CP&L

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FEB 14 1996

SERIAL: BSEP 96-0066
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 1-96-002

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 and is submitted in accordance with the format set forth in NUREG-1022, September 1983.

Please refer any questions regarding this submittal to Mr. George Honma at (910) 457-2741.

Sincerely,

W. Levis
Director — Site Operations
Brunswick Nuclear Plant

SFT/wrm

Enclosures
1. Licensee Event Report
2. Summary of Commitments

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

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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 4

TITLE (4)

Unit 1 Manual Reactor Scram Due to Main Turbine Vibration

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
01	23	96	96	-- 02 --	00	02	14	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	01
POWER LEVEL (10)	28

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)	X	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, 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

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
X	YES (If yes, complete EXPECTED SUBMISSION DATE).	NO		05	31	96

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On January 23, 1996, with the Unit 1 reactor operating at 28% power and undergoing a planned reactor shutdown to correct previously identified control rod slow scram insertion times, main turbine bearings experienced increasing vibration. When vibration levels approached the procedural limits, a manual reactor scram was initiated at approximately 0657 hours. Reactor water level momentarily decreased below the low level 1 setpoint (162.5"), resulting in Primary Containment Isolation System (PCIS) Group 2 (Drywell Floor and Equipment Drains) and Group 6 (Containment Atmospheric Control) valve isolations. In addition, a Group 8 (Shutdown Cooling) isolation signal occurred; however, the Group 8 valves were already closed at the time of the shutdown. The plant responded as expected. Following completion of repairs to the Unit 1 hydraulic control units and Plant Nuclear Safety Committee review of the event recovery effort, Unit 1 reactor power ascension commenced. On January 25, 1996, at 0435 hours, the Unit 1 main generator was synchronized to the electrical grid system.

Investigation results indicate that the increased turbine vibration resulted from rubs in the low pressure turbine diaphragm and shaft packing. At the time of the event the unit was being operated at low power and the steam temperature and pressure entering the low pressure turbines was lower than it had been at full power. Seasonal changes in circulating water temperature led to lower turbine exhaust hood temperatures and pressures. The reduced power level in combination with the lower exhaust hood temperatures and pressures resulted in reduced clearances between the fixed and rotating turbine parts. These operating conditions led to the packing rubs.

Formal root cause analyses to document the cause of the turbine vibration and control rod slow scram insertion times are in progress. The results of these analyses, including corrective actions to prevent recurrence, will be provided in a supplement to this LER.

The safety significance of the scram event is minimal in that the plant responded as designed and the Emergency Core Cooling Systems were operable at the time of the event.

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

TITLE

Unit 1 Manual Reactor Scram Due to Main Turbine Vibration

INITIAL CONDITIONS

Unit 1 was operating at 28% reactor power and undergoing a planned reactor shutdown. The shutdown was initiated on January 23, 1996, to allow repairs of the Control Rod Drive (CRD) system hydraulic control units. The Unit 1 Residual Heat Removal, High Pressure Coolant Injection, Core Spray, and Reactor Core Isolation Cooling systems were operable.

EVENT NARRATIVE

On January 20, 1996, Technical Specification required control rod scram time testing on ten percent of the Unit 1 control rods was performed. Twelve of the fourteen controls rods in the sample exceeded the Technical Specification core-wide average limit (0.358 seconds) for insertion to notch 46 by approximately 0.043 seconds; however, the core average Technical Specification limit was not exceeded. To determine whether a generic problem existed with control rod scram times, an engineering investigation team was assembled on January 21, 1996. Utilizing fault tree analysis techniques, the investigation team developed a diagnostic plan to determine the cause of the timing indications. On January 21 and 22, 1996, a series of tests were performed on additional selected control rods. These tests validated that the delay in scram times was a generic problem and attributable to the Scram Pilot Solenoid Valve (SSPV) assemblies. Details of the testing performed on January 21 and 22, 1996, are provided in NRC Information Notice 96-07 and INPO Operational Experience Report 7652 dated January 26, 1996.

On January 23, 1996, based on the results of the diagnostic testing, BNP management decided to shutdown the Unit 1 reactor to replace the SSPV exhaust diaphragms and continue diagnostic testing. While reducing reactor power, several of the main turbine bearings experienced increasing vibration. With vibration levels at approximately 11.6 mils on bearing number 5 and approaching the procedural limits, a manual reactor scram as inserted at approximately 0657 hours. Reactor water level momentarily decreased below the low level 1 setpoint (162.5"), resulting in Primary Containment Isolation System (PCIS) Group 2 (Drywell Floor and Equipment Drains) and Group 6 (Containment Atmospheric Control) valve isolations. In addition, a Group 8 (Shutdown Cooling) isolation signal occurred; however, the Group 8 valves were already closed at the time of the shutdown. Startup level control was placed in service to maintain normal reactor vessel operating level. At approximately 0710 hours, the PCIS isolations were reset and the affected systems returned to service. The reactor was maintained in the Hot Shutdown mode of operation until repairs to the SSPVs could be completed. A Site Incident Investigation Team (SIIT) was organized to investigate the scram and determine the actions necessary for restart of the Unit 1 reactor.

Following the scram, the five percent insertion time data of 79 control rods were retrieved and applied to the core average. The data indicated that the core-wide average five percent insertion time was approximately 0.380 seconds, which exceeded the Technical Specification 3.1.3.3 limit for insertion to notch 46 (0.358 seconds). Additionally, 25 two-by-two control rod arrays exceeded the Technical Specification 3.1.3.4 limit for average scram insertion time. After replacement of the SSPVs the core average measurement following reactor startup was 0.309 seconds.

Following completion of repairs to the Unit 1 SSPVs and Plant Nuclear Safety Committee (PNSC) review of the SIIT report, Unit 1 reactor power ascension commenced. On January 25, 1996, at 0435 hours, the Unit 1 main generator was synchronized to the electrical grid system.

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

CAUSE OF EVENT

Investigation results indicate that the increased turbine vibration resulted from rubs in the low pressure turbine diaphragm and shaft packing. At the time of the event the unit was being operated at low power and the steam temperature and pressure entering the low pressure turbines was lower than it had been at full power. Seasonal changes in circulating water temperature led to lower turbine exhaust hood temperatures and pressures. The reduced power level in combination with the lower exhaust hood temperatures and pressures resulted in reduced clearances between the fixed and rotating turbine parts. These operating conditions led to the packing rubs.

An inspection was performed on the Unit 1 low pressure turbines in conjunction with rotor replacement in the Spring of 1995. The steam packing in the diaphragms and shaft packing were replaced at that time. When the unit was started in May, rubbing occurred in the packing in both low pressure turbines. This rubbing is often experienced in steam turbines after packing replacement. By operating the turbine in a careful manner the packing was rubbed out providing additional clearance and reducing vibration. Even though the unit was operated at full power during the summer months, some packing with less than adequate clearance for the low power and reduced circulating water temperature condition remained. The conditions that contributed to this event were assessed by company and vendor personnel and the turbine was determined to be acceptable for restart. During unit restart no packing rubs or unusual vibration were experienced. The clearance required to operate the unit during expected seasonal and load conditions is believed to have been achieved.

The primary cause of the slow control rod scram insertion times has been isolated by diagnostic and laboratory testing to be adherence of the SSPV's exhaust diaphragm to the valve seat. This adherence phenomena has been demonstrated in independent tests at both the Automatic Switch Company (ASCO) and General Electric (GE) and appears to be characteristic of the diaphragm material. The Unit 1 SSPVs are the "dual type" ASCO solenoid valves. The Unit 1 valves were replaced during the 1995 spring refuel outage with a new design ASCO solenoid valve and had been in-service for approximately 8 months. The newly designed valve uses a Viton diaphragm instead of the original Buna-N diaphragm. The Viton replacement diaphragm is the result of a Boiling Water Reactor Owners Group (BWROG) effort and is recommended by GE SIL 585 and qualified by GE NEDC 32365P. The exact mechanism underlying the Viton adherence characteristic is still under investigation.

CORRECTIVE ACTIONS

During Unit 1 power ascension following the scram, additional vibration monitoring equipment was installed to monitor the main turbine performance and provide detailed diagnostic vibration information. The turbine vibration data remained acceptable during reactor restart.

A formal root cause analysis of the turbine vibration event is on-going to ensure contributing factors and appropriate corrective actions have been identified. The results of this analysis will be provided in the supplement to this LER.

An engineering evaluation was performed to provide an operability assessment of the Unit 1 CRD system with new Viton diaphragms installed. The evaluation provided a basis for the acceptability of replacing the Unit 1 degraded SSPV diaphragms with new diaphragms. The 137 inboard and outboard SSPV exhaust diaphragms and end caps were replaced prior to startup of Unit 1. Testing of the control rod scram insertion times during reactor startup power ascension demonstrated that the new diaphragms restored the control rod insertion times well within the Technical Specification limits. Additionally, the Plant Nuclear Safety Committee will review industry data and the results of the root cause analysis for the slow scram insertion times within 90 days from startup to determine whether accelerated testing or other actions are warranted.

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CORRECTIVE ACTIONS (Cont.)

BNP is working with GE and the GE BWROG Regulatory Response Group to identify the exact failure mechanism and to develop a long term resolution to this problem. In addition, BNP has contracted the services of two independent research laboratories to assist in the root cause investigation. The results of these efforts and BNP's root cause analysis of the event will be provided in a supplement to this LER.

SAFETY ASSESSMENT

The safety significance of the scram event is minimal. Following the scram the plant responded as designed and consistent with the analyses presented in the Updated Final Safety Analysis Report. In addition, the Unit 1 Residual Heat Removal, High Pressure Coolant Injection, Core Spray, and Reactor Core Isolation Cooling systems were operable at the time of the event. The manual scram was inserted prior to exceeding the limits for turbine vibration as established by plant procedure. Post scram testing and turbine data review indicate no turbine system component abnormalities resulted from the increased turbine vibration condition.

Control rods are inserted to assure thermal limits are not exceeded during design transients. Troubleshooting to date has confirmed that the slow insertion time condition is being caused by delayed operation of the SSPV. This delayed operation results in a delay in the start of control rod motion but does not affect the speed of travel once motion has begun nor result in a complete failure of the valve to operate. In addition, BNP engineering evaluation has determined that the insertion times obtained from the testing performed from January 20 through January 22, 1996, would not have challenged the core licensing basis nuclear safety criteria. Additional GE analyses determined that the observed degradation of the 5% average scram insertion time does not impact any safety analysis or threaten any safety limits.

PREVIOUS SIMILAR EVENTS

No previous similar events involving a manual reactor shutdown due to main turbine vibration has been identified.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Control Rod Drive	AA
Main Turbine System	TA
Hydraulic Control Unit	HCU

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
LER 1-96-02 will be revised to include the results of the root cause analysis for the slow control rod insertion times and the turbine vibration.	5/31/96