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 Two | DOCKET NUMBER (2) 0|5|0|0|0| 3| 6| 8 | PAGE (3) 1 OF 0|5

TITLE (4) Setpoint Discrepancies for Pressurizer Code Safety Valves Discovered During In-Situ Testing Following Heatup After Refueling Outage

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| 4	3| 0	8| 8	8| 8	0| 1| 2	0| 0	0| 7	2| 1	8| 8		0|5|0|0|0
										0|5|0|0|0

OPERATING MODE (9) 3

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

POWER LEVEL (10) 0|0|0

__ 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)	X Other (Specify in Abstract below and in Text, NRC Form 366A) Voluntary Rep.
__ 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)
Patrick C. Rogers, Nuclear Safety and Licensing Specialist	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

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)

On 4/30/88 after heatup following refueling outage 2R6, an in situ test was performed on pressurizer safety valve 2PSV-4633. The lift setpoint of the valve was determined to be 2455 psia which is lower than the minimum allowable Technical Specification (TS) value of 2475 psia. 2PSV-4633 was adjusted to comply with TS requirements. Testing was then performed on the other pressurizer safety valve, 2PSV-4634, and the lift setpoint was determined to be 2580 psia which is higher than the maximum allowable TS value of 2525 psia. 2PSV-4634 was adjusted to comply with TS requirements. 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. Prior to being installed during 2R6, valve 2PSV-4633 had been adjusted to have a lift setpoint of 2484 psia at an off-site facility. The setpoint discrepancy for 2PSV-4634 was attributed to operational setpoint drift. Valve 2PSV-4634 had been adjusted in situ on 9/26/86 and no maintenance or testing had been performed on the valve until the finding of the high setpoint on 4/30/88. An evaluation will be performed to determine the feasibility of expanding the TS tolerance for the pressurizer safety valve lift setting. This occurrence is being voluntarily reported for information.

IE22
1/1

I. Description of Event

A. Unit Status

On 4/30/88, Arkansas Nuclear One, Unit Two (ANO-2) was in Mode 3 operation with reactor coolant system (RCS) pressure at 2250 psia and RCS temperature at 545° Fahrenheit.

B. Component Identification

This event involves the discovery of lift setpoints for the pressurizer ASME code safety valves (2PSV-4633 and 2PSV-4634) [AB-RV] outside the limits specified in the Technical Specifications (TS). The Technical Specifications require a lift setting of 2500 psia plus or minus one percent (2475 psia to 2525 psia). The valves are Model Number HB-86-BP safety valves manufactured by Crosby Valve and Gauge Company [C710].

C. Sequence of Events

On 4/30/88 while in Mode 3 after heatup to Hot Standby conditions following refueling outage 2R6, pressurizer safety valve 2PSV-4633 was tested and the lift setpoint of the valve was determined to be 2455 psia which is lower than the allowable Technical Specification value of 2475 psia. 2PSV-4633 was adjusted to increase the setpoint to a value of 2510 psia. After resetting 2PSV-4633, testing was performed on the other safety valve, 2PSV-4634, and the lift setpoint was determined to be 2580 psia which is higher than the maximum allowable Technical Specification value of 2525 psia. 2PSV-4634 was adjusted to decrease the lift setpoint to a value of 2515 psia.

II. Event Analysis

A. Event Cause

The ANO-2 pressurizer is equipped with two ASME code safety valves which function to prevent overpressurization of the RCS during transients and accident conditions. The valves are mounted on two nozzles located on top of the pressurizer. The pressurizer nozzle on which the valve is mounted determines whether the valve is designated as 2PSV-4633 or 2PSV-4634. Currently, ANO-2 utilizes three pressurizer safety valves, two installed on the pressurizer and one spare valve. The valves are periodically alternated between the two pressurizer nozzles. Typically, one of the two installed valves is removed each refueling outage, and the spare valve is installed in its place. The valve which was removed during the outage is refurbished and tested as necessary and then becomes the spare valve. At the next refueling outage the valve which was not removed during the last refueling outage is replaced with the spare valve. This practice allows for one of the valves to be available for refurbishment during non-outage time periods.

Refurbishment normally takes place at an off-site facility. The off-site facility presently being used by Arkansas Power and Light Company (AP&L) is Wyle Laboratories (Wyle). The safety valve is sent to Wyle where the valve is inspected and tested upon receipt to determine the extent of refurbishment needed. Necessary repairs are made and the valve is tested for seat leakage and to determine the lift setpoint. After refurbishment, the valve is adjusted so that the lift setpoint is within the Technical Specification tolerance. The valve is then shipped back to AP&L. The final lift setting testing of the valve at Wyle is performed with the valve heated to a temperature which approximates the ambient conditions present when the valve is installed on the pressurizer and the RCS is at nominal operating temperature and pressure. This testing is considered to be a "hot" setting of the valve and can be used to meet the Technical Specification surveillance requirements in lieu of in situ testing. However, it is AP&L's current policy to perform in situ testing of both code safety valves following each refueling outage.

2PSV-4633 was replaced during the recently completed refueling outage 2R6. The valve used for replacement was a spare which as discussed above had been previously refurbished and set by Wyle. A review of the Wyle test records for the valve indicated the final lift pressure for the valve prior to return to AP&L was approximately 2480 psia. After heatup to Hot Standby conditions following 2R6, the in situ test performed by AP&L maintenance determined that the lift setpoint for 2PSV-4633 was 2455 psia. The cause of the difference in the setpoint determined by Wyle and the setpoint determined by the in situ test performed by AP&L could not be determined but may be related to the different test methods used by Wyle and AP&L or other factors such as differences in actual temperatures of valve components when the valve is tested in situ versus simulating valve temperatures by heating the valve when tested by Wyle.

Following adjustment of 2PSV-4633 and subsequent testing to verify the lift setpoint was within the Technical Specification limits, in situ testing of 2PSV-4634 was performed on 4/30/88. The lift setpoint was found to be 2580 psia. An evaluation of this discrepancy revealed that 2PSV-4634 was previously adjusted by AP&L maintenance personnel to comply with Technical Specification requirements on 9/26/86. No further testing or maintenance had been performed on the valve until the findings on 4/30/88. The reason for the change in lift setpoint for 2PSV-4634 could not be conclusively determined. However, the setpoint difference could be attributed to setpoint drift which occurred during Cycle 6 operation or during the plant cooldown and heatup related to Refueling Outage 2R6.

Minor setpoint drift such as observed on 2PSV-4634 was identified in IE Information Notice 86-92 and was considered normal for 18 months of plant operation. Also, IE Information Notices 86-05 and 86-56 identified similar occurrences discovered during testing of main steam safety valves at some power plants.

Another contributing cause of the setpoint discrepancies could be related to the method used to determine the initial valve lift setpoint versus the final lift setpoint following any adjustment to the valve during in situ testing. Only one lift test of the valves is performed to determine the initial lift setpoint; whereas, after valve adjustments are made, a minimum of two lift tests are required to verify the final setpoint.

B. Safety Significance

The pressurizer safety valves function 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.

In order to assess the safety significance of the high safety valve lift setpoint, a sensitivity study of the effects of pressurizer safety valve lift settings on the results of the feedwater line break accident analysis was performed for AP&L by Combustion Engineering, the Nuclear Steam Supply System vendor for ANO-2. Although not a detailed quantitative analysis, this sensitivity study indicated that the out of tolerance lift setting of 2580 psia for 2PSV-4634 would not have resulted in pressure exceeding the RCS safety limit of 2750 psia in the event of a postulated feedwater line break accident. Additionally, the feedwater line break accident analysis utilizes several assumption inputs which cause the analysis results to be appropriately conservative in nature. For example, the analysis assumes a higher initial reactor thermal power level, higher initial RCS cold leg temperatures, a lower initial RCS flow rate and a higher initial RCS pressure than allowed by the current Technical Specifications. Also, the RCS moderator temperature coefficient (MTC) is assumed to be zero in the analysis whereas the actual MTC for operating Cycle 6 was negative from the beginning of the cycle to the end of the cycle. Therefore, based on the significant conservatisms used in the feedwater line break accident analysis and the results of the analysis sensitivity studies performed, the safety significance of the high lift setting for 2PSV-4634 is considered to be minimal.

The low lift setting of 2455 psia found on 2PSV-4633 was conservative with respect to overpressure protection. Also, the setting was not significantly low enough such that premature lifting of the valve following a transient would likely occur. Furthermore, this valve was initially placed in service during 2R6, and the incorrect setting was discovered and corrected prior to operational modes which would produce challenges to the valves. Consequently there was no safety significance of the low setting for 2PSV-4633.

C. Root Cause

The root cause(s) of the low setpoint for 2PSV-4633 or the high setpoint for 2PSV-4634 could not be conclusively determined.

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Arkansas Nuclear One, Unit Two	0|5|0|0|0| 3| 6| 8	8| 8	--	0| 1| 2	--	0| 0	0|4 OF 0|5

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

D. Reportability

TS 3.4.3.b allows for operation in Hot Standby (Mode 3) for the purpose of setting the pressurizer code safety valves under ambient (hot) conditions provided a preliminary cold setting of the valve was made prior to heatup. The test performed at Wyle is actually a "hot" setting and is more than adequate to meet the preliminary cold setting requirements. Adjustments of 2PSV-4633 to comply with Technical Specifications were made as allowed, therefore, the requirements of Technical Specifications were met and the out of tolerance lift setpoint found for 2PSV-4633 is not reportable.

Setpoint testing of 2PSV-4634 in order to comply with Technical Specification surveillance requirements was not required until next refueling. However, as a good maintenance practice and because of previous discoveries of safety valve setpoint discrepancies at ANO and other utilities, in situ testing was also performed on 2PSV-4634. The evaluation of the high setpoint did not reveal any conclusive evidence to indicate when the valve setpoint change occurred. Therefore, per the guidance in NUREG 1022, the event date was considered the date of discovery. Consequently, no reporting criteria for this occurrence was identified. However, these findings are being reported voluntarily for information due to possible generic concerns related to safety valve setpoint discrepancies.

III. Corrective Actions

A. Immediate

Since the existing plant conditions at the time of discovery of the safety valve setpoint discrepancies were those required for valve testing and adjustments, no immediate actions were necessary.

B. Subsequent

Adjustments were made to 2PSV-4633 to return the lift setpoint to a value within the allowable Technical Specification tolerance. The as-left lift setpoint for 2PSV-4633 was determined to be approximately 2510 psia. The adjustments and followup testing were completed on 4/30/88.

Adjustments were made to 2PSV-4634 to return the setpoint to a value within the allowable Technical Specification values. The as-left lift setpoint for 2PSV-4634 was determined to be approximately 2515 psia. The adjustments and followup testing were completed on 4/30/88.

Subsequent evaluations were performed regarding the setpoint discrepancies; however, no definitive evidence was revealed as to when the setpoint discrepancies occurred or the exact root cause of either setpoint discrepancy.

C. Future

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.

Additionally, 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. Previous Events

Similar events regarding pressurizer safety relief valve setpoint discrepancies were previously identified in LER 50-368/86-012 and LER 50-313/86-007.

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Arkansas Nuclear One, Unit Two	0|5|0|0|0| 3| 6| 8	8| 8	0| 1| 2	0| 0	0|5 OF 0|5

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

B. Additional Information

The pressurizer safety valve installed on the 2PSV-4633 nozzle during Cycle 6 operation had exhibited a small amount of seat leakage during the cycle; therefore, this valve was removed during 2R6 and sent to Wyle for refurbishment. Upon receipt at Wyle, the valve was tested prior to refurbishment and the lift setpoint was determined to be approximately 2555 psia. 2PSV-4633 had been previously adjusted to within required tolerances during in situ testing on 9/26/86 and no further testing or maintenance was performed on the valve prior to its removal and shipping to Wyle during 2R6.

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

No supplemental report is planned.

ARKANSAS POWER & LIGHT COMPANY

July 21, 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 Report 50-368/88-012-00

Gentlemen:

Attached is the subject report concerning setpoint discrepancies for pressurizer safety valves discovered during in situ testing following heatup after a refueling outage. This report is being voluntarily submitted for information to identify possible generic concerns.

Very truly yours,

J M Levine

J. M. Levine
Executive Director,
Nuclear Operations

JML:PCR:dm
attachment

cc w/att: Regional Administrator
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