

F 00105/198

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)  
DISTRIBUTION FOR INCOMING MATERIAL

50-296

REC: OREILLY J P  
NRC

ORG: FOX H S  
TN VALLEY AUTH

DOCDATE: 04/28/78  
DATE RCVD: 05/02/78

DOCTYPE: LETTER NOTARIZED: NO

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SUBJECT:

LTR 1 ENCL 1

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

REVIEWER INITIAL: XJM  
DISTRIBUTOR INITIAL: OL

\*\*\*\*\* DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS \*\*\*\*\*

INCIDENT REPORTS  
(DISTRIBUTION CODE A002)

FOR ACTION: BR CHIEF LEAR\*\*W/4 ENCL

INTERNAL: REG FILE\*\*W/ENCL  
I & F\*\*W/2 ENCL  
SCHROEDER/IPPOLITO\*\*W/ENCL  
NOVAK/CHECK\*\*W/ENCL  
KNIGHT\*\*W/ENCL  
HANAUER\*\*W/ENCL  
EISENHUT\*\*W/ENCL  
SHAO\*\*W/ENCL  
KREGER/J. COLLINS\*\*W/ENCL  
K SEYFRIT/IE\*\*W/ENCL

NRC PDR\*\*W/ENCL  
MIPC\*\*W/3 ENCL  
HOUSTON\*\*W/ENCL  
EEB\*\*W/ENCL  
BUTLER\*\*W/ENCL  
TEDESCO\*\*W/ENCL  
BAER\*\*W/ENCL  
VOLLMER/BUNCH\*\*W/ENCL  
ROSA\*\*W/ENCL

EXTERNAL: LPDR'S  
ATHENS, AL\*\*W/ENCL  
TIC\*\*W/ENCL  
NSIC\*\*W/ENCL  
ACRS CAT B\*\*W/16 ENCL

COPIES NOT SUBMITTED PER  
REGULATORY GUIDE 10.1

DISTRIBUTION: LTR 45 ENCL 45  
SIZE: 1P+1P+2P

CONTROL NBR: 781240050

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

9



TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

REGULATORY DOCKET FILE COPY

APR 28 1978

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

Dear Mr. O'Reilly:

TEENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 3 -  
DOCKET NO. 50-296 - FACILITY OPERATING LICENSE DPR-68 - REPORTABLE  
OCCURRENCE REPORT BFR0-50-296/788

The enclosed report provides details concerning relief valve 3-1-31 which failed to reseal 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,

TEENNESSEE VALLEY AUTHORITY

H. S. Fox  
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

US NRC  
ESTIMATION SERVICES  
BRANCH

1978 MAY 2 AM 11 41

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781240050

4002  
5/11

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SECTION

14 11 11 S 11 11

SAVINGS

11 11 11

NRC FORM 356  
(7-77)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

EXHIBIT A

CONTROL BLOCK: 1 (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

01 8 9 14 15 25 26 30 31 32 33 34 35 36 37 38 39 40 41 42 43 44 45 46 47 48 49 50 51 52 53 54 55 56 57 58 59 60

LICENSEE CODE: A I B R F 3 LICENSE NUMBER: 0 0 0 - 0 0 0 0 0 - 0 0 LICENSE TYPE: 4 1 1 1 1 CAT 59: 4

CON'T: 0 1 REPORT SOURCE: L 6 COCKET NUMBER: 0 5 0 0 0 2 9 6 7 EVENT DATE: 0 4 1 1 5 7 8 REPORT DATE: 0 4 2 8 7 8

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

02 During a reactor scram, relief valve 3-1-31 failed to reseal until reactor

03 pressure reached 280 psig. The moderator cooldown rate exceeded the 100°F/

04 hr. rate allowed by Technical Specification 3.6.A.1. The torus reached a

05 maximum temperature of 118°F. Similar failures of valves to reseal 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.)

09 SYSTEM CODE: C C CAUSE CODE: E CAUSE SUBCODE: B COMPONENT CODE: V A L V E X COMP. SUBCODE: X VALVE SUBCODE: B

17 LER/RO REPORT NUMBER: 7 1 8 EVENT YEAR: 7 1 8 SEQUENTIAL REPORT NO.: 0 0 8 OCCURRENCE CODE: 0 1 REPORT TYPE: T REVISION NO.: 0

ACTION TAKEN: C FUTURE ACTION: Z EFFECT ON PLANT: A SHUTDOWN METHOD: B HOURS: 0 1 5 3 ATTACHMENT SUBMITTED: N NPD-4 FORM SUB.: N PRIME COMP. SUPPLIER: N COMPONENT MANUFACTURER: T 0 2 0

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

10 The valve which malfunctioned was a Target Rock, model 67F, serial No. 184.

11 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.

14

15 FACILITY STATUS: E % POWER: 1 0 0 OTHER STATUS: NA METHOD OF DISCOVERY: A DISCOVERY DESCRIPTION: N/A

16 ACTIVITY RELEASED OF RELEASE: Z AMOUNT OF ACTIVITY: NA LOCATION OF RELEASE: NA

17 PERSONNEL EXPOSURES NUMBER: 0 0 0 TYPE: Z DESCRIPTION: NA

18 PERSONNEL INJURIES NUMBER: 0 0 0 DESCRIPTION: NA

19 LOSS OF OR DAMAGE TO FACILITY TYPE: L DESCRIPTION: NA

20 PUBLICITY ISSUED: N DESCRIPTION: NA

NRC USE ONLY

NAME OF PREPARER

PHONE:

7510

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.



