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

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10CFR50.59
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
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South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
10CFR50.59 Summary Report

Pursuant to the requirements of 10CFR50.59, the attached report contains a brief description and summary of the 10CFR50.59 evaluations of changes, tests and experiments conducted at the South Texas Project.

There are no commitments in this letter.

If there are any questions regarding this summary report, please contact Jim Morris at (361) 972-8652 or me at (361) 972-7298.

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Attachment: 10CFR50.59 Evaluation Summaries May 2009 – July 2011

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NRR

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10CFR50.59 Evaluation Summaries
May 2009 – July 2011

Summaries of the following 10CFR50.59 Evaluations are provided in this attachment:

No.	Condition Report Number	Subject
1.	07-2887-59	Change to UFSAR Table 10.3-1, Main Steam System Power Supply Failure Modes And Effects Analysis.
2.	07-17012-4	Change to UFSAR 15.6.3 Steam Generator Tube Rupture Margin to Overfill Analysis
3.	08-15047-3	Change to UFSAR Table 15.0-4 and Table 16.1-3
4.	09-7557-3	Change to UFSAR Table 15.3-2
5.	09-18094-3	Change to UFSAR to add the results of additional analysis for Containment steam line break analysis for Unit 1 Cycle 16 and Unit 2 Cycle 14.
6.	09-18620-8	Change to UFSAR to reflect the revised containment atmospheric pressure and temperature analysis for the steam-line break event.
7.	00-10937-88 Revision 1	Replacement of the Main Feedwater Isolation Valve circuitry
8.	10-17816-1	Change to the UFSAR Section 3.6 to add the statement, "An assumed single failure of safety related active component concurrent with the non-mechanistic break is not required".

10CFR50.59 Evaluation Summaries May 2009 – July 2011*

1. 07-2887-59 - Change to UFSAR Table 10.3-1, Main Steam System Power Supply Failure Modes and Effects Analysis.

Description: The subject of this evaluation is the revision of UFSAR Table 10.3-1 to permit a manual operator action to close the Main Steam Isolation Valves (MSIVs) to the Moisture Separator Re-heater (MSR) in the event of a loss of 120VAC to power panels DP005 and DP006 or a loss instrument air. The change ensures that the plant complies with the assumptions for the UFSAR Section 15 dose analysis and prevents the potential for an increase in frequency of a steam line break event as discussed in UFSAR section 15.1.5.

Summary: MSIV's to the MSR valves are marked with an Emergency Operating Procedure (EOP) tag to facilitate their identification while performing actions required by the EOP's. Closing these valves will reduce the steam release path from the steam generator with a failed open MSIV to the condenser to an acceptable level. These valves are ultimately powered from the 13.8 KV transformer, which is lost on a loss of offsite power. With offsite power available, the valve can be manually closed from the control room. Without offsite power, these valves can be manually closed locally.

The proposed change can be performed with no adverse affects on any UFSAR described functions and can therefore be implemented without prior NRC approval.

2. 07-17012-4 - Change to UFSAR 15.6.3 Steam Generator Tube Rupture Margin to Overfill Analysis

Description: The change to UFSAR section 15.6.3 revises the description and results of the Steam Generator Tube Rupture margin to overfill analysis and plant Emergency Operating Procedures. The following are the key changes to the analysis.

a. Decrease in decay heat assumed in the analysis
b. Eliminating the step to initiate charging during cool down phase of an event
c. Added a step to control Reactor Coolant System (RCS) pressure to minimize RCS leakage to ruptured Steam Generator.
d. Eliminate the assumption of turbine runback prior to reactor trip in the analysis

* - 10 CFR 50.59 reporting requirements state, "A report must be submitted at intervals not to exceed 24 months". It has been greater than 24 months since STP Nuclear Operating Company submitted its previous 10 CFR 50.59 Summary Report. This oversight is being addressed in the STP Corrective Action Program.

Summary: The proposed change was performed in response to the Westinghouse Nuclear Safety Advisory Letter 07-11 with regard to the decay heat assumed in the steam generator tube rupture margin to overfill analysis. To regain lost margin, the operators are now directed to not start charging flow during the cooldown phase of the event. A review of the Westinghouse Owners Group (WOG) background document found this would not result in a failure of the reactor coolant pump seals. The procedure was further enhanced to provide operator guidance with regard to managing RCS pressure after the primary and secondary pressure in the ruptured steam generator are equalized. Finally, the assumption of turbine runback prior to reactor trip was removed since STP does not have this feature. The methodology used to perform the reanalysis did not change from the current methodology described in WCAP 10698-P-A. The results of the evaluation show that the changes do not require prior NRC approval.

3. **08-15047-3 - Change to UFSAR Table 15.0-4 and Table 16.1-3**

Description: The proposed change revises the delay time for the Power Range Neutron Flux Positive Rate Reactor Trip (PFRT) on UFSAR Tables 16.1-3 and 15.0-4. The change is being made to address the revised uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power Overpressure analysis discussed in Westinghouse Nuclear Safety Advisory Letter (NSAL) 09-01.

Summary: The revised analysis reduced the delay time of 3 seconds assumed in the original analysis to 0.65 seconds to ensure the peak RCS pressure did not exceed 110% of design pressure for the uncontrolled Rod cluster Control Assembly Bank Withdrawal at Power. A review of the circuitry found the PFRT and high neutron flux have essentially the same time response of 0.5 seconds stated on UFSAR Table 16.1-3. The results of the evaluation show that the change does not require prior NRC approval.

4. **09-7557-3 - Change to UFSAR Table 15.3-2**

Description: The proposed change revises the maximum reactor coolant system pressure for the locked rotor event presented in UFSAR Table 15.3-2.

Summary: In NSAL 09-2, Westinghouse identified the locked rotor RCS over-pressurization analysis incorrectly assumed that the initial pressurizer water level did not impact the results. The analysis discussed in UFSAR section 15.3.3 for the locked rotor overpressure event assumed an initial pressurizer level of 55% plus a 7.1% instrument uncertainty (62.1 %). The actual indicated water level at full power is 57%. Applying the same 7.1% instrument uncertainty, the correct water level should be 64.1 %. To address the issue identified in the NSAL, Westinghouse performed a generic assessment that determined an increase of 10% in indicated initial pressurizer level equated to an increase in Reactor Coolant System (RCS) pressure of 41 psi. The methodology used in the assessment is the same as was used for the analysis in the UFSAR. Applying this sensitivity, the peak RCS pressure for this analysis was increased by 9 psi from 2637 psia to 2646 psia to account for the increase of 2% in the initial pressurizer water level not assumed in the analysis presented in the UFSAR. The increase in pressure to 2646 psia is below the acceptance limit of 2748.5 psia (110% design pressure). The results of the evaluation show that the change does not require prior NRC approval.

5. **09-18094-3 - Change to UFSAR to add the results of additional analysis for Containment steam line break analysis for Unit 1 Cycle 16 and Unit 2 Cycle 14.**

Description: The proposed change revises the UFSAR to reflect the results of analysis for the containment steam line break pressure and temperature analysis for Unit 1 Cycle 16 and Unit 2 Cycle 14.

Summary: The analysis was performed to address a more positive Moderator Density Coefficient (MDC) than was assumed in the analysis of record. An evaluation was performed that developed a penalty factor to account for the more positive MDC. This penalty factor was applied to the current mass and energy releases, resulting in an increase in mass and energy releases. The increase in mass and energy releases resulted in an increase in containment pressure and temperature for the steam line break event using the currently licensed CONTEMPT methodology. The results of the analysis show that the peak containment pressure increases from 27.3 psi to 30.0 psi, which remains bounded by the Loss of Coolant Accident (LOCA) peak pressure of 41.2 psi. The results also show that the peak containment temperature increases from 328°F to 329°F. However, sufficient margin exists to accommodate this increase without exceeding the design basis of any equipment or structures. The results of the evaluation show that the change does not require prior NRC approval.

6. **09-18620-8 - Change to UFSAR to reflect the revised containment atmospheric pressure and temperature analysis for the steam-line break event.**

Description: The proposed change revises the UFSAR to reflect the revised containment atmospheric pressure and temperature analysis for the main steam line break (MSLB) event.

Summary: The revised analysis was performed to remove the penalty factor applied to the mass and energy releases discussed in 10CFR50.59 evaluation 09-18094-3. The analysis consists of two parts. The first part addresses an increase in mass and energy releases due to the more positive MDC than originally assumed in the current analysis of record. This analysis uses the currently approved RETRAN methodology. The second part evaluates the impact of the RETRAN mass and energy releases on containment atmospheric pressure and temperature. The second part uses the GOTHIC computer code in place of the currently licensed CONTEMPT computer code. The GOTHIC analysis used methodology that incorporated the constraints and limitations identified in topical reports and SERs previously approved by the NRC.

The results of the analysis show that the peak containment pressure stays the same at 30 psig, which remains bounded by the LOCA peak pressure of 41.2 psig. The results also show that the peak containment temperature decreases from 329°F to 299°F, due to margin afforded by the GOTHIC methodology. The results of the evaluation show that the changes do not require prior NRC approval.

7. 00-10937-88 Revision 1 - Replacement of the Main Feedwater Isolation Valve circuitry

Description: Revise the 10CFR50.59 evaluation for the change to the Main Feedwater Isolation Valves (MFIVs) solenoid control circuit operation and testing circuitry to reflect a revised Failure Modes and Effects Analysis (FMEA).

Summary: During preparation of revision 0 of this evaluation, the FMEA failed to identify a failure mode that could result in the main feedwater isolation valves failing open. This failure could result in an increase in the frequency of an accident previously evaluated in the UFSAR (Chapter 15.1.1 & 15.1.2, Feedwater system malfunction causing a reduction in feedwater temperature or increase in feedwater flow) or increase the likelihood of a occurrence of a malfunction of an SSC important to safety. The evaluation demonstrated that the increase in frequency of a feedwater malfunction event was less than 10% and increase in the likelihood of occurrence of a malfunction was less than a factor of 2. Therefore, the proposed change can be implemented under 10CFR 50.59 without prior approval from the NRC.

8. 10-17816-1 - Change to UFSAR Section 3.6.2

Description: Revised UFSAR Section 3.6 to add the statement, "An assumed single failure of safety related active component concurrent with the non-mechanistic break is not required".

Summary: During the 2010 Component Design Basis Inspection (CDBI), the NRC identified a change to the UFSAR required a full 10CFR50.59 evaluation. The proposed change addresses the methodology used in the evaluation of High Energy Line Break Failure Modes and Effects Analysis for breaks in superpipe. The proposed change involves only a change to a method of evaluation. The 50.59 evaluation concluded that prior NRC approval was not required.