CP&L

Carolina Power & Light Company

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

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DRP-71
LICENSEE EVENT REPORT 1-92-019

Gentlemen:

In accordance with Title 10 of 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 submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours,

J. W. Spencer, General Manager
Brunswick Nuclear Project

ST/

Enclosure

cc: Mr. S. D. Ebneter
Mr. R. H. Lo
BSEP NRC Resident Office

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NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 4/30/92
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE RECORDS AND REPORTS MANAGEMENT BRANCH (P-530), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant Unit 1 and Unit 2	05000324	1

TITLE (4) Safety Relief Valves Tested At Wyle Laboratories Exceeded Technical Specification Limit Due To Pilot Disc-To-Seat Bonding

EVENT DATE (5)			LER NUMBER (6)					REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR		SEQ. NO.		REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	21	92	92	-	019	-	000	09	20	92	BSEP-Unit 2	05000325

OPERATING MODE (9)	4	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)
POWER LEVEL (10)	0%		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)	X	50.73(a)(2)(vii)		OTHER (Specify in Abstract and Text)
			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
Steve F. Tabor, Regulatory Compliance Specialist	(919) 457-2178

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	
B/X	AD	RV	T020	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 Cold Shutdown of Units 1 and 2 eleven Safety Relief Valves (SRVs) were removed from each unit and shipped to Wyle Laboratories for set pressure testing and recertification. 'As-received' testing was performed by Wyle Laboratories using Wyle Test Procedure No. 1025 with the following results: 3 of the Unit 1 SRV pilots (C, G, and L) had a setpoint drift of between 1.5% and 3.5%. The Unit 1 SRV pilots A, E, F, and K had setpoint drifts of 8.1%, 5.7%, 9.1% and 8.6% respectively. The Unit 1 SRV H pilot did not initially lift at a steam lift pressure of 1250 psig. 3 of the Unit 2 SRV pilots (E, G, and L) had a setpoint drift of between 1.3% and 3.8%. The Unit 2 SRV pilots A, B, C, F, and J had setpoint drifts of 5.3%, 4.1%, 9.9%, 7.1%, and 6.3% respectively. The remaining SRV pilot setpoints were found within the Technical Specification allowable setpoint drift tolerance of (+/-) 1%.

The SRVs are manufactured by the Target Rock Corporation. The setpoint drift is a generic concern with the Target Rock two-stage SRVs installed in Boiling Water Reactors. The cause of the drift is attributed to oxygen induced bonding of the pilot disc-to-seat surface. Carolina Power & Light (CP&L) is pursuing a modification to the SRV pilot disc surface.

Although the Technical Specification limiting condition for operation was exceeded, the test results are bounded by a 1986 General Electric analysis which determined that the setpoint drift of the SRVs did not create a potential for exceeding the ASME code reactor pressure vessel limit of 1375 psig. Previous similar events have been reported in LERs 2-84-007, 1-85-033, 2-86-001, 1-87-011, 2-88-005, 1-88-030, 2-89-018, 1-91-002, and 2-91-017.

TITLE

Safety Relief Valves Tested At Wyle Laboratories Exceeded Technical Specification Limit Due To Pilot Disc-To-Seat Bonding

INITIAL CONDITIONS

During Cold Shutdown of Units 1 and 2 eleven Safety Relief Valves (SRVs) were removed from each unit and shipped to Wyle Laboratories for set pressure testing and recertification.

EVENT NARRATIVE

Technical Specifications require that the safety relief valve function of all reactor coolant system safety relief valves be operable with lift settings within +/- 1% of their setpoints. This requirement is intended to prevent the reactor coolant pressure, as measured in the reactor vessel dome, from exceeding the safety limit of 1325 psig. Eleven Target Rock Corporation two-stage safety relief valves are installed in each unit to perform this function.

'As-received' testing was performed by Wyle Laboratories using Wyle Test Procedure No. 1025 with the following results (summary on attachment).:

The Unit 1 SRV pilots C, G, and L had a setpoint drift of between 1.5% and 3.5%.

The Unit 1 SRV pilots A, E, F, and K had setpoints of 8.1%, 5.7%, 9.1%, and 8.6% respectively.

The Unit 1 SRV H pilot did not initially lift at a steam lift pressure of 1250 psig.

The Unit 2 SRV pilots E, G, and L had a setpoint drift of between 1.3% and 3.8%.

The Unit 2 SRV pilots A, B, C, F, and J had setpoints of 5.3%, 4.1%*, 9.9%, 7.1%, and 6.3% respectively.

The remaining SRV pilots had a setpoint drift within the allowable Technical Specification tolerance of +/- 1%.

Special diagnostic testing was performed on the Unit 1 SRV H pilot assembly that did not initially lift. Unit 1 SRV H pilot assembly initially did not open at pressure up to 1270 psig (setpoint 1115 psig). The pilot assembly was then tested with the pilot disc unloaded and the disc lifted at 72 psig steam force below the disc. The actual drift was conservatively estimated at 22.5%.

*Actual drift was tested to be 4.1% although based on observations during testing the drift could have been as high as 11.1%. The calculated average drift as delineated on the Attachment is conservatively based on the 11.1% value.

CAUSE OF EVENT

SRV setpoint drift is caused by oxygen induced pilot disc-to-seat bonding. This bonding has been a recognized industry concern with Target Rock Two-Stage SRV pilot assemblies, model number 7567F, since the early 1980s.

CORRECTIVE ACTIONS

The SRV pilots will be recertified prior to startup of Units 1 and 2.

CP&L is working to mitigate the SRV pilot disc-to-seat bonding independently of General Electric and the Boiling Water Reactor Owner's Group (BWROG) by pursuing a modification to the SRV pilot disc surface. This modification mitigates the oxygen induced bonding rate on the disc-to-seat interface. To date, a total of 26 autoclave tests have been conducted under actual steam scouring conditions at the Target Rock Corporation test facility on Long Island, New York. The results to date have been positive, but further surface measurements and testing are required prior to actual implementation. CP&L is planning an in-plant test (two modified SRV pilot valves per Unit, non-Automatic Depressurization System (ADS) valves) as soon as the optimal modification methodology is chosen. The goal of CP&L's effort is to mitigate SRV setpoint drift to less than (+/-) 3%.

SAFETY ASSESSMENT

Although the Technical Specification limiting condition for operation was exceeded, the test results are bounded by General Electric (GE) analysis of a 1986 event (LER 2-86-001 and SRV Setpoint Drift Evaluation Report). In this evaluation the highest percent drift for any SRV was 22.8% and the average drift was 13%. The GE analysis determined that the peak vessel pressure would have remained below the ASME code reactor vessel pressure limit of 1375 psig. Relative to the event as reported herein, the average drift was 5.7% for Unit 1 and 4.4% for Unit 2 with the highest percent drift for any SRV at 22.5%.

PREVIOUS SIMILAR EVENT

Previous similar events include LERs 2-84-007, 1-85-033, 2-86-001, 1-87-011, 2-88-005, 1-88-030, 2-89-018, 1-91-002, and 2-91-017.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Reactor Core System	AC
Relief Valve	AC/RV

ATTACHMENT

UNIT ONE

SRV	NAMEPLATE SETPOINT	AS-FOUND LIFT POINT	PERCENTAGE OF DRIFT
B21-F013A	1105	1195	+8.1
B21-F013B	1125	1132	+0.62
B21-F013C	1105	1122	+1.5
B21-F013D	1115	1111	-0.36
B21-F013E	1115	1179	+5.7
B21-F013F	1105	1206	+9.1
B21-F013G	1105	1144	+3.5
B21-F013H	1115	Special Test	+22.5
B21-F013J	1125	1118	-0.62
B21-F013K	1115	1211	+8.6
B21-F013L	1125	1152	+2.4

UNIT TWO

SRV	NAMEPLATE SETPOINT	AS-FOUND LIFT POINT	PERCENTAGE OF DRIFT
B21-F013A	1105	1164	+5.3
B21-F013B	1125	1250	+11.1*
B21-F013C	1105	1214	+9.9
B21-F013D	1115	1123	+0.72
B21-F013E	1115	1157	+3.8
B21-F013F	1105	1184	+7.1
B21-F013G	1105	1119	+1.3
B21-F013H	1115	1118	+0.27
B21-F013J	1125	1196	+6.3
B21-F013K	1115	1112	-0.27
B21-F013L	1125	1156	+2.8

Average percent drift for all Unit 1 SRVs is 5.7%.
Average percent drift for all Unit 2 SRVs is 4.4%.

*This percentage of drift is a conservative calculation based on observed performance during the test. The calculated average is conservatively based on the 11.1% value.