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

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0 1 | REPORT SOURCE L (6) | 0 5 0 0 0 2 6 7 (7) | 0 6 0 4 8 0 (8) | 0 3 2 8 8 4 (9)
DOCKET NUMBER | EVENT DATE | REPORT DATE

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 On June 4, 1980, during normal power operation while performing SR 5.2.16a-m, Loop 2
0 3 steam generator penetration leakage appeared to be greater than 400 pounds per day.
0 4 Further testing revealed the leakage was internal to the steam generator and not
0 5 through the seals. Leakage was greater than 700 pounds per day on several occasions
0 6 during February and March, 1981, and resulted in a plant shutdown on March 22, 1981,
0 7 for further investigation. These events were reported as operation in a degraded
0 8 mode of LCO 4.2.9 per Fort St. Vrain Technical Specification AC 7.5.2(b)2. No affect on public health or safety. No similar reportable occurrences.

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	H B (11)	X (12)	Z (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 0	—	0 3 0	/	0 3	X	—	3

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
X (18)	Z (19)	A (20)	A (21)	0 2 1 6	Y (23)	Y (24)	N (25)	G 0 6 3 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The purified helium leakage is internal to the pe[illegible]ation interspace and occurred
1 1 between the interspace and associated cold reheat steam piping. Public Service
1 2 Company Change Notice (CN) 1436 modified the steam generator interspace to operate at
1 3 a pressure slightly greater than cold reheat steam. Revisions to LCO 4.2.7 and 4.2.9 were approved to allow operation in this manner and set limits on possible releases of
1 4 primary coolant activity. No further corrective action is anticipated or required.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	E (28)	0 6 7 (29)	N/A	B (31)	Regular Scheduled Surveillance

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

8404090146 840328
PDR ADOCK 05000267
S PDR

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	

NAME OF PREPARER [signature]

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GPO 917-026

x'

FORT ST. VRAIN NUCLEAR GENERATING STATION
PUBLIC SERVICE COMPANY OF COLORADO
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PLATTEVILLE, COLORADO 80651-9298

| REPORT NO. 50-267/80-30/03-X-3

| Revised Final

IDENTIFICATION OF OCCURRENCE:

During performance of plant test T-147, the steam generator penetration interspace leakage appeared to be in excess of the limit allowed by LCO 4.2.9 and the variance granted by the Nuclear Regulatory Commission on June 5, 1980.

This was reported as operation in a degraded mode of LCO 4.2.9 per Fort St. Vrain Technical Specification AC 7.5.2(b)2.

EVENT DESCRIPTION:

On May 28, 1980, during performance of the scheduled surveillance on PCRV leakage, Loop 2 steam generator penetration was found to have a leak rate of 8.23 pounds per hour or 198 pounds per day. This was well within the 400 pounds per day limit of LCO 4.2.9.

On June 4, 1980, the surveillance was again performed and indicated steam generator penetration leakage to be 17.83 pounds per hour. This was 428 pounds per day and in excess of the LCO 4.2.9 limit of 400 pounds per day. A special test, T-145, using a pressure decay method, was written, approved, and performed later that day and showed the leakage rate to be 10.74 pounds per hour or 258 pounds per day, within the limits of LCO 4.2.9. T-145 was a more accurate test and was used at this time to insure Technical Specification compliance.

During performance of T-145, it was noted that the pressure decay rate did not decrease as reactor pressure was approached. This indicated that a primary closure seal was not leaking. Further investigation revealed a noncondensible gas was in the hot reheat sample line. A sample was taken, analyzed, and determined to be clean helium. The leakage flow path was believed to be from the penetration interspace to the cold reheat piping and out of the steam generator via the hot reheat piping (see Figure 1).

On June 5, 1980, T-145 was again performed and revealed a leakage rate of 10.37 pounds per hour or 249 pounds per day. A letter was sent to the Nuclear Regulatory Commission explaining the situation and requesting temporary relief from LCO 4.2.9, permitting up to 700 pounds per day instead of 400 pounds per day for the suspected internal leakage path.

The Nuclear Regulatory Commission granted the requested temporary relief, agreed to the four administrative controls proposed by Public Service Company, and added one additional administrative control. The five administrative controls agreed to are as follows:

1. SR 5.3.7, Secondary Coolant Activity, be conducted once each 72 hours in lieu of once per week.

2. A pressure decay test be conducted for the Loop 2 steam generator penetration closures on a weekly basis. This pressure decay test was utilized to determine the leakage rate. Leakage rate increases of 25% over previous values required conducting pressure decay tests daily until it was established that the leak rate had reached an equilibrium value within the 700 pounds per day.

3. SR 5.2.16, PCRV Closure Leakage, be conducted once every two weeks for Loop 2 steam generator penetration closure rather than monthly as a comparison to the pressure decay tests.

4. Radiation process monitors for the reheat steam system will be monitored once per shift for indication of primary coolant leakage into the secondary system.

5. Check and record the interspace differential pressure once per shift to comply with LCO 4.2.7.

To collect all the required data, T-145 was expanded and renumbered T-147. Both T-147 and SR 5.2.16 were performed during each power increase, as required, and also as the leakage appeared to be somewhat dependent on power level.

The leakage was within the 700 pounds per day limit, except for three occasions during power level increases:

1. On February 23, 1981, SR 5.2.16a-M was performed and indicated 902.4 pounds per day leakage. Test T-147 was performed as part of this surveillance and indicated the leakage to be 680.4 pounds per day. Later the same day, T-147 was again performed and indicated the leakage to be 777.6 pounds per day.

2. On February 25, 1981, T-147 was performed and indicated leakage of 709.7 pounds per day.

3. On March 4, 1981, T-147 was performed and indicated leakage of 709.7 pounds per day.

Plant operation was changed in each of these three situations and the leakage rate was reduced to less than the 700 pound per day limit.

On March 21, 1981, at 0227 hours, during plant testing, a turbine trip occurred. Prior to this trip, reactor power had been 69%, and the Loop 2 steam generator interspace leakage rate had been 568 pounds per day. The plant was restored to a stable condition, power increased, and the generator placed back on line at 0702 hours the same day. Reactor power was increased, and normal plant operation resumed during the remainder of the day shift. At 1600 hours, T-147 was performed and indicated the steam generator interspace leakage to be 1312 pounds per day at 57% reactor power. Various changes were made to plant operations, but the steam generator interspace leakage rate remained above the 700 pounds per day limit. An orderly plant shutdown was initiated at 1832 hours with the turbine generator off line at 0358 hours on March 22, 1981, and the reactor manually scrammed from approximately 2% power at 1145 hours. The reactor was depressurized and investigation began on the cause of the excessive steam generator Loop 2 interspace leakage.

CAUSE
DESCRIPTION:

Other.

The purified helium leakage was internal to the Loop 2 steam generator penetration and occurred between the penetration interspace and the cold reheat steam piping internal to the penetration (see Figure 1).

CORRECTIVE
ACTION:

Public Service Company Change Notice (CN) 1436 was initiated in November, 1981, and has installed instrumentation, piping, valves, and control equipment to allow operation of the steam generator penetration interspaces at a pressure slightly greater than cold reheat but less than reactor pressure. Additional capability to monitor the interspace helium was also provided. These modifications maintain steam generator interspace helium leakage to within the limits of LCO 4.2.9.

Revisions to the Fort St. Vrain Technical Specifications, LCO 4.2.7 and LCO 4.2.9, were proposed by Public Service Company in January, 1982, and approved by the Nuclear Regulatory Commission in March, 1982, as Amendment No. 26 to the Fort St. Vrain Facility Operating License DPR-34.

Amendment No. 26 revised the Technical Specifications to:

(1) permit the interspace between primary and secondary closures of the steam generator modules to be maintained at a pressure slightly above cold reheat steam pressure; and

(2) set a limit on the possible release of primary coolant activity through the primary closure seals of no greater than 1.4 curies per day.

No further corrective action is anticipated or required.

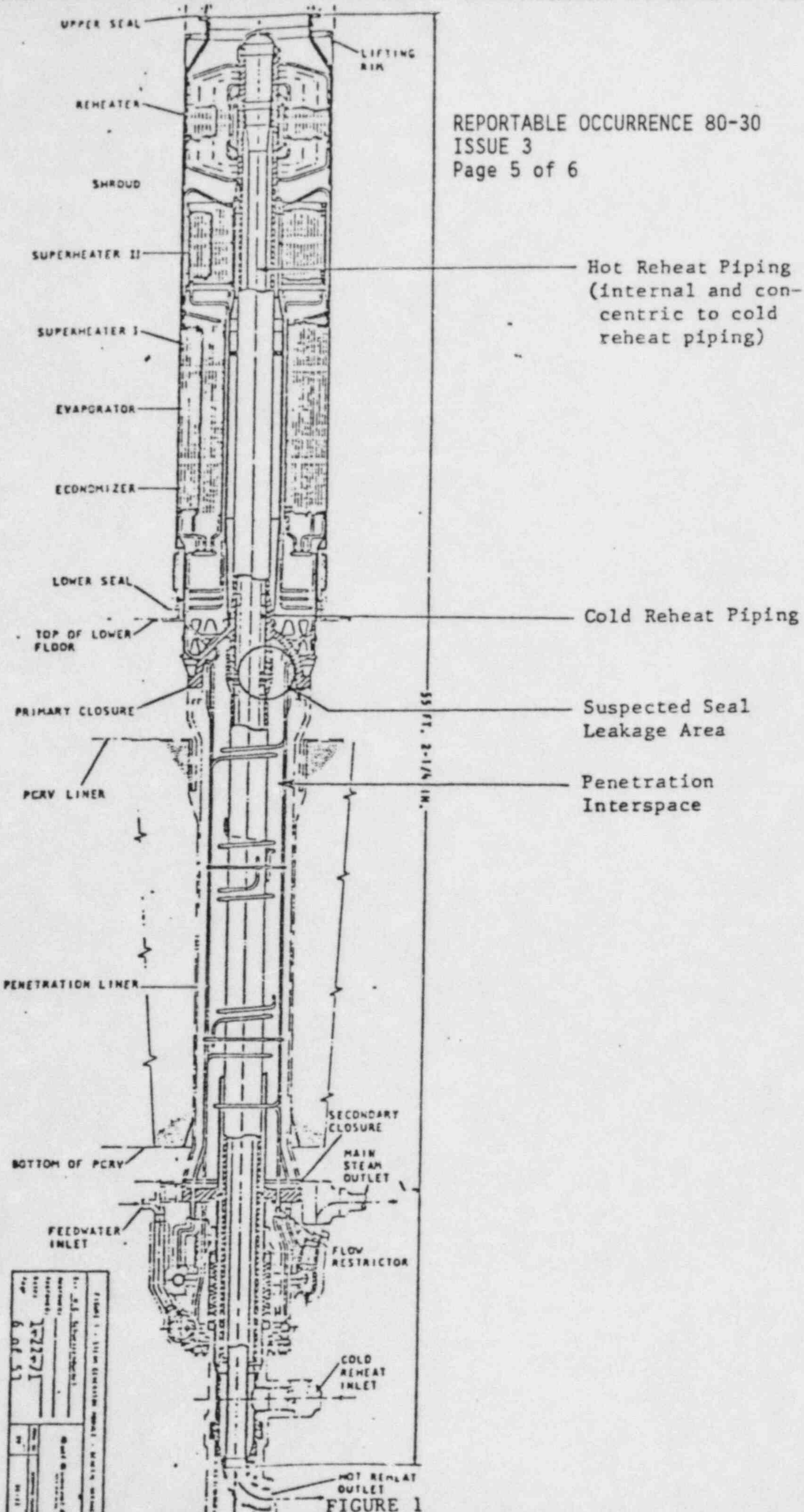
UPPER SEAL
LIFTING RIM
REHEATER
SHROUD
SUPERHEATER II
SUPERHEATER I
EVAPORATOR
ECONOMIZER
LOWER SEAL
TOP OF LOWER FLOOR
PRIMARY CLOSURE
PCRV LINER
PENETRATION LINER
SECONDARY CLOSURE
BOTTOM OF PCRV
MAIN STEAM OUTLET
FEEDWATER INLET
FLOW RESTRICTOR
COLD REHEAT INLET
HOT REHEAT OUTLET
55 FT. 2-1/4 IN.
Hot Reheat Piping (internal and concentric to cold reheat piping)
Cold Reheat Piping
Suspected Seal Leakage Area
Penetration Interspace

FIGURE 1

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