Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

April 7, 2016
NOC-AE-16003351
10 CFR 50.90

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

South Texas Project
Unit 1
Docket No. STN 50-498
License Amendment Request to Revise Technical Specification 5.3.2 to
Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1

References:

1. Letter; G. T. Powell to NRC Document Control Desk, "Emergency License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1 Cycle 20"; December 3, 2015; (NOC-AE-15003315) (ML15343A347)
2. Letter; G. T. Powell to NRC Document Control Desk, "Response to Request for Additional Information and Supplement to South Texas Project (STP) Unit 1 Emergency License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1 Cycle 20"; December 9, 2015; (NOC-AE-15003318) (ML15344A304).
3. Letter; L. Regner to D. Koehl, "South Texas Project Unit 1 – Issuance of Amendment Re: Revision to Technical Specifications for One Operating Cycle Operation with 56 Control Rods (Emergency Circumstances)"; December 11, 2015; (AE-NOC-15002765) (ML15343A128).
4. Letter; L. Regner to STP Nuclear Operating Company, "Summary of March 2, 2016, Public Meeting With STP Nuclear Operating Company to Discuss a Proposed License Amendment to Operate South Texas Project, Unit 1, Permanently With 56 Control Rods (CAC No. MF7361)"; March 24, 2016; (AE-NOC-16002820) (ML 16081A205).

By References 1 and 2, STP Nuclear Operating Company (STPNOC) requested approval of an emergency license amendment to Technical Specification (TS) 5.3.2 to require the Unit 1 Cycle 20 core to contain 56 full-length control rods with no full-length control rod assembly in core location D-6. By Reference 3, the NRC approved the emergency license amendment request and issued Amendment 208 to South Texas Project Operating License NPF-76.

Pursuant to 10 CFR 50.90 STPNOC hereby requests a license amendment to South Texas Project Operating License NPF-76 to require the Unit 1 core to contain 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6. Currently, TS 5.3.2 requires the Unit 1 core to contain 57 full-length control rod assemblies except for the Unit 1 Cycle 20 core, which shall contain 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6. A pre-submittal meeting between the NRC Staff and STPNOC was held on March 3, 2016. The NRC concerns and questions raised in this meeting, as documented in Reference 4, have been addressed and answered in the Enclosure to this letter.

A001
NRR

STI: 34288696

STPNOC evaluated options for repairing or replacing Control Rod D-6 as an alternative to operating with Control Rod D-6 permanently removed as requested herein and concluded that the consequences and uncertainties associated with a Control Rod Drive Mechanism (CRDM) repair or replacement are significant. Repairing or replacing the D-6 CRDM would require the use of specialized remote tooling and processes that do not currently exist and would necessitate cutting and welding on the Reactor Coolant System pressure boundary.

Using current reactor core design methodologies, STPNOC can design future reactor cores with 56 control rods that remain within the criteria established in the current STPNOC Updated Final Safety Analysis Report Chapter 15 accident analyses and associated safety margins. Given these conclusions, STPNOC determined that the most prudent course of action would be to continue safely operating Unit 1 with Control Rod D-6 permanently removed.

Approval of this license amendment request is requested by March 2, 2017, prior to the start of the next Unit 1 refueling outage (1RE20), currently scheduled to begin in March 2017. Once approved, the amendment shall be implemented prior to entering Mode 5 from Mode 6 during startup from 1RE20. STPNOC is only requesting NRC approval of the proposed change to TS 5.3.2. All design changes and supporting safety analyses discussed in this document will be performed in accordance with the station design change process, core reload design process, and the current licensing basis.

The Enclosure to this letter provides a technical and regulatory evaluation of the proposed amendment. Attachment 1 to the Enclosure contains the proposed TS page markup.

The proposed amendment has been reviewed and approved by the STPNOC Plant Operations Review Committee and has undergone an independent Organizational Unit Review.

In accordance with 10 CFR 50.91, STPNOC is notifying the State of Texas of this license amendment request by transmitting a copy of this letter and Enclosure to the designated State Official.

There is one commitment in this letter provided in Attachment 2 to the Enclosure.

If there are any questions or if additional information is needed, please contact Drew Richards at (361) 972-7666 or me at (361) 972-7344.

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

Executed on April 7, 2016

James Connolly

James Connolly
Site Vice President

amr/JWC

Enclosure: Evaluation of the Proposed Change

cc:
(paper copy)

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, TX 76011-4511

Lisa M. Regner
Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North (O8H04)
11555 Rockville Pike
Rockville, MD 20852

NRC Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code: MN116
Wadsworth, TX 77483

(electronic copy)

Morgan, Lewis & Bockius LLP
Steve Frantz, Esquire

U.S. Nuclear Regulatory Commission
Lisa M. Regner

NRG South Texas LP
Chris O'Hara
Jim von Suskil
Skip Zahn

CPS Energy
Kevin Pollo
Cris Eugster
L. D. Blaylock

Crain Caton & James, P.C.
Peter Nemeth

City of Austin
Elaina Ball
John Wester

Texas Dept. of State Health Services
Helen Watkins
Robert Free

ENCLOSURE

Evaluation of the Proposed Change

Subject: **License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1**

Attachments:
1. Technical Specification Markup
2. List of Commitments

1.0 Summary description

This evaluation supports a request to amend Operating License NPF-76 for South Texas Project (STP) Unit 1 by revising Technical Specification (TS) 5.3.2, "Control Rod Assemblies," to require the Unit 1 core to contain 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6. The current TS 5.3.2 requirement is for Unit 1 to operate with 56 full-length control rod assemblies for Cycle 20 only. The Unit 2 requirement to operate with 57 full-length control rod assemblies will remain unchanged.

After evaluating the safety implications and repair options, STP Nuclear Operating Company (STPNOC) decided to permanently remove Control Rod D-6 and operate Unit 1 with 56 full-length control rods. NRC approval is required to change TS 5.3.2 to permanently require Unit 1 to contain 56 full-length control rods. Approval of the proposed amendment is requested by March 2, 2017, prior to the start of the next Unit 1 refueling outage (1RE20), currently scheduled to begin in March 2017.

2.0 Detailed description

2.1. Proposed amendment

The proposed amendment would revise TS 5.3.2, "Control Rod Assemblies," to require Unit 1 to operate with 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6, in lieu of the requirement to contain 57 full-length control rod assemblies. STPNOC is also proposing to remove the footnote, which permitted the Unit 1 Cycle 20 core to contain 56 full-length control rod assemblies with no full-length control rod assembly installed in core location D-6. STPNOC performed a thorough review of the TS and has determined that no other TS changes are required.

STPNOC is only requesting NRC approval of the proposed change to TS 5.3.2. Supporting design changes and safety analyses discussed in this document will be performed in accordance with the station design change process, core reload design process, and the current licensing basis.

2.2. Control Rod D-6 issue

On November 18, 2015, in preparation for restarting Unit 1 following refueling outage 1RE19, STPNOC performed control rod drop time surveillance testing. During this testing, Control Rod D-6 in Shutdown Bank A did not function as expected. During subsequent troubleshooting activities, Control Rod D-6 was unable to be moved using normal methods. Control Rod D-6 was later able to be moved to the bottom of the core (i.e., fully inserted). Unit 1 was subsequently cooled down to Mode 6 and the reactor head was disassembled. All 57 Unit 1 Control Rod Drive Mechanisms (CRDMs) were inspected to determine the extent of condition. It was determined that the issue with Control Rod D-6 is due to deformation of the CRDM rod holdout (RHO) ring, which is used during rapid refueling operations (see Section 3.1 of this Enclosure for a description of rod lockout and the rapid refueling feature at STP). No similar deformation has been observed on the other 56 Unit 1 CRDMs.

Previously, on November 11, 2012, during Unit 1 refueling outage 1RE17, Control Rod D-6 failed to fully insert into the core when the CRDM allowed the control rod to ratchet into the core during rod unlocking operations. While performing testing on November 12, 2012, Control Rod D-6 dropped to the bottom of the core. Following further testing and evaluation, Control Rod D-6 performed as designed; the control rod passed all surveillance testing during the following operating cycle (Cycle 18).

On March 17, 2014, during Unit 1 refueling outage 1RE18, Control Rod D-6 was unable to be locked out in preparations for a rapid refueling. Visual inspection of the Control Rod D-6 CRDM determined that the RHO ring was damaged and was moving with the stationary gripper pole, causing increased stationary gripper latch closure times. The decision was made to continue with a non-rapid refueling and the refueling outage continued without the control rods being locked out. Following further evaluation, it was determined that the rod drop function and the ability to step Control Rod D-6 was not impaired; the only affected function was the ability to lock out Control Rod D-6 to perform a rapid refueling.

On November 18, 2015, while preparing to restart Unit 1 during refueling outage 1RE19, Control Rod D-6 in Shutdown Bank A did not function as expected. Subsequent inspections showed deformation of the rod holdout ring in the CRDM for Control Rod D-6, which would require replacement of the CRDM to repair the condition. STPNOC requested an emergency license amendment (see References 6.52 and 6.53), which was approved by the NRC on December 11, 2015 (Reference 6.54).

STPNOC has determined that the STP reactor cores can be safely designed and operated for the life of the plant with 56 control rods. Replacing the D-6 CRDM would require cutting and welding on the Reactor Coolant System (RCS) pressure boundary. Therefore, STPNOC determined that the most prudent course of action would be to continue safely operating Unit 1 with Control Rod D-6 permanently removed.

Cause evaluation

The event during 1RE17 occurred due to a failure of the Rod Holdout Mode Selector (RHMS) switch during performance of the rod unlocking procedure. The switch failure was caused by surface contamination on the switch contacts. Failure of the RHMS switch resulted in Shutdown Bank A Group 2 receiving partial stepping current orders intended for a different bank in the associated rod control power cabinet. The Shutdown Bank A stationary gripper coils received stepping demands while the Shutdown Bank A movable gripper coils received no demand to energize, resulting in the Shutdown Bank A Group 2 control rods dropping into the reactor core. The simultaneous release of the Shutdown Bank A RHO ring and cycling of the stationary gripper latches as the control rods were falling caused them to ratchet into the core with the stationary gripper latches impacting the RHO ring. The impacting of the latches resulted in deformation of the RHO ring.

As a corrective action to prevent recurrence following 1RE17, STPNOC replaced the RHMS switches with newly manufactured switches and installed jumpers to bypass the contacts for stationary multiplexing signals. The procedure for rapid refueling rod holdout operation was revised to provide instructions for exercising the RHMS switches and for removing and reinstalling the jumpers as required to perform a rod holdout. Based on further issues in 1RE18 and 1RE19, STPNOC will no longer perform rapid refueling operations for Unit 1 (see Attachment 2).

3.0 Technical evaluation

3.1. System description

Unit 1 currently contains 56 full-length control rod assemblies divided into four control banks (Control Banks A, B, C, D) and five shutdown banks (Shutdown Banks A, B, C, D, E). The control banks are normally used for reactor startup and shutdown and Control Bank D is used for short-term control during normal at-power operation. The shutdown banks provide additional negative reactivity to meet shutdown margin requirements. During operation in Modes 1 and 2,

the shutdown banks are fully withdrawn from the core in accordance with TS 3.1.3.5 and as specified in the Core Operating Limits Report (COLR).

Prior to its removal for Unit 1 Cycle 20 (Reference 6.54), Control Rod D-6 was located in Shutdown Bank A and was located in the core as shown in Figure 1.

Figure 1: Control Rod Locations

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

Control Bank	Number of rods	Shutdown Bank	Number of rods
A	8	SA	**7**
B	8	SB	8
C	8	SC	4
D	5	SD	4
		SE	4

Each control rod is moved by a 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, which move the control rods within the reactor core by sequencing power to the three coils of each mechanism to produce a stepping rod motion. 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 or withdraw control rods in the core to control reactivity and provide the required shutdown margin. Mechanically, each control rod location includes a guide tube, which is an assembly that houses and guides the control rod through the upper internals.

In the STP installation, a fourth coil is installed as part of the Rod Holdout Control System, which allows STPNOC to perform rapid refueling. In a rapid refueling, all control rods are held in the fully withdrawn position by the CRDMs and the reactor vessel head and upper internals are lifted together in one polar crane lift. One of the components of the rapid refueling feature is a separate rod holdout ring below the stationary gripper latch assembly, which is positioned to hold the stationary gripper latch closed on the control rod drive after electrical power is removed. After withdrawing the control rods into the reactor vessel upper internals, the rod holdout coil is energized, the RHO ring moves up into a notch behind the closed stationary gripper latches, the stationary coil and then the rod holdout coil are de-energized, and the control rods become mechanically locked in place. After refueling, when the reactor vessel head is set on the vessel flange, the control rods are unlocked and fully inserted.

The STPNOC Updated Final Safety Analysis Report (UFSAR) refers to control rods as Rod Cluster Control Assemblies (RCCAs). Generally, "RCCA" refers to the group of individual neutron absorber rods fastened at the top end to a common spider assembly and "control rod" refers to the entire assembly including the control rod drive shaft. However, for the purposes of this submittal, the terms "control rod" and "RCCA" can be considered to be synonymous.

STPNOC varies the control rod fully withdrawn position to distribute RCCA cladding wear associated with flow-induced vibration within the reactor vessel upper internal control rod guide tubes. This position is referred to as the Full Out Position (FOP); it is controlled by the Rod Insertion Limit sections of the COLR and may vary between 249 and 259 steps withdrawn. An RCCA position at or above 255 steps corresponds to at or above the top of the active fuel region and 249 steps corresponds to a position approximately four inches inside the active fuel region. The FOP strategy is defined during the core reload design phase and the positions of the RCCAs are explicitly modeled in the cycle-specific core design analysis. When the RCCAs are positioned at an FOP above the active fuel region, the at-power core power distributions are not impacted by the removal of RCCA D-6. When the RCCAs are positioned at an FOP where rod tips are slightly inside the active fuel region, the at-power core power distribution is minimally impacted by the removal of RCCA D-6. The relative fuel assembly power increase in the D-6 fuel assembly with no control rod installed is on the order of 0.007 with FOP at 249 steps withdrawn.

3.2. Current licensing basis

As described in UFSAR Section 4.2.2.3.1, *Rod Cluster Control Assembly*, the RCCAs are divided into two categories: control and shutdown. The control groups compensate for reactivity changes due to variations in operating conditions of the reactor; i.e., power and temperature variations. Together, the control and shutdown groups provide adequate shutdown margin.

As described in UFSAR Section 4.3.2.4.12, *Rod Cluster Control Assemblies*, only full-length assemblies are employed in this reactor. The RCCAs are used for shutdown and control purposes to offset fast reactivity changes. The allowed control bank reactivity insertion is limited at full power to maintain shutdown capability. All shutdown RCCAs are fully withdrawn before withdrawal of the control banks.

The RCCA at Unit 1 core location D-6 was removed for Unit 1 Cycle 20 and TS 5.3.2 was revised by issuance of Unit 1 License Amendment 208 (Reference 6.54).

3.3. Process for evaluating plant design and core reload design impacts

Using the engineering design change process, design changes to the plant are reviewed for impacts to the UFSAR, including the Chapter 15 accidents. Per the engineering design change process, the responsible engineer would request that the Nuclear Fuel and Analysis group perform an impact review on plant design changes that may affect core reload design or UFSAR Chapters 6 and 15 safety analyses. If the change affects the UFSAR Chapters 6 and 15 safety analyses, the impacts of the change are evaluated in the core reload design process to ensure UFSAR Chapter 15 safety analyses remain bounding.

The Westinghouse Reload Safety Evaluation Methodology (WCAP-9272-P-A; Reference 6.51) is integrated into the STPNOC design change processes through the core reload design process, which governs design control activities unique to the design of reload cores. Using this process, recent or planned plant design changes are reviewed for potential impact to core parameters. The core reload design process is performed for each new fuel cycle regardless of whether there are any plant design changes that could impact the core design for that cycle.

The removal of RCCA D-6 was evaluated using both the design change process and the core reload design process. The Unit 1 Cycle 20 core was initially designed for 57 control rods. Using the design change and core reload design processes together, STPNOC was able to implement the removal of RCCA D-6 safely. The need for a flow restrictor in the D-6 guide tube was identified and implemented using the plant design change process. Using the core reload design process, a thimble plug on the D-6 fuel assembly was installed and STPNOC determined that key safety parameters were met and UFSAR Chapter 15 safety analyses remained bounding.

The engineering design change process for removal of RCCA D-6 will be used during 1RE20 for drawing updates. No additional physical changes to the reactor are required because RCCA D-6 has already been removed from the Unit 1 design. For Unit 1 Cycle 21 and future Unit 1 core reloads, STPNOC will use the core reload design process to evaluate the reactor core design (with RCCA D-6 removed) to ensure that the reactor core design remains bounded by the UFSAR Chapter 15 safety analyses.

3.4. Evaluation of physical impacts

3.4.1. Physical configuration changes

STPNOC performed the following work activities in support of removing Control Rod D-6 in December 2015:

- Control rod drive shaft unlatched from the RCCA and CRDM,
- Drive shaft completely removed from the reactor vessel,
- Flow restrictor installed on the top of the Control Rod D-6 guide tube to maintain proper reactor coolant flow in the upper internals,
- RCCA removed from the fuel assembly located in core location D-6,
- Thimble plug installed in the fuel assembly at core location D-6 to minimize changes to reactor coolant flow through the fuel assembly,
- Display card for RCCA D-6 removed from the Digital Rod Position Indication (DRPI) display,
- Plant computer point for RCCA D-6 modified, and
- Rod control system fuses for control power to the D-6 CRDM removed.

To support the permanent removal of Control-Rod D-6, STPNOC plans to perform the following work activities and physical configuration changes to support long-term operation:

- Remove thimble plug from the fuel assembly at core location D-6,
- Change the plant computer rod supervision program software to remove input from RCCA D-6 to the rod deviation alarm, and
- Spare in place the cabling for the D-6 CRDM, DRPI, and rod holdout cabling.

3.4.2. *Thermal-hydraulic impacts*

The thermal-hydraulic reactor internal vessel evaluation is not impacted by removal of the control rod drive shaft and RCCA as long as the flow restrictor at the top of the guide tube housing is installed. The core bypass flow will remain unaffected by the installation of the flow restrictor. The hydraulic equivalence between the D-6 upper guide tube with a control rod drive shaft installed versus a flow restrictor installed ensures that there will be no impact on rod drop times at other core locations and the current TS 3.1.3.4 rod drop time limits will continue to be met.

The installation of a flow restrictor at the top of the D-6 guide tube housing maintains RCS flow characteristics through the upper internals guide tube that are hydraulically equivalent to the previous RCS flow configuration with a control rod drive shaft installed. The flow restrictor assembly is a readily-available part designed so that flow entering or exiting the upper plenum will be essentially unchanged when compared to the original guide tube housing plate design with the control rod drive shaft in place. The flow restrictor assembly is a passive device and has been used successfully in other plants (e.g., Beaver Valley Units 1 and 2, DC Cook Units 1 and 2, Farley Unit 2, Salem Units 1 and 2) after a drive shaft had been removed from a guide tube location for part-length control rod deletion.

The thimble plug device that was installed in the fuel assembly at core location D-6 for Cycle 20 will be removed to reduce fuel component handling during core refueling. An evaluation was performed to show that the thimble plug removal causes a very small increase in core bypass flow, which remains bounded by the flow assumed in the safety analyses.

3.4.3. *Seismic and structural impacts*

The installed flow restrictor is structurally adequate and meets allowable ASME Code stress limits. Materials for the flow restrictor parts were designed and fabricated to meet the intent of ASME Code Subsection NG. Since the flow restrictor is not a core support structure, it is not required to conform to the ASME Boiler and Pressure Vessel Code. Material properties were taken from Section II, Part A of the Code. The flow restrictor is classified as ANSI Safety Class III.

The upper guide tube flow restrictor assembly is manufactured from stainless steel, which is the same material as the guide tube, and is compatible with fluid conditions in the reactor vessel upper head. Because the flow restrictor assembly and the guide tube are both the same material, there will be no differential thermal expansion. The installation procedure for the flow restrictor ensures that the specified 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 lock cup, hex bolt preload) provide assurance that the flow restrictor is securely installed and will not result in the generation of loose parts.

There is no potential interference between an installed guide tube flow restrictor and a thermal sleeve at core location D-6 because the guide tube funnel has a larger diameter.

The removal of the control rod drive shaft and RCCA and the installation of the flow restrictor does not impact the functionality nor the structural integrity of the reactor vessel upper internals. Therefore, there is no impact on the current reactor vessel internals analyses.

The dynamic analysis (seismic and Loss of Coolant Accident, discussed in UFSAR Section 3.9.1.4.8) of the D-6 CRDM was performed using the Reactor Equipment System Model (RESM). Review of the RESM shows that it remains valid after removal of the Control Rod D-6 drive shaft and RCCA. Removal of the control rod drive shaft reduces the overall weight of the CRDM; however, given the magnitude of the CRDM weight, the overall impact of weight reduction is negligible. Therefore, the CRDM dynamic stress evaluation due to seismic and Loss of Coolant Accident excitations in the current CRDM Design Report remains valid with removal of the Control Rod D-6 drive shaft and RCCA.

Removal of the Control Rod D-6 drive shaft and RCCA has a negligible effect on thermosiphoning inside the CRDM housing during normal operating conditions (i.e., water circulation inside the CRDM due to temperature differential between the outside and inside of the CRDM housing). Additionally, thermal transients caused by rod up and down motion, which dominate the thermal response of the CRDM, are eliminated. Since Control Rod D-6 is in a shutdown bank, the up and down motions would be minimal during operation at power. Therefore, the thermal stress evaluations in the current analysis (CRDM Design Report) and UFSAR Section 3.9.5 remain valid after the removal of the D-6 control rod drive shaft and RCCA.

3.4.4. RCS water volume impact

The change in reactor vessel metal volume due to removal of RCCA D-6 is estimated to be:

+ 0.024 gal	flow restrictor addition
- 2.08 gal	control rod drive shaft removal
- 1.87 gal	rod cluster control assembly removal
- 3.93 gal	net change in metal volume (inverse change in RCS volume)

The total RCS water volume is approximately 100,000 gallons; an increase of less than four gallons results in a change of less than 0.004%, which is considered negligible. Note that the net change in metal volume is slightly different than stated in the emergency license amendment request submitted in December 2015; this difference is due to the planned removal of the thimble plug in future cycles.

3.4.5. Reactor vessel mass impacts

Removal of the D-6 control rod drive shaft and RCCA will reduce the overall weight of the reactor vessel. The control rod drive shaft and RCCA have a combined weight of approximately 300 pounds and the weight of the reactor vessel head is approximately 350,000 pounds. The impact of the weight reduction (less than 0.09%) on the current reactor equipment system

model used in the seismic and loss of coolant accident analyses of the reactor internals is negligible (UFSAR Section 3.9.5.3).

3.4.6. *Potential long term impacts*

The following potential impacts have been reviewed and found to have a minimal impact on safety:

Corrosion product build-up in the guide tube

Based on industry operating experience and inspections of reactor internals at STP Unit 1 and Unit 2 (each with more than 25 years of operation), corrosion product build-up in the guide tube and the guide cards has not been observed and is not expected. Corrosion product build-up in the D-6 guide tube will not increase due to installation of the flow restrictor, which maintains hydraulically equivalent RCS flow, at the top of the D-6 guide tube housing (Section 3.4.2 of this Enclosure). There are other United States Westinghouse PWRs with RCCA guide tubes that do not have a control rod drive shaft installed. These plants have been in operation for many years with no negative effects related to corrosion product build-up. Several of these units later inspected and re-activated spare guide tubes and have since functioned acceptably.

Reactor vessel thermal sleeve wear

The removal of RCCA D-6 may increase the potential for thermal sleeve wear. Since there is no control rod drive shaft to limit the depth of thermal sleeve outer diameter wear against the reactor vessel head adapter insider diameter, outer thermal sleeve wear can be greater at core locations with an unrodded reactor head penetration. Outer core locations with an unrodded reactor head penetration (core rows 2 and 14 and core columns B and P, refer to Figure 1) are known to have greater thermal sleeve outer diameter wear than inner core locations with an unrodded reactor head penetration. The D-6 core location is an inner unrodded reactor head penetration so the likelihood for significant thermal sleeve wear is less. The Unit 1 replacement reactor vessel head design includes wear pads on the thermal sleeve inside the reactor head penetration. Thermal sleeve inspection frequency are not impacted by removal of RCCA D-6.

Flow induced vibration of the core exit thermocouples / conduits

The thermal hydraulic reactor internal vessel evaluation is not impacted by removal of the D-6 control rod drive shaft and RCCA with the guide tube flow restrictor installed. Since the hydraulic condition of the upper head is not changed, there is no increase in flow-induced vibration of the core exit thermocouples or their associated conduits.

Trapped air in the control rod travel housing that may increase the local concentration of oxygen before going into solution

Removal of the Control Rod D-6 drive shaft results in the potential of more air being trapped in the rod housing during filling of the RCS at the conclusion of a refueling outage. This air volume would be relatively small and would be compacted during pressurization of the RCS to normal operating conditions. Trapped air would dissolve during operation and be absorbed into the RCS. An evaluation shows that it would take approximately three years for a trapped air bubble inside the CRDM rod housing to diffuse into the trapped coolant in the CRDM. Therefore, the trapped air bubble will not significantly impact the trapped coolant in the CRDM. Also, since the chemistry environment in the CRDM is only influenced by bulk RCS coolant, the current CRDM housing stress evaluations remain valid.

Summary

Based on the above evaluation of potential long term impacts and the review of the flow restrictor design and installation procedure, no special periodic inspections are deemed necessary other than visual inspection concurrent with the STP PWR Reactor Internals Aging Management Program.

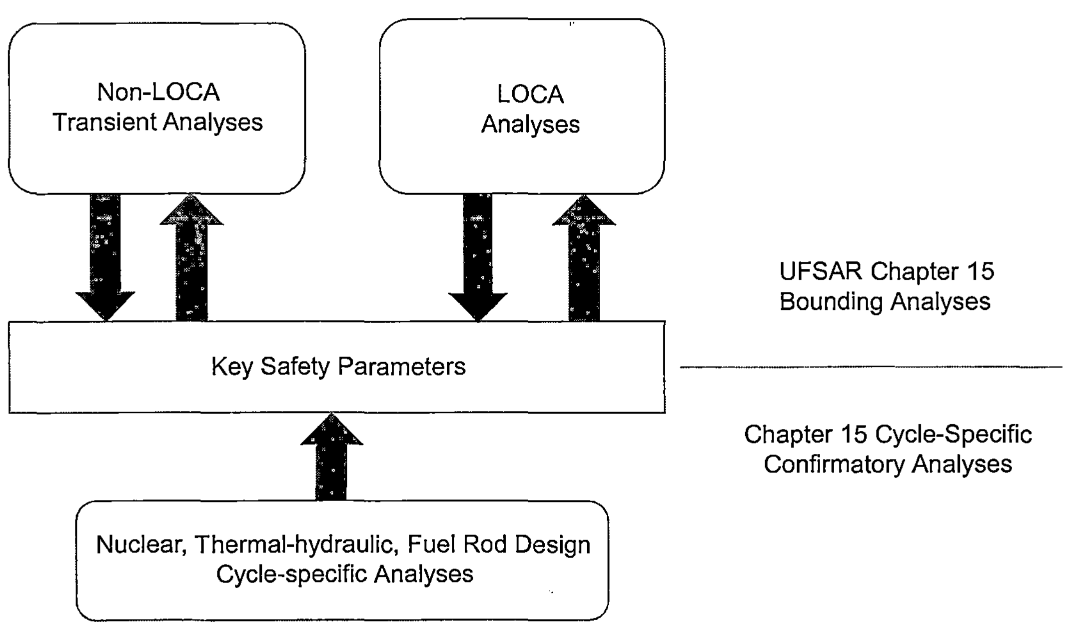
3.5. Evaluation of core reload design impacts

STPNOC uses the NRC-approved core reload design evaluation methodology described in WCAP-9272-P-A (Reference 6.51) to ensure each core reload design is acceptable. WCAP-9272-P-A was developed to use a systematic process that determines if changes associated with a reload core design impact the response characteristics of the reactor core to such an extent as to invalidate the reference safety analysis.

The WCAP-9272-P-A methodology uses bounding reference safety analyses, which are not dependent upon a minimum number of RCCAs in a core, a particular RCCA configuration, or the existence of a symmetric RCCA pattern. These types of RCCA pattern details are not included in the list of key safety parameters assumed in the bounding reference safety analysis, in which individual RCCAs are not explicitly modeled. This level of detail is explicitly accounted for each core reload design in the nuclear design calculations to confirm that the key safety parameter values assumed in the reference analyses remain bounding.

Figure 2 illustrates the relationships between the bounding analyses, the key safety parameters and the cycle-specific design analysis. The individual, cycle-specific, analyses produce parameter values that are compared to the limiting key safety parameters. The cycle-specific values must be bounded by the corresponding key safety parameters.

Figure 2: Reload Safety Analysis Methodology



3.5.1. Use of WCAP-9272-P-A methodology

The safety analysis reported in UFSAR Chapter 15 is performed using core kinetic characteristics, control rod worths, and core power distributions that historically bound most core reload designs. The WCAP-9272-P-A process identifies which of these parameters are impacted by a core reload design and identifies them as key safety parameters. The basis of selection of a parameter as a key safety parameter for a given accident is that the parameter could change as a result of core rearrangement, and if it changed, it could affect the accident consequence. The key safety parameters form the basis for determining whether the reference safety analysis applies for a reload cycle following changes in fuel assemblies and configuration[1], and performance or setpoint changes in the reactor plant systems. The values of the key safety parameters used as inputs for the reference, or bounding, safety analyses were selected to conservatively bound the values expected in subsequent cycles.

During the core reload design process, cycle-specific values for the key safety parameters are generated and compared to the bounding values used in the reference safety analyses. For each reload cycle, values of the key safety parameters are determined for the reload core during the nuclear, thermal and hydraulic, and fuel rod design processes. If all reload safety parameters for a core are conservatively bounded, the reference safety analysis is assumed to be valid and no revisions are necessary. Using this bounding analysis concept, the impact of differences from the reference core are evaluated in place of a complete new safety analysis of each reload core.

When a reload key safety parameter is not bounded, each accident, which includes the parameter, is separately evaluated to determine the impact of the deviation on the accident. If the magnitude of the effect is not easily quantifiable by the bounding evaluation, then a reanalysis is performed to ensure the required margin of safety is maintained for each affected accident. For reload cores that do not closely approximate the reference core, this bounding analysis approach greatly narrows down the areas that need to be examined to ensure the safety of the reactor core. In case the reference analysis is not bounding, the key safety parameters may play a second role in that the sensitivity of an accident to a given key safety parameter might be used to quantitatively evaluate the impact of the parameter on the safety of the reactor core.

This process is used to determine if the change in core design adversely impacts the bounding key safety parameters assumed in the UFSAR Chapter 15 safety analyses as well as to determine the effects on the reference safety analysis when a reload parameter is not bounded. In addition, the impact on Departure from Nucleate Boiling (DNB) due to the change in power distribution attributable to the new core design is also evaluated. The evaluation is documented in the reload evaluation to confirm the acceptability of the new core design.

---

[1] Core configuration includes:

- Fuel assembly parameters: enrichment, burnup, mechanical design, core location
- Core components such as: control rods, thimble plugging devices, burnable neutron absorbers, neutron sources

3.5.2. Capability of the codes used in the application of the WCAP-9272-P-A methodology

The WCAP-9272-P-A methodology is used to determine core operating limits as specified in TS 6.9.1.6b. The nuclear design analytical methods and codes used in the application of the WCAP-9272-P-A methodology are:

- ANC (WCAP-10965-P-A),
- APOLLO (WCAP-13524-P-A), and
- PARAGON/NEXUS (WCAP-16045-P-A, Addendum 1-A).

These analytical methods and codes were rigorously benchmarked and qualified for a variety of reactor types, fuel types, and burnable poison types. These computer codes have been verified for use with asymmetrical core configurations similar to the one created by the removal of RCCA D-6. The codes are applicable to this core configuration and are fully capable of assessing the nuclear characteristics for STP Unit 1 cores with RCCA D-6 removed.

3.5.3. Key safety parameters

During the generation of a reload core loading pattern, when a preliminary loading pattern that meets the required energy output is established, an evaluation is performed to ensure the following criteria are satisfied:

- The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) with all-rods-out and Control Bank D inserted to the insertion limit are below specified limits, with allowance for variation in the actual burnup of the previous cycle.
- The Moderator Temperature Coefficient (MTC) satisfies the Technical Specification requirement.
- Sufficient control rod worth is available to meet the N-1 control rods shutdown margin criterion.

Because RCCA D-6 is a shutdown bank control rod and is withdrawn to the full out position at power, its removal has a negligible impact on the core power peaking factors. The MTC and the control rod worths will be impacted as discussed below.

3.5.3.1. Core reactivity key safety parameters

The values of the core reactivity key safety parameters evaluated for a reload core depends on the burnup of the previous cycle, the number and enrichment of fresh fuel assemblies, the loading pattern of burned and fresh fuel, and the number and location of any burnable poisons. As outlined in WCAP-9272-P-A, these parameters are evaluated for each cycle-specific core loading pattern. The impacts of the removal of RCCA D-6 on these core reactivity key safety parameters are summarized in Table 1 below.

Table 1: Core reactivity key safety parameters		
Key Safety Parameter	WCAP-9272 Section	Impact from removal of RCCA D-6
MTC/Moderator Density Coefficient (MDC)	3.3.1.1	Slight impact to the most positive MDC because this parameter is conservatively calculated assuming all RCCAs are inserted into the core. A cycle-specific evaluation of the MTC/MDC values with the removal of Control Rod D-6 will be performed for each core reload design to confirm the most positive MDC remains bounding.
Fuel temperature (Doppler) coefficient	3.3.1.2	Negligible impact because RCCA D-6 would be withdrawn to the full out position during power operation, so its absence only negligibly impacts power distribution and fuel temperature.
Boron worth	3.3.1.3	No impact on the boron worth as a function of boron concentration assumed in the UFSAR 15.4.6 analysis for the Chemical and Volume Control System malfunction event.
Effective delayed neutron fraction	3.3.1.4	Potential slight increase to the fission rate in the top portion of one assembly, depending on the full out position. This increase in a low power area (the top end of the fuel assembly at core location D-6) negligibly impacts power sharing and the delayed neutron fraction.
Prompt neutron lifetime	3.3.1.5	Potential slight increase to the fission rate in the top portion of one assembly, depending on the full out position. This increase in a low power area (the top end of the fuel assembly at core location D-6) negligibly impacts power sharing and the core average prompt neutron lifetime.

In addition to the core reactivity parameters and coefficients discussed in WCAP-9272-P-A, the removal of RCCA D-6 also impacts the available shutdown margin. Maintaining the shutdown margin within the limits specified by TS 3.1.1.1 ensures the safety analysis described in UFSAR Chapter 15 remains bounding. Shutdown margin is an input to rod ejection, main steam line break, and dilution events. The impact of the removal of RCCA D-6 on shutdown margin is summarized below in Table 2.

Table 2: Additional core reactivity key safety parameter		
Key Safety Parameter	UFSAR Section	Impact from removal of RCCA D-6
Shutdown margin	15.4.6	In order to meet the COLR shutdown margin limits, the required RCS shutdown margin boron concentrations for Operating Modes 3, 4, and 5 will be higher with Control Rod D-6 removed.

3.5.3.2. <u>Control rod worth key safety parameters</u>

The shutdown banks are withdrawn from the core during power operation. Since RCCA D-6 was in a shutdown bank and would be withdrawn to the full out position during power operation, its

presence would have a negligible effect on the overall core power distribution and its removal will similarly have a negligible effect on the power distribution.

Reload core designs typically result in changes in individual RCCA worths and control and shutdown bank worths from cycle to cycle. These changes can be attributed to differences in the neutron flux distribution (thus, reactivity importance) resulting from the unique loading pattern of burned and fresh fuel assemblies for each core and the fuel depletion occurring during the reload fuel cycle. Changes in control rod worths may also affect rod insertion allowance, trip reactivity, differential rod worths, shutdown margin, and shutdown control rod worth. Table 3.3 of WCAP-9272-P-A lists the limiting directions for each of the control rod worth parameters that are evaluated for each reload core to confirm the UFSAR Chapter 15 analyses remain bounding.

The impacts of the removal of RCCA D-6 on the control rod worth key safety parameters are summarized in Table 3 below.

Table 3: Control rod worth key safety parameters

Key Safety Parameter	WCAP-9272 Section	Impact from removal of RCCA D-6
Insertion limits	3.3.2.1	No impact because RCCA D-6 would be withdrawn to the full out position during power operation.
Total rod worth	3.3.2.2	Reduction of the total rod worth – this key safety parameter is evaluated on a cycle-specific basis to ensure shutdown margin and trip reactivity limits are met.
Trip reactivity	3.3.2.3	Reduction of 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 -- a cycle-specific evaluation is performed to confirm the trip reactivity remains bounded by the key safety parameter value.
Differential rod worths	3.3.2.4	No impact because RCCA D-6 was in a shutdown bank and this parameter is calculated for control banks.

3.5.3.3. Nuclear design key safety parameters for specific events

WCAP-9272-P-A, Table 3.4, lists each event and the key safety parameters that are evaluated for each event. Table 4 below summarizes the effects of the removal of RCCA D-6 on the nuclear design key safety parameters for the listed events.

Table 4: Nuclear design key safety parameters			
Event	Key Safety Parameter	WCAP-9272 Section	Impact from removal of RCCA D-6
Loss of coolant accident	Heat flux hot channel factor (F_Q)	3.3.3.1	Negligible impact to core hot channel factor parameters and axial power distributions because RCCA D-6 was in a shutdown bank and would have been fully withdrawn during power operation.
Uncontrolled boron dilution accident	Maximum critical boron concentration with N-1 control rods inserted	3.3.3.2	Increases the N-1 critical boron concentration. The removal of RCCA D-6 effectively creates an N-2 situation. This key safety parameter is confirmed for each fuel cycle to ensure the UFSAR analyses remain bounding.
Single RCCA events: • Single RCCA withdrawal during full power operations • Statically misaligned RCCA during power operations • Dropped RCCA during full power operations	Maximum Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	3.3.3.3	Negligible impact because RCCA D-6 was in a shutdown bank and would have been fully withdrawn during power operation.
Dropped bank during full power operations	Maximum Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)	3.3.3.3	Decreased worth for Shutdown Bank A. This key safety parameter is confirmed for each fuel cycle to ensure the UFSAR analysis remains bounding.
Control rod ejection accident	Maximum ejected rod worth Heat flux hot channel factor (F_Q) at hot zero power (HZP) and hot full power (HFP)	3.3.3.4	No impact on analysis of fuel rods in DNB analysis because RCCA D-6 was in a shutdown bank and the ejected rod is assumed to be in a control bank

Table 4: Nuclear design key safety parameters			
Event	Key Safety Parameter	WCAP-9272 Section	Impact from removal of RCCA D-6
Steam line break accident	Core reactivity insertion and core power distribution (e.g., shutdown margin; MTC/MDC; power feedback and trip reactivity; power peaking factors)	3.3.3.5	Increase in post-trip core reactivity due to the removal of RCCA D-6. The power distribution at HZP with all rods in is also impacted. These key safety parameter items are confirmed for each core reload design to ensure the UFSAR Chapter 15 safety analysis remains bounding.

In addition to the limits discussed in WCAP-9272-P-A, a limit is placed on the effective neutron multiplication factor (k_{eff}) following a control rod ejection event. The safety limit for k_{eff} at HZP for RCCA Ejection is less than 0.999 to ensure that the reactor can be brought to a subcritical condition following a reactor trip from an RCCA ejection at either HFP or HZP (HZP is limiting). The rod ejection safety analysis assumes that, in addition to the ejected rod, an additional rod adjacent to the ejected rod is stuck out of the core (N-2 event).

A calculation is performed to confirm that the core will not return critical during a steam generator tube rupture event with a stuck open steam generator power operated relief valve. The calculation determines the critical boron concentration given a bounding temperature for the event in an N-1 configuration. The calculated boron concentration is then compared to the RCS boron concentration determined by the reference analysis. Removal of RCCA D-6 increases the core reactivity in the calculation.

Table 5 provides a summary of the impacts of the removal of RCCA D-6 on these additional nuclear design key safety parameters.

Table 5: Additional nuclear design key safety parameter		
Event	Key Safety Parameter	Impact from removal of RCCA D-6
Control rod ejection accident	Effective neutron multiplication factor	Removal of RCCA D-6 effectively creates an N-3 situation. This key safety parameter is confirmed for each fuel cycle. Based on a multi-cycle margin assessment, there will be sufficient margin to meet the limit in future core reload designs with RCCA D-6 removed.
Steam generator tube rupture	Boron concentration at the limiting (N-1) condition	Increases the N-1 critical boron concentration. The removal of RCCA D-6 effectively creates an N-2 situation. This key safety parameter is confirmed for each fuel cycle to ensure the UFSAR analysis remains bounding.

3.5.3.4. <u>Thermal and hydraulic analyses key safety parameters</u>

The thermal and hydraulic analyses are performed to assess the impact of the reload core on the design basis acceptance limits. The removal of Control Rod D-6 and the removal of the thimble plug previously installed at core location D-6 are discussed below.

The hydraulic evaluation of the reload core requires a review of the new fuel assembly design (nozzles, grids, fuel rods, etc.) to be inserted in to the core. This design is compared with the design of the fuel assemblies remaining in the core to ensure the new fuel assemblies are hydraulically compatible with the fuel assemblies remaining in the core. As discussed in Section 3.4.2 of this Enclosure, removal of the thimble plug from the D-6 core location will have a minimal impact on the hydraulic characteristics of the core and the DNB limits will not be exceeded. Therefore, the removal of the thimble plug from the fuel assembly in this location will not result in a violation of any safety limits for reload cores.

The effects of the removal of Control Rod D-6 on the thermal and hydraulic key safety parameters are summarized in Table 6 below.

Table 6: Thermal and hydraulic analysis key safety parameters

Key Safety Parameter	WCAP-9272 Section	Impact from removal of RCCA D-6
Engineering hot channel factors	4.3.1	No impact to heat flux engineering hot channel factor (F_{Q}^{E}) and the enthalpy rise engineering hot channel factor ($F_{\Delta H}^{E}$) because RCCA D-6 removal does not impact fuel pellet physical characteristics and because RCCA D-6 was in a shutdown bank and its removal has a negligible impact on at-power core power distributions.
Axial fuel stack shrinkage	4.3.2	No impact because removal of RCCA D-6 does not impact the physical design of the fuel.
Fuel temperatures	4.3.3	No impact on the physical fuel design parameters and negligible impact on the at-power core power distributions assumed in the calculation of fuel temperatures.
Rod internal pressure	4.3.4	No impact because the removal of RCCA D-6 does not affect the physical design of the fuel and its removal has a negligible impact on the at-power core power distributions assumed in the calculation of rod internal pressure.
Core limit lines	4.3.5	No impact because the removal of RCCA D-6 has a negligible impact on the parameters presented in Table 4-2 of WCAP-9272-P-A.

3.5.3.5. <u>DNB analysis for specific events</u>

For the reference analyses, a sub-channel thermal hydraulic analysis using THINC or VIPRE thermal-hydraulic computer codes is performed using bounding power shapes to ensure the Departure from Nucleate Boiling Ratio (DNBR) stays above the acceptance limit for the following events:

- Zero and full power steam line break (UFSAR Section 15.1.5),
- Loss of forced RCS flow (UFSAR Section 15.3.2),
- Uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition (UFSAR Section 15.4.1),
- Control rod out of position (UFSAR Section 15.4.3),

- Dropped RCCA (UFSAR Section 15.4.3), and
- RCCA ejection accidents (UFSAR Section 15.4.8).

With the exception of the steam line break event, the cycle-specific power shapes are compared to the bounding power shapes to ensure they remain bounded. If the cycle-specific event is no longer bounding, an evaluation that may use a sub-channel thermal hydraulic code is performed to ensure the DNBR acceptance limit is satisfied. For the steam line break event, a thermal hydraulic analysis using VIPRE and cycle-specific power shapes is performed for each core reload design to ensure the DNB limits are satisfied.

3.5.4. UFSAR Chapter 15 safety analysis

The reference analysis approach is fundamental to the reload evaluation process. The process determines if a reload is bounded by existing safety analyses in order to confirm applicable safety criteria are satisfied. The methodology systematically identifies parameter changes that may invalidate existing safety analyses and identifies postulated accidents that need to be reevaluated.

The re-evaluation may be of two types. If the parameter is only slightly out of bounds, or the transient is relatively insensitive to the parameter, a simple quantitative evaluation may be made, which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large or is expected to have a more significant or not easily quantifiable effect on the accident, a re-analysis of the accident is performed. If the accident is re-analyzed, the analysis methods follow standard procedures and will employ analytical methods that have been used in previous submittals to the NRC. These are the methods that have been presented in the UFSAR. The re-analyzed accident must continue to meet the appropriate safety limit for the event in order to be considered to have acceptable results.

For the UFSAR Chapter 15 bounding analyses, the number of control rods or control rod pattern are not used as inputs. Instead, these parameters are inputs to the cycle-specific core reload design analyses. The resulting cycle-specific values for the key safety parameters are then compared to the limiting values for the key safety parameters to ensure the UFSAR Chapter 15 analyses remain bounding. Therefore, a change in the number of control rods or control rod pattern does not necessarily cause a revision of the UFSAR Chapter 15 bounding analyses.

The impact of the removal of RCCA D-6 on the UFSAR Chapter 15 safety analyses is summarized in Table 7 below. A similar table was provided in the emergency license amendment request submitted in December 2015. Since this license amendment request applies to all future Unit 1 core reload designs versus one specific core (Unit 1 Cycle 20), there are necessary differences between this table and the one provided in December 2015. A discussion of the impact on key safety parameters by the removal of RCCA D-6 is provided in Section 3.5.3 of this Enclosure.

Table 7: Impact on UFSAR Chapter 15 safety analysis

#	UFSAR / WCAP	Description	Comments
1	**15.1.1** 5.3.9	Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature	Bounded by UFSAR Section 15.1.2 based on UFSAR stating this transient is less severe than the transients analyzed in UFSAR Sections 15.1.2 and 15.1.3. Based on the evaluations below for UFSAR Sections 15.1.2 and 15.1.3, the permanent removal of RCCA D-6 does not alter the conclusion of UFSAR Section 15.1.1.
2	**15.1.2** 5.3.9	Feedwater System Malfunctions Causing an Increase in Feedwater Flow	Most positive MDC and trip reactivity are potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
3	**15.1.3** 5.3.9	Excessive Increase in Secondary Steam Flow	No impact. Core DNB limits are not challenged by this event.
4	**15.1.4** 5.3.14	Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System	Bounded by UFSAR Section 15.1.5 based on UFSAR Section 15.1.4 stating this transient is less severe than the double-ended steam line rupture transient analyzed in UFSAR Section 15.1.5. Based on the evaluation below for UFSAR Section 15.1.5, the permanent removal of RCCA D-6 does not alter the conclusion of UFSAR Section 15.1.4.
5	**15.1.5** 5.3.14	Spectrum of Steam System Piping Failures Inside and Outside Containment	Shutdown margin and power distribution impacted during the return to power. DNB is impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
6	**15.2.1** N/A	Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow	Not applicable for STP
7	**15.2.2** 5.3.7	Loss of External Electrical Load	Bounded by UFSAR Section 15.2.3 based on UFSAR Section 15.2.2.1 stating the loss of load event is less severe than the turbine trip event. UFSAR Section 15.2.2.2 refers to the turbine trip for the method of analysis.
8	**15.2.3** 5.3.7	Turbine Trip	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
9	**15.2.4** N/A	Inadvertent Closure of Main Steam Isolation Valves	Bounded by UFSAR Section 15.2.5 based on UFSAR Section 15.2.4 stating the inadvertent closure of main steam isolation valves event would cause a turbine trip and is described in UFSAR Section 15.2.5.

Table 7: Impact on UFSAR Chapter 15 safety analysis

#	UFSAR / WCAP	Description	Comments
10	**15.2.5** N/A	Loss of Condenser Vacuum and Other Events Causing Turbine Trip	Bounded by UFSAR Section 15.2.3 based on UFSAR Section 15.2.5 stating the conclusions in UFSAR Section 15.2.3 apply to the loss of condenser vacuum.
11	**15.2.6** 5.3.8	Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power)	Bounded by UFSAR Section 15.3.2 for DNB and UFSAR Section 15.2.7 for pressurizer overfill based on statements in the UFSAR Section 15.2.6.
12	**15.2.7** 5.3.8	Loss of Normal Feedwater Flow	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
13	**15.2.8** 5.3.15	Feedwater System Pipe Break	Trip reactivity and most positive MDC potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
14	**15.3.1** 5.3.5	Partial Loss of Forced Reactor Coolant Flow	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety analysis parameters are not impacted. Bounded by UFSAR Section 15.3.2 based on statements in UFSAR Section 15.3.1.2.
15	**15.3.2** 5.3.5	Complete Loss of Forced Reactor Coolant Flow	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety analysis parameters are not impacted.
16	**15.3.3** 5.3.16	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted
17	**15.3.4** N/A	Reactor Coolant Pump Shaft Break	Bounded by UFSAR Section 15.3.3 based on UFSAR Section 15.3.4 stating results are no worse than the locked rotor event.
18	**15.4.1** 5.3.1	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
19	**15.4.2** 5.3.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Most positive MDC and trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
20	**15.4.3** 5.3.3 5.3.12	Rod Cluster Control Assembly Misoperation	Key safety parameters are not impacted.

Table 7: Impact on UFSAR Chapter 15 safety analysis

#	UFSAR / WCAP	Description	Comments
21	**15.4.4** 5.3.6	Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
22	**15.4.6** 5.3.4	Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant	Boron concentration and shutdown margin for return to criticality potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
23	**15.4.7** N/A	Inadvertent Loading of a Fuel Assembly into an Improper Position	No impact. Inadvertent loading is detected using incore instrumentation when the shutdown banks are withdrawn.
24	**15.4.8** various	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Trip reactivity and N-2 K_{eff} potentially impacted and will be evaluated for each core reload design. No other key safety parameters impacted.
25	**15.5.1** 5.3.11	Inadvertent Operation of ECCS During Power Operation	No impact. UFSAR Section 15.5.1 concludes "Spurious SI without immediate reactor trip has no effect on the RCS."
26	**15.5.2** N/A	Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Event is used to determine operator action sufficient to preclude pressurizer overfill. Event is not sensitive to core parameters.
27	**15.6.1** 5.3.10	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Trip reactivity potentially impacted and will be evaluated for each core reload design. All other key safety parameters are not impacted.
28	**15.6.2** N/A	Failure of Small Lines Carrying Primary Coolant Outside Containment	Event not sensitive to core parameters. Analysis parameters are not impacted.
29	**15.6.3** N/A	Steam Generator Tube Rupture	Boron concentration at the limiting condition is impacted and will be evaluated for each core reload design.
30	**15.6.5** 5.3.13	Loss of Coolant Accidents	Analysis parameters are not impacted.
31	**15.7** N/A	Radioactive Release From a Subsystem or Component	Analysis parameters are not impacted.

Table 7: Impact on UFSAR Chapter 15 safety analysis

#	UFSAR / WCAP	Description	Comments
32	**15.8** N/A	Anticipated Transients Without Scram	The generic nature of the Anticipated Transients Without Scram (ATWS) analysis reported in UFSAR Section 15.8 means there are no plant-specific inputs needing to be checked on a reload core design basis. Therefore the permanent removal of RCCA D-6 does not affect inputs used in the generic ATWS analysis.
Note: UFSAR Sections 15.4.5 and 15.6.4 apply to boiling water reactors and are not applicable to STP.			

3.6. Multi-cycle margin assessment

STPNOC performed multi-cycle margin assessment to determine the effect removal of RCCA D-6 would have had on previous reactor core designs for STP Unit 1. To evaluate the impact of only 56 control rods on core design, the four most recent Unit 1 core designs (Unit 1 Cycles 17, 18, 19, and 20) were re-analyzed with only 56 control rods. This assessment supports the conclusion that STPNOC can safely design reactor cores with RCCA D-6 removed.

STP Unit 1 operates on a nominal eighteen-month fuel cycle, varying between about seventeen and nineteen months. Accordingly, reactor cores are typically designed for lengths of about 480 to 520 effective full power days. The four most recent Unit 1 core designs are typical core designs and are indicative of future core designs.

The results of this assessment indicate that removal of RCCA D-6 generally reduces the margin to the limiting values of the key safety parameters. However, this does not reduce the margin between the bounding UFSAR Chapter 15 analyses and respective safety limits. The cycle-specific impact of RCCA D-6 removal is shown to be manageable and the results show that future core reloads can be safely designed.

3.6.1. Multi-cycle margin assessment of shutdown margin

A summary of the shutdown margin calculated at End of Cycle (EOC) for the four representative cycles (performed for the limiting HZP main steam line break accident) is provided in Table 8 below.

The key safety parameter limit for shutdown margin reactivity is 1.3% $\Delta\rho$. Based on this assessment, STPNOC anticipates there will be sufficient margin to meet the shutdown margin limit with RCCA D-6 removed and there would be no impact on STP Unit 1 core design strategies.

Table 8: Summary of shutdown margin at EOC								
	Unit 1 Cycle 17		Unit 1 Cycle 18		Unit 1 Cycle 19		Unit 1 Cycle 20	
	D-6 present	D-6 removed	D-6 present	D-6 removed	D-6 present	D-6 removed	D-6 present	D-6 removed
Control Rod Worth (%Δρ)								
All Rods Inserted minus Worst Stuck Rod (N-1)	7.48	7.00	7.10	6.72	6.94	6.60	7.10	6.81
Less 10%	6.73	6.30	6.39	6.05	6.25	5.94	6.39	6.13
Control Rod Requirements (%Δρ)								
Reactivity defects	3.67	3.67	3.61	3.61	3.60	3.60	3.60	3.60
Rod insertion allowance	0.37	0.37	0.37	0.37	0.39	0.39	0.37	0.37
Total requirements	4.04	4.04	3.98	3.98	3.99	3.99	3.97	3.97
Shutdown margin (%Δρ)	2.69	2.26	2.41	2.07	2.26	1.95	2.42	2.16
Safety analysis limit (%Δρ)	1.30	1.30	1.30	1.30	1.30	1.30	1.30	1.30

3.6.2. Multi-cycle margin assessment of N-2 subcriticality

The safety limit for k_{eff} at HZP for RCCA ejection is less than 0.999 to ensure the reactor can be brought to a subcritical condition following a reactor trip from an RCCA ejection at either HFP or HZP. The N-2 rod worth is the sum of the ejected rod worth plus the highest adjacent rod worth. The key safety parameter limit for N-2 subcriticality is 0.999 and is met by confirming the N-2 rod worth is less than the actual shutdown margin (SDM) plus 90% of the worst stuck rod worth.

Table 9 provides a summary of N-2 subcriticality calculated at EOC (limiting case) for the four representative cycles. Note that 1 pcm = 1 percent millirho = 1.0×10^{-5} $\Delta\rho$; MWD/MTU is megawatt-days per metric ton uranium.

Table 9: Summary of N-2 subcriticality at EOC					
Unit 1 Cycle	Burnup (MWD/MTU)	RCCA Condition	N-2 Rod Worth (pcm)	Confirming Actual SDM & Worst Stuck Rod (pcm)	Margin (%)
17	20400	D-6 present	2342	3529	34
		D-6 removed	2928	3292	11
18	19450	D-6 present	2273	3301	31
		D-6 removed	2815	3053	8
19	19400	D-6 present	2337	3187	27
		D-6 removed	2747	2937	6
20	18610	D-6 present	2175	3308	34
		D-6 removed	2659	3096	14

The four representative Unit 1 cycles meet the limit with and without RCCA D-6 present. The smallest margin to the N-2 subcriticality limit is 6% and would have occurred for Cycle 19 with RCCA D-6 removed. The average margin to the N-2 subcriticality limit with RCCA D-6 present is 31.5% and is reduced to 10% with RCCA D-6 removed.

3.6.3. Multi-cycle margin assessment of trip reactivity following rod ejection

Table 10 provides a summary of the HZP trip reactivity at EOC (limiting case) following a rod ejection for the four representative cycles.

The HZP trip reactivity key safety parameter value used in the rod ejection analysis is 2.000% $\Delta\rho$. With RCCA D-6 present, there is margin ranging from 49% to 66% with an average of 58.5%. With RCCA D-6 removed, the margin ranges from 22% to 31% with an average of 27%.

Table 10: Summary of HZP trip reactivity following rod ejection at EOC				
Unit 1 Cycle	Burnup (MWD/MTU)	RCCA Condition	HZP Trip Reactivity (pcm)	Margin (%)
17	20400	D-6 present	3317	66
		D-6 removed	2605	30
18	19450	D-6 present	3145	57
		D-6 removed	2504	25
19	19400	D-6 present	2983	49
		D-6 removed	2449	22
20	18610	D-6 present	3240	62
		D-6 removed	2630	31

Table 11 provides a summary of the HFP trip reactivity at Beginning of Cycle (BOC) and EOC following a rod ejection for the four representative cycles.

The HFP trip reactivity key safety parameter value used in the rod ejection analysis is 4.000% $\Delta\rho$. With RCCA D-6 present, there is margin ranging from 26% to 36% and with RCCA D-6 removed, the margin ranges from 12% to 21%.

Table 11: Summary of HFP trip reactivity following rod ejection at BOC and EOC

Unit 1 Cycle	RCCA Condition	HFP Trip Reactivity at BOC (pcm)	Margin (%)	HFP Trip Reactivity at EOC (pcm)	Margin (%)
17	D-6 present	5211	30	5433	36
	D-6 removed	4716	18	4854	21
18	D-6 present	5037	26	5268	32
	D-6 removed	4543	14	4719	18
19	D-6 present	5071	27	5038	26
	D-6 removed	4574	14	4489	12
20	D-6 present	5112	28	5256	31
	D-6 removed	4625	16	4663	17

3.7. Impact on operator actions

There is currently no impact to operator actions or emergency operating procedures (EOPs) resulting from the removal of Control Rod D-6. A cycle-specific evaluation will be performed as part of the core reload design process to determine if changes are required to EOP emergency boration values for a stuck rod. There are no changes required to the UFSAR Chapter 15 accident analysis inputs to the EOPs and no new operator actions are created.

4.0 Regulatory evaluation

4.1. Applicable regulatory requirements and criteria

Technical Specification 5.3.2, Control Rod Assemblies

The *Control Rod Assemblies* section of TS is a Design Feature required per 10 CFR 50.36(c)(4). The proposed change does not eliminate the design feature, which is the full-length control rod assemblies, but the required number of control rod assemblies is changed. As outlined in Section 3.0 of this Enclosure, all safety analysis limits will be confirmed to be bounding on a cycle-specific basis using the methodologies prescribed in TS 6.9.1.6.

10 CFR 50, Appendix A, General Design Criteria

Criterion 4—Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

The removal of Control Rod D-6 from the reactor vessel does not impact the response to accidents involving missiles or pipe breaks since the reactor shutdown function of the control rod system remains acceptable with 56 control rods. The design of the reactor vessel, reactor internals, and fuel assemblies to withstand the effects of missiles and pipe breaks is not impacted by the removal of Control Rod D-6 since there is negligible impact to the thermal hydraulics of the reactor vessel and the internals. Also, there is no adverse impact to the physical design of the reactor vessel and the internals due to the removal of the control rod drive shaft and RCCA and the installation of the guide tube flow restrictor.

Thus, the requirements of General Design Criteria (GDC) 4 are met with respect to the design of the system against the adverse effects of missile hazards inside the containment, pipe whipping and jets caused by broken pipes, and adverse environmental conditions resulting from high- and moderate-energy pipe breaks during normal plant operations, anticipated operational occurrences, and accident conditions.

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.

This criterion is satisfied because the removal of Control Rod D-6 negligibly impacts the at-power core power distribution, shutdown margin is maintained, and the design and safety limits for the UFSAR Chapter 15 accidents remain satisfied.

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.

This criterion is satisfied because removal of Control Rod D-6 does not impact the ability to detect or control core power distribution and the at-power nuclear reactivity feedback coefficients (e.g., Doppler or Moderator Temperature Coefficients) remain unchanged because the effect of Control Rod D-6 removal is similar to the original condition with Control Rod D-6 fully withdrawn from the reactor core during power operations.

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

Since Control Rod D-6 was located in a shutdown bank that is withdrawn at power, the removal of this control rod will not result in power oscillations that can result in conditions exceeding specified acceptable fuel design limits. Also, detection and suppression of reactor power oscillations is not impacted since Control Rod D-6 was in a shutdown bank and would have been fully withdrawn from the reactor core at power.

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

The removal of Control Rod D-6 from the reactor vessel does not impact the fail-safe function of the remaining 56 control rods, which will reliably maintain an adequate reactor protection system. The mechanical removal of the control rod drive shaft and RCCA and installation of the

flow restrictor in the guide tube do not have any mechanical impact on the function of the remaining 56 control rods. The electrical removal from service of Control Rod D-6 will involve removing control power to the respective stationary, lift, and movable coils. The remaining 56 control rods are not impacted by this electrical change and will continue to meet their design function. The STPNOC design change processes ensure that the plant modifications involve only Control Rod D-6 and do not affect other control rods.

Thus, the requirements of GDC 23 are met by maintaining the capability to insert the control rods upon failure of the drive mechanisms or induced failure by an outside force (e.g., loss of electric power, instrumentation air, fire, radiation, extreme heat, pressure, cold, water, and steam).

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Fuel design limits will be confirmed to be bounding on a cycle-specific basis using the NRC-approved methodologies described in TS 6.9.1.6. The protection system (reactor trip function) remains fully capable of performing its function with 56 control rods and fuel design limits are not exceeded for analyzed malfunctions of the STP reactivity control systems.

Thus, the requirements of GDC 25 are met by ensuring no fuel design limits are exceeded for any single malfunction or rod withdrawal accident.

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

This criterion is satisfied because removal of Control Rod D-6 does not impact the ability of the reactivity control systems to reliably control reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions, such as a single stuck rod, specified acceptable fuel design limits are not exceeded. Shutdown margin and safety analysis limits will be confirmed to be bounding throughout the fuel cycle using the NRC-approved methodologies described in TS 6.9.1.6. The reactivity control systems (e.g., rod control, reactor trip, RCS boron addition) continue to perform their design and safety functions with removal of Control Rod D-6.

Thus, the requirements of GDC 26 are met by demonstrating the ability to control reactivity changes to ensure that, under normal operation and anticipated operational occurrences with the appropriate margin for malfunction (such as stuck rods), no fuel design limits are exceeded and the reactor can be maintained subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under

postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

This criterion is satisfied because the removal of Control Rod D-6 does not impact the ability of the reactivity control systems to reliably control reactivity changes and adequate shutdown margin is maintained when considering highest stuck rod worth. Shutdown margin and safety analysis limits will be confirmed to be bounding throughout the fuel cycle using the NRC-approved methodologies described in TS 6.9.1.6.

Thus, the requirements of GDC 27 are met by demonstrating the ability to reliably control reactivity changes under accident conditions to ensure no fuel design limits are exceeded and the capability to cool the core is maintained.

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.

This criterion is satisfied because the removal of Control Rod D-6 will be evaluated for each core reload design to ensure trip reactivity insertion rate, shutdown margin, and the safety analysis limits remain bounding for the rod ejection, rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition events for the entire fuel cycle.

Thus, the requirements of GDC 28 are satisfied by demonstrating the ability to reliably control the amount and rate of reactivity change to ensure no reactivity accident will damage the reactor coolant pressure boundary or disturb the core or the core appurtenances such as to impair coolant flow.

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.

The removal of Control Rod D-6 from the reactor vessel does not impact the ability of the reactivity control systems to reliably control reactivity changes and perform their safety functions. The mechanical removal of the control rod drive shaft and the RCCA and the installation of the flow restrictor in the guide tube do not have any mechanical impact on the function of the remaining 56 control rods. The electrical removal from service of Control Rod D-6 will involve removing control power to the respective stationary, lift, and movable coils. The remaining 56 control rods are not impacted by this electrical change and will continue to meet their design and safety functions. The modification to remove Control Rod D-6 from service will not adversely impact the capability of the remaining 56 control rods to reliably meet their safety functions in the event of anticipated operational occurrences.

Thus, the requirements of GDC 29 are met by demonstrating a high probability of control rod insertion under anticipated operational occurrences.

Other

The requirements of 10 CFR 50.62(c) concerning an alternate rod injection system as stated in Standard Review Plan 4.6 were also assessed.

Removal of Control Rod D-6 does not impact either the reactor protection system or ATWS Mitigation System Actuation Circuitry because the trip reactivity remains bounding. The changes to other parameters described in the license amendment request do not impact the ATWS analysis. The all-rods-out MTC is not impacted by removal of Control Rod D-6. Therefore, the requirements of 10 CFR 50.62(c)(1) continue to be met and there is no impact to the ATWS analysis.

The requirements of 10 CFR 50.62(c)(2) are not applicable since STP is a Westinghouse pressurized water reactor. Also, 10 CFR 50.62(c)(3), (4), and (5) are applicable to boiling water reactors and therefore not applicable to STP.

4.2. Precedence

A similar request for a Technical Specification change to allow the removal of the center control rod assembly for Arkansas Nuclear One - Unit 1 (ANO-1) (Reference 6.49) was approved by the NRC (Reference 6.50) as Amendment Number 103. The STPNOC proposed license amendment is not for removal of a center control rod; however, the precedence does evaluate the impact on shutdown margin, reactor core thermal-hydraulics, and structural integrity of the RCS.

The NRC approved (Reference 6.54) a similar license amendment request (References 6.52 and 6.53) for STP Unit 1 for operation during Cycle 20 only (Amendment Number 208). The proposed license amendment is similar with the exception that the proposed amendment would be a permanent change for Unit 1 operation beyond Cycle 20.

4.3. No significant hazards consideration determination

STP Nuclear Operating Company (STPNOC) is proposing an amendment to Unit 1 Technical Specification (TS) 5.3.2, *Control Rod Assemblies*, to require Unit 1 to contain 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6. Currently, TS 5.3.2 requires the Unit 1 core to contain 57 full-length control rod assemblies except for Unit 1 Cycle 20, which requires 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6.

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

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

   Response: No. STPNOC has performed a multi-cycle assessment on previous Unit 1 reactor cores and evaluated the consequences associated with removal of Control Rod D-6. The assessment indicates that removal of Control Rod D-6 does impact reactivity parameters (e.g., shutdown margin and trip reactivity); however, sufficient margin exists to ensure the Updated Final Safety Analysis Report (UFSAR) accident analysis limits continue to be met. The physical changes associated with the removal of Control Rod D-6 do not impact the probability of occurrence of a previously evaluated accident. 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 STP Unit 1 with Control Rod D-6 removed will not create the possibility of a new or different kind of accident from any accident previously evaluated. To preserve the reactor coolant system flow characteristics in the reactor core, a flow restrictor will be installed at the top of the D-6 guide tube housing. Installation of this component will not prevent the remaining 56 control rods from performing the required design function of providing adequate shutdown margin. No new operator actions are created as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No. Operation of STP Unit 1 with Control Rod D-6 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 a UFSAR design basis or safety limit and does not change any setpoint at which automatic actuations are initiated. STPNOC will continue to confirm all safety analysis limits remain bounding on a cycle-specific basis using an NRC-approved Westinghouse core reload evaluation methodology. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC 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, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 References

6.1. Technical Specification 3.1.1.1, *Shutdown Margin.*

6.2. Technical Specification 3.1.3.4, *Rod Drop Time.*

6.3. Technical Specification 3.1.3.5, *Shutdown Rod Insertion Limit.*

6.4. Technical Specification 3.9.1, *Boron Concentration.*

6.5. Technical Specification 5.3.2, *Control Rod Assemblies.*

6.6. Technical Specification 6.9.1.6, *Core Operating Limits Report (COLR).*

6.7. UFSAR Section 3.9.1.4.8, *Evaluation of the Control Rod Drive Mechanisms.*

6.8. UFSAR Section 3.9.5, *Reactor Pressure Vessel Internals.*

6.9. UFSAR Section 3.9.5.3, *Design Loading Categories.*

6.10. UFSAR Section 4.2.2.3.1, *Rod Cluster Control Assembly.*

6.11. UFSAR Section 4.3.2.4.12, *Rod Cluster Control Assemblies.*

6.12. UFSAR Section 15.1.1, *Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature.*

6.13. UFSAR Section 15.1.2, *Feedwater System Malfunctions Causing an Increase in Feedwater Flow.*

6.14. UFSAR Section 15.1.3, *Excessive Increase in Secondary Steam Flow.*

6.15. UFSAR Section 15.1.4, *Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing a Depressurization of the Main Steam System.*

6.16. UFSAR Section 15.1.5, *Spectrum of Steam System Piping Failures Inside and Outside Containment.*

6.17. UFSAR Section 15.2.1, *Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow.*

6.18. UFSAR Section 15.2.2, *Loss of External Electrical Load.*

6.19. UFSAR Section 15.2.3, *Turbine Trip.*

6.20. UFSAR Section 15.2.4, *Inadvertent Closure of Main Steam Isolation Valves.*

6.21. UFSAR Section 15.2.5, *Loss of Condenser Vacuum and Other Events Causing a Turbine Trip.*

6.22. UFSAR Section 15.2.6, *Loss of Nonemergency AC Power to the Plant Auxiliaries (Loss of Offsite Power).*

6.23. UFSAR Section 15.2.7, *Loss of Normal Feedwater Flow.*

6.24. UFSAR Section 15.2.8, *Feedwater System Pipe Break.*

6.25. UFSAR Section 15.3.1, *Partial Loss of Forced Reactor Coolant Flow.*

6.26. UFSAR Section 15.3.2, *Complete Loss of Forced Reactor Coolant Flow.*

6.27. UFSAR Section 15.3.3, *Reactor Coolant Pump Shaft Seizure (Locked Rotor).*

6.28. UFSAR Section 15.3.4, *Reactor Coolant Pump Shaft Break.*

6.29. UFSAR Section 15.4.1, *Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition.*

6.30. UFSAR Section 15.4.2, *Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power.*

6.31. UFSAR Section 15.4.3, *Rod Cluster Control Assembly Misoperation.*

6.32. UFSAR Section 15.4.4, *Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature.*

6.33. UFSAR Section 15.4.5, *A Malfunction or Failure of the Flow Controller in a BWR Loop That Results in an Increased Reactor Coolant Flow Rate.*

6.34. UFSAR Section 15.4.6, *Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant.*

6.35. UFSAR Section 15.4.7, *Inadvertent Loading of a Fuel Assembly into an Improper Position.*

6.36. UFSAR Section 15.4.8, *Spectrum of Rod Cluster Control Assembly Ejection Accidents*

6.37. UFSAR Section 15.5.1, *Inadvertent Operation of ECCS During Power Operation.*

6.38. UFSAR Section 15.5.2, *Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory.*

6.39. UFSAR Section 15.6.1, *Inadvertent Opening of a Pressurizer Safety or Relief Valve.*

6.40. UFSAR Section 15.6.2, *Failure of Small Lines Carrying Primary Coolant Outside Containment.*

6.41. UFSAR Section 15.6.3, *Steam Generator Tube Rupture.*

6.42. UFSAR Section 15.6.4, *Radiological Consequences of Main Steam Line Failure Outside Containment (BWR).*

6.43. UFSAR Section 15.6.5, *Loss of Coolant Accidents.*

6.44. UFSAR Section 15.7, *Radioactive Release From a Subsystem or Component.*

6.45. UFSAR Section 15.8, *Anticipated Transients Without Scram.*

6.46. STPNOC Procedure 0PEP01-ZE-0003, "Core Reload Design Process," Revision 17.

6.47. STPNOC Procedure 0PGP04-RX-0227, "PWR Reactor Internals Aging Management Program," Revision 0.

6.48. STPNOC Procedure 0PGP04-ZE-0309, "Design Change Package," Revision 35.

6.49. Letter; John F. Stolz (ANO - 1) to NRC; "Request for a Technical Specification Change to Allow the Removal of Center Control Rod Assembly" (1CAN078609); dated July 18, 1986, NRC Legacy ADAMS accession number 8607280077.

6.50. Letter; Guy S. Vissing (NRC) to Gene Campbell (ANO-1); "Amendment No. 103 to Facility Operating License No. DPR-51 to allow the removal of the center control rod assembly from Arkansas Nuclear One"; dated November 14, 1986; NRC Legacy ADAMS accession number unavailable.

6.51. "Westinghouse Reload Safety Evaluation Methodology;" WCAP-9272-P-A; July 1985.

6.52. Letter; G. T. Powell to USNRC Document Control Desk; "Emergency License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1 Cycle 20;" NOC-AE-15003315 (ML15343A347); dated December 3, 2015.

6.53. Letter; G. T. Powell to USNRC Document Control Desk; "Response to Request for Additional Information and Supplement to Unit 1 Emergency License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1 Cycle 20;" NOC-AE-15003318 (ML15344A304); dated December 9, 2015.

6.54. Letter; L. Regner to D. Koehl; "South Texas Project Unit 1 – Issuance of Amendment Re: Revision to Technical Specifications for One Operating Cycle Operation with 56 Control

Rods (Emergency Circumstances);" AE-NOC-15002765 (ML15343A128); dated December 11, 2015.

6.55. Liu, Y. S., et al., "ANC - A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A, September 1986.

6.56. Nguyen, T. Q., et. al., "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A, June 1988.

6.57. "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," WCAP-14565, Revision 0, October 1999.

6.58. Ouisloumen, M., et al., "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (Proprietary) and WCAP-16045-NP-A (Non-Proprietary), August 2004.

6.59. Yarbrough, M. B., et al., "APOLLO - A One Dimensional Neutron Diffusion Theory Program," WCAP-13524-P-A, Revision 1-A, September 1997.

6.60. Zhang, B., et al., "Qualification of the NEXUS Nuclear Data Methodology," WCAP-16045-P-A, Addendum 1, November 2005.

Attachment 1

Technical Specification Markup

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 193 fuel assemblies. Each fuel assembly shall consist of a matrix of zircaloy, ZIRLO™ or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy, ZIRLO™ or stainless steel filler rods for fuel rods, in accordance with NRC-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 non-limiting core regions.

CONTROL ROD ASSEMBLIES

The Unit 1 core shall contain 56 full-length control rod assemblies with no full-length control rod assembly installed in core location D-6. The Unit 2 core shall contain 57 full-length control rod assemblies.

5.3.2 ~~The core shall contain 57* full-length control rod assemblies.~~ The full-length control rod assemblies shall contain a nominal 158.9 inches of absorber material. The absorber material within each assembly shall be silver-indium-cadmium or hafnium. Mixtures of hafnium and silver-indium-cadmium are not permitted within a bank. All control rods shall be clad with stainless steel tubing.

5.4 (NOT USED)

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

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

~~* The Unit 1 Cycle 20 core shall contain 56 full-length control rod assemblies with no full-length control rod assembly installed in core location D-6.~~

Attachment 2

List of Commitments

List of Commitments

The following table identifies the action to which STP Nuclear Operating Company has committed. Statements in this submittal with the exception of those in the table below are provided for information purposes and are not considered commitments.

Commitment	Scheduled Completion Date	Condition Report Action
Revise applicable procedures to ensure that rapid refueling operations are not performed for STP Unit 1.	March 2, 2017	16-2792-21