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

August 26, 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 - SUMMARY OF JUNE 28-30, 2016, REGULATORY AUDIT AND REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 5.3.2 TO ALLOW LONG-TERM OPERATION WITH 56 FULL-LENGTH CONTROL ROD ASSEMBLIES (CAC NO. MF7577)

Dear Mr. Powell:

By letter dated April 7, 2016, as supplemented by letter dated May 25, 2016, STP Nuclear Operating Company, the licensee for South Texas Project, Unit 1, submitted a license amendment request (LAR) to allow long-term operation of Unit 1 with 56 full-length control rod assemblies instead of the originally designed 57 control rods.

The U.S. Nuclear Regulatory Commission (NRC) staff determined that a regulatory audit, conducted in accordance with the Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits," would be useful to better understand the LAR and more efficiently make a regulatory decision. Accordingly, the NRC staff conducted a regulatory audit on June 28-30, 2016, at the Westinghouse Corporation offices in Rockville, Maryland. The audit included the examination of supporting calculations and bases documents to verify information and identify material to be docketed to support the basis for the NRC staff's regulatory decision on the LAR.

Enclosure 1 to this letter describes the results of the NRC staff's audit. During the audit, the NRC staff identified key technical issues and e-mailed a draft request for additional information (RAI) to your staff on August 10, 2016. Enclosure 2 to this letter includes the final RAI. The RAI was discussed with Mr. Drew Richards of your staff on August 18, 2016, and a mutually agreed upon date for the RAI response was determined to be September 29, 2016.

If you have any questions, please contact me by telephone at 301-415-1906 or by e-mail at lisa.regner@nrc.gov.

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. Audit Summary
2. Request for Additional Information

cc w/encls: Distribution via Listserv

SUMMARY OF JUNE 28-30, 2016, REGULATORY AUDIT

REVIEW OF LICENSE AMENDMENT REQUEST TO REVISE

TECHNICAL SPECIFICATIONS TO ALLOW LONG-TERM OPERATION

WITH 56 FULL-LENGTH CONTROL ROD ASSEMBLIES

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1.0 BACKGROUND

By letter dated April 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16110A297), as supplemented by letter dated May 25, 2016 (ADAMS Accession No. ML16162A196), STP Nuclear Operating Company (STPNOC), the licensee for South Texas Project (STP), Unit 1, submitted a license amendment request (LAR) to revise the licensing bases to allow operation with 56 full-length control rod assemblies.

The Unit 1 reactor has been operating since December 2015 with 56 full-length control rod assemblies following U.S. Nuclear Regulatory Commission (NRC) approval of an emergency license amendment request dated December 3, 2015.[1] The April 6, 2016, submittal requests long-term operation in the 56-control-rod configuration instead of the originally-designed 57-control-rod configuration.

The NRC staff conducted a regulatory audit of the STPNOC LAR and supporting documentation on June 28-30, 2016, to gain a better understanding of the licensee's request. The audit was conducted at Westinghouse Corporation offices in Rockville, Maryland, to review the STPNOC reload design change process calculations, safety analyses calculations, and shutdown margin calculations which form the bases of the statements contained in the LAR.

The following NRC staff members participated in the audit:

- Lisa Regner – Audit Lead, Project Manager
- Eric Oesterle – Technical Lead, Branch Chief
- Jeremy Dean – Technical Lead, Branch Chief
- Joshua Kaizer – Technical Reviewer, Reactor Engineer
- Matthew Hardgrove – Technical Reviewer, Reactor Systems Engineer
- Joshua Borromeo – Technical Reviewer, Reactor Systems Engineer
- George Thomas – Technical Reviewer, Senior Reactor Systems Engineer
- Ian Tseng – Technical Reviewer, Mechanical Engineer

[1] The NRC staff's safety evaluation dated December 11, 2015, is in ADAMS at Accession No. ML15343A128.

STPNOC was represented by the following personnel:

- Lance Sterling – STPNOC
- Drew Richards - STPNOC
- Charles Albury – STPNOC
- Nathan Hall – STPNOC
- Brian Guthrie – Westinghouse Corporation
- Danielle Schmitt– Westinghouse Corporation

The audit facilitated an expedited review of the LAR and helped the NRC staff develop a clear understanding of the information provided by the licensee. The audit was conducted in accordance with the guidance in the NRR Office Instruction LIC-111 and consistent with the draft audit plan dated June 6, 2016 (ADAMS Accession No. ML16159A023).

During the audit, the NRC staff identified technical issues and generated a draft request for information (RAI), which was e-mailed to STPNOC staff on August 10, 2016 (ADAMS Accession No. ML16228A002). The finalized version of the RAI is included as Enclosure 2 to the letter transmitting this audit summary.

2.0 TECHNICAL ISSUES REVIEWED

The basis of this audit was the LAR provided by STPNOC requesting operation of STP Unit 1 with 56 full-length control rod assemblies, and Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR).

During the audit, the STPNOC staff provided a history of the rod control cluster assembly (RCCA) malfunction during refueling outage 1RE17 in November 2012 and 1RE19 in November 2015. Westinghouse staff provided an overview of the "Westinghouse Methodology Impacts from the Removal of the RCCA from Core Location D-6," which is available in ADAMS at Accession No. ML16188A368. The STPNOC team provided a summary of the differences between the emergency LAR and the permanent LAR, including the removal of the thimble plug in accordance with the As Low As Reasonably Achievable (ALARA) program, the permanent instrumentation and control changes, and the additional analyses discussed below. Additionally, the STPNOC staff discussed changes to the submittal which would be addressed in the response to the NRC's RAI. The changes include removal of the dropped bank during full power operations event in LAR Table 4, and the addition of missing legacy items to LAR Table 7.

The focus of the NRC staff's audit was to ensure that the licensee identified the inputs to the safety analysis that were impacted by the control rod removal, ensure that these impacts were adequately addressed, and to ensure that the methods used to analyze the control rod removal had the capability to model the core in this configuration.

As discussed in the Westinghouse presentation, the safety analysis codes use a point kinetics neutronics model without consideration of control rod pattern. Various parameters from the safety analyses of record are used as input into the core neutronics codes (i.e., ANC and PARAGON/NEXUS) which have the capability to model an asymmetric control rod pattern. The output from the neutronics codes provide the key safety parameters identified in WCAP-9272, which are compared to the key safety parameters used as input to the safety analyses. If the results from the neutronics codes are found to not bound those used in the safety analyses, an evaluation or complete re-analysis is completed for that safety analysis.

The NRC staff identified additional information needed to complete its review regarding how the licensee determined the shutdown margin (SDM). Specifically, the licensee was requested to provide a discussion of how the SDM was calculated for Cycle 20 and the multi-cycle assessment. This concern was captured in question 1 of Enclosure 2.

The NRC staff also identified additional information needed in LAR Table 7. Specifically, the licensee was requested to provide additional details on the impacts to specific accident analyses for key safety parameters. This concern was captured in question 2 of Enclosure 2.

The NRC staff noted during the discussion that the licensee evaluates the total rod worth on a cycle-specific basis each operating cycle. The NRC and STPNOC staff discussed how the licensee evaluated the impact to total rod worth in relation to the permanent removal of control rod D-6, and the impact to the design basis analyses in the UFSAR. This concern was captured in question 3 of Enclosure 2.

The NRC staff requested additional information on why key safety parameters in LAR Table 5 were included in the STP LAR since they are not incorporated into the WCAP-9272-P-A methodology. These appear to be important, and the licensee stated they are calculated regardless of the WCAP-9272-P-A methodology. This concern is captured in question 4 of Enclosure 2.

The NRC staff identified a concern with the use of the neutronics codes for the core configuration proposed by STPNOC (i.e., one control rod removed at location D-6). Specifically, the NRC staff requested verification that the neutronics codes had the capability to adequately model the N-2 configuration for a rod eject accident. The NRC staff also discussed its concern with a local return to power in a specific region of the core with the one rod removed and another stuck rod. This concern was captured in question 5 of Enclosure 2.

The NRC staff identified a concern with the main steam line hot zero power methodology. The postulated accident most susceptible to being impacted by the removal of the control rod is the main steam line break hot zero power accident. The NRC staff asked for clarification of the method used to analyze this accident and asked the licensee to demonstrate how WCAP-9272 was implemented to evaluate this accident on a cycle specific basis. The staff was specifically concerned about where in the process the removal of the control rod was captured and how that was evaluated. This concern was captured in question 6 of Enclosure 2.

3.0 DOCUMENTS REVIEWED

- Reload Safety Analysis Checklist (RSAC) Transmittals (Cycle 20)
- RSAC Violations (Cycle 20)
- Reload Safety Evaluation (RSES) for each safety analysis group
- SDM Analysis (Cycle 20 and Multi-Cycle Assessment)
- Design Initialization (DI) Meeting Minutes: NF-TF-15-36

- RSAC Preparation and Evaluation Guidance for Transient Analysis: NONLOCA-SAS-17.0
- RSAC-HFP and HZP SLB Calculations for South Texas Unit 1 (TGX) Cycle 20: CN-TG20-013
- Redesign – RSAC – HZP SBL, Trip Reactivity versus Power and Most Positive MDC for South Texas Unit 1 (TGX) Cycle 20: CN-TG20-053
- Thermal Hydraulic Design RSAC Confirmation for South Texas Unit 1 Cycle 20: CN-TG20-025
- METCOM – Section 6.22 Steam Line Break Analysis, Revision 71, March 2014-Design Initialization (DI) Meeting Minutes: NF-TF-15-36

4.0 CONCLUSION

The NRC staff found that the audit helped the staff to better understand certain aspects of the licensee's submittal, and to clarify several NRC staff concerns and questions. There was open communication throughout the audit and it was conducted in accordance with the draft audit plan with no known deviations.

In Enclosure 2 to the letter transmitting this audit summary, there is the final RAI to the licensee resulting from the audit. The NRC staff may have additional questions once the detailed review of the LAR and supplements is complete.

Attachment:
Audit Report Questions

REQUEST FOR ADDITIONAL INFORMATION

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1. In the license amendment request (LAR) dated April 7, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16110A297), as supplemented by letter dated May 25, 2016 (ADAMS Accession No. ML16162A196), STP Nuclear Operating Company (STPNOC, the licensee) stated, in part, that the required reactor coolant system shutdown margin (SDM) boron concentrations for Operating MODES 3, 4, and 5 will be higher with control rod D-6 removed. Additionally, Table 8 of the LAR provides a summary of the SDM calculated at the end of cycle for the four representative cycles performed for the limiting hot zero power (HZP) main steam line break accident. It is unclear to the U.S. Nuclear Regulatory Commission (NRC) staff how the SDM was determined. The calculated SDM is evaluated for each core reload design to satisfy the General Design Criterion (GDC) 28, "Reactivity limits," of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. The NRC staff requests that the licensee provide a discussion on how the SDM was calculated for Cycle 20 and the multi-cycle assessment.

2. Table 7 of the LAR discusses the impact of control rod D-6 on the key safety parameters related to Updated Final Safety Analysis Report (UFSAR) Chapter 15 safety analyses. The discussion only references to the bounding UFSAR Chapter 15 analyses with no further discussion of how the removal of the control impacts the inputs/assumptions of the analyses. The NRC staff requests that the licensee provide the following:

    a. For each Chapter 15 analysis:

        1. Please discuss how consideration of control rod D-6 was previously incorporated into each accident analysis (e.g., control rod D-6 was part of the shutdown bank that was inserted into the core following reactor trip initiated by a turbine trip).

        2. Please discuss how the removal of the control rod impacts the key safety parameters (e.g., since the shutdown banks are assumed to insert during this event, the overall trip reactivity is decreased with the removal of control rod D-6).

        3. Please provide the basis for events that are not impacted by removal of control rod D-6.

        4. If there is an impact on the key safety parameters, please provide an estimate of the magnitude of the change to the key safety parameter.

b. For the control rod ejection accident, Table 7 identifies "various" as the sections of the reload methodology. Please specify these locations and identify the key safety parameters related to this accident (i.e., those impacted and not impacted by the control rod D-6 removal).

c. The NRC staff identified several discrepancies between the key safety parameters impacted identified in Table 7 and the key safety parameters identified for each accident in the reload methodology. Please discuss why there are differences between the documents as identified below (note that for this request for additional information, Table 7 of the LAR is abbreviated as Table 7):

   1. Feedwater System Malfunctions (reduction in feedwater temperature and increased feedwater flow): Table 7 identifies trip reactivity as impacted while the reload methodology does not identify trip reactivity as a key safety parameter.

   2. Loss of External Load and Turbine Trip: Table 7 identifies trip reactivity as impacted while the reload methodology does not identify trip reactivity as a key safety parameter. The reload methodology identifies moderator density coefficient (MDC) as a key safety parameter while Table 7 does not identify MDC as impacted.

   3. Feedwater System Pipe Break: Table 7 identifies trip reactivity as impacted while the reload methodology does not identify trip reactivity as a key safety parameter. The reload methodology identifies SDM as a key safety parameter while Table 7 does of the LAR does not identify SDM as impacted.

   4. Partial Loss of Forced Reactor Coolant Flow, Complete Loss of Forced Reactor Coolant Flow, and Rod Cluster Control Assembly (RCCA) Misoperation: The reload methodology identifies MDC as a key safety parameter while Table 7 does not identify MDC as impacted.

   5. Startup of Inactive Reactor Coolant Loop at an Incorrect Temperature: The reload methodology identifies MDC and SDM as a key safety parameter while Table 7 does not identify MDC and SDM as impacted.

d. In the column in Table 7, several comments state that an analysis is bounded by another. Please discuss the basis for why these analyses are bounded by another and confirm that these bounding assumptions are unchanged with the removal of the control rod.

3. The licensee stated, in part, in Table 3 of the LAR that the total rod worth is evaluated on a cycle-specific basis to ensure that the SDM and trip reactivity limits are met. It is unclear to the NRC staff whether the licensee has evaluated the influence and impact of total rod worth on control rod D-6 in relation to the UFSAR Chapter 15 analyses. The calculated total rod

worth is evaluated for each core reload design to satisfy GDC 28, "Reactivity limits." The NRC staff requests that the licensee provide clarification of the removal of control rod D-6 on total rod worth and the parameters in relation to the UFSAR Chapter 15 analyses.

4. Table 5 of the LAR contains additional nuclear design key safety parameters that are not part of the reload methodology. Please clarify if these additional parameters in Table 5 have been analyzed in past reloads. Also, please clarity if these additional parameters in Table 5 have been incorporated into the reload guidance such that they will be analyzed in future reloads.

5. GDC 10, "Reactor design," of Appendix A to 10 CFR Part 50 states that 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. Such margin is commonly demonstrated using computational models to simulate how the system would behave during normal operation and anticipated operational occurrences. Because the results of these simulations are used to confirm that such margin exists, the simulations themselves and the computer models which are used to perform them must be trustworthy.

   The Advanced Nodal Code (ANC) is used to perform analyses for these scenarios and will have a change in inputs due to the removal of the D-6 control rod. Please provide justification for the continued use of ANC with the removal of the D-6 control rod. This should include a demonstration that any change to the simulations considered (i.e., N-1 to N-2 rods out) are within the capabilities of ANC and the scope of the initial approval of ANC.

6. For the Hot Zero Power Main Steam Line Break (HZP MSLB), STPNOC uses multiple computer codes to simulate the scenario. Please provide further details on the methodology for performing the HZP MSLB analysis. Specifically, address how the analysis of record, generated by RETRAN, was used in conjunction with ANC and VIPRE to ensure that there was margin to DNB.

If you have any questions, please contact me by telephone at 301-415-1906 or by e-mail at lisa.regner@nrc.gov.

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

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1. Audit Summary
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DATE	8/24/16	8/26/16	8/26/16

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