Form 1062.01A

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) Arkansas Nuclear One, Unit 2 | DOCKET NUMBER (2) |0|5|0|0|0| 3| 6| 8| PAGE (3) |1|OF|0|4

TITLE (4) Pressurizer Code Safety Valves Discovered Outside Technical Specifications Required Lift Settings Due To Indeterminate Cause

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| 9	2| 5	8| 6	8| 6	0| 1| 2	0| 1	0| 9	1| 5	8| 8		0|5|0|0|0
										0|5|0|0|0

OPERATING MODE (9): 1

POWER LEVEL (10): 1|0|0

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

20.402(b)	20.405(c)	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 (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	X 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: Larry A. Taylor, Nuclear Safety and Licensing Specialist

Telephone Number (Area Code): 5|0|1|9|6|4|-|3|1|0|0

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
X	A| B	R| V	C| 7| 1| 0	Y					

SUPPLEMENT 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 fifteen single-space typewritten lines) (16)

Evaluation of transient data following a reactor scram on September 24, 1986 (see LER 50-368/86-011) identified indications that a pressurizer code safety valve may have partially lifted during the event at a pressure less than the Technical Specification required minimum lift setting. On September 25, 1986 at 0528 hours, mechanical maintenance personnel performed in situ testing to determine the lift settings for the valves. These test indicated that the setpoints for both valves were below the required range. The valves were declared inoperable, Technical Specification 3.0.3 was entered, and a plant cooldown was commenced. Extensive investigation and evaluation of this event by Arkansas Power & Light, the valve vendor, Wyle Laboratories, and an independent engineering firm could not conclusively identify the cause of the incorrect pressurizer code safety valves lift settings. As a result of the discovery of high piping loading on one of the safety valves, a modification was completed during refueling outage 2R6 to reduce the loading to more closely reflect the manufacturer's recommended limits.

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I. Description of Event

A. Plant Status

At the time of discovery of this event on September 25, 1986, Arkansas Nuclear One, Unit 2 (ANO-2) was in Operational Mode 3 (Hot Standby). ANO-2 was being maintained in Hot Standby to perform testing on the pressurizer [PZR] code safety valves [RV].

B. Component Identification

The ANO-2 pressurizer is equipped with two code ASME safety valves (2PSV-4633 and 2PSV-4634). These valves operate to prevent the reactor coolant system (RCS) [AB] from being pressurized above its safety limit during design basis events. The code safety valves are totally enclosed, back pressure compensated, spring loaded safety valves. They were manufactured by Crosby Valve and Gauge Company (manufacturer code: C710) as model number HB-86-BP. The valves are required by Technical Specifications (TS) to have a lift setpoint of 2500 psia plus or minus one percent (2475 to 2525 psia).

C. Sequence of Events

On September 24, 1986 at 0822 hours, a reactor scram from 100 percent power on high RCS pressure occurred on ANO-2 following a trip of the main turbine due to a secondary system transient (see LER 50-368/86-011). Evaluation of the transient data following the scram identified indications that a pressurizer code safety valve may have partially lifted during the event at a pressure less than the Technical Specification required minimum value for the safety valves lift setpoint. Actions were initiated to determine the lift settings for the valves.

On September 25, in situ testing indicated that the lift settings for both pressurizer code safety valves were below the Technical Specification required minimum value. Both valves were declared inoperable and TS 3.0.3 was entered. A plant cooldown was commenced and ANO-2 reached Operational Mode 4 (Hot Shutdown) at 1217 hours. The unit was maintained in Hot Shutdown to allow additional safety valve testing.

On September 26, mechanical maintenance personnel, with assistance from a representative from the valve manufacturer, performed additional in situ valve testing to verify the lift settings for the valves. These tests reconfirmed that the lift settings were below the Technical Specification required minimum value. The valves were adjusted to increase the lift settings and were tested satisfactorily. Upon completion of these actions, a plant startup was performed. The unit entered Operational Mode 1 (Power Operation) on September 27.

II. Event Cause

A. Event Analysis

The ANO-2 pressurizer code safety valves function to reduce pressure of the RCS if necessary by discharging steam to a quench tank located in the containment building. The discharge piping for each valve to the quench tank contains a resistance temperature detector and an acoustic detector used to provide indications of flow through the safety valves. The review of the transient data for these monitoring devices following the reactor scram on September 24 indicated flow had occurred from valve 2PSV-4633. The maximum RCS pressure reached during the reactor scram event was approximately 2400 psia. Based on this information, it was suspected that a premature, partial lift of 2PSV-4633 had occurred.

Both pressurizer code safety valves had been replaced during a recent refueling outage (2R5) which was completed in August 1986. Valve 2PSV-4633 had been removed from the pressurizer and refurbished by Wyle Laboratories during the outage. Valve 2PSV-4634 had been previously refurbished by Wyle Laboratories in early 1986. The certification test reports supplied to

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

Arkansas Power & Light (AP&L) by Wyle Laboratories following refurbishment and testing of the valves indicated acceptable final lift settings. No further testing or adjustments were performed on the valves until the findings on September 25. The in situ testing performed on September 26 indicated that the as-found lift settings for the safety valves were 2425 psia (2PSV-4633) and 2450 psia (2PSV-4634). After adjustment and retesting, the as-left lift settings were 2510 psia (2PSV-4633) and 2503 psia (2PSV-4634).

As a result of this event, several actions were taken to determine the cause of the incorrect lift settings. A review of the procedures and test methods used by Wyle Laboratories to measure and adjust the lift settings of the valves was performed. Nothing was identified that would have resulted in the incorrect lift settings discovered on September 25. Additionally, an independent engineering firm was contracted to assist AP&L with a longer term evaluation of pressurizer code safety valve problems experienced at ANO, Unit 1 (ANO-1) and ANO-2 (see also LER 50-313/86-007).

The first part of this evaluation included a detailed static piping analysis of the installed configuration of 2PSV-4633, 2PSV-4634, and associated piping. This NUPIPE analysis indicated that the external loads applied to 2PSV-4634 by restrained thermal expansion plus deadweight loads were well within the manufactuer's recommeded limits. However, the applied loading to 2PSV-4633, though still within the manufacturer's stress allowables, was noted to be particularly high. These high loads were postulated to have potentially contributed to a persistent leakage problem with the valve installed on this pressurizer nozzle. Additionally, the valve vendor stated that loads high enough to cause leakage could potentially cause some reduction in lift settings of safety valves. Secondly, an evaluation of ANO and Wyle Laboratories procedures used for valve removal, disassembly and testing was performed. No reasons for the incorrect lift settings could be identified as a result of these evaluations.

Another objective of the long-term evaluation was to determine whether off site or in situ valve tests are more accurate for determining valve lift setpoint. The investigation indicated little difference between the two test methods, assuming the procedures were properly performed. Several other safety valve test methods currently available were also evaluated. Crosby, Dresser, and Furmanite offer on site testing of safety valves using sophisticated computer programs to provide a permanent record of valve testing. Various utilities have reported good results with all three methods. It was concluded that any of the above methods are acceptable when performed properly.

B. Root Cause

Extensive investigation and evaluation of this event by AP&L, the valve vendor, Wyle Laboratories, and an independent engineering firm could not conclusively identify the cause of the incorrect pressurizer code safety valves lift settings. However, it was determined that high valve loading on 2PSV-4633 could have resulted in some reduction of the lift setting for this valve.

C. Safety Significance

The RCS has two code safety valves to provide overpressure protection. These valves are flange mounted on nozzles located on the top of the pressurizer. The valves operate to prevent the RCS from being pressurized above its safety limit of 2750 psia during design basis events. A review of current transient and accident analysis for ANO-2 indicates that a postulated main feedwater line break accident results in the highest calculated peak RCS pressure of any analyzed events. The out of specification lift settings discovered in this event were in a conservative direction with respect to protecting the RCS pressure boundary from an overpressure condition. During the reactor scram transient on September 25, no abnormal decrease in RCS pressure had occurred and additional operator action was not required as a result of the premature lifting of 2PSV-4633. Additionally, the as-found lift settings of the safety valves were within two to three percent of the Technical Specification required 2500 psia lift setting. For these reasons, this event is considered to have minimal safety significance.

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LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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		Year	Sequential Number	Revision Number	
Arkansas Nuclear One, Unit Two	0|5|0|0|0| 3| 6| 8	8| 6	0| 1| 2	0	0|4 OF 0|4

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

D. Basis for Reportability

In Operational Mode 3, TS 3.4.3 requires that all pressurizer code safety valves be operable with a lift setting of 2500 plus or minus 1 percent. As a result of both pressurizer code safety valves being declared inoperable, TS 3.0.3 was entered and a plant cooldown was performed. This constituted operation prohibited by Technical Specifications and is reportable under 10CFR50.73(a)(2)(i)(B).

III. Corrective Actions

A. Immediate

Upon discovery of the incorrect lift settings, both valves were declared inoperable and actions were taken to comply with Technical Specifications.

B. Subsequent

Following the identification of the inoperable safety valves, the lift settings were adjusted to within Technical Specification limits and the valves were declared operable. Extensive investigations and evaluations were performed to determine the cause of the incorrect lift settings. During the last refueling outage (2R6), a modification was completed to reduce the loading on 2PSV-4633 to more closely reflect the manufacturer's recommended limits.

C. Future

As a result of the discovery of out of specification lift settings for both pressurizer safety valves following completion of 2R6 (see Similar Events below), additional corrective actions have been initiated. AP&L plans to evaluate the feasibility of amending the Technical Specification limits on the required lift setpoints for pressurizer safety valves to increase the allowable tolerances. Based on the results of these evaluations, a Technical Specification change request will be developed and submitted if appropriate. Also, AP&L plans to continue to evaluate alternative testing equipment and methods for performing in situ valve testing to determine if improvements in accuracy and repeatability can be achieved.

IV. Additional Information

A. Similar Events

A similar event regarding pressurizer safety valve setpoint discrepancies on ANO, Unit 1 occurred December 21, 1986 and was documented in LER 50-313/86-007 (Information Report).

On April 30, 1988 after heatup following refueling outage 2R6, an in situ test indicated that the lift settings for the pressurizer safety valves were 2455 psia (2PSV-4633) and 2580 psia (2PSV-4634) (see LER 50-368/88-012, Voluntary Report). Prior to being installed during 2R6, valve 2PSV-4633 had been adjusted to have a lift setpoint of 2484 psia at Wyle Laboratories. Valve 2PSV-4634 had been adjusted in situ on September 26, 1986 and no maintenance or testing had been performed on the valve until the finding of the high setpoint on April 30, 1988. The root cause of the setpoint discrepancy for 2PSV-4633 could not be conclusively determined but is thought to be related to differences in test methods used by an off-site facility for valve setpoint measurements and by AP&L for in situ testing. The setpoint discrepancy for 2PSV-4634 was attributed to operational setpoint drift.

B. Supplemental Information

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

ARKANSAS POWER & LIGHT COMPANY

September 15, 1988

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

SUBJECT: Arkansas Nuclear One - Unit 2
Docket No. 50-368
License No. NPF-6
Licensee Event Re[illegible] No. 50-368/86-012-01

Gentlemen:

Attached is the subject supplemental report concerning pressurizer code safety valves discovered outside the Technical Specification required lift settings due to an indeterminate cause.

Very truly yours,

J M Levine

J. M. Levine
Executive Director,
Nuclear Operations

JML:DAH:den

cc: U. S. Nuclear Regulatory Commission
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Washington, DC 20555

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