NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Haddam Neck	0 5 0 0 0	1 OF 0 4

TITLE (4)
Failure of Automatic Initiation of Auxiliary Feedwater

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)
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0 9	1 0	8 5	8 5	—	0 0 9	—	0 0	1 0	1 0	85		0 5 0 0 0

OPERATING MODE (9)	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
1	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10): 1 0 0	20.405(a)(1)(i)	50.36(c)(1)	X 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
Joseph H. Evola, 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	B A	F S V	A 6 0 9	Y							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO				

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

Abstract

During the first performance of a preventive maintenance procedure in which each of the auxiliary feedwater auto-initiation solenoid operated valves (SOVs) is cycled, one of the SOVs failed to change state upon deenergization. (This procedure was developed per manufacturer's recommendations as a result of our November 2, 1984, event in which two feedwater bypass valves failed to open automatically. The cause was determined to be sticking SOVs). This failure would have prevented one of the four feedwater bypass valves from automatically opening upon receipt of an Auxiliary Feedwater System auto-initiation signal. It did not and would not have prevented normal valve control using the main control board controller.

After being tapped once, the stuck SOV changed state. It was then satisfactorily cycled several times.

Using the same preventive maintenance procedure, the six SOVs were satisfactorily cycled on September 11, 13, 16, and 19 and once per week since then. Periodic cycling will continue until either the SOVs are replaced with an upgraded model or the cause of the sticking problem on the existing SOVs is determined and corrected.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Haddam Neck	0 5 0 0 0 2 1 3	8 5	— 0 0 9	— 0 0	0 2 OF 0 4

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

Event Description

On the afternoon of September 10, 1985, with the plant in Mode 1 at 100 percent power, one of the six solenoid-operated valves (EIIS Component ID "FSV") in the Auxiliary Feedwater System (AFW) (EIIS System ID "BA") automatic initiation scheme failed to operate. This failure was detected during the first performance of the preventive maintenance procedure developed to periodically cycle each of the six SOVs to prevent a sticking problem similar to that described in our LER No. 85-005.

This SOV failure would have prevented the number 1 feedwater bypass valve from opening automatically (normal valve control from the main control board was not affected).

The six SOVs are normally energized. Upon receipt of an auto-initiation signal, they are deenergized and change state to open six air-operated valves--two to start the two auxiliary feedwater pumps (EIIS Component ID "P") and four to open the four feedwater bypass valves (EIIS Component ID "FCV").

Reportability

This event is reportable in accordance with:

1. 10CFR50.73(a)(2)(i)(B). The number 1 feedwater bypass valve would not have actuated to its correct position upon receipt of an auxiliary feedwater auto-initiation signal. This condition is implicitly prohibited by Technical Specification Sections 3.8 and 4.8.

2. 10CFR50.73(a)(2)(v). Although operator action could have provided auxiliary feedwater to the number 1 steam generator, the system would have failed to automatically remove residual heat.

Failure Cause

The failure was a stuck SOV which, upon deenergization, failed to change state. The SOV is an ASCO (NPRDS Manufacturer Code A609) model number NP-8320A-185E. We have not yet determined the exact cause of the malfunction.

Corrective Action

Using the same preventive maintenance procedure, the six SOVs have been cycled at least once per week since September 10. Periodic cycling will continue until either the SOVs are replaced with upgraded models (not subject to a sticking problem) or the specific cause of the existing sticking problem is determined and, if possible, corrected.

The licensee considers that periodic cycling will preclude any further sticking problem because:

1. In all tests subsequent to the failure, the six SOVs have operated satisfactorily.

2. The SOV manufacturer recommends periodic cycling to prevent sticking. They have not defined "periodic" or recommended a specific cycling interval.

The latter point is very important, because prior to September 10, the last time these six SOVs were cycled was on May 16, 1985. September 10 was the first performance of a new preventive maintenance (cycling) procedure. This procedure was written after our LER number 85-005 (April 1, 1985), but was not approved until August 29, 1985, due to a review of plant technical specification requirements of the safety evaluation for the procedure.

References

1. LER number 85-005-00, dated April 1, 1985, which describes the similar failure of two of the six SOVs in the auxiliary feedwater auto-initiation system.

2. J. F. Opeka letter to J. A. Zwolinski, dated September 20, 1985, which discusses a number of issues regarding the auxiliary feedwater system. This letter discusses this event as well as a wiring deficiency discovered on the same date. The wiring deficiency is not reportable under 10CFR50.72 or 50.73 because overall system operability was not and is not affected.

Generic Implications

This sticking SOV problem could impact the operability of any system using the same model valve in a similar application (normally energized, deenergized to "fail" to safety position). No other systems at our plant use these SOVs in such an application.

Safety Assessment

Auto-Initiation Logic

Auto-initiation of AFW logic is actuated when two-out-of-four steam generator low level alarms occur (credited in the safety analysis) or both main feed pump breakers open (not credited in safety analysis). In the event that the auto-initiation logic fails to open the steam generator bypass valves, as in the November 2, 1984, and September 10, 1985, occurrences, the control room operator can take manual control of those valves and maintain steam generator level. Plant procedures require the control room operator to commence feeding the steam generators via the bypass valves to restore levels to 25-50% on the narrow range level.

Four-Loop Operation

Following a reactor trip from four-loop operation, reactor coolant pumps (RCP) in loops 1 and 3 trip by design. Given a total loss of feedwater, most of the decay heat is removed by the two active steam generators in the loops with the running RCPs (loops 2 and 4). For four-loop operation, the present design basis analysis show that injecting AFW into all four steam generators produces acceptable results. The effects of having fewer than four operating steam generators with both AFW pumps have not been specifically analyzed. This is consistent with the single failure assumption of loss of one AFW pump made in the Facility Description Safety Analysis (FDSA)--other potential single failures were not considered. These other potential single failures are being evaluated as part of the FDSA Chapter 10 reanalysis effort scheduled to be completed by June 1986.

The case of no AFW flow to all four steam generators for 10 minutes has been analyzed and found to be acceptable if the operator takes action at that time to initiate AFW flow. Since operator action (manually initiating AFW flow) could be performed within 10 minutes, the subject event is not considered to be unanalyzed.

Three-Loop Operation

For three-loop operation, the present analyses show that, with one AFW pump operating, injecting AFW into the three operable steam generators produces acceptable results. The effects of having fewer than three operable steam generators with both AFW pumps operating have not been specifically analyzed. This is consistent with the single failure assumption of loss of one AFW pump made in the FDSA--other potential single failures were not considered. These other potential single failures are being evaluated as part of the FDSA Chapter 10 reanalysis effort scheduled to be completed by June 1986.

The most severe loss of feedwater event in three-loop operation occurs when the isolated loop contains an RCP that would normally be running following reactor trip (i.e., loop 2 or 4). When this is the case, there is forward flow through the loop with the operating RCP and reverse flow through loops 1 and 3. As a result, most of the decay heat load is placed on the single steam generator in the loop with the running RCP. Because of the heat load, the inventory in that steam generator boils down rapidly. The inventories in the steam generators in loops 1 and 3, (containing the idle RCPs) boil down much more slowly. Since the auto-initiation signal for AFW utilized two-out-of-three logic, auto-initiation was predicted to occur later than 10 minutes. As a result, the auto-initiation of AFW results in more severe consequences (PORVs were predicted to lift) than a similar case assuming operator action to start AFW at 10 minutes. In order to meet the licensee's acceptance criterion that the PORVs should not lift for this event, it is necessary for the operator to manually initiate AFW prior to 10 minutes. Since operator action (manually initiating AFW flow) could be performed within 10 minutes, the subject event is not considered to be unanalyzed.

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT

RR#1 • BOX 127E • EAST HAMPTON, CONN. 06424

October 9, 1985

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 50-213/85-009-00

Gentlemen:

This letter forwards the Licensee Event Report 85-009-00, required to be submitted within thirty days, pursuant to the requirements of Connecticut Yankee Technical Specifications.

Very truly yours,

Richard H. Graves
Station Superintendent

RHG:JHE/ssg
Attachment: LER 85-009-00

cc: Dir. T. E. Murley, Region I

IE22
1/1