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
Manager, Nuclear Production

DW/cls

Enclosure

cc: Director, MIPC

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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. 50-267/82-049/01-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 Occurrence 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
Technical Services Engineering Supervisor

Reviewed By: _____
Edwin D. Hill
Station Manager

Approved By: _____
Don Warembourg
Manager, Nuclear Production