NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Virgil C. Summer Nuclear Station	0 5 0 0 0 3 9 5	1 OF 0 3

TITLE (4)
Pressurizer Safety Valve Setpoints Out Of Tollerance

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 NAMES	DOCKET NUMBER(S)
0 3	3 1	8 7	8 7	0 0 5	0 0	0 4	3 0	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)		THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
6		20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)		20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
0 0 0		20.405(a)(1)(ii)	50.36(c)(2)	X 50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
		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
	AREA CODE
W. R. Higgins - Associate Manager, Regulatory Compliance	8 0 3 — 3 4 5 - 4 0 4 2

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 fifteen single-space typewritten lines) (16)

On March 31, 1987, South Carolina Electric & Gas Company (SCE&G) was notified that all three pressurizer safety valves were tested and the as-found setpoints were outside the Technical Specification limits of 2485 ± 1% psig.

Preliminary analysis results indicate the differences in test methods used to measure the setpoints of the valves in 1984 and 1987 are the most probable cause of the different values.

To assure operability of the pressurizer safety valves during the fourth operating cycle, SCE&G had the setpoint of the three valves verified at Wyle Laboratories (two valves) and Crosby Valve and Gage Co. (one valve). In addition, SCE&G will conduct in situ testing of the safety valves during startup following the third refueling outage.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

PLANT IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

EQUIPMENT IDENTIFICATION:

Pressurizer Safety Valves XVR-8010-RC (Valves A, B and C)

IDENTIFICATION OF EVENT:

All pressurizer safety valves were tested and the as-found setpoints were outside 2485 ± 1% psig. (Technical Specification Section 3.4.2, "Safety Valves.")

EVENT DATE: 03/31/87

REPORT DATE: 04/30/87

This report was initiated by Off-Normal Occurrence Number 87-034.

CONDITIONS PRIOR TO EVENT:

Mode 6 - third refueling outage

DESCRIPTION OF EVENT:

On March 31, 1987 with the plant in Mode 6 for the third refueling outage, the Licensee was notified that all three pressurizer safety valves were tested and the as-found setpoints were outside the Technical Specification limits of 2485 ± 1% psig but less than the safety limit of 2735 psig. On April 1, 1987, South Carolina Electric & Gas Company (SCE&G), reported per 10CFR50.72(b)(2)(i) the deviation of the pressurizer safety valves setpoints outside the Technical Specification limits. Although no safety significance determination had been made at that time, SCE&G thought it prudent to report the anomalous condition of the safety valves. Three safety valves had been removed and tested. The as-found setpoints were as follows:

Valve	Valve Serial	Nameplate Set Pressure (PSIG)	Set Pressure Runs (PSIG)			
			1	2	3	4
8010A	0076	2485	2606	2614	2525	2534
8010B	0079	2485	*	2724	2717	2726
8010C	0077	2485	*	2694	2681	2678

*Steam test pressure less than actuation pressure.

CAUSE OF EVENT:

SCE&G considers the use of different testing methods to be a primary contributor to this situation. These valves were last tested using the hot nitrogen method and setpoints verified for the second operating cycle operability in April 1984. The 1987 test at Wyle involved the use of steam.

SCE&G performed a detailed review of pressurizer safety valve testing methodology in response to this situation. Although not conclusive, a preliminary analysis indicates a significant difference in as-measured setpoints that varies as a function of methods utilized to simulate in-plant environmental conditions. It is this testing methodology difference which is most likely the cause of the disparity between the as-set and as-found setpoints.

ANALYSIS OF EVENT:

Westinghouse reviewed all Final Safety Analysis Report (FSAR) Chapter 15 transients which explicitly take credit for the pressurizer safety valves opening and relieving steam. These transients are:

1) Loss of External Electrical Load/Turbine Trip
2) Reactor Coolant Pump Shaft Seizure (Locked Rotor)
3) Feedwater System Pipe Break

A consequence evaluation using the as-found setpoints was performed utilizing available, plant design features and current analytical tools. The results of this consequence evaluation demonstrates that the conclusions in the FSAR for these transients remain unchanged.

CORRECTIVE ACTION:

SCE&G is conducting an evaluation to determine the reason for the high setpoints. Preliminary results indicate that the differences in test methods used to measure the setpoints of the valves in 1984 and in 1987 are the most probable cause of the different values.

To assure operability of the pressurizer safety valves during the fourth operating cycle, SCE&G had the setpoint of the three valves verified at Wyle Laboratories (two valves) and Crosby Valve & Gage Co (one valve). In addition, SCE&G will conduct in situ testing of the safety valves during startup after the third refueling outage. This testing will be conducted at hot plant conditions using a steam medium and a calibrated pressure assist device, together with a test method which best simulates actual plant conditions.

10 CFR 50.73

SCE&G
A SCANA Company

South Carolina Electric & Gas Company
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(803) 748-3513

Dan A. Nauman
Vice President
Nuclear Operations

April 30, 1987

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

SUBJECT: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
LER 87-005

Dear Sir:

Attached is Licensee Event Report No. 87-005 for the Virgil C. Summer Nuclear Station. This report is submitted pursuant to the requirements of 10 CFR 50.73(a)(2)(vii).

Should there be any questions, please call us at your convenience.

Very truly yours,

J. G. Connelly for
D. A. Nauman

MDB/DAN:bjh

Attachment

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