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)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	1 OF 0 6

TITLE (4) Engineered Safety Features Actuations - Procedural Deficiency Causes Two Main Feedwater Isolations And An Auxiliary Feedwater Actuation

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
1 2	2 6	8 7	8 7	0 5 1	0 0	0 1	2 2	8 8		0 5 0 0 0
										0 5 0 0 0

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

OPERATING MODE (9): 4
POWER LEVEL (10): 0

[ ] 20.402(b)	[ ] 20.405(c)	[X] 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)	[ ] 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	
Merlin G. Williams - Superintendent of Regulatory, Quality and Administrative Services	3 1 6	3 6 4 - 8 8 3 1

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	S B	I S V	A 3 9 1	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 fifteen single-space typewritten lines) (16)

On December 26, 1987, three Engineered Safety Features actuations occurred. At approximately 0510 CST, a Main Feedwater Isolation Signal (FWIS) occurred when Steam Generator 'A' (S/G) level swelled above the hi-hi setpoint. At approximately 0517 CST, a Reactor trip signal and Auxiliary Feedwater Actuation Signal (AFAS) occurred when S/G 'A' level shrank below the lo-lo level setpoint. At approximately 0830 CST, a second FWIS occurred when S/G 'A' level again swelled above the hi-hi level setpoint.

The root cause of the first FWIS (0510 CST) and the AFAS (0517 CST) were procedural deficiency. The procedure for testing the MSIVs has been enhanced to reduce the effects of swell by requiring a low pressure differential across the MSIV and less than 50 percent S/G level prior to opening the MSIV.

The root cause of the second FWIS (0830 CST) is a combination of personnel error and procedural deficiency. The licensed operator controlling the S/G level allowed the S/G level to reach approximately 66 percent prior to opening the MSIV. The procedure did not address a level requirement prior to this event. The procedure enhancements should prevent both of these events from reoccurring.

8802010215 880122
PDR ADOCK 05000482
S PDR

☆ U.S.GPO:1984-0-454-481/18750

INTRODUCTION

On December 26, 1987, three Engineered Safety Features [JE](ESF) actuations occurred. At approximately Ø51Ø CST, a Main Feedwater [SJ] Isolation Signal (FWIS) occurred when Steam Generator 'A' [AB-SG](S/G) level swelled above the hi-hi level setpoint. At approximately Ø517 CST, a Reactor [AB-RCT] trip signal and Auxiliary Feedwater [BA] Actuation Signal (AFAS) occurred when S/G 'A' level shrank below the lo-lo level setpoint. At approximately Ø83Ø CST, a second FWIS occurred when S/G 'A' again swelled above the hi-hi level setpoint. All of these events are directly related to procedural inadequacies associated with testing of the Main Steamline Isolation Valves [SB-ISV](MSIV). These events are being reported pursuant 1ØCFR 5Ø.73(a)(2)(iv) as unplanned actuations of ESF equipment. The plant was in Mode 4, Hot Shutdown, preparing to enter Mode 3, Hot Standby, following the Refuel II Outage, at the time of these events.

DESCRIPTION OF EVENTS

On December 26, 1987, preparations were being made for entry into Mode 3. Steam flow to the Main Condenser [SG-COND] via the steam dumps had been established to control the Reactor Coolant System [AB] temperature. The four MSIVs were closed with the bypass valves [SB-V] open in preparation for the fast closure retest of the 'A' MSIV after an initial failure. Even though the bypass valves were open, a high differential pressure may still exist across the MSIV due to the difference in size between the Main Steamline and the bypass valve line. When the 'A' MSIV was opened for the test at approximately Ø517 CST, the resulting increased steam demand caused a pressure decrease in S/G 'A' which caused rapid swell in the S/G. The FWIS occurred when S/G 'A' indicated level reached the hi-hi level setpoint. This also initiated a Main Turbine [TA-TRB] trip signal.

The FWIS was reset and Control Room [NA] operators reduced steam dump demand to decrease the level in S/G 'A'. This decreased steam demand caused a pressure increase in S/G 'A' which caused the level in S/G 'A' to decrease rapidly and the S/G to shrink. This caused the indicated level to decrease below the lo-lo level setpoint at approximately Ø517 CST causing an AFAS and Reactor trip signal. The AFAS was reset. The motor-driven Auxiliary Feedwater Pump [BA-P] was placed in standby at approximately Ø52Ø CST after the S/G levels were stabilized.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	8 7	— 0 5 1 —	0 0	0 3 OF 0 6

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

The steam dumps were then isolated and the atmospheric Power Operated Relief Valves [AB-RV](PORVs) were used for RCS temperature control in an attempt to reduce the effects of S/G swell when the MSIV was opened for testing. The swell effects may be minimized because the PORVs are upstream of the MSIVs while the steam dumps are downstream. The 'A' MSIV was opened for the fast close retest at 0805 CST. The retest was declared unsatisfactory at 0825 CST when an open/close dual indication occurred on the Main Control Board [IC-MCBD]. A work request was written and maintenance was called to investigate the dual indication. The steam dumps were unisolated at approximately 0828 CST so that the atmospheric PORVs could be closed. The FWIS occurred at 0830 CST due to S/G 'A' hi-hi level. It was identified that the 'A' MSIV had not completely closed. This caused excess steam demand when the steam dumps were opened and caused S/G 'A' indicated level to swell. After discovering that the 'A' MSIV was still partially open, the Control Room operators did a slow close of 'A' MSIV and verified that it full closed. This quickly reduced S/G 'A' indicated level below the hi-hi level setpoint. The unit was stabilized and the FWIS was reset at approximately 0835 CST prior to entry into Mode 3 at 0852 CST.

ROOT CAUSE AND CORRECTIVE ACTIONS

The root cause of the first (0510 CST) FWIS and the AFAS (0517 CST) were procedural deficiency. The procedure for testing the MSIV has been enhanced to reduce the effects of swell by requiring minimum differential pressure across the MSIV and less than 50 percent S/G level prior to opening the MSIV. During this event, the S/G level swelled from approximately 46 percent to above the hi-hi setpoint of 78 percent in a matter of seconds. After the MSIV was opened, operator action could not have prevented the FWIS. The procedures required less than or equal to 25 pounds per square inch across the MSIVs prior to opening. By adding another step to require operators to have a minimum differential pressure across the MSIV prior to opening, future occurrences may be avoided.

The root cause of the second FWIS (0830 CST) is a combination of personnel error and procedural deficiency. The licensed operator controlling the S/G level allowed the S/G level to reach approximately 66 percent prior to opening the MSIV. The procedures did not address a minimum level prior to this event. The procedure enhancements should prevent these events from reoccurring.

ADDITIONAL INFORMATION

There was no damage to plant equipment or release of radioactivity as a result of this event. All safety systems performed as designed and therefore at no time did conditions develop that could have posed a threat to the health or safety of the public.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	8 7	— 0 5 1	— 0 0	0 4 OF 0 6

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

Table 1 shows previous FWIS along with the operation underway and the power level during each event. These events are grouped into five categories as described in Table 1. The corrective actions taken appear to be decreasing the yearly number of events. Licensee Event Report (LER) 86-024 is considered similar to this because of the MSIV testing in progress. The corrective action taken in LER 86-024 is now seen as only being a short term corrective action in this case. It is believed that procedure enhancement is a long term corrective action to prevent future occurrences.

The cause of the dual indication on the MSIV was determined to be an O-ring blown out on the hydraulic accumulator [SB-ACC]. The O-ring was replaced and the MSIV then retested satisfactorily. The MSIVs are manufactured by Anchor-Darling Valve Company. The hydraulic accumulator is also manufactured by Anchor-Darling Valve Company, Part Number W-30500. The cause of the failed O-ring could not be determined.

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

TABLE 1

FEEDWATER ISOLATIONS

LER	EVENT (POWER LEVEL)	CATEGORY
85-020-00	Transfer from auxiliary feedwater pump to startup feedwater pump (0%)	5
85-031-00	Failure to stabilize level prior to performing transient (12%, Reactor Trip)	3
85-039-00	Steam dump testing (6%, Reactor Trip)	4
85-042-00	Transfer from bypass to main feedwater control valves (16%, Reactor Trip)	1
85-043-00	Leakage past main feedwater control valves (2%)	2
85-044-00	Loaded main gener .tor greater than reactor power could supply (8%)	5
85-046-00	Reverse flow through steam generator check-valves. Steam generator pressure greater than feedwater header pressure (0%)	5
85-050-00	Leakage past main feedwater control valves (4%)	2
85-064-00	Unpreheated feedwater added to steam generator (5%, Reactor Trip)	5
85-072-00	Failure to stabilize level prior to performing transient (27%, Reactor Trip)	3
*86-024-00	MSIV testing (0%)	4
86-030-00	Leakage past main feedwater control valves (1%)	2
86-037-00	Transfer from bypass to main feedwater control valves (20%, Reactor Trip)	1
86-042-00	Transfer from bypass to main feedwater control valves (13%, Reactor Trip)	1
86-068-00	Transfer from bypass to main feedwater control valves (6%, Reactor Trip)	1

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Wolf Creek Generating Station	0 5 0 0 0 4 8 2	8 7	— 0 5 1 —	0 0	0 6 OF 0 6

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

TABLE 1 (continued)

LER	EVENT (POWER LEVEL)	CATEGORY
87-037-00	Transfer from bypass to main feedwater control valves and transfer from startup to main feed-water pump (2 events)(7% and 4)	5 and 1
87-042-00	Operator not attentive to steam generator level (0%)	5

Category:

1. Transfer from bypass to main feedwater regulator valves
2. Leakage past main feedwater regulator valves
3. Failure to stabilize S/G level before imposing further transients
4. Testing of MSIVs or steam dumps
5. Other

*Previous Similar Occurrence

WOLF CREEK
NUCLEAR OPERATING CORPORATION

Bart D. Withers
President and
Chief Executive Officer

January 22, 1988

WM 88-0027

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

Subject: Docket No. 50-482: Licensee Event Report 87-051-00

Gentlemen:

The attached Licensee Event Report (LER) is submitted pursuant to 10 CFR 50.73 (a) (2) (iv) concerning an Engineered Safety Features actuation.

Very truly yours,

BDWithers

Bart D. Withers
President and
Chief Executive Officer

BDW/llk

Attachment

cc: J. E. Cummins (NRC), w/a
R. D. Martin (NRC), w/a
P. W. O'Connor (NRC), 2 w/a

IE22
1/1