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 and Reactor Trip on Low Steam Generator Level

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 6	0 6	8 5	8 5	0 3 9	0 0	0 7	0 5	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 0 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
Merlin G. Williams - Superintendant of Regulatory, Quality and Administrative Services	AREA CODE 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)

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)

At 2253 CDT on June 6, 1985, a Reactor Trip, Auxiliary Feedwater Actuation Signal and Steam Generator Blowdown and Sample Isolation Signal were initiated due to low-low water level in Steam Generator (S/G) "D". The Reactor Trip coupled with a low Reactor Coolant System average temperature initiated a Feedwater Isolation Signal. All required Engineered Safety Features and Reactor Protection System equipment responded properly. The low-low S/G water level occurred as a result of S/G level oscillations while performing testing of the steam dump control system.

The plant was in Mode 1 at a Reactor power level of approximately 6 percent at the time of this event. The Reactor Coolant System was at normal operating pressure and approximately 560 degrees F.

There was no damage to plant equipment or release of radioactivity as a result of this event. All actuated safety systems performed as required thus preventing any adverse condition which may have posed a threat to the public health or safety.

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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 3 9 —	0 0	0 2 OF 0 3

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

At 2253 CDT on June 6, 1985, a Reactor Trip, Auxiliary Feedwater Actuation Signal (AFAS) and Steam Generator Blowdown and Sample Isolation Signal (SGBSIS) were initiated due to low-low level in Steam Generator (S/G) "D". The Reactor Trip coupled with a low Reactor Coolant System average temperature (Tavg) initiated a Feedwater Isolation Signal (FWIS).

The plant was initially in Mode 2 with startup testing in progress per SU7-AB02, "Dynamic Automatic Steam Dump Control". Main Feedwater Pump "A" in manual control was supplying feedwater to the Steam Generators via the Feedwater Control Valve (FCV) Bypass valves in automatic control. In accordance with the test procedure, steam dump control had been shifted from steam pressure control mode to the Tavg control mode to reduce Tavg 5 degrees to approximately 557 degrees F. After an initial increase in S/G water levels during this shift - due to "swell", S/G levels stabilized and testing continued. At 2222 CDT the plant entered Mode 1 (greater than 5 percent Reactor power) as reactor power level was increased to approximately 6 percent in support of SU7-AB02. During this power escalation S/G water levels began to swing as the FCV Bypass valve controls attempted to maintain the levels in response to changes in steam flow and reactor power. The swings in S/G water levels became increasingly larger, and attempts to stabilize them by taking manual control of the FCV Bypass valves were not successful.

As reactor power reached approximately 6 percent, the S/G water levels began to decrease rapidly. With the FCV Bypass valves near full open and S/G water levels continuing to decrease, the motor driven auxiliary feedwater pumps (AFWP) were manually started to provide additional feedwater flow. Shortly thereafter, at 2253 CDT, the water level in S/G "D" reached the low-low level trip point and a Reactor Trip, AFAS and SGBSIS were automatically initiated. Since the Reactor Coolant System Tavg was low (560 degrees F), a FWIS was also initiated. The motor driven Auxiliary Feedwater Pumps were already in operation and all other required Engineered Safety Features (ESF) and Reactor Protection System equipment responded properly.

At 2300 CDT S/G "C" water level also reached the low-low level trip point. Since 2-of-4 S/G's had now exceeded the low-low level setpoint, a Turbine-driven AFWP AFAS was initiated. Again, all ESF equipment responded properly.

AT 2306 CDT the Main Steam Line Isolation Valves (MSLIV's) were manually fast-closed to stop steam flow from the S/G's as a contributor to Reactor Coolant System cooldown. Reactor Coolant System Tavg reached a minimum temperature of approximately 538 degrees F and then was returned to normal operating temperature.

Steam Generator water levels were returned to normal by approximately 0023 CDT on June 7, 1985 and normal feedwater flow was reestablished in accordance with plant operating procedures at 0318 CDT.

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Wolf Creek Generating Station	0 5 0 0 0 4 8 2	8 5	— 0 3 9	— 0 0	0 3 OF 0 3

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

Subsequent review of this event determined that the cause of the S/G water level swings which resulted in the reactor trip was over compensation by the automatic controls for the FCV Bypass valves to changes in S/G water level (shrink and swell) induced by testing of the steam dump valve control system. This occurred because the FCV Bypass valve control system had not been dynamically "tuned" for automatic operation. Dynamic tuning was scheduled to be performed following testing of the Steam Dump Control System.

Prior to resuming startup testing per SU7-AB02, "Dynamic Automatic Steam Dump Control", startup test SU7-AB01.1 "Automatic Steam Generator Level Control" was performed. This procedure was successfully completed on June 11, 1985, thus ensuring optimum adjustment and level control stability of the Steam Generator FCV Bypass valve control system in automatic mode. Steam dump testing was subsequently completed successfully with no further incident.

There was no damage to plant equipment or release of radioactivity as a result of this event. All actuated safety systems performed as required thus preventing any adverse condition which may have posed a threat to the public health or safety.

KANSAS GAS AND ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

July 5, 1985

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

Mr. R.P. Denise, Director
Wolf Creek Task Force
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

KMLNRC 85-173
Re: Docket No. STN 50-482
Subj: Licensee Event Report 85-039-00

Dear 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,

Glenn L Koester

Glenn L. Koester
Vice President - Nuclear

GLK:dab

Enclosure

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

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