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

March 15, 2016

Mr. Dennis L. Koehl
President and CEO/CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 1 – STAFF AUDIT SUMMARY RELATED TO REQUEST FOR EMERGENCY LICENSE AMENDMENT TO ALLOW OPERATION FOR ONE CYCLE FOLLOWING REMOVAL OF ONE INOPERABLE CONTROL ROD (CAC NO. MF7142)

Dear Mr. Koehl:

By letter dated December 3, 2015, the STP Nuclear Operating Company (STPNOC, the licensee) submitted an emergency license amendment request to allow operation of South Texas Project, Unit 1 for Operating Cycle 20 following the removal of one control rod assembly.

The U.S. Nuclear Regulatory Commission (NRC) staff conducted a regulatory audit at the Westinghouse Corporation Offices in Rockville, Maryland, on December 7 and 8, 2015, in order to gain a better understanding of the licensee's approach to implement a risk-informed evaluation of the effects of debris on emergency core cooling system and the containment spray system operation following a loss-of-coolant accident. A specific goal was to review the STPNOC reload design change process calculations and shutdown margin calculations which formed the bases of the claims made in the license amendment request.

The enclosure to this letter describes the results of the NRC staff's audit and some of the key technical issues highlighted by the staff during the audit. The NRC staff issued the emergency license amendment with the associated safety evaluation on December 11, 2015.

If you have any questions, please contact me at 301-415-1906 or via 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

Enclosure:
Staff Audit Summary

cc w/encl: Distribution via Listserv

STAFF AUDIT SUMMARY

EMERGENCY AMENDMENT REQUEST TO REVISE

TECHNICAL SPECIFICATIONS FOR ONE OPERATING CYCLE

OPERATION WITH 56 CONTROL RODS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1.0 Background

By application dated December 3, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15343A347), as supplemented by letter dated December 9, 2015 (ADAMS Accession No. ML15344A304), STP Nuclear Operating Company (STPNOC, the licensee) requested changes to the Technical Specifications (TSs), Appendix A to Facility Operating License No. NPF 76, for the South Texas Project (STP), Unit 1.

The proposed amendment would modify TS 5.3.2, "Control Rod Assemblies," by adding a footnote allowing the use of 56 full-length control rods for operating cycle Unit 1 Cycle 20, instead of 57 full-length control rods, and that there would be no full-length control rod in the assembly at core location D-6. This amendment was necessitated by emergent issues identified during control rod testing following completion of the recent refueling outage (1RE19). The proposed amendment would allow the plant to operate for one cycle, approximately 18 months, while repair plans are completed for the control rod drive mechanism (CRDM).

The licensee requested that the U.S. Nuclear Regulatory Commission (NRC) staff process this submittal under emergency circumstances as provided in Title 10 to the *Code of Federal Regulations* (10 CFR), Subsection 50.91(a)(5).

The NRC staff conducted a regulatory audit at the Westinghouse Corporation offices in Rockville, Maryland, on December 7 and 8, 2015, to review the STPNOC reload design change process calculations and shutdown margin calculations which form the bases of the claims made in the license amendment request (LAR).

The following NRC staff members participated in the audit:

- Lisa Regner – Audit Lead, Project Manager
- Jeremy Dean – Technical Lead, Branch Chief
- William MacFee – Technical Reviewer, General Engineer
- George Thomas – Technical Reviewer, Senior Reactor Systems Engineer
- Ian Tseng – Technical Reviewer, Mechanical Engineer

STPNOC was represented by the following personnel:

- Charles Alsbury – STPNOC
- Daniell Smidtt – Westinghouse Corporation

The audit facilitated an expedited review of the emergency LAR and to develop a clear understanding of the information provided by the licensee consistent with the Office of Nuclear Reactor Regulation Office Instruction LIC-111, *Regulatory Audits* (ADAMS Accession No. ML082900195). During the audit, the NRC staff identified technical issues and generated a request for information (RAI). The RAI was transmitted to the licensee by e-mails dated December 7 and 9, 2015 (ADAMS Accession Nos. ML16026A586 and ML16026A587, respectively), and is included in the attachment to this audit summary. The licensee provided responses to the RAIs by letter dated December 9, 2015 (ADAMS Accession No. ML15344A304).

By letter dated December 11, 2015 (ADAMS Accession No. ML15343A128), the NRC staff issued the amendment and associated safety evaluation allowing STP, Unit 1, to operate for Cycle 20 with 56 control rods.

2.0 Technical Issues Reviewed

Shutdown Margin

The NRC staff reviewed Westinghouse report WCAP-18075-P, Rev. 0, "Nuclear Design and Core Management of the South Texas Project 1 Nuclear Power Plant Cycle 20," which details the Nuclear Design and Core Management for STP, Unit 1, for Cycle 20, the plant's next operating cycle. The document provided the critical boron value (defined as the amount of boron necessary to have the reactor just critical or k=1), calculated at specific temperatures and fuel burnup. This value is used to determine the required shutdown margin, which then is used to calculate a value of minimum boron concentration to maintain the required shutdown margin specified in the core operating limits report (COLR). A shutdown margin value of 1.14 was determined, which provides the ratio of the critical boron concentration to minimum shutdown boron concentration.

The NRC staff discussed with STPNOC staff the boron worth impacts discussed in the submittal and if the boron worth changes were still bounded by the Updated Final Safety Analysis Report (UFSAR) analysis. The NRC staff reviewed the supporting documentation, Westinghouse calculation file, CN-TG20-049 (dated November 23, 2015), which provides the analysis of the limiting case analysis: a boron dilution accident. The NRC staff reviewed the analysis showing that the values assumed in the UFSAR accidents are not impacted by the change in critical boron concentrations for Cycle 20.

Moderator Density Coefficient

The LAR stated that the most positive moderator density coefficient was impacted by the proposed removal of the control rod D-6. Since some STP UFSAR Chapter 15 *Accident Analysis* design-basis accidents assume the most positive moderator density coefficient and others assume a lower moderator density coefficient, the NRC staff reviewed the extent of these impacts. The NRC staff reviewed the UFSAR Table 15.0-2 for the moderator density coefficient assumptions and method the licensee used to calculate the "least positive moderator density coefficient." This analysis is done under the conservative assumption of all rods fully withdrawn from the core; therefore, the least positive moderator density coefficient is not impacted with control rod D-6 removed.

The NRC staff reviewed the summary of design basis analyses impacted by the proposed removal of control rod D-6 listed in Table 4 of the LAR. The NRC staff discussed with STPNOC staff the submittal comments in Table 4 concerning the impacts on UFSAR Chapter 15 accidents.

The NRC staff questioned the licensee's comment that there was "no impact" on the UFSAR Chapter 15.1.3, "Excessive Increase in Secondary Steam Flow." The STPNOC staff clarified that the UFSAR section 15.1.3.2 showed that the event does not challenge the departure from nucleate boiling limits and therefore, the event was not analyzed in the core reload design change process.

The NRC staff questioned why the licensee did not discuss the impacts to moderator density coefficient, local core effects, and departure from nucleate boiling ratios in the UFSAR Chapter 15.1.5, "Spectrum of Steam System Piping Failures Inside and Outside Containment," in the LAR. The STPNOC staff provided the following supporting documentation for the NRC staff to review.

- CN-TG20-025 "THD RSAC [reload safety analysis checklist] Confirmation for South Texas Unit 1 Cycle 20"
- CN-TG20-013 "RSAC – HFP and HZP SLB Calculations for South Texas Unit 1 Cycle 20"
- CN-TG020-053 "Redesign- RSAC – HZP SLB, Trip Reactivity Shape, Trip Reactivity vs Power and Most Positive MDC for South Texas Unit 1 Cycle 20"

The assumptions used in the evaluation of the steam line break accident are listed in UFSAR Chapter 15.1.5.2, including the assumption that the most negative moderator temperature coefficient is used with the least negatively reactive rod stuck out. Since the most positive moderator density coefficient and shutdown margin are impacted, the NRC staff needed clarification on the values impacted. The LAR Table 4 stated that shutdown margin and all other analysis parameters remained bounded by the design basis analyses in the UFSAR. The licensee showed that most positive moderator density coefficient does impact the assumptions in the steam line break accident but that the revised value is bounded by the assumptions in the UFSAR.

The NRC staff generated RAI question 1 in the enclosure to this document to provide documentation that the licensee analyzed the impacts and found that the revised moderator density coefficient remained bounded by the UFSAR analysis.

Departure from Nucleate Boiling

The NRC staff reviewed the calculation files for the impacts to the departure from nucleate boiling value reported in the LAR. The calculation files detailed the assumptions, calculation steps, and results in determining if the safety limits were met. One portion of the Steam Line Break Analysis was a departure from the nucleate boiling analysis showing that the UFSAR limits were not challenged. The departure from nucleate boiling analysis showed why the ratio for the steam line break event decreased from 3.011 to 1.811. The NRC staff determined that more information was needed for the staff to complete its review, and RAI question 2 in the enclosure to this document was written.

Rod Ejection Analysis

The licensee provided the following calculation files which re-analyzed the rod ejection event analysis with control rod D-6 removed:

- CN-TG20-019 "RSAC – Rod Ejection for South Texas Unit 1 (TGX) Cycle 20"
- CN-TG20-050 "Redesign – RSAC-SDM, Rod Ejection, and Trip Reactivity Following RWSC for South Texas Unit 1 (TGX) Cycle 20"

The licensee performed the analysis using the methodology in WCAP-9272-P-A which is the licensing basis methodology in STP's UFSAR. The calculation showed that the shutdown margin for subcriticality was impacted but still bounded for the event.

The NRC staff reviewed the Uncontrolled RCCA [rod cluster control assembly] Bank Withdrawal from a Subcritical and Low Power Condition event calculation files:

- CN-TG20-020 "RWAP/RWSC/Trip Reactivity for South Texas Unit 1 (TGX) Cycle 20"
- CN-TG20-050 "Redesign – RSAC-SDM, Rod Ejection, and Trip Reactivity Following RWSC for South Texas Unit 1 (TGX) Cycle 20"

The NRC staff reviewed the calculation file assumptions and initial conditions stated in the UFSAR as compared with the reload methodology.

Flow Restrictor

The licensee stated in the LAR that it would use a flow restrictor to simulate the flow effects of a control rod in the upper guide tube once the control rod is removed. The NRC staff reviewed the thermal-hydraulic equivalency of the upper internal flow restrictor using the Vendor Technical Information Document, LTR-RID-15-250, "South Texas Unit 1 – Upper guide Tube Top Plate Flow Restrictor Installation Evaluation" (STP Document number 0042-0100240WN). The technical document discussed the equivalency of the thermal hydraulic properties between the flow through annulus between the drive rod and the guide tube and the flow through the flow restrictor without the drive rod. The NRC staff also reviewed the design, analysis, material properties, and installation procedure for the flow restrictor installed at the top of the D-6 guide tube (STP Document numbers 0042-0100234WN, 0042-0100239WN, 0042-0100240WN, and FCN-TGX-40549), which provides the licensee's basis that the flow restrictor is structurally adequate and describes the design features to prevent the generation of loose parts.

The NRC staff generated RAI question 5 in the enclosure to this document to provide documentation of the design features and materials used for fabrication, which will help the NRC staff to assess whether the flow restrictor is securely installed to prevent the generation of loose parts, and whether thermal expansion and material compatibility are concerns.

Dynamic Analysis

Removal of the control rod D-6 drive shaft and 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. The NRC staff reviewed documentation on the Reactor Equipment System Model to assess whether the the current dynamic analyses remains valid after the removal of the control rod D-6 drive shaft and RCCA.

The NRC staff generated RAI question 6 in the enclosure to this document to provide documentation that current dynamic analysis of the CRDM was performed by the licensee using a model that does not include the weight of the control rod drive shaft in the CRDM assembly. The NRC staff reviewed this to assess the consistency between the model and the proposed configuration and to determine whether the current dynamic analyses remain valid.

3.0 Conclusion

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

The NRC discussed with STP staff that the review of this emergency amendment was conducted with consideration that it could allow operation for one cycle only if found acceptable. The NRC staff noted that if STPNOC requested to operate for a longer period without the D-6 control rod and drive shaft, the NRC staff considers this a configuration change and additional staff review may be required due to the change in scope.

In the Attachment to this document, the NRC staff provided questions to the licensee, which the licensee agreed to answer on the docket so the NRC staff could complete its review.

Attachment:
Audit Report Questions

AUDIT REPORT QUESTIONS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1. Confirm that the most positive moderator density coefficient remains bounding for the moderator feedback effects assumed in Chapter 15.1.5, "Spectrum of Steam System Piping Failures Inside and Outside Containment."

2. For the Departure from Nucleate Boiling analysis related to Steam Line Break, provide more detailed information on why the departure from nucleate boiling ratio changed from 3.011 to 1.811.

3. Provide the impacts to the reactor protective system from the modifications to the Digital Rod Position Indication system associated with the removal of control rod D-6.

4. Provide the impacts to operator actions or emergency operating procedures as a result of the removal of control rod D-6.

5. In order for the staff to assess the potential of the proposed flow restrictor generating loose parts, provide a description of any relevant design features of the flow restrictor, and justify how the thermal expansion of the flow restrictor and its component parts has been addressed.

6. In order for the staff to verify the structural adequacy of the D-6 Control Rod Drive Mechanism (CRDM) housing without the drive rod when subjected to Loss of Cooling Accident or seismic excitations, provide a description of relevant analyses that were performed to model the CRDM without the mass of the drive rod.

7. On page 19 of 22 of your December 3, 2015, emergency amendment request, you state that 10 CFR 50.62(c)(3) concerning an alternate rod injection system as stated in Standard Review Plan, Section 4.6 are for boiling water reactors and do not apply to the South Texas Project Unit 1. The staff agrees that the BWR portion of the Anticipated Transient Without Scram (ATWS) rule, 10 CFR 50.62(c)(3), does not apply; however, the remainder of the ATWS rule, 10 CFR 50.62, should be considered by STP since it does apply to pressurized water reactors. Provide the impact on ATWS for STP considering the removal of control rod D-6

Attachment

If you have any questions, please contact me at 301-415-1906 or via 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

Enclosure:
Staff Audit Summary

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ADAMS Accession No.: ML16029A272 * by email

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DATE	3/2/16	2/5/16	3/14/16
OFFICE	NRR/DE/EMCB/BC*	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME	TLupold	RPascarelli	LRegner
DATE	3/3/16	3/14/16	3/15/16

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