



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

October 27, 1999
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10CFR50.90

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498 and STN 50-499
Response to Verbal Request for Additional Information -
Proposed Amendments on Replacement Steam Generator Water Level Trip Setpoint Differences,
South Texas Project, Units 1 and 2

- References: 1) Letter from T.H. Cloninger to U.S. Nuclear Regulatory Commission dated July 22, 1998, (NOC-AE-000163)
- 2) Letter from U.S. Nuclear Regulatory Commission to W. T. Cottle, dated April 15, 1999
- 3) Letter from S. E. Thomas to U.S. Nuclear Regulatory Commission dated June 16, 1999, (NOC-AE-000548)

Attached is South Texas Project Nuclear Operating Company's (STPNOC) response to verbal questions posed by the U. S. Nuclear Regulatory Commission (NRC) regarding this STPNOC submittal (Reference 1).

On April 15, 1999, the NRC sent a request for additional information (RAI) (Reference 2), and STPNOC answered it (Reference 3) on June 16, 1999. On July 1, 1999, STPNOC, Westinghouse, and the NRC conducted discussions of the subject submittal in a telephone conference held on July 1, 1999. In preparation for this conference, the staff verbally provided four questions. Before the conference, the staff obtained a satisfactory answer for Question 4 and withdrew it, leaving three questions. STPNOC and the staff could not find a mutually agreeable answer for Question 1 during the conference, but subsequently resolved it in a meeting at NRC White Flint offices. During the telephone conference, STPNOC provided satisfactory answers to

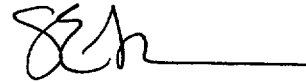
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Questions 2 and 3, and the NRC asked that we provide those answers in written form. After the conference, the NRC verbally posed one additional question. This is Question 5. STPNOC agreed to provide the answer along with the answers to Questions 2 and 3. Attached to this letter (Attachment 1) are the questions with their respective answers.

As mentioned during recent telephone conversations between STPNOC and the NRC, Westinghouse has discovered that there was an input error made during a RETRAN run for design basis event analyses associated with this amendment request. Westinghouse identified and corrected the root cause of the problem. They also reviewed all other related analyses and verified them to be correct. The input error had a minor affect on certain material provided in the STPNOC response (Reference 3) to the NRC RAI (Reference 2). STPNOC has prepared a supplement to correct the inaccuracies and will transmit it as soon as the draft has been reviewed and approved.

Nothing contained in this response is a commitment unless so specified in separate correspondence.

If questions arise, please contact Mr. M. E. Kanavos (512) 972-7181, or me (512) 972-7162.



S. E. Thomas
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SET/MEK/MTVN

Attachment: 1. Responses to NRC Verbal Request for Additional Information (VRAI)

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ATTACHMENT 1

RESPONSES TO NRC

VERBAL REQUEST FOR ADDITIONAL INFORMATION (VRAI)

Verbal Request for Additional Information

Replacement Steam Generator Water Level Trip Setpoint Differences

South Texas Project, Units 1 and 2

1. ***Attachment 1, Page 10 of 13, 2nd par, 1st sentence says, "STPNOC does not believe the requirements of 10CFR50.36(c)(2)(ii)(B) apply to STP because STP is excluded by the provisions of 10CFR50.36(c)(2)(iii).***

Mr. Weiss disagrees.

Response to VRAI 1.

This issue was resolved during a July 22, 1999, meeting between STPNOC and NRC at NRC White Flint offices. The NRC's decision is expressed in an NRC letter (Reference 1), thus will not be repeated here.

2. ***Specification 3.4.4.a (RCS Relief Valves), says, "With one or both PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours."***

This allows the block valve upstream of the pressurizer PORVs to remain closed for an unlimited time. How can STP take credit for the pressurizer PORVs in mitigating accidents?

Response to VRAI 2.

STP does not take credit for pressurizer PORVs in mitigating accidents. However, as with all control systems, STPNOC models Pressurizer PORVs in non-LOCA transient analyses if their function would tend to increase the severity of the transient. This is the reason that we model pressurizer PORVs in analyses of the loss of normal feedwater (LONF) and feedline rupture (FLB) events.

3. *Attachment 1, Page 6 of 13, shows Table 4, "Time Sequence of Events for the FLB Event with Maximum Reactivity Feedback and Offsite Power Available." The last line of the table lists, "Pressurizer Reaches Water-Solid Condition," and shows a corresponding time of 2796 seconds.*

Response to VRAI 3.

2796 seconds is 46 minutes and 36 seconds. The event analysis calculates this value, and we provided it to demonstrate that it is significantly greater than 30 minutes (1800 seconds). The importance of this is that our Condition IV FLB analysis shows that the pressurizer does not go water solid during the first thirty minutes following initiation of the design basis transient. This clearly demonstrates that STP operators have sufficient time following event initiation to prevent pressurizer overfill.

4. *STPNOC supplement to TSC-205, NOC-AE-000462, "Auto Action of SG PORVs."*

Response to VRAI 4.

(Request withdrawn by NRC.)

5. *Address each of the three conditions listed in the conclusions section of the NRC staff SER for WCAP-14882.*

The NRC staff concludes in the SER for WCAP-14882 (Ref. 3) that the "use of RETRAN as described in WCAP-14882 is acceptable for licensing calculations and RETRAN may be used to replace the LOFTRAN computer code in Westinghouse reload methodology provided that the following conditions are met:

- a. The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of this code for other analytical purposes will require additional justification.*
- b. WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification.*
- c. Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 4. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."*

Response to VRAI 5.

Each of these conditions are addressed below, as they relate to the South Texas Project Model Δ94 Replacement Steam Generator Program.

- a. The non-LOCA transients explicitly analyzed with RETRAN for this program include the following: feedwater system malfunctions, steam system piping failures, turbine trip, loss of offsite power, loss of normal feedwater flow, and feedwater system pipe break. All of these events are listed in Table 1 of the SER; therefore, no additional justification is required.
- b. The South Texas Project nuclear units are 4-loop, Westinghouse-designed, pressurized water reactors that are currently in commercial operation. Therefore, no additional justification is required.
- c. The non-LOCA RETRAN analyses were performed in accordance with the methodologies discussed in WCAP-14882 (Ref. 2) and WCAP-9272-P-A (Ref. 4).

References

1. USNRC Letter, "South Texas Project, Units 1 and 2 (STP) – Proposed Technical Specification (TS) Change on Replacement Steam Generator (SG) Water Level Trip Setpoint (TAC Nos. MA2500 and MA2501)," Thomas W. Alexion (USNRC) to William T. Cottle (STPNOC), dated August 26, 1999
2. WCAP-14882 Revision 0, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," Huegel, D. S., et al., June 1997.
3. USNRC Letter, "Acceptance for Referencing of Licensing Topical Report WCAP-14882, 'RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analysis' (TAC NO. M99107)," Akstulewicz, F. (USNRC) to Sepp, H. (W), February 11, 1999.
4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," Bordelon, F. M., et al., Approved July 1985.