WOLF CREEK
NUCLEAR OPERATING CORPORATION

John A. Bailey
Vice President
Nuclear Operations

April 9, 1990
NO 90-0100

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-137
Washington, D. C. 20555

Subject: Docket No. 50-482: Licensee Event Report 85-058-01

Gentlemen:

The attached Licensee Event Report (LER) is submitted pursuant to 10 CFR 50.73(a)(2)(iv) concerning a Reactor trip and Engineered Safety Feature actuation as a result of an electrical spike on a power range Nuclear Instrumentation channel. This report is a revision to LER 85-058-00 and revises the description of corrective actions taken regarding the repositioning of Auxiliary Feedwater valves.

Very truly yours,

John A. Bailey

John A. Bailey
Vice President
Nuclear Operations

JAB/jra

Attachment

cc: R. D. Martin (NRC), w/a
D. Persinko (NRC), w/a
D. V. Pickett (NRC), w/a
M. E. Skow (NRC), w/a

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NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/88

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 4

TITLE (4) Unit Trip Caused By A Spike On One Nuclear Instrumentation Channel While Another Was In Test

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 7	3 1	8 5	8 5	— 0 5 8 —	0 1					0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 8 6

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)	[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 - Manager Plant Support	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
B	I G	R J X	P 3 2 3	N					

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 fifteen single-space typewritten lines) (16)

On July 31, 1985, at approximately 0348 CDT, a Reactor trip, Main Turbine trip, Auxiliary Feedwater Actuation, Feedwater Isolation, and a Steam Generator Blowdown and Sample Isolation occurred due to an electrical spike on one channel of the power range nuclear instrumentation, while another channel was out of service for surveillance testing. All required Reactor Protection System and Engineered Safety Features equipment responded properly. Prior to this occurrence, the plant was in Mode 1, Power Operation, at a Reactor power level of approximately eighty-six percent.

The spike on the nuclear instrumentation channel was the result of a faulty power supply which has been replaced. The faulty power supply has been returned to the vendor for inspection and evaluation.

There was no damage to plant equipment or release of radioactivity as a result of this event, and at no time did conditions develop that may have posed a threat to the health or safety of the public.

Revision 1 to this report is being submitted to revise the description of corrective actions taken regarding repositioning of certain Auxiliary Feedwater valves.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

On July 31, 1985, at approximately 0348 CDT, a Reactor trip and Engineered Safety Features actuation occurred as a result of an electrical spike on a power range Nuclear Instrumentation channel (NI)[IG] while another channel was out of service.

Prior to this event, the plant was in Mode 1, Power Operation, at a reactor power level of approximately eighty-six percent. At approximately 0340 CDT, power range NI channel 43 had been taken out of service for surveillance testing. With one channel out of service, the trip logic is satisfied by a trip signal from one of the other three in-service channels.

At 0348 CDT, power range channel 42 spiked low, satisfying the negative rate trip logic, and resulted in a Reactor trip and Main Turbine trip. The Reactor trip, coupled with a low Reactor Coolant System [AB] average temperature, initiated a Feedwater Isolation. When Steam Generator "C" [AB-SG] water level reached the low-low level setpoint, a motor-driven Auxiliary Feedwater Actuation and Steam Generator Blowdown and Sample Isolation occurred. Shortly thereafter, a turbine-driven Auxiliary Feedwater Actuation occurred when Steam Generator "B" water level also reached the low-low level actuation setpoint.

All required Engineered Safety Features and Reactor Protection System equipment performed their intended functions. One equipment problem was noted - there was no indication on the Main Control Board of Auxiliary Feedwater [BA] flow to Steam Generator "C". However, the expected Auxiliary Feedwater flow to Steam Generator "C" was indicated at the Auxiliary Shutdown Panel.

During this event, pressurizer [AB-PZR] level decreased to approximately twenty percent, and Reactor Coolant System average temperature reached a minimum of 550 degrees Fahrenheit. Water levels in all four Steam Generators reached the low-low level setpoint. The power operated relief valve [SB-RV] on Steam Generator "C" opened for approximately three minutes during this transient. Normal feedwater flow was restored at approximately 0431 CDT.

During the restoration of normal feedwater flow to the steam generators, valve AL-HV-07, Auxiliary Feedwater flow control to Steam Generator "A" [BA-FCV], was throttled closed and could not be reopened from the Control Room. This valve had to be partially opened manually before normal control could be established. The probable cause of this occurrence was a limit switch setting misadjustment which has been corrected. Further discussion of this situation is provided in Licensee Event Report 85-054-00.

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

Two other valves, AL-HV-8 and AL-HV-10, turbine driven Auxiliary Feedwater flow discharge valves to Steam Generators "B" and "C", could not be reopened from the Control Room once they were closed and the turbine driven Auxiliary Feedwater Pump had been secured. These valves are normally in the full open position, and are throttled closed during restoration of normal feedwater flow. In this instance, the valves had to be partially opened manually prior to restoration of normal control from the Control Room. An investigation identified a potential reverse differential pressure condition which could restrict valve opening if the turbine-driven Auxiliary Feedwater Pump [BA-P] was secured while the valves are in the fully closed position. The investigation also confirmed that the valves would be capable of remote opening if the turbine-driven Auxiliary Feedwater Pump was energized because the potential reverse differential pressure condition could not exist and since the valves are flow-assisted to open. As an administrative control to prevent this situation, the operating procedure governing securing of the turbine driven Auxiliary Feedwater Pump was revised to ensure the flow discharge valves are partially open prior to securing the Pump.

Subsequently, in 1987, the actuator springs in the valves were replaced with more powerful springs to allow the actuators to open the valves under greater reverse differential pressure conditions. With the installation of this modification, the administrative controls described above are no longer necessary and are being deleted from the applicable procedures.

Subsequent investigations into the problems with power range NI channel 42 revealed the existence of a faulty power supply [IG-RJX]. The power supply was replaced, and NI channel 42 was tested demonstrating operability and returned to service.

The faulty power supply was Model UPMD-X54W, manufactured by Power Designs, Inc. Troubleshooting identified that the power supply had experienced internal arcing, resulting in a zero output signal. The rate circuitry sensed the output changes and initiated a high-rate trip. The power supply was disassembled and tested, but no further arcing was observed. It has been returned to the vendor for inspection and evaluation.

Subsequent investigations also revealed that the lack of indication of Auxiliary Feedwater flow to Steam Generator "C" was due to a failed flow indicator which has been replaced. There was adequate flow to Steam Generator "C" as evidenced by the similarity between the level traces of all four Steam Generators, computer point indications, and Auxiliary Shutdown Panel indication.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

ESTIMATED BURDEN PER RESPONSE TO COMPLY WTH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

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

There was no damage to plant equipment or release of radioactivity as a result of this event, and at no time did conditions develop that may have posed a threat to the health and safety of the public.

There have been no previous similar occurrences.