December 20, 2001

Mr. William T. Cottle
President and Chief Executive Officer
STP Nuclear Operating Company
South Texas Project Electric
Generating Station
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT UNITS 1 AND 2: REQUEST FOR APPROVAL OF POWER UPRATE AND REVISION TO THE TECHNICAL SPECIFICATIONS SUPPORTING THE POWER UPRATE (TAC NOS. MB2899 AND MB2903)

Dear Mr. Cottle:

By letter dated August 22, 2001, STP Nuclear Operating Company, the licensee for the South Texas Project Units 1 & 2, submitted information and requested approval of increasing the plant operating power level by 1.4 percent, and a license amendment supporting associated revisions to Technical Specifications. The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the licensee's requests and requires additional information in order to complete its evaluation.

The enclosed request for additional information (RAI) was discussed with the licensee's staff, during telephone calls on December 13, 17, and 19, 2001. During the calls, it was agreed that the licensee will respond to the enclosed RAI in two steps. The response to questions that can be answered with limited additional effort will be targeted for January 20, 2002. The response to the remainder of the questions will be targeted for the first week of February 2002. If circumstances result in a need to revise the target dates, please contact me promptly at (301) 415-1476.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosure: RAI Questions

cc w/encls: See next page

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SOUTH TEXAS PROJECT (STP)

REQUEST FOR ADDITIONAL INFORMATION

Instrumentation & Controls

1. Section 3.6 in Attachment 6 states that with respect to the CROSSFLOW Ultrasonic flow measurement (UFM) uncertainties, Uncertainty calculations have been performed and determined a mass flow accuracy of better than 0.5 percent of rated flow for STP Units 1 and 2. The Westinghouse calculation in WCAP-15633 used 1 percent instrument uncertainty for the CROSSFLOW. Please explain, why the plant could not attain the calculated accuracy of 0.5 percent.

2. Section 3.3 in Attachment 6 states that with an inoperable CROSSFLOW UFM, plant operation at a core thermal power level of 3853 MWt may continue for 24 hours. If the CROSSFLOW UFM is not restored in 24 hours, plant procedures would require reactor power to be reduced to a level less than or equal to 3838 MWt. During this 24 hours, the power measurement uncertainty is 1.0 percent, as calculated in WCAP-15697. What is the impact of a sustained overpower event during the 24 hour period on core damage frequency and large early release frequency when secondary side power calorimetric measurement uncertainty is 1.0 percent, instead of 0.6 percent?

3. Power Range Neutron Flux High setpoint values are revised in Table 3.7-1 of the STP Technical Specifications to reflect the power uprate. Please confirm that the reactor trip setpoint of this function does not need revision.

4. Provide description of the programs and procedures that will control calibration of the CROSSFLOW system and the pressure and temperature instrumentation whose measurement uncertainties affect the power calorimetric uncertainties determined in the Westinghouse calculations WCAP-15633, Revision 0 and WCAP-15697, Revision 0. In this description, please include the procedures for:

    a. Maintaining calibration,
    b. Controlling software and hardware configuration,
    c. Performing corrective actions,
    d. Reporting deficiencies to the manufacturer, and
    e. Receiving and addressing manufacturer deficiency reports.

Reactor Systems

1. Attachment 6, Section 2, Table 2.1-1 presents 4 steady-state plant conditions. However, the table does not describe the cases. Please state the 4 cases presented and describe the major methodologies and assumptions used to generate its calculated values. Also include the current design parameters and assumptions for STP Units 1 and 2 as presented in Table 2.1-1.

2. Attachment 6, Section 6.1, Nuclear Steam Supply System Fluid Systems, states that various reactor coolant system (RCS) parameters remain unaffected by the power uprate because they are bounded by the values calculated for the Model Delta 94 steam generators. Please provide a reference to the approvals or show that the parameters were calculated using methods or processes that were previously approved by the NRC.

3. Attachment 6, Section 6.2.1, Main Steam System - Steam Generator (SG) Power-Operated Relief Valves (PORV), states that an evaluation of the installed capacity of the SG PORVs indicates that the original design basis cooldown capacity can still be achieved for the uprated conditions, however, sufficient bases were not provided to support that conclusion. Please provide the original design basis cooldown capacity and the uprated capacity or a reference to this information.

4. Attachment 6, Section 6.2.2, Steam Dump System - Condenser Steam Dump Valves, states that the condenser steam dump capacity continues to meet the sizing criterion for the uprated plant conditions, however, sufficient bases were not provided to support that conclusion. Please state the total uprated steam dump capacity, in percentage of the maximum steam flow and in lb/hr, at the uprated conditions.

5. Attachment 6, Section 6.2.4, Auxiliary Feedwater System (AFS), states that evaluations of the limiting transients and accidents have confirmed that the current AFWS design basis performance remains acceptable, however, sufficient bases were not provided to support your conclusion. At the uprated conditions, state the limiting transients for the AFWS design basis, the limiting minimum flow requirements of the AFWS for the limiting transients, and the AFWS performance for these transients.

6. Attachment 6, Section 8.3.1.9, Chemical and Volume Control System Malfunction, states that an evaluation of the Mode 1 analysis showed that the power uprate has an insignificant impact on the automatic reactor trip time used in the analysis. However, sufficient bases were not provided to support this conclusion. Provide the technical bases which lead to this conclusion.

7. Attachment 6, Section 7.2, Vessel Integrity - Neutron Irradiation, states that the 32 effective full-power year fluence was reevaluated to account for the power uprate in the revised pressure temperature curves and the calculation of the RT_{PTS}. Did the methodology used in this reevaluation comply with the guidance in Regulatory Guide (RG) 1.190, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I? Provide a reference to the approved methodology used.

8. Attachment 6, Section 7.10.1, Nuclear Design, states that "...adequate margin to the limits associated with all reload safety analysis parameters that are evaluated for each cycle have been confirmed..." For the power uprated cycle, please provide the overall peaking factor (vs the old) and the departure from nucleate boiling ratio (DNBR) (vs the old). Will the new cycle contain different fuels? Is the new cycle a transition (mixed fuel) cycle? Will the new cycle contain a lead test assembly?

9. It appears that the South Texas reactor cores will consist of 17x17 XL robust fuel assembly (RFA) and 17x17 XL V5H fuel after the power uprate. Please provide the licensing bases and justifications that the fuel will conform to all the applicable

regulations so that the (1) fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained. Also, please describe the licensing rationale that was used to justify the transition from V5H to RFA fuel.

10. As described in Section 3 of Attachments 2 and Section 7.10.3 of Attachment 6, the core thermal-hydraulic analyses and evaluations for the 1.4-percent power uprate were performed with: (1) the assumption of core designs composed of RFAs, (2) use of the WRB-2M correlation for the DNB analysis, (3) use of the revised thermal design procedure (RTDP) DNB methodology, and (4) use of the WRB-1 DNB correlation for the standard and Vantage 5 Hybrid (V5H) fuel type for which the WRB-2M DNB correlation is not applicable.

    a. The NRC staff safety evaluation for the acceptance of the WRB-2M correlation described in WCAP-15025-P-A states that the WRB-2M correlation may only be used for the Modified 17x17 Vantage 5H (V5H) fuel without further justification. Provide a comparison of the RFA and modified V5H fuel designs, and justification for the use of WRB-2M for the RFA design.

    b. Provide justification for application of the WRB-1 correlation to the V5H fuel.

    c. How do you assure that the WRB-2M and WRB-1 correlations are not applied outside its ranges of applicability, including pressure, local mass velocity, local quality, and fuel design?

11. It appears that there will be mixed cores of standard, V5H, and RFA fuel designs in the future fuel cycles.

    a. What is the basis for assuming the core designs are composed only of RFAs?

    b. How is it determined that the limiting channel having the minimum DNBR would not occur in the standard or V5H fuel?

    c. What is the mixed-core DNBR penalty value used in the thermal-hydraulic calculation assuming RFAs?

    d. How is the mixed core penalty value determined?

    e. Has the mixed core penalty been applied in the safety analysis assuming the RFA cores? If not, why not?

12. Attachment 6, Section 7.10.3 states that to support operation at the uprated conditions with the use of WRB-2M DNB correlation, revised RTDP DNBR design limits were calculated. In addition, the safety analysis limits were revised to create an increased DNB margin.

    a. Provide WCAP-13441 which was stated to provide the basis for the RTDP uncertainties.

b. Provide the revised RTDP DNBR design limits and safety analysis limits for both typical and thimble cells.

c. Describe how these DNBR design limits are derived based on the power measurement uncertainty using the CROSSFLOW UFM for the feedwater flow.

d. What are the RTDP design DNBR limits when the CROSSFLOW UFM is out of service? How are these limits accounted for in the safety analyses?

e. If the limiting hot channel occurs in the standard or V5H fuel design, and the WRB-1 correlation is used for these fuel designs, what are the values for the RTDP design and safety analysis DNBR limits?

f. Describe how the analysis conforms to the restrictions stated in the NRC staff safety evaluation accepting the use of the RTDP methodology described in WCAP-11397-P-A.

13. Attachment 6, Section 7.10.3 states that the DNBR analyses at the 1.4-percent uprate conditions showed that the DNB design basis continues to be met. However, sufficient technical bases were not provided to support this conclusion. Please provide the analyses and evaluations that were performed which lead to your conclusion. Also, provide the input from the THINC-IV calculations used in the RTDP analysis.

14. Attachment 6, Section 8.3, non-loss-of-coolant-accident (Non-LOCA) Analysis, states that nominal values of initial conditions are assumed in accident analyses that are performed to demonstrate meeting the DNB acceptance criteria. However, the same section states that some non-LOCA analyses are currently analyzed with an explicit 2-percent power measurement uncertainty, thus not requiring re-analysis for the power uprate. These transients include: (1) loss of alternating current (AC) power and loss of normal feedwater, (2) startup of an inactive reactor coolant loop, and (3) chemical and volume control system malfunction that results in increasing reactor coolant inventory. Please clarify whether these three transients were analyzed with the RTDP methodology or the deterministic methodology.

15. Attachment 6, Section 8.3 states that the core thermal limits and the resulting overtemperature ΔT (OT ΔT) and overpower ΔT (OP ΔT) setpoints are essential inputs to the non-LOCA analyses. It also states that a revised set of core thermal limits was developed because of the 1.4-percent power uprate, and that the OT ΔT and OP ΔT setpoints need not be revised.

a. Clarify how the core thermal limits are input in the non-LOCA safety analyses.

b. Specify the core thermal limits and provide the technical bases which supported your conclusion that the trip setpoints need not be revised.

16. Attachment 6, Section 7.3.1 states that the revised design conditions (primarily T_{cold}) will have a negligible effect on the rod control cluster assembly drop time, and the time will still be less than the current value of 2.8 seconds required by the Technical Specifications. However, sufficient technical bases were not provided to support these conclusion. Provide the technical bases that support these conclusions.

17. Attachment 6, Section 7.3.1 states that the resulting bypass flow velocities exiting or entering the pressure relief holes on the baffle plates did not significantly change and still meet the fuel interface requirements for the 1.4-percent uprate conditions. Provide the technical bases that support these conclusions.

18. For accidents and transients that the existing analyses of record do not bound plant operation at the proposed uprated power level:

    I. This section covers the transient and accident analyses that are included in the plant's Updated Final Safety Analysis Report (UFSAR) (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing of its plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, and flooding).

    II. For analyses that are covered by the NRC approved reload methodology for the plant:

        a. Identify the transient/accident that is the subject of the analysis;

        b. Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate;

        c. Provide an explicit commitment to submit the analysis for NRC review, prior to implementation of the power uprate, if NRC review is deemed necessary by the criteria in 10 CFR 50.59;

        d. Provide a reference to the NRC's approval of the plant's reload methodology; and

        e. Provide tables containing the following information:

            1. A summary of the initial conditions and assumptions for all transients reanalyzed that will differ from the NRC approved UFSAR due to uprated power operations. This table should identify the conditions or assumptions that have changed, its values used in the cycle before and after the uprate, and a brief justification for any values that move in an non-conservative direction.

            2. A summary of the results of all transients reanalyzed at the uprated power level. Include the applicable safety limit values used as the acceptance criteria and provide the values obtained at the current rated thermal power (RTP) and the uprated RTP. Provide a discussion for any transients that result in a reduced margin of safety.

III. For analyses that are not covered by the reload methodology for the plant, provide a detailed discussion for each analysis:

a. Identify the transient or accident that is the subject of the analysis.

b. Identify the important analysis inputs and assumptions (including its values), and explicitly identify those that changed as a result of the power uprate.

c. Confirm that the limiting event determination is still valid for the transient or accident being analyzed.

d. Identify the methodologies used to perform the analyses, and describe any changes in those methodologies.

e. Provide references to NRC staff approvals of the methodologies in Item Item d, above.

f. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology.

g. Describe the sequence of events and explicitly identify those that changed as a result of the power uprate.

h. Describe and justify the chosen single-failure assumption.

i. Provide plots of important parameters and explicitly identify those that changed as a result of the power uprate.

j. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head, valve isolation capabilities) required to support the analysis.

k. Discuss the results and acceptance criteria for the analysis, including any changes from the previous analysis. In addition, provide the results of the re-analyses including primary and secondary system peak pressure, minimum DNBR, and/or amount of fuel failed.

19. The NRC concluded that you have not provided enough information for us to agree with your conclusion that South Texas continues to comply with the ATWS Rule. In sensitivity studies performed by Westinghouse as the basis for the ATWS rule, an increase in rated power for units similar to South Texas can result in significantly higher peak RCS pressures. This condition, coupled with a positive moderator temperature coefficient operation may result in the South Texas units exceeding the American Society of Mechanical Engineers (ASME) Stress Level C Limit of 3200 psig during an ATWS event. This limit is the basis for the ATWS rule. Provide a plant-specific analysis demonstrating the ability of the South Texas units to meet the basis for the ATWS Rule,

10 CFR 50.62. In the analysis, state the assumed initial conditions for power level, moderator temperature coefficient and plant operating conditions for each of the units. Show that the units will not exceed the ASME Stress Level C Limit of 3200 psig. In addition, include a discussion and applicable values of the unfavorable exposure time, if any, and ATWS Core Damage Frequency for the South Texas units as a result of the proposed power uprate.

20. To show that the referenced generically approved LOCA analysis methodologies apply specifically to the South Texas plants, provide a statement that the South Texas plants and its vendor have **ongoing processes** which assure that LOCA **analysis input values** for peak cladding temperature sensitive parameters **bound the as-operated plant values** for those parameters.

21. What is the decay heat source assumed in the design of the emergency core cooling system (ECCS) switchover from the injection mode to the ECCS sump recirculation mode for the current power rating? Does this assumed heat source change for the uprated power? Is the timing of the switchover affected? Please explain.

22. Attachment 6, Section 8.2.2, “Post-LOCA Long-Term Core Cooling,” states that the boron concentration in the recirculating coolant is maintained at adequate levels to keep the core subcritical post-LOCA. It also states that this detail will be confirmed in the core reload licensing process. What is the basis for calculating the boron concentration during the core reload licensing process? In addition, what are the assumptions used for the calculations? In particular, please identify what the reactor power (at time of LOCA initiation) is assumed to be. Also, during this process, how are issues with boron precipitation handled?

23. You reference CENPD-397-P-A Revision 01 in your amendment request. This topical report, including the NRC staff’s safety evaluation, contains criteria that shall be addressed by licensees referencing the topical report. Please list each criterion and state how each is satisfied. (Where appropriate, you may reference specific sections of your application. Also, provide the technical bases that support the use of a Combustion Engineering methodology at a Westinghouse plant.)

24. Attachment 6, Section 3.3 states that when the CROSSFLOW equipment is out-of-service “... power level is based upon the methodology and instrumentation configuration assumed in WCAP-15697....” WCAP-15697 also uses “assume” and provides values, including a power calorimetric uncertainty. Please:

    a. Confirm that your methodology is actually what is provided in WCAP-15697 or identify any differences.

    b. Where “assume” is used, please justify this usage or provide plant-specific information.

    c. For each item contributing to the power calorimetric uncertainty determination, provide a comparison of the uncertainty-of-record used in your licensing basis and the values provided in WCAP-15697 and justify any differences.

d. WCAP-15697 states “... no allowance is made for venturi fouling.” We found no other reference to venturi fouling in your request. Please describe how you address this topic.

e. The WCAP-15697 power calorimetric uncertainty appears to be inconsistent with the 3838 MWt you state will be the maximum allowable power if the CROSSFLOW system is unavailable and is not restored within 24 hours. Please explain.

f. WCAP-15697 states “Reactor power is monitored by the performance of a secondary side heat balance (power calorimetric) at least once every 24 hours.” In Section 3.2, you state a comparison is made periodically. If the NRC staff assume periodically means every 24 hours, and consider the Section 3.3 statement referenced in Item (e), immediately above, approximately 48 hours may pass between a calibration of thermal power and a power reduction. Please address this observation and justify the actual times.

25. WCAP-15697 references Prairie Island data and other plant reactor coolant pump data regarding pump thermal energy generation. Please provide references for the plant data other than Prairie Island. The data the NRC staff have seen appear to indicate that the reactor coolant pump thermal energy changes early in plant life due to impeller smoothing. Explain how this effect is taken into account in determining the pump thermal energy contribution for the power uprate?

<u>Structural Questions.</u>

1. The licensee evaluated the impact of the power uprate on the minimum wall thickness of the steam generator tubes. However, the licensee did not provide the value of the minimum wall thickness. In addition, the licensee did not relate the minimum wall thickness to the tube repair limits in the South Texas Units 1 and 2 technical specifications. The licensee needs to provide the value of the minimum wall thickness and confirm that the minimum wall thickness is within the tube repair limits in the South Texas technical specifications.

2. NRC has issued the following generic communications regarding SG tube plugs: NRC Information Notice 89-65, "Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox;" NRC Information Notice 89-33, "Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs;" NRC Bulletin No. 89-01, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs," and Supplements 1 and 2; NRC Information Notice 94-87, "Unanticipated Crack in A Particular Heat of Alloy 600 Used for Westinghouse Mechanical Plugs for Steam generator Tubes".

   The licensee discussed evaluation of mechanical tube plugs under the power uprated conditions.

   a. Clarify if tube plugs have been used in the South Texas Unit 1 replacement SGs,

   b. Discuss if any of the above NRC generic communications are applicable to the tube plugs used in the South Texas replacement SGs and the steps that have been taken to meet the NRC staff's recommendations in the above generic communications,

c. Discuss any degradation detected in tube plugs and the associated repair method other than those discussed in Item (b).

3. Clarify whether the SG tubes under the power uprated conditions satisfy the structural integrity discussed in RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes.”

4. The licensee evaluated structural integrity of the steam generator tubes under the power uprated conditions; however, the NRC staff is not clear if the structural integrity evaluation included evaluating leakage integrity of SG tubes. Confirm the acceptability of the leakage integrity of the SG tubes under the power uprated conditions.

5. Discuss the impact of the power uprated conditions on (a) the degradation in the steam generator tubes, (b) the intervals of SG tube inspections, and (c) the condition monitoring and operational assessments of the SG tubes.

6. In your submittal you have indicated that the models in the CHECWORKS predictive code were revised to incorporate flow and process system conditions that are determined for 1.4 percent power uprate conditions. What was the predicted change of the wear rates calculated by the revised code for the components most susceptible to flow-accelerated corrosion?

<u>Accident Analyses</u>

During a conference call on December 19, 2001, the NRC staff pointed out discrepancies in the licensee's statement in Attachment 6, Section 11.2, "Accident Analyses," and the information contained in the updated safety analysis report (SAR). The licensee has stated in Section 11.2 that radiological source terms for all the analyses (except small line failure outside the containment) were determined at power level of 4100 MWt. Contrary to the above, the NRC staff noted, and informed the licensee during the December 19, 2001 phone call, that some accidents reported in the updated SAR were not labeled as evaluated at power level of 4100 MWt. The licensee indicated that the radiological source terms used in the analyses are based on 4100 MWt, and the SAR page(s) have either been updated or will be updated accordingly. Please provide the updated SAR page(s) reflecting the correct power level used to evaluate the radiological consequences resulting from these design-basis accidents and any other additional information concerning use of updated power level of 4100 MWt.