NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

UPDATE REPORT:
PREVIOUS REPORT DATE: 5-10-84

LICENSEE EVENT REPORT

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

0	1	N C B E P 2 (2)	0 0 - 0 0 0 0 0 - 0 0 (3)	4 1 1 1 1 (4)	CAT (5)
7	8	LICENSEE CODE 9–14	LICENSE NUMBER 15–25	LICENSE TYPE 26–30	57 CAT 58

CON'T

0	1	REPORT SOURCE L (6)	DOCKET NUMBER 0 5 0 - 0 3 2 4 (7)	EVENT DATE 0 8 0 4 8 2 (8)	REPORT DATE 0 5 2 4 8 4 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2	While manually opening the suppression pool suction supply valve to B pump of B loop
0 3	RHR, 2-E11-F004B, the valve stem spun freely beyond the full open valve position. The
0 4	unit was in cold shutdown at the time of this discovery. This event did not affect
0 5	the health and safety of the public.
0 6	
0 7	
0 8	Technical Specifications 3.5.3.2, 6.9.1.9b

	SYSTEM CODE	CAUSE CODE	CAUSE SUBCODE	COMPONENT CODE	COMP. SUBCODE	VALVE SUBCODE
0 9	C F (11)	E (12)	X (13)	V A L V E X (14)	E (15)	D (16)

(17) LER/RO REPORT NUMBER	EVENT YEAR		SEQUENTIAL REPORT NO.		OCCURRENCE CODE	REPORT TYPE		REVISION NO.
	8 2	—	0 8 8	/	0 3	L	—	3

ACTION TAKEN	FUTURE ACTION	EFFECT ON PLANT	SHUTDOWN METHOD	HOURS (22)	ATTACHMENT SUBMITTED	NPRD-4 FORM SUB.	PRIME COMP. SUPPLIER	COMPONENT MANUFACTURER
A (18)	X (19)	Z (20)	Z (21)	0 0 0 0	N (23)	Y (24)	N (25)	A 3 9 1 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0	IGSCC of the valve stem material, resulting from a hardness factor of the material in
1 1	excess of the manufacturer's maximum specifications which is attributed to improper
1 2	heat treating during manufacturing, had allowed a complete fracture of the stem to
1 3	occur approximately six inches from the valve gate. A new valve stem was installed
1 4	and the valve was returned to service.

	FACILITY STATUS	% POWER	OTHER STATUS (30)	METHOD OF DISCOVERY	DISCOVERY DESCRIPTION (32)
1 5	G (28)	0 0 0 (29)	NA	A (31)	Operator Surveillance

	ACTIVITY RELEASED	CONTENT OF RELEASE	AMOUNT OF ACTIVITY (35)	LOCATION OF RELEASE (36)
1 6	Z (33)	Z (34)	NA	NA

PERSONNEL EXPOSURES

	NUMBER	TYPE	DESCRIPTION (39)
1 7	0 0 0 (37)	Z (38)	NA

PERSONNEL INJURIES

	NUMBER	DESCRIPTION (41)
1 8	0 0 0 (40)	NA

LOSS OF OR DAMAGE TO FACILITY (43)

	TYPE	DESCRIPTION
1 9	Z (42)	NA

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PUBLICITY

	ISSUED	DESCRIPTION (45)	NRC USE ONLY
2 0	N (44)	NA	

NAME OF PREPARER M. J. Pastva, Jr. PHONE: 919-457-9521

GPO 917-926

LER ATTACHMENT - RO #2-82-88

Facility: Unit 2 Event Date: August 4, 1982

During a unit refueling outage, while manually opening the suppression pool suction supply valve to the B pump of the B loop RHR, 2-E11-F004B, it was discovered that the valve stem turned freely beyond the full open valve position. A subsequent inspection of the valve internal workings revealed the valve stem, made of 410SS, had completely fractured approximately 6" from the valve stem T-head. A new valve stem was installed and the valve was satisfactorily cycled, determined to be operable, and returned to service.

A document search of the maintenance history associated with this valve determined the failed valve stem was originally supplied with the valve, which is manufactured by Anchor Darling. The failed valve stem was then sent to the Company's Shearon Harris E&E Center for a metallurgical/failure analysis evaluation. A fracture analysis of the broken F004B valve stem showed the stem had failed from IGSCC. It was found that IGSCC had reduced the valve stem cross-sectional area to 30% of original and the fracture was then completed by a sudden shear. An analysis of the valve stem material showed the material had higher than specified surface hardness which is felt to be a contributor to the stem failure. It is felt the high stem material hardness was due to improper heat treating during the manufacture of the stem material. In addition, it was noted that the stem displayed excessive surface pitting.

Following the receipt of the laboratory findings, a site engineering task force was formed and an extensive document search was initiated at the Brunswick site to identify all Anchor Darling valves in use at the facility and classify them according to heat treatment batch. The Anchor Darling valve stems at the Brunswick site were matched to a specific heat treatment batch number.

An in-place hardness testing program was begun in accordance with an approved special procedure. Two stems from each of the 36 heat treatment batches were chosen as samples. In batches with four or less stems, only one stem was chosen. Also, in seven batches, all stems were smaller than 1 1/4" diameter. These stems could not be tested in place due to constraints placed on the testing device. In these seven batches, only one stem from each batch was selected for removal and testing. The basis for this was the excessive plant impact of mass valve disassembly and the fact that no small stems had been found with excessive hardness to date.

Within two weeks of beginning the testing program, over 60 valve stems were tested, representing samples from 34 of 36 heat treatment batches. Of the 34 batches tested, 5 were identified as having excessive hardness in stems 1 3/4" diameter and larger. The listing of these valves by batch number is presented in Appendix A. (A priority listing of these valves is presented in Appendix B.) No stems 1 1/2" and smaller showed high hardness. On this basis, 2 batches out of 36 which contained only small stems and nonsafety-related valves were not sampled.

Corrective Actions Performed or Planned

As a result of this event, Category 1 stems on Unit 1 were replaced during the 1983 refueling/maintenance outage.

In addition, two Category 2 valve stems on Unit 1, 1-E11-F020A and F020B, were replaced during the Unit 1 outage.

The Unit 2 Categories 1 and 2 valve stems are presently scheduled for replacement during the current Unit 2 refueling outage.

The remaining valve stems will be replaced as the valves become available for maintenance.

APPENDIX A

BSEP VALVES WITH EXCESSIVE STEM HARDNESS

Heat Treatment Batch No.	Valve No.	Qty.	Stem Blank Size
65912-A	E21-F015A and B	4	2 1/4"
66095-A	E21-F007A and B	4	2 1/8"
	E11-F020A and B	4	2 1/8"
67374-A	E51-F022	2	1 3/4"
	*E11-F083	2	1 5/16"
72972-A	E11-F010	2	2 1/16"
	E11-F004A, B, C, and D	8	2 1/16"
74940	E11-F016A and B	4	3"
	E41-F008	2	3 1/2"
	E11-F024A and B	4	3 1/2"
	E11-F048A and B	4	4"

*Valve stem unavailable for testing assumed to be hard.

APPENDIX B

ANCHOR DARLING VALVE STEM REPLACEMENT PRIORITY

Category Items

1 Valves whose failure could cause the loss of a safety function and failure would not be detected.

E21-F007A and B	Core spray injection, normally open
E11-F016A and B	Drywell spray, normally closed

2 Valves whose failure could cause loss of safety function but testing would detect failure.

E21-F015A and B	Core spray full flow test, normally closed
E11-F020A and B	RHR torus suction, normally open
E11-F004A, B, C, and D	RHR torus suction, normally open
E11-F024A and B	Torus test/cooling, normally closed
E11-F048A and B	RHR heat exchanger bypass, normally open

3 Valves whose failure would not cause the loss of a safety function but would prevent required operability testing.

E41-F008	HPCI flow test to CST, normally closed
E51-F022	RCIC flow test to CST, normally closed

4 Valves whose failure would not lead to the loss of a safety function.

E11-F083	RHR suction fill, normally closed
E11-F010	RHR cross-tie, locked closed

CP&L

Carolina Power & Light Company

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Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429

May 24, 1984

FILE: B09-13510C
SERIAL: BSEP/84-1308

Mr. James P. O'Reilly, Administrator
U. S. Nuclear Regulatory Commission
Region II, Suite 3100
101 Marietta Street N.W.
Atlanta, GA 30303

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2
DOCKET NO. 50-324
LICENSE NO. DPR-62
SUPPLEMENT TO LICENSEE EVENT REPORT 2-82-88

Dear Mr. O'Reilly:

In accordance with Section 6.9.1.9b of the Technical Specifications for Brunswick Steam Electric Plant, Unit 2, the enclosed supplemental Licensee Event Report is submitted. This original report fulfilled the requirement for a written report within thirty (30) days of a reportable occurrence and both are in accordance with the format set forth in NUREG-0161, July 1977.

Very truly yours,

C. R. Dietz, General Manager
Brunswick Steam Electric Plant

RMP/ag/LETJ05

Enclosure

cc: Mr. R. C. DeYoung
NRC Document Control Desk

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