FEB 15 1983

MEMORANDUM FOR: C. Trammell, Project Manager, Operating Reactors Branch 3, NRR

FROM: P. Wagner, Project Manager, Reactor Project Branch 1

SUBJECT: TIA ON FORT ST. VRAIN (FSV) STEAM GENERATOR TUBE LEAK

50-267

Attached is a copy of the LER that Public Service Company of Colorado (PSC) submitted on December 22, 1982. This LER is preliminary and provides details of the events leading to identifying and "plugging" the leaking steam generator tube. As I explained at the OR Events Briefing on December 16, 1982, the "plugging" operation involves the removal of sections from the feedwater lead in and the steam lead out tubes followed by capping both ends of each. This operation actually removes 3 of the 54 tubes from service in the affected module. The leak occurred in Module B-2-3, and was identified in the Superheater II section of the tubes in subheader M.

This was the second tube leak to occur at FSV; the first occurred on November 30, 1977, in Module B-1-1. Information about the first leak is contained in the following letters:

a. PSC to NRC dated December 9, 1977

b. GA to NRC dated December 21, 1977

c. NRC to GA dated January 6, 1978

d. GA to NRC dated January 30, 1978

I doubt that this information would be of much assistance in the review of the recent leak since the earlier leak was probably caused by a problem during fabrication, but I can provide copies if you so desire.

As stated in the attached LER, PSC will provide a final report on the recent leak. This final report will contain PSC's evaluation of the leak and provide a discussion of the problems involved with inspecting the FSV steam generators.

RPB1	RPB1	DRRP&EP/RIV
PWagner	GMadsen	JGagliardo
2/11/83	2/11/83	2/15/83

H005

In addition, the subheader sections removed for the "plugging" operation were sent to GA for metalography studies. I expect to receive a copy of GA's findings around April 1, 1983, and will provide you with a copy upon receipt.

/S/

Philip C. Wagner, Project Manager
Reactor Project Branch 1

Attachment: As stated

cc:
J. Gagliardo
G. Madsen
R. Clark, ORB3
D. Wessman, AD/SA

Public Service Company of Colorado

16805 Road 19 1/2, Platteville, Colorado 80651-9298

December 22, 1982
Fort St. Vrain
Unit No. 1
P-82551

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. 50-267/82-049, Preliminary, submitted per the requirements of Technical Specification AC 7.5.2(a)3.

Also, please find enclosed one copy of the Licensee Event Report for Reportable Occurrence Report No. 50-267/82-049.

Very truly yours,

Don Warembourg

Don Warembourg
Manager, Nuclear Production

DW/cls

Enclosure

cc: Director, MIPC

Dupe

~~8301180364~~

REPORT DATE: December 22, 1982 REPORTABLE OCCURRENCE 82-049
Determined ISSUE 0
OCCURRENCE DATE: December 8, 1982 Page 1 of 4

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

REPORT NO. 50-267/82-049/01-T-0

Preliminary

IDENTIFICATION OF OCCURRENCE:

On December 8, 1982, while operating at 1% reactor power, a secondary side to primary side leak was discovered in the economizer-evaporator-superheater (EES) section of the B-2-3 module in the Loop 2 steam generator. The leak is assumed to have developed following a reactor scram transient which occurred on September 30, 1982.

This event is reportable per Fort St. Vrain Technical Specification AC 7.5.2(a)3.

CONDITIONS PRIOR TO OCCURRENCE:

Post reactor scram cooldown (assumed).

DESCRIPTION OF OCCURRENCE:

On September 30, 1982, while the plant was operating at 70% reactor power with a turbine generator load of 215 MW(e), a reactor scram occurred as a result of plant protective system surveillance testing.

Primary coolant moisture levels began to rise shortly after the scram but for no apparent reason. Based on previous experience, however, it was assumed that the cause of the increase was an unobserved upset in the helium circulator auxiliary system which, in turn, caused water to enter the reactor vessel.

Continued high primary coolant moisture levels in combination with unusual amounts of water being removed by the helium purification system, indicated the possible presence of an additional source of water into the reactor vessel. Based on this indication, investigations into other possible sources of moisture entering the reactor vessel were initiated.

On December 8, 1982, it was determined that a secondary side to primary side leak existed in the B-2-3 module of the Loop 2 steam generator.

APPARENT CAUSE
OF OCCURRENCE:

Component failure.

ANALYSIS OF
OCCURRENCE:

Since the reactor has remained in a shutdown or low power condition from the time of occurrence, the degradation of the primary coolant pressure boundary due to the discovered steam generator leak has posed no danger to the health and safety of the public. The only time that primary coolant was detected on the secondary side was during the portion of the "moisture source" investigation that finally identified the leak in the Loop 2 steam generator. At that time, the steam generator was isolated and dumped. Primary coolant from the dump tank was processed through the gas waste system in a normal manner after sampling.

Had the reactor been operated at power with the leak, the normal operation of the secondary side pressure above the primary side pressure would have precluded any egress of primary coolant from the reactor vessel via the Loop 2 steam generator.

The water ingress into the reactor vessel due to the said leak, resulted in operation under a degraded mode of LCO 4.2.11 and was reported in Reportable Occurrance No. 82-044.

Although the actual cause of the leak is not identified, it appears to be random in nature. Once plugged, the leak should have no impact on future operation.

CORRECTIVE
ACTION:

The affected steam generator was isolated.

Efforts to identify the precise location of the leak and to make necessary repairs are in progress. Upon completion of these efforts, a final report will be submitted.

FAILURE DATA/SIMILAR
REPORTED OCCURRENCES:

Similar Reportable Occurrence is RO 77-42.

PROGRAMMATIC IMPACT:

The reactor has remained in a shutdown or low power condition since September 30, 1982. This status will continue until the necessary repairs are made.

CODE IMPACT:

None

Prepared By: Frank J. Novachek
Reactor Engineer

Reviewed By: Charles Fuller
Charles Fuller
Technical Services Engineering Supervisor

Reviewed By: Hill by M. Block
Edwin D. Hill
Station Manager

Approved By: Don Warembourg
Don Warembourg
Manager, Nuclear Production

NRC FORM 366 (7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

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

0 1	LICENSEE CODE: C O F S V 1 (2)	LICENSE NUMBER: 0 0 - 0 0 0 0 0 - 0 0 (3)	LICENSE TYPE: 4 1 1 1 0 (4)	CAT (5)

CON'T

0 1	REPORT SOURCE: L (6)	DOCKET NUMBER: 0 5 0 0 0 2 6 7 (7)	EVENT DATE: 1 2 0 8 8 2 (8)	REPORT DATE: 1 2 2 2 8 2 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

While operating at 1% reactor power, a secondary to primary side leak was discovered in the EES section of the B-2-3 module in the Loop 2 steam generator. The leak is assumed to have developed following a reactor scram transient on September 30, 1982. No effect on public health or safety. Accompanying occurrences reported in RO 82-044. Similar report is RO 77-42.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	H B (11)	E (12)	B (13)	H T E X C H (14)	F (15)	Z (16)

(17) LER/RO REPORT NUMBER

EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
8 2	—	0 4 9	/	0 1	T	—	0

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	D (19)	C (20)	Z (21)	0 0 0 0	Y (23)	N (24)	N (25)	G 0 6 3 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

The actual cause is not identified, but appears to be random in nature. The affected steam generator was isolated. Efforts to identify the precise location of the leak and to make necessary repairs are in progress. Upon completion of these efforts, a final report will be submitted.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	0 0 0 (29)	N/A	C (31)	Special Investigation

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	N/A	

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	N/A

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	N/A

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	N/A

PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	N (44)	N/A	

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(303) 785-2224