STP Nuclear Operating Company

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

February 9, 2004
NOC-AE-04001677
10CFR50.73
10CFR50.72

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Unit 1
Docket Nos. STN 50-498
Licensee Event Report 1-03-006
Unanalyzed Condition That Significantly Degraded Plant Safety
Regarding the Natural Circulation Cool Down Rate

Pursuant to 10CFR50.73, the South Texas Project submits the attached Unit 1 Licensee Event Report 1-03-006 regarding an unanalyzed condition that significantly degraded plant safety regarding the natural circulation cool down rate for Unit 1 and Unit 2. This event did not have an adverse effect on the health and safety of the public.

There are no commitments contained in this event report. Resulting corrective actions will be handled in accordance with STP Corrective Action Program.

If there are any questions on this submittal, please contact S. M. Head at (361) 972-7136 or me at (361) 972-7849.

E.D. [signature]

E. D. Halpin
Plant General Manager

KJT/

Attachment: LER 1-03-006 (South Texas, Unit 1)

IE22

cc:

(paper copy)

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NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a

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4. TITLE

Unanalyzed condition that significantly degraded plant safety regarding the natural circulation cool down rate

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME South Texas Unit 2	DOCKET NUMBER 05000 499
12	09	2003	2003 -	06 -	00	02	09	2004	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR ': (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)	X	50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36©(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36©(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36©(2)		50.73(a)(2)(v)(B)	X	OTHER Part 21
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Ken Taplett	361-972-8416

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On December 9, 2003, Unit 1 and Unit 2 were operating at 100% power. During the performance of a fire hazards reactor coolant system (RCS) cool down analysis, it was discovered that the current procedural cool down rate of 50°F/hr does not satisfy the design basis requirements when less than 4 steam generators are available during natural circulation cool down conditions. This condition placed both Unit 1 and Unit 2 in an unanalyzed condition that significantly degraded plant safety.

The root causes of this event were the failure of the Westinghouse Owner's Group to fully address RCS loop stagnation conditions for natural circulation cool down and the failure to re-analyze the natural circulation cool down cases when a change was made to increase the cool down rate to 50°F/hr.

Corrective actions were taken to limit the natural circulation cool down rate and to complete long-term cooling analyses to validate the revised rate.

This event resulted in no personnel injuries, no offsite radiological releases, and no damage to safety-related equipment. There were no challenges to plant safety.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

I. DESCRIPTION OF EVENT

A. REPORTABLE EVENT CLASSIFICATION

This event is reportable pursuant to 10CFR50.73(a)(2)(ii)(B), any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety.

B. PLANT OPERATING CONDITIONS PRIOR TO EVENT

STP Unit 1 and Unit 2 were in Mode 1 operating at 100% power.

C. STATUS OF STRUCTURES, SYSTEMS OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

All structures, systems and components were determined to be OPERABLE at the start of the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

During the performance of a fire hazards reactor coolant system (RCS) cool down analysis, it was discovered that the current procedural cool down rate of 50°F/hr does not satisfy the design basis requirements when less than 4 steam generators are available during natural circulation cool down conditions. The analysis demonstrated that the active RCS loop could be cooled down to the residual heat removal (RHR) system cut-in temperature of 350°F; but the inactive loop would flash to steam. Therefore, controlled cool down could not be performed.

STP was originally designed as a reactor vessel head (RVH) T_{HOT} plant. With the increase in the number of steam generator tubes when the steam generators were replaced, the net RCS flow through the core increased. In order to reduce the fuel uplift forces, the flow bypass nozzles in the reactor vessel lower internal flange were modified to allow for more flow to the upper head area. The increase in bypass flow to the upper head classified STP as a T_{COLD} plant.

Westinghouse performed an evaluation for changing the cool down rate from 25°F/hr to 50°F/hr and determined the change to be acceptable based on Westinghouse Owner's Group guidelines for other T_{COLD} plants. The change of the reactor vessel head from T_{HOT} to T_{COLD} allowed changing the cool down rate from 25°F/hr to 50°F/hr for natural circulation cool down in accordance with Emergency Operating Procedure Background documents.

While performing a natural circulation cool down analysis at STP, it was discovered that the plant could not achieve cool down with only two available steam generators with a 50°F/hr cool down rate.

Water in the tube side of the two steam generators that were not receiving auxiliary feedwater (AFW) would stagnate with hot water and flash to steam during the depressurization process. This would result in filling the pressurizer and causing a loss of RCS pressure control. Cool down and

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depressurization of the RCS in this configuration would take substantially longer than that assumed in the current analyses. A long cool down time could result in depletion of the AFW storage tank inventory prior to reaching the residual heat removal (RHR) cut-in conditions contrary to plant design requirements.

Although there is adequate core cooling, the plant would be in a condition that is outside the existing design basis and not specifically addressed by the Emergency Operating Procedures. The evaluation affects safe shutdown for fire and long term cooling analyses that result in loss of AFW flow to one or more steam generators.

Following discovery of the event, the natural circulation cool down rate was conservatively reduced. Long-term cooling analyses are being performed to validate the cool down rate to successfully meet the functional objective of bringing the plant to cold shutdown conditions.

This event was discovered at 1538 on December 9, 2003. Notification was made to the Nuclear Regulatory Commission at 1732 on December 9, 2003.

E. METHOD OF DISCOVERY OF EACH COMPONENT, SYSTEM FAILURE, OR PROCEDURAL ERROR

This condition was identified during the performance of a fire hazards reactor coolant system (RCS) cool down analysis.

II. EVENT DRIVEN INFORMATION

A. SAFETY SYSTEMS THAT RESPONDED

Not Applicable.

B. DURATION OF SAFETY SYSTEM INOPERABILITY

Not Applicable.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

This event did not result in personnel injuries, radiation exposure, onsite or offsite radiological releases nor damage to important safety related equipment. The higher cool down rate is acceptable with forced circulation in all the steam generators. South Texas has not had to cool the plant down with less than 100% steam generator availability. However, this event resulted in a condition that could have prevented fulfillment of a safety function. The safety function is the ability of the AFW storage tank to provide sufficient inventory to cool the RCS down to RHR cut-in conditions.

There is no impact or change in core damage frequency since the higher cool down rate was acceptable for the plant conditions at the time (i.e., all four steam generators were available for cool down).

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III. CAUSE OF THE EVENT

1. Failure of the Westinghouse Owner's Group (WOG) to identify all limiting scenarios in the WOG Guidelines for Natural Circulation Cool down. Specifically, the WOG Guidelines addressed upper head cooling and did not fully address loop stagnation.

2. Failure to re-analyze the natural circulation cool down cases when a change was made increase the cool down rate to 50°F/hr.

IV. CORRECTIVE ACTIONS

A. Administrative requirements were put in place to limit the natural circulation cool down rate.

B. Complete long-term cooling analyses to validate and change the natural circulation cool down rate.

C. Prepare a design change package to revise the EOP Set Point Document to change the natural circulation cool down rate.

D. Revise plant operations procedures that require the operators to use a plant cool down rate.

V. PREVIOUS SIMILAR EVENTS

A. The Westinghouse methodology that was used to determine the worth of the highest worth stuck rod in the STP analysis could result in a non-conservative shutdown boron concentration.

B. The Westinghouse methodology for the STP loss of normal feedwater/loss of offsite power analysis did not include the zero per cent steam generator tube plugging case that resulted in filling of the pressurizer.

VI. ADDITIONAL INFORMATION

The inability to cool down during natural circulation with less than all Steam Generators available could impact other plants that have used the WOG Guidelines. Westinghouse is currently investigating the need to notify specific plants of the natural circulation cool down unanalyzed condition. [Reference: CAPs-ACA-03-351-M007, Rev. 0, Inactive Loop Flow Stagnation During Natural Circulation Cooldown (Westinghouse internal document)] In addition, STP is developing a direct work request for the WOG to address the ability of plants to cool down to RHR cut-in conditions with a dry steam generator.