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): ESF Actuation - Reactor Trip, Auxiliary Feed Actuation, Feedwater Isolation

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	0 9	8 5	8 5	— 0 4 9 —	0 1	0 9	0 4	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 4 7

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

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 9, 1985, at approximately 1115 CDT, a Reactor Trip, Main Turbine Trip, Feedwater Isolation, Auxiliary Feedwater Actuation, and Steam Generator Blowdown and Sample Isolation occurred due to low-low water level in Steam Generator "C". A Turbine-driven Auxiliary Feedwater Actuation occurred when water level in Steam Generator "A" also reached the low-low level setpoint.

Prior to this event, the plant was in Mode 1, Power Operation, at a Reactor power level of approximately forty-seven percent. During Power Ascension Testing, feedwater flow to the "C" Steam Generator began decreasing rapidly, resulting in the steam generator water level reaching the low-low level setpoint for Engineered Safety Features Actuation.

Subsequent investigations determined that the loss of feedwater flow was the result of installation and removal of test equipment in the feedwater control circuit cabinets. All required Engineered Safety Features and Reactor Protection System equipment performed their intended safety function, and the actuated systems were restored to normal at approximately 1230 CDT.

To prevent future occurrences of this type, personnel have been cautioned about the effects of installing and removing testing equipment.

There was no damage to plant equipment or release or 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.

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☆U.S.GPO:1984-0-454-481/18759

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

On July 9, 1985, at approximately 1115 CDT, an Engineered Safety Features Actuation and a Reactor Trip occurred due to low-low water level in Steam Generator (S/G)[AB-SG]"C".

Prior to this event, the plant was in Mode 1, Power Operation. Power Ascension Testing was in progress in accordance with SU7-008.2, "Power Coefficient Determination", which involves ramping the Main Turbine [TA-TRB] load down forty megawatts, allowing conditions to stabilize, and then ramping the Main Turbine load back up forty megawatts at five percent per minute. The ramp down was completed successfully, and the plant was stable at forty-seven percent reactor power, with the S/G's being maintained at a level of approximately fifty percent by the Main Feedwater Control Valves [SJ-FCV], which were in automatic control.

As the ramp back up was attempted, feedwater flow fluctuations occurred on all four Steam Generators. The operators took manual control of the Main Feedwater Control Valves for steam generators "A", "B", and "C", but water level in S/G "C" continued to decrease and reached the low-low level setpoint for Engineered Safety Features Actuation. This resulted in a Reactor Trip, Main Turbine Trip, motor-driven Auxiliary Feedwater Actuation, Steam Generator Blowdown and Sample Isolation and a Feedwater Isolation. Shortly thereafter, S/G "A" water level reached the low-low level setpoint, initiating a Turbine-driven Auxiliary Feedwater Actuation.

All required Engineered Safety Features and Reactor Protection System equipment responded properly with the exception of the Intermediate Range Nuclear Instrumentation [IG] channel SE-NI-35B, which would not allow Permissive P-6 to energize the Source Range Nuclear Instruments following the trip. The Source Range Nuclear Instruments had to be manually energized.

During this event, pressurizer [AB-PZR] level decreased to approximately sixteen percent, and the Reactor Coolant System [AB] average temperature reached a minimum of 538 degrees Farenheit. Levels in all four Steam Generators reached the low-low level setpoint during this event. The actuated systems were restored to normal configuration per plant procedures, and normal feedwater flow was restored at approximately 1230 CDT.

An investigation into this event determined that the initial feedwater flow fluctuations in all S/G's were the result of a personnel activity in installing and removing test recorders in the feedwater control circuits. The test leads being utilized had also created a "short", resulting in a loss of program level signal to the controller for Main Feedwater Control Valve "C" which caused a loss of feedwater to that S/G. This scenario has been reenacted with an identical loss of program level signals, although no ESF actuations occurred due to plant conditions at the time. Circuit cards which were exposed to the test signals have been replaced.

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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

The problems experienced with the intermediate range nuclear instrumentation channel SE-NI-35B were investigated by Instrumentation and Control (I&C) personnel and found to be the result of dirty connectors. These connectors were cleaned, and SE-NI-35B has been returned to operable status.

This event is considered to be a unique situation, caused by the installation and removal of test equipment. The importance of being aware of the consequences of installing and removing test equipment will be stressed to I&C personnel, and further emphasis will be placed on this event by incorporation of this report into I&C Required Reading.

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.

KANSAS GAS AND ELECTRIC COMPANY
THE ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

September 4, 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 85-213
Re: Docket No. STN 50-482
Subj: Licensee Event Report 85-049-01

Gentlemen:

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

Yours very truly,

Glenn L Koester

Glenn L. Koester
Vice President - Nuclear

GLK:see

Enclosure

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

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