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 5

TITLE (4)
Engineered Safety Features Actuation - Safety Injection and Reactor Trip

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 1	0 8	8 7	8 7	— 0 0 2 —	0 0	0 2	0 9	8 7		0 5 0 0 0
										0 5 0 0 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

OPERATING MODE (9): 1

POWER LEVEL (10): 1 0 0

	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)	X	50.73(a)(2)(vii)	X	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
	20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		T.S. 6.9.2 & 3.5.2b Special Report
	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	A B	P C V	C 7 1 0	N						

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 1818 CST on January 8, 1987, with the unit operating at approximately 100 percent power, a low steamline pressure signal occurred during calibration testing and initiated a Safety Injection signal and Main Steamline Isolation. This resulted in a Reactor trip, Main Feedwater Isolation, Containment Isolation Phase A, Containment Purge Isolation Signal, Control Room Ventilation Isolation, Emergency Diesel Generator start, Auxiliary Feedwater actuation, Essential Service Water actuation, and Steam Generator Blowdown and Sample Isolation. An Unusual Event was declared at 1818 CST and terminated at 1850 CST. The appropriate local and state agencies were notified. The Reactor Coolant System transient was maintained within design pressure by the pressurizer power operated relief valves (PORV). Suspected stroke reversal of both PORV block valves caused a trip of their respective circuit breakers. This occurrence was subsequently resolved. Because of indications that a PORV did not reclose, its associated block valve was maintained closed. Power was not removed from the block valve within the time required by Technical Specifications through licensed personnel error. The PORV position indication was determined to be erroneous and was corrected by repairing the valve position indication mechanism.

The cause of this event was a cognitive personnel error in isolating the incorrect steamline pressure transmitter. Corrective actions include improved instrument identification tags and procedural enhancements. In addition, further evaluation of contributing factors is being conducted. This report also fulfills the reporting requirements of Technical Specification 3.5.2. Similar previous events were discussed in Licensee Event Reports 85-012-00 and 85-021-00. There was no threat to the health and safety of the public.

IE22

☆U.S.GPO:1984-0-464-481/18759

At 1818 CST on January 8, 1987, due to a cognitive personnel error during calibration testing, a low steamline pressure signal on Steam Generator (SG)[AB-SG] C initiated a Safety Injection (SI) Signal and Main Steamline Isolation. This resulted in a Reactor [AB-RCT] trip, Main Feedwater [SJ] Isolation, Containment Isolation [JM] Phase A, Containment Purge [VA] Isolation Signal, Emergency Diesel Generator [EK-DG] start, Control Room Ventilation [VI] Isolation, Auxiliary Feedwater System [BA] actuation, Essential Service Water System [BI] actuation, and Steam Generator Blowdown [WI] and Sample Isolation.

At the time of this event, the plant was in Mode 1, Power Operation at approximately 100 percent power. Reactor Coolant System pressure and average temperature were approximately 2250 psig and 588 degrees Fahrenheit respectively. Pressurizer [AB-PZR] level was in automatic control at approximately 61 percent. Pressurizer pressure control was in automatic mode, with backup heater Bank 'A' in manual mode.

At the time of this event, Instrumentation and Control (I&C) technicians were performing calibration of Steamline Pressure Transmitter AB-PT-0535 [SB-PT] under the direction of a lead I&C Technician in the Control Room. Channel AB-PT-0535 was placed in "Test" position in the Safeguards Cabinet located in the Control Room, providing one signal of the two out of three logic necessary to initiate a SI signal. A cognitive personnel error resulted in Pressure Transmitter AB-PT-0534 being isolated by closing the sensing line valve at the steamline. This transmitter is located on Auxiliary Building [NF] elevation 2026' in the immediate vicinity of Pressure Transmitter AB-PT-0535. This provided the two out of three coincidence necessary to initiate the SI signal.

Due to the initiation of a SI signal while in Mode 1, an Unusual Event was declared at 1818 CST. The appropriate local and state authorities were subsequently notified. At 1826 CST, it was observed by the control room operators that a pressurizer Power Operated Relief Valve (PORV)[AB-PCV], BB-PCV-455A, had lifted during the transient and apparently failed to reclose, based on position indication on the Main Control Board (MCB)[IB-MCBD]. PORV Block Valve [AB-V] BB HV-8000A, was maintained closed in order to isolate the flow path to the affected PORV. This PORV was declared inoperable at 1836 CST.

At 1835 CST, the SI pumps, residual heat removal pumps and emergency diesel generators were secured. After plant conditions had stabilized, the Unusual Event was terminated at 1850 CST. At 2012 CST, it was recognized that power had not been isolated from Block Valve BB HV-8000A as required by Action Statement 'b' of Technical Specification 3.4.4 within one hour of PORV BB-PCV-455A being declared inoperable. Power isolation from this valve was accomplished at 2025 CST.

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

The RCS transient was limited to a maximum pressure of approximately 2410 psig by the pressurizer PORV's and did not drop below approximately 2175 psig during the transient. The approximate maximum RCS hot leg temperature was 618 degrees Fahrenheit. Maximum pressurizer level was approximately 78 percent.

Subsequent inspection of PORV BB-PCV-455A at 0120 CST on January 9, 1987, found that the valve had closed, although indicating lights showed "Open". At 0111 CST on January 10, 1987, continued troubleshooting efforts demonstrated that indication of valve position on the Main Control Board did not represent the true position of the valve plug. By 0540 CST, sufficient valve stroking exercises of PORV BB-PCV-455A had been performed and evaluation had been completed to establish a plan for correction of the problem. Maintenance personnel entered the reactor containment in order to check the limit switches on PORV BB HV-455A. By 1145 CST, it was determined that PORV BB-HV-455A was cycling properly, but its position indication mechanism was not operating properly.

During the RCS transient, both PORV Block Valves BB-HV-8000A & B tripped their circuit breakers. The closure of both block valves was noted by indicating lights when the circuit breakers were re-energized. During the investigation to determine the reason for the tripping of the circuit breakers, it was recognized that the block valve control circuitry was designed to allow the motor to reverse its direction of travel during operation. The pressure transient following the reactor trip decayed rapidly when the PORV's opened, which caused the block valves to start closing on low pressurizer pressure. The PORV's reclosed, producing an increase in pressure in the pressurizer. It is believed that the block valve control circuit may have attempted to reverse direction of the block valves during stroke, due to fluctuating pressure signals during the pressure transient. This may have caused the motor operators to experience a large current up to a maximum of twice locked rotor conditions, exceeding the instantaneous trip setting on the circuit breakers and resulting in trip of the circuit breakers. At 1800 CST on January 10, 1987, subsequent to a design review, the instantaneous current trip settings on the circuit breakers were increased from 45 amps to 58 amps. This modification to the instantaneous current trip settings was intended to preclude circuit breaker trips during normal valve operations. This current trip setting correction was confirmed by cycling one of the block valves through a test which attempted to reproduce the high current conditions which occurred during the transient.

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 7	— 0 0 2	— 0 0	0 4 OF 0 5

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

At 1911 CST, Block Valves BB HV-8000A and B were closed in order to facilitate repair of the position indication mechanism of PORV BB PCV-455A, thereby entering Action Statements 'c' and 'd' of Technical Specification 3.4.4. At 1405 CST on January 11, 1987, RCS cooldown was initiated in compliance with Technical Specification 3.4.4 while work on the PORV was being completed. AT 1720 CST, the work on the position indication mechanism including replacement of magnetic rod and switch assembly in PORV BB PCV-455A was completed. This valve is a Solenoid Operated Pressure Relief Valve manufactured by Crosby Valve and Gage Company. At 1817 CST, electrical power was restored to Block Valve BB HV-8000B and PORV BB PCV-456A was returned to service, exiting Action Statements 'c' and 'd' of Technical Specification 3.4.4. At 1840 CST, the RCS cooldown was terminated, Block Valve BB-HV-8000A was opened and energized, and at 2017 CST PORV BB PCV-455A was declared operational exiting Action Statement 'b' of Technical Specification 3.4.4. Reactor startup was commenced and at 2327 CST on January 11, 1987, Mode 2, Startup, was reached.

This event is being reported pursuant to 10CFR 50.73(a)(2)(iv) as an event that resulted in automatic activation of Engineered Safety Features, 10CFR 50.73(a)(2)(vii) with respect to the PORV block valves as a single cause or condition whereby two independent trains became inoperable in a single system designed to mitigate the consequences of an accident and 10CFR 50.73(a)(2)(i)(b) as an operation prohibited by Technical Specification 3.4.4, Action Statement 'b' when PORV Block Valve BB HV-8000A was closed due to an erroneous PORV open indication without removing power within the one hour period allowance. In addition, this event is being reported pursuant to Technical Specifications 6.9.2 and 3.5.2, Action Statement 'b', which requires the submittal of a report describing the circumstances of the actuation and the total accumulated actuation cycles in the event the Emergency Core Cooling System (ECCS) is actuated and injects into the RCS.

The ECCS actuation described in this report is the fourth actuation that has occurred while the unit was at or near operating temperature and pressure. The current value of the usage factor for each Safety Injection nozzle is less than 0.7.

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

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 7	— 0 0 2	— 0 0	0 5 OF 0 5

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

The cause of this SI was a cognitive personnel error during performance of routine calibration testing. As an immediate corrective action, larger, more legible instrument identification tags are being added to steam line pressure and feedwater flow transmitters. In addition, the calibration procedure is being revised to accomplish isolation of the pressure transmitter at the steamline first. This serves to energize control room alarms and status indications providing diverse verification of the correct channel to be placed in test in the control room. As a longer term corrective measure, further recommendations for correcting the factors contributing to this event are being developed. The failure to remove the electrical power from Block Valve BB HV-8000A was a cognitive personnel error by a licensed operator. The cause of this error was personnel oversight in failing to recall the requirement to remove power to the block valve. Contributing to this was the high level of operator attention required during the stabilization and recovery from this event. The oversight was discovered during review of the plant status by the same personnel who committed the error. This Licensee Event Report will be added to required reading for all plant operators and I&C technicians. In addition, engineering evaluation of the PORV block valve control circuit is continuing to determine if any further corrective action is needed and to identify any other valve control circuit which could be affected by similar conditions.

Previous occurrences of personnel error causing a low steam line pressure signal resulting in SI are discussed in Licensee Event Reports 85-012-00 and 85-021-00. The pressure transient was a brief event adequately controlled and at no time during this event did conditions develop that may have posed a threat to the health and safety of the public.

WOLF CREEK
NUCLEAR OPERATING
CORPORATION

February 9, 1987

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

Letter: ET 87-0039
Re: Docket No. 50-482
Subj: Licensee Event Report 87-002-00

Gentlemen:

The enclosed Licensee Event Report is submitted pursuant to 10 CFR 50.73 (a) (2) (i), 10 CFR 50.73 (a) (2) (iv), and 10 CFR 50.73 (a) (2) (vii). In addition, this report is being submitted pursuant to Wolf Creek Generating Station Technical Specifications 6.9.2 and 3.5.2, Action Statement 'b'.

Yours very truly,

John A Bailey

John A. Bailey
Vice-President Engineering
and Technical Services

JAB:see

Attachment

cc: PO'Connor (2)
JCummins
RMartin

IE22
1/1