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LTR 1

SUBJECT: FORWARDING LICENSEE EVENT REPT (RO 50-296/78-008) ON 04/15/78 CONCERNING

RELIEF VALVE 3-1-31 WHICH FAILED TO RESEAT UNTIL PRESSURE REACHED 280 PSIG

DURING A REACTOR SCRAM... W/ATT.

PLANT NAME: BROWNS FERRY - UNIT 3

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INCIDENT REPORTS

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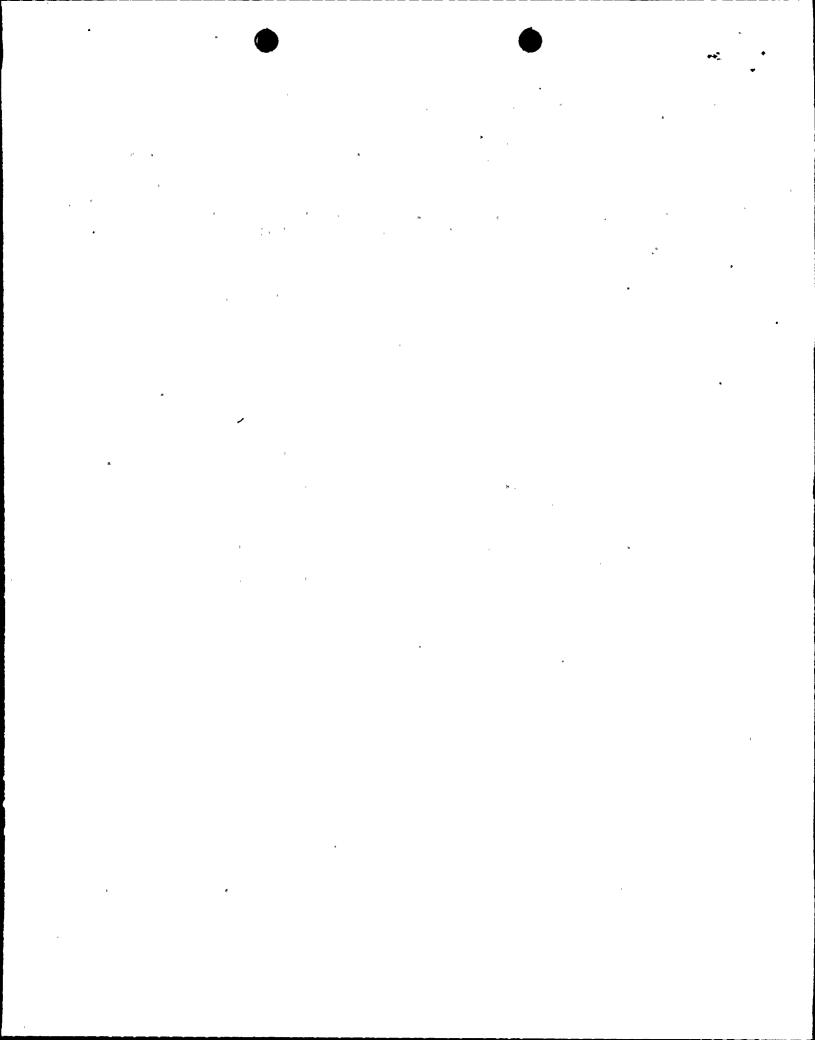
ENCL 45

CONTROL NBR: 781240050

SIZE: 1P+1P+2P

\*\*\*\*\*\*\*\* THE END

\*\*\*\*\*\*\*\*



## TENNESSEE VALLEY AUTHORIT

CHATTANOOGA, TENNESSEE 37401

## REGULATORY DOCKET FILE COPY

APR 28 1978 ir. James P. O'Reilly, Director U.S. Nuclear Regulatory Commission Office of Inspection and Enforcement Region II 230 Peachtree Street, NV., Suite 1217 Atlanta, Georgia 30303

Dear Mr. O'Reilly:

TEMPLESEE VALLEY AUTHORITY - BROWNS FERRY MUCLEAR PLANT UNIT 3 -DOCKET NO. 50-296 - FACILITY OPERATING LICENSE DPR-68 - REPORTABLE OCCURRENCE REPORT BFRO-50-296/788

The enclosed report provides details concerning relief valve 3-1-31 which failed to reseat until reactor pressure reached 280 psig during a reactor scram. This report is submitted in accordance with Browns Ferry unit 3 Technical Specifications 6.7.2.a.(2) and 6.7.2.a.(9).

Very truly yours,

TENNESSEE VALLEY AUTHORITY

H. S. Pox Director of Power Production

Enclosure (3) cc (Enclosure):

Director (3)

Office of Management Information and Program Control U.S. Nuclear Regulatory Commission

Washington, DC 20555

Director (40) Office of Inspection and Enforcement U.S. Nuclear Regulatory Commission Washington, DC 20555

THE POINT c<sub>P</sub> - N 833.W ا را کا در غوا از ایک مختل

U.S. NUCLEAR REGULATORY COMMISSION

## LICENSEE EVENT REPORT

EXHIBIT A

	LATIDIT A
	CONTROL BLOCK: [ ] [ ] (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)
01	A 1 B R F 3 2 0 0 - 0 0 0 0 0 0 0 3 4 1 1 1 1 1 1 3 5 CAT 59 5
O I	SOUNCE LOCKET NUMBER  EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (1)  [During a reactor scram, relief valve 3-1-31 failed to reseat until reactor
	pressure reached 280 psig. The moderator cooldown rate exceeded the 100°F/
اوأوا	hr. rate allowed by Technical Specification 3.6.A.1. The torus reached a
013	maximum temperature of 118°F. Similar failures of valves to reseat have
06	previously been reported in BFAO-50-260/7349, BFAO-50-260/749, BFAO-50-260/
07	7429, BFAO-50-260/7430, and BFRO-50-260/783. There was no hazard to the
08	public health and safety. (See attached report.)
	SYSTEM CAUSE SUBCODE COMPONENT CODE SUBCODE SU
	The second report of the secon
10	The valve which malfunctioned was a Target Rock, model 67F, serial No. 184.
	It was replaced with serial No. 154. The cause of the malfunction is not
12	known at this time. Tests and inspections will be performed to determine
13	the cause, and results will be reported.
<u>ina</u>	
, , , , , , , , , ,	ACILITY SPOWER OTHER STATUS  O
	LEASED OF RELEASE AMOUNT OF ACTIVITY (35)      33
,	PERSONNEL EXPOSURES NUMBER TYPE DESCRIPTION (39)
117	9 11 12 13 80 PERSONNEL INJURIES
18	NUMBER   DESCRIPTION (41)   NA
7 8	LOSS OF OR DAMAGE TO FACILITY 43 TYPE DESCRIPTION
io	L 100 NA
2101	PUBLICITY  DESCRIPTION (45)  NA
استساميتما	
٠,	. NAME OF PREPARER

\*Revision

## Other Events During the Occurrence

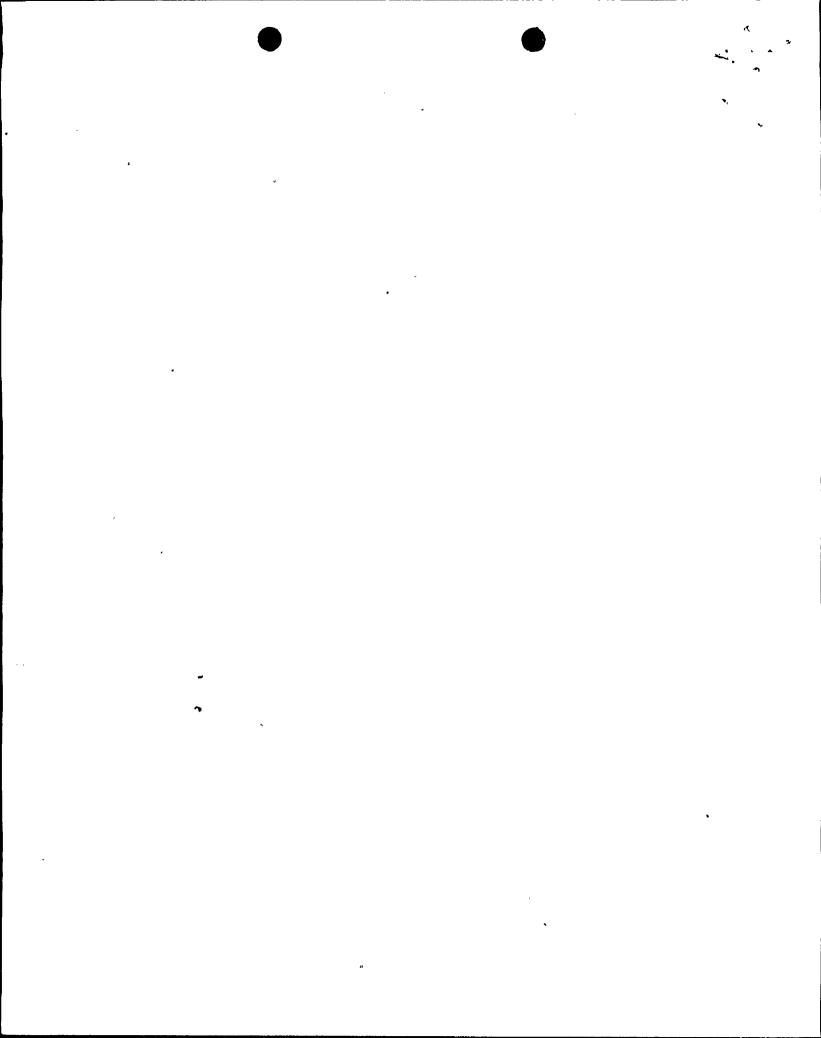
The supply line to the condensate ring header in the unit 3 torus room failed at a welded joint following the reactor scram. The ring header provides reactor quality water to the HPCI and RCIC. The weld failure occurred in the proximity of a junction of a 24-inch and a 20-inch pipe at elevation 552 above core spray pumps A and C and resulted in the loss of about 80,000 gallons of condensate. Water in the core spray pumps A and C room rose to approximately 12 inches and then overflowed the door base and into the torus floor area. Since the floor drain system is common to other corner rooms and the HPCI room, subsequent equalizing of the spilled condensate throughout the basement area resulted.

No apparent cause was evident. A similar event occurred on November 23, 1977.

Preliminary evaluation concluded the most probable cause was weld fatigue caused by line movement during repetitive operations of the HPCI system. The corrective action being taken is to remove the junction and replace it with a straight pipe.

The failure of this pipe has been analyzed in a study conducted by TVA titled "Concluding Report on Effects of Postulated Pipe Failure Outside of Containment for Units 2 & 3 for the Browns Ferry Nuclear Plant - DED-TM-PF2" dated March 1, 1974.

The object of the studies performed was to show that the plant could be placed in and maintained in a cold shutdown condition following a postulated pipe failure in any of the high or low energy lines outside of containment. This evaluation included the failure of the condensate storage tank suction line to the HPCI and RCIC systems and showed that its failure would not prevent placing the plant in or maintaining it in a cold shutdown condition.



' All equipment functioned as required or remained operable during and ensuing the scram. No personnel received any radiation or contamination exposure as a result of the failure.

