CP&L

Carolina Power & Light Company
P.O. Box 10429
Southport, NC 28461-0429

APR 14 1997

SERIAL: BSEP 97-0134
10 CFR 50.73

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
LICENSEE EVENT REPORT [illegible] 002

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. Keith Jury, Manager - Regulatory Affairs, at (910) 457-2783.

Sincerely,

William Levis
Director — Site Operations
Brunswick Steam Electric Plant

SFT/sft

Enclosures
1. Licensee Event Report
2. List of Regulatory Commitments

IE22
1/1

pc (with enclosures):

U. S. Nuclear Regulatory Commission
ATTN.: Mr. Luis A. Reyes, Regional Administrator
101 Marietta Street, N.W., Suite 2900
Atlanta, GA 30323-0199

U. S. Nuclear Regulatory Commission
ATTN: Mr. C. A. Patterson, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461

U. S. Nuclear Regulatory Commission
ATTN.: Mr. David C. Trimble, Jr. (Mail Stop OWFN 14H22)
11555 Rockville Pike
Rockville, MD 20852-2738

The Honorable J. A. Sanford
Chairman - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Brunswick Steam Electric Plant, Unit 1	05000325	1 OF 4

TITLE (4)

Core Spray Header Differential Pressure Instrumentation Inoperable

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
03	13	97	97	-- 02 --	00	04	14	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
MODE (9)		20.2201(b)	20.2203(a)(2)(v)	X 50.73(a)(2)(i)	50.73(a)(2)(viii)
POWER		20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
LEVEL (10)	95	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)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steve Tabor, Senior Analyst - Licensing	(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

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED	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)

On March 9,1997, the Unit 1 Core Spray Nozzle A Differential Pressure Switch, 1-E21-PDS-N004A, was identified to be indicating -154" H_2O instead of the -108" H_2O value previously considered normal. This instrument generates a control room alarm on an in-vessel line break which would cause the reading to be positive. Technical Specifications require that the 1-E21-PDS-N004A alarm setpoint be based on the normal reading. Based on preliminary review it was determined that the -154" H_2O should be considered normal, and as a result the instrument setpoint was believed to be outside Technical Specification requirements. At 2030 hours, the instrument was conservatively declared inoperable. On March 13, 1997, an engineering evaluation was completed and confirmed that the 1-E21-PDS-N004A instrument setpoint was outside of the Technical Specification value. The evaluation also found that the most probable cause is a bi-stable condition where voiding of the in-vessel Core Spray piping is triggered by vessel conditions. The piping can be either full of water or full of steam and/or non-condensable gases depending on the local hydraulic/thermodynamic conditions at the core spray nozzles. This accounts for readings being stable at two different "normal" conditions. Since the -154" H_2O reading was the more conservative normal condition, the setpoint was changed based on this value. Data indicates that -154" H_2O has been a normal reading for this operating cycle, therefore, the setpoint was not in compliance with requirements of Technical Specifications from approximately November of 1996.

The safety significance of this event is considered minimal since, for most core spray line breaks in the vessel annulus, the piping would not remain completely voided and the alarm would have actuated. Additionally, any line break which should have resulted in an alarm would cause core flow to bypass the dryer-separator and affect reactor power sufficiently to alert operators to the problem.

TITLE

Core Spray Header Differential Pressure Instrumentation Inoperable

INITIAL CONDITIONS

On March 9, 1997, Unit 1 was operating at 95% power with the Automatic Depressurization, the Low Pressure Coolant Injection, the High Pressure Coolant Injection, and the Reactor Core Isolation Cooling systems operable.

EVENT NARRATIVE

On March 9, 1997, during a plant walk down by an Auxiliary Operator, the Unit 1 Core Spray Nozzle A Differential Pressure Switch, 1-E21-PDS-N004A, reading of -154" H_2O was observed and questioned. 1-E21-PDS-N004A senses differential pressure between an above core plate sensing line and a sensing line connected to the Core Spray sparger and provides an alarm input in the event of a line break of the Core Spray piping within the boundary of the reactor pressure vessel, but outside the boundary of the core shroud (i.e., results in core flow diversion). Prior to the last Unit 1 outage, the readings for 1-E21-PDS-N004A during normal full power operation had been between -91" H_2O and -115" H_2O. The instrument setpoint had been established at +31 ± 7" H_2O based on a normal value of -108" H_2O. This is consistent with the Technical Specification requirement that the setpoint be 5 ± 1.5 psi above the normal reading.

Based on preliminary review it was determined that the -154" H_2O should be considered normal, and as a result the instrument setpoint was believed to be outside Technical Specification requirements. At 2030 hours, the instrument was conservatively declared inoperable. On March 13, 1997, an engineering evaluation was completed and confirmed that the 1-E21-PDS-N004A instrument setpoint was outside Technical Specifications. This evaluation also determined that two different "normal" conditions can occur. Since the -154" H_2O reading was the more conservative normal condition, the 1-E21-PDS-N004A alarm setpoint was changed based on this value. Given that -154" H_2O has been a normal reading for this operating cycle, the setpoint was not in compliance with requirements of Technical Specifications from November of 1996. Applying the 5 ± 1.5 psi above normal value requirement to the -154" H_2O normal value, it was determined that the allowable range for the setpoint was +26.2" H_2O to -57" H_2O. Review of the last as-left value of the setpoint identified it above the allowed setpoint of +26.2" H_2O. As a result, this event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(i) as a condition prohibited by Technical Specifications.

CAUSE OF EVENT

An investigation, using troubleshooting and fault tree analysis techniques, was performed to determine the cause of this event. This effort determined that no physical problem existed with the 1-E21-PDS-N004A or its associated sensing lines. In addition, the investigation confirmed that an actual loss of integrity of the in-vessel piping was not the cause of the new reading, as the shift would be in the opposite direction (more positive) and there was no detectable change in the power-to-flow relationship. Other possible plant parameter changes such as reactor pressure, core flow, void fraction, etc. were considered and no direct link to the cause of the condition could be established.

It was identified that the change in stable indications very closely matched what would be expected if the contents of the Core Spray downcomer piping in the vessel annulus were to change from water to either steam or non-condensable gases. Both conditions can be considered "normal" based upon local hydraulic/thermodynamic conditions at the Core Spray nozzles. Due to this close match, and since other postulated causes had been eliminated, the most probable cause of the shift in indication was determined to be voiding of the in-vessel piping. A review of data has determined the following:

Time/condition	Piping Condition
January 1985	Full
From 8/95 to 9/96	Full
11/06/96 during power ascension	Full
11/13/96 shortly after startup	Full
12/31/96 near full power	Voided
01/29/97 one day after a down power to approximately 60%	Full
03/09/97	Voided
03/14/97 at the start of a down power	Voided
At approximately 65% power	Full
03/16/97 day shift at near full power	Full
03/16/97 - 03/17/97 night shift at near full power	Voided
03/17/97 at approximately 15:00 at near full power	Full
03/18/97 at approximately 15:00 at near full power	Voided
03/27/97 at near full power	Voided

The above data indicates that the most likely cause of this event is a change in state in the 1-E21-PDS-N004A instrument piping contents that resulted from a change in the local hydraulic/thermodynamic conditions at the Core Spray nozzles. This accounts for readings being stable at two different "normal" conditions. It is possible that the fluid in the piping has alternated between the two conditions in the past, but due to infrequent monitoring it was not identified earlier. During a reactor down-power on March 14, 1997, Engineering personnel conducted a pre-planned effort to monitor for this condition. The change in piping condition from voided to full was observed, validating the most likely cause.

Using the voided piping normal reading of -154" H_2O is conservative (i.e., it will cause the alarm to actuate for a pipe break in either condition); consequently, this reading is now being used as the basis for the alarm setpoint. Additionally, use of this new setpoint is expected to provide adequate indication for anticipated flow diversion scenarios.

CORRECTIVE ACTIONS

The 1-E21-PDS-N004A instrument setpoint of +31 ± 7" H_2O based on a -108" H_2O normal reading was changed to +2.1" H_2O (-5.0 to +9.1" H_2O) based on a -154" H_2O normal reading. This setpoint is based on the most conservative of the two normal conditions. The basis for the setpoint change is documented in a formal Engineering evaluation and associated 10 CFR 50.59 safety evaluation.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Brunswick Steam Electric Plant, Unit 1	05000325	97	-- 02	-- 00	4 OF 4

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

Other possibly affected instrumentation was reviewed with the following results:

The Unit 2 Loop A Core Spray sparger has the same basic configuration as the Unit 1 sparger. The 2-E21-PDS-N004A instrument setpoint of -9.5 ± 7" H_2O based on a -148" H_2O normal reading was reviewed. The normal reading used for the setpoint determination was apparently taken with the piping voided. Since regular data gathering was started in August of 95, 2-E21-PDS-N004A readings have been between -115" H_2O and -100" H_2O, indicating that the piping has been full. Since the setpoint was established based on the more conservative of the "normal" readings, the setpoint was found to be acceptable as-is.

The Unit 1 and Unit 2 Loop B Core Spray spargers have spray nozzles above the Core Spray header piping instead of below the piping like the Loop A spargers. For both units, all Loop B readings have indicated full piping conditions. This indicates that until a voided piping reading is noted, there is no basis for considering piping voided as a normal condition. Therefore, there is no need to implement any changes to the Loop B instrumentation or setpoints.

SAFETY ASSESSMENT

The safety significance of this event is considered minimal since for most Core Spray line breaks in the vessel annulus, the piping would not remain completely voided and the alarm would have actuated. Additionally, any line break that should have resulted in an alarm, would cause core flow to bypass the dryer-separator and affect reactor power sufficiently to alert operators to the problem. This condition did not result in operation outside of the Technical Specifications for the Unit 1 loop B or the Unit 2 loops A and B of Core Spray.

PREVIOUS SIMILAR EVENTS

None

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Core Spray	BM

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
NRC DOCKET NOS. 50-325 AND 50-324
OPERATING LICENSE NOS. DPR-71 AND DPR-62
LICENSEE EVENT REPORT 1-97-002

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Carolina Power & Light (CP&L) Company in this document. Any other actions discussed in the submittal represent intended or planned actions by CP&L. 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 Steam Electric Plant of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
None	