NRC Form 366 (9-83)

UPDATE REPORT - PREVIOUS REPORT DATE 7/24/84

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 2

TITLE (4)
Reactor Manual Scram due to Main Steam Relief Valve Lifting

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 6	2 7	8 4	8 4	— 0 2 7 —	0 1	0 8	2 8	8 4		0 5 0 0 0
										0 5 0 0 0

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

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)
D. L. Smith	2 0 5 7 2 9 - 0 8 6 5

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)

[ ] YES (If yes, complete EXPECTED SUBMISSION DATE) [X] NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

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

During normal unit startup following a short unit outage, while approaching 1 percent power with approximately 400 psig reactor pressure, a main steam relief valve began to unseat. It continued to leak causing the torus temperature to approach technical specification limits and the unit was manually scrammed at 350 psig. The relief valve subsequently reseated at 100 psig. No unusual occurrences followed.

The valve was replaced and the unit restarted. The relief valve was tested at Wyle Laboratories; it performed normally. It was then disassembled and all internal parts were found to be normal. No reason, either electrical or mechanical, could be found to explain the premature valve actuation. No further corrective action is necessary.

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1/1

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 4	— 0 2 7 —	0 1	0 2 OF 0 2

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

On June 27, 1984, unit 1 was in startup at 1 percent power, unit 2 was at 49 percent power, and unit 3 in a refueling outage. Unit 1 was the only unit affected by this event.

Unit 1 was in startup and approaching 1 percent power at approximately 400 psig reactor pressure, when the pressure control valve 1-4 (RV) unseated. As reactor (RCT) pressure decreased, torus (BT) temperature was approaching its technical specification limit; therefore, a reactor manual scram was initiated at 350 psig. The valve subsequently reseated at 100 psig. No unusual events followed, and there was no safety significance to this event.

The relief valve was completely and thoroughly tested, both electrically and mechanically, by TVA, Wyle, and Target Rock personnel. The initial test actuation pressure was recorded to be 2.2 percent above acceptable criteria. This is a common problem industry wide for first time actuation of relief valves after a long period of standby readiness. Subsequent actuations were all within limits. After it was tested, the valve was disassembled and inspected. No reason could be found for premature valve actuation. The valve was recertified and placed into service on unit 3. No further corrective action is required.

Responsible Plant Section - N/A

Previous Similar Events

BFRO-50-259/7726
BFRO-50-259/7713
BFRO-50-260/7503
BFRO-50-260/7430
BFRO-50-260/7429
BFRO-50-259/7349
BFRO-50-259/7314

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401
Browns Ferry Nuclear Plant
P. O. Box 2000
Decatur, Alabama 35602

August 28, 1984

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

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 1 - DOCKET NO. 50-259 - FACILITY OPERATING LICENSE DPR-33 - REPORTABLE OCCURRENCE REPORT BFRO-50-259/84027 R1

The enclosed updated report provides additional details concerning reactor manual scram due to main steam relief valve lifting. This report was originally submitted accordance with 10 CFR 50.73 (a)(2)(iv).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

G. T. Jones

G. T. Jones
Plant Manager
Browns Ferry Nuclear Plant

Enclosure
cc (Enclosure):
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

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