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 3

TITLE (4)

Reactor Trip and ESF 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)
0 7	2 3	8 5	8 5	— 0 5 4 —	0 0	0 8	2 2	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 7 5

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 - 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
B	E E	X F M R	S 2 5 0	N					

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 July 23, 1985, at approximately 0808 CDT, a Reactor Trip, Main Turbine trip, Auxiliary Feedwater actuation, Feedwater Isolation and Steam Generator Blowdown and Sample Isolation occurred as a result of a low-low water level condition in all four steam generators. Plant response throughout the event was normal and all required Reactor Protection System and Engineered Safety Features equipment responded properly.

Prior to this event the plant was operating normally on automatic controls in Mode 1, Power Operation, at seventy-five percent reactor power.

The root cause of this event was failure of a wire supplying power to a 120 volt instrument AC distribution panel. This resulted in a loss of governor control power to Main Feedwater Pump "B", allowing the pump to slow down and decrease feedwater flow to the Steam Generators. The wire failure has been attributed to a faulty crimped connection. This connection has been repaired and an inspection for other faulty connections did not reveal any additional problems.

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

IE22
1/1

U.S.GPO:1984-0-454-481/18759

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 5	— 0 5 4	— 0 0	0 2 OF 0 3

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

On July 23, 1985, at approximately 0808 CDT, a Steam Generator [AB-SG] low-low water level condition initiated a Reactor Trip, Main Turbine trip, Auxiliary Feedwater actuation, Feedwater Isolation and Steam Generator Blowdown and Sample Isolation.

Prior to the event, the plant was operating normally on automatic controls in Mode 1, Power Operation, at seventy-five (75) percent reactor power. Steam Generator water levels were being controlled at approximately 50 percent narrow range level (equivalent to 61 percent wide range level).

At approximately 0806 CDT, annunciation of an undervoltage condition on 120 volt instrument AC distribution panel PN08[EE-PL] was received. Shortly thereafter, water levels in all four steam generators began decreasing, and it was noted that Main Feedwater Pump (MFP)[SJ-P] "B" had slowed from approximately 4400 revolutions per minute (RPM) to 800 RPM. Efforts to restore "B" MFP to normal speed were unsuccessful. Both motor driven Auxiliary Feedwater Pumps (AFP)[BA-P] were manually started and the main generator [TB-TG] load was rapidly reduced to lower steam flow to the level of available feedwater flow. However, Steam Generator water levels continued to decrease and S/G "C" reached the low-low setpoint at approximately 0808 CDT initiating the Reactor trip and Engineered Safety Features actuation.

The motor driven Auxiliary Feedwater Pumps were already in operation and all other required Reactor Protection System and Engineered Safety Features equipment responded properly. The turbine driven Auxiliary Feedwater Pump [BA-P] started automatically approximately one second later when Steam Generators "A" "B" and "D" reached the low-low water level setpoint.

Plant response to the Reactor trip transient was normal. Steam Generator water levels reached a minimum of 45 percent wide range level during the event and were recovered to normal at approximately 0915 CDT. The actuated systems were restored to normal configurations per plant procedures and normal feedwater flow was re-established by approximately 0935 CDT.

The cause of this event was the failure of a stranded wire on the secondary side of a power supply transformer feeding 120 volt instrument AC distribution panel PN08. The wire overheated and partially separated adjacent to a crimped terminal lug within the vendor supplied wiring in the transformer housing. This caused an undervoltage condition in phase "B" of distribution panel PN08 and resulted in the loss of governor control power (supplied by phase "B") to Main Feedwater Pump "B". Loss of governor control power allowed the Main Feedwater Pump to slow down, decreasing feedwater flow to the steam generators.

The probable cause of the overheating was a faulty crimp connection between the terminal lug and the wire. The connection was repaired by installing a new terminal lug and reconnecting the wire. Inspections of this and other similar transformers for faulty connections or evidence of overheating did not reveal any additional problems.

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 5	— 0 5 4 —	0 0	0 3 OF 0 3

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

The transformer and wiring were supplied by Solidstate Controls Inc. and the transformer is a model RT-481260N. There have been no previous Licensee Event Reports associated with these transformers.

There was no damage to plant equipment or release of radioactivity as a result of this event. At no time did conditions develop which could have posed a threat to the health or safety of the public.
During restoration from this event, as Steam Generator water levels were increasing toward normal, valve AL-HV-07, auxiliary feedwater flow control to Steam Generator "A" [BA-FCV], was throttled to the closed position and could not be reopened from the Control Room. The valve was manually opened slightly, restoring normal control to the Control Room. On July 31, 1985, during restoration from another Reactor trip (to be described in Licensee Event Report 85-058-00), this same problem occurred again.

Subsequent to each occurrence, investigation identified the probable cause as "open" limit switch misadjustment which prevented the valve from opening after it had torqued closed. Positive identification of the cause was not possible since in each case the valve had been manually operated and manual operation of the valve without first de-energizing the motor operator changes the limit switch and control settings. Following each occurrence the limit switch and control settings were adjusted and the valve was demonstrated to operate correctly through its entire stroke. Following adjustment after the second occurrence, the valve was also tested under normal flow conditions and operated correctly through its entire stroke. This test will be repeated at the next opportunity to confirm that limit switch settings have not changed.

Valve AL-HV-07 is normally open and is throttled to control auxiliary feedwater flow to the Steam Generator. The valve is closed only during restoration from an Auxiliary Feedwater Actuation as Steam Generator levels are approaching normal. The valve was supplied by Masoneilan International Inc. and is a Model 90-207X1 Globe valve with a Limitorque SMB00-5 operator using modulating position controls.

KG&E
THE ELECTRIC COMPANY
KANSAS GAS AND ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

August 22, 1985

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

Mr. R.P. Denise, Director
Division of Reactor Safety and Projects
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

KMLNRC 85-202
Re: Docket No. STN 50-482
Subj: Licensee Event Report 85-054-00

Gentlemen:

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

If you have any questions concerning this matter, please contact me or Mr. Otto Maynard of my staff.

Yours very truly,

CCMason
for Glenn L. Koester
Vice President - Nuclear

GLK:see

Enclosure

xc: PO'Connor (2), w/a
JCummins, w/a

IE22
1/1