



CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

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

June 20, 1991
Re: 10CFR50.73(a)(2)(ii)

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 50-213/85-003-01

Gentlemen:

This letter forwards the Licensee Event Report 85-003-01, required to be submitted, pursuant to the requirements of Connecticut Yankee Technical Specifications.

Very truly yours,

John P. Stetz
Station Director

JPS/mlg

Attachment: LER 50-213/85-003-01

cc: Mr. Thomas T. Martin
Regional Administrator, Region I
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J. T. Shedlosky
Sr. Resident Inspector
Haddam Neck

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LICENSEE EVENT REPORT (LER)

FACILITY NAME (1) **Haddam Neck** DOCKET NUMBER (2) **0 5 0 0 0 2 1 3 1** OF **0 5**

TITLE (4) **Low Reactor Coolant Flow Trip Setpoint**

EVENT DATE (6)			LER NUMBER (8)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)		
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAMES		DOCKET NUMBER(S)
02	12	85	85	003	01	06	20	09			0 5 0 0 0
											0 5 0 0 0

OPERATING MODE (8) **1**

POWER LEVEL (10) **1 0 0**

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5. (Check one or more of the following) (11)

<input type="checkbox"/> 20.402(b)	<input type="checkbox"/> 20.405(e)	<input type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(b)
<input type="checkbox"/> 20.405(a)(1)(iii)	<input type="checkbox"/> 50.36(a)(1)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)	<input type="checkbox"/> 73.71(c)
<input type="checkbox"/> 20.405(a)(1)(iv)	<input type="checkbox"/> 50.36(a)(2)	<input checked="" type="checkbox"/> 50.73(a)(2)(v)	OTHER (Specify in Abstract below and in Text, NRC Form 365A)
<input type="checkbox"/> 20.405(a)(1)(v)	<input type="checkbox"/> 50.73(a)(2)(i)	<input type="checkbox"/> 50.73(a)(2)(vi)(A)	
<input type="checkbox"/> 20.405(a)(1)(vi)	<input checked="" type="checkbox"/> 50.73(a)(2)(ii)	<input type="checkbox"/> 50.73(a)(2)(vii)(B)	
<input type="checkbox"/> 20.405(a)(1)(vii)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(viii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME **J. Stanford, I & C Manager** TELEPHONE NUMBER **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	MANUFAC TURE	REPORTABLE TO NRRDS	CAUSE	SYSTEM	COMPONENT	MANUFAC TURE	REPORTABLE TO NRRDS

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15)

MONTH	DAY	YEAR

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

ABSTRACT

While at 100% power, two channels of the low reactor coolant flow trip logic of the Reactor Protection System were discovered to have trip setpoints that were not in accordance with technical specifications. The technical specification states that the low reactor coolant flow trip shall be set at greater than or equal to 90% of normal loop flow.

The immediate corrective action taken was to reduce power below the P8 permissive (84% power) to allow recalibration of affected trip logic relays.

The root cause of the trip setpoint error was inadequate definition of the acceptance criteria in the calibration procedure.

Additional corrective action which was taken included (1) clarification of the acceptance criteria in the Instrument and Control procedures used to set the low flow trip setpoint before these procedures were used again, (2) review of all surveillances to determine which procedures must have their acceptance criteria upgraded and clarified, and (3) trending the RCS flow rate.

The purpose of this supplemental report is to document the significant changes that have taken place since 1985 which justify alleviating the need to continue RCS flow rate trending.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Haddam Neck	DOCKET NUMBER (2) 0 5 0 0 0 2 1 3 8 5	LER NUMBER (8)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		0 0 3	0 1	0 2	OF	0 5

TEXT (if more space is required, use additional NRC Form 306A's) (17)

BACKGROUND

On February 12, 1985, while the reactor was at 100% of full power, it was discovered that two channels of a portion of the Reactor Protection System (RPS) (EHS System Code JC) had incorrect trip setpoints. This portion was the low reactor coolant flow reactor trip logic (EHS System Code JC). The problem was identified when a review of the indicating differential pressure across the Reactor Coolant System (RCS) of the steam generators (a measure of RCS flow) was not consistent with the low flow trip setpoints.

REPORTABILITY

This event is reportable because:

1. The reactor was operating at power with portion of the Reactor Protection System outside the design basis of the plant (50.73(a)(2)(ii)).
2. The absence of clearly stated acceptance criteria in the plant procedure used to calibrate the low reactor coolant flow trip could have prevented the fulfillment of the safety function of a reactor trip in the time frame analyzed during a total loss of Reactor Coolant Pump (RCP) flow (50.73(a)(2)(v)).
3. There were incorrect setpoints in two channels of the subject RPS which are designed to shut down the reactor (50.73(a)(2)(vii)).

RCS LOGIC DESCRIPTION

Reactor protection for the loss of reactor coolant flow is provided by the low differential pressure across the RCS side of the steam generator. Each reactor coolant loop is instrumented with one differential pressure transmitter and three trip relays. Above reactor permissive P8 (84% of full power), low differential pressure indicated by two out of three trip relays in one RCS loop is needed to satisfy the trip logic. Above 10% power (P7) and below 84% power (P8), low differential pressure in two RCS loops is needed to complete the trip logic.

During three RCS loop operation, (three-loop licensed power level is 65% of four-loop licensed power), loss of flow in one operating (at power) loop will complete the trip logic.

TECHNICAL SPECIFICATION

Technical Specification 2.4 states that the low reactor coolant flow trip shall be set no less than 90% of normal loop flow. This value is applicable to both four and three RCS loop operation.

The licensee assumes that the design flow rate was used in the accident analyses. Flow measurements performed at the plant indicate that actual flow is approximately 3% higher than design. Since the Technical Specification states "90% of normal loop flow", the licensee has interpreted this to mean "indicated flow" not "design flow". Therefore, in order to be in accordance with the Technical Specifications, the trip setpoint will be set at 90% of indicated mass flow (81% of the indicated full flow differential pressure).

FACILITY NAME (1) Haddam Neck	DOCKET NUMBER (2) 0 5 0 0 0 2 1 3 8 5	LER NUMBER (8)			PAGE (3)		
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER			
			0 0 3	0 1	0 3	OF	0 5

TEXT IF more space is required, use additional NRC Form 306A's (17)

FAILURE CAUSE EVALUATION

The root cause of the trip setpoint is inadequate definition of the acceptance criteria in the calibration procedure.

THREE-LOOP OPERATION

When the plant is in three-loop operation, the loop differential pressure is higher than for four-loop operation. This increase in differential pressure is approximately 11%.

Prior to February 12, 1985 the low reactor coolant flow trip setpoint was not changed from the four-loop setting when operating with three loops. In subsequent three-loop operation, the low flow trip setpoint will be recalibrated for the three-loop mode of operation.

EVALUATION OF THE DATA

A summary of the RCS data is presented below.

Loop	Differential Pressure (psi)	Trip Setpoint (psi)	
		Before 2-12-85	After 2-12-85
1	15.6	13.2	13.2
2	17.9	13.2	14.6
3	18.0	13.5	14.6
4	16.0	13.7	13.7

Prior to February 12, 1985, the trip setpoints for loops 2 and 3 were set at 86% and 87% of normal loop flow. Recalibration restored the trip setpoints to 90% of normal loop flow. Trip setpoints for loops 1 and 4 were set at 92% and 93% of normal loop flow. No recalibration was required for loops 1 and 4.

ASSESSMENT OF SAFETY CONSEQUENCES

Three loss of flow transients are evaluated in the safety analysis. The impact of the non-conservative low flow trip setpoints is presented below.

1. Loss of Flow RCPs From 100% Power With Four Loops Operating

The analysis assumes the reactor trip signal is generated when any loop flow reduces to 85% of its initial value (90% setpoint minus 5% uncertainty). Since the low flow trip channels for loops 1 and 4 were conservatively set at 92% and 93% of normal loop flow, and actuation of a reactor trip is a 1 out of 4 logic, there is no impact on this analysis caused by the trip setpoint error. Also, loss of flow in all four loops would most probably be caused by station electrical disturbance which actuated separate redundant reactor trip relays.

LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

FACILITY NAME (1) Haddam Neck	DOCKET NUMBER (2) 0 5 0 0 0 2 1 3	LER NUMBER (6)			PAGE (3)	
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		8 5	0 0 3	0 1	0 4	OF 0 5

TEXT (If more space is required, use additional NRC Form 3054's) (17)

2. Loss of Three RCPs With Three Loops Operating

Although the non-conservative low flow trip setpoint is a DNB penalty, the DNBR margin more than compensates for this penalty. There is no impact of safety significance caused by the trip setpoint error.

3. Loss of One RCP at 75% Power With Four Loops Operating

Since the loss of flow in one loop initiated from 84% power is not expected to result in DNBR less than 1.3, even in the case of no reactor trip there is no effect of safety significance caused by the trip setpoint error.

4. Effect on Other Loss of Flow Events

There is the potential for the setpoint error to have an effect on the loss of one or two pumps while operating at 100% power in four-loop operation and at 65% power while in three-loop operation. These events are not part of the Connecticut Yankee design basis. The delay in the reactor trip that will take place as the result of the setpoint error has not been analyzed. However, even though the reactor trip would be slightly delayed due to the lower trip setpoint (actual versus analysis), the loss of flow in four loops while in four loop operation and the loss of flow in three loops while in three-loop operation events are expected to bound the results of these above cases.

5. Redundant Trips

In addition to the low differential pressure reactor protection circuit, reactor trip signals are generated from the opening of the reactor coolant pump breaker and low voltage on the generator bus supplying the RCP. These additional reactor protection circuits were operational during the discovery of the event.

CORRECTIVE ACTION

The immediate corrective action taken at the time of discovery was to declare channels for loops 2 and 3 inoperable and reduce power to permissive P7, 10% of rated thermal power. When the core power was reduced below permissive P8 (84% of rated thermal power), the Instrument and Control Department recalibrated the low flow trip in loop 2. After successful calibration and test of this loop, loop 3 was then recalibrated. Power reduction was terminated at that time because the minimum number of operable channel logic for that power level was satisfied. Following successful recalibration and test of loop 3, power ascension to 100% commenced.

Additional corrective action which was taken included (1) clarification of the acceptance criteria of the Instrument and Control procedures used to set the low flow trip setpoint before these procedures were used again, (2) review of all surveillances to determine which procedures must have their acceptance criteria upgraded and clarified, and (3) trending the RCS flow rate.

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		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER		
		8 5	0 0 3	0 1	0 5	OF 0 5

TEXT (If more space is required, use additional NRC Form 306A's) (17)

Since 1985, five significant changes have taken place at Haddam Neck which alleviate the need to continue reactor coolant system flow rate trending. Each is discussed below.

1. Loop/channel setpoint and uncertainty analyses have been developed for all Reactor Protection System functions. All calculations and analyses have been developed in accordance with applicable industry standards.
2. Two of three phases of the Reactor Protection System upgrade have been completed. The upgrade includes total replacement of the Reactor Coolant System flow monitoring hardware. The new configuration replaces a one channel per loop configuration with three flow channels per loop. Each channel provides inputs to two redundant reactor trip matrix trains. Unlike its predecessor, the upgraded system allows trip setpoint verification while at power.
3. Station Technical Specifications have been upgraded to the Westinghouse standard format, amendment 125, issued April 26, 1990. The Reactor Protection System setpoints, allowable drift and surveillance requirements are clearly defined. The Reactor Coolant System low flow setpoints are verified at least once per six weeks on a staggered basis.
4. A major effort was undertaken to upgrade all station procedures. This project was completed in 1989. Inter-department round table discussions and INPO guidelines were utilized to adequately define procedure structure and address human factor considerations. Particular attention was given to acceptance criteria.
5. A comprehensive Reactor coolant System flow measurement program was established in 1986. A detailed heat balance is used to measure Reactor coolant System volumetric flow following each refueling outage. The effects of measurement uncertainty are considered as specified in NUREG/CR-3659. A Reactor Coolant System flow surveillance is conducted at least once per 12 hours while operating at power. Both the refueling and shiftly surveillances have shown Reactor Coolant System flow rate to be invariant.

LER 50-213/85-003-00 identified the need to trend Reactor Coolant System flow rate to preclude the possibility of the process signal drifting high such that the trip setpoint would be non-conservative. Haddam Neck believes this event is no longer a concern for the reasons outlined above. Accordingly, the Reactor Coolant System flow rate trending has been discontinued.