New Hampshire
Yankee

Ted C. Feigenbaum
President and
Chief Executive Officer

NYN-90209

December 10, 1990

United States Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Reference: Facility Operating License No. NPF-86, Docket No. 50-443

Subject: Licensee Event Report (LER) No. 90-025-00: Reactor Trip Due to Steam Generator Low-Low Level Signal

Gentlemen:

Enclosed please find Licensee Event Report (LER) No. 90-025-00 for Seabrook Station. This submittal documents an event which occurred on November 9, 1990, and is being reported pursuant to 10CFR50.73(a)(2)(iv).

Should you require further information regarding this matter, please contact Mr. Allen L. Legendre, Lead Engineer - Compliance, at (603) 474-9521, extension 2373.

Very truly yours,

Ted C. Feigenbaum

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Enclosures: NRC Forms 366, 366A

TCF:WJT/act

1/1

IE22

cc: Mr. Thomas T. Martin
Regional Administrator
United States Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Mr. Noel Dudley
NRC Senior Resident Inspector
P.O. Box 1149
Seabrook, NH 03874

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Records Center
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Atlanta, GA 30339

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SEABROOK STATION	0 5 0 0 0 4 4 3	1 OF 0 3

TITLE (4)
REACTOR TRIP DUE TO STEAM GENERATOR LOW-LOW LEVEL SIGNAL

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)
1 1	0 9	9 0	9 0	0 2 5	0 0	1 2	0 7	9 0		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): 1
POWER LEVEL (10): 1 0 0

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

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)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	
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	
Allen L. Legendre, Lead Engineer - Compliance, Extension 2373	6 0 3	4 7 4 - 9 5 2 1

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC-TURER	REPORTABLE TO NPRDS
B	J B	P S F	C 6 3 5	Y					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) — X NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On November 9, 1990, at 2:31 p.m. EST, while in Mode 1 at 100% power, a reactor trip with turbine-generator trip occurred. The trip was initiated by a steam generator low-low narrow range level signal.

The initiating event was caused by fatigue failure of a control air pipe nipple. This nipple connects the air booster relay to the valve actuator for feedwater flow control valve, 1-FW-FCV-520. Following the failure of the nipple, valve 1-FW-FCV-520, failed closed as designed, due to a loss of control air. The valve closure resulted in a loss of feedwater to the "B" steam generator. A reactor trip occurred as designed, when the steam generator water level dropped below the low-low narrow range level setpoint. Subsequent to the reactor trip, a Main Feedwater Isolation occurred due to high-high steam generator water level signal spikes. In addition, an Emergency Feedwater Actuation occurred as designed, due to the loss of feedwater to a steam generator.

The root cause of the event has been determined to be vibration induced fatigue failure of the pipe fitting due to a less than optimal design location. Corrective actions include relocation of the air booster relays for all four feedwater flow control and bypass valves. In addition, a walkdown of the secondary side will be performed during 100% power operation to identify similar control air arrangements that may be affected by excessive vibration.

This is the first event of this type at Seabrook Station.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
SEABROOK STATION	0 5 0 0 0 4 4 3	9 0	— 0 2 5	— 0 0	0 3 OF 0 3

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

A walkdown of the secondary side will be performed during 100% power operation to identify similar control air arrangements that may be affected by excessive vibration. This walkdown is expected to be completed by March 1, 1991.

Plant Conditions

At the time of this event, the plant was in Mode 1, Power Operation at 100%, with an RCS temperature of 587 degrees Fahrenheit and pressure of 2,235 psig.

This is the first event of this type at Seabrook Station.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
SEABROOK STATION	0 5 0 0 0 4 4 3	9 0	— 0 2 5 —	0 0	0 2 OF 0 3

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

On November 9, 1990 at 2:31 p.m. EST, a reactor trip with a turbine-generator trip occurred while the plant was at 100% power. The reactor trip was initiated by a low-low narrow range level signal for the "B" Steam Generator.

Description of Event

Prior to the event, the plant was at 100%, with plant systems in a steady state condition. The initiating event was caused by a failure of a control air pipe nipple. This nipple connects the air booster relay to the valve actuator for feedwater flow control valve, 1-FW-FCV-520. Following the failure of the nipple, valve 1-FW-FCV-520, failed closed due to a loss of control air. The feedwater flow control valves are 16 inch, air operated plug valves that are designed to fail closed on a loss of control air. The valve closure resulted in a loss of feedwater to the "B" steam generator. A reactor trip with a turbine-generator trip occurred as designed, when the steam generator water level dropped below the low-low level setpoint (17% narrow range).

Following the reactor trip and turbine trip a Main Feedwater Isolation occurred. Pressure pulses were created by the rapid closure of the turbine control valves. These pressure pulses were transmitted through the steam flow transmitters' water filled lines and sensed by the high pressure side of the steam generator narrow range level transmitter. This resulted in the steam generator high-high level signal. Actual steam generator levels did not approach the high-high level setpoint at any time. Additionally, an Emergency Feedwater Actuation occurred as designed, due to the loss of feedwater to a steam generator.

Safety Consequences

There were no adverse safety consequences as a result of this event. All the applicable trips and interlocks associated with the reactor trip functioned as designed.

All operator actions were determined to be appropriate to ensure the safety of the plant. At no time during this event was there any impact on the health and safety of plant employees or the public.

Root Cause

The root cause of the event has been determined to be a vibration induced fatigue failure of the nipple due to a less than optimal design location. The nipple was sent to a vendor for a metallurgical analysis to verify that vibration was indeed the primary cause of the failure.

Corrective Actions

After the trip, the plant was placed in HOT STANDBY in accordance with operating procedure OS1001.11 "Post Trip to Hot Standby". An event evaluation and post trip review were immediately initiated. A Human Performance Evaluation System (HPES) analysis as well as a root cause analysis were also initiated.

A design change was implemented to relocate the air booster relay for all four feedwater flow control valves and bypass valves to an area with less vibration.