

LICENSEE EVENT REPORT

UPDATE REPORT:
PREVIOUS REPORT DATE: 8-31-83

CONTROL BLOCK: _____ 1

(PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

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

CON'T
0 1 REPORT SOURCE L 6 0 5 0 - 0 3 2 4 7 0 8 0 4 8 2 8 0 5 1 0 8 4 9
7 8 60 61 DOCKET NUMBER 68 69 EVENT DATE 74 75 REPORT DATE 80

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 80

0 9 SYSTEM CODE C F 11 CAUSE CODE E 12 CAUSE SUBCODE X 13 COMPONENT CODE V A L V E X 14 COMP. SUBCODE E 15 VALVE SUBCODE D 16
7 8 9 10 11 12 13 14 15 16 17
17 LER/RO REPORT NUMBER 8 2 21 22 SEQUENTIAL REPORT NO. 0 8 8 24 26 OCCURRENCE CODE 0 3 28 29 REPORT TYPE L 30 REVISION NO. 2 32
ACTION TAKEN A 18 FUTURE ACTION X 19 EFFECT ON PLANT Z 20 SHUTDOWN METHOD Z 21 HOURS 0 0 0 0 37 40 ATTACHMENT SUBMITTED N 23 NRPD-4 FORM SUB. Y 24 PRIME COMP. SUPPLIER N 25 COMPONENT MANUFACTURER A 3 9 1 26
33 34 35 36 37 40 41 42 43 44 47

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. 80

1 5 FACILITY STATUS G 28 % POWER 0 0 0 29 OTHER STATUS NA 30 METHOD OF DISCOVERY A 31 DISCOVERY DESCRIPTION Operator Surveillance 32
7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

1 6 ACTIVITY CONTENT Z 33 RELEASED OF RELEASE Z 34 AMOUNT OF ACTIVITY NA 35 LOCATION OF RELEASE NA 36
7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

1 7 PERSONNEL EXPOSURES NUMBER 0 0 0 37 TYPE Z 38 DESCRIPTION NA 39
7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

1 8 PERSONNEL INJURIES NUMBER 0 0 0 40 DESCRIPTION NA 41
7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

1 9 LOSS OF OR DAMAGE TO FACILITY TYPE Z 42 DESCRIPTION NA 43
7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

2 0 PUBLICITY ISSUED N 44 DESCRIPTION NA 45
7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32

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

IE 22

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. 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 70% 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 listing of these valves by system 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. Failure of these valves could cause the associated systems from performing their intended functions.

LER ATTACHMENT - RO #2-82-88 (Cont'd)

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

<u>Heat Treatment Batch No.</u>	<u>Valve No.</u>	<u>Qty.</u>	<u>Stem Blank Size</u>
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-FC07A 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 |



Carolina Power & Light Company

Brunswick Steam Electric Plant
P. O. Box 10429
Southport, NC 28461-0429
May 10, 1984

84 MAY 22 P12:28

FILE: B09-13510C
SERIAL: BSEP/84-1139

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/LFTJ05

Enclosure

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

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