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)
Brunswick Steam Electric Plant Unit No. 1	0 5 0 0 0 3 2 5	1 OF 0 3

TITLE (4)
Unit No. 1 Reactor Scram Due to High Pressure

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 5 0 0 0
0 2	0 3	8 4	8 4	— 0 0 2 —	0 0	0 3	0 2	8 4		0 5 0 0 0

OPERATING MODE (9): 1

POWER LEVEL (10): 0 2 6

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)	X 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)	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
M. J. Pastva, Jr. Regulatory Technician	AREA CODE 9 1 9 — 4 5 7 - 9 5 2 1

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	T G	T B G	B 4 5 0	No					

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE) NO

EXPECTED SUBMISSION DATE (15): MONTH DAY YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)

On February 3, 1984, at 0044, a Unit No. 1 reactor scram occurred due to high reactor vessel pressure (1049 psig recorded). At the time, the unit was operating at 26% power and preparations were in progress to roll and place the unit main turbine into service.

A unit scram recovery was carried out with the lowest recorded reactor level of 163". The main steam flow path to the main condenser was rendered inoperable by this event; however, the Emergency Core Cooling Systems were operable along with the Condensate Feedwater System. The event could have been more severe had it occurred at 100% power. The reactor scram resulted from closure of the main turbine bypass valves caused by a loss of Electrohydraulic Control (EHC) System fluid actuation supply (FAS) pressure to the valves. The FAS header tube to the valves had failed due to two through-wall circumferential cracks in the toe weld of the socket side of the tube where it meets the tube coupling union, attributed to cyclic loading fatigue failure of the tubing material. The subject tubing section was replaced. Inspections of other Unit No. 1 EHC System tubing did not reveal additional similar problems. Appropriate short-term and long-term corrective actions in response to this event will be implemented during respective future outages on each unit. This event did not affect the health and safety of the public.

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Brunswick Steam Electric Plant Unit No. 1	0 5 0 0 0 3 2 5	8 4	— 0 0 2	— 0 0	0 2 OF 0 3

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

On February 3, 1984, at approximately 0040, actions were progressing in preparation for rolling the Unit No. 1 main turbine and placing it into service. At the time, the unit reactor power level was approximately 26% with three main turbine bypass valves open for reactor pressure control. A plant auxiliary operator reported to the Unit No. 1 Control Room that a large high pressure leak existed in the Unit No. 1 main turbine Electrohydraulic Control (EHC) System tubing located in the EHC System pump room. A reduction in reactor power level by reactor control rod insertion was begun. At 0044 a Unit 1 reactor scram occurred due to high reactor vessel pressure (1049 psig recorded). It was determined that the high reactor pressure resulted from closure of the unit main turbine bypass valves caused by a loss of EHC System fluid actuation supply (FAS) pressure to the valves.

A unit scram recovery, in accordance with the scram recovery procedure of Emergency Instruction EI-31, was carried out. During the scram recovery, the lowest recorded reactor vessel level was 163". The main turbine bypass valve closure, resulting from the EHC piping failure, rendered the unit main condenser inoperable as a heat sink for the reactor, however, Unit No. 1 Emergency Core Cooling Systems (ECCS) were operable and available for reactor decay heat removal. In addition, the main condenser was still available to support operation of the turbine-driven feed pumps and the main condenser Off-Gas System steam jet air ejectors.

The reactor system transient encountered during the event was less severe than it could have been had the unit been operating at 100% power, in which case, Maximum Critical Power Ratio (MCPR) thermal limits could have placed the unit in a limiting control rod pattern. In this event, MCPR thermal power levels were not as limiting due to the lower power level.

The subject EHC System tubing failure occurred on the system FAS line to the unit main turbine bypass valves and their two 7.5-gallon and one 10-gallon capacity FAS header accumulators. The failure consisted of two through-wall circumferential cracks of 0.5" and 0.97" in surface length originating on opposite sides of the tube and located in the toe weld of the socket side of the subject FAS tube where the tube meets the coupling union of the FAS high pressure block manifold. The failure of the tubing, which is Type 304 stainless steel in the fully annealed condition, is attributed to fatigue of the tubing material caused by cyclic loading. The plane of the crack growth appears to have been caused by back-and-forth movement of the tube out of the tube row plane. Measurements of the fracture surface feature spacing together with fracture mechanics and stress analysis indicate the stress levels which led to the failure were locally well beyond the yield strength of the tubing material. It is felt these stress levels may have resulted from a two-inch, back-and-forth movement of the tube elbow. This two-inch estimate was made assuming the elbow is located approximately four feet above the FAS high pressure manifold. It was determined there were approximately 2,500 to 3,000 load cycles leading to the subject failure which corresponds to a once-a-day actuation of the main turbine bypass valves.

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

The subject section of FAS tubing with the encountered cracking failure was replaced along with the tubing fitting and was returned to service. In addition, inspections of the Unit No. 1 EHC System operating fluid tubing were performed to determine if other similar problems existed, although none were found.

During respective future Unit Nos. 1 and 2 outages, short-term resolution to this event consisting of installing additional supports for the EHC System tubing of each unit will be performed. In addition, appropriate long-term modifications to the EHC tubing supports of each unit will be developed and installed during respective future outages of each unit.

CP&L

Carolina Power & Light Company

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

March 2, 1984

FILE: B09-13510C
SERIAL: BSEP/84-0478

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 NO. 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-84-2

Dear Mr. O'Reilly:

In accordance with Title 10 to the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

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

MJP/dgr/MSCGC1

Enclosure

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

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