

**North  
Atlantic**

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The Northeast Utilities System

NYN- 96014

February 23, 1996

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

Attention: Document Control Desk

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

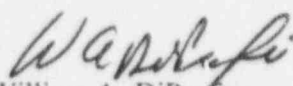
Subject: Licensee Event Report (LER) No. 96-001-00, "Automatic Reactor Trip"

Gentlemen:

Enclosed, please find Licensee Event Report (LER) No. 96-001-00 for Seabrook Station. This submittal documents an event which occurred on January 27, 1996. This event is being reported pursuant to 10CFR50.72 (b)(2)(ii).

Should you require further information regarding this matter, please contact Mr. Anthony M. Callendrello, Licensing Manager, at (603) 474-9521, extension 2751.

Very truly yours,

  
William A. DiProffio  
Station Director

WAD/JMPjr:sm

Enclosures: NRC Forms 366/366A

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United States Nuclear Regulatory Commission  
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February 23, 1996  
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cc: Mr. Thomas T. Martin  
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United States Nuclear Regulatory Commission  
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## LICENSEE EVENT REPORT (LER)

(See reverse for required number of  
digits/characters for each block)ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY  
INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE  
INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY.  
FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND  
RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION,  
WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-  
0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)

Seabrook Station

DOCKET NUMBER (2)

05000443

PAGE (3)

1 OF 4

TITLE (4)

Automatic Reactor Trip

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 NAME	DOCKET NUMBER
01	27	96	96	-- 001	-- 00	02	27	96		05000
OPERATING MODE (9)		1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)		100	20.2201(b)			20.2203(a)(2)(v)			50.73(a)(2)(viii)	
			20.2203(a)(1)			20.2203(a)(3)(i)			50.73(a)(2)(ii)	
			20.2203(a)(2)(i)			20.2203(a)(3)(ii)			50.73(a)(2)(iii)	
			20.2203(a)(2)(ii)			20.2203(a)(4)			50.73(a)(2)(iv)	
			20.2203(a)(2)(iii)			50.36(c)(1)			50.73(a)(2)(v)	
			20.2203(a)(2)(iv)			50.36(c)(2)			50.73(a)(2)(vii)	
									<input checked="" type="checkbox"/> OTHER	
									Specify in Abstract below or in NRC Form 366A	

## LICENSEE CONTACT FOR THIS LER (12)

NAME

Mr. Anthony M. Callendrello, Licensing Manager

TELEPHONE NUMBER (Include Area Code)

603-474-9521 extension 2751

## 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

## 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 15 single-spaced typewritten lines) (16)

On January 27, 1996 at 0945 an automatic reactor trip occurred at 100% power. The reactor trip occurred due to high Pressurizer pressure. The turbine combined intercept valves closed in response to the insertion of a fast close signal due to position mismatch. Concurrently, the turbine control valves closed due to the insertion of a slow close signal. The resulting turbine load rejection caused the Reactor Coolant System temperature and pressure to increase. This event was reported to the NRC pursuant to 10CFR50.72(b)(2)(ii), actuation of the Reactor Protection System and Engineered Safety Feature system. There were no adverse safety consequences as a result of this event.

The turbine combined intercept valve and control valve closure was caused by a failure of a circuit card in the turbine speed control circuit in the Electro-Hydraulic Control system. This resulted in a large speed error signal and valve close signal to the combined intercept valves and control valves.

The root cause of this event was determined to be random failure of an Electro-Hydraulic Control speed control circuit card.

North Atlantic has taken immediate corrective actions including replacing both of the Electro-Hydraulic Control speed control circuit cards. Long term actions include performing several evaluations related to the design and handling of Electro-Hydraulic Control circuit cards.

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Seabrook Station	05000443	96	-- 001	-- 00	2 OF 4

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

On January 27, 1996, at 0945, with the unit operating at 100% power, the low pressure turbine (TRB) combined intercept valves (CIVs) closed in response to the insertion of a fast close signal due to position mismatch. The main turbine control valves (CVs) concurrently closed on insertion of a slow close signal. The CIV and CV closure was initiated by a false high speed error signal (valve closure demand) which resulted in a CIV position error of greater than five percent causing a CIV fast closure.

The CIV and CV closure was initiated by failure of one of two "low value gate" circuit cards in the turbine speed control circuit in the Electro-Hydraulic Control (EHC) (TG) system. This resulted in a large speed error signal and valve close signal to the CIVs and CVs.

Following closure of the CVs and CIVs, the resulting turbine load rejection caused the Reactor Coolant System (RCS) (AB) temperature and pressure to increase. An automatic reactor (RCT) trip occurred due to high Pressurizer (PZR) pressure. Both Pressurizer Power Operated Relief Valves (PORVs) cycled open for a brief period in response to the primary side transient. Both valves functioned normally and exhibited no evidence of leakage after final closure. The primary safety valves were not challenged during this event. The highest primary system pressure was 2,387 psig. The condenser steam dump valves opened as designed, reducing the RCS temperature to the no load value and the transient was terminated rapidly.

A transient associated with the EHC system for the main turbine preceded the reactor trip and several associated alarms. The highest priority alarm was "EHC MALFUNCTION". The Unit Shift Supervisor (USS) acknowledged the EHC alarm report from the Control Room Operator (CRO). The control rods were observed stepping inward. The operating crew determined that the turbine load was decreasing. The load quickly went to zero Mwe and the CRO promptly reported this to the USS. The USS immediately ordered a reactor trip. Before the Senior Control Room Operator (SCRO) manipulated the reactor trip switch, the reactor automatically tripped on high pressurizer pressure. The turbine tripped approximately one-tenth of a second after the reactor trip.

The following Engineered Safety Features (ESF) actuations occurred immediately following the reactor trip, as expected.

Emergency Feedwater (EFW) on Steam Generator Lo-Lo Level

Feedwater Isolation (FWI) on Low  $T_{avg}$  Coincident with a Reactor Trip

The USS announced the reactor trip and entry into Emergency Response Procedure, "Reactor Trip And Safety Injection." The crew performed their immediate actions, continued with the procedure and concluded no safety injection had occurred, and was not required. The USS announced a transition from the Emergency Response Procedure to the Reactor Trip Response Procedure. The following plant conditions were noted following the reactor trip.

#### MAIN GENERATOR HYDROGEN LEAK

At 1223 a report was sent to the Control Room from the Work Control Coordinator (WCC) about a possible hydrogen leak in the Turbine Building. The USS immediately dispatched the Fire Brigade Leader (FBL) and the duty Fire Fighter to investigate the potential leak. The operating crew verified all available Turbine Building roof fans were operating to disperse the hydrogen. The FBL called the Control Room and confirmed that the leak was hydrogen from a Main Generator Hydrogen Cooler flange.

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

The Shift Manager reviewed the Emergency Response Manual Procedure, "Classification Of Emergencies. The Shift Manager discussed the potential for ignition of the hydrogen with the FBL. Because the ceiling of the Turbine Building is 50 feet above the top of the generator and the roof fans were running, the FBL indicated that it was unlikely that any pockets of hydrogen would accumulate and that ignition was unlikely.

The Shift Manager concluded that the hydrogen leak would not adversely impact station operations and informed the crew of his decision. At approximately 1309 the Shift Manager determined that the plant conditions did not meet the initiating criteria for an Unusual Event declaration. He then instructed the USS to use the emergency evacuation portion of the generator hydrogen system to quickly lower the pressure of the generator to minimize hydrogen leakage. The operating crew then used the normal procedure for purging the generator with CO<sub>2</sub> to remove all hydrogen from the generator.

The generator hydrogen cooler flange leak was determined to be a result of a gradual relaxation of the flange bolt torque and was not related to the plant trip.

**CONDENSATE/FEEDWATER PRESSURE TRANSIENT**

The modifications performed on the main feed pump (MFP) speed controls and the FRV and FWIV stroke times following the June 18, 1995 unit trip were very effective. The pressure transient resulting from the feedwater isolation (FWI), was reduced in magnitude and duration in comparison to previous events.

The feedwater heater tube side thermal relief valves, were not damaged during this event as they had been on previous unit trips.

Safety Consequences

There were no adverse safety consequences as a result of this event. Plant equipment functioned as designed. Plant equipment functioned as designed and operator actions were reviewed and determined to be conservative and correct.

Root Cause

A Root Cause evaluation was performed as an immediate response to this event. The evaluation concluded that the circuit card failure appears to be random in nature. This is based on detailed troubleshooting of the two circuit cards removed and a review of the industry operating experience.

Corrective ActionsImmediate

Following the plant trip, the immediate corrective actions included determining the source of the failure. A General Electric (GE) representative, assisted by North Atlantic Technicians and System Engineers, performed a comprehensive troubleshooting and repair effort of the EHC system. The troubleshooting isolated the failure to the low value gates for the speed error EHC circuit cards. The failure was duplicated during initial troubleshooting efforts, but later cleared and could not be repeated during later troubleshooting efforts. It was not finally determined which of the two circuit cards was the source of the fault therefore, both circuit cards were replaced.



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

A review was performed of the maintenance recommendations in the General Electric (GE) Manual for the EHC system and it was confirmed that there is no requirement or recommendation for replacing printed circuit cards or components as a preventive maintenance activity.

Prior to the plant restart, the main generator hydrogen leaks were repaired.

Long Term

Further investigation of the circuit cards that were removed from the system will be performed to attempt to identify the failed card, the exact component on the card that failed and the failure mechanism. This information will be used to determine if further corrective action is warranted for circuit cards in other protective systems in the plant.

North Atlantic is evaluating an upgrade to the Mark V EHC control system during the sixth refueling outage. As appropriate, the scope of EHC system testing and calibration will be enhanced during the fifth refueling outage.

Plant Conditions

At the time of this event, the plant was in operational Mode 1, at 100% power, with a Reactor Coolant System temperature of 587 degrees Fahrenheit and a pressure of 2235 psig.

Prior Events

There have been three EHC related plant trips at Seabrook Station since power ascension testing. The first event occurred on July 5, 1990 (LER 90-018) and was related to excessive vibration of EHC pressure switches due to their mounting locations. The second event occurred on August 22, 1990 (LER 90-022) and was related to a loss of voltage on the EHC bus during troubleshooting activities in the EHC cabinet. The third event occurred on June 2, 1991 and was related to a stuck open oil trip solenoid valve during weekly surveillance testing. Based on the evaluations of these events and the information related to the current trip, there is no apparent relationship to the current event.