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) |
|---|---|---|
| Catawba Nuclear Station, Unit 1 | 0 5 0 0 0 4 1 3 | 1 OF 0 4 |

TITLE (4)

Reactor Trip Due to Low-Low 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 6 | 2 2 | 8 5 | 8 5 | — 0 4 4 — | 0 0 | 0 7 | 2 2 | 8 5 | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9): 1

POWER LEVEL (10): 0 1 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) | X 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) | 50.72(b)(2)(ii) |
| 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 | |
| Roger W. Ouellette, Associate Engineer - Licensing | 7 0 4 | 3 7 3 - 7 5 3 0 |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | 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)


On June 22, 1985, at 1708 hours, a Steam Generator (S/G) Low-Low Level Reactor Trip occurred. Unit 1 was in Mode 1 at approximately 15% power and recovering from a reactor trip which had occurred earlier, at 0105 hours. When the Unit 1 Nuclear Control Operator (NCO) noticed that S/G levels were decreasing, he requested that the Unit 2 NCO increase the speed of the Main Feedwater Pump Turbine (CFPT), but did not communicate to him the mode of speed control. This caused the Unit 2 NCO to mistakenly operate the wrong controller. Before this error could be realized, the reactor tripped on Low-Low S/G C level. Because of the failure to effectively communicate the speed control status, this incident is classified as a Personnel Error.

This incident is reportable pursuant to 10 CFR 50.73, Section (a)(2)(iv), and 10 CFR 50.72, Section (b)(2)(ii).




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The amount of Main Feedwater (CF) supplied to the Steam Generators (S/G's) is regulated by the CF control bypass valves (CF30, 39, 48, and 57) or the CF control valves (CF28, 37,46, and 55), depending on power level. During startup, the low power Feedwater Control System positions the CF bypass control valve to each S/G based on S/G water level and neutron flux measurements. Using this system, the bypass feedwater control valves regulate the CF flow at low reactor power until the steam and feedwater flow signals are sufficient to swap control to the CF control valves (This evolution is performed at approximately 15% reactor power). Also, CF flow delivery will be switched from the auxiliary feedwater (CA) nozzles to the CF nozzles on the S/G's at approximately 17% feed flow during power escalation.

When feeding the CA nozzles at the control valve transfer point of 15% full power, a feedwater pump discharge pressure of approximately 1225 psig and a main steam (SM) /CF pressure differential of approximately 100 psid is required to deliver the appropriate flow to the S/G's. CF pump discharge pressure and the SM/CF pressure differential can be varied by manipulation of CF pump turbine (CFPT) speed. There are several methods of speed control available which can be selected by the use of a mode control switch for each CF pump on the main control board. The modes available on the switch are Alternate Speed (ALT) and Manual/Auto (M/A). In the ALT position, speed is varied by manually adjusting a control potentiometer supplied by General Electric. In the M/A switch position CFPT speed is controlled by the use of a Westinghouse M/A selector station slave controller associated with each CF pump. In the manual mode on the selector station, each pump can be independently controlled by operator manipulation. In the automatic mode, speed is controlled by the speed demand controller which takes input from the differential pressure (D/P) error signal between SM/CF header pressure and programmed D/P from compensated steam flow. The signal from the speed demand controller is supplied to the CFPT master controller, which is also a Westinghouse M/A selector station. This configuartion allows each slave controller to automatically control the speed of their associated pump, either simultaneously or separately.

During startup, when the S/G's are being fed through the CA nozzles and flow is regulated by the CF bypass control valves, manual control of CFPT speed is required. The Unit Fast Recovery Procedure uses the GE Potentiometer as the means for manual control instead of the Westinghouse M/A selector station because it allows more fine tuning of CFPT speed.

On June 22, 1985, at approximately 1700 hours, Unit 1 was at approximately 15% power, recovering from the reactor trip which had occurred at 0105 hours on the same day. At 1703 hours, Nuclear Control Operator (NCO) A noticed that S/G levels were dropping due to increasing core heat dissipation. He attempted to transfer feedwater control from the CF bypass control valves (CF30, 39, 48, and 57) to the CF control valves (CF28, 37, 46, and 55) in order to increase feed flow. NCO A then requested assistance from NCO B in transferring control to the CF control valves. The NCO's noticed that CFPT 1B discharge header pressure was approximately 1100 psig and SM/CF DP was approximately 30 psid. These conditions were not adequate for CF entry into the CA nozzles of the S/G's; therefore levels continued to drop. The Unit 2 NCO was requested by NCO A to increase the speed of CFPT 1B. The Unit 2 NCO began to adjust the GE control potentiometer which is the normal method of CFPT speed control during startup. CFPT 1B discharge header pressure did not increase as a result of his actions. It was not communicated to him that on the previous shift, the speed control for CFPT 1B had been placed in manual on the Westinghouse controller. Before the use of the wrong controller could be realized, at 1708:56:357 hours, the reactor tripped on low-low level in S/G C.

As a result of the reactor trip, the main turbine, CFPT 1B, and main generator breakers tripped. In addtion, the motor driven CA pumps and the turbine driven CA pump auto-started. Also, feedwater isolation occurred, and letdown isolation occurred when pressurizer level decreased below 17%.

Operations personnel responded to the incident by entering the following procedures: EP/1/A/5000/01 (Reactor Trip of Safety Injection), EP/1/A/5000/01A (Reactor Trip Response) and AP/1/A/5500/02 (Turbine-Generator Trip. Because the Reactor Coolant (NC) System was rapidly cooled by the CA System, a substantial shrink of the NC System occurred causing pressurizer level to drop significantly. Operations personnel re-established level by opening valves NI9A and NI10B (centrifugal charging pump to cold leg discharge isolation) to provide makeup from the Volume Control Tank (VCT).

To recover from the incident, CA pumps 1A and 1B, along with the CA turbine driven pump, were secured. Feedwater isolation was reset. Valves NI9A and NI10B were closed after pressurizer level was restored to approximately 25%. Letdown isolation was reset, and CFPT 1B was returned to service. Reactor startup commenced at 2000 hours, and the Unit reached Mode 1 at 0333 hours on June 23, 1985.

This incident has been classified as a Personnel Error, because NCO A failed to communicate to the Unit 2 NCO that the CFPT 1B speed was being controlled by the Westinghouse controller in the manual position, instead of the GE speed control potentiometer which is normally used during startup prior to switching the controls to automatic. During the preceding shift, the controls were switched from the GE potentiometer to the Westinghouse M/A selector station while in the process of unit recovery after the reactor trip earlier that day. When the shifts changed at 0700 hours, the status of the CFPT 1B speed controls was discussed and accepted by the oncoming shift. Since Unit 1 and Unit 2 have different turnover meetings, the Unit 2 NCO was unaware that the CFPT 1B speed control was in an unexpected configuration.

A contributing factor in this event was the NCO's failure to request assistance from other control room personnel in ample time when he saw there was a problem in controlling steam generator level. It is probable that the mode of speed control on CFPT 1B could have been properly determined if more time had been available to evaluate the problem of no response from the GE potentiometer when the Unit 2 NCO took control.

## CORRECTIVE ACTION

1. Systems affected by the Reactor Trip were returned to normal alignment.
2. Unit 1 reactor was returned to Mode 1.
3. Operations personnel involved with this incident were counselled on the improvement of communications among control room personnel and the importance of requesting assistance when abnormal plant conditions arise.
4. An update will be issued to all licensed personnel to re-emphasize the need for improved communication when personnel not involved in Unit turnovers are requested to perform activities.

## SAFETY ANALYSIS

Following the Reactor Trip, NC System pressure decreased to a low of 2009.6 psig, after which the pressure began to increase to no-load conditions. Letdown isolation occurred as designed when pressurizer level decreased below 17%, to a low of 10.8%. Makeup water from the VCT was added to increase level to 25%. NC System temperature dropped to 530 degrees F following the trip. This was probably caused by the rapid feeding of the S/G's to restore level and the entry of make-up water from the Volume Control Tank. Steam Generator levels were restored to no load conditions by the automatic actuation of the CA pumps, thus providing adequate heat removal capabilities. Feedwater isolation also occurred due to Reactor Trip coincident with Tave less than 564 degrees F. The health and safety of the public were not affected by this incident.

**DUKE POWER COMPANY**
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July 22, 1985

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

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413

Gentlemen:

Pursuant to 10 CFR 50.73 Section (a) (1) and (d), attached is Licensee Event Report 413.85-44 concerning a Reactor Trip due to Low-Low steam generator level. This event was considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

H.B. Tucker /HBT

Hal B. Tucker

RWO:slb

Attachment

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