UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 21, 2016

Mr. G. T. Powell
Executive Vice President and CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 1 - ISSUANCE OF AMENDMENTS RE: PERMANENT OPERATION WITH 56 CONTROL RODS (CAC NO. MF7577)

Dear Mr. Powell:

The Commission has issued the enclosed Amendment No. 211 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1 (STP-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 7, 2016, as supplemented by letters dated May 25 and September 28, 2016.

The amendment revises TS 5.3.2, "Control Rod Assemblies," to allow the STP-1 core to contain only 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6. The proposed amendment would allow permanent operation of STP-1 with 56 full-length control rods.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-498

Enclosures:
1. Amendment No. 211 to NPF-76
2. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 211
License No. NPF-76

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)*, acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated April 7, 2016, as supplemented by letters dated May 25 and September 28, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 211, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance of this amendment and shall be implemented prior to entering Mode 5 from Mode 6 during startup from refueling outage 1RE20.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating
License No. NPF-76 and the
Technical Specifications

Date of Issuance: December 21, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 211 TO

FACILITY OPERATING LICENSE NO. NPF-76

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

Replace the following pages of the Facility Operating License No. NPF-76 and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License No. NPF-76

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
5-6	5-6

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 211, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Safety Parameter Display System (Section 18, SSER No. 4)*

Before startup after the first refueling outage, HL&P[**] shall perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to issues as described in Section 18.2 of SER Supplement 4.

(6) Supplementary Containment Purge Isolation (Section 11.5, SSER No. 4)

HL&P shall provide, prior to startup from the first refueling outage, control room indication of the normal and supplemental containment purge sample line isolation valve position.

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* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

** The original licensee authorized to possess, use and operate the facility was HL&P. Consequently, historical references to certain obligations of HL&P remain in the license conditions.

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

5.3.2 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. 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:

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 211 TO

FACILITY OPERATING LICENSE NO. NPF-76

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1.0 INTRODUCTION

By letter dated April 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16110A297), as supplemented by letters dated May 25 and September 28, 2016 (ADAMS Accession Nos. ML16162A196 and ML16285A406, respectively), STP Nuclear Operating Company (STPNOC), the licensee for South Texas Project, Unit 1 (STP-1), requested a revision to the STP-1 Technical Specifications (TSs) to allow operation with 56 full-length control rod (CR) assemblies with no full-length CR assembly in core location D-6. The supplemental letter dated September 28, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 19, 2016 (81 FR 46967).

Specifically, the proposed revision to TS 5.3.2, "Control Rod Assemblies" would allow STP-1 to operate with 56 full-length CR assemblies with no full-length CR assembly in core location D-6, in lieu of the requirement to contain 57 full-length CR assemblies. Unit 2 will not be impacted by this change. STPNOC is also proposing to remove the footnote, which permitted the STP-1 core to contain 56 full-length CR assemblies with no full-length CR assembly installed in core location D-6 for Cycle 20 only.

By letter dated December 11, 2015 (ADAMS Accession No. ML15343A128), the NRC staff approved an emergency amendment for STP-1 to revise TS 5.3.2 to allow one cycle of operation with 56 full-length CRs while STPNOC considered alternatives to restore the control rod drive mechanism (CRDM) D-6 function. In the April 7, 2016, license amendment request (LAR), STPNOC stated that it had evaluated the options for repairing or replacing CRDM D-6 and concluded that these options were not feasible due to significant uncertainties involved in cutting and welding of the reactor coolant system boundary and manufacture of specialized

Enclosure 2

remote tooling and processes to perform the replacement. The licensee, therefore, requested in the LAR that the NRC allow STP-1 to operate permanently with 56 full-length CRs.

2.0 REGULATORY EVALUATION

Under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, whenever a holder of a license wishes to amend the license, including technical specifications in the license, an application for amendment must be filed, fully describing the changes desired and following as far as applicable, the form prescribed for original applications. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Both the common standards for licenses in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be reasonable assurance that the activities authorized by the license will not endanger the health and safety of the public.

Under 10 CFR 50.36(a)(1), each applicant for a license authorizing operation of a utilization facility shall include in its application proposed technical specifications in accordance with the requirements of 10 CFR 50.36. The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." The TS, per 10 CFR 50.36(c)(1)-(5), are required to include items in the following categories: (1) safety limits, limiting safety systems settings, and control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in the categories of (1) safety limits, limiting safety systems settings, and control settings, (2) limiting conditions for operation, or (3) surveillance requirements.

The design of the Control Rod Assemblies is a design feature which, if altered, would have a significant effect on are safety, and is not covered by items under 10 CFR 50.36(c)(1)-(3). Accordingly, as required per 10 CFR 50.36(c)(4), Technical Specification (TS) 5.3.2, "Control Rod Assemblies," addresses design of the Control Rod Assemblies.

Section 3.1, "Conformance with NRC General Design Criteria," of the STP-1 Updated Final Safety Analysis Report (UFSAR), Rev. 14, contains an evaluation of the design bases as measured against the NRC General Design Criteria (GDC) for Nuclear Power Plants in Appendix A to 10 CFR Part 50. Section 3.1 of the UFSAR states that there are some cases where conformance to a particular criterion is not directly measurable, and in these cases, the conformance of plant design to the interpretation of the criterion is discussed. The UFSAR further states that based on the content in Section 3.1, it is concluded that the STP-1 nuclear power plant fully satisfies and is in compliance with the GDC. The NRC staff determined that four of the GDC were particularly applicable to the review of the LAR, and these four are discussed below.

GDC 10, "Reactor design," states that "[t]he 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."

Section 3.1.2.2.1.1 of STP-1 UFSAR, Rev. 14, provides the following evaluation against GDC 10:

> The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:
>
> 1. Preclude significant fuel damage during normal core operation and operational transients (Condition I)* or any transient conditions arising from occurrences of moderate frequency (Condition II)*.
>
> 2. Ensure return of the reactor to a safe state following a Condition III* event with only a small fraction of fuel rods damaged although sufficient fuel damage might occur to preclude immediate resumption of operation.
>
> 3. Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV)**. Chapter 4 discusses the design bases and design evaluation of reactor components including the fuel and reactivity control materials. Section 3.9 discusses the design bases and design evaluation of the reactor vessel internals and the control rod drive mechanisms. Details of the control and protection systems instrumentation design and logic are discussed in Chapter 7. This information supports the accident analyses of Chapter 15 which show that the acceptable fuel design limits are not exceeded for Condition I and II occurrences.
>
> ----
>
> * Defined by ANSI N18.2 – 1973
>
> ** Defined by ANSI N18.2-1973 Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided by reactor trip functions using the measured axial power imbalance as an input.

GDC 11, "Reactor Inherent Protection," states that "[t]he 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."

Section 3.1.2.2.2.1 of STP-1 UFSAR, Rev. 14, provides the following evaluation against GDC 11:

> Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the nonpositive operational limit on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel; the nonpositive moderator temperature coefficient of reactivity is assured by administratively controlling the dissolved absorber concentration or by burnable poisons.
>
> These reactivity coefficients are discussed in Section 4.3.

GDC 26, "Reactivity control system redundancy and capability," states that "[t]wo 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."

Section 3.1.2.3.7.1 of STP-1 UFSAR, Rev. 14, provides the following evaluation against GDC 26:

> Two Reactivity Control Systems are provided. These are rod cluster control assemblies (RCCAs) and chemical shim (boric acid). The RCCAs are inserted into the core by the force of gravity.
>
> During operation, the shutdown rod banks are fully withdrawn. The Control Rod System automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The shutdown rod banks along with the control banks are designed to shut down the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences, thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to be in the fully withdrawn position.
>
> The Chemical and Volume Control System (CVCS) will maintain the reactor in the cold shutdown state independent of the position of the control rods and can compensate for xenon burnout transients.
>
> Details of the construction of the RCCSs are presented in Chapter 4 and the operation is discussed in Chapter 7. The means of controlling the boric acid concentration is described in Chapter 9. Performance analyses under accident conditions are included in Chapter 15.

GDC 28, "Reactivity limits," states that "[t]he 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 be 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 the reactor coolant temperature and pressure, and cold water addition."

Section 3.1.2.3.9.1 of STP-1 UFSAR, Rev. 14, provides the following evaluation against GDC 28:

> The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values that prevent rupture of the RCPB or disruptions of the core or vessel internals to a degree that could impair the effectiveness of emergency core cooling.
>
> The maximum positive reactivity insertion rates for the withdrawal of RCCAs and the dilution of the boric acid in the RCS are limited by the physical design characteristics of the RCCAs and of the CVCS. Technical specifications on shutdown margin and on RCCA insertion limits and bank overlaps as functions of power provide additional assurance that the consequences of the postulated accidents are no more severe than those presented in the analyses of Chapter 15. Reactivity insertion rates, dilution, and withdrawal limits are also discussed in Section 4.3. The capability of the CVCS to avoid an inadvertent excessive rate of boron dilution is discussed in Chapter 15.
>
> Assurance of adequate core-cooling capability following Condition IV accidents, such as rod ejections, steam line break, etc., is provided through analysis to demonstrate that the RCPB stresses remain within faulted condition limits as specified by applicable ASME Codes. Structural deformations are checked also and limited to values that do not jeopardize the operation of necessary safety features. Condition IV accidents are discussed in Section 15.0.1.4.

The NRC staff reviewed the licensee's proposed change to its design feature technical specification, which reflect the physical changes in the design feature, would continue to provide reasonable assurance of public health and safety. The staff considered in part if the physical changes made by the removal of the CRs would cause STP-1 no longer to adhere to the design prescribed in UFSAR. The staff used NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 4.6, "Functional Design of Control Rod Drive System" (ADAMS Accession No. ML070540139) to inform the staff's review.

3.0 TECHNICAL EVALUATION

3.1 Background

On November 18, 2015, STPNOC was conducting normal pre-operational testing and surveillances on safety equipment in preparation for restart of STP-1 following refueling outage 1RE19. During routine CR drop time surveillance while the reactor was still shut down, CR D-6 of Shutdown Bank A did not function as expected. STPNOC operators were initially unable to move CR D-6 by normal methods, but subsequently, operators were able to move CR D-6 to the bottom of the core (i.e., fully inserted). STP-1 was subsequently cooled down and the reactor head was disassembled. The inability to move CR D-6 was determined by the licensee to be due to deformation of the CRDM rod holdout ring, which is used during rapid refueling operations.[1] Of the 57 CRDMs in STP-1, all were inspected to determine the extent of the damage. No other deformation was observed on the other 56 STP-1 CRDMs.

By letter dated December 3, 2015 (ADAMS Accession No. ML15343A347), STPNOC requested an emergency license amendment to revise the STP-1 TS for one cycle of operation, Cycle 20, with only 56 CRs. The NRC staff approved the emergency license amendment via letter dated December 11, 2015. The NRC staff review and approval of the emergency license amendment was limited to Cycle 20 and did not address the indefinite removal of this CR.

In this safety evaluation (SE), the NRC staff has evaluated the technical issues associated with the methodologies used by the licensee, the impacted key safety parameters (i.e., inputs) used for safety analyses, and possible impacts to the Updated Final Safety Analysis Report (UFSAR) Chapter 15 safety analyses to determine if the operation of STP-1 is safe considering long-term removal of the CR D-6.

3.2 Proposed Changes

TS 5.3.2, "Control Rod Assemblies," currently states, in part, that "[t]he core shall contain 57* full-length control rod assemblies." There is a footnote (*) for TS 5.3.2 that states "[t]he 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."

The proposed change to TS 5.3.2, "Control Rod Assemblies," will remove the footnote and revise the statement to read, "[t]he 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."

3.3 Structural Evaluation

3.3.1 Dynamic Analysis

Removal of the CR D-6 drive shaft and rod cluster control assembly (RCCA) reduces the weight of the CRDM, which can impact the dynamic analyses that predict the stresses in the CRDM, reactor vessel, vessel supports, and reactor internals when subjected to seismic or loss-of-coolant-accident excitations.

As stated in the submittal, and as audited during a regulatory audit conducted by the NRC staff on December 7-8, 2015,[2] the current dynamic analysis of the CRDM was performed using the licensee's reactor equipment system model. Removal of the CR drive shaft reduces the overall weight of the CRDM, which generally reduces dynamic loads. Furthermore, the overall reduction of weight due to removal of the CR drive shaft is relatively small given the magnitude of the entire CRDM assembly. Because the model remains valid, the staff agrees that the current CRDM dynamic stress evaluations, due to seismic and loss-of-coolant-accident (LOCA) excitations, remains valid for the proposed configuration.

[1] A more detailed description of the core design, CRDM, rapid refueling operation, and the damage to the CR D-6 is provided in the NRC staff's SE for Amendment No. 208, dated December 11, 2015.
[2] Audit report dated March 15, 2016 (ADAMS Accession No. ML16029A272).

Other current reactor vessel, vessel supports, and reactor internals analyses are also unaffected by the mass impacts of removing the CR D-6 drive shaft and RCCA due to the relatively small change in mass when compared to the mass of the reactor vessel head. The NRC's staff's review of UFSAR Section 3.9.1, "Special Topics for Mechanical Components," reveals that the models used in these analyses are lumped-mass models where the mass of the CRDMs, including the drive shaft and the RCCA, are lumped at the center of gravity of the reactor vessel head. In the submittal, the licensee estimated that the mass of the drive shaft and RCCA is approximately 300 pounds, while the mass of the reactor vessel head is approximately 350,000 pounds. Given the small percentage change (less than 0.09 percent) of the lumped mass to be modeled, the NRC staff finds that the impact of the weight reduction on the model used to analyze the reactor vessel, vessel supports, and reactor internals is negligible.

3.3.2 Reactor Coolant Pressure Boundary

The NRC staff has reviewed the proposed changes and determined that the reactor coolant pressure boundary is unaffected.

3.3.3 Thimble Plug

A thimble plug was installed on the fuel assembly in core location D-6 for Cycle 20, and will be removed for future fuel cycles to reduce fuel component handling during core refueling. The removal of the thimble plug is acceptable from a structural standpoint because it returns the configuration to the originally licensed configuration, and is consistent with the surrounding fuel assemblies.

3.3.4 Flow Restrictor

The flow restrictor added at the top of the D-6 guide tube must be assessed for structural adequacy. In the submittal, the licensee stated that the flow restrictor materials and design meet American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Subsection NG, and were analyzed using material properties that were taken from Section II, Part A of the ASME Code to meet the allowable stress limits. During a regulatory audit conducted on December 7-8, 2015, the NRC staff reviewed the flow restrictor design and structural analysis documents. The staff verified that the flow restrictor materials and design meet the ASME Code. The staff has determined that the flow restrictor is structurally adequate because it meets the relevant requirements of the ASME Code, Subsection NG.

Thermal expansion and material compatibility must also be assessed to provide assurance of the structural adequacy of the added flow restrictor. In the submittal, the licensee detailed that the flow restrictor is made of the same grade of stainless steel as used to fabricate the guide tubes to which the flow restrictor is mounted. Because the same material is used, there is no differential thermal expansion between these two components, and material compatibility of the flow restrictor with the fluid conditions in the reactor vessel upper head are assured.

3.3.5 Potential Long-Term Impacts

The NRC staff reviewed the licensee's assessments of potential long-term impacts and sources of degradation such as corrosion product build-up in the guide tube, thermal sleeve outer surface wear, flow-induced vibration, and long-term material degradation.

The NRC staff evaluated the potential for an increase the amount of corrosion product build-up in the guide tube. Based on operating experience at STP Unit 1 and 2, and of other similar Westinghouse pressurized-water reactor plants in the United States which have locations without a CR drive shaft installed, any potential increases in corrosion product build-up is not expected to have substantive safety impacts. Furthermore, the added flow restrictor is not expected to further increase the amount of corrosion product build-up in the guide tube because the previous flowrate and hydraulic flow conditions will remain unchanged.

The NRC staff evaluated the removal of the CR D-6 drive shaft on the potential for thermal sleeve outer surface wear. The missing drive shaft can increase the potential for thermal sleeve outer surface wear against the inside of the reactor vessel head adapter since the drive shaft is not present to limit the depth of the wear. Operating experience has shown that more thermal sleeve wear occurs at outer diameter core locations than inner core locations including outer locations without CR drive shafts. The CR D-6 thermal sleeve is located at an inner core location, thus the wear effect at location D-6 is expected to be bounded by the known wear at unrodded outer core locations. Because the potential wear is expected to be bounded by other core locations, the existing analyses and the current thermal sleeve inspection plan remain acceptable for this license amendment.

The effects of flow-induced vibration are not expected to change due to the installation of the guide tube flow restrictor which maintains the previous hydraulic flow conditions in the upper head.

Long-term material degradation of the added guide tube flow restrictor is not expected since the flow restrictor is made of the same grade of stainless steel as the guide tubes and other upper internals. In the submittal, the licensee described a number of design features and installation procedures in the submittal that provide long-term assurance against the generation of loose parts. The hex bolt that is used for attachment is preloaded to a specified torque, which, in conjunction with the locking fingers, securely locks the flow restrictor to 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 loosening due to vibration. The NRC staff has reviewed the submittal, and audited the design and analysis documents of the flow restrictor during a regulatory audit conducted on December 7-8, 2015, and finds that the materials and design features of the flow restrictor provide assurance that the flow restrictor is securely installed and will not result in the generation of loose parts.

Based on the above, the NRC staff find the potential long-term impacts and sources of degradation due to the removal of the CR D-6 will be minimal, and, therefore, concludes that the change is acceptable.

3.3.6 Structural Evaluation Conclusion

The NRC staff concludes that the licensee's removal of the CR D-6 drive shaft and RCCA, and installation of the flow restrictor and removal of the thimble plug in the D-6 location is acceptable as a permanent change from a structural standpoint because (1) the change does not invalidate the current models used in the dynamic stress evaluations, (2) the change does not affect the reactor coolant pressure boundary, (3) the added components have been adequately analyzed for structural adequacy, and (4) the changes and added components have been adequately analyzed to address concerns of potential long-term impacts and degradation.

3.4 Accident Analysis Evaluation

The NRC staff focused on the effects of the permanent removal of CR D-6 on the UFSAR Chapter 15, "Accident Analyses," and the NRC-approved methodologies used by STP-1 to evaluate the change. The NRC staff's objective was to determine if the inputs to the safety analyses that are impacted by the removal of CR D-6 remain bounding, and that methodologies used to perform the safety analysis could adequately account for the removed CR.

3.4.1 Methodologies Review

The licensee stated that for the UFSAR Chapter 15 bounding analyses, the number of CRs and the CRs pattern are not used as inputs. Instead, these parameters are inputs to the cycle-specific core reload analysis. The NRC staff reviewed each scenario and the associated methodology and verified that the number of CRs and CR pattern would be appropriately accounted for in the cycle-specific core reload analysis through the power related inputs from the Advanced Nodal Code (ANC), which does have the capability of explicitly modeling CR removal.

During its review, the NRC staff verified that the removal of CR D-6 has a minor impact on the core reload analysis. The methodology which is impacted is ANC, as it models the CRs and calculates the various rod powers. The ANC is used for neutronics analyses in order to generate parameters such as the critical boron concentration, CR worth, reactivity coefficient, assembly average power and exposure, assembly peak power, peaking factor, and axial power shape. As the removal of CR D-6 could have an impact on the calculation of these parameters, the ANC must be capable of adequately modeling the given scenarios with the CR removed.

It was not clear to the NRC staff that this was within the capabilities and qualifications of the ANC or that it was within the initial scope of approval of the ANC; therefore, by letters dated August 26 and September 15, 2016 (ADAMS Accession Nos. ML16214A291 and ML16246A095, respectively), a request for additional information (RAI) (and correction letter) was issued to provide justification for the continued use of the ANC with the removal of the CR D-6. The NRC staff requested the licensee to demonstrate that any change to the simulations considered (i.e., N-1 to N-2 rods inserted) were within the capabilities and qualifications of the ANC and the scope of the initial approval of the ANC.

In a letter dated September 28, 2016, STPNOC provided further details about the ANC and its approval. The licensee confirmed that the solution method used by the ANC is equally valid for cores with any distribution of CRs. Further, the ANC has previously been approved to model much more challenging core configurations such as the case for Combustion Engineering plants where differences in peaking factors resulting from the N-1 condition are typically much larger than those expected in a Westinghouse plant. Because STPNOC adequately demonstrated that the ANC is capable of performing the neutronics analysis with CR D-6 removed, the NRC staff has determined that the analysis methodologies used by STPNOC are capable of appropriately modeling the accident and transient scenarios.

3.4.2 Impact on Key Safety Parameters

The licensee identified the inputs to the safety analyses that were potentially impacted by the removal of CR D-6 via use of the WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology," March 1978. These potentially impacted inputs are identified as key safety parameters in WCAP-9272. Although WCAP-9272 was not specifically designed to evaluate changes such as this CR D-6 removal, the NRC staff reviewed the parameters identified by the licensee and compared those to inputs in the UFSAR Chapter 15 Accident Analyses that could potentially be impacted by the removal of CR D-6 and found that the licensee had adequately identified the impacted parameters (i.e. the key safety parameters).

The removal of CR D-6 has a direct impact on several of the key safety parameters assumed in the UFSAR Chapter 15 analyses. The licensee evaluated all the key safety parameters found in WCAP-9272 for the removal of CR D-6 for impacts through the end of licensed operation of the facility. The licensee determined that a portion of the key safety parameters were not impacted, and the NRC evaluated and verified that the key safety parameters identified by the licensee were not impacted. Additionally, the licensee presented an analysis of the key safety parameters that were impacted by the removal of CR D-6 and the NRC evaluated this analysis. The NRC staff's review of these parameters is discussed in detail below.

The key safety parameters assumed in the transient and accident analyses in UFSAR Chapter 15 that are impacted by the removal of CR D-6 are:

- Shutdown Margin,
- Boron Worth,
- Total Rod Worth,
- Trip Reactivity,
- Moderator Density Coefficient, and
- Power Distribution

The effect of the proposed change on these parameters is described below.

3.4.2.1 Shutdown Margin

The removal of the CR D-6 shutdown bank rod has a direct impact on the shutdown margin (SDM) available. Section 2.3 of the STP-1 Core Operating Limit Report (COLR) for Cycle 20 submitted in the licensee's letter dated November 12, 2015 (ADAMS Accession No. ML15334A346), provides the SDM limits to validate the safety analyses. For MODES 1 and 2, the SDM has to be greater than the 1.3 percent delta rho (% $\Delta\rho$) limit stated in the licensee's COLR. The licensee recalculated the SDM before and after removal of CR D-6, and the value changed from 2.42 to 2.16 for Cycle 20. Given the reduction in margin, the new SDM value of

2.16% Δρ is greater than the COLR limit for MODES 1 and 2 of 1.3 % Δρ, and was, therefore, determined acceptable by the NRC.

In the supplement dated September 28, 2016, the licensee stated that the cycle-specific SDM is determined via a full-core, 3-dimensional (3-D) ANC model. The SDM calculation determines the location of the highest worth CR and evaluates it as follows:

SDM = (N–1 Rod Worth) – Power Defect – Void – Rod Insertion Allowance
Where N is the number of CRs

N-1 Rod Worth is the total rod worth minus the worth of the most reactive RCCA which is assumed to be fully withdrawn. This parameter is calculated explicitly using the 3-D ANC model leaving the most reactive CR fully withdrawn. N-1 Rod Worth is reduced in the calculation by 10% uncertainty for conservatism. N-1 Rod Worth is impacted by the removal of CR D-6 by creating an N-2 situation with CR D-6 and the highest worth rod being stuck out.

Power Defect is the positive reactivity associated with a decrease in reactor core power due to temperature feedback effects from a decrease in fuel and moderator temperature. This parameter is calculated explicitly using the 3-D ANC model at All Rods Out condition and therefore is not impacted by the removal of the RCCA from core location D-6.

Void is the term applied to conservatively increase the Power Defect term. The void defect accounts for the positive reactivity insertion that occurs due to increased moderation when voids collapse when tripping to zero power. This term is a constant value and is not impacted by the removal of the RCCA from core location D-6.

Rod Insertion Allowance is the term which accounts for the allowed partial insertion of the CRs as a function of power. This parameter is calculated explicitly using the 3-D ANC model. This parameter is calculated at power, thus, the removal of the CR D-6 has no impact on the calculated rod insertion allowance because it is in a shutdown bank and shutdown banks are fully withdrawn when the core is at power.

The NRC staff reviewed the capability for the ANC to model the reactor core with one CR removed, and determined that the ANC possesses this capability. Further, the licensee previously demonstrated for Cycle 20, with CR D-6 removed that SDM requirements are met. The NRC staff concludes that the licensee adequately addressed the SDM with CR D-6 permanently removed. The staff next determined if, with the amendment granted, the licensee would continue to conform to GDC 10 as described in the UFSAR with respect to shutdown margin. Specifically, the staff considered whether adequate SDM was in place to preclude significant fuel damage during normal core operation and operational transients, or any transient conditions arising from occurrences of moderate frequency. As previously discussed in this section, the licensee demonstrated adequate SDM. Accordingly, the staff found that the licensee would continue to conform to GDC 10 with respect to shutdown margin.

3.4.2.2 Boron Worth

The NRC staff previously evaluated the boron worth for STP-1 specific to a single cycle for the removal of CR D-6 under the emergency amendment for revising the TS for one operating cycle with 56 CRs. The NRC staff determined that the boron worth is not impacted for the analyses in UFSAR Chapter 15. Thus, the impact on boron worth as a result of the removal of CR D-6 is acceptable because it remains within the bounds of the assumed values in Chapter 15 of the UFSAR.

For this amendment, the NRC staff reviewed the removal of CR D-6 for the duration of the operating license. Further, the licensee previously demonstrated for Cycle 20, with CR D-6 removed, that boron worth requirements were met. The staff next determined if, with the amendment granted, the licensee would continue to conform to GDC 26 as described in the UFSAR, with respect to boron worth. Specifically, the staff considered if adequate boron worth exists, with the removal of CR D-6, to ensure that fuel design limits are not exceeded. The insertion of control rods will continue to have adequate boron worth to shut down the reactor core. Accordingly, the staff found that the licensee would continue conform with GDC 26 with respect to boron worth.

3.4.2.3 Total Rod Worth

The licensee stated in Table 3 of the application that the total rod worth is evaluated on a cycle-specific basis to ensure the SDM and trip reactivity limits are met. Total rod worth is discussed previously in this SE under Section 3.4.1, "Shutdown Margin."

For this amendment, the NRC staff reviewed the removal of CR D-6 for the duration of the operating license. The NRC staff reviewed the capability for the ANC to model the reactor core with one CR removed, and determined that the ANC possesses this capability. The NRC staff concludes that the licensee adequately addressed the total rod worth with CR D-6 permanently removed.

The NRC staff next determined if, with the amendment granted, the licensee would continue to conform with GDC 10 as described in the UFSAR with respect to total rod worth. Specifically, the staff considered whether adequate total rod worth exists to ensure SDM and trip reactivity are sufficient to preclude significant fuel damage during normal core operation and operational transients, or any transient conditions arising from occurrences of moderate frequency. The licensee demonstrated adequate total rod worth as previously discussed in this section of the SE. Accordingly, the staff found that the licensee would continue conform with GDC 10 with respect to total rod worth.

3.4.2.4 Trip Reactivity

The NRC staff previously evaluated the trip reactivity for STP-1 specific to a single cycle for the removal of CR D-6 under the emergency amendment for revising the TS for one operating cycle with 56 CRs. The NRC staff determined that the new trip reactivity values with CR D-6 removed are acceptable because they are still bounded by the assumed values in Chapter 15 of the UFSAR. The application states that a cycle-specific evaluation will be performed for each core reload design to confirm the trip reactivity remains bounded.

For this amendment, the NRC staff reviewed the removal of CR D-6 for the duration of the operating license. The NRC staff reviewed the capability for the ANC to model the reactor core with one CR removed, and determined that the ANC possesses this capability. Further, the licensee previously demonstrated for Cycle 20, with CR D-6 removed, that trip reactivity worth requirements are met. The NRC staff concludes that the licensee adequately addressed the trip reactivity with CR D-6 permanently removed.

The NRC staff next determined if, with the amendment granted, the licensee would continue to conform with GDC 26 and GDC 28 as described in the UFSAR with respect to trip reactivity. Specifically, the staff considered whether adequate trip reactivity worth remains in place for the shutdown banks along with the control banks to shut down the reactor with adequate margin under normal operating conditions or anticipated operational occurrences. Adequate trip reactivity worth also ensures that specified fuel design limits are not exceeded. The trip reactivity worth remains adequate, as discussed in this section, because the ANC platform has the capability to model the core adequately with one CR removed and is capable of ensuring trip reactivity within bounds for each operating cycle. Accordingly, the staff found that the licensee would continue conform with GDC 26 and GDC 28 with respect to trip reactivity.

3.4.2.5 Moderator Density Coefficient

The NRC staff has evaluated the Moderator Density Coefficient (MDC) for STP-1 specific to a single cycle for the removal of CR D-6 under the emergency amendment for revising the TS for one operating cycle with 56 CRs. The NRC staff determined that the new value for most positive MDC is still conservative as it is below the assumed most positive MDC used in the UFSAR Chapter 15 analyses, and is thus acceptable. The licensee stated in its application that a cycle-specific evaluation of the MDC values with the removal of CR D-6 will be performed for each core reload design to confirm the most positive MDC remains bounding.

For this amendment, the NRC staff reviewed the removal of CR D-6 for the duration of the operating license. The NRC staff reviewed the capability for the ANC to model the reactor core with one CR removed, and determined that the ANC possesses this capability. The NRC staff concludes that the licensee adequately addressed the MDC with CR D-6 permanently removed.

The NRC staff next determined if, with the amendment granted, the licensee would continue to conform with GDC 10 as described in the UFSAR with respect to moderator density coefficient. Specifically, the staff considered whether the MDC with CR D-6 removed will have appropriate margin to assure that specified acceptable fuel design limits are not exceed during any condition of normal operation, including the effects of anticipated operational occurrences. As discussed above, the staff found that the licensee would continue conform with GDC 10 with respect to moderator density coefficient because the new value for most positive MDC is still conservative and is, therefore, acceptable.

3.4.2.6 Power Distribution

The CRs for pressurized-water reactors are positioned in a fully withdrawn position during power operation. To distribute RCCA cladding wear associated with flow-induced vibration within the reactor vessel upper head internal CR guide tubes, the fully withdrawn position is varied from cycle to cycle. This fully withdrawn position can have the CR tips just above the top of active fuel or just inside the top of active fuel. The fully withdrawn position of the CRs potentially impacts the power distribution since it could impact the power at the top of the core. For cycles with the CR tips just above the top of active fuel, power distribution will not be impacted with the removal of CR D-6. For cycles with the CR tips just inside the top of active fuel, power distribution will be impacted. The estimated impact, relative to the presence of CR D-6, is small, as the power change is on the order of +0.7 percent for the D-6 fuel assembly.

The NRC staff has reviewed the UFSAR with respect to this power change to determine if a reactivity change of this magnitude impact the UFSAR safety analyses. The NRC staff determined that a change of this magnitude for the CR tips below the top of active fuel will remain bounded by the UFSAR Chapter 15 Safety Analyses. Additionally, the NRC staff concludes that the nuclear design analytical methods used for each reload are capable of explicitly modeling the fully withdrawn position of each CR and thus, the impact on the power distribution. The NRC staff concludes that the licensee adequately addressed the power distribution with CR D-6 permanently removed.

The NRC staff next determined if, with the amendment granted, the licensee would continue to conform with GDC 10 as described in the UFSAR with respect to power distribution. Specifically, the staff considered whether adequate power distribution exists to ensure that the removal of CR D-6 precludes significant fuel damage during normal core operation and operational transients, or any transient conditions arising from occurrences of moderate frequency. As discussed above, the staff found that the licensee would continue to conform with GDC 10 with respect to power distribution because the impacts relative to the removal of CR D-6 is small and the power distribution change will remain bounded by Chapter 15 analyses.

3.5 Impact on Accident Analyses

Table 7 of the LAR details the UFSAR Chapter 15 accident analyses that the licensee reviewed for impacts due to the removal of CR D-6. The licensee's table provided the event number, UFSAR section, reference to applicable reload methodology section, and the licensee's evaluation of the impact to key safety parameters related to the specific UFSAR Chapter 15 design basis accident. The NRC staff created a similar table in this SE, Table 1, adding a column for the NRC staff determination associated with the licensee's evaluation of the UFSAR Chapter 15 accident analysis.

The NRC staff reviewed the licensee's evaluation provided in the application, the referenced NRC-approved reload methodology, and the relevant Chapter 15 UFSAR sections. The NRC staff compared the impacted parameters stated in the application with the corresponding key safety parameters in the reload methodology to verify that the parameters for the event were addressed and that they were bounded by the UFSAR. The Control Rod Ejection Accident (CREA) and Hot Zero Power Main Steam Line Break (HZP MSLB) accidents are the most sensitive UFSAR Chapter 15 accidents to the CR D-6 removal. The details of these accidents are presented following Table 1.

The CREA and HZP MSLB are the UFSAR Chapter 15 limiting analyses for the removal of CR D-6 because both are analyzed at zero power, shutdown conditions where CR D-6 would be inserted into the core and the additional negative reactivity of the rod will be absent during these significant positive reactivity insertion events.

Table 1. Impact on UFSAR Chapter 15 Accident Analysis

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.1.1	5.3.9	Feedwater System Malfunctions Causing a Reduction in Feedwater Temperature	Bounded by UFSAR Sections 15.1.2 and 15.1.3, and terminated without safety or protection systems.	The NRC staff found the licensee's evaluated impact to be acceptable because the removal of the shutdown rod will not affect the rate of heat removal from the core, and thus the bounding assumption is not affected. The NRC staff determined that the removal of the shutdown rod will not impact the rate of heat removal from the core because the accident is terminated without safety or protection systems.
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 are evaluated for each core reload design. Most positive MDC changes with the amount of control rods in the core, resulting in less than a 1% impact. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for each key safety parameter, the NRC staff confirmed that the impact for MDC is 1% and trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the specified acceptable fuel design limit (SAFDL), and thus is acceptable for long-term operation with CR D-6 removed.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.1.3	5.3.9	Excessive Increase in Secondary Steam Flow	No impact. Core DNB [departure from nucleate boiling] limits are not challenged by this event. Event is initiated from a full power condition with CR D-6 fully withdrawn. Event is terminated without safety or protection systems.	The NRC staff determined that the removal of CR D-6 has no impact on the event based on the initiating conditions and mitigating strategies. Thus, the NRC staff determined that removal of CR D-6 has no impact on this accident because the accident is initiated from a full power condition and CR D-6 is fully withdrawn.
15.1.4	5.3.14	Inadvertent Opening of a Steam Generator Relief or Safety Valve Causing Depressurization of the Main Steam System	Bounded by UFSAR Section 15.1.5.	The NRC staff determined that the bounding assumptions are not impacted with the removal of CR D-6 because the accident is not as severe as the spectrum of steam system piping failure inside and outside containment accident discussed below for UFSAR Section 15.1.5. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.
15.1.5	5.3.14	Spectrum of Steam System Piping Failures Inside and Outside Containment	SDM is impacted during the return to power. DNB is impacted and will be evaluated for each core reload design. SDM is decreased because the amount of negative reactivity available to shut down the reactor is reduced. Power distribution is affected and results in a higher power level and a lower DNB ratio after a return to criticality.	The NRC staff's evaluation is provided in Section 3.5.2 of this SE.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.2.1	N/A	Steam Pressure Regulatory Malfunction or Failure that Results in Decreasing Steam Flow	Not Applicable to STP-1.	The NRC staff determined that event is not applicable to STP-1 and thus the removal of CR D-6 has no impact on long-term operation.
15.2.2	5.3.7	Loss of External Electrical Load	Bounded by UFSAR Section 15.2.3.	The NRC staff determined that the bounding assumptions are not impacted with the removal of CR D-6 because the accident is not as severe as the turbine trip accident discussed below for UFSAR Section 15.2.3. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.
15.2.3	5.3.7	Turbine Trip	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change to the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC staff confirmed there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.2.4	5.3.7	Inadvertent Closure of Main Steam Isolation Valves	Bounded by UFSAR Section 15.2.5.	The NRC staff determined that the bounding assumptions are not impacted with the removal of CR D-6 because the accident is not as severe as the loss of condenser vacuum and other events causing turbine trip accident discussed below for UFSAR Section 15.2.5. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.
15.2.5	N/A	Loss of Condenser Vacuum and Other Events Causing Turbine Trip	Bounded by UFSAR Section 15.2.3.	The NRC staff determined that the bounding assumptions are not impacted with the removal of CR D-6 because the accident is not as severe as the turbine trip accident discussed below for UFSAR Section 15.2.3. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.
15.2.6	5.3.8	Loss of Non-Emergency AC [Alternating Current] 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.	The NRC staff determined that the bounding assumptions are not impacted with the removal of CR D-6 because the accident is not as severe as the complete loss of forced reactor coolant flow and loss of normal feedwater flow accidents discussed below for UFSAR Sections 15.3.2 and 15.2.7, respectively. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.

UFSAR Section	WCAP-927 2 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.2.7	5.3.8	Loss of Normal Feedwater Flow	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff has confirmed that the impact for trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.
15.2.8	5.3.15	Feedwater System Pipe Break	Most positive MDC and trip reactivity are potentially impacted and are evaluated for each core reload design. Most positive MDC changes with the amount of control rods in the core, resulting in less than a 1% impact. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for each key safety parameter, the NRC staff confirmed that the impact for MDC is 1% and trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.
15.3.1	5.3.5	Partial Loss of Forced Reactor Coolant Flow	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC determined that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.3.2	5.3.5	Complete Loss of Forced Reactor Coolant Flow	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC staff determined that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without D-6.
15.3.3	5.3.16	Reactor Coolant Pump Shaft Seizure (Locked Rotor)	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC staff confirmed there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.
15.3.4	N/A	Reactor Coolant Pump Shaft Break	Bounded by UFSAR Section 15.3.3.	The NRC staff determined that the bounding assumptions are not impacted with the removal of CR D-6 because the accident is not as severe as the reactor coolant pump shaft seizure (locked rotor) accident discussed above for UFSAR Section 15.3.3. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
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 evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.
15.4.2	5.3.2	Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power	Most positive MDC and trip reactivity are potentially impacted and are evaluated for each core reload design. Most positive MDC changes with the amount of control rods in the core, resulting in less than a 1% impact. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for each key safety parameter, the NRC staff confirmed that the impact for MDC is 1% and trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.
15.4.3	5.3.3 5.3.12	Rod Cluster Control Assembly Misoperation	Key safety parameters are not impacted.	The NRC staff confirmed that there are no impacts on the key safety parameters; therefore, the NRC staff determined the removal of CR D-6 has no impact on this accident analysis.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.4.4	5.3.6	Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation without CR D-6.
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 SDM for return to criticality are potentially impacted and are evaluated for each core reload design. Maximum boron concentration increases to compensate for the change in reactivity due to the removal of CR D-6, with a magnitude of change of less than 40 parts per million (ppm). SDM is decreased because the amount of negative reactivity available to shut down the reactor is reduced, with the change being less than 20%.	Based on the change for the key safety parameters, the NRC staff confirmed that the impact for SDM is 20% and boron concentration is less than 40 ppm. The SDM has a noticeable change based on the removal of CR D-6; however, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation.
15.4.7	N/A	Inadvertent Loading of a Fuel Assembly into an Improper Position	No impact as an inadvertent loading is detected using incore instrumentation when the shutdown banks are withdrawn.	The NRC staff confirmed that there are no impacts on the key safety parameters; therefore, the NRC staff determined the removal of CR D-6 has no impact on this accident analysis.
15.4.8	Various	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Trip reactivity and N-2 K_{eff} are potentially impacted and are evaluated for each core reload design.	The NRC staff's evaluation is provided in Section 3.5.1 of this SE.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
15.5.1	5.3.11	Inadvertent Operation of ECCS During Power Operation	No impact.	The NRC staff confirmed that there are no impacts on the key safety parameters; therefore, the NRC staff determined the removal of CR D-6 has no impact on this accident analysis.
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.	The NRC staff confirmed that there are no impacts on the key safety parameters; therefore, the NRC staff determined the removal of CR D-6 has no impact on this accident analysis.
15.6.1	5.3.10	Inadvertent Opening of a Pressurizer Safety or Relief Valve	Trip reactivity potentially impacted and evaluated for each core reload design. Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip, resulting in less than a 2% impact.	Based on the change for the key safety parameter, the NRC staff confirmed that the impact for trip reactivity is 2%. However, the NRC staff confirmed that there is an adequate amount of margin in relation to the SAFDL, and thus is acceptable for long-term operation with CR D-6 removed.
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.	The NRC staff confirmed that there are no impacts on the key safety parameters; therefore, the NRC staff determined the removal of CR D-6 has no impact on this accident analysis.

UFSAR Section	WCAP-9272 Section	Description	Qualitative Influence and Impact on Accident	NRC Staff Determination
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. Maximum boron concentration increases to compensate for the change in reactivity due to the removal of CR D-6, with a magnitude of change of less than 40 ppm.	Based on the change for the key safety parameter, the NRC staff confirmed that the sensitivity impact is negligible to the removal of CR D-6. The NRC staff determined that the change in boron concentration is acceptable because there is an adequate amount of margin in relation to the SAFDL.
15.6.5	5.3.13	Loss of Coolant Accidents	Analysis parameters are not impacted.	The NRC staff confirmed that there are no impacts on the key safety parameters; therefore, the NRC staff determined the removal of CR D-6 has no impact on this accident analysis.
15.7	N/A	Radioactive Release From a Subsystem or Component	Analysis parameters are not impacted. Control rods are not assumed in any of these events.	The NRC staff confirmed that the bounding assumptions are not impacted with the removal of CR D-6 because the accident does not consider control rods in the analysis. Thus, the NRC staff determined that this accident is not impacted for long-term operation of STP-1 with CR D-6 removed.
15.8	N/A	Anticipated Transients Without SCRAM [ATWS]	The generic nature of the 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. The event is terminated without safety or protection systems. Therefore, the permanent removal of CR D-6 does not affect inputs used in generic ATWS analysis.	The NRC staff confirmed that the ATWS analysis assumes that CRs do not insert and, thus, are not credited. The NRC staff confirmed that the bounding assumptions are not impacted with the removal of CR D-6 and, therefore, the analysis remains valid for long-term operation with CR D-6 removed.

3.5.1 Control Rod Ejection Accident

The CREA is defined as a mechanical failure of a CR mechanism pressure housing that results in the ejection of a CR and drive shaft. For the CREA, the ejected rod additionally ejects another rod, and thus both rods will not insert, causing an N-2 situation. This assumption by STPNOC of N-2 is a conservative assumption that is beyond what is required by the NRC, which only requires N-1. The result of the ejection of a CR and drive shaft is a rapid positive reactivity insertion coupled with an adverse core power distribution. This reaction can lead to localized fuel rod damage. The analysis of the CREA are done both at beginning and end of the cycle at zero power and full power. The acceptance criteria for this event is to demonstrate that the core can be brought to a subcritical condition (i.e., $k_{eff} \leq 0.999$), which is accomplished by showing that the N-2 rod worth (sum of ejected rod worth plus the highest adjacent rod worth) is less than the combination of the SDM and 90 percent of the worst stuck rod worth. The difference between the SDM and 90 percent of the worst stuck rod and the N-2 rod worth is the N-2 subcriticality margin.

Note that the CREA also constitutes a break in the reactor coolant boundary, which is considered a LOCA. The LOCA impacts related to the CREA are discussed at the end of this section.

By letter dated September 28, 2016, the licensee stated that the removal of CR D-6 impacts two key safety parameters: SDM and trip reactivity. The SDM is impacted which impacts the N-2 subcriticality margin. For the licensee to confirm that the acceptance criteria is met, the N-2 rod worth is shown to be less than the actual SDM plus 90 percent of the worst stuck rod worth, which is conservative. With the removal of CR D-6, the N-2 rod worth increases and the SDM decreases, thus reducing the margin between the two parameters. The licensee provided the impact of removing the CR D-6 on the N-2 subcriticality margin for previous cycles and for the current cycle (Cycle 20). While there is a change in the amount of margin when removing CR D-6, the licensee demonstrated that the current and previous cycles still had adequate margin to meet the acceptance criteria.

Trip reactivity as a function of rod position is decreased because the shutdown banks are assumed to insert into the reactor core following a reactor trip on turbine trip. The change in the impact from CR D-6 present and removed was less than 2 percent for Cycle 20, which continued to meet the key safety parameter value for STP-1.

The licensee noted that the following key safety parameters are not impacted by the removal of CR D-6 because the shutdown banks are assumed to be withdrawn when determining these parameters:

- Doppler Power Coefficient,
- Moderator Temperature Coefficient,
- Delayed Neutron Fraction,
- Initial Hot-Spot Fuel Temperature,
- Ejected Rod Worth, and
- Ejected Rod Hot Channel Factor.

The NRC staff reviewed the parameters with respect to removal of CR D-6 and confirmed that these parameters are calculated under the worst-case conditions of the shutdown banks fully withdrawn and, thus, they are not impacted. Therefore, long-term operation without CR D-6 will not impact these parameters since they are calculated assuming CR D-6 is already removed from the core.

The key safety parameters, SDM and trip reactivity, are calculated on a cycle-specific basis using the NRC-approved methodology, the ANC. As discussed in Section 3.3, the NRC staff found that the ANC had the capability and was qualified to be able to model the removal of CR D-6. Since the licensee has demonstrated that STP-1 met those limits in previous and current cycles, and that the ANC has the capability and was qualified to model the removal of CR D-6, the NRC staff has determined that the licensee has adequately evaluated the CREA and has the ability to continue to evaluate the CREA for each future operating cycle.

In its review, the NRC staff determined that the impacts on the key safety parameters of SDM and trip reactivity for CREA continue to be bounded by the existing safety analysis. Since the CREA remains bounding with the removal of CR D-6 and the as-mentioned impacts on the listed key safety parameters, the NRC staff concludes that the licensee's analysis is acceptable.

The NRC staff next determined if, with the amendment granted, the licensee would continue to conform to GDC 10 as described in the UFSAR with the control rod ejection accident. Specifically, the staff considered whether adequate trip reactivity worth remains in place so that the shutdown banks and the control banks can shut down the reactor with adequate margin under normal operating conditions or anticipated operational occurrences. Adequate trip reactivity worth ensures that specified fuel design limits are thus not exceeded. As discussed in this section, the trip reactivity worth remains adequate to provide assurances that the consequences of postulated accidents are no more severe than those presented in the analyses of Chapter 15 with 56 control rods.

Additionally, adequate SDM is in place to preclude significant fuel damage during normal core operation and operational transients, or any transient conditions arising from occurrences of moderate frequency. The licensee demonstrated adequate SDM as previously discussed in this section of the SE. Accordingly, the staff found that the licensee would continue to conform with GDC 10 with respect to the control rod ejection accident.

In addition to the impacts discussed above, the ejection of a CR constitutes a break in the reactor coolant boundary with the effects and consequences similar to that of a LOCA. The break location for a rod eject accident is not the limiting case for the design basis LOCA which is discussed in Section 15.6.5 of the STP-1 UFSAR. Additionally, the limiting LOCA calculation assumes a worst-case scenario with no credit for insertion of CRs to shut down the plant, thus the removal of CR D-6 has no impact, specifically, to the SDM. The NRC staff reviewed the inputs necessary to complete the LOCA analyses, and determined that none are impacted by the removal of CR D-6 such that the results of the LOCA analyses are affected.

3.5.2 Hot Zero Power Main Steam Line Break

The UFSAR Chapter 15 analysis for an MSLB is the most limiting cooldown transient, and thus analyzed at zero power with no decay heat and the largest steam pipe rupture to provide the worst-case scenario. An MSLB accident is when there is piping failure in the nuclear steam supply system. The steam release from the MSLB would result in an initial increase in steam flow, which decreases during the accident as the steam pressure decreases. The increased energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. The analysis of the MSLB assumes that one CR remains stuck and does not insert. This assumption increases the possibility that the core will become critical and return to power. A cycle-specific evaluation for HZP MSLB is performed as part of the fuel reload process.

The removal of the CR D-6 would impact this event as two RCCAs would remain stuck outside the core instead of one. This would increase the local power and likely decrease the margin to the departure from nucleate boiling (DNB) limit. While the licensee stated that the analysis of record did not change, it did not explain how the additional stuck rod was evaluated. Therefore, the licensee was asked to address how the analysis of record provided by RETRAN represented the removal of CR D-6, and how that output was used in conjunction with the ANC and VIPRE codes to determine the margin to DNB.

By letter dated September 28, 2016, STPNOC provided additional information on the analysis methodology. The RETRAN statepoint is used along with cycle-specific information (e.g., the removal of the CR D-6 and an asymmetric cooldown assumption) as inputs to the ANC which then calculates the power distribution resulting from an HZP MSLB. The limiting stuck RCCA location is also determined considering the asymmetric cooldown assumption. The resulting power distribution from the most limiting ANC power search calculation is then used by VIPRE to determine the resulting heat flux of the event and ensure that the heat flux is below the heat flux which would cause DNB as predicted by the approved DNB correlation (including uncertainties). The NRC staff found that while the RETRAN analysis of record has not changed, an analysis is performed each cycle to confirm that the DNB limit for the HZP MSLB would remain below its applicable limit and that analysis would account for impacts due to the removal of the CR D-6.

3.6 Regulatory Criteria Compliance Summary

The NRC staff reviewed the application and supplemental information for the proposed permanent removal of CR D-6 for STP-1 to operate with 56 full-length CRs to meet NRC regulations. The NRC staff's scope was on the structural analysis, the inputs to the safety analyses that are impacted by the removal of CR D-6 such that they remain bounding for long-term operation, and on the NRC-approved methodologies such that they could account for asymmetric CR patterns.

The NRC staff confirmed that the UFSAR Chapter 15 analyses remain bounding with the removal of CR D-6 as discussed in Section 3.5 of this SE.

Sections 3.1.2.2.1 of STP-1 UFSAR, Rev. 4, describes how the plant satisfies and is in conformance with GDC 10, "Reactor Design." The NRC staff confirmed the change from 57 to 56 full-length control assemblies would not result in STP-1 operating outside of the evaluation against Criterion 10 documented in STP-1 UFSAR Section 3.1.2.2.1.1, and UFSAR Chapter 4 (discussing the design bases and design evaluation of reactor components including the fuel and reactivity control materials); UFSAR Section 3.9 (discussing the design bases and design evaluation of the reactor vessel internals and the control rod drive mechanisms); UFSAR

Chapter 7 (discussing details of the control and protection systems instrumentation design and logic); and UFSAR Chapter 15 (showing that the acceptable fuel design limits are not exceeded for Condition I and II occurrences, which are Defined by ANSI N18.2 – 1973).. Therefore, the licensee met the intent of GDC 10 for reactor design by demonstrating that adequate SDM and total rod worth can be calculated with CR D-6 removed using the ANC. The NRC staff confirmed that the ANC is an adequate code to assure the fuel design limits are not exceeded during any condition of normal operation, including anticipated operational occurrences as discussed in Section 3.4.1 of this SE.

Section 3.1.2.2.2 of STP-1 UFSAR, Rev. 14, describes how STPEGS nuclear power plant satisfies and is in conformance with GDC 11, "Reactor Inherent Protection." The NRC staff confirmed the change from 57 to 56 full-length control assemblies would not result in STP-1 being outside of the design basis of Criterion 11. The licensee met the intent of GDC 11 for reactor inherent protection by demonstrating that adequate boron worth can be calculated with CR D-6 removed using the ANC to maintain the negative reactivity feedback as discussed in Section 3.4.2.2 of this SE.

Section 3.1.2.3.7 of STP-1 UFSAR, Rev. 14, describes how the plant satisfies and is in conformance with GDC 26, "Reactivity Control System Redundancy and Capability." The NRC staff confirmed the change from 57 to 56 full-length control assemblies would not result in STP-1 being outside of the design basis of Criterion 26. The licensee met the intent of GDC 26 for reactivity control system redundancy and capability since the removal of CR D-6 does not impact the design of the plant which has independent reactivity control systems. The boron worth and trip reactivity provides independent reactivity control systems of different design principles (i.e., CRs and a reactivity control system capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes to assure acceptable fuel design limits are not exceeded).

Section 3.1.2.3.9.1 of STP-1 UFSAR, Rev. 14, describes how the plant satisfies and is in conformance with GDC 28, "Reactivity Limits." The NRC staff confirmed the change from 57 to 56 full-length control assemblies would not result in STP-1 being outside of the design basis of Criterion 28. Therefore, the licensee met the intent of GDC 28 reactivity limits since the trip reactivity maintains the appropriate limits on the potential rate of reactivity increase to assure that the effects of postulated reactivity 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 structure, or other reactor pressure vessel internals to impair significantly the capability to cool the core. The licensee demonstrated that the postulated reactivity accidents continue to meet the SAFDLs and the licensee has the ability to evaluate the SAFDLs each cycle using their reload methodology as discussed in Section 3.5 of this SE.

3.7 Technical Evaluation Conclusion

The NRC staff evaluated the licensee's proposed change for the STP-1 core to contain 56 full-length CR assemblies with no full-length CR assembly in core location D-6. Based on the considerations discussed above, the NRC staff determined that the proposed revision for STP-1 to operate with 56 full-length CR assemblies is acceptable for long-term operation. The amended design feature TS 5.3.2, which reflects the change to the facility, along with the existing provisions in license, provides reasonable assurance that the licensee will comply with the Commission's regulations and that the health and safety of the public will not be endangered.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Texas official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on July 19, 2016 (81 FR 46967). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

Principal Contributors: Matthew Hardgrove, NRR/DSS/SRXB
Josh Borromeo, NRR/DSS/SRXB
Josh Kaizer, NRR/DSS/SNPB
Ian Tseng, NRR/DE/EMCB

Date: December 21, 2016

December 21, 2016

Mr. G. T. Powell
Executive Vice President and CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 1 - ISSUANCE OF AMENDMENTS RE: PERMANENT OPERATION WITH 56 CONTROL RODS (CAC NO. MF7577)

Dear Mr. Powell:

The Commission has issued the enclosed Amendment No. 211 to Facility Operating License No. NPF-76 for the South Texas Project, Unit 1 (STP-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 7, 2016, as supplemented by letters dated May 25 and September 28, 2016.

The amendment revises TS 5.3.2, "Control Rod Assemblies," to allow the STP-1 core to contain only 56 full-length control rod assemblies with no full-length control rod assembly in core location D-6. The proposed amendment would allow permanent operation of STP-1 with 56 full-length control rods.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/
Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-498

Enclosures:
1. Amendment No. 211 to NPF-76
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION:
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MHardgrove, NRR/DSS/SRXB
JBorromeo, NRR/DSS/SRXB
JKaizer, NRR/DSS/SNPB

ADAMS Accession No.: ML16319A010 * by Safety Evaluation **by e-mail

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA	NRR/DSS/STSB/BC**	NRR/DSS/SRXB/BC*	NRR/DSS/SNPB/BC*
NAME	LRegner	JBurkhardt	AKlein (RGrover for)	EOesterle	RLukes
DATE	11/14/2016	11/15/2016	12/16/2016	10/28/2016	10/28/2016
OFFICE	NRR/DE/EMCB/BC*	OGC (NLO)*	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM	
NAME	JQuichocho (RHsu for)	DRoth	RPascarelli	LRegner	
DATE	11/01/2016	12/15/2016	12/21/2016	12/21/2016	

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