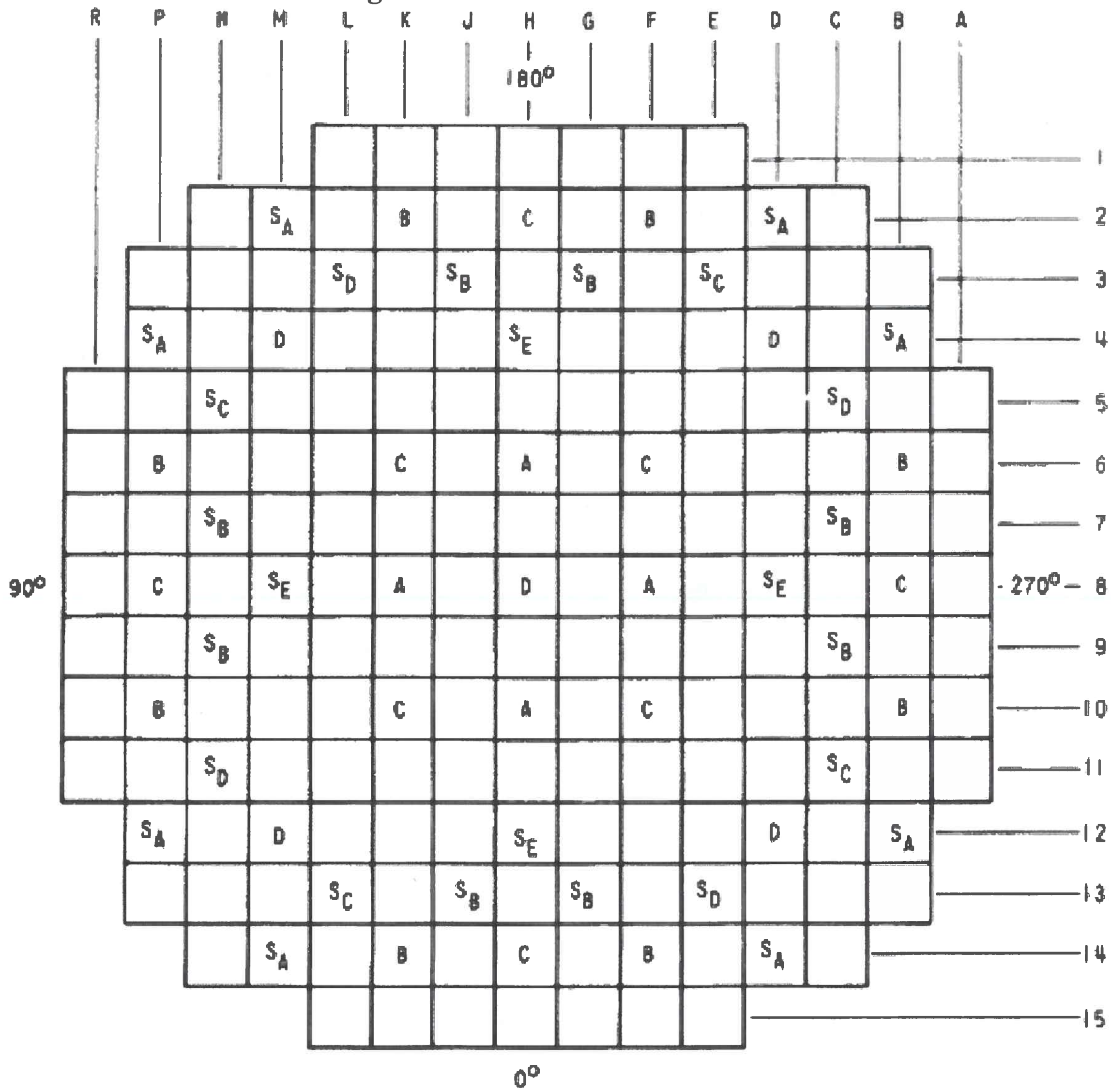

Table 1 – Control and Shutdown Bank Rods

CONTROL BANK	NUMBER OF RODS
A	4
B	8
C	8
D	5 (4)[1]
TOTAL	25 (24)[1]

SHUTDOWN BANK	NUMBER OF RODS
SA	8
SB	8
SC	4
SD	4
SE	4
TOTAL	28

[1] Numbers in parentheses indicate the number of rods without the H-08 control rod in the core

3.2 Current Licensing Basis

The reload safety analysis applies NRC-approved codes and analytical methods to design the reload core. The NRC-approved codes and analytical methods used to generate the reload safety evaluation are identified in TS 5.6.5, "Core Operating Limits Report," and are also listed in the cycle-specific COLR.

The reload safety analysis methods are not invalidated by the removal of the H-08 control rod from the Cycle 28 core design because these methods are not dependent on a particular RCCA configuration. Reload safety analysis methods and supporting computer codes remain applicable to model and evaluate the 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 are performed each cycle to ensure core protective and operating limits remain satisfied and safety analysis limits remain bounded.

As described in Final Safety Analysis Report (FSAR) Section 4.2.2.3.1, "Rod Cluster Control Assembly":

The rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes associated with variations in operating conditions of the reactor, i.e., power and temperature variations. Two nuclear design 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 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 banks provide adequate shutdown margin.

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

The number of rod cluster assemblies is shown in Table 4.3-1A. The rod cluster control assemblies are used for shutdown and control purposes to offset fast reactivity changes associated with:

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

The allowed control bank reactivity insertion is limited at full power to maintain shutdown capability. As the power level is reduced, 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. For further discussion, refer to the COLR on rod insertion limits.

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 shown in Figure 4.3-36. All shutdown rod cluster control assemblies are withdrawn before withdrawal of the control banks is initiated. In going from zero to 100-percent power, control banks A, B, C, and D are withdrawn sequentially. The limits of rod positions and further discussion on the basis for rod insertion limits are provided in the COLR.

3.3 Impact on Safety Analysis

The removal of the H-08 control rod from Control Bank D is considered to apply for the entirety of Cycle 28 operation and impacts the nuclear design and safety analysis characteristics for this reload core design. As such, the reload safety evaluation 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 FSAR Chapter 15, for the H-08 control rod removed. The nuclear design parameter changes associated with core operation with the H-08 control rod removed were evaluated against a set of bounding values contained in the pertinent accident and transient analyses for the plant. The results of those evaluations are discussed in this section.

NRC-approved reload safety analysis codes and methods were used to determine if the change in core design parameters remain bounded by the key safety parameters assumed in the FSAR Chapter 15 safety analysis. Additionally, impacts on margins to fuel thermal and power peaking limits related to departure from nucleate boiling (DNB) and centerline fuel melt (CFM) safety criteria due to the change in power distribution attributable to operation with the H-08 control rod removed were evaluated.

An evaluation of impacts to core and fuel performance, as well as the impact to the safety analyses described in FSAR Chapter 15 and safety analysis parameters, is summarized in the cycle-specific reload safety evaluation documentation to confirm the acceptability of reactor operation with the new core configuration. There were no changes in analytical methods or safety analysis limits used to perform the core reload safety evaluation for Cycle 28 with the H-08 control rod removed. The Cycle 28 core design (with the H-08 control rod removed) was performed with full core models. Results of the safety analysis impact evaluation are described in the following subsections.

It should be noted that the TS Limiting Conditions for Operation (LCOs) and associated Surveillance Requirements (SRs) are not impacted by the proposed change. As described in Section 2 above, a temporary note is proposed to be added to TS 4.2.2 allowing for operation with the RCCA in location H-08 removed for Cycle 28.

3.3.1 Shutdown Margin

The proposed change impacts the available shutdown margin (SDM). TS LCO 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 FSAR remains bounding. An evaluation of the impact on the reduction of SDM due to the removal of the H-08 control rod has been performed, and the results are presented in Table 2. The SDM is reduced from 2.289 $\%\Delta\rho$ to 2.154 $\%\Delta\rho$ at the beginning-of-cycle (BOC) and from 1.861 $\%\Delta\rho$ to 1.479 $\%\Delta\rho$ at the end-of-cycle (EOC). These reduced values remain bounding relative to the 1.3 %Δρ limit specified in the COLR. By maintaining this limit, the safety analysis described in Chapter 15 of the FSAR remains bounding with regards to SDM for accidents initiated in MODES 1 and 2.

Table 2 – Comparison of Effect on Shutdown Margin

	With H-08		Without H-08	
	BOC	EOC	BOC	EOC
Control Rod Worth (%Δρ)				
Available Rod Worth Less Worst Stuck Rod	4.860	6.148	4.534	5.560
[A] Less 10%	4.374	5.533	4.080	5.004
Control Rod Requirements (%Δρ)				
Reactivity Defects	2.035	3.622	1.876	3.475
Void Allowance	0.050	0.050	0.050	0.050
[B] Total Requirements	2.085	3.672	1.926	3.525
[C] Available Shutdown Margin [A-B]	2.289	1.861	2.154	1.479
[D] Required Shutdown Margin	1.300	1.300	1.300	1.300
Excess Shutdown Margin [C-D]	0.989	0.561	0.854	0.179

The COLR provides the required SDM limits for MODES 3, 4, and 5, and MODE 2 with $K_{eff} < 1.0$. Per the Cycle 28 COLR, SDM must be greater than or equal to 1.3% in MODES 3 and 4, and it must be greater than or equal to 1.0% in MODE 5.

Operationally, the required RCS SDM boron concentrations for MODES 3, 4, and 5 will be higher with the H-08 control rod removed in order to meet the COLR SDM limits. Table 3 below provides the minimum required shutdown boron concentration with all rods in (ARI) minus the most reactive stuck rod for 1.3% SDM and 1.0% SDM for beginning-of-cycle, middle-of-cycle (MOC), and end-of-cycle conditions.

Table 3 – Minimum Required Shutdown Boron Concentration with ARI minus the Most Reactive Stuck Rod (1.3% SDM and 1.0% SDM)

	Required Boron with H-08 (ppm)		Required Boron without H-08 (ppm)	
1.3% SDM	350°F	557°F	350°F	557°F
BOC	1215	1072	1245	1116
MOC	1038	860	1083	923
EOC	399	106	448	171
1.0% SDM	68°F	200°F	68°F	200°F
BOC	1230	1220	1253	1244
MOC	1063	1050	1094	1088
EOC	504	462	524	500

3.3.2 Boron Concentration and Boron Worth

The removal of the H-08 control rod was evaluated for impact on required boron concentration and differential boron worth as a function of boron concentration in a rodded configuration. The removal of the H-08 control rod increases the required boron concentration and reduces boron worth, as a function of boron concentration, when RCCAs are inserted into the core. This results in a change in boron concentration requirements in the RCS for MODES 1, 2, 3, 4, and 5, which impacts the boron dilution accident described in FSAR section 15.4.6.

Although the boron concentration requirements are revised, the analysis described in FSAR section 15.4.6 remains bounding for MODES 1 through 5 for the removal of the H-08 control rod. No other changes are made regarding this analysis. Therefore, the removal of the H-08 control rod does not impact the results presented in the FSAR section 15.4.6.

3.3.3 Trip Reactivity

The removal of the H-08 control rod 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 the normalized trip reactivity as a function of time after the RCCAs begin to fall is presented in the FSAR. An evaluation of the effects of the removal of the H-08 control rod 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 used in the safety analyses remains bounding. Table 4 provides a comparison of the trip reactivity as a function of rod position for Cycle 28 with and without the H-08 control rod inserted. The trip reactivity is maintained above the limit at all rod positions analyzed. Also, Table 5 provides the minimum trip worth for Cycle 28 with and without the H-08 control rod inserted. In both cases, all limits are maintained. Therefore, the removal of the H-08 control rod does not impact the trip reactivity assumed in FSAR Chapter 15 events.

Table 4 – Trip Reactivity vs. Position

Rod Position (Fraction of Insertion)	Bounding Burnup	Trip Reactivity with H-08 (%ΔK)	Trip Reactivity without H-08 (%ΔK)	Limit (%ΔK)
0.00	BOC	0.000	0.000	≥ 0.000
0.03	BOC	0.023	0.023	> 0.001
0.06	BOC	0.045	0.045	> 0.005
0.10	BOC	0.063	0.063	> 0.012
0.20	EOC	0.094	0.095	> 0.041
0.30	EOC	0.119	0.119	> 0.087
0.40	EOC	0.159	0.156	> 0.117
0.60	EOC	0.374	0.374	> 0.292
0.80	EOC	1.352	1.341	> 1.144
0.90	BOC	2.944	2.859	> 2.750
0.95	BOC	4.033	3.838	> 3.720
1.00	BOC	4.553	4.300	> 4.000

Table 5 – Trip Reactivity vs. Power

Limit (%ΔK)		Minimum Trip Reactivity with H-08 (%ΔK)	Minimum Trip Reactivity without H-08 (%ΔK)
100% RTP	> 4.000	4.556	4.303
90% RTP	> 4.000	4.422	4.213
50% RTP	> 2.800	3.397	3.266
0% RTP	> 1.300	2.466	2.365

3.3.4 Moderator Temperature Coefficient (MTC)

FSAR 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. The consequences of accidents that cause core overcooling must be evaluated when the MTC is negative.

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. The most conservative combination appropriate to the accident is then used for the analysis.

The removal of the H-08 control rod slightly impacts the moderator temperature coefficient calculated at the conservative bounding conditions determined for the FSAR accident analyses. Moderator temperature coefficient results for Cycle 28 with and without the H-08 control rod are shown in Table 6 and confirm that the limit assumed in the safety analysis remains bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in the FSAR sections 15.1.2, 15.1.5, 15.2.2, 15.3.2, 15.4.1, and 15.4.2.

Table 6 – Moderator Temperature Limit Summary for Cycle 28 with and without H-08 Control Rod

Limit Description	Limit	Reload Values with H-08	Reload Values without H-08
Most positive HFP MTC	< 0 pcm/F	-3.784	-3.784
HFP error-adjusted rod insertion limit (Bank D at 149 steps withdrawn) EOC	> -40.4 pcm/F	-34.684	-34.684
Near-EOC MTC at 300 ppmB	> -40.4 pcm/F	-28.482	-28.479
Near-EOC MTC at 60 ppmB	> -45.5 pcm/F	-33.259	-33.259
Most positive HZP MTC	< 5 pcm/F	3.936	3.936

3.3.5 FSAR Chapter 15 Accident Analyses Impacts from Removal of the H-08 Control Rod

Removal of the H-08 control rod for Cycle 28 has an impact for most comparisons to FSAR Chapter 15 accident analysis parameters routinely evaluated as part of the reload design process. Cycle-specific evaluations were performed to determine if the change in core design adversely impacts bounding key safety parameters assumed in the FSAR Chapter 15 safety analysis and impacts on DNB and fuel thermal limits due to the change in power distribution. The bounding key safety parameters are developed in FSAR 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 the H-08 control rod during Cycle 28 does not impact the results presented in the FSAR Chapter 15 accident analyses. Results and discussion of the FSAR Chapter 15 accident analyses for Cycle 28 with the H-08 control rod removed are provided below.

3.3.5.1 Hot Zero Power (HZP) Steam Line Break (SLB) Accident

For the 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 that 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.

The removal of the H-08 control rod 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. The Westinghouse reload methodology determines if the linear power generation in the core remains bounding for the reload core. Cycle-specific linear power generation parameters are presented in Table 7 which shows that the limit assumed in the safety analysis remains bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.1.5.

Table 7 – HZP SLB Linear Power Generation Results

Limit (kW/ft)	Reload Values with H-08	Reload Values without H-08
< 22.46	18.23	14.60

3.3.5.2 Hot Full Power (HFP) Steam Line Break

For the HFP SLB, the same consequences associated with the HZP SLB can be observed, resulting in an increase in core power potentially leading to CFM and DNB.

The removal of the H-08 control rod impacts the consequences of this event in the same manner it impacts those of the HZP SLB. In accordance with the Westinghouse reload methodology, WCAP-9272, it is determined whether linear power generation in the core remains bounding for the reload core. Cycle-specific linear power generation parameters are presented in Table 8 which shows that the limit assumed in the safety analysis remains bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.1.5.

Table 8 – HFP SLB Linear Power Generation Results

Limit (kW/ft)	Reload Values with H-08	Reload Values without H-08
< 22.46	19.82	19.82

3.3.5.3 Locked Rotor Accident (LRA)

The LRA postulated is an instantaneous seizure of a reactor coolant pump rotor. The LRA is analyzed assuming offsite power lost conditions. Analysis is performed to determine the $F\Delta H$ parameter (enthalpy rise hot channel factor) for the reload core. The removal of the H-08 control rod could impact the localized reactor core power distribution for the LRA event, potentially resulting in DNB.

The results of cycle-specific LRA evaluations are presented in Table 9 and confirm that positive $F\Delta H$ margin exists. The limiting case was not impacted by the removal of the H-08 control rod. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.3.3.

Table 9 – LRA $F\Delta H$ Parameter Results

Limit	Reload Values with H-08	Reload Values without H-08
< 1.59	1.47	1.47

3.3.5.4 Rod Withdrawal from Subcritical (RWFS)

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 in the detailed plant analysis is calculated to compare to limits assumed in Chapter 15 accident analyses. The removal of the H-08 control rod will impact the localized reactor core power distribution for events where a power excursion occurs.

The results of cycle-specific evaluations for the RWFS accident are presented in Table 10. A comparison of the cycle-specific evaluation results and the cycle-specific limit confirms that positive margin exists, and more margin exists without the H-08 control rod. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.4.1.

Table 10 – RWFS Reactivity Insertion Rate Results

Limit (pcm/in)	Reload Values with H-08	Reload Values without H-08
< 113.3	46.9	43.1

3.3.5.5 Rod Withdrawal at Power (RWAP)

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 (RPS) is designed to terminate any such transient before DNB occurs.

The maximum positive reactivity insertion rate at power is less than that observed at subcritical conditions. The removal of the H-08 control rod will impact the localized reactor core power distribution for events where a rod power maneuver occurs.

The results of cycle-specific evaluations for RWAP accident are presented in Table 11 and confirm that positive margin exists, and more margin exists without the H-08 control rod. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.4.2.

Table 11 – RWAP Reactivity Insertion Rate Results

Limit (pcm/in)	Reload Values with H-08	Reload Values without H-08
< 146.7	21.7	21.2

3.3.5.6 Single Rod Withdrawal (SRW)

Withdrawal of a single RCCA 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. The event is analyzed for the highest worth Control Bank D rod withdrawn from the insertion limit with the reactor initially at full power. The removal of the H-08 control rod will impact the localized reactor core power distribution for events where a single Control Bank D rod withdrawal occurs.

The results of cycle-specific evaluations for single RCCA withdrawal accident are presented in Table 12 and confirm that positive margin exists. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.4.3.

Table 12 – SRW FΔH Parameter Results

Limit	Reload Values with H-08	Reload Values without H-08
< 1.65	1.45	1.46

3.3.5.7 Misaligned Rod (MAR)

Misaligned control rod events result in asymmetric radial peaking that could result in DNB. The removal of the H-08 control rod will impact the localized reactor core power distribution for events where a single Control Bank D rod misalignment occurs.

The results of cycle-specific parameter evaluations for the statically misaligned RCCA accident are presented in Table 13 and confirm that positive margin exists. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.4.3.

Table 13 – MAR FΔH Parameter Results

Limit	Reload Values with H-08	Reload Values without H-08
< 1.98	1.63	1.63

3.3.5.8 Dropped Rod Accident (DRA)

Following a dropped rod event in manual rod control, the plant will establish a new equilibrium condition. 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.

The DRA is evaluated for all the dropped rod combinations of control and shutdown bank, and a peaking evaluation is performed to compare to applicable peaking limits to ensure DNB would not occur for DRA. The removal of the H-08 control rod will impact the localized reactor core power distribution for DRA.

The results of cycle-specific parameter evaluations for DRA are presented in Table 14 and confirm that positive margin exists. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.4.3.

Table 14 – DRA Pre-Drop FΔH Parameter Results

Limit	Reload Values with H-08	Reload Values without H-08
> 1.53	1.57	1.61

3.3.5.9 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.

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. Xenon effects are considered in the analysis.

The REA is evaluated for the plant and control bank conditions described above and bounding REA safety analysis limits, and a peaking evaluation is performed. The removal of the H-08 control rod will impact the localized reactor core power distribution for REA. An REA accident cannot occur in the H-08 location with the H-08 control rod removed.

The results of cycle-specific parameter evaluations for REA are presented in Table 15 and confirm that the limits assumed in the safety analysis remains bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in FSAR section 15.4.8.

Table 15 – REA Parameter Results

Limit Description	Limit	Reload Values with H-08	Reload Values without H-08
HZP BOC Max Ejected Rod Worth	< 0.77 %Δρ	0.31	0.26
HZP BOC Max FQ	< 11.0	4.8	4.2
HZP EOC Max Ejected Rod Worth	< 0.90 %Δρ	0.50	0.43
HZP EOC Max FQ	< 20.0	16.6	15.0
HFP BOC Max Ejected Rod Worth	< 0.20 %Δρ	0.04	0.04
HFP BOC Max FQ	< 6.30	2.02	2.04
HFP EOC Max Ejected Rod Worth	< 0.25 %Δρ	0.06	0.06
HFP EOC Max FQ	< 6.40	3.29	3.26

3.3.6 Miscellaneous Safety Analysis Neutronic Parameters

Miscellaneous Safety Analysis neutronic parameters such as delayed neutron data (beta and prompt neutron lifetime), Doppler temperature coefficients, and fuel temperatures are not significantly impacted by the change in core configuration. These parameters are driven more directly by the core design. Cycle-specific parameter evaluations of these safety analysis values show negligible changes and confirm that the values assumed in the safety analysis remain bounding.

3.3.7 Safety Analysis Evaluation Summary

To summarize, the impact of the removal of the H-08 control rod in Cycle 28 on the nuclear design and safety analysis, including the FSAR Chapter 15 events accident analyses, has been evaluated using the NRC-approved methods described in TS 5.6.5. These NRC-approved reload safety evaluation methods were used to determine if the proposed change in core configuration adversely impacts the bounding key safety parameters assumed in the FSAR Chapter 15 Safety Analysis and impacts on DNB and CFM due to the change in power distribution attributable to the new core design with the H-08 control rod removed. Cycle-specific parameter evaluations for FSAR Chapter 15 Safety Analysis parameters confirm that the values assumed in the safety analysis remain bounding for all FSAR Chapter 15 Safety Analysis accidents, and consequently, there is no change to the accident dose analysis.

Therefore, removal of the H-08 control rod for Cycle 28 does not impact the results presented in FSAR Chapter 15. Table 16 presents a summary of the impact of removal of the H-08 control rod on each Chapter 15 Safety Analysis accident.

Table 16 – Impact on FSAR Chapter 15 Accident Analyses

#	FSAR	Description	Comments
1	15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	Cycle-specific reload evaluations verify the AOR remains bounding.
2	15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	Cycle-specific reload evaluations verify the AOR remains bounding.
3	15.1.3	Excessive Increase in Secondary Steam Flow	Cycle-specific reload evaluations verify the AOR remains bounding.
4	15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Cycle-specific reload evaluations verify the AOR remains bounding.
5	15.1.5	Steam System Piping Failure	Cycle-specific reload evaluations verify the AOR remains bounding.
6	15.2.1	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	No impact. There are no steam pressure regulators whose failure or malfunction could cause a steam flow transient.
7	15.2.2	Loss of External Electrical Load	Cycle-specific reload evaluations verify the AOR remains bounding.
8	15.2.3	Turbine Trip	Cycle-specific reload evaluations verify the AOR remains bounding.
9	15.2.4	Inadvertent Closure of Main Steam Isolation Valves	Cycle-specific reload evaluations verify the AOR remains bounding.
10	15.2.5	Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	Cycle-specific reload evaluations verify the AOR remains bounding.
11	15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	Cycle-specific reload evaluations verify the AOR remains bounding.
12	15.2.7	Loss of Normal Feedwater Flow	Cycle-specific reload evaluations verify the AOR remains bounding.
13	15.2.8	Feedwater System Pipe Break	Cycle-specific reload evaluations verify the AOR remains bounding.
14	15.3.1	Partial Loss of Forced Reactor Coolant Flow	Cycle-specific reload evaluations verify the AOR remains bounding.
15	15.3.2	Complete Loss of Forced Reactor Coolant Flow	Cycle-specific reload evaluations verify the AOR remains bounding.
16	15.3.3	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Cycle-specific reload evaluations verify the AOR remains bounding.
17	15.3.4	Reactor Coolant Pump Shaft Break	Cycle-specific reload evaluations verify the AOR remains bounding.
18	15.4.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	Cycle-specific reload evaluations verify the AOR remains bounding.

19	15.4.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Cycle-specific reload evaluations verify the AOR remains bounding.
20	15.4.3	Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)	Cycle-specific reload evaluations verify the AOR remains bounding.
21	15.4.4	Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature	No impact. The Callaway Technical Specifications do not permit operations in Modes 1 and 2 with less than four reactor coolant loops operating. This transient, initiated from Mode 3 conditions, is not limiting with respect to DNB. As such, an explicit analysis of this event is deemed not necessary (as noted in the FSAR).
22	15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	No impact. This section is not applicable to the Callaway Plant.
23	15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Cycle-specific reload evaluations verify the AOR remains bounding.
24	15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in Improper Position	No impact. These analyses do not consider reactor trip or control rod insertion and, as such, are not impacted by the removal of the H-08 control rod.
25	15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Cycle-specific reload evaluations verify the AOR remains bounding.
26	15.5.1	Inadvertent Operation of the Emergency Core Cooling System During Power Operation	Cycle-specific reload evaluations verify the AOR remains bounding.
27	15.5.2	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Cycle-specific reload evaluations verify the AOR remains bounding.
28	15.5.3	A Number of BWR Transients	No impact. This section is not applicable to the Callaway Plant.
29	15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Cycle-specific reload evaluations verify the AOR remains bounding.
30	15.6.2	Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary that Penetrate Containment	No impact. The analysis is limited to rupture of the chemical and volume control system (CVCS) letdown line at a point outside of containment. As such, this analysis is not impacted by the removal of the H-08 control rod.

31	15.6.3.1	Steam Generator Tube Rupture with Postulated Stuck-Open Atmospheric Steam Dump Value	No impact. Although reactor trip is credited to achieve subcriticality, the removal of the H-08 control rod does not impact the assumed trip reactivity. As such, this analysis is not impacted by the removal of the H-08 control rod.
32	15.6.3.2	Steam Generator Tube Rupture with Failure of Faulted Steam Generator AFW Control Valve	No impact. Although reactor trip is credited to achieve subcriticality, the removal of the H-08 control rod does not impact the assumed trip reactivity. As such, this analysis is not impacted by the removal of the H-08 control rod.
33	15.6.4	Spectrum of BWR Steam System Piping Failures Outside of Containment	No impact. This section is not applicable to the Callaway Plant.
34	15.6.5	Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Cycle-specific reload evaluations verify the AOR remains bounding.
35	15.6.6	A Number of BWR Transients	No impact. This section is not applicable to the Callaway Plant.
36	15.7.1	Radioactive Waste Gas Decay Tank Failure	No impact. The postulated accident is unaffected by the removal of the H-08 control rod.
37	15.7.2	Radioactive Liquid Waste System Leak or Failure	No impact. The postulated accident is unaffected by the removal of the H-08 control rod.
38	15.7.3	Postulated Radioactive Release Due to Liquid Tank Failures	No impact. The postulated accident is unaffected by the removal of the H-08 control rod.
39	15.7.4	Fuel Handling Accidents	No impact. The postulated accident is unaffected by the removal of the H-08 control rod.
40	15.8	Anticipated Transients without Scram	Cycle-specific reload evaluations verify the AOR remains bounding.

3.3.8 Impact on Operating Analysis Support

The reload safety evaluation methodology and computer code package (PARAGON/ANC9) currently used are applicable to model and evaluate the as-designed/operated configuration of the plant (reactor). Cycle-specific reload evaluations of TS limits and core operating limits without the H-08 control rod for Cycle 28 are performed to ensure applicable safety analysis limits remain satisfied. The PARAGON/ANC9 models that calculate reactivity parameter and power distribution performance are not impacted nor invalidated due to removal of the H-08 control rod from Cycle 28 core, and the methodology is not dependent upon control bank configuration. The NRC-approved methods used to determine COLR limits are not constrained by the removal of the H-08 control rod since explicit modeling of the core is employed in the verification of margin to thermal and peaking limits and in development of power peaking related core monitoring factors.

There will not be any differences in the COLR limits for TS 3.2.1 and 3.2.2 for core monitoring due to removal of the H-08 control rod from Cycle 28 core. Explicit modeling of the new core configuration is used in the generation of the cycle-specific peaking factor limits.

3.3.9 Conclusion

The reload safety evaluations for Cycle 28 with the H-08 control rod removed validated all cycle-specific safety analysis limits and determined the FSAR Chapter 15 accident analyses remain bounding with respect to the Cycle 28 safety analysis physics parameters and core thermal-hydraulic parameters with the H-08 control rod removed.

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

If required, the H-08 control rod would be removed from service by performing the following work items, which would be evaluated in accordance with appropriate Callaway 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
- 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 would 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 would continue to function for all other control rods.

3.5 Evaluation of Potential Design Impacts

3.5.1 Thermal-Hydraulic Impacts

When the H-08 control rod and driveshaft are removed from service, no flow restrictor will be installed in the H-08 control rod guide tube in the reactor vessel upper internals. Bypass flow analysis was performed to confirm that there is a negligible increase in core bypass flow and that the design core bypass limits are not exceeded. The removal of the H-08 drive rod results in a hole which has a cross-sectional area of 2.4 in^2. This results in a 2.57% increase in the upper support plate flow area parameter, which corresponds to a 0.239% increase in the flow in the upper head region. Including the increased bypass flow, considering the removal of core location H-08 from service for one cycle, results in a negligible increase in the total core bypass flow, and the design core bypass limits are not exceeded. It is estimated that the total core bypass flow will increase by approximately 0.08% at maximum.

3.5.2 Seismic and Structural Impacts

There is no impact on the functionality or structural integrity of the reactor vessel upper internals with the removal of the control rod drive shaft and RCCA at core location H-08. There is no impact on the current reactor vessel internals analyses.

FSAR Section 3.7(N).3.14 discusses the CRDM housing dynamic analysis (seismic and LOCA). Removal of the control rod drive shaft has negligible impact on the structural integrity of the CRDM housing dynamic analysis, so the current analysis would remain bounding with removal of the H-08 control rod.

3.5.3 Other Considerations

The changes in RCS water volume and metal mass are not appreciably impacted by removal of the H-08 RCCA and driveshaft.

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 the H-08 control rod during Cycle 28. 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 the H-08 control rod. However, SDM will be maintained within the limits provided in the COLR and as required by TS 3.1.1. As shown in Table 2 (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

The safety evaluations performed for the Cycle 28 H-08 RCCA removal validated that the impacts to the nuclear design parameters are within the bounds of those already assumed in the FSAR Chapter 15 accident analyses. No new or revised operator actions are required to meet the safety analyses' acceptance criteria. As a result, there are no changes required to the emergency operating procedures or the operator actions assumed for these accidents.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements / Criteria

The regulatory requirements and/or guidance documents associated with this amendment application include the following:

- GDC Criterion 10 – Reactor Design: *The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.*

  The reactor core and associated coolant, control, and protection systems are designed to the following criteria:

  - No fuel damage will occur during normal core operation and operational transients (Condition I) or any transient conditions arising from occurrences of moderate frequency (Condition II) beyond the small fraction of clad defects (1 percent) for which the plant shielding, cleanup, and radwaste systems are designed. Fuel damage, as used here, is defined as penetration of the fission product barrier (i.e., the fuel rod clad). Conditions I and II, as used here, are defined by ANSI N18.2-1973. The small number of clad defects that may occur are within the capability of the plant cleanup system and are consistent with the plant design bases.
  - The reactor can be returned to a safe shutdown state following a Condition III event with only a small fraction of the fuel rods damaged, although sufficient fuel damage might occur to preclude the immediate resumption of operation. Condition III, as used here, is defined by ANSI N18.2-1973.
  - The core will remain intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV). Condition IV, as used here, is defined by ANSI N18.2-1973.

  The reactor trip system is designed to actuate a reactor trip whenever necessary to ensure that the fuel design limits are not exceeded. The core design, together with the process and decay heat removal systems, provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater, and loss of both normal and preferred power sources.

  A Cycle 28 redesign reload analysis was performed in accordance with the methods described in TS 5.6.5 and confirmed that the fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences, with the H-08 control rod removed.

- GDC Criterion 11 – Reactor Inherent Protection: *The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.*

  Whenever the reactor is critical, prompt compensatory reactivity feedback effects are assured by the negative fuel temperature effect (Doppler effect) and by the operational limit on the moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design, using low-enrichment fuel. The moderator temperature coefficient of reactivity is dependent upon core characteristics, such as fuel loading, the dissolved absorber (boron) concentration, and burnable poisons.

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

- GDC Criterion 12 – Suppression of Reactor Power Oscillations: *The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.*

  Power oscillations of the fundamental mode are inherently eliminated by negative Doppler and nonpositive moderator temperature coefficients of reactivity.

  Oscillations, due to xenon spatial effects, in the radial, diametral, and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and nonpositive moderator temperature coefficients of reactivity.

  Oscillations, due to xenon spatial effects, may occur in the axial first overtone mode. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions, using the measured axial power imbalance as an input.

  If necessary to maintain axial imbalance within the limits of the Callaway Technical Specifications, i.e., imbalances which are alarmed to the operator and are within the imbalance trip setpoints, the operator can suppress xenon axial oscillations by control rod motions and/or temporary power reductions.

  Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

  This criterion remains satisfied as the removal of the H-08 control rod was not found to result in power oscillations which would otherwise result in conditions exceeding specified acceptable fuel design limits.

- GDC Criterion 23 – Protection System Failure Modes: *The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.*

  The protection system is designed with consideration of the most probable failure modes of the components under various perturbations of the environment and energy sources. Each reactor trip channel is designed on the de-energize-to-trip principle so that loss of power, disconnection, open channel faults, and the majority of the internal channel short-circuit faults cause the channel to go into its tripped mode.

Similarly, that portion of the engineered safety features actuation system provided for actuation of auxiliary feedwater system is designed to fail into a safe state, except for the final output relays. The relays are energized to actuate as are the pumps and motor-operated valves of the actuated equipment.

This criterion remains satisfied because the removal of the H-08 control rod 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 shutdown capability. The mechanical removal of the control rod drive shaft does not have any mechanical impact on the function of the remaining 52 control rods. The electrical removal from service of the H-08 control rod 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 the H-08 control rod and do not affect other control rods.

Therefore, 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.

- GDC Criterion 25 – Protection System Requirements for Reactivity Control Malfunctions: *The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.*

  The protection system is designed to limit reactivity transients so that the fuel design limits are not exceeded. Reactor shutdown by control rod insertion is completely independent of the normal control function since the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. Thus, in the postulated accidental withdrawal of a control rod or control rod bank (assumed to be initiated by a control malfunction) neutron flux, temperature, pressure, level, and flow signals would be generated independently. Any of these signals (trip demands) would operate the breakers to trip the reactor.

  Analyses of the effects of possible malfunctions are discussed in Chapter 15 of the Callaway FSAR. These analyses show that for postulated boron dilution during refueling, startup, manual or automatic operation at power, hot standby, or cold shutdown, the operator has ample time to determine the cause of dilution, terminate the source of dilution, and initiate reboration before the shutdown margin is lost. Either manual or automatic controls can be used to terminate dilution and initiate boration. The analyses show that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system.

  This criterion remains satisfied because an operating cycle Cycle 28 redesign reload analysis, performed according to methods referenced in TS 5.6.5, confirms 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.

- GDC Criterion 26 – Reactivity Control System Redundance and Capability: *Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.*

  Two reactivity control systems are provided. These are RCCAs and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.

  During operation, the shutdown rod banks are fully withdrawn. Using the rod control system, the operator maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks, along with the control banks, are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in the core life is assumed in all analyses, and the most reactive rod cluster is assumed to be in the fully withdrawn position.

  The boron system will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.

  This criterion remains satisfied because removal of the H-08 control rod 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 the H-08 control rod.

- GDC Criterion 27 – Combined Reactivity Control Systems Capability: *The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.*

  The facility is provided with means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. Combined use of the rod cluster control system and the chemical shim control system permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck full out upon trip for this determination.

This criterion remains satisfied because removal of the H-08 control rod does not impact the ability of the reactivity control systems to reliably control reactivity changes. Further, adequate SDM is analytically shown to be maintained even when considering the highest stuck rod worth. Evaluations of the removal of the H-08 control rod during Cycle 28 demonstrate that SDM and safety analysis limits are met throughout the fuel cycle.

- GDC Criterion 28 – Reactivity Limits: *The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.*

  The maximum reactivity worth of the control rods and the maximum rates of reactivity insertion employing control rods and boron removal are limited to values that prevent any reactivity increase from rupturing the reactor coolant system boundary or disrupting the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

  The appropriate reactivity insertion rate for the withdrawal of RCCAs and the dilution of the boric acid in the reactor coolant systems are specified in the Technical Specifications for the facility. The Technical Specifications in combination with the COLR include appropriate graphs that show the permissible withdrawal limits and overlap of the RCCA banks as a function of power.

  Core cooling capability following accidents, such as rod ejection, steam line break, etc., is assured by keeping the reactor coolant pressure boundary stresses within faulted condition limits, as specified by applicable ASME codes. Structural deformations are also checked and limited to values that do not jeopardize the operation of needed safety features.

  This criterion remains satisfied because removal of the H-08 control rod has been evaluated to ensure trip reactivity insertion rate, shutdown margin, and the safety analysis limits remain met for the FSAR Chapter 15 accidents for the entire fuel cycle (Cycle 28).

- GDC Criterion 29 – Protection Against Anticipation Operational Occurrences: *The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.*

  The protection and reactivity control systems have an extremely high probability of performing their required safety functions in any anticipated operational occurrences. Diversity and redundancy, coupled with a rigorous quality assurance program and analyses, support this probability as does operating experience in plants using the same basic design. Failure modes of system components are designed to be safe modes. Loss of power to the protection system results in a reactor trip.

This criterion remains satisfied because the removal of the H-08 control rod 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 do not have any mechanical impact on the function of the remaining 52 control rods. The electrical removal from service of the H-08 control rod 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 the H-08 control rod and do not affect other control rods. Therefore, a high probability of control rod insertion continues to exist under anticipated operational occurrences, even with the removal of the H-08 control rod during Cycle 28.

- 10 CFR 50.62(c)(1): *Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an Anticipated Transient Without Scram (ATWS). This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.*

  The requirements of 10 CFR 50.62(c)(1) applicable to Callaway continue to be met. Removal of the H-08 control rod does not impact Anticipated Transient Without Scram (ATWS) Mitigation System Actuation Circuitry, and changes to parameters described in the license amendment request (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 Callaway, and subsections (c)(3) through (c)(5) are applicable only to boiling water reactors.)

There are no changes being proposed in this amendment application such that conformance or commitments to the regulatory requirements and/or guidance documents above would come into question. The evaluations documented herein confirm that Callaway Plant will continue to comply with all applicable regulatory requirements.

In conclusion, based on considerations discussed herein, (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) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.2 Precedent

Ameren Missouri 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.

NRC Letter to Tennessee Valley Authority, "Sequoyah Nuclear Plant, Unit 1 – Issuance of Exigent Amendment No. 348 to Operate One Cycle with One Control Rod Removed (EPID L-2019-LLA-0239)," dated November 21, 2019 (ML19319C831).

NRC Letter to South Texas Project, “South Texas Project Unit 1 - Issuance of Amendment Re: Revision to Technical Specifications for One Operating Cycle Operation with 56 Control Rods (Emergency Circumstances) (TAC No. MF7142),” dated December 11, 2015 (ML15343A128).

In addition, a similar License Amendment Request was submitted for McGuire Nuclear Station Unit 2 (ML18254A182), but this was ultimately withdrawn when repair efforts were successful.

4.3 No Significant Hazards Consideration Determination

Ameren Missouri has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

Response: No

The proposed license amendment would add a note to Callaway Unit 1 Technical Specification (TS) 4.2.2, “Control Rod Assemblies,” to permit Cycle 28 to contain 52 control rods, i.e., with no control rod in core location H-08. Currently, TS 4.2.2 requires 53 control rod assemblies in each reactor core.

This proposed license amendment would allow for the temporary removal of the control rod in core location H-08 during Cycle 28. Operation of Callaway Cycle 28 with the H-08 control rod removed will not involve a significant increase in the probability or consequences of an accident previously evaluated. Shutdown margin (SDM) is reduced by the absence of the H-08 control rod but remains bounded by the limits specified by the Core Operating Limits Report (COLR). Because the impacts on the cycle-specific nuclear design parameters are bounded by the conservative input values used in the Final Safety Analysis Report (FSAR) accident analyses, the current accident analyses remain bounding. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will not alter or prevent the ability of the remaining 52 control rods from performing their intended functions to mitigate the consequences of an initiating event within the assumed acceptance limits. The mechanical removal of the control rod drive shaft and Rod Cluster Control Assembly (RCCA) does not have any mechanical impact on the function of the remaining 52 control rods. The electrical removal from service of the H-08 control rod 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 the H-08 control rod and do not affect other control rods.

The proposed changes do not alter any accident analysis assumptions discussed in the FSAR. Shutdown Margin (SDM) is reduced by the absence of the H-08 control rod but remains bounded by the limits specified by the COLR.

Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

Operation of Callaway Cycle 28 with the H-08 control rod removed will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed change involves no physical change beyond the removal of the H-08 control rod. The safety evaluations performed for Cycle 28 with the H-08 control rod removed validated that the impacts to the nuclear design parameters were within the bounds of those already assumed in the FSAR Chapter 15 accident analyses. The change in core bypass flow through the upper head region has been evaluated, and it has been determined that the increase is negligible and there is no impact to the safety analysis due to this negligible increase. The current accident analyses remain bounding. All plant equipment will continue to meet applicable design and safety requirements.

Therefore, it is concluded that this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed change does not alter any FSAR design basis or safety limit and does not change any setpoint at which automatic actuations are initiated. The proposed change has been evaluated for effects on available shutdown margin, boron worth, trip reactivity as a function of time, and moderator temperature coefficient. The results of these evaluations show that adequate margin is maintained such that the proposed change would not cause a design basis or safety limit to be altered or exceeded.

Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

In consideration of all the above, Ameren Missouri concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and on that basis, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the considerations discussed above, 1) there is a 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.

5.0 ENVIRONMENTAL EVALUATION

The proposed change 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 change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change 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 change.