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C. K. McCoy
Vice President, Nuclear
Vogtle Project

October 24, 1996

LCV-0896

Docket Nos. 50-425

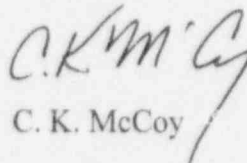
U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
LICENSEE EVENT REPORT
PRESSURIZER SAFETY VALVE'S
LIFT SETPOINT FOUND OUT OF TOLERANCE

In accordance with the requirements of 10 CFR 50.73, Georgia Power Company submits the enclosed report related to an event which was discovered on September 25, 1996.

Sincerely,


C. K. McCoy

CKM/NJS

Enclosure: LER 50-425/1996-005

xc: Georgia Power Company
Mr. J. B. Beasley, Jr.
Mr. M. Sheibani
NORMS

U. S. Nuclear Regulatory Commission
Mr. S. D. Ebner, Regional Administrator
Mr. L. L. Wheeler, Licensing Project Manager, NRR
Mr. C. R. Ogle, Senior Resident Inspector, Vogtle

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PDR ADOCK 05000425
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LICENSEE EVENT REPORT (LER)

EXPIRES: 04/30/96

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REQUIRED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F22), 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.

FACILITY NAME (1)

Vogtle Electric Generating Plant - Unit 2

DOCKET NUMBER (2)

500004251 OF 4

PAGE (3)

TITLE (4)

PRESSURIZER SAFETY VALVE'S LIFT SETPOINT FOUND OUT OF TOLERANCE

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(S)
09	25	96	96	005	00	10	24	96		500004251

OPERATING MODE (9)

0

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

POWER LEVEL (10)

0

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	X 50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2033(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2033(c)(1)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME

Mehdi Sheibani, Nuclear Safety and Compliance

TELEPHONE NUMBER (include area code)

706 826-1320 9

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
B	A/B	R/V	C7110	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE)

X

NO

EXPECTED SUBMISSION DATE (15)

MONTH DAY YEAR

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

During testing at a vendor lab, the pressurizer safety valves (PSVs) were found to open at pressures above their setpoints. As found setpoints were 1.4 percent, 2.3 percent, and 3.2 percent above the specified setpoints. Following discussions with the NSSS vendor, personnel were unable to conclude that any safety analyses could support the as-found lift setpoint of 3.2 percent above the specified setpoint. As a result, it was determined that a condition may have existed that placed the unit in an unanalyzed condition that significantly compromised plant safety.

The cause of this event was determined to be a drift in the valves' lift setpoints. A possible contributing cause to this condition was wear of bellows and spindle assemblies in two of the three valves and a difference in the ambient temperatures employed during valve testing. The worn bellows and spindles were replaced, the PSVs' setpoints were readjusted, and the valves were reinstalled in the unit.

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Vogtle Electric Generating Plant - Unit 2	05000425	96	005	00	2	OF	4

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

A. REQUIREMENT FOR REPORT

This report is required per 10 CFR 50.73 (a)(2)(ii)(A) because a condition was found, while the unit was shutdown, that may have placed the unit in an unanalyzed condition that significantly compromised plant safety.

B. UNIT STATUS AT TIME OF EVENT

At the time of the discovery of this event, Unit 2 was defueled. Other than that described herein, there was no inoperable equipment that contributed to the occurrence of this event.

C. DESCRIPTION OF EVENT

At the end of the fifth fuel cycle, the pressurizer safety valves (PSVs) were removed from the unit and sent to an independent lab for testing. On September 21st and 22nd, 1996, the lab reported that the PSVs had failed to open within the prescribed range of pressures during the initial lift tests. The setpoint pressure is 2485 psig, with a tolerance range of plus or minus 1 percent. Valve 2PSV-8010A lifted at 2.3 percent above the setpoint. Valve 2PSV-8010B lifted at 1.4 percent above the setpoint, and valve 2PSV-8010C lifted at 3.2 percent above the setpoint.

The as-found lift setpoints of 1 to 3 percent above the original setpoints, while unanalyzed, are not believed to significantly compromise plant safety. However, following discussions with the NSSS vendor, personnel were unable to conclude that any safety analyses could support the as-found lift setpoint of 3.2 percent above the specified setpoint for 2PSV-8010C. Therefore, assuming the lift pressure setpoints found at the lab were the same as when the PSVs were installed, on September 25, 1996, at 1500 EDT, personnel determined that the unit may have operated in an unanalyzed condition that significantly compromised plant safety. The NRC Operations Center was notified of this condition at 1736 EDT.

D. CAUSE OF EVENT

The cause of this event was determined to be a drift in the valves' lift setpoints. A possible contributing cause to this condition was wear of bellows and spindle assemblies in valves 2PSV-8010A and 2PSV-8010C. Another possible contributing factor was a procedural change that stipulated a different test temperature than had previously been used. The PSVs' setpoints had been

LICENSEE EVENT REPORT (LER)
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TEXT (If more space is required, use additional copies of NRC Form 366A)(17)

previously set when the ambient air and valve body temperatures were at 140 degrees F. When the tests were conducted on September 21st and 22nd, the ambient air and valve body temperatures were approximately 106 degrees F, which more closely corresponds to plant operating conditions. The root cause of this event is undetermined. Georgia Power Company is presently evaluating these results and if additional information on the root cause and corrective action is identified a supplement to this LER will be submitted.

E. ANALYSIS OF EVENT

Maintaining the power operated relief valves in an operable condition during the last fuel cycle ensures that capacity was available to relieve a reactor coolant system (RCS) overpressurization event. Furthermore, the RCS has been tested to 125 percent of design pressure, providing additional assurance that the system could maintain its integrity in a pressure transient. Finally, no overpressurization event occurred during the last operating cycle. Based on these considerations, there was no adverse affect on plant safety or on the health and safety of the public as a result of this event.

F. CORRECTIVE ACTION

- 1) Bellows and spindles were replaced on 2PSV-8010A and 2PSV-8010C. The lift setpoints were readjusted for all three PSVs at an ambient temperature of approximately 106 degrees F, and the valves were reinstalled in the unit.
- 2) The NSSS vendor is performing calculations to evaluate an increase in the PSV setpoint tolerance.
- 3) Georgia Power Company will evaluate the present maintenance practices to determine methods to improve the reliability of of these valves.

G. ADDITIONAL INFORMATION

- 1) Failed Components:
PSVs manufactured by Crosby Valve and Gage Company.
Model #HB-BP-86
- 2) Previous Similar Events:
None

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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

3) Energy Industry Identification System Code:
Reactor Coolant System - AB