Follow-up Report

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 3	0 5 0 0 0 2 9 6	1 OF 0 2

TITLE (4)
Residual Heat Removal Valve (3 FCV-74-67) Stem Broke

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 NAMES	DOCKET NUMBER(S)
0 2	2 8	8 4	8 4	0 0 4	0 1	0 6	1 5	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9): N

POWER LEVEL (10): 0 9 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)	[ ]	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)	X	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 - 0 7 8 8

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 O	V	W 0 3 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)

During the unit 3 cycle 5 refueling outage on February 28, 1984, the Residual Heat Removal (RHR) Outboard Loop II Isolation Valve stem was found to be broken upon disassembly. (The unit was removed from service on September 7, 1983.) The valve (FCV-74-67) is a 24-inch gate valve which is manufactured by Walworth Company. The most probable cause for the valve stem to break was due to overstress or extreme loading conditions.

The metallurgical engineers performed a failure analysis of the stem breaks which indicated the failure was due to overloading. The stem was broken in two places. Metallurgical examinations did not find any evidence of fatigue or corrosion attack of the fracture surfaces.

Operation Instructions state to use the loop not previously used when going into shutdown cooling. Loop I was used per the shift engineer's log.

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

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

Unit 1 was operating at 95.5 percent power, unit 2 was at 90.2 percent power, and unit 3 was in a refueling outage. Only unit 3 was affected by the event.

On February 28, 1984 at 1030, while unit 3 was in a refueling outage, it was discovered that the valve stem on Low Pressure Coolant Injection System (BO) Loop II injection valve (FCV) was broken. The valve stem was broken in two places, once below the stem packing area and once at the gate connection. The upper stem break surface was battered from cycling the valve after the stem broke.

The valve (FCV-74-67) is a 24-inch gate valve which is manufactured by Walworth Company. The valve stem is made of 410 stainless steel and will be replaced with 17-4 PH stainless which is a stronger and more durable material. From visual examination of the stem break surface, an oxide deposit was present over approximately 50 percent of the total stem cross-sectional break, which indicated that the valve stem had been cracked for some period of time. Metallurgical examination did not find any evidence of fatigue or corrosion attack on the fracture surface.

The unit was removed from service on September 7, 1983 for refueling using Loop I RHR during shutdown cooling. Operation instructions require that the loop not previously used be used when going into shutdown cooling. The shift engineer's and unit operator's logs were reviewed to verify which loop was used September 7, 1983 on unit 3. Loop I RHR was used during shutdown. RHR system is used for shutdown cooling when the reactor pressure is below 100 psig and the reactor temperature is between 200°F and 240°F.

The unit was in cold shutdown at the time and fuel was removed from the vessel. There were no serious safety problems from this event. Since this valve is only cycled during cold shutdown, it is doubtful that a valve failure could go undetected. Failure of the valve during a design basis accident is within the single failure criteria.

The investigation of the limitorque, limitorque switches, and valve disc has been completed; and no abnormal conditions were found. For additional details, see the Metallurgical Report attached.

Prior to the end of the next refueling outage, similar Walworth valve stems will be checked to determine if they are magnetic. If any are found to be magnetic, a metallurgist will measure the hardness of the stem and evaluate the suitability for continued use.

<u>Responsible Plant Section</u> - N/A

<u>Previous Similar Events</u> - None

METALLURGICAL REPORT

ANALYTICAL

BROWNS FERRY NUCLEAR PLANT
UNIT 3
METALLURGICAL ANALYSIS OF
FLOW CONTROL VALVE 3-74-67
STEM FAILURE

Prepared By Terry R. Woods Date 5/25/84

Submitted By E. F. Harwell Date 5/29/84

Introduction

During routine outage maintenance, FCV 3-74-67 was found to be inoperable. Upon disassembly, the valve stem was found to be fractured in two different places. One of the fractures occurred in the backseat region where there is a transition from a cylindrical portion to the rectangular portion of the stem. The other fracture occurred near the top of the stem (see attached photographs). By examination of the fracture surfaces, it appears that the stem had first partially failed in the transition (backseat) region since approximately half of the fracture surface in this region showed oxides (hematite) on the fracture surface. These oxides were not corrosion products of the stem material but were corrosion products from the system. The fracture near the top is believed to be a secondary fracture that occurred some considerable time after the first failure.

Metallurgical analysis of the failed stem by the NCO metallurgists indicates the material to not be the type specified for this application and was also improperly heat treated. The failure mode was overload fracture.

Discussion

The subject 24 inch valve is the residual heat removal (RHR) supply outboard isolation valve. This valve is normally closed during operation. The valve is manufactured by Walworth and is cast stainless steel. Stems of type 316 stainless steel are specified for this valve originally, and type 17-4 pH material has been subsequently specified for replacement stems. The chemistry of the subject stem listed below indicates that the material is type 410 martensitic stainless steel.

X-Ray Fluorescence Analysis

Element	Wt %
Manganese	0.46
Phosphorus	0.02
Silicon	0.28
Nickel	0.29
Chromium	12.44
Molybdenum	0.05
Vanadium	0.05

LECO Analysis

Element	Wt %
Carbon	0.12
Sulfur	0.02

This type of stainless steel is normally specified for stems by the valve manufacturer for use in cast carbon steel valves.

The measured mechanical properties and average hardness value of the stem is listed below.

Charpy V-Notch

Temperature (°F)	Ft-Lb
212	10
Room	7

Tensile Strength (KSI)	Yield Strength (KSI)
157	156

Elongation (%)

18

Rockwell Hardness

Average - 36 HR_c

The combination of mechanical properties is unusual because of the small difference between yield and ultimate tensile strengths, low impact values, a large amount of elongation, and excessive hardness. This combination of mechanical properties is the result of improper heat treatment. Improper heat treatment probably occurred during tempering or there may have been a complete lack of tempering.

Examination of the fracture surfaces using scanning electron microscopy (SEM) revealed a predominantly dimpled rupture which indicates overload. This compares favorably with examination of the tensile specimen fracture surfaces which showed large amounts of dimpled rupture. There was evidence of unusual longitudinal cracking in both the tensile specimens and valve stem. In addition, some cleavage was also noted on the tensile specimens; however, there was no indication of fatigue or corrosion on the fracture surfaces of the stem. Attached are photographs pertaining to the subject stem.

Conclusion and Recommendations

Failure is the result of overload on a stem with poor mechanical properties. These poor mechanical properties are caused by improper heat treatment. These overload fractures occurred on two separate occasions which were closing and subsequently opening of the valve.

TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

June 15, 1984

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

Dear Sir:

TENNESSEE VALLEY AUTHORITY - BROWNS FERRY NUCLEAR PLANT UNIT 3 - DOCKET NO. 50-296 - FACILITY OPERATING LICENSE DPR-68 - REPORTABLE OCCURRENCE REPORT BFRO-50-296/84004 R1

The enclosed report provides follow-up details concerning broken residual heat removal valve (3 FCV-74-67) stem. This report is submitted in accordance with 10 CFR 50.73 (a)(2)(ii) and is 10 CFR Part 21 reportable.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

G. T. Jones

G. T. Jones
Power Plant Superintendent
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, GA 30303

INPO Records Center
Suite 1500
1100 Circle 75 Parkway
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NRC Inspector, Browns Ferry Nuclear Plant

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