North Atlantic

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The Northeast Utilities System
July 26, 2000

Docket No. 50-443

NYN-00062

CR 00-07526

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Seabrook Station
Licensee Event Report (LER) 00-004-00
Manual Reactor Trip and ESF Actuation
Due to a Steam Generator Low-Low Level Indication

Enclosure 1 contains Licensee Event Report (LER) 00-004-00 for an event that occurred at Seabrook Station on June 26, 2000. This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv). Commitments associated with this letter are contained in Enclosure 2.

Should you require further information regarding this matter, please contact Mr. James M. Peschel, Manager-Regulatory Programs at (603) 773-7194.

Very truly yours,

NORTH ATLANTIC ENERGY SERVICE CORP.

Ted C. Feigenbaum
Executive Vice President and
Chief Nuclear Officer

cc: H. J. Miller, NRC Regional Administrator
R. M. Pulsifer, NRC Project Manager, Project Directorate 1-2
R. K. Lorson, NRC Senior Resident Inspector

IE22

ENCLOSURE 1 TO NYN-00062

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Seabrook Station	05000443	1 of 3

TITLE (4)

Manual Reactor Trip and ESF Actuation Due to a Steam Generator Low-Low Level Indication

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
06	26	00	00	004	00	07	26	00	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9): 1

POWER LEVEL (10): 100

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

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	X 50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
James M. Peschel, Manager - Regulatory Programs	(603) 773-7194

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
B	JD	CBD	W120	Y					

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

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On June 26, 2000 at approximately 1851 with the reactor operating at 100 percent power (mode 1), the "A" Main Feedwater Pump (1FW-P-32A) tripped. The trip of 1FW-P-32A resulted in a setback to 55 percent power. Approximately 1 minute and 25 seconds after the trip of 1FW-P-32A, plant operators received a low level alarm on the "C" steam generator due to the inability of the "B" Main Feedwater Pump (1FW-P-32B) to maintain feed water inventory to the steam generators in proportion to the reactor power level. Shortly thereafter, a manual reactor trip was initiated by plant operators as steam generator levels approached the low-low level trip setpoint. This event (Event Number 37117) was initially reported to the Nuclear Regulatory Commission on June 26, 2000 at 1923 pursuant to the requirements of 10 CFR 50.72(b)(2)(ii). This event is additionally reportable pursuant to the requirements of 10 CFR 50.73(a)(2)(iv).

1FW-P-32A tripped as a result of a momentary steam generator High-High level signal originating from the solid state protection system (SSPS). A number of corrective actions have been completed and identified. There were no adverse safety consequences as a result of this event. Plant equipment functioned as designed and operator actions were determined to be appropriate. This is the first event of this type reported within the past two years.

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

I. Description of Event

On June 26, 2000 at approximately 1851 with the reactor operating at 100 percent power (mode 1), the "A" Main Feedwater Pump [SK] (1FW-P-32A) tripped. The trip of 1FW-P-32A resulted in a setback to 55 percent power. Approximately 1 minute and 25 seconds after the trip of 1FW-P-32A, plant operators received a low level alarm on the "C" steam generator due to the inability of the "B" Main Feedwater Pump (1FW-P-32B) to maintain feed water inventory to the steam generators [SB] in proportion to the reactor power level. Shortly thereafter, a manual reactor trip was initiated by plant operators (at approximately 80 percent reactor power) as steam generator levels approached the low-low level trip setpoint.

The Reactor Control System [JD] functioned as expected with the control rods fully inserting. The emergency feedwater system [BA] actuated as expected. No primary or secondary safety valves lifted during the transient. The required sources of off-site power remained operable during the event.

This event (Event Number 37117) was reported to the Nuclear Regulatory Commission on June 26, 2000 at 1923 pursuant to the requirements of 10 CFR 50.72(b)(2)(ii) as a condition that resulted in a manual or automatic actuation the reactor protection system or Engineered Safety Feature. Event Number 37117 was updated on June 27, 2000 at 0015 to document an abnormal post-trip response relating to the failure of the intermediate and source range nuclear instrumentation to energize and to correct the event time. Subsequent review of this event indicates that the intermediate range nuclear instrumentation was energized as expected. This event is additionally reportable pursuant to the requirements of 10 CFR 50.73(a)(2)(iv). This event did not result in a safety system functional failure of equipment needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

II. Cause of Event

1FW-P-32A tripped as a result of a momentary steam generator High-High level signal originating from the solid state protection system [JG] (SSPS). This SSPS signal caused a master slave relay and slave relay to actuate a trip of the main feedwater pump and actuation of an annunciator indicating a main feedwater isolation. This momentary signal did not immediately result in an isolation of the Main Feedwater isolation valves because the control circuit includes a 5-second time delay.

Subsequent testing of the SSPS boards (Universal and Safeguards Output) indicates that the Safeguard Driver Board (A517) had degraded to the point where its output voltage was fluctuating. The voltage drop was long enough to actuate the slave relay.

III. Analysis of Event

The consequences of this event were minimal. A function of the Feedwater system is to maintain the proper steam generator water inventory automatically during power operations. The two main feedwater pumps are 60 percent capacity, variable speed, steam turbine driven horizontally mounted centrifugal pumps. The basic function of the reactor protection circuits associated with low-low steam generator water level is to preserve the steam generator heat sink for the removal of long-term residual heat. The emergency feedwater system actuated

as expected to maintain and restore steam generator levels consistent with plant operating conditions following the reactor trip.

There were no adverse safety consequences as a result of this event. Plant equipment functioned as designed and operator actions were determined to be appropriate to ensure the safety of the plant. At no time during this event was there an impact on the health and safety of plant employees or the public.

IV. Corrective Action

1. A logic test of the "A" train SSPS was performed. This is a self-test of the system using test pulses. The system tested satisfactorily with no problems found.

2. A recorder was installed to monitor voltage at various intermediate points for the circuit in question. The voltage values were as expected.

3. The two SSPS circuit boards were replaced with new boards. The new printed circuit boards were satisfactorily tested prior to installation.

4. A logic test of the "A" train SSPS was satisfactorily repeated.

5. A logic test of the circuit was performed in its entirety to verify a change in state of the subject master relay.

6. The two replaced boards were inspected to determine if transistor contamination could have caused eratic output. No visible evidence of contamination was identified.

7. Failure analysis of the subject Safeguards Driver Board will be performed at a sub-component level to determine the cause of the voltage fluctuation.

V. Additional Information

None.

Similar Events

This is the first event of this type within the within the past two years. This is the fifth reactor trip (manual or automatic) and ESF actuation that has occurred at Seabrook Station due to a steam generator low-low level indication. The previous events were described in LERs 90-025-00, 92-017-00, 93-01-00, and 97-012-00. A review of the causes of these events indicates that they are unrelated to this event.

Manufacturer Data

Westinghouse Universal Printed Circuit Board (Part Number: 6056D21G01)