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)
H. B. Robinson, Unit 2	0 5 0 0 0 2 6 1	1 OF 0 3

TITLE (4)

Reactor Trip - High Steam Generator Level

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 1	1 5	8 6	8 6	— 0 0 2	— 0 0	0 2	1 4	8 6		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9):

POWER LEVEL (10): 7 4 . 5

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)	X 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)	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)
Roy C. Abbott	8 0 3 3 8 3 - 4 5 2 4

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)		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

On January 15, 1986, the Plant was operating at approximately 75% power. At 1026 hours, a turbine trip/reactor trip occurred because of a high level in "A" Steam Generator. At the time of the trip, Plant personnel were performing MST-013 "Steam Generator Water Level Protection Channel Testing." The evolution in progress was to place "A" Feedwater Control in manual. Prior to performing this function, the level in "A" Steam Generator began rising. Feedwater Control for "A" Steam Generator was placed in manual, and an attempt was made to close the valve. This effort was unsuccessful resulting in a trip on high "A" Steam Generator level.

An extensive evaluation of the event identified the cause to be either personnel error or equipment malfunction; neither one of the causes could be eliminated. Therefore, corrective action has been initiated to address both. Specifically, personnel involved will review this event, and the controlling procedure will be improved from a human factors standpoint. Additionally, a loose connection on a test jack in the circuit was corrected, and a toggle switch, which if intermittently defective could have caused the event, was replaced.

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NRC:86-514

☆U.S.GPO:1984-0-454-481/18759

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
H. B. Robinson, Unit No. 2	0 5 0 0 0 2 6 1	8 6	— 0 0 2	— 0 0	0 2 OF 0 3

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

Event Description

At 1026 hours on January 15,1986, the Plant was at approximately 75% power. Two I&C Technicians with one Control Operator were performing MST-013 "Steam Generator Water Level Protection Channel Testing". The evolution in progress was Step 7.3.3 in the procedure which directs the technician to request the operator to place the Feedwater Flow Station Switch for "A" Steam Generator FC-478-F in the manual mode until further notice. While the operator was reviewing this step in the procedure, he noticed that "A" Steam Generator level on LI-474 was at 74% and increasing. In an effort to reduce the increasing flow to "A" Steam Generator, the operator placed "A" Feedwater Regulating Valve (Switch FC-478-F) in manual and attempted to close the valve. This effort was unsuccessful resulting in a trip on high Steam Generator level in "A" Steam Generator.

A discussion of the procedure subsequent to Step 7.3.3 is necessary to better understand the evaluation of the event.

The next step in the procedure, Step 7.3.4, directs the I&C Technician to open the signal injection test jacks cover plate and the cover for toggle switch CT-476 and to check for the annunciator (RTGB) for the open cover plate. The toggle switch is used to disconnect the normal transmitter signal input to the level channel and connect the signal injection test jacks to the channel. Step 7.3.5 instructs the technician next to put toggle switch CT-476 in the TEST position and check for the annunciator (RTGB) for the test switch. Step 7.3.6 involves connecting the test potentiometer to the test jacks.

Review of the strip chart recorder for "A" Steam Generator Level (same channel as that to be tested) showed that the indicated level went to zero and then returned to about 30% just prior to the trip. This is the response that would be expected if the toggle switch were put in the TEST position (Step 7.3.5) followed by input of a test signal (Step 7.3.6).

The Technician in the Hagan Racks acknowledged lifting the test jacks cover plate (first part of Step 7.3.4), while waiting for the Operator to go to manual, but stated that he did nothing more. The Technician in the Control Room confirmed receiving this annunciator (only the signal injection test jacks cover is annunciated). The Technician in the Hagan Racks stated he did not open the individual cover for toggle switch CT-476 nor did he put switch CT-476 to the TEST position. The Technician in the Control Room stated that the annunciator signaling CT-476 in the TEST position never actuated. No one else in the Control Room could recall acknowledging the annunciator for CT-476 being in the TEST position.

A loose connection was identified on a test jack on LC-476, which is a comparator in the control loop, but it could not be confirmed that the loose connection caused or contributed to the event.

Following the event, the MST was completed and the level channel returned to service.

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

Cause/Corrective Action

A special Trip Review Committee was established as a part of the Trip Reduction Program to review the circumstances of this event. The Trip Review Committee enlisted the assistance of personnel from the Corporate Nuclear Safety Section to investigate the event using Management Oversight and Risk Tree (MORT) methodology. This investigation included interviews with individuals involved.

This investigation resulted in a reduction of possible causes to two; personnel error or equipment malfunction. Because the statements of the individuals involved suggest an equipment malfunction while the circumstantial evidence suggests a personnel error, elimination of either cause was not possible.

Corrective action, therefore, was initiated to address both possible causes.

The equipment involved was tested and the following corrections were made. A loose connection on a test jack in the circuit was identified and tightened. Toggle Switch CT-476, which if intermittently defective could have caused the event, was replaced. The procedure controlling the work will be revised to provide additional controls to ensure equipment is in the proper configuration prior to testing and to improve the overall quality of the procedure from a human factors perspective. Other Maintenance Surveillance Tests which are similarly performed at power and affect the control circuits (most tests are for protection logic and do not affect the control circuits) will be reviewed for similar improvements.

Additionally, this event will be reviewed with the personnel involved to ensure that they understand the significance of the events that could result from personnel error.

CP&L

Carolina Power & Light Company

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Washington, D. C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
LICENSEE EVENT REPORT 86-002

Dear Sir:

In accordance with 10CFR50.73, Licensee Event Report System, the enclosed Licensee Event Report is submitted. This report fulfills the requirements for a written report within (30) days of a reportable event and is in accordance with the format set forth in NUREG-1022, September, 1983.

Very truly yours,

R. E. Morgan
General Manager
H. B. Robinson S. E. Plant

RCA:jch

Enclosure

cc: INPO
J. N. Grace
H. E. P. Krug

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