The Light company
Houston Lighting & Power South Texas Project Electric Generating Station P. O. Box 289 Wadsworth, Texas 77483

June 13, 1994
ST-HL-AE-4813
File No.: G26
10CFR50.73

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

South Texas Project
Unit 2
Docket No. STN 50-499
Licensee Event Report 94-004
Engineered Safety Features Actuation due to Filling
a Steam Generator to Approximately 90% Full

Pursuant to 10CFR50.73, Houston Lighting & Power (HL&P) submits the attached Unit 2 Licensee Event Report 94-004 regarding Engineered Safety Features Actuation due to filling a Steam Generator to approximately 90% full. Although this actuation did not have an adverse effect on the health and safety of the public, it does not meet our standards for expected operational performance.

If you should have any questions on this matter, please contact Mr. J. M. Pinzon at (512) 972-8027 or me at (512) 972-8664.

J. F. Groth
Vice President,
Nuclear Generation

JMP/esh

Attachment: LER 94-004 (South Texas, Unit 2)

IE22
1/1

c:

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NRC FORM 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
South Texas Unit 2	05000 499	1 OF 4

TITLE (4) Engineered Safety Features Actuation due to Filling a Steam Generator to Approximately 90% Full

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
05	12	94	94	-- 004 --	00	06	13	94		05000
									FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	4
POWER LEVEL (10)	0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.402(b)		20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
	20.405(a)(1)(i)		50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
	20.405(a)(1)(ii)		50.36(c)(2)	50.73(a)(2)(vii)	OTHER
	20.405(a)(1)(iii)		50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iv)		50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
	20.405(a)(1)(v)		50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jairo Pinzon - Staff Engineer	(512) 972-8027

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

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

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

On May 12, 1994 at 2210 hours, Unit 2 was in Mode 4 at 0% power. A Steam Generator Hi-Hi Level (P-14) signal was generated while performing Resistance Temperature Detector cross calibrations. Steam Generator levels were being maintained between 70% and 80% in accordance with the controlling procedure using Auxiliary Feedwater. Inadvertently, the P-14 setpoint was reached, actuating a P-14 signal. Because Main Feedwater was not in service and the main turbine was not on line, components actuated by a P-14 signal were already in their actuated condition, thus no components changed status. The cause of this event was inadequate control board awareness and turnover communications. Corrective actions included crew briefings, sitewide distribution of lessons learned and constructive discipline of involved personnel.

DESCRIPTION OF THE EVENT

On May 12, 1994 at 2210 hours, Unit 2 was in Mode 4 at 0% power. A Steam Generator Hi-Hi Level (P-14) signal was generated in response to a high level in the Steam Generators.

Steam Generator levels were being maintained between 70% and 80%, using Auxiliary Feedwater in accordance with the controlling procedure while performing Resistance Temperature Detector cross calibrations. The secondary reactor operator, who was in charge of maintaining Steam Generator levels, was relieved by the primary operator. The relieving operator noticed that the level on the 2D Steam Generator was nearing the lower end of the band and began filling the Steam Generator. The secondary operator returned and resumed his position, but was not informed of the Steam Generator fill evolution in progress. Shortly thereafter, a Main Control Board "Hi-Hi SG level" alarm was received. Auxiliary Feedwater to the Steam Generator was secured. All four channels of Steam Generator Hi-Hi level received an actuation signal. At the time of the actuation signal, all the components actuated by the P-14 signal were in the condition covered by the actuation, thus no components changed status.

The event was reviewed and determined to be not reportable as an Engineered Safety Features actuation since no components changed status and the plant was in a mode in which P-14 was not required. Therefore, this event was not initially reported to the Nuclear Regulatory Commission. On May 16, 1994, after additional review, it was determined that this event was reportable since the actuation logic had been fulfilled. A notification was made to the Nuclear Regulatory Commission on May 16, 1994 at 2155 hours.

CAUSE OF EVENT

The causes of this event were an inadequate turnover and failure to adequately monitor an evolution in progress.

Because there was no change of status of any Engineered Safety Features component and the plant was in a mode in which P-14 was not required, the first review determined the event was not reportable. Internal guidance stated that a change of status of a component was required to be considered an actuation.

ANALYSIS OF EVENT

The unplanned actuation of an Engineered Safety Feature (ESF) is reportable pursuant to 10CFR50.73(a)(2)(iv). Feedwater Isolation provides protection to the turbine by isolating feedwater to prevent moisture carry over and also to prevent water accumulation in the steamlines. In addition, the Feedwater Isolation Valves are designed to provide containment isolation in the event of a Steam Generator tube rupture and to terminate the supply of water to the Steam Generators to prevent over cooling of the Reactor Coolant System. There was no adverse radiological or safety consequence as a result of this event.

CORRECTIVE ACTIONS

The following corrective actions have been taken as a result of the inadvertent Engineered Safety Features actuation:

1. Crew briefings discussing this event were conducted in both units. Discussions included:

   a. the need for thorough turnovers
   b. reinforcing management expectations for monitoring of evolutions in progress

2. Lessons learned from this event were provided to station personnel.

3. Action was taken consistent with the Constructive Discipline Program for individuals involved in the event whose performance did not meet expected standards.

The following corrective action has been taken as a result of the failure to recognize this event as a reportable event:

1. The reportability guidance on Engineered Safety Features actuations has been enhanced to clearly state that fulfillment of the actuation logic for an Engineered Safety Features component is reportable even though there may be no change of status of that component.

ADDITIONAL INFORMATION

During the past two years there have been three previous events reported to the Nuclear Regulatory Commission regarding inadequate control board awareness, inadequate communications, or inadequate shift turnovers.

Unit 1 Licensee Event Report 92-015 was submitted documenting an unplanned Engineered Safety Features actuation of a Component Cooling Water pump. The cause of this event was attributed to inattention to operating conditions exacerbated by procedural conditions which required extra attention by the operators.

Unit 1 Licensee Event Report 93-013 was submitted documenting a failure to meet the requirements of Technical Specifications due to performing positive reactivity changes in Mode 5 without an operable Centrifugal Charging Pump. A contributing cause was determined to be an inadequate review of the Operability Tracking Log during shift change.

Unit 2 Voluntary Licensee Event Report 93-012 was submitted documenting a loss of spent fuel pool cooling for approximately 13 hours. Causes were determined to be: insufficient control board awareness, and failure to identify this problem during the shift change control board walkdown or during subsequent normal control board walkdowns.