CP&L

Carolina Power & Light Company
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SERIAL: BSEP 96-0430
10 CFR 50.73

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-96-013

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Licensee Event Report. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence.

Please refer any questions regarding this submittal to Mr. Mark Turkal at (910) 457-3066.

Sincerely,

W. L.

W. Levis, Director - Site Operations
Brunswick Nuclear Plant

SFT/sft

Enclosures
1. Licensee Event Report
2. Summary of Commitments

cc: Mr. S. D. Ebneter, Regional Administrator, Region II
Mr. D. C. Trimble, NRR Project Manager - Brunswick Units 1 and 2
Mr. C. A. Patterson, Brunswick NRC Senior Resident Inspector
The Honorable H. Wells, Chairman - North Carolina Utilities Commission

IE22 1/1

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 1	05000325	1 OF 3

TITLE (4)

Safety Relief Valves Tested At Wyle Laboratories Exceeded Technical Specification Setpoint Limits

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 NAME	DOCKET NUMBER 05000
10	31	96	96	-- 13 --	00	11	18	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9): 05

POWER LEVEL (10): 0

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.36(c)(2)	X	50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve F. Tabor, Sr. Analyst, Regulatory Affairs	(910) 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	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 15 single-spaced typewritten lines) (16)

During the Unit 1 B111R1 refuel outage, eleven Unit 1 Safety Relief Valves (SRVs) were removed and shipped to Wyle Laboratories for set pressure testing and recertification. Technical Specification 3/4.4.2 requires the SRVs to be operable with lift settings within +/- 1% of their lift setpoints. The as-found test results indicate that the 11 valves exceeded the Technical Specification setpoint requirement.

The SRVs are manufactured by the Target Rock Corporation. Setpoint drift is a generic concern with the Target Rock two-stage SRVs. The cause of the drift is attributed to oxygen induced bonding of the pilot disc-to-seat surface. SRV pilot assemblies with modified pilot disc surfaces were certified and installed prior to startup from the unit refuel outage. The disc modification involves applying a platinum coating to the pilot disc surface. Previous testing and evaluation determined that disc modification in this manner lowers the oxidation rate and thus inhibits SRV pilot disc-to-seat bonding. In addition, the SRV lift setpoint drift allowed by Technical Specification was increased to +/- 3% as part of the Power Uprate Project related amendment to the Technical Specifications.

Although the SRV setpoint drift allowed by the Technical Specification was exceeded, the as-found test results are bounded by a 1986 General Electric analysis which determined that the setpoint drift of the SRVs would not result in exceeding the ASME code reactor pressure vessel limit of 1375 psig.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

TITLE

Safety Relief Valves Tested At Wyle Laboratories Exceeded Technical Specification Setpoint Limits

INITIAL CONDITIONS

During the Unit 1 B111R1 refuel outage, eleven Unit 1 Safety Relief Valves (SRVs) were removed and shipped to Wyle Laboratories for set pressure testing and recertification.

EVENT NARRATIVE

On October 31, 1996, the results of the testing performed by Wyle Laboratories identified that the 11 valves exceeded the Technical Specification setpoint requirement. The test data is provided in Attachment 1.

This event is being reported in accordance with the requirements of 10 CFR 50.73 (a) (2) (vii) as an event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems designed to mitigate the consequences of an accident.

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.

The valve vendor has noted that the model 7567F Target Rock Two-Stage pilot valve is capable of performing within +/- 1% of its setpoint relative to repeatability during certification testing. However, the vendor noted that nuclear industry service experience has determined that the model 7567F experiences a larger drift after having been subjected to the environment of an operating nuclear unit. The Target Rock position on this issue is that +/- 3% drift is the normal range of drift for this design.

CORRECTIVE ACTIONS

The SRV pilot valve assemblies were replaced with certified spares prior to startup of Unit 1.

As committed in LER 1-92-019, CP&L initiated an effort to resolve the SRV pilot disc-to-seat bonding issue. This effort included the modification of the pilot disc surfaces in six SRV pilot assemblies, three of which were installed in each unit during previous refuel outages. The modification involved platinum particle deposition on the SRV pilot disc surface by means of ion beam implantation. The modified disc surface acts as a catalyst to render the pilot disc-to-seat interface less susceptible to oxygen induced bonding.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

The Unit 2 modified discs were removed after unit operation from May of 1993 to March of 1994. The Unit 1 modified discs were removed during the B110R1 refuel outage which ended in May of 1995 following a full cycle of operation. Metallurgical evaluation of the removed modified SRV pilot valve discs was performed to determine if the platinum coating had been damaged, abraded, or removed during the operating cycles. There was no evidence of material loss or damage. Additionally, "as received" set pressure testing of the surface treated disc assemblies was performed at Wyle Laboratories. Test results revealed that the modified disks lifted at less than or equal to +1% of their lift setpoint and within the industry average of +/- 3%.

Based on the results of the testing performed on the modified discs, CP&L decided to modify the pilot valve disc surfaces on the eleven SRVs installed in Unit 1 and Unit 2. SRVs with modified pilot disc surfaces were installed prior to the startup from the Unit 2 refuel outage (B212R1). SRVs with modified pilot disc surfaces were installed in Unit 1 prior to startup from the B111R1 refuel outage.

As committed in LER 1-95-07, CP&L is pursuing a change to the Technical Specification setpoint drift tolerance based on the Power Uprate Project. The Power Uprate Project provides the basis for a change to the Technical Specification 3/4.4.2 to increase the allowable setpoint drift from +/- 1% to +/- 3%. The Power Uprate Project related change to the Unit 1 Technical Specifications was approved prior to startup from the B111R1 refuel outage. The corresponding change to the Unit 2 Technical Specification is scheduled to be implemented by completion of the B213R1 refuel outage in 1997.

SAFETY ASSESSMENT

Although the SRV setpoint drift allowed by the Technical Specification was exceeded, the Wyle Laboratories test results are bounded by General Electric (GE) analysis of a 1986 event (LER 2-86-001 and SRV Setpoint Drift Evaluation Report). The GE analysis determined that the peak vessel pressure would have remained below the ASME code reactor vessel pressure limit of 1375 psig without the actuation of four SRVs and with the actuation of the other seven SRVs occurring above the nameplate setpoint pressure. The overall average setpoint drift in the 1986 analysis was 13%. In the Wyle test the average setpoint drift of the 11 SRVs was +4.6%.

PREVIOUS SIMILAR EVENTS

Similar events have been previously 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, 2-91-017, 1-92-019, 1-95-007, and 2-96-002.

EIIS COMPONENT IDENTIFICATION

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

LER 1-96-13

ATTACHMENT 1

SRV	NAMEPLATE SETPOINT	AS-FOUND LIFT POINT	PERCENTAGE OF DRIFT
B21-F013A	1105	1092	-1.2
B21-F013B	1125	1188	+5.6
B21-F013C	1105	1128	+2.1
B21-F013D	1115	1132	+1.5
B21-F013E	1115	1205	+8.1
B21-F013F	1105	1149	+4.0
B21-F013G	1105	1191	+7.8
B21-F013H	1115	1250	+12.1
B21-F013J	1125	1111	-1.2
B21-F013K	1115	1222	+9.6
B21-F013L	1125	1145	+1.8

Enclosure
List of Regulatory Commitments

The following table identifies those actions committed to by Carolina Power & Light Company in this document. Any other actions discussed in the submittal represent intended or planned actions by Carolina Power & Light Company. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Manager-Regulatory Affairs at the Brunswick Nuclear Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
NONE	