STP Nuclear Operating Company

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

May 25, 2016
NOC-AE-16003382
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
Supplement to License Amendment Request to Revise Technical Specification 5.3.2
to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1 (CAC No. MF7577)

References:

1. Letter; J. Connolly to NRC Document Control Desk, "License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1"; April 7, 2016; (NOC-AE-16003351) (ML16110A297).
2. Letter; L. Regner to D. Koehl, "South Texas Project, Unit 1 - Supplemental Information Needed For Acceptance of Requested Licensing Action Re: Request To Operate Permanently With 56 Control Rods (CAC No. MF7577)"; May 12, 2016; (AE-NOC-16002853) (ML16127A452).

By Reference 1, STP Nuclear Operating Company (STPNOC) requested approval of a license amendment to Technical Specification 5.3.2 to require the Unit 1 core to contain 56 full-length control rods with no full-length control rod assembly in core location D-6. By Reference 2, the NRC requested supplemental information necessary to enable the staff to make an independent assessment regarding the applicability of the proposed license amendment. STPNOC is providing the requested supplemental information as an Enclosure to this letter. Also included in the Enclosure is a correction to information provided in Table 4 of Reference 1 regarding the "dropped bank during full power operations" event.

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) 972-7344.

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

Executed on May 25, 2016

James Connolly
James Connolly
Site Vice President

amr/JWC

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

A001
NRR

STI: 34321383

cc:
(paper copy)

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

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

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

(electronic copy)

Morgan, Lewis & Bockius LLP
Steve Frantz, Esquire

U.S. Nuclear Regulatory Commission
Lisa M. Regner

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

CPS Energy
Kevin Pollo
Cris Eugster
L. D. Blaylock

Crain Caton & James, P.C.
Peter Nemeth

City of Austin
Elaina Ball
John Wester

Texas Dept. of State Health Services
Helen Watkins
Robert Free

ENCLOSURE

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

STPNOC correction regarding "dropped bank during full power operations" event

In Table 4 of the Enclosure to the original LAR ("License Amendment Request to Revise Technical Specification 5.3.2 to Allow Operation with 56 Full-Length Control Rod Assemblies for Unit 1"; April 7, 2016; ML16110A297), the Maximum Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) for the "dropped bank during full power operations" event was identified as a key safety parameter that is confirmed each fuel cycle. This statement is incorrect for the reasons discussed below.

The "dropped bank during full power operations" event is not explicitly analyzed for STP because a dropped Rod Cluster Control Assembly (RCCA) bank is bounded by a single dropped RCCA (reference STP UFSAR Section 15.4.3.2.2). Per STP off-normal procedure 0POP04-RS-0001 (Control Rod Malfunction), control room operators will trip the reactor if more than one control rod drops into the reactor core while in Mode 1 or Mode 2. This is an immediate operator action which will be performed from memory by licensed reactor operators.

The line item in Table 4 for "Dropped bank during full power operations" in the original LAR should be deleted.

NRC Item #1

1. Provide an explanation of how the value for a bounding key safety parameter was initially determined for input into the safety analyses. For example, the bounding shutdown margin originally input into the safety analyses was 1.3 percent delta rho. Explain how the value of 1.3 was initially determined. The key safety parameters include the following:

   a. Moderator temperature coefficient/moderator density coefficient
   b. Shutdown margin
   c. Trip reactivity

   Confirm that if these key safety parameters are impacted by the removal of the control rod, then the new value for the key safety parameter would be rerun through the analysis to determine the new result.

STPNOC Response

Key safety parameter limits were generically defined in the 1970s for Westinghouse 2-loop, 3-loop, and 4-loop PWRs. The initial shutdown margin (SDM) limit for STP was 1.75% $\Delta\rho$. In 1992, the SDM limit was updated to 1.3% $\Delta\rho$ when the analyses of record were revised due to fuel product and peaking factor changes. This value was chosen because it bounds the SDM value for typical reload cores and it allows the acceptance criteria to be met. All subsequent reload cores since 1992 have met the 1.3% $\Delta\rho$ limit, including the STP Unit 1 Cycle 20 core with Control Rod D-6 removed. The same philosophy for determining key safety parameter values was used for other parameters including moderator temperature coefficient, moderator density coefficient, and trip reactivity; the specific values that were chosen were considered bounding for future cores and have resulted in satisfying acceptance criteria.

During the core reload design process, any key safety parameter violations are transmitted to the cognizant safety analysis group for evaluation. Collaboration between the impacted functional groups may identify additional margin that may be available to support evaluation or reanalysis of impacted analyses. If no margin is identified, a reanalysis is performed and the analysis of record is updated.

Following removal of Control Rod D-6, the STP Unit 1 Cycle 20 core design was evaluated and several key safety parameters were impacted. However, the impacted parameters remained bounded by the key safety parameter limits and no evaluations or revisions to the analyses of record were required.

NRC Item #2

2. Provide summaries of the evaluations performed from the supporting calculations and documentation for each of these design basis accident events analyzed and provide the reference number:

    a. Uncontrolled boron dilution accident
    b. Dropped bank during full power operations
    c. Steam line break accident
    d. Control rod ejection accident
    e. Steam generator tube rupture

STPNOC Response

a. Uncontrolled boron dilution accident (STP UFSAR Section 15.4.6)

For analysis in Modes 1 and 2, calculations confirm that sufficient operator action time is available between indication of a boron dilution event and a complete loss of shutdown margin. For analysis in Modes 3, 4, and 5, calculations confirm the minimum allowable operator action time between receipt of the flux multiplication alarm and a complete loss of shutdown margin. The key parameters for this analysis are as follows:

Modes 1 and 2

- Maximum critical boron concentration at hot zero power assuming all rods inserted with the most reactive control rod stuck out of the core (final boron concentration)
- Minimum critical boron concentration with control rods at the rod insertion limits (initial boron concentration)

Modes 3, 4, and 5

- Maximum critical boron concentration within the Mode-dependent temperature range assuming all rods inserted with the most reactive control rod stuck out of the core (final boron concentration)
- Maximum allowable differential boron worth as a function of critical boron concentration meeting the existing variable shutdown margin requirements
- Variable shutdown margin requirements as a function of critical boron concentration

Removal of Control Rod D-6 reduces by one the number of RCCAs inserted into the reactor core for the "N rods in" and the "N-1 rods in" configurations. The reduction in negative reactivity insertion increases the critical boron concentration with N-1 rods inserted and results in a reduction in margin to the maximum boron concentration limits associated with the uncontrolled boron dilution accident. Analysis of this event for the STP Unit 1 Cycle 20 core with Control Rod D-6 removed shows that the key safety parameter limits are not exceeded. The reload safety evaluation process confirms that the cycle-specific values remain bounded by the assumptions of the boron dilution safety analysis. Violation of any reload parameter would require evaluation or re-analysis of the boron dilution event prior to core reload.

References:

1. Westinghouse calculation, "Redesign – RSAC – Boron Dilution for South Texas Unit 1 (TGX) Cycle 20."
2. Westinghouse calculation, "South Texas Units 1 and 2 Boron Dilution Evaluation to Support 10% SGTP."

3. Westinghouse calculation, "South Texas Unit 1 D94 RSG: Boron Dilution."
4. Westinghouse calculation, "South Texas Unit 1 (TGX) Cycle 11 Reload Safety Evaluation."
5. Westinghouse calculation, "Miscellaneous Non-LOCA Evaluations in Support of 1.4% Power Uprating at South Texas Units 1 and 2 (TGX/THX)."

b. Dropped bank during full power operations

As discussed earlier in this Enclosure, the "dropped bank during full power operations" event is not analyzed for STP.

c. Steam line break accident (STP UFSAR Section 15.1.5)

Transient analysis of steamline break accident uses the RETRAN-02 code to develop transient values of reactor core average heat flux, reactor core pressure, vessel inlet temperature, reactor core flow rate, and reactor core boron concentration. Each of these "statepoints" are analyzed using the ANC computer code to calculate cycle-specific reactivities and radial power distributions. The departure from nucleate boiling ratio (DNBR) value is then calculated using the VIPRE-W code. The accident-specific key safety parameters include:

- Moderator density coefficient (modeled using stuck-rod coefficients),
- Doppler defect (modeled using stuck-rod coefficients),
- Boron worth (modeled using stuck-rod coefficients),
- Doppler temperature coefficient, and
- Shutdown margin.

Removal of Control Rod D-6 reduces by one the number of RCCAs inserted into the reactor core for the "N rods in" and the "N-1 rods in" configurations. This effectively results in an analysis of the steamline break for an "N-2" configuration. The increased perturbation of post-break power distribution and increase in post-break core reactivity results in a reduction of margin to the peaking factor and DNBR limits associated with the steamline break event. Analysis of this event for the STP Unit 1 Cycle 20 core with Control Rod D-6 removed shows that the key safety parameter limits were still met for this configuration. For future core designs, calculations will be performed each cycle to confirm that the key safety parameter limits are met.

References:

1. Westinghouse calculation, "Redesign - RSAC - HZP SLB, Trip Reactivity Shape, Trip Reactivity vs Power and Most Positive MDC for South Texas Unit 1 (TGX) Cycle 20."
2. Westinghouse calculation, "South Texas Unit 1 D94 RSG: Steamline Break – Core Response."
3. Westinghouse calculation, "South Texas Unit 1 D94 RSG: Steamline Break – Core Response - Plant-Specific HZP Analysis."
4. Westinghouse calculation, "South Texas Unit 1 D94 RSG: Updated Steamline Break – Core Response Plant-Specific HZP Analysis."
5. THD RSAC Confirmation for South Texas Unit 1 Cycle 20.

d. Control rod ejection accident (STP UFSAR Section 15.4.8)

This event is analyzed for four different conditions: beginning-of-life and end-of-life with each at hot full power (HFP) and hot zero power (HZP). Transient analysis of the control rod ejection event is performed using the TWINKLE (WCAP-7979-P-A) and FACTRAN computer codes. The TWINKLE code is used to perform a one-dimensional calculation to determine core power considering various total core feedback effects (e.g., Doppler reactivity and moderator reactivity). The TWINKLE results are used as inputs to FACTRAN, which is a detailed fuel and cladding transient heat transfer code, to calculate hot-spot fuel enthalpy and temperature transient.

The cycle-specific analysis of this event determines the ejected rod worth and hot channel factor following ejection of an RCCA from the applicable rod insertion limits during power operation in Modes 1 and 2. Control Rod D-6 is in a shutdown bank and shutdown banks are fully withdrawn during operation in Modes 1 and 2; therefore, removal of Control Rod D-6 does not impact the parameters of the control rod ejection accident analysis.

The removal of Control Rod D-6 does, however, result in a reduction of margin to the N-2 subcriticality requirement associated with the control rod ejection accident, resulting in the removal of three (versus two) RCCAs from the complete set available when the analysis of record was performed. Analysis of this event for the STP Unit 1 Cycle 20 reactor core using the ANC code with Control Rod D-6 removed, showed that this key safety parameter limit was still met for this configuration. For future core designs, calculations will be performed each cycle to confirm that the key safety parameter limit is met.

<u>References</u>:

1. Westinghouse calculation, "TGX/THX Rod Ejection for Analysis for Vantage 5H Fuel."
2. Westinghouse calculation, "South Texas (TGX/THX) Rod Ejection for 3% TDF Reduction."
3. Westinghouse calculation, "Redesign - RSAC - SDM, Rod Ejection, and Trip Reactivity Following RWSC for South Texas Unit 1 (TGX) Cycle 20."
4. Westinghouse calculation, "Rod Ejection N-2 Subcriticality and Trip Reactivity Calculations for South Texas Unit 1 LAR for the Permanent Removal of the RCCA Located in D-6 - Deliverable Attachment."

e. Steam generator tube rupture (STP UFSAR Section 15.6.3)

The transient analysis for this event is performed using the RETRAN-02 computer code under the following limiting core conditions:

- end-of-life,
- cold zero power (350° F),
- all control rods (minus the most reactive, fully inserted control rod), and
- no xenon.

Removal of Control Rod D-6 reduces by one the number of RCCAs inserted into the reactor core for the "N rods in" and the "N-1 rods in" configurations. The reduction in negative reactivity insertion increases the critical boron concentration with N-1 rods inserted and results in a reduction in margin to the maximum boron concentration limits associated with the steam generator tube rupture accident.

Analysis of this event for the STP Unit 1 Cycle 20 core with Control Rod D-6 removed shows that the key safety parameter limits were still met for this configuration. For future core designs, calculations will be performed each cycle to confirm that the key safety parameter limit is met.

References:

1. STPNOC calculation "Steam Generator Tube Rupture Mass Release."
2. STPNOC calculation "Steam Generator Tube Rupture Margin to Steam Generator Overfill."
3. Westinghouse calculation, "Redesign - Revision 1 ** BORDER for South Texas Unit 1 (TGX) Cycle 20 - Deliverable Attachment."

NRC Item #3

3. Provide an explicit discussion for each safety analysis methodology regarding the assumptions made when developing the methodology for symmetric versus asymmetric control rod patterns (i.e., that would result from operation with one control rod removed). If no assumptions were made or if it was assumed that the control rod pattern was symmetric, provide a discussion of why that methodology is still applicable given the proposed new plant configuration.

STPNOC Response

The Nuclear Design analytical methods and codes used in the application of the WCAP-9272-P-A methodology (WCAP-16045-P-A; WCAP-16045-P-A, Addendum 1-A; and WCAP-10965-P-A) were rigorously benchmarked and qualified for a variety of reactor types (Westinghouse 2-, 3- and 4-loop, Combustion Engineering); fuel types (various lattice types and fuel rod diameters); and burnable poison types (integral fuel burnable absorber, wet annular burnable absorber, Pyrex, Gadolinia).

The Advanced Nodal Code (ANC) (WCAP-10965-P-A) is a nodal neutronics code for multidimensional reactor core calculations used for core nuclear design, including the prediction of design parameters such as:

- reactivity,
- assembly average power,
- control rod power and flux,
- Doppler coefficients,
- moderator coefficients,
- boron worth,
- control rod worth,
- burnable absorber worth, and
- fuel depletion.

ANC uses a nodal expansion method to solve the two-group diffusion equations. With this method, the neutron currents and average neutron fluxes for a node are determined from continuous homogeneous neutron flux profiles described by fourth order polynomial expansions for each of the x, y, and z directions across the node. Discontinuity factors are used to modify the homogeneous cross-sections to preserve the node surface fluxes and neutron currents that would be obtained from an equivalent heterogeneous model. ANC also employs a pin-power recovery process which uses an analytic solution to the two-group diffusion equations coupled with pin power information from the discrete model applied to the calculated node average power. ANC accurately reconstructs the results of fine mesh models using these methods.

The lattice code used to provide multi-group data to ANC has been updated to PHOENIX-P (WCAP-11596-P-A) and more recently to PARAGON/NEXUS (WCAP-16045-P-A, Addendum 1-A). With each update of the lattice code, the qualification of ANC includes a broader spectrum of reactor, fuel, and burnable absorber designs.

ANC is intended to be used for all nuclear design calculations, including off-normal condition analyses. These analyses include (but are not limited to) control rod worths and power distributions for ejected rod, stuck rod, and dropped rod conditions. The results discussed in WCAP-16045-P-A, Addendum 1-A; WCAP- 11596-P-A; and WCAP-10965-P-A demonstrate the

ability of ANC to accurately predict reactivity and power distribution in the presence of strong power gradients, which are typical of stuck, ejected, or dropped control rod configurations. These states provide an extreme test of the ANC flux solution.

WCAP-16045-P-A, Addendum 1-A; WCAP-11596-P-A; and WCAP-10965-P-A demonstrate that ANC is an accurate analytical tool for multidimensional nuclear calculations performed in the design, safety analyses, and operational follow of PWR cores. The intended usage of ANC encompasses all applications described in the reload safety evaluation methodology topical report.

The Nuclear Design methodology used to confirm the key safety parameter limits for those events listed in Item # 2 uses a three-dimensional nodal neutronics model that includes explicit modeling of the core configuration, including RCCA pattern. A full-core model is used for asymmetric calculations (e.g., "all-rods-in-minus-one" (N-1) calculations, asymmetric temperature distributions during a steamline break, etc.). No assumptions regarding RCCA bank symmetry are made in these analyses since the actual configuration is modeled. The one exception is trip reactivity following control rod ejection which was generically confirmed for certain standard RCCA patterns. As part of the STP Unit 1 Cycle 20 redesign to remove Control Rod D-6, the trip reactivity following rod ejection for the specific RCCA configuration was confirmed to be bounded by the value assumed in the generic analysis. Following permanent removal of Control Rod D-6, the trip reactivity following rod ejection will be confirmed every fuel cycle. The process, mechanics, and capability to model an N-2 configuration in a three-dimensional nodal model are not different from the analyses currently performed on a cycle-specific basis.

A change in the number of RCCAs is represented by the broad spectrum of reactor, fuel, and burnable absorber designs as well as the off-normal condition analyses evaluated in WCAP-16045-P-A, Addendum 1-A; WCAP-11596-P-A; and WCAP-10965-P-A. The capabilities of the codes/methodology and calculation uncertainties used in the methodology are not impacted.

Uncontrolled Boron Dilution

For the Uncontrolled Boron Dilution event, Reactor Coolant System boron concentrations are confirmed under various conditions. Analysis of this event uses a full-core model with specific reactor core and RCCA configurations. No assumptions are made with respect to RCCA symmetry. As discussed above, removal of Control Rod D-6 does not impact the capabilities of the codes/methodology or calculation uncertainties used in the methodology.

In Modes 3 through 5, indication to the operator that a boron dilution event is in progress is a flux multiplication alarm. The boron dilution analysis assumes a minimum allowable operator action time between receipt of the flux multiplication alarm and complete loss of shutdown margin, and calculates the maximum allowable differential boron worth as a function of critical boron concentration for the existing variable shutdown margin requirements (also a function of critical boron concentration). No assumption is made regarding control rod pattern symmetry.

Determination of the differential boron worth is not impacted by Control Rod D-6 removal because the analysis is performed assuming an all-rods-out configuration to maximize the differential boron worth.

In Modes 3 through 5, the time between event initiation and receipt of the flux multiplication alarm is determined using limiting plant empirical source range inverse count rate ratio (ICRR) data as a function of time. The solution technique makes no assumption regarding control rod pattern symmetry. The empirical ICRR data is generated while monitoring the approach to

criticality during boron dilution with the shutdown banks removed from the reactor core. Therefore, removal of Control Rod D-6 has no impact on the use of the ICRR data.

Hot Zero Power (HZP) Steamline Break

The Nuclear Design analysis of this event uses a full-core model with specific core and RCCA configurations. No assumptions are made with respect to RCCA symmetry. As discussed above, removal of Control Rod D-6 does not impact the capabilities of the codes/methodology or calculation uncertainties used in the methodology.

For the HZP steamline break core response analysis, the thermal-hydraulic code RETRAN-02 is used to model the core response. RETRAN-02 uses a point kinetics neutronics model and cannot directly model an asymmetric control rod pattern. For this event, which assumes minimum shutdown margin with the most reactive control rod stuck out of the reactor core, the limiting case is generated using assumed stuck-rod reactivity feedback coefficients to model moderator density, Doppler power defect, and boron worth characteristics. As part of the reload safety evaluation process, Nuclear Design implicitly confirms acceptability of the stuck-rod reactivity feedback coefficients by confirming that an acceptable power match exists between the most limiting ANC power search calculation and the maximum HZP steamline break statepoint power level from the RETRAN-02 results. An unacceptable power mismatch requires evaluation or reanalysis of the event using adjusted stuck-rod reactivity feedback coefficients. This approach is consistent with the analysis methodology for the HZP steamline break core response described in WCAP-9226-P-A, Revision 1; the reload methodology for this event described in WCAP-9272-P-A; and the qualification of RETRAN-02 for use in analyzing HZP steamline break core response as described in WCAP-14882-P-A.

Permanent removal of the control rod D-6 has the potential to affect the RETRAN-02 reactivity feedback model as implemented using the stuck rod feedback reactivity coefficients. For each core reload analysis, the stuck rod coefficients applied in the HZP steamline break analysis are confirmed via demonstration of an acceptable power match using ANC. An unacceptable power mismatch requires evaluation or reanalysis of the event using adjusted stuck-rod coefficients. For the STP Unit 1 Cycle 20 reload, the reactivity feedback model was confirmed acceptable without requiring stuck rod coefficient adjustment or reanalysis.

Control Rod Ejection

The Nuclear Design analysis of this event uses a full-core model with specific core and RCCA configurations. No assumptions are made with respect to RCCA symmetry for the cycle-specific trip reactivity calculation or the balance of the Nuclear Design analysis as discussed in the STPNOC response to Item # 2. Previously, the trip reactivity following rod ejection was generically confirmed for certain standard RCCA patterns. As part of the STP Unit 1 Cycle 20 redesign to remove Control Rod D-6, the trip reactivity following rod ejection for the specific RCCA configuration was confirmed to be bounded by the value assumed in the generic analysis. Following permanent removal of Control Rod D-6, the trip reactivity following rod ejection will be confirmed every fuel cycle. As discussed above, removal of Control Rod D-6 does not impact the capabilities of the codes/methodology or calculation uncertainties used in the methodology.

For the transient analysis, the neutron kinetics code TWINKLE is used to model the core response. Although TWINKLE is capable of modeling in one, two, or three dimensions, the control rod ejection analysis methodology only uses one-dimensional axial geometry to calculate core-average nuclear power. Therefore, an asymmetric control rod pattern cannot be modeled explicitly. The potential impact of removal of Control Rod D-6 is limited to the

calculation of ejected control rod worth, confirmation of trip reactivity, and confirmation of trip reactivity versus rod position, as discussed below:

- The ejected control rod worth input to TWINKLE reflects the maximum allowed bank insertion at a given power level as determined by rod insertion limits. However, since Control Rod D-6 is in a shutdown control rod bank, there is no impact on the ejected rod worth calculation because the control rod is already fully withdrawn from the reactor core.
- Trip reactivity is calculated assuming all control rods are inserted except for the highest worth ejected control rod and an adjacent control rod, both of which are assumed to be fully withdrawn. Calculations of the HZP and HFP trip reactivity values for rod ejection analyses were generically performed and confirmed to apply to the STP control rod pattern such that cycle-specific confirmation was not historically required. However, removal of Control Rod D-6 impacts the generic calculation such that cycle-specific confirmation will be required for both HZP and HFP conditions. Confirmation of trip reactivity was performed for the STP Unit 1 Cycle 20 reload and will be implemented in the reload safety evaluation process in future cycles with Control Rod D-6 permanently removed.
- Because trip reactivity is affected, trip reactivity as a function of inserted rod position is also affected by the removal of Control Rod D-6. The reload safety evaluation process implicitly confirms that the cycle-specific curve for trip reactivity versus rod position is bounded by the curve assumed in the control rod ejection analysis. Analysis of this event for the STP Unit 1 Cycle 20 core with Control Rod D-6 removed confirms that the trip reactivity versus rod position curve is acceptable without requiring reanalysis. Violation of this reload parameter will require evaluation or re-analysis of the rod ejection event prior to core reload.

Steam Generator Tube Rupture

The Nuclear Design analysis of this event uses a full-core model with specific core and RCCA configurations. No assumptions are made with respect to RCCA symmetry. As discussed above, removal of Control Rod D-6 does not impact the capabilities of the codes/methodology or calculation uncertainties used in the methodology.

The transient analysis for this event is performed with the RETRAN-02 computer code (WCAP-14882-P-A) in accordance with the methodology prescribed in WCAP-10698-P-A, and WCAP-10698-P-A, Supplement 1. RETRAN-02 uses a point kinetics neutronics model and does not directly model an asymmetric control rod pattern or the number of RCCAs in the reactor core. The impact of the number of RCCAs and their arrangement is inherently considered in the reactivity parameters generated by the three-dimensional ANC model and used in the RETRAN-02 model.

NRC Item #4

4. Provide a discussion of any evaluations that have been performed under Title 10 of the *Code of Federal Regulations* Section 50.59 "Changes, tests, and experiments," if applicable, as a result of a removal of a control rod that may impact or may have impacted the analyses discussed above.

STPNOC Response

There were no 10CFR50.59 evaluations performed as a result of removing Unit 1 Control Rod D-6. Prior to initial startup of STP Unit 1 Cycle 20, the following 10CFR50.59 screenings were performed after NRC approval of the December 2015 emergency LAR:

Title	Reference	Discussion
"Remove Unit 1 Control Rod D6 from Service (Mechanical Scope)"	DCP 15-25420-8, Supplement 0	*No impact to analyses discussed above* Evaluated changes: • Unlatching and removal of D-6 control rod drive shaft • Installation of flow restrictor on top of control rod guide tube
"Remove power from shutdown rod D-6 and remove indication/alarms that would occur with the removal of rod D-6"	DCP 15-25420-9, Supplement 0	*No impact to analyses discussed above* Evaluated changes: • Removal of control rod D-6 indication in the main control room • Preventing DRPI alarms which would occur as a result of control rod D-6 removal • Removal of power to control rod D-6 stationary, lift, and movable coils • Plant computer system changes (inputs to Rod Supervisory Application) to allow system to perform as designed • Updating rod trace equipment to start recording based on movement of control rod F-12 instead of control rod D-6
"Unit 1 Cycle 20 Reload Safety Evaluation (Modes 1-5)"	RSE-U1, Revision 5	*Impacted accident analyses are discussed in the License Amendment Request and this supplement* Applicable evaluated changes (for Unit 1 Cycle 20): • Axial offset limits • Core axial power shape model • Maximum RCS boron concentration • Removal of RCCA D-6 and installation of thimble plug • Operations with 56 RCCAs instead of 57 RCCAs • F_{xy} limits for rated thermal power

Using the methodology in WCAP 9272-P-A, a Reload Safety Evaluation (RSE) is performed prior to each operating cycle. Within the RSE, a 10CFR50.59 screening is performed for cycle-specific changes; for changes that do not screen out, a 10CFR50.59 evaluation is performed.