CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

RR#1 • BOX 127E • EAST HAMPTON, CT 06424-9341

December 18, 1990
Re: 10CFR50.73(a)(2)(i)(B)

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Reference: Facility Operating License No. DPR-61
Docket No. 50-213
Reportable Occurrence LER 50-213/90-029-00

Gentlemen:

This letter forwards the Licensee Event Report 90-029-00, required to be submitted, pursuant to the requirements of Connecticut Yankee Technical Specifications.

Very truly yours,

John P. Stetz
Station Director

JPS/dl

Attachment: LER 50-213/90-029-00

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

J. T. Shedlosky
Sr. Resident Inspector
Haddam Neck

IE22
111

NRC Form 366 (9-83)

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)
Haddam Neck	0 5 0 0 0 2 1 3	1 OF 0 4

TITLE (4): Leak on Sensing Line to Reactor Coolant Flow Transmitters Due to Loose Fitting

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)
										0 5 0 0 0
1 1	2 0	9 0	9 0	0 2 9	0 0	1 2	1 8	9 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)	[ ] 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)	[X] 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	
J. Leger, Associate Engineer	2 0 3	2 6 7 - 2 5 5 6

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
X	J C	T B G	P 0 7 0	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)

ABSTRACT

On November 20, 1990, at 0815 hours, with the plant in Mode 1 at 100 percent power, operators observed that the containment radiation monitors indicated an increase in gas and particulate activity. A containment inspection was performed and it was determined that there was a leak at a union on a sensing line to the reactor coolant loop No. 1 flow transmitters. In accordance with Technical Specification 3.0.3 a load reduction was commenced at 1030 hours to place the plant in a mode in which the reactor coolant low flow reactor trip is not required to be operable (below 10 percent power). A loss of flow trip signal for loop No. 1 was inserted at 1315 hours once the 74 percent power level permissive was inactive. The cause of the event was an improperly assembled tubing union. The root cause is unknown. Short term corrective action consisted of inspecting a sample of similar unions and remaking the loose union which was followed by a leak test at system pressure. Long term corrective action consists of performing additional inspections of unions and tubing runs during the next cold shutdown. This event is reportable under 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications.

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BACKGROUND INFORMATION

The Haddam Neck plant is a four loop pressurized water reactor which has three reactor coolant flow channels per loop. When two out of three of the channels sense a low flow condition, as measured by the differential pressure across each steam generator, a low flow reactor trip signal is generated with the loss of flow in one loop above 74 percent power or the loss of flow in two loops between 10 percent and 74 percent power. The three flow transmitters (EIIS Code: FIT) share a common sensing line. The three channels per loop configuration, installed as part of the reactor protection system (EIIS Code: JC) upgrade during the 1989/1990 refueling outage, replaced a one channel per loop configuration. The installation was inspected and leak tested prior to startup.

EVENT DESCRIPTION

On November 20, 1990, at 0815 hours, with the plant in Mode 1 at 100 percent power, Operators observed that the containment radiation monitors indicated an increase in gas and particulate activity. At approximately 0930 the containment sump level was observed to be increasing slightly. At approximately 1020 hours a containment inspection was performed and it was determined that there was a leak at a 1/2" union on the low side sensing line to the reactor coolant loop No. 1 flow transmitters. Since the action statement for Technical Specification Table 3.3-1 (Item 7) does not address the inoperability of more than one flow channel, Technical Specification 3.0.3 was implemented and a load reduction was commenced at 1030 hours to place the plant in a mode in which the low flow reactor trip is not required (below 10 percent power). A reactor coolant loop No. 1 loss of flow trip was inserted at 1315 hours once the 74 percent power level permissive was inactive. At approximately 1400 hours, a repair team entered the containment building and isolated the loop No. 1 flow sensing line. The union was tightened and a satisfactory leak test was performed at system pressure. The loss of flow trip was removed and the channel was placed back in service at 1444 hours. The load reduction was terminated at 1446 hours at 39 percent power.

CAUSE OF THE EVENT

The cause of the event was an improperly assembled tubing union. The union required one full turn to remake the connection. The root cause is unknown.

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

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

SAFETY ASSESSMENT

This event is reportable under 10CFR50.73(a)(2)(i)(B) as a condition prohibited by the plant's Technical Specifications in that the action statement for Technical Specification Table 3.3-1 (Item 7) does not address the inoperability of more than one flow channel; therefore, it was necessary to implement Technical Specification Section 3.0.3.

The reactor protection system will initiate a low flow reactor trip for loss of flow in one coolant loop above 74 percent reactor power, or loss of flow in two loops below 74 percent reactor power but above 10 percent reactor power. Loss of flow in a loop is detected by three different methods:

1. Reactor coolant pump (RCP) breaker position.

2. Undervoltage on the bus supplying the reactor coolant pumps.

3. Differential pressure across each steam generator.

The loop No. 1 flow instruments were placed in trip after reactor power was reduced below the 74 percent interlock.

During the incident, loop No. 1 flow indicated approximately 6 percent above actual flow due to the leak in the low pressure side flow sensing line. This instrument error delays a low flow reactor trip as detected by loop No. 1 low steam generator differential pressure by less than 0.1 seconds. The outcome of the single loop loss of flow safety analysis is not affected by this increased time delay.

Response to a multiple loop loss of flow accident is not affected by this incident since the flow instruments in loops Nos. 2, 3, and 4 were not affected. This is the most probable loss of flow accident.

Response to a loss of flow in loop No. 1 caused by bus undervoltage or by opening of the RCP breaker was not affected.

Based on the above, this event has negligible safety significance.

CORRECTIVE ACTION

Immediate corrective action was to commence a load reduction and insert a loss of flow trip on reactor coolant loop No. 1. Short term corrective action consisted of isolating the leak, remaking the union connection, successfully performing a leak test and inspecting a sample of other similar unions for the same problem. No other unions were found to be leaking. Long term corrective action consists of performing additional inspections of unions and tubing runs during the next cold shutdown.

ADDITIONAL INFORMATION

Component: 1/2 inch union fitting
Manufacturer: Parker
Model No: CPI 8-8 HBZ SS
EIIS Code: TBG

PREVIOUS SIMILAR EVENTS

None