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 - Feedwater Isolation, Auxiliary Feedwater Actuation, Steam Generator Blowdown and Sample Isolation.

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

OPERATING MODE (9): 1

POWER LEVEL (10): 0 1 2

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)	☒ 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)	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

At 1752 CDT on June 11, 1985, an Engineered Safety Features Actuation Signal was initiated due to high-high water level in Steam Generator "C". This initiated a Feedwater Isolation Signal, a Main Turbine Trip signal, and Main Feedwater Pump (MFP) Trip signals. The MFP trip caused an Auxiliary Feedwater Actuation and a Steam Generator Blowdown and Sample Isolation to be initiated. The Main Turbine was not in operation at the time of this event, and all required Engineered Safety Features equipment responded properly.

The plant was in Mode 1, Power Operation, at a reactor power level of approximately twelve percent at the time of the event and the Reactor Coolant System was at normal operating pressure and temperature.

There was no damage to plant equipment or release of radioactivity as a result of this event. At no time did this event pose 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)
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TEXT (If more space is required, use additional NRC Form 366A's) (17)

At 1752 CDT on June 11, 1985, Engineered Safety Features Actuation Signals were initiated when water level in Steam Generator (S/G) [AB-SG] "C" increased to the high-high level actuation setpoint.

Prior to this event, the plant was in Mode 1, Power Operation, at a power level of approximately twelve (12) percent and the Reactor Coolant System [AB] was at normal operating pressure and temperature. Dynamic testing of the Steam Dump Control System [JI] had just completed, and steam generator water levels ranged from 69 percent in S/G "C" to 46 percent in S/G "A". Feedwater was being supplied by Main Feedwater Pump [SJ-P] "A" and feedwater flow control was via the Main Feedwater Control Valve Bypass valves [SJ-FCV] in automatic control. At that time, reactor power level was automatically increased from approximately 3 percent to approximately 12 percent via increased steam flow from the Steam Dump Valves [SB-PCV] in preparation to roll the Main Turbine [TA-TRB]. As a result of the increase in steam demand, S/G water levels initially increased due to "swell", then began a downward trend due to the subsequent increased rate of feedwater introduction ("shrink"). As the automatic feedwater controls responded to the increased steam demand and decreasing S/G water levels, steam generator water levels began to rapidly increase.

The initial high water level in S/G "C" (69 percent), coupled with the rapid increase in S/G water levels which occurred during the power escalation due to increased steam demand, resulted in a high-high S/G "C" water level trip. This initiated a Feedwater Isolation Signal (FWIS), a Main Turbine Trip signal, and Main Feedwater Pump (MFP) Trip signals. The MFP trip signals initiated an Auxiliary Feedwater Actuation and a Steam Generator Blowdown and Sample Isolation Signal.

The Main Turbine and Main Feedwater Pump "B" were not in operation at the time of the event. All required Engineered Safety Features equipment responded properly.

Following the FWIS and the trip of the MFP, reactor power level was reduced to approximately 2 percent. Steam Generator "C" water level reached a maximum of 83 percent. Water levels in all four S/G's then began to decrease and the turbine-driven Auxiliary Feedwater Pump [BA-P] was manually started at 1755 CDT to provide additional feedwater flow. The water level in S/G "A", initially at 46 percent, decreased to a minimum of 26 percent, and did not reach the low-low level Reactor Trip setpoint.

Steam generator water levels were returned to normal by approximately 1810 CDT. Reactor Coolant System average temperature reached a minimum of 545 degrees F and was returned to normal operating temperature by 1811 CDT. All actuated plant systems were restored to normal configurations per plant operating procedures by 1910 CDT.

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

Investigation into this event concluded that S/G water levels had not completely stabilized following the completion of dynamic steam dump testing before an additional transient was imposed on the plant. Large differences between S/G water levels and their trends existed prior to the increase in steam demand in preparation to roll the Main Turbine.

While no specific action or error was involved in the initiation of this event, the conditions leading to the event will be presented in an upcoming Training Week for each operating crew stressing the need to minimize the introduction of additional transients while plant systems are in a dynamic state of change, particularly while at low plant power levels. In addition, this Licensee Event Report will be assigned as required reading to operating crew personnel.

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

KGE THE ELECTRIC COMPANY
KANSAS GAS AND ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

July 10, 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 85-178
Re: Docket No. STN 50-482
Subj: Licensee Event Report 85-031-00

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,

CR Mason
for Glenn L. Koester
Vice President - Nuclear

GLK:dab

Enclosure

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

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