REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR: 8705040314 DOC. DATE: 87/04/27 NOTARIZED: NO DOCKET #
FACIL: 50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270

AUTH. NAME	AUTHOR AFFILIATION
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TUCKER, H. B.	Duke Power Co.

RECIP. NAME	RECIPIENT AFFILIATION

SUBJECT: LER 87-002-00: on 870326, Unit 2 tripped from 90% full power on RCS pressure. Caused by failure in "A" loop BTU limit circuit. Stabilize unit at hot shutdown condition & test BTU limit, inputs & connections of modules. W/870427 ltr.

DISTRIBUTION CODE: IE22D COPIES RECEIVED: LTR 1 ENCL 1 SIZE: 6
TITLE: 50.73 Licensee Event Report (LER), Incident Rpt, etc.

NOTES: AEOD/Ornstein: 1cy. 05000270

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	NRR/PMAS/PTSB	1	1	REG FILE 02	1	1
	RES SPEIS, T	1	1	RGN2 FILE 01	1	1
EXTERNAL:	EG&G GROH, M	5	5	H ST LOBBY WARD	1	1
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NOTES:		1	1			

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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)
OCONEE NUCLEAR STATION - UNIT 2	0 5 0 0 0 2 7 0	1 OF 0 4

TITLE (4)
REACTOR TRIP FROM HIGH RCS PRESSURE CAUSED BY EQUIPMENT FAILURE

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 3	2 6	8 7	8 7	— 0 0 2 —	0 0	0 4	2 7	8 7		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N
POWER LEVEL (10): 0 9 6

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)	XX 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	
PHILIP J. NORTH, LICENSING	7 0 4	3 7 3 - 7 4 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	J A			NO					

SUPPLEMENTAL REPORT EXPECTED (14)

YES *(If yes, complete EXPECTED SUBMISSION DATE)* XX NO

EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

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

On March 26, 1987 at 2333 hours, Unit 2 tripped from 96% Full Power on high Reactor Coolant System (RCS) pressure. An open connection in the "A" loop BTU limit section of the Integrated Control System (ICS) caused the ICS to respond by decreasing feedwater flow to the Steam Generators. This decrease in feedwater flow caused a decrease in the heat transfer rate, leading to an increase in RCS temperature and pressure. The unit tripped on high RCS pressure.

The immediate corrective action was to stabilize the unit at hot shutdown conditions. The supplemental corrective action was to investigate the cause of the transient.

The root cause of this event was an intermittent poor connection in the ICS BTU limit loop "A" circuitry.

No Technical Specification limits were exceeded, and there was no release of radioactivity. As such, the health and safety of the public was not affected.

IE22

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
OCONEE NUCLEAR STATION - UNIT 2	0 5 0 0 0 2 7 0	8 7	— 0 0 2	— 0 0	0 2 OF 0 4

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

Background

The Integrated Control System (ICS) provides coordination of Reactor, Steam Generator feedwater control, and Turbine for all operating conditions. The ICS includes four subsections.

A. Unit Load Demand Control

B. Integrated Master Control

C. Steam Generator Feedwater Control (SGFC)

D. Reactor Control

The subsection of the ICS in which the failure occurred, leading to the reactor trip, is the SGFC. The SGFC receives a Feedwater Demand Signal from Integrated Master Control, modifies the signal to obtain desired steam conditions, and applies the modified signal to position Feedwater Flow Controls (feedwater control valves, feedwater pump speed).

The purpose of the BTU limit circuitry is to protect the Main Turbine from moisture in the steam coming from the Steam Generator. The BTU limits limit the demand for feedwater by considering the conditions of:

- Reactor Coolant Flow
- Feedwater Temperature
- Steam Generator Pressure
- Reactor Outlet Temperature (T_{hot})

Description of Occurrence

On March 26, 1987 at 2332, Unit 2 was operating at 96% Full Power; limited by high Steam Generator levels. At approximately 2333 hours, with the Integrated Control System (ICS) in the integrated mode, a Main Feedwater (FDW) System transient occurred. FDW demand for the "A" loop went to approximately zero thus causing a reduction of FDW flow to the "A" Steam Generator. This caused a reduction in the heat transfer capability of the Steam Generator which resulted in increased Reactor Coolant System (RCS) temperature and pressure. At 2333:36 the reactor tripped on high RCS pressure. After the trip, reactor power decreased to a post-trip decay heat level, with all control and safety rods inserted.

Pressurizer level decreased from 220 inches to a post-trip minimum of approximately 90 inches. Level was increased to 190 inches by manually starting the 2A High Pressure Injection (HPI) pump at 2335:26. The 2A HPI pump was secured at 2337:23.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
OCONEE NUCLEAR STATION - UNIT 2	0 5 0 0 0 2 7 0	8 7	— 0 0 2	— 0 0	0 3 OF 0 4

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

The Turbine Header pressure was reduced to approximately 960 psig to reseat one of the Main Steam Relief Valves (MSRVs). The unseated MSRV did not remove enough steam from the Steam Generator (SG) to cause an overcooling of the RCS.

Following the trip, Steam Generator level in SG "B" decreased to approximately 25 inches as expected. The Steam Generator level in SG "A" decreased to approximately 62 inches, then due to an increase in the ICS FDW "A" Demand, the "A" FDW Startup Valve opened and the level in the "A" SG increased to 160 inches. Operations personnel took prompt manual action to stop the refeed.

Cause of Occurrence

The root cause of this incident was a failure in the "A" loop BTU limit circuit. An open circuit intermittently occurred at the input of the T_{Hot} signal into the BTU limit summer module. None of the other inputs to the BTU limit circuitry could have had an open circuit (or failed to 0 volts) and caused the FDW demand signal on loop "A" to go to the value that it did. Personnel checked all of the inputs to the BTU limits, the modules in the BTU limits, and the connections of the modules. No problems could be found. It is concluded that the nature of the failure in the connector is such that examination would not detect the loose connection.

Post-trip the BTU limit and FDW Demand "A" loop circuitry did not perform as expected. Feedwater Master Demand was zero, yet Feedwater Demand on the "A" loop called for about 1.5×10^6 lb/hr. It could not be determined whether the BTU limit circuitry generated this demand. If the BTU limit circuitry did not generate this signal, then it should have over-ridden it and driven "A" loop demand to zero. This failure caused the refeed of the Steam Generator.

Over the past five years one other incident has occurred that was due to a failure of a BTU limit signal. This incident involved the actual failure of an input to the BTU limit circuit. A comparison of corrective actions for this incident vs the earlier one is not applicable because the failure mode is not the same. This incident is considered a recurring event with low frequency.

Corrective Action

The immediate corrective action was to stabilize the unit at hot shutdown conditions.

Supplemental corrective action was to test the BTU limit modules, the BTU limit inputs, and the connections of the modules.

Analysis of Occurrence

The unit was stabilized at hot shutdown after the reactor trip. Integrated Control System control stations were in auto before the trip and Control Operators responded to mitigate the effects of the transient. The pressurizer

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
OCONEE NUCLEAR STATION - UNIT 2	0 5 0 0 0 2 7 0	8 7	0 0 2	0 0	0 4 OF 0 4

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

relief and code safety valves were not challenged. Main steam pressure was properly controlled by the Main Steam Relief Valves and the Turbine Bypass Valve System. RCS temperature dropped from a pre-trip maximum (in the hot leg) of 600°F to a post-trip minimum of 550°F. Technical Specifications' maximum cooldown rate of 50°F per 0.5 hours on RCS temperature (T_{cold}) was not approached. The pressurizer level reached a minimum level of approximately 90". The pressurizer level stabilized at approximately 190 inches with the use of the "2A" High Pressure Injection (HPI) pump. The Reactor Coolant System pressure reached a minimum of about 1856 psig before being brought back up and stabilized at 2180 psig. The Reactor Protective System reacted in accordance with the design features to prevent Reactor Coolant System overpressurization, no Technical Specification limits were exceeded, and there was no release of radioactivity; therefore, the health and safety of the public were not affected.

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HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

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April 27, 1987

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Oconee Nuclear Station, Unit 2
Docket No. 50-270
LER 270/87-02

Gentlemen:

Pursuant to 10CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 270/87-02 concerning a Unit 2 reactor trip from high reactor coolant system pressure at 96% full power.

This report is submitted in accordance with §50.73(a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

Hal B. Tucker

Hal B. Tucker

PJN/162/jgm

Attachment

IE22
1/1

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