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April 25, 2001
LIC-01-0040

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

Reference: Docket No. 50-285

SUBJECT: Licensee Event Report 2001-001 Revision 0 for the Fort Calhoun Station

Please find attached Licensee Event Report 2001-001, Revision 0, dated April 25, 2001. This report is being submitted pursuant to 10 CFR 50.73(a)(2)(i)(B). If you should have any questions, please contact me.

Sincerely,

S. K. Gambhir

S. K. Gambhir
Division Manager
Nuclear Operations

SKG/EPM/epm

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
L. R. Wharton, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
INPO Records Center
Winston and Strawn

IE22

NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
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 person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Fort Calhoun Nuclear Station Unit Number 1	05000285	1 OF 3

TITLE (4)
Primary Safety Valves Outside Lift Setting Acceptance Range

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	26	2001	2001	- 001 -	00	04	25	2001	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	5
POWER LEVEL (10)	0

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

	20.2201(b)		20.2203(a)(3)(ii)		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(c)(1)(i)(A)		50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	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)	X	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)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
James Geschwender, Component Testing Engineer	402-533-6857

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

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)**

At 1117 hours on March 26, 2001, during a scheduled refueling outage, it was discovered that the "as-found" lift pressure of primary safety valve (PSV) RC-142 was outside its specified lift setting acceptance range. Fort Calhoun Station (FCS) Technical Specification (TS) 2.1.6(1) specifies that this valve is to have its lift setting adjusted to ensure valve opening at 2485 pounds per square inch gauge (psig) +/- 1 percent (i.e., a range of 2460.2 to 2509.8 psig). The "as-found" lift pressure was 2459.9 psig, which is 1.01 percent below the nameplate lift pressure.

At 1225 hours on March 29, 2001, it was discovered that the "as-found" lift pressure of PSV RC-141 was outside its specified lift setting acceptance range. TS 2.1.6(1) specifies that this valve is to have its lift setting adjusted to ensure valve opening at 2530 psig +/- 1 percent (i.e., a range of 2504.7 to 2555.3 psig). The "as-found" lift pressure was 2485.9 psig, which is 1.74 percent below the nameplate lift pressure.

It has been concluded that the root cause of this event was normal setpoint variance.

Corrective actions included additional lift testing of RC-142 to obtain repeatable lift pressures within the +/- 1 percent tolerance range. RC-141 was disassembled, cleaned, inspected, and refurbished, followed by testing and adjustment to obtain repeatable lift pressures within the +/- 1 percent tolerance range.

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

BACKGROUND

Overpressure protection at the Fort Calhoun Station (FCS) is ensured by means of primary safety valves (PSV) (EIIS: RV), secondary safety valves, and the reactor protection system. Technical Specification (TS) 2.1.6(1) specifies that the reactor shall not be made critical unless the two PSVs are operable with their lift settings adjusted to ensure valve opening at 2485 pounds per square inch gauge (psig) +/- 1 percent and 2530 psig +/- 1 percent. The two PSVs are RC-141 and RC-142. The lift setting criterion for RC-141 is 2530 psig +/-1 percent (i.e., a range of 2504.7 to 2555.3 psig), and the criterion for RC-142 is 2485 psig +/- 1 percent (i.e., a range of 2460.2 to 2509.8 psig).

EVENT DESCRIPTION

During the 2001 refueling outage, PSVs RC-141 and RC-142 were removed from service and sent off-site to Wyle Laboratories for setpoint verification. On March 26, 2001, at 1117 hours, the "as-found" lift pressure for RC-142 was discovered to be 2459.9 psig (or 1.01 percent below the nameplate lift pressure value). The next four lifts (at 2472.7, 2471.9, 2485.5, and 2472.3 psig respectively) were found to be within the required setpoint tolerance without adjustment.

On March 29, 2001, at 1225 hours, the "as-found" lift pressure for RC-141 was found to be 2485.9 psig (or 1.74 percent below the nameplate lift pressure value). The next four lifts (at 2498.3, 2513.0, 2522.4, and 2501.2 psig respectively) averaged almost 1 percent higher than the initial lift without adjustment. The valve was subsequently disassembled, cleaned, inspected, and refurbished. No significant deficiencies were noted in the condition of the valve. The disc and nozzle were acceptable as found, but were replaced because they were approaching minimum tolerances. The valve was reassembled and tested. The three "as-left" lifts (at 2516.8, 2517.8, and 2529.6 psig respectively) were within the required setpoint tolerance.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(i)(B).

SAFETY SIGNIFICANCE

Nuclear safety was not adversely impacted by this condition. The PSVs provide overpressure protection for the reactor coolant system. Both valves had "as-found" opening pressures below their nameplate setpoints, ensuring that overpressure protection would have been provided. Although both valves opened below the specified tolerance, the opening pressures were well above the pressure at which the reactor is designed to trip (i.e., nominal 2350 psi absolute), and at which the power operated relief valves (PORVs) are designed to open (i.e., nominal 2350 psi absolute). Therefore, the condition could not have initiated a transient or otherwise adversely affected plant operation.

CONCLUSION

It was concluded that the root cause of this event was normal setpoint variance. This conclusion was reached after reviewing test results for RC-141 and RC-142, and disassembly, cleaning, and inspection of RC-141 by the vendor field service representative.

A number of factors can contribute to minor safety valve setpoint variance, including slight variations in alignment of internal components, variations in temperature gradients within the valve, imperfections in disc-to-seat contact, and others. Even when considerable care is exercised, the process of removing, shipping, and preparing the valves for off-site steam set pressure testing may affect alignment of internal components. Even when reasonable measures are taken to control and monitor temperature conditions during testing, temperature gradients within the valve (especially temperatures of internal components such as the nozzle and spring) can not be expected to be identical from test to test. Due to these and other factors, it is not practical to perfectly diagnose, predict, or eliminate all setpoint variance. American Society of Mechanical Engineers (ASME) / American National Standards Institute (ANSI) OM-1, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," requires a determination of cause if a primary safety valve is out of tolerance by 3 percent or more. Setpoint variations below this threshold can generally be regarded as normal setpoint variance. (Reference Electric Power Research Institute (EPRI) report EPRI TR-105872, "Safety and Relief Valve Testing and Maintenance Guide," Section 5.0, "Failure Modes and Failure Cause Analysis," subsection 5.4, "Failure Modes Analysis" for additional discussion of this issue.)

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Fort Calhoun Nuclear Station Unit Number 1	05000285	2001	- 001	- 00	3 OF 3

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

A review of past test results for RC-141 and RC-142 determined that one or both valves have been outside the specified +/- 1 percent range in 1975, 1976, 1977, 1980, 1983, 1984, 1985, 1987, 1992, 1996, 1999 and 2001. These incidents were due to a number of different causes. A more detailed review of recent testing indicates that since 1993, only one "as-found" lift was outside +/- 3 percent of nameplate setpoint (RC-142 in 1996, which was determined to have been caused by incorrect installation of insulation before the test).

EPRI report EPRI TR-105872, subsection 5.4, discusses the occurrence of setpoint variation in which an "as-found" safety valve lift is outside of a technical specification tolerance of +/-1 percent, but within the typical +/- 3 percent vendor design tolerance for safety valves. The report indicates that, "this lifting phenomenon appears to be normal for the types of valves being used in the industry." The EPRI report discusses only one method of responding to this issue. That method is to request relief from TS.

CORRECTIVE ACTIONS

RC-141 was disassembled, cleaned, inspected, and refurbished. The valve was subsequently reassembled and tested. The "as-left" lift pressures for both RC-141 and RC-142 met the TS 2.1.6(1) requirements. TS relief is being considered.

SAFETY SYSTEM FUNCTIONAL FAILURE

This event did not result in a safety system functional failure in accordance with NEI 99-02.

PREVIOUS SIMILAR EVENTS

PSVs RC-141 and RC-142 "as-found" lift pressures have been found outside their +/- 1 percent tolerance range caused by drift on previous occasions as documented by LERs 1977-028, 1983-001, 1987-014, 1992-023 (This LER reported that additional out of specification lifts were recorded in 1975, 1980, 1984, and 1985, but not reported by LER. It is not known if these failures were due to setpoint drift or some other issue.), 1993-013, and 1999-003.