U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

Line	Field	Entry
0 1	LICENSEE CODE (2)	C O F S V 1
	LICENSE NUMBER (3)	0 0 - 0 0 0 0 0 - 0 0
	LICENSE TYPE (4)	4 1 1 1 2 0
	CAT (5)	
CON'T 0 1	REPORT SOURCE (6)	L
	DOCKET NUMBER (7)	0 5 0 0 0 2 6 7
	EVENT DATE (8)	0 8 1 7 7 9
	REPORT DATE (9)	0 5 1 6 8 4

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On August 17, 1979, at approximately 1522 hours, plant personnel inadvertantly grounded
0 3 instrument panel I-36, blowing the panels fuses and causing a voltage perturbation on
0 4 instrument bus 2. This resulted in a Loop I shutdown, reactor scram, and loss of
0 5 forced circulation for approximately three minutes. This event was reportable per
0 6 Fort St. Vrain Technical Specification AC 7.5.2(a)5. Similar reports are RO's: 76-01,
0 7 77-14, and 79-17. There was no effect on public health and safety. No accompanying
0 8 occurrence.

Line	Field	Entry
0 9	SYSTEM CODE (11)	E D
	CAUSE CODE (12)	A
	CAUSE SUBCODE (13)	C
	COMPONENT CODE (14)	Z Z Z Z Z Z
	COMP. SUBCODE (15)	Z
	VALVE SUBCODE (16)	Z

LER/RO REPORT NUMBER (17)	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	7 9	—	0 2 8	/	0 1	X	—	1

ACTION TAKEN (18)	FUTURE ACTION (19)	EFFECT ON PLANT (20)	SHUTDOWN METHOD (21)	HOURS (22)	ATTACHMENT SUBMITTED (23)	NPRD-4 FORM SUB. (24)	PRIME COMP. SUPPLIER (25)	COMPONENT MANUFACTURER (26)
X	Z	A	C	0 0 6 9	Y	N	Z	Z 9 9 9

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The ground was caused by personal error of a non-licensed maintenance personnel. The
1 1 ground was corrected, power restored to the instrument panel and action taken to
1 2 return the plant to normal conditions. The effects of the upset were analyzed and were
1 3 determined to have no adverse impact on reactor internal components. No further
1 4 corrective action is anticipated or required.

Line	Field	Entry
1 5	FACILITY STATUS (28)	E
	% POWER (29)	0 6 8
	OTHER STATUS (30)	N/A
	METHOD OF DISCOVERY (31)	A
	DISCOVERY DESCRIPTION (32)	Personnel observation
1 6	ACTIVITY RELEASED (33)	Z
	CONTENT OF RELEASE (34)	Z
	AMOUNT OF ACTIVITY (35)	N/A
	LOCATION OF RELEASE (36)	N/A
1 7	PERSONNEL EXPOSURES NUMBER (37)	0 0 0
	TYPE (38)	Z
	DESCRIPTION (39)	N/A
1 8	PERSONNEL INJURIES NUMBER (40)	0 0 0
	DESCRIPTION (41)	N/A
1 9	LOSS OF OR DAMAGE TO FACILITY TYPE (42)	Z
	DESCRIPTION (43)	N/A
2 0	PUBLICITY ISSUED (44)	N
	DESCRIPTION (45)	N/A
	NRC USE ONLY	

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NAME OF PREPARER Frank Novachek PHONE: (303) 785-2224

Public Service Company of Colorado

16805 WCR 19 1/2, Platteville, Colorado 80651

50-267

May 16, 1984
Fort St. Vrain
Unit #1
P-84146

RECEIVED
MAY 22 1984

Mr. John T. Collins, Regional Administrator
Region IV
Nuclear Regulatory Commission
611 Ryan Plaza Drive
Suite 1000
Arlington, Texas 76011

REFERENCE: Facility Operating License
No. DPR-34

Docket No. 50-267

Dear Mr. Collins:

Enclosed please find a copy of Reportable Occurrence Report No. 50-267/79-028, Final, submitted per the requirements of Technical Specification AC 7.5.2(a)5.

Very truly yours,

Don Warembourg

Don Warembourg
Manager, Nuclear Production

DWW/djm

Enclosure

cc: Director, MIPC

H005
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FORT ST. VRAIN NUCLEAR GENERATING STATION
PUBLIC SERVICE COMPANY OF COLORADO
16805 WELD COUNTY ROAD 19 1/2
PLATTEVILLE, COLORADO 80651-9298

REPORT NO. 50-267/79-28/01-X-1

Final

IDENTIFICATION OF OCCURRENCE:

On Friday, August 17, 1979, with the plant operating at 68% thermal power and 212 MW electrical power, an instrument panel, I-36, was shorted to ground and tripped. This resulted in a reactor scram and an approximate three minute interruption of reactor cooling.

This was reportable per Fort St. Vrain Technical Specification AC 7.5.2(a)5.

CONDITIONS PRIOR TO OCCURRENCE:

Steady state power.

The major plant parameters at the time of the event were as follows:

Reactor Power	68% Power
Electrical Power	212 MWe
Secondary Coolant Pressure	2,400 psig
Secondary Coolant Temperature	1,000 °F
Secondary Coolant Flow (1A and 1C Boiler Feedpumps Supplying)	1,400,000 #/hour
Primary Coolant Pressure	645 psig
Primary Coolant Circulator Temperature	666 °F
Primary Coolant Core Outlet Temperature	1,334 °F
Primary Coolant Flow	77%
Circulator 1A	6,700 RPM
Circulator 1B	6,700 RPM
Circulator 1C	6,400 RPM
Circulator 1D	6,400 RPM

Actions that lead to the event were as follows:

Circulator trips on circulator speed/feedwater flow program had been experienced as a result of circulator speed cable resistance changes at the circulator connector. As the resistance changed, an imbalance of the speed modifier output would appear as a low circulator speed signal to the control system. As the speed would appear to decrease, the control system, through the speed controller (SC), would call for further opening of the circulator steam turbine speed valve (SV). As the speed valve would open to correct the apparent low speed condition, the actual circulator speed would increase even though the circulator speed had actually been at the proper setpoint for the existing plant conditions. The plant protective system (PPS) circulator speed/feedwater flow program trip was unaffected by the loss of speed signal to the control system. As a result of the actual circulator speed increase, the plant protective system functioned to trip the circualtor as designed.

To avoid unnecessary circulator trips, it was decided to connect an electrical high selector (>) on the output of the steam turbine and water turbine speed pulse to current transmitters. This provides a speed signal to the control system from either of the identical speed elements (SE) on the circulator so that the steam turbine speed elements becoming imbalanced would not cause a programmed circulator overspeed trip.

NOTE: This problem with circulator speed cables causing erroneous speed indications is similar to that reported in Reportable Occurrence Report No. 50-267/78-05. Here however, the speed control circuits and not the Plant Protective circuits were affected.

The required approvals were received, and preparations were made to wire the high selector (>) into the I-36B instrument panel. The speed controller for 1D circulator was placed in manual control during the installation.

DESCRIPTION OF
OCCURRENCE:

Sequence of Events

Time	Event
	The high selector (>) was being wired into I-36B when a wire was inadvertently shorted to ground. This interrupted AC power to I-36B and I-36A and resulted in a voltage dip on instrument bus 2 which is the AC power supply to I-36B and I-36A.

1522:55:55 The voltage dip on instrument bus 2 caused a spurious trip of the Loop 1 steam generator penetration pressure high circuitry. This initiated a Loop 1 shutdown which initiated 1A and 1B circulator steam turbine trips and a half-load turbine runback. There was no steam water dump due to duration time of the penetration pressure high trip signal.

1522:56:26 The Loop 1 steam generator penetration high trip returned to normal.

1523:02:38 Reactor scrammed due to spurious hot reheat temperature high signals in scram channels B and C as observed by Reactor Operator.

1523:02:38 Loop 2 circulator speeds decreased due to a decrease in cold reheat steam pressure. This decrease was due to loss of control signal power to PV-2230 (Loop 2 main steam bypass pressure control valve) which prevented normal steam flow to the bypass flash tank during the turbine runback. The auxiliary boiler was in manual at a low flow rate which prevented use of steam from the 150 pound header supply.

NOTE: At this time, the turbine was in the runback mode, circulator speed valves were apparently closing due to loss of control signal power, and circulator speed was decreasing as was feedwater flow. This indicates that with the ISS switch in POWER, Loop 2 circulator speed/flow trip setpoints were being approached.

1523:09:21 1D circulator steam turbine trip from circulator speed/feedwater flow program.

1523:10:54 1C circulator steam turbine trip from circulator speed/feedwater flow program.

1523:10:54 Two-loop trouble scram and turbine trip initiated by last and final circulator trip.

1523:36 Loop 2 feedwater flow isolated for less than 5 seconds. Loop 2 feedwater flow was isolated when the Reactor Operator closed FV-2206 (Loop 2 feedwater flow control valve). The Reactor Operator closed FV-2206 by selecting manual control on FC-2206.

1524:19:37 The ISS handswitch was placed in the low power position as indicated by alarm printer output (alarms returning to normal status).

1525:35 1C circulator speed manually increased on steam turbine to restore primary coolant flow.

1525:40 1D circulator speed manually increased on steam turbine.

Secondary coolant flow on Loop 2 decreased during this incident, but feedwater temperature (TT-2206) indicates feedwater flow remained in service. Since PV-2230 (Loop 2 main steam bypass valve) failed to open, the Loop 2 main steam safety relief valves lifted to vent pressure and maintain Loop 2 secondary flow.

Primary coolant flow was lost for 2 minutes and 26 seconds. The Reactor Operator manually restarted 1C circulator as noted above. No automatic water turbine start occurred because the ISS switch had been placed in low power position.

It should be noted that 1C circulator inlet helium temperatures exceeded 800 degrees fahrenheit for approximately 12 minutes. The general design limit for circulator inlet helium temperatures is 800 degrees fahrenheit.

APPARENT CAUSE
OF OCCURRENCE:

| Personnel Error.

| The inadvertent grounding of the I-36 instrument panel resulted in a
| loss of power to the panel and the voltage perturbation on instrument bus 2, which led to a Loop 1 shutdown, reactor scram, and a temporary loss of forced cooling.

ANALYSIS OF
OCCURRENCE:

| The inadvertent grounding of instrument panel I-36 has been
| attributed to personnel error. The following loss of I-36A and I-36B and the voltage perturbation on instrument bus 2 resulted in the loss of control of forced circulation in Loop 2. The plant responses were as expected under these circumstances. However, the Loop 1 trip due to steam generator penetration pressure high, and the reactor scram due to hot reheat temperature high, were not expected and resulted from spurious noise signals generated in the plant protective system circuitry.

NOTE: After the occurrence, the plant protective system logic circuits were tested and all found to be operating satisfactorily.

During this occurrence, several significant developments arose; a loss of forced circulation occurred during which the power-to-flow ratio increased to approximately 1.92, 1C circulator inlet temperature exceeded 800 degrees fahrenheit for approximately 11 minutes, and the steam generator cross-over temperature exceeded 950 degrees fahrenheit for approximately 25 minutes.

| The duration of the increased power-to-flow ratio condition caused by the loss of forced circulation did not exceed the limits of | Specification SL 3.1. The power-to-flow ratio did increase to 1.92 momentarily, but quickly decreased to less than 1.0. The following table gives the data for this transient.

Power-To-Flow Ratio	Time at Power-To-Flow Ratio (Seconds)	Allowable Time at Power-To-Flow Ratio
1.125 - 1.15	1	100 Hours
1.2 - 1.225	1	100 Hours
1.275 - 1.30	1	60 Hours
1.350 - 1.375	1	27 Hours
1.425 - 1.450	1	13 Hours
1.50 - 1.525	1	6 Hours
1.525 - 1.55	1	1.35E4 Seconds
1.55 - 1.60	1	8.28E3 Seconds
1.60 - 1.65	1	5.58E3 Seconds
1.65 - 1.70	1	3.60E3 Seconds
1.70 - 1.75	1	2.52E3 Seconds
1.75 - 1.80	1	1.62E3 Seconds
1.80 - 1.85	1	1.30E3 Seconds
1.85 - 1.90	1	9.36E2 Seconds
1.90 - 1.95	0	7.56E2 Seconds
1.95 - 2.00	0	5.76E2 Seconds

| This power-to-flow transient was analyzed in the weekly surveillance and was not reportable in itself.

| General Atomic Company has analyzed the high circulator inlet temperatures and has determined that there were no adverse effects on the reactor internal components.

CORRECTIVE
ACTION:

Plant personnel removed the ground from I-36 instrument panel, replaced the blown fuses, and restored power to the panel.

| The Reactor Operator restored primary coolant flow by starting 1C and 1D circulators on steam.

| General Atomic Company analyzed the effects of circulator inlet temperatures exceeding 800°F, and concluded that there were no adverse effects on reactor internal components.

| No further corrective action is anticipated or required.

FAILURE DATA/SIMILAR REPORTED OCCURRENCES:

Similar occurrences were reported in Reportable Occurrence Report Numbers 76-01, 77-14, and 79-17.

PROGRAMMATIC IMPACT:

None

CODE IMPACT:

None

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