NRC FORM 366 (6-1998) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001
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 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the 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. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4)

Turbine/Reactor Trip on High Steam Generator Level Due to Instrument 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 NAME	DOCKET NUMBER 05000 --
11	22	1998	1998 –	013 --	00	12	16	1998	FACILITY NAME	DOCKET NUMBER 05000 --

OPERATING MODE (9)	N	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)
POWER LEVEL (10)	28%		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)
E. S. Grecheck, Site Vice President	(757) 365-2000

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	JB	IMOD	Westinghouse Electric Corp.	Y					

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

On November 22, 1998, at 04:30, with Unit 1 at 28% power, control room annunciators alarmed indicating an incorrect level in steam generator (SG) "B" and a feedwater/steam flow mismatch. In response to an apparent increase in steam flow, the main feedwater regulating valve, 1-FW-FCV-1488, began to open further. Although a control room operator began to manually control 1-FW-FCV-1488 to try to prevent a high level condition, the "B" SG reached its high level turbine trip setpoint. The Unit 1 turbine automatically tripped, which was immediately followed by an automatic reactor trip. When the reactor coolant system cooled to the low average temperature (low T_{avg}) setpoint of 543°F, a safety injection (SI) actuation occurred. The SI initiation resulted from the low T_{avg} condition coincident with an apparent high steam flow condition. The SI actuation was spurious since it resulted from an invalid signal. The event was caused by a short circuit in the summator for the main steam line "C" loop channel III flow transmitter. Approved Root Cause Evaluation recommendations, designed to prevent the recurrence of a similar event, will be implemented. The NRC was notified pursuant to 10 CFR 50.72 (b)(2)(ii) on November 22, 1998 at 07:20. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv).

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

1.0 DESCRIPTION OF THE EVENT

On November 22, 1998, at 04:30, with Unit 1 at 28% power, control room annunciators [EIIS-IB] alarmed indicating an incorrect level in steam generator (SG) "B" [EIIS-AB,SG] and a difference between the feedwater and steam flow parameters. In response to an apparent increase in steam flow, the main feedwater regulating valve [EIIS-SJ,FCV], 1-FW-FCV-1488, began to open further. Although a control room operator began to manually control 1-FW-FCV-1488 to try to prevent a high level condition, the "B" SG reached its high level turbine trip setpoint. As designed, the Unit 1 turbine [EIIS-TA,TRB] automatically tripped, which was immediately followed by an automatic reactor trip [EIIS-JC].

The auxiliary feedwater pumps [EIIS-BA-P] started on low-low SG water level and provided flow to the SGs. When the reactor coolant system (RCS) cooled to the low average temperature (low T_{avg}) setpoint of 543°F, a safety injection (SI) [EIIS-BQ] actuation occurred. The SI initiation resulted from the low T_{avg} condition coincident with an apparent high steam flow condition (one steam flow instrumentation channel [EIIS-JB,CHA] for each of SGs "A" and "C" had been placed in the tripped condition, prior to the event, to facilitate system maintenance). Emergency diesel generators (EDG) [EIIS-EK,EDG] Nos. 1 and 3 automatically started upon SI initiation. Following verification that the SI actuation had been spurious, the SI was terminated and the EDGs were shutdown.

The RCS reached a minimum temperature of approximately 535°F and subsequently stabilized at 547°F. The reactivity shutdown margin was calculated following the RCS cooldown to ensure that Technical Specification and administrative shutdown margin limits were satisfied.

The following discrepancies were noted during the post-trip response:

- Intermediate position was indicated in the control room when 1-FW-FCV-1488 was fully closed.
- "C" SG steam line pressure indication was lower than the actual value.
- "B" SG steam line flow was indicated when no flow was present.

The NRC was notified pursuant to 10 CFR 50.72 (b)(2)(ii) on November 22, 1998 at 07:20. This report is being submitted pursuant to 10 CFR 50.73 (a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features and the reactor protection system.

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2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event resulted in no safety consequences or implications. The SI actuation was spurious since it resulted from an invalid signal (i.e., an actual high steam flow condition did not exist). Appropriate operator actions were taken in accordance with emergency operating procedures to ensure the performance of system automatic actions and to respond to abnormal conditions. The unit was quickly brought to a stable, no-load condition. Therefore, the health and safety of the public were not affected at any time during this event.

3.0 CAUSE

A Category 1 Root Cause Evaluation (RCE) was initiated on November 22, 1998, to determine the cause of this event and to recommend corrective actions. The RCE has preliminarily concluded that the event was caused by a short circuit in the summator for the main steam line "C" loop channel III flow transmitter [EIIS-JB,FIT], 1-MS-FT-1494. The short circuit resulted in circulating ground currents which caused the "B" loop channel III steam flow parameter to be greater than the actual value. The "B" loop channel III was affected through its power supply, which is common to the "C" loop channel III. As a result of the false steam flow indication, 1-FW-FCV-1488 opened rapidly to increase feedwater flow to the "B" SG. The level in the "B" SG increased to the high level turbine trip setpoint before control room operators could intervene.

The 1-MS-FT-1494 summator had been replaced and was in the process of being returned to service when the event occurred. The RCE investigation revealed that the module repair testing procedure did not include the defective portion of the summator's circuit board. As a result, the fault was not identified before installation.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the reactor trip, control room operators acted promptly to place the unit in a safe, shutdown condition in accordance with emergency and other operating procedures.

The Shift Technical Advisor monitored the critical safety function status trees to ensure that plant parameters remained acceptable.

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5.0 ADDITIONAL CORRECTIVE ACTIONS

The1-MS-FT-1494 summator was replaced. The installation of a new summator corrected the "C" SG steam line pressure and "B" SG steam line flow indication discrepancies.

The limit switches [EIIS-ZIS] for 1-FW-FCV-1488 were adjusted and the valve was tested satisfactorily.

The RCE team is evaluating unit conditions and systems response contributing to the RCS cooldown following the reactor trip.

6.0 ACTIONS TO PREVENT RECURRENCE

Approved RCE recommendations, designed to prevent the recurrence of a similar event, will be implemented.

7.0 SIMILAR EVENTS

None

8.0 MANUFACTURER/MODEL NUMBER

Westinghouse Electric Corporation
Signal Summator
Assembly No. 4111084-001

9.0 ADDITIONAL INFORMATION

Unit 1 was returned to service on November 23, 1998.

Unit 2 was operating at 100% power and was not affected by this event.