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 4	8 5	8 5	— 0 4 4 —	0 0	0 7	1 2	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 0 0 8

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	MANUFAC-TURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	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)

At 0631 CDT on June 14, 1985, an Engineered Safety Features Actuation was initiated due to high-high water level in Steam Generator "B". This initiated a Feedwater Isolation, a Main Turbine Trip, and a Main Feedwater Pump (MFP) Trip. The MFP trip caused an Auxiliary Feedwater Actuation and a Steam Generator Blowdown and Sample Isolation to be initiated. All required Engineered Safety Features equipment responded properly.

The plant was in Mode 1, Power Operation, at a reactor power level of sixteen (16) percent as indicated on the nuclear instrumentation. Based on test program recommendations from the supplier (Westinghouse), the nuclear instrumentation was adjusted conservatively high, and the actual reactor power level at the time of the event was approximately eight (8) percent. This caused a power mismatch during subsequent loading of the Main Turbine Generator which initiated this event.

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.

IE22
1/1

At 0631 CDT on June 14, 1985, an Engineered Safety Features Actuation was initiated when water level in Steam Generator (S/G) [AB-SG] "B" increased to the high-high level actuation setpoint.

Prior to this event, the plant was in Mode 1, Power Operation. The Reactor Coolant System [AB] was at normal operating temperature and pressure, and loading of the Main Generator [TB-GEN] was in progress. Reactor power level as indicated on the nuclear instrumentation [IG] was approximately sixteen (16) percent but, in accordance with supplier (Westinghouse) test program recommendations, these instruments were adjusted conservatively high pending final calibration by heat balance to be accomplished later in the test program. Actual reactor power level was approximately eight (8) percent as determined by reactor core delta temperature measurements. This lower power level caused a power mismatch upon subsequent loading of the Main Generator which resulted in the Engineered Safety Features Actuation.

Feedwater flow to the steam generators was via the Main Feedwater Control Bypass Valves [SJ-FCV] in automatic control. Also, in anticipation that increased feedwater flow would be required during Main Generator loading, additional feedwater flow was introduced to the steam generators via manual control of the Main Feedwater Control Valves [SJ-FCV], leading to oscillations in S/G water levels. As the Main Generator was loaded, insufficient steam flow was available from the steam generators due to the actual reactor power level. This resulted in a significant decrease in steam line pressure and a decrease in Reactor Coolant System average temperature (Tavg), and necessitated increasing reactor power level to maintain Tavg in the required band. Since the control signal for the Main Feedwater Control Bypass Valves is compensated by reactor power level while in the automatic control mode, a resultant increase in feedwater flow via the Main Feedwater Control Bypass Valves occurred.

At 0631 CDT, the increased rate of feedwater flow and the oscillating S/G water levels coupled with "swell" in S/G levels due to the decreased steam line pressure, resulted in a high-high water level trip in S/G "B". This initiated a Feedwater Isolation, a Main Turbine Trip, and Main Feedwater Pump (MFP) trips. The trip of the MFP's caused an Auxiliary Feedwater Actuation and a Steam Generator Blowdown and Sample Isolation to be initiated. All required Engineered Safety Features equipment responded properly.

Following the Feedwater Isolation, reactor power level was reduced to approximately two (2) percent. The maximum steam generator water level reached during this event was 79 percent in S/G "B". All S/G water levels were returned to normal and the actuated plant systems were restored to normal configurations by 0708 CDT.

The cause of this event was the higher than actual indication of reactor power level on the nuclear instrumentation and the use of this indication for plant control. The nuclear instrumentation was subsequently calibrated to indicate actual reactor power level at the 30 percent testing plateau.

NRC Form 366A
(9-83)

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

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 4 4	— 0 0	0 3 OF 0 3

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. At no time did this event pose a threat to the public health or safety.

KANSAS GAS AND ELECTRIC COMPANY

GLENN L. KOESTER
VICE PRESIDENT - NUCLEAR

July 12, 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 85-180
Re: Docket No. STN 50-482
Subj: Licensee Event Report 85-044-00

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,

CJMason
for Glenn L. Koester
Vice President - Nuclear

GLK:dab

Enclosure

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

IE22
1/1