Nuclear Operating Company

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

December 9, 2015
NOC-AE-15003318
10 CFR 50.90

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

South Texas Project
Unit 1
Docket No. STN 50-498
Response to Request for Additional Information and Supplement to
South Texas Project (STP) Unit 1 Emergency License Amendment Request to
Revise Technical Specification 5.3.2 to Allow Operation with
56 Full-Length Control Rod Assemblies for Unit 1 Cycle 20

References:

1. Letter; G. T. Powell to USNRC Document Control Desk; "Emergency License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1 Cycle 20;" NOC-AE-15003315; dated December 3, 2015.
2. E-mail; L. Regner to W. Brost, C. Albury; "DRAFT Request for Additional Information - Emergency Amendment;" dated December 8, 2015.
3. E-mail; L. Regner to D. Richards; "Additional STP RAI for Emergency CR Amendment;" dated December 9, 2015.

By Reference 1, STP Nuclear Operating Company (STPNOC) requested approval of an emergency license amendment to Technical Specification (TS) 5.3.2 to require the Unit 1 Cycle 20 core to contain 56 full-length control rods with no full-length control rod assembly in core location D-6. By References 2 and 3, the NRC staff sent requests for additional information (RAIs) to complete its review. STPNOC's response to references 2 and 3 is provided in Attachment 1 to this letter.

The No Significant Hazards Consideration determination provided in the Enclosure to Reference 1 has been revised and is provided in Attachment 2 to this letter. Please replace Section 4.3 of the Enclosure to Reference 1 in its entirety with the information provided in Attachment 2. The revised No Significant Hazards Consideration determination has been reviewed and approved by the STPNOC Plant Operations Review Committee and has undergone an independent Organizational Unit Review.

STI: 34250901

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There are no commitments in this letter.

If there are any questions or if additional information is needed, please contact Drew Richards at (361) 972-7666 or me at (361) 972-7566.

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

Executed on December 9, 2015

G. T. Powell

G. T. Powell
Site Vice President

amr/GTP

Attachments:

1. Response to Request for Additional Information
2. Revised No Significant Hazards Consideration determination

cc:
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Attachment 1

Response to Request for Additional Information

RAI #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

STP Response

The Moderator Density Coefficient (MDC) used in the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analyses is described on Table 15.0-2, which refers to Figure 15.0-6 for specific accidents. The MDC for steam line break analysis differs from the values on UFSAR Figure 15.0-6 in that the steam line break MDC is variable based on detailed neutronics calculations for the steam line break configuration as opposed to the limiting values presented in Figure 15.0-6. Using the NRC approved methodology in WCAP-9272-P-A for the main steam line break analysis for a reload core, a detailed Advanced Nodal Code (ANC) calculation is performed to confirm that the maximum reference state point power level calculated in the transient analysis remains applicable for the reload core. This ensures that the MDC assumed in the transient analysis remains applicable. For the STP Unit 1 Cycle 20 redesign, the maximum reference state point power level was found to be acceptable. Therefore, the MDC assumed in the transient analysis remains acceptable.

Based on the discussion above, the comment for item 5 of Table 4 in the Enclosure to the emergency license amendment request should be revised to state the following:

> "Shutdown margin remains bounding. *MDC remains acceptable.* All other analysis parameters *remain acceptable*."

RAI #2

For the Departure from Nucleate Boiling analysis related to Steam Line Break, provide additional information on why the Departure from Nucleate Boiling Ratio changed from 3.011 to 1.811.

STP Response

The change in the Departure from Nucleate Boiling Ratio (DNBR) from 3.011 to 1.811 due to the removal of Control Rod D-6 can be attributable to the increase in power associated with this change.

DNBR is defined as the ratio of the critical heat flux (CHF) predicted by the W-3 correlation to the actual heat flux:

$$DNBR = (q''_{DNB,predicted})/(q''_{actual})$$

For the cases with and without Control Rod D-6, the change in the predicted CHF is minor. However, the actual heat flux for the event changes significantly. The results of the Advanced Nodal Code (ANC) analysis show that the core power increases from 5.3% of rated thermal power to 9.2%. This is a percent increase of more than 70%. Though there are other factors, such as the radial and axial power distributions, that partially offset the impact of the core power increase, the actual heat flux still increases quite significantly. Because the actual heat flux is the denominator of the ratio, the final DNBR value decreases. As stated in the emergency license amendment request, the results of the analysis show that the DNB ratio was reduced from 3.011 for the analysis with Control Rod D-6 inserted to 1.811 with Control Rod D-6 removed, which is still above the limit of 1.495. Therefore, the results of the steam line break from zero power with Control Rod D-6 removed remains bounding.

RAI #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 D6.

STP Response

Modifications to the Digital Rod Position Indication (DRPI) system resulting from the removal of Control Rod D-6 will have no impact to the reactor protection system (RPS). DRPI is a non-safety related system that is independent of the rod control and reactor protection systems. DRPI measures the actual position of each control rod using a detector consisting of discrete coils mounted concentrically to the control rod drive pressure housing. Each detector is essentially a hollow tube with assemblies of coiled wire slipped over the tube and spaced out along its length. The coils are located axially along the pressure housing and magnetically sense the tip of the control rod drive shaft through its centerline. DRPI provides indication of control rod position and alarms to inform the Control Room operators of misaligned control rods or system malfunctions. A failure of DRPI will not result in the rod control or reactor protection systems failing to perform intended safety functions.

RAI #4

Provide the impacts to operator actions or emergency operating procedures as a result of the removal of control rod D6.

STP Response

There is no impact to operator actions or emergency operating procedures (EOPs) resulting from the removal of Control Rod D-6. An evaluation was performed and determined that no changes are required to EOP emergency boration values for a stuck rod. There are no changes required to the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accident analysis inputs to the EOPs and no new operator actions are created.

RAI #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.

STP Response

The upper guide tube flow restrictor assembly is manufactured from stainless steel, which is the same material as the guide tube, and is compatible with fluid conditions in the reactor vessel upper head. Because the flow restrictor assembly and the guide tube are both the same material, there will be no differential thermal expansion. The installation procedure for the flow restrictor ensures that the specified hex bolt preload is obtained, securely locking the flow restrictor in place at the top of the guide tube. A locking cup, which is tack welded to the flow restrictor, is crimped onto the hex bolt to prevent hex bolt rotation. The capture features of the flow restrictor (i.e., locking fingers, hex bolt lock cup, hex bolt preload) provide assurance that the flow restrictor is securely installed and will not result in the generation of loose parts.

RAI #6

In order for the staff to verify the structural adequacy of the D6 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.

STP Response

The current control rod drive mechanism (CRDM) housing structural analysis uses a reactor system model which includes a detailed representation of the reactor internals and the mass of the head assembly appropriately lumped at the vessel head center of gravity (CG). Attached to this model at the head CG are three detailed representations of the CRDMs: a long; medium; and short CRDM, with the length determined by the head penetration adapter and the CRDM location on the domed vessel head. These CRDM elements supply the loads used in the CRDM housing structural analysis.

The dynamic analysis of the CRDM was performed using the reactor equipment system model (RESM). The total weight of the CRDM used in the RESM does not include the weight of the control rod drive shaft in the CRDM assembly. Removal of the D-6 RCCA and control rod drive shaft will have no impact on the CRDM model. Therefore, the RESM remains valid after removal of the D-6 RCCA and control rod drive shaft. In summary, the CRDM dynamic stress evaluation (due to seismic and loss of coolant accident excitations) remains valid after the removal of the D-6 RCCA and control rod drive shaft.

RAI #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 D6

STP Response

Removal of Control Rod D-6 does not impact either the reactor protection system or ATWS Mitigation System Actuation Circuitry. As stated in Table 4 of the emergency license amendment request (LAR), the trip reactivity remains bounding. The changes to other parameters described in the LAR do not impact the ATWS analysis. The all-rods-out moderator temperature coefficient is not impacted by removal of Control Rod D-6. Therefore, the requirements of 10 CFR 50.62(c)(1) continue to be met and there is no impact to the ATWS analysis.

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

Attachment 2

Revised No Significant Hazards Consideration determination

4.3 No significant hazards consideration determination

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

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

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

   Response: No. Operation of STP Unit 1 Cycle 20 with Control Rod D-6 removed will not involve a significant increase in the probability or consequences of an accident previously evaluated. STPNOC has evaluated the reactivity consequences associated with removal of Control Rod D-6 and determined that the amount of shutdown margin would be reduced but remains bounded by the limit provided in the Core Operating Limits Report. The impact on adjacent control rod worth was also evaluated and there is an increase in the rod worth of the most reactive stuck rod assumed in some accident analyses; however, the Updated Final Safety Analysis Report (UFSAR) accident analysis limits continue to be met. The probability of occurrence of a previously evaluated accident is not impacted by removal of Control Rod D-6. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No. Operation of STP Unit 1 Cycle 20 with Control Rod D-6 removed will not create the possibility of a new or different kind of accident from any accident previously evaluated. To preserve the reactor coolant system flow characteristics in the reactor core, a flow restrictor will be installed at the top of the D-6 guide tube housing and a thimble plug will be installed on the fuel assembly located in core location D-6. Installation of these components will not prevent the remaining 56 control rods from performing the required design function of providing adequate shutdown margin. No new operator actions are created as a result of the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No. Operation of STP Unit 1 Cycle 20 with Control Rod D-6 removed will not involve a significant reduction in a margin of safety. The margin of safety is established by setting safety limits and operating within those limits. The proposed change does not alter a UFSAR design basis or safety limit and does not change any setpoint at which automatic actuations are initiated. STPNOC has evaluated the impact of the proposed change on available shutdown margin, boron worth, rod worth, trip reactivity as a function of time, and the most positive moderator density coefficient; the results of these evaluations show that the proposed change does not exceed or alter a design basis or safety limit. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, STPNOC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.