Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-20-042

April 17, 2020

10 CFR 50.90
10 CFR 50.91

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

Sequoyah Nuclear Plant, Unit 2
Renewed Facility Operating License No. DPR-79
NRC Docket No. 50-328

Subject: **Sequoyah Nuclear Plant Unit 2 – Exigent License Amendment Request to Revise Technical Specification 4.2.2, "Control Rod Assemblies" (SQN-TS-20-05)**

In accordance with the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," and 10 CFR 50.91(a)(6), "Notice for public comment; State consultation," Tennessee Valley Authority (TVA) is submitting an exigent request for an amendment to Renewed Facility Operating License (RFOL) No. DPR-79 for Sequoyah Nuclear Plant, Unit 2 (SQN2) for Nuclear Regulatory Commission (NRC) approval. The proposed amendment would revise Technical Specification (TS) 4.2.2, "Control Rod Assemblies," to permit the SQN2 Cycle 24 (U2C24) core to contain 52 full length control rods with no full length control rod assembly in core location H-08 for one cycle.

On August 27, 2019, at 0109 while operating at 100 percent power, the control rod in the Sequoyah Nuclear Plant, Unit 1 (SQN1) core location H-08 (H-08 control rod) unexpectedly dropped into the core, resulting in an automatic reactor trip of SQN1. Testing and inspections performed during the SQN1 refueling outage Cycle 23 (U1R23) determined that wear of the control rod drive mechanism (CRDM) stationary gripper latch mechanism resulted in the inability to maintain the control rod in the fully withdrawn or nearly fully withdrawn position. Because in-situ replacement of the affected CRDM would be a first-of-a-kind activity in the United States and would require special tooling that is unavailable at this time, TVA opted to remove the H-08 control rod. SQN1 was issued Exigent License Amendment 348 (ML19319C831) to allow SQN1 to operate for Operating Cycle 24 (U1C24) with 52 full length control rod assemblies instead of 53 full length assemblies.

An extent of condition examination performed during the SQN2 Cycle 23 refueling outage (U2R23) revealed similar wear indications on its H-08 CRDM gripper latch mechanism. Therefore, TVA has decided to remove the H-08 control rod from SQN2 and operate for U2C24 with 52 full length control rod assemblies instead of 53 full length assemblies. SQN2 is currently in Mode 6. Approval of this license amendment request (LAR) is required for SQN2 to enter Mode 5 and resume power operation at the conclusion of the U2R23 refueling outage.

The Enclosure to this letter provides a technical and regulatory evaluation of the proposed amendment. Attachments 1 and 2 to the Enclosure contain the proposed TS page markup and clean TS pages, respectively.

TVA is requesting approval of the proposed amendment on an exigent basis pursuant to 10 CFR 50.91(a)(6) and is requesting approval by April 24, 2020. The TS change will provide a one-time allowance that will be in effect for the duration of U2C24. Once approved, the amendment shall be implemented within 24 hours. TVA is requesting NRC approval of the proposed change to TS 4.2.2; as all other design changes and supporting safety analyses discussed in this exigent LAR were performed in accordance with the current licensing basis. TVA plans to submit a future LAR to make this a permanent configuration.

TVA determined that there are no significant hazards consideration associated with the proposed change and that the TS change qualifies for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosure to the Tennessee State Department of Environment and Conservation.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Kimberly D. Hulvey, Senior Manager, Fleet Licensing, at (423) 751-3275.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 17th day of April 2020.

Respectfully,

James Barstow

James Barstow
Vice President, Nuclear Regulatory Affairs & Support Services

Enclosure:

Evaluation of the Proposed Change

cc: See Page 3

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Sequoyah Nuclear Plant
NRC Project Manager – Sequoyah Nuclear Plant
Director, Division of Radiological Health - Tennessee State Department of Environment and Conservation

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNIT 2

EVALUATION OF THE PROPOSED CHANGE

Subject: **Sequoyah Nuclear Plant Unit 2 - Exigent License Amendment Request to Revise Technical Specification 4.2.2, "Control Rod Assemblies" (SQN-TS-20-05)**

Contents

ATTACHMENTS
1. Proposed TS Changes (Mark-Ups) for SQN Unit 2
2. Proposed TS Changes (Final Typed) for SQN Unit 2

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Facility Operating License DPR-79 for Sequoyah Nuclear Plant, Unit 2 (SQN2), by adding a note to SQN2 Technical Specification (TS) 4.2.2, “Control Rod Assemblies,” to permit the SQN2 Cycle 24 (U2C24) core to contain 52 full length control rods with no full length control rod in core location H-08, in lieu of the current requirement of 53 full length control rods. A SQN operating cycle is nominally 18 months.

On August 27, 2019, at 0109 while operating at 100 percent power, the control rod in the Sequoyah Nuclear Plant, Unit 1 (SQN1) core location H-08 (H-08 control rod) unexpectedly dropped into the core, resulting in an automatic reactor trip of SQN1. Testing and inspections performed during the SQN1 refueling outage Cycle 23 (U1R23) determined that wear of the control rod drive mechanism (CRDM) stationary gripper latch mechanism resulted in the inability to maintain the control rod in the fully withdrawn or nearly fully withdrawn position. Because in-situ replacement of the affected CRDM would be a first-of-a-kind activity in the United States and would require special tooling that was unavailable at that time, TVA opted to remove the SQN1 H-08 control rod. SQN1 was issued Exigent License Amendment 348 (ML19319C831) to operate U1C24 with 52 full length control rod assemblies instead of 53 full length assemblies.

Extent of condition examinations on SQN2 during the SQN2 refueling outage Cycle 23 (U2R23) refueling outage revealed similar wear indications on its H-08 control rod CRDM gripper latch mechanism. Therefore, because in-situ replacement of the affected CRDM remains as described for the SQN1 H-08 control rod, TVA has decided to remove the H-08 control rod from SQN2 and operate for U2C24 with 52 full length control rod assemblies instead of 53 full length assemblies. TVA plans to submit a future LAR to make this a permanent configuration.

2.0 DETAILED DESCRIPTION

2.1 PROPOSED TECHNICAL SPECIFICATION CHANGE

The proposed amendment would revise TS 4.2.2 to add a note permitting operation with 52 full length control rods during U2C24, in lieu of the requirement for 53 full length control rods. TVA has reviewed the SQN2 TS and has determined that no additional TS changes are required.

The proposed TS note is as follows:

> Operation with 52 full length control rod assemblies (with no control rod assembly installed in core location H-08) is permitted during Cycle 24.

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

Attachment 1 provides a marked-up version of the affected page of SQN2 TS 4.2.2 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.

2.2 CIRCUMSTANCES ESTABLISHING NEED FOR THE PROPOSED EXIGENT AMENDMENT

As provided for in 10 CFR 50.91(a)(6)(vi), TVA is required to explain the exigency and why this cannot be avoided.

SQN1 was initially unable to startup from the U1R23 outage in November 2019 due to unreliable RCCA performance in core location H-08. The control rod in core location H-08 was unable to be reliably held in the withdrawn position due to worn stationary gripper latch mechanisms in the control rod drive system, and unexpectedly dropped three times during Mode 3 testing activities at the end of U1R23. The SQN1 H-08 control rod was unable to maintain a withdrawn position reliably, and was unlikely to remain in position for the entire cycle.

The replacement of a CRDM of similar design and installation configuration had not been performed in the United States. In-situ replacement of the H-08 control rod CRDM required specially modified tooling, similar in nature to the original manufacturing tooling, which did not currently exist. Additionally, the planning associated with a CRDM replacement activity would have required fabrication of mockups to test the effectiveness of the tooling, methods, and procedures. The planning and preparation process was expected to require a lead time on the order of months. Therefore, the above repair/replacement discussion provided the basis for exigency.

Consideration was given to operating SQN1 during U1C24 with the H-08 control rod fully inserted in the core. This option was not considered to be viable for the following reasons:

- The core would be susceptible to radial xenon oscillations that would challenge operator responses,
- Uneven depletion of fuel assemblies would have had a significant impact on the core design for future fuel cycles with regard to safety/operating margins and fuel economy, and
- The impact on core power distribution would have likely required operation at a reduced power level.

TVA determined that the safest option was to operate SQN1 during U1C24 with the H-08 control rod removed. Based on the above, this option was unavoidable and was exigent in nature.

In response to the SQN1 operating experience, an extent of condition exam was planned for the U2R23 refueling outage. A sampling of 14 CRDM locations was selected for the scope of this inspection. This sample population included H-08, the eight surrounding locations, a subset of control bank “D”, and locations in shutdown banks in order to best understand which locations are most susceptible to this wear mechanism. A set of examination criteria was developed based on all available data to assess the potential risk that SQN2 CRDMs may be in the same material condition as the SQN1 H-08 CRDM. This included a review of CRDM coil current trace data, rod drop times, guide card wear, and thermal sleeve wear data. With this supporting information, an inspection procedure and decision flow chart was prepared by TVA and the original equipment manufacturer. This material directed that any CRDM with a latch arm tip thickness worn to zero thickness (as was observed in the SQN1 H-08 CRDM inspections) and also had unexpected CRDM coil trace results would be graded as a high risk of a control rod drop. The SQN2 H-08 CRDM met both of these criteria, a condition that could not have been known until inspection results were obtained during the outage.

Inspections were completed for all 14 CRDM locations selected for the U2R23 refueling outage examination scope. The 13 remaining locations were inspected, reviewed, and determined to have adequate latch arm tip thickness with minimal wear reported. Based on the inspection and review, the risk condition of all remaining inspected CRDM stationary gripper latch mechanisms was determined to be low based on the examination criteria discussed previously.

Because the condition of the SQN2 CRDM stationary gripper latch mechanisms was not completely known pre-outage, reasonable preparations were made with the original equipment manufacturer in the event that CRDM repair or replacement would have been shown to be the best option. Many of the same challenges regarding domestic precedence, tooling, qualifications, and implementation that existed at the time of the SQN1 H-08 operating experience still exist today. A repair method has also been considered, which would replace only the internal latch assembly instead of the whole CRDM. While this slightly reduces the scope of work required, implementation schedules and challenges remain similar to the CRDM replacement option.

In summary, the extent of condition examination of the SQN2 H-08 control rod revealed similar wear indications on the CRDM stationary gripper latch mechanism. It is therefore believed that SQN2 H-08 control rod will not reliably maintain in the withdrawn position, resulting in SQN2 being unable to startup or maintain power operation. As with SQN1, TVA has determined that the safest option is to operate SQN2 during U2C24 with the H-08 control rod removed. This option is similarly unavoidable and is exigent in nature. TVA plans to provide the technical evaluation supporting permanent removal of the SQN2 H-08 control rod in a future license amendment request.

3.0 TECHNICAL EVALUATION

3.1 SYSTEM DESCRIPTION

SQN2 normally contains 53 full-length control rod assemblies divided into four control banks (Control Banks A, B, C, D) and four shutdown banks (Shutdown Banks A, B, C, D). Of the eight 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 control rod in core location H-08, U2C24 will contain 52 full length control rod assemblies as shown in the table to Figure 1.

Each control rod is moved by a full length 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.

Should both sets of grippers be de-energized simultaneously, the corresponding control rod would drop into the core. The primary function of the CRDMs is to insert, withdraw, or hold control rods 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 houses and guides the control rod through the upper internals.

The full length Rod Control System receives rod speed and direction signals from the T_{avg} control system (contained within the Distributed Control System). The automatic rod speed demand signal varies over the corresponding range of 5 to 45 inches per minute (8 to 72 steps/minute) depending on the magnitude of the error signal. The rod direction demand signal is determined by the positive or negative value of the error signal. Manual control is provided to move a control bank in or out at a prescribed fixed speed.

Figure 1 - Control Rod Locations

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

90°

0°

Bank Identifier	Number of Locations	Bank Identifier	Number of Locations
SA	8	CA	4
SB	8	CB	8
SC	4	CC	8
SD	4	CD	8

Note: The control rod in core location H-08 will be removed for U2C24.

3.2 CURRENT LICENSING BASIS

Framatome performs the reload licensing analysis for Sequoyah Nuclear Plant and applies NRC-approved codes and analytical methods to license the reload core. The

NRC-approved codes and analytical methods used to generate the reload safety evaluation are included in TS 5.6.3 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 Sequoyah Unit 2 Cycle 24 (U2C24) 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 without the H-08 control rod for U2C24 were performed to ensure core protective and operating limits remain satisfied and safety analysis limits remain bounded.

As described in Updated Final Safety Analysis Report (UFSAR) Section 4.2.3.2.1, “Reactivity Control Components”:

The rod cluster control assemblies are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variation 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 instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

As described in UFSAR Section 4.3.2.5.2, “Rod Cluster Control Assemblies”:

The number of Rod Cluster Control Assemblies is shown in Table 4.3.2-1. The Rod Cluster Control Assemblies are used for shutdown 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. *Unprogrammed fluctuations in boron concentration, coolant temperatures, or xenon concentration (with rods not exceeding the allowable rod insertion limits),*
4. *Reactivity ramp rates resulting from load changes.*

The allowed full length 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 Technical Specifications 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 4.3.2-36. 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 SQN Technical Specifications.

3.3 IMPACT ON THE SAFETY ANALYSIS

The removal of the H-08 control rod from Control Bank D is considered to apply for the entirety of U2C24 operation and impacts all 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 UFSAR 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 document.

NRC-approved reload safety analysis codes and methods were used to determine if the change in core design parameters adversely impacted the bounding key safety parameters assumed in the 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. The cycle-specific power distribution Maneuvering Analysis (MA) was also evaluated to determine the acceptability of the TS / COLR operating limits related to the LOCA and loss of forced reactor coolant flow accident initial condition criteria.

Evaluation of impacts to core and fuel performance, as well as the impact to the safety analyses described in UFSAR Chapter 15 and safety analysis parameters, are summarized in the cycle-specific reload safety evaluation documentation to confirm the acceptability of safe 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 U2C24 with the H-08 control rod removed. The analysis supporting the evaluation of these impacted parameters was performed using an NRC-approved methodology described in TS 5.6.3. The U2C24 COLR will be submitted to the NRC 30 days after issuance, as prescribed in TS 5.6.3. Results of the safety analysis impact evaluation are described in the following subsections.

With one exception, all Technical Specifications remain unchanged as a result of the H-08 control rod being removed. The exception is an added note to TS 4.2.2 for the removal of the H-08 control rod.

Shutdown Margin

The proposed change impacts the available shutdown margin (SDM). 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 UFSAR 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 1. The SDM is reduced from 2.812 %ΔK/K to 2.253 %ΔK/K, which remains bounded by the 1.6 %ΔK/K limit for MODES 1 and 2 specified in COLR. By maintaining the 1.6 %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 the H-08 control rod inserted, is 1.226 %ΔK/K in core location C-11. With the H-08 control rod removed, the worth of the most reactive stuck rod in an N-1 configuration at core location F-08 and symmetric locations is 1.176 %ΔK/K.

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

	U2C23 RCCA in H-08 (%ΔK/K)	U2C24 RCCA in H-08 (%ΔK/K)	U2C24 NO RCCA in H-08 (%ΔK/K)
Available Rod Worth			
Total Rod Worth	8.936	8.277	7.563
Maximum Stuck Rod Worth	1.515	1.226	1.176
Variable ARO	0.060	0.050	0.049
Maximum Insertable Worth	7.361	7.001	6.338
Uncertainty	0.589	0.560	0.507
Total Available Worth	6.772	6.441	5.831
Required Rod Worth			
Power Defect	2.696	2.634	2.632
Off-Nominal Allowance	0.300	0.300	0.300
Rod Insertion Allowance	0.517	0.511	0.467
Cooldown Defect	0.191	0.184	0.179
Total Required Worth	3.704	3.629	3.578
Available Shutdown Margin	3.068	2.812	2.253
Mandatory Shutdown Margin	1.600	1.600	1.600

COLR Sections 2.1.1 and 2.1.2 also provide the required SDM limits for MODES 3, 4, and 5, and MODE 2 with $K_{eff} < 1.0$. Per these sections, SDM must be greater than or equal to 1.6 %ΔK/K in MODES 3 and 4, and it must be greater than or equal to 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 steam line break event from hot zero power (HZP) and the boron dilution event.

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 1.6 %ΔK/K, the steam line break event remains bounding. As discussed above, the removal of the H-08 control rod does not result in an SDM of less than the limit of 1.6 %ΔK/K. A key parameter for the Boron Dilution event is SDM. An evaluation of the effect on SDM with the H-08 control rod 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 will be higher with the H-08 control rod 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 for 1.6% SDM and 1.0% SDM 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 1.6% SDM (Modes 1-4) and 1.0% SDM (Mode 5) Conditions

1.6% SDM	**RCCA in H-08 Required Boron (ppm)**			**NO RCCA in H-08 Required Boron (ppm)**		
		350 °F	**547 °F**		**350 °F**	**547 °F**
BOC		1452	1260		1523	1368
MOC		1232	997		1245	1079
EOC		482	64		485	128
1.0% SDM	**RCCA in H-08 Required Boron (ppm)**			**NO RCCA in H-08 Required Boron (ppm)**		
	50 °F	**135 °F**		**50 °F**	**135 °F**	
BOC	1516	1491		1527	1510	
MOC	1267	1249		1268	1250	
EOC	609	582		611	584	

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

The removal of the H-08 control rod was evaluated for impact on SDM boron concentration requirements and differential boron worth as a function of boron concentration in a rodded configuration. The removal of the H-08 control rod increases the SDM boron concentration requirement to compensate for the loss in the available total RCCA negative reactivity and to compensate for the reduction in boron worth when the H-08 control rod is removed. The increase in the SDM boron concentration requirements in the RCS for MODES 1, 2, 3, 4, and 5, ensures the removal of control rod H-08 does not impact the results presented in the uncontrolled boron dilution accident (UBDA).

Note: Post-LOCA Subcriticality boron concentrations are calculated at a conservative all rods out (ARO) configuration; therefore, these results are not impacted by the removal of the H-08 control rod.

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 normalized trip reactivity as a function of time after the RCCAs begin to fall is presented in the UFSAR. The curve of trip reactivity as a function of time used in the safety analyses is verified to be bounding by a cycle-specific calculation of the minimum trip worth at hot full power (HFP) and HZP. An evaluation of the effects of the removal of the H-08 control rod shows that the minimum trip worth is greater than the limit of 4000 pcm and therefore the curve of trip reactivity as a function of time after the RCCAs begin to fall used in the safety analyses remains bounding. Table 3 provides a comparison of the minimum trip worth for U2C24 with and without the H-08 control rod inserted. Therefore, the removal of the H-08 control rod does not impact the trip reactivity assumed in UFSAR Chapter 15 events.

Table 3 - Minimum Trip Worth Results

	HFP Minimum Trip Worth (pcm)	HZP Minimum Trip Worth (pcm)
RCCA in H-08	5830	5089
NO RCCA in H-08	5053	4364
Limit	> 4000	> 4000

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 uncontrolled bank withdrawal transient from any power level, loss of electrical load, 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 condition. The most conservative combination appropriate to the accident is then used for the analysis.

The removal of the H-08 control rod only slightly impacts the MTC calculated at the conservative bounding conditions determined for the UFSAR accident analyses. MTC results for U2C24 with the H-08 control rod in and out are shown in Table 4 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 UFSAR for the above listed events.

Table 4 - MTC Limit Summary for U2C24 with and without the H-08 Control Rod

Limit Description	Limit (pcm/°F)	U2C24 Reload Values- RCCA in H-08 (pcm/°F)	U2C24 Reload Values- NO RCCA in H-08 (pcm/°F)
Most positive HFP MTC	< 0.0	-7.98	-7.98
HFP error-adjusted rod insertion limit (Bank CD at 182 steps withdrawn (SWD)) EOC	> -45	-34.11	-34.01
Near-EOC MTC at 300 ppmB	>-38	-27.24	-27.29
Near-EOC MTC at 60 ppmB	>-42	-31.50	-31.50
Most positive HZP MTC	< 0.0	-1.25	-1.25

<u>UFSAR Chapter 15 Accident Analyses Impacts from Removal of the H-08 control rod</u>

Removal of the H-08 control rod from U2C24 has an impact on most comparisons to UFSAR Chapter 15 accident analysis parameters routinely evaluated as part of the reload safety evaluation. In addition to SDM, MTC, trip reactivity, boron concentration, and boron worth accident analysis parameters discussed above, the impact of removal of the H-08 control rod on control rod worth, margin to peaking limits (DNB and CFM), and other accident analysis parameters was also evaluated. The removal of the H-08 control rod impacts these parameters by reactivity effects on calculated boron concentrations, control rod position reactivity worth, or power distribution effects due to the modified control rod pattern during rodded power maneuvers.

Cycle-specific evaluations were performed to determine if the change in core configuration adversely impacts bounding key safety parameters assumed in the UFSAR Chapter 15 safety analysis and impacts on DNB and fuel thermal margins due to the change in power distribution. The bounding key safety parameters were developed in UFSAR Chapter 15 accident analyses of record (AORs) 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 from U2C24 does not impact the results presented in the UFSAR Chapter 15 accident analyses. Results and discussions of the UFSAR Chapter 15 accident analyses for U2C24 with the H-08 control rod removed are provided below.

1. HZP 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 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 Framatome reload core methodology for the SLB event from zero power 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) 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.

Nuclear Enthalpy Rise Hot Channel Factor (FΔH) and Total Heat Flux Channel Factor (Fq) are presented in Table 5 as key parameters for DNB and CFM evaluations, though they are not compared against limits on a cycle-specific basis. Due to the decrease in both FΔH and Fq with the H-08 control rod removed relative to the case where the H-08 control rod was present in the core, it was determined that DNB and CFM evaluations for the H-08 removal case were not necessary. The rodded H-08 case was determined to be bounding for DNB and CFM.

Cycle-specific parameter evaluations for the HZP SLB accident are also presented in Table 5. The cycle-specific evaluation confirms the limits assumed in the safety analysis remain bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in the UFSAR section for the HZP SLB accident.

Table 5 - HZP Steam Line Break Parameter Results

	Maximum FΔH	Maximum Fq	Most Positive Steam Line Break Reactivity ($)	Most Negative INBW* (ppm/%ΔK/K)
RCCA in H-08	12.468	23.001	-1.30	-105
NO RCCA in H-08	11.320	22.603	-2.08	-105
Limit	-	-	< +0.057801	> -125

* Inverse Boron Worth (INBW)

2. Steam Line Break Coincident with Bank Withdrawal at Power (SLB c/w RWAP as defined the SQN UFSAR)

   The SLB c/w RWAP accident postulates that an uncontrolled bank withdrawal occurs due to a fault in the automatic rod control system caused by adverse conditions resulting from a steam line break. This uncontrolled bank withdrawal causes an increase in core power which could lead to possible 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 c/w RWAP. The Framatome reload core methodology for the SLB c/w RWAP event uses safety analysis and nuclear design methods to determine if the reference transient analysis state points (reactor power level, control rod position, inlet temperature, pressure, flow, and reactivity) remain bounding for the reload core.

Cycle-specific parameter evaluations for the SLB c/w RWAP accident are presented in Table 6. The cycle-specific evaluation confirms the parameters assumed in the safety analysis remain bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in the UFSAR section for the SLB c/w RWAP accident.

Table 6 - SLB c/w RWAP Parameter Evaluation Results

	BOC Reactivity Defect (pcm)	EOC Reactivity Defect (pcm)	EOC HFP MTC (pcm/℉)	MTC at Limiting Reactivity Insertion Rate (pcm/℉)
RCCA in H-08	-151	-264	-34.11	-26.2
NO RCCA in H-08	-171	-286	-34.01	-28.5
Limit	< 59.0	< 28.0	> -45.0, < -22.0	< -22.0

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. A DNB analysis is then performed by comparing the cycle-specific power peaking factors at ARO and RIL conditions to bounding power peaking factors. The removal of the H-08 control rod could impact the localized reactor core power distribution for the LRA event.

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

Table 7 - LRA DNB Pin Census Results

	Maximum Fraction of Pins Failed
RCCA in H-08	8.44%
NO RCCA in H-08	9.26%
Limit	< 10%

4. 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 the H-08 control rod will impact the localized reactor core power distribution for events where a power excursion occurs.

Cycle-specific evaluations for the UCBW from subcritical accident are presented in Table 8. A comparison of the cycle-specific value and the limit confirms that positive margin exists. Therefore, the removal of the H-08 control rod does not impact the results presented in the UFSAR section on UCBW accident from subcritical.

Table 8 - UCBW from Subcritical Reactivity Insertion Rate and Peaking Evaluation Results

	Maximum HZP Reactivity Insertion Rate (pcm/sec)	**Maximum HZP Radial Pin Power**
RCCA in H-08	45.04	1.8916
NO RCCA in H-08	40.97	1.8996
Limit	< 57	< 1.9596

5. Uncontrolled Bank Withdrawal (UCBW) at Power

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because 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 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 the H-08 control rod will impact the localized reactor core power distribution for events where a rod power maneuver occurs.

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

Table 9 - UCBW at Power Reactivity Insertion Rate Results

	BOC Maximum HFP Reactivity Insertion Rate (pcm/sec)	**EOC Maximum HFP Reactivity Insertion Rate (pcm/sec)**
RCCA in H-08	9.89	17.87
NO RCCA in H-08	8.87	16.16
Limit	< 75	< 75

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

RCCA misoperation accidents include:

a. Withdrawal of a single RCCA
b. Statically misaligned RCCA
c. One or more dropped RCCAs within the same group or a dropped RCCA bank

a. Withdrawal of a Single RCCA

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. A DNB analysis is performed using the power peaking factors at the rod insertion limit for the reload core. 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. Note: for the H-08 control rod removed in U2C24, a single RCCA withdrawal accident event cannot occur for this core location.

Cycle-specific evaluations for single RCCA withdrawal accident are presented in Table 10 and confirm that positive margin exists. Therefore, the removal of the H-08 control rod does not impact the results presented in the UFSAR section on single RCCA withdrawal accident.

Table 10 - Single Rod Withdrawal DNB Pin Census Results

	Maximum Fraction of Pins Failed
RCCA in H-08	1.3%
NO RCCA in H-08	1.3%
Limit	< 5%

b. Statically Misaligned RCCA

Statically misaligned control rod events result in asymmetric radial peaking that could result in DNB. A DNB analysis is performed at allowable HFP rod positions for the reload core. 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. Note: for the H-08 control rod removed in U2C24, a statically misaligned RCCA accident event cannot occur for this core location.

Cycle-specific parameter evaluations for the statically misaligned RCCA accident are presented in Table 11 and confirm that positive margin exists. Therefore, the removal of the H-08 control rod does not impact the results presented in the UFSAR section on the statically misaligned RCCA accident.

Table 11 – Statically Misaligned RCCA DNB Peaking Margin Results

	Minimum DNB Peaking Margin
RCCA in H-08	9.48%
NO RCCA in H-08	10.22%
Limit	> 0%

c. One or more Dropped RCCAs within same Group or Dropped RCCA Bank (DRA)

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 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. Note: for the H-08 control rod removed in U2C24, a DRA accident event cannot occur for control rod in this core location. Also, the Control Bank D rod group previously containing H-08 will go from 5 to 4 control rods with H-08 excluded.

Cycle-specific parameter evaluations for DRA 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 the UFSAR section on DRA.

Table 12 - DRA DNB Peaking Margin Results

	Minimum DNB Peaking Margin
RCCA in H-08	0.801%
NO RCCA in H-08	0.768%
Limit	> 0%

7. <u>Rod Ejection Accident (REA)</u>

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 Sequoyah 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, that lessens the potential ejected 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. Xenon effects are considered in the analysis. 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 neutronics codes approved for reload design analysis.

The REA is evaluated for the plant and control bank conditions described above and bounding REA safety analysis limits; a peaking evaluation is performed to determine the number of pins in DNB for REA. The removal of the H-08 control rod will impact the localized reactor core power distribution for REA. Note: For the H-08 control rod removed in U2C24, an REA accident will not occur in the H-08 location for this configuration.

Cycle-specific parameter evaluations for REA are presented in Table 13 and confirm that the limits assumed in the safety analysis remain bounding. Therefore, the removal of the H-08 control rod does not impact the results presented in the UFSAR section on REA.

Table 13, REA Max Ejected Rod Worth, Peaking Evaluation, and Pin Census Results

	HFP Ejected Rod Results				
	Maximum BOC Ejected Rod Worth (pcm)	**Maximum EOC Ejected Rod Worth (pcm)**	**Maximum BOC Fq After Event**	**Maximum EOC Fq After Event**	**Maximum Fraction of Pins Failed**
RCCA in H-08	32.865	43.935	2.243	2.558	0.49%
NO RCCA in H-08	35.565	46.086	2.268	2.552	0.40%
Limit	< 200	< 210	< 7.11	< 7.88	< 10%

	HZP Ejected Rod Results			
	Maximum BOC Ejected Rod Worth (pcm)	**Maximum EOC Ejected Rod Worth (pcm)**	**Maximum BOC Fq After Event**	**Maximum EOC Fq After Event**
RCCA in H-08	540.240	797.297	8.538	20.604
NO RCCA in H-08	453.516	742.440	7.517	19.427
Limit	< 750	< 910	< 14.05	< 24.8

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.

Safety Analysis Evaluation Summary

To summarize, the impact of the removal of the H-08 control rod in U2C24 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 5.6.3. These NRC-approved reload safety evaluation methods were used to determine if the change in core configuration 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 the H-08 control rod removed. Cycle-specific parameter evaluations for UFSAR Chapter 15 Safety Analysis parameters confirm that the values assumed in the safety analysis remain bounding for all UFSAR Chapter 15 Safety Analysis accidents.

Therefore, removal of the H-08 control rod for U2C24 does not impact the results presented in UFSAR Chapter 15. Table 14 presents a summary of the impact of removal of the H-08 control rod on each Chapter 15 Safety Analysis accident. Observed cycle-specific results from the UFSAR Chapter 15 Safety Analysis technical evaluation with the 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 U2C24 core.

- REA Ejection Accident (REA) ejected rod worths slightly increased for the HFP cases and were reduced for HZP cases. Peaking results were reduced for all cases with exception of the HFP BOC case, which increased but maintained margin to the safety limit. These changes are due to the power shifting more towards the center of the core during the REA due to the removal of the control rod in core location H-08. This power distribution change reduced the peaking for the REA for most cases and changed the ejected rod worths only slightly. 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 in the H-08 location.

- DRA DNB margins decreased due to the removal of the H-08 control rod, but still had significant margin to the limit. The DRA Control Bank D rod group previously containing H-08 will go from 5 to 4 control rods with H-08 excluded.

- HZP SLB peaking results were reduced because power was anchored toward center of core with no control rod in H-08 and maximum stuck rod out. The reduction in peaking of the No H-08 case relative to the rodded H-08 case meant the rodded H-08 DNB and CFM calculations bounded the No H-08 case, and therefore the No H-08 DNB and CFM margin calculations were not necessary. The HZP SLB reactivity decreased versus safety analysis limits due to the absence of the H-08 control rod and had significant margin to the limit.

- The SLB c/w RWAP parameter evaluation results with no control rod in H-08 were less limiting than the rodded H-08 evaluation. Statepoint reactivity differences resulting from the bank withdrawal decreased relative to the limit due to the absence of H-08 from the withdrawn Control Bank D. Safety analysis required MTC ranges continued to be satisfied with the removal of the H-08 control rod.

- The increase in SDM boron concentration requirements ensures the UBDA for modes 1 through 5 remains bounding for the removal of the H-08 control rod.

- Single Rod Withdrawal accident saw no change in the number of failed fuel pins in the pin census.

- Uncontrolled Control Bank Withdrawal (UCBW) at power saw reduced maximum reactivity insertion rates due to the absence of the H-08 control rod from Control Bank D and increased margins to the safety analysis limits.

- UCBW from subcritical saw the maximum reactivity insertion rate decrease due to the removal of the H-08 control rod. However, maximum radial pin power increased because the absence of the H-08 control rod allowed power to move strongly to the center of the core. The higher calculated maximum radial pin power with the H-08 control rod removed satisfied the limit.

- SDM and maximum insertable 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 14 - Impact on UFSAR Chapter 15 Accident Analyses

#	UFSAR	Description	Comments
1	15.2.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From A Subcritical Condition	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
2	15.2.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
3	15.2.3	Rod Cluster Control Assembly Misalignment	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
4	15.2.4	Uncontrolled Boron Dilution	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific

#	UFSAR	Description	Comments
			evaluations verify the AOR remains bounding.
5	15.2.5	Partial Loss of Forced Reactor Coolant Flow	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
6	15.2.6	Startup Of An Inactive Reactor Coolant Loop	Technical Specification 3.4.4 requires that all four reactor coolant pumps be operating in Modes 1 and 2; therefore, power operation with an inactive loop is precluded. This event was originally included in the UFSAR when potential operation with a loop out of service was anticipated. It remains for historical purposes but is not actively maintained.
7	15.2.7	Loss Of External Electrical Load And/Or Turbine Trip	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
8	15.2.8	Loss of Normal Feedwater	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
9	15.2.9	Loss of Off-Site Power to the Station Auxiliaries	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
10	15.2.10	Excessive Heat Removal Due to Feedwater System Malfunctions	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
11	15.2.11	Excessive Load Increase	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
12	15.2.12	Accidental Depressurization of the Reactor Coolant System	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
13	15.2.13	Accidental Depressurization of the Main Steam System	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
14	15.2.14	Spurious Operation of the Safety Injection System at Power	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
15	15.3.1	Loss of Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System	Removal of the H-08 control rod has no effect on the LOCA AOR. Cycle-specific evaluations verify the AOR remains bounding.

#	UFSAR	Description	Comments
16	15.3.2	Minor Secondary System Pipe Breaks	Bounded by UFSAR Section 15.4.2.1
17	15.3.3	Inadvertent Loading of a Fuel Assembly into an Improper Position	No impact. Inadvertent loading is detected using incore instrumentation during startup testing.
18	15.3.4	Complete Loss of Forced Reactor Coolant Flow	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
19	15.3.5	Waste Gas Decay Tank Rupture	No impact. There are no relevant analysis parameters affected.
20	15.3.6	Single Rod Cluster Control Assembly Withdrawal at Full Power	Removal of the H-08 control rod has no effect on system AOR. Cycle specific evaluations verify the AOR remains bounding.
21	15.3.7	Steam Line Break Coincident with Rod Withdrawal at Power (SLB c/w RWAP)	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
22	15.4.1	Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)	Removal of the H-08 control rod has no effect on the LOCA AOR. Cycle-specific evaluations verify the AOR remains bounding.
23	15.4.2.1	Rupture of a Main Steam Line	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
24	15.4.2.2	Major Rupture of a Main Feedwater Pipe	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
25	15.4.3	Steam Generator Tube Rupture	No impact. The NSSS response to this event is not affected by the removal of the H-08 control rod because control rods are not explicitly modeled in the analysis.
26	15.4.4	Single Reactor Coolant Pump Locked Rotor	Removal of the H-08 control rod has no effect on system AOR. Cycle-specific evaluations verify the AOR remains bounding.
27	15.4.5	Fuel Handling Accident	No impact. There are no relevant analysis parameters affected.
28	15.4.6	Rupture Of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)	Removal of the H-08 control rod has no effect on system AOR.

Impact on Operating Analysis Support

SQN2 licensing basis safety analysis methods are used to perform cycle-specific calculations in the Maneuvering Analysis (MA) and Nuclear Design Report (NDR) to support TS and COLR limits, associated Surveillance Requirements, and to generate data to support the startup and operation of the U2C24 core. Removal of the H-08 control rod from the U2C24 core design does not invalidate the methods used to develop the nuclear design models for U2C24 nor in the performance of the cycle-specific reload MA and NDR as described in the applicable methodology reports.

The reload safety evaluation methodology and computer code package (CASMO-3/NEMO) currently used are applicable to model and evaluate the as-designed/operated configuration of the plant. Cycle-specific reload evaluations of TS limits and core operating limits without the H-08 control rod for U2C24 are performed to ensure applicable safety analysis limits remain satisfied. The CASMO-3/NEMO models that calculate reactivity parameter and power distribution performance are not impacted nor invalidated due to removal of the H-08 control rod from U2C24 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 H-08 because 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.

NDR data and operational characteristics for U2C24 will be different with the H-08 control rod removed from the 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 the H-08 control rod because explicit modeling of the as-designed core configuration is employed in the generation of the NDR data. These differences will be reflected in the cycle-specific NDR to identify expected changes in core design and behavior.

The cycle-specific maneuvering analysis (MA) results showed acceptable analysis margins to power peaking limits with the current overpower delta-temperature (OPΔT) / overtemperature delta-temperature (OTΔT) trip settings, COLR Axial Flux Difference (AFD) limits, and Rod Insertion Limits with the H-08 control rod removed from the U2C24 core. No differences are expected 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 the U2C24 core. Explicit modeling of the new core configuration is used in the generation of the cycle-specific peaking factor limits.

Conclusion

The reload safety evaluations for the U2C24 with the H-08 control rod removed validated all cycle-specific safety analysis limits, and determined the UFSAR Chapter 15 accident analyses remain bounding with respect to the U2C24 safety analysis physics parameters, MA, NDR, 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

The H-08 control rod will be removed from service by performing the following work items, which will be evaluated in accordance with appropriate TVA 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 located in core location H-08
- Install a flow restriction plate in the H-08 control rod guide tube (CRGT)

- Remove H-08 control rod inputs to the Rod Position Indication (RPI) system
- Modify plant computer position indication and alarm points for the H-08 control rod
- Remove visual indications of rod position and rod bottom light for the H-08 control rod on the Main Control Room M-4 panel
- Remove rod control system fuses for control power to the H-08 CRDM

Modifications to the Rod Position Indication (RPI) system configuration will ensure that H-08 will not impact any alarms or annunciators. The Integrated Computer System computer will also be reprogrammed to account for the H-08 control rod being removed. Modifications to the RPI system and Rod Control system related to the removal of the H-08 control rod will have no impact to the ability to manipulate the remaining control rods or the ability to trip the reactor via the reactor protection system (RPS).

These changes are reviewed and approved by SQN engineering using TVA procedures for design changes.

3.5 EVALUATION OF POTENTIAL DESIGN IMPACTS

Flow Restrictor

When the H-08 control rod and driveshaft are removed from service, a flow restrictor will be installed in the H-08 control rod guide tube in the reactor vessel upper internals. The installed flow restrictor is a standard component used to hydraulically simulate the CRDM drive shaft clearance with the guide tube housing opening. This will establish hydraulically equivalent flow conditions in the upper internals when the drive shaft is removed. A generic structural analysis of the restrictor plate/orifice assembly has been performed using a bounding pressure differential load for the faulted service condition (Loss of Coolant Accident (LOCA)). This analysis conservatively assumed no orifice holes in the assembly to maximize the differential pressure load. The analysis demonstrated that all membrane and bending, bearing, and shear stress intensities satisfy the requirements of the 1989 Edition of ASME Section III. Bolting preload adequate to resist assembly separation was also demonstrated for the maximum LOCA pressure loads. The generic analysis has been reviewed and confirmed to bound SQN plant-specific service conditions.

Materials for the flow restrictor assembly conform to the ASME Code, Section II, Part A. The restrictor assembly is manufactured from 304 stainless steel, which is the same material as the guide tube, and is compatible with fluid conditions in the reactor vessel upper internals. Because the restrictor assembly and the guide tube are both the same material, there will be no differential thermal expansion.

Installation of the restrictor is controlled to ensure that the required hex bolt preload is obtained, securely locking the flow restrictor in place at the top of the guide tube. A locking cup, which is tack welded to the flow restrictor, is crimped onto the hex bolt to prevent hex bolt rotation. The capture features of the flow restrictor (i.e., locking fingers, hex bolt cup, hex bolt preload) provide assurance that the flow restrictor is securely installed and will not result in the generation of loose parts.

The reactor internals at SQN are designed and analyzed to the requirements of Section 3.9.3 of the Updated Final Safety Analysis Report (UFSAR), “NSSS Components Not Covered by the ASME Code.” The basis for the design stress and deflection criteria is summarized in Section 4.2.2.5 of the UFSAR. While the restrictor assembly does not perform a core support or safety function, it is classified as ANSI Safety Class 3. All of the calculated stresses are within the ASME Code allowables. The restrictor assembly materials, fabrication, and design analysis discussed above meet the intent of ASME Code Subsection NG consistent with the SQN design basis and UFSAR design summary.

Thermal-hydraulic impacts

Installation of the flow restrictor as described above will ensure the flow area and hydraulic resistance normally provided by the driveshaft in the guide tube will be maintained.

A bypass flow analysis was performed to determine the impact of removing the control rod in core location H-08. This analysis shows that the core bypass flow increases slightly but remains below the analyzed bounding value. Therefore, all DNB analyses remain bounding following the removal of the H-08 control rod.

In addition, the increase in the core bypass flow has the potential to affect the system transient analyses, and a disposition of events was performed for the Chapter 15 events. The bypass flow is a less significant parameter in the system analyses than it is in the DNB analyses. Framatome determined that the existing large and small break LOCA analyses remain bounding. Furthermore, the non-LOCA UFSAR Chapter 15 accident analyses continue to be applicable considering the incremental increase in bypass flow due to the removal of the H-08 control rod.

Due to the flow restrictor maintaining the thermal-hydraulic configuration of the reactor vessel upper internals, there will be no impact to rod drop times at other core locations as a result of the removal of the H-08 RCCA and associated control rod drive shaft. Therefore, Technical Specification Surveillance Requirement 3.1.4.3 will continue to be met.

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 as long as a flow restrictor is installed in its place. 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 H-08 control rod.

Other Considerations

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

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 U2C24. 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 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

The safety evaluations performed for the U2C24 H-08 RCCA removal validated that the impacts to the nuclear design parameters were within the bounds of those already assumed in the UFSAR Chapter 15 accident analyses. No new or revised operator actions were 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

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 U2C24 core has been evaluated with and without the H-08 control rod assembly per the methodologies set forth in TS 5.6.3, "Core Operating Limits Report (COLR)."

SQN2 TS 3.1.4, "Rod Group Alignment Limits," requires all shutdown and control rods to be operable. Because the control rod in location H-08 would be removed under the proposed change, this TS requirement would not be applicable to that control rod position. As such, no changes to TS 3.1.4 are required.

The requirements of 10 CFR 50.62(c) applicable to SQN2 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 SQN2, and subsections (c)(3) through (c)(5) are applicable only to boiling water reactors.

4.1.1 General Design Criteria

SQN was designed to meet the intent of the Proposed General Design Criteria (GDC) for Nuclear Power Plant Construction Permits published in July 1967. The SQN construction permit was issued in May 1970. The UFSAR, however, addresses the NRC GDCs published as Appendix A to 10 CFR 50 in July 1971. Conformance with the GDCs is described in Section 3.1.2 of the UFSAR.

Each criterion listed below is followed by a discussion of the design features and procedures that meet the intent of the criteria. Any exception to the 1971 GDCs resulting from the earlier commitments is identified in the discussion of the corresponding criterion.

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.

Compliance

The reactor core with its related coolant, control, and protection systems is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The Reactor Trip System is designed to actuate a reactor trip, when necessary, for any anticipated combination of plant conditions, to ensure that fuel design limits are not exceeded. The core design, together with reliable process and decay heat removal systems, provides 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 power.

A U2C24 redesign reload analysis was performed in accordance with the methods described in TS 5.6.3 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.

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.

Compliance

The fuel temperature coefficient is negative and the moderator temperature coefficient of reactivity is non-positive for power operating conditions, thereby providing negative reactivity feedback characteristics.

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.

<u>Criterion 12 - Suppression of Reactor Power Oscillations</u>

The reactor core and associated coolant, or, 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.

Compliance

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and non-positive moderator temperature coefficient 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 non-positive moderator temperature coefficients of reactivity.

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

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 because, as per the COLR analysis, the removal of the H-08 control rod will not result in power oscillations, which would result in conditions exceeding specified acceptable fuel design limits.

<u>Criterion 23 - Protection System Failure Modes</u>

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.

Compliance

The Protection System is designed with due 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 loss of power, disconnection, open channel faults, and the majority of internal channel short-circuit faults cause the channel to go into its tripped mode.

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.

<u>Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions</u>

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.

Compliance

Reactor shutdown by full length rod insertion is completely independent of the normal control function, because the trip breakers interrupt power to the rod mechanisms regardless of existing control signals. The Protection System is designed to limit reactivity transients so that fuel design limits are not exceeded.

In addition, the analysis presented in SQN UFSAR Chapter 15 shows that for postulated dilution during refueling, startup or manual or automatic operation at power, 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.

This criterion remains satisfied because, a Unit 2 operating cycle U2C24 redesign reload analysis, performed according to methods referenced in TS 5.6.3, 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.

<u>Criterion 26 - Reactivity Control System Redundance and Capability</u>

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.*

Compliance

Two Reactivity Control Systems are provided. These are rod cluster control assemblies (RCCA) and chemical shim (boration). The RCCA are inserted into the core by the force of gravity.

During operation the shutdown rod banks are fully withdrawn. The full length Control Rod System maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks along with the full length 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 core life is assumed in all analyses and the most reactive rod cluster is assumed to stick in out of core position.

The boron chemical shim is unaffected and will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for all 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.

<u>Criterion 27 - Combined Reactivity Control Systems Capability</u>

The Reactivity Control Systems shall be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), 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.

Compliance

Sufficient capability is provided to control reactivity for any anticipated cooldown transient, i.e., accidental opening of a steam bypass or relief valve or safety valve stuck open. This capability is achieved by a combination of RCCA and automatic boron addition via the ECCS with the most reactive control rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to supplement the RCCA in maintaining the shutdown margin for the long-term conditions of xenon decay and plant cooldown.

This criterion remains satisfied, because the removal of the H-08 control rod 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 the H-08 control rod during U2C24 demonstrate that SDM and safety analysis limits are met throughout the fuel cycle.

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.

Compliance

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 rupture of the Reactivity Control (RC) System boundary or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.

Assurance of core cooling capability following accidents, such as rod ejection, steam line break, etc., is given by keeping the reactor coolant pressure boundary stresses within faulted condition limits as specified by applicable ASME codes. Structural deformations are checked also 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, 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 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.

Compliance

The Protection and Reactivity Control Systems are designed to ensure an extremely high probability of fulfilling their intended functions. The design principles of 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.

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 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, 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 U2C24.

4.2 PRECEDENT

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

NRC letter to TVA, “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

Tennessee Valley Authority (TVA) is proposing an amendment to Sequoyah Nuclear Plant (SQN) Unit 2 Technical Specification (TS) 4.2.2, “Control Rod Assemblies,” to permit Unit 2 Cycle 24 (U2C24) to contain 52 full length control rods with no full length control rod in core location H-08. Currently, TS 4.2.2 requires 53 full length control rod assemblies 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 change involve a significant increase in the probability or consequences of an accident previously evaluated?*

   Response: No.

   Operation of SQN, Unit 2, Cycle 24 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 Updated Final Safety Analysis Report (UFSAR) 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.

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 SQN, Unit 2, Cycle 24 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 and the safety evaluations performed for U2C24 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 UFSAR Chapter 15 accident analyses. The current accident analyses remain bounding. Additionally, by installing a flow restrictor in the H-08 upper internals control rod guide tube, the hydraulic characteristics of the reactor vessel upper internals hydraulic characteristics are unchanged and all 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 change involve a significant reduction in a margin of safety?*

Response: No.

Operation of SQN, Unit 2, Cycle 24 with the H-08 control rod removed will not involve a significant reduction in a margin of safety. The margin of safety is established by setting safety limits and operating within those limits. The proposed change does not alter any UFSAR 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 the proposed change does not exceed or alter a design basis or safety limit. Therefore, the proposed change does not significantly reduce a margin of safety.

Based on the above, TVA concludes that the proposed amendment does not involve a 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.

4.4 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.

5.0 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. 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.

Enclosure

ATTACHMENT 1

Proposed TS Changes (Mark-Ups) for SQN Unit 2

4.0 DESIGN FEATURES

4.1 Site Location

The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of 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. Sequoyah is authorized to place a limited number of lead test assemblies into the reactor as described in the Framatome-Cogema Fuels report BAW-2328, beginning with the Unit 2 Operating Cycle 10 core.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium, and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

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 U-235 enrichment of 5.0 weight percent;

--NOTE--
Operation with 52 full length control rod assemblies (with no control rod assembly installed in core location H-08) is permitted during Cycle 24.

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ATTACHMENT 2

Proposed TS Changes (Final Typed) for SQN Unit 2

4.0 DESIGN FEATURES

4.1 Site Location

The Sequoyah Nuclear Plant is located on a site near the geographical center of Hamilton County, Tennessee, on a peninsula on the western shore of Chickamauga Lake at Tennessee River mile (TRM) 484.5. The Sequoyah site is approximately 7.5 miles northeast of the nearest city limit of Chattanooga, Tennessee, 14 miles west-northwest of Cleveland, Tennessee, and approximately 31 miles south-southwest of TVA's Watts Bar Nuclear Plant.

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4.2.2 Control Rod Assemblies

--NOTE--
Operation with 52 full length control rod assemblies (with no control rod assembly installed in core location H-08) is permitted during Cycle 24.

The reactor core shall contain 53 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium, and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

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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 U-235 enrichment of 5.0 weight percent;

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

b. A k_{eff} less than critical when flooded with unborated water and a k_{eff} less than or equal to 0.95 when flooded with water containing 300 ppm soluble boron. For some accident conditions, the presence of dissolved boron in the pool water may be taken into account by applying the double contingency principle which requires two unlikely, independent, concurrent events to produce a criticality accident; and

c. A nominal 8.972 inch center to center distance between fuel assemblies placed in the high density fuel storage racks.

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water;

c. $k_{eff} \leq 0.98$ under optimum moderation conditions; and

d. The arrangement of 146 storage locations shown in Figure 4.3.1.2-1. The cells shown as empty cells in Figure 4.3.1.2-1 shall have physical barriers installed to ensure that inadvertent loading of fuel assemblies into these locations does not occur.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 722 ft.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 2091 fuel assemblies. In addition, no more than 225 fuel assemblies will be stored in a rack module in the cask loading area of the cask pit.