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)
Browns Ferry - Unit 1	0 5 0 0 0 2 5 9	1 OF 0 4

TITLE (4)
Automatic Reactor Scram Due To Loss Of Feedwater

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAMES	DOCKET NUMBER(S)
0 1	1 6	8 5	8 5	— 0 1 6 —	0 0	0 5	3 1	8 5		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	N
POWER LEVEL (10)	1 0 0

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
Jimmy B. Walker	2 0 5 7 2 9 - 2 5 3 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	B N	T R B	T 1 4 7	Y	B	S B	F I	T 0 6 8	Y
B	J K	L I C	G 0 8 0	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 fifteen single-space typewritten lines) (16)

Unit 1 scrammed during normal operation due to low reactor water level while at 100 percent power.

The high pressure steam to the reactor feed pump turbines isolated due to problems with the master level controller which resulted in a loss of feedwater to the reactor vessel. The vessel level continued decreasing, due to the feedwater flow loss, past the isolation setpoint. The primary containment isolation system functioned as designed. The High Pressure Coolant Injection System initiated and returned the water level to the high trip setpoint in approximately six minutes. The Reactor Core Injection Coolant System initiated but tripped immediately on mechanical and electrical overspeed.

The root cause for the unit scram was believed to be a cold solder joint in the master level controller which resulted in loss of motive steam to the reactor feed pump turbines and subsequent loss of feedwater to the vessel. The cold solder joint was resoldered and no further problems have occurred.

IE22
1/1

NRC Form 366
(9-83)

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Browns Ferry - Unit 1	0 5 0 0 0 2 5 9	8 5	— 0 1 6	— 0 0	0 2 OF 0 4

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

Unit 1 was operating at 100 percent power, unit 2 was in a refueling outage, and unit 3 was at 100 percent power. Unit 1 was the only unit affected by this event.

A red phone report was made to NRC within one hour after the event, but the written report was not submitted in the normal timeframe due to administrative errors in preparing the licensee event report.

On January 16, 1985, at 1440 during normal operation, unit 1 scrammed due to low reactor water level after a loss of the feedpumps.

The root cause of this event was believed to be in the master level controller (LIC) (LIC 46-5) which caused the steam loss to the reactor feed pump turbines. Following the loss of high pressure steam to the reactor feed pump turbines, the reactor scrammed on low reactor level. The level continued to drop past the setpoint for Main Steam Isolation Valves (MSIV)(FCV) closure, High Pressure Coolant Injection (HPCI) (BG) System initiation, and Reactor Core Isolation Cooling (RCIC) (BN) System initiation. The primary containment isolation system (PCIS) functioned properly. Both HPCI and RCIC initiated, however, RCIC failed to inject. HPCI injected for approximately six minutes, and the water level returned to the high level trip setpoint. The PCIS was reset within 40 seconds of the scram and the MSIVs reopened.

RCIC initiated but tripped immediately on electrical and mechanical overspeed, as well as high turbine exhaust pressure. The mechanical and electrical overspeed pickup mechanisms were checked for damage, and both appeared to be in good condition. The setpoint of the high turbine exhaust pressure switch was checked and found to be operating correctly. The RCIC controller, turbine exhaust pressure switches, and pressure transmitter were functionally checked and no problems were found. Both the speed feedback magnetic pickup and the electronic overspeed pickup were inspected and replaced as a precautionary measure. It could not be determined why the high turbine exhaust pressure trip came in. The flow test surveillance instruction was performed with no problems, and a number of RCIC quick starts was performed. This condition could not be repeated, and after successful completion of the RCIC operability surveillance test, the system was declared operable. The mechanical tripping mechanism was believed to be set too sensitive prior to the scram and caused the RCIC to trip prematurely.

The reactor pressure was maintained below the relief valve (RV) setpoints by manually controlling the reactor steam pressure by opening 2 of the 13 safety relief valves after the MSIVs closed. After one of these relief valves (MSRV 1-23) closed, the acoustic monitor channel for this relief valve still indicated flow in the relief valve tailpipe.

Upon investigation, the relief valve was verified closed and the acoustic monitor light was still indicating flow. The light cleared when the acoustic monitor power source was turned off and back on. This reset the light to the normal (off) position, and the operator was able to reset his flow alarm.

The manufacturer (Technology for Energy Corporation) indicated that no other users of this equipment (TEC Model 914) had reported this problem. TEC was, however, able to simulate this latch-up behavior in laboratory conditions under high signal strength and gain setting. Their recommendation was to momentarily turn the power off the power monitor module should this occur again, or to reduce the gain of the 914 module. These monitors have redundant relief valve tailpipe temperature sensors.

The recirculation (AD) system protective relaying scheme for pump 1A MG set failed to operate properly. The normal MG set drive motor was tripped by low reactor water level, which results in a loss of field relay to drop out and the generator lock out relay to operate. However, neither relay operated, apparently due to the alternate breaker contact fingers not being fully made up. This configuration allowed the alternate breaker to accept an auto transfer from the normal to alternate supply for MG set 1A drive motor. This auto transfer created a disagreement between the actual position and the position of their control switches. The alternate breaker was later manually tripped by the assistant shift engineer at the breaker. The root cause of this failure was the apparent unsuccessful engagement of the alternate breaker contact fingers. The protective relaying scheme was later checked and found to operate properly.

Unit 1 returned to service on January 21, 1985, at 0835. Diagnostics continued on the feedwater system since no definitive cause could be ascertained during the shutdown.

On January 26, 1985, the applicable portion of refuel test instruction (RTI) (to verify proper controller operation) was performed on unit 1 feedwater system. The system functioned properly at that time and passed the RTI criteria successfully.

On January 30, 1985, unit 1 was operating at 1014 MWe, some 14 days after the initial feedwater problem, at 0832 the reactor level dropped three different times approximately 10 minutes apart due to feedwater controller runback. After the third transient occurred, the operator immediately placed the master controller to manual and returned the level to normal. The master level controller was replaced with the controller from unit 2.

The controller was bench tested again and worked properly at first. All electrolytic capacitors were checked. One capacitor was found to be out of tolerance and leaking. The capacitor was replaced. The controller was returned to the test configuration and was still functioning properly. Later the controller became erratic. This time the problem was traced to a cold solder joint connection of a wire. The joint was resoldered, and the controller left in a test situation throughout the night with no further problems.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Browns Ferry - Unit 1	0 5 0 0 0 2 5 9	8 5	— 0 1 6	— 0 0	0 4 OF 0 4

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

All safety systems functioned as designed during the scram. The RCIC serves as a backup to the HPCI as a source of feedwater makeup during primary system isolation conditions. Failure of the RCIC represents a loss of redundancy of the high pressure injection systems. All other safety systems were fully operable and performed their design function.

The scram was caused by the intermittent transient caused by the cold solder joint. This is considered to be a random failure, and no further action is planned.

Responsible Plant Section - N/A

Previous Events - None

TENNESSEE VALLEY AUTHORITY
Browns Ferry Nuclear Plant
P. O. Box 2000
Decatur, Alabama 35602

May 31, 1985

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

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT (BFN) UNIT 1 - DOCKET NO. 50-259 - FACILITY OPERATING LICENSE DPR-33 - REPORTABLE OCCURRENCE REPORT BFRO-50-259/85016

The enclosed report provides details concerning automatic reactor scram due to loss of feedwater. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

Robert L. Leonard
for G. T. Jones
Plant Manager
Browns Ferry Nuclear Plant

Enclosures
cc (Enclosures):
Regional Administrator
U. S. Nuclear Regulatory Commission
Office of Inspection and Enforcement
Region II
101 Marietta Street, Suite 2900
Atlanta, Georgia 30303

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

NRC Resident Inspector, BFN

IE22
1/1