CP&L

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

SERIAL: BSEP 97-0294
10 CFR 50.73

JUL 07 1997

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

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
DOCKET NO. 50-325/LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-97-006

Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.73, Carolina Power & Light Company submits the enclosed Licensee Event Report.

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

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

IE22
1/1

140039

pc (with enclosures):

U. S. Nuclear Regulatory Commission
ATTN.: Mr. Luis A. Reyes, Regional Administrator
Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, GA 30303

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

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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Engineered Safety Feature Actuation Due to Loss of Emergency Bus E-2

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
06	08	97	97	-- 006 --	00	07	07	97	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1
POWER LEVEL (10)	100

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 71.95
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
B	EK	27	G080	Y		B	DC	PSF	N152	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)

On June 8, 1997, at 1441 hours, with Units 1 and 2 operating at 100% power, a momentary loss of 4160 volt emergency bus E-2 occurred while manually starting Emergency Diesel Generator (EDG) #2. The loss of emergency bus E-2 resulted in the following unplanned, automatic Engineered Safety Feature (ESF) system actuations: Group 1 Division II logic trip (Main Steam Line) and closure of the 1-B32-F020, Reactor Sample Valve; Reactor Protection System Division II trip; Secondary Containment isolation and start of both Standby Gas Treatment Trains; Group 2 Division II isolation (Drywell Floor and Equipment Drains); Group 3 isolation (Reactor Water Cleanup); Group 6 Division II and applicable containment monitor isolations (Containment Atmospheric Control); and Group 10 Division II isolation (Pneumatic Nitrogen). In addition, EDG #2 auto-started and connected to emergency bus E-2 as designed. At 1907 hours, with EDG #2 supplying power to emergency bus E-2 loads, a ¼ inch fuel oil line on EDG #2 developed a non-isolable leak. At 1913 hours, EDG #2 was manually tripped resulting in the loss of emergency bus E-2 and the same ESF actuations that resulted from the first loss of emergency bus E-2. During both events, the affected systems operated as required. Investigation into the first event determined that the normally closed trip contacts on emergency bus E-2 undervoltage relay 27PK failed to open when the relay was energized due to a less than adequate contact gap. The inadequate contact gap setting is believed to have existed since receipt of the relay from the vendor. Investigation into the failed fuel oil line determined that a pipe-nipple connection used to mount an EDG #2 fuel oil pressure indicator had failed due to fatigue. Corrective actions include replacement of the degraded relay and fuel oil line pipe-nipple, inspection of other similar relays installed in class 1E applications, relay calibration procedure enhancements, and replacement and inspection of similar EDG pipe-nipple applications. Problems were not identified with the relays and pipe-nipples installed in similar applications.

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

TITLE

Engineered Safety Feature Actuation Due to Loss of Emergency Bus E-2

INITIAL CONDITIONS

On June 8, 1997, Units 1 and 2 were operating at 100% reactor power. Testing of the emergency diesel generators (EDG) was in progress to support a planned maintenance outage on EDG #4.

EVENT NARRATIVE

On June 8, 1997, quick start testing of EDGs 1, 2, and 3 commenced at 1335 hours in accordance with surveillance procedure 0PT-12.8, "Diesel Generator Operability Test," to support a scheduled outage of EDG #4. Testing of EDG #2 commenced at approximately 1435 hours. At 1441 hours, the control room operator placed the EDG #2 select switch in the local-manual position to support starting of EDG #2. Immediately following this evolution, the master and slave breakers for the 4160 volt emergency bus E-2 tripped, de-energizing emergency bus E-2. EDG #2 auto-started and, by design, connected to emergency bus E-2. The loss of emergency bus E-2 resulted in the following Engineered Safety Feature (ESF) system actuations:

- Group 1 Division II logic trip (Main Steam Line) and closure of the 1-B32-F020, Reactor Sample Valve
- Reactor Protection System Division II trip
- Secondary Containment isolation and start of both Standby Gas Treatment Trains
- Group 2 Division II isolation (Drywell Floor and Equipment Drains)
- Group 3 isolation (Reactor Water Cleanup)
- Group 6 Division II and applicable containment monitor isolations (Containment Atmospheric Control)
- Group 10 Division II isolation (Pneumatic Nitrogen).

At 1907 hours, with EDG #2 still supplying power to emergency bus E-2, the control room received a report of a non-isolable fuel oil leak of approximately one quart per minute from a ¼ inch EDG #2 fuel oil line. The leaking fuel oil line connects the engine driven fuel pump to the EDG #2 fuel oil pressure gauge 2-FO-PI-1994. At 1910 hours, while attempting to place a rag around the leaking fuel oil line and suppress the spray of fuel oil, the fuel oil line completely failed. At 1913 hours, EDG #2 was manually tripped, resulting in a second loss of emergency bus E-2. Again, the affected systems operated as required and the above ESF system actuations occurred. Tripping EDG #2 stopped the fuel oil leak since the fuel oil was no longer circulating through the engine. At 1926 hours, after the suspected failure mechanism associated with the first loss of emergency bus E-2 was determined, emergency bus E-2 was successfully paralleled to the off-site power source. Following repair of the failed fuel oil line, EDG #2 was declared operable on June 9, 1997, at 0330 hours.

This event is being reported in accordance with the requirements of 10 CFR 50.73 (a)(2)(iv), in that the loss of emergency bus E-2 events resulted in the unplanned automatic actuation of ESF systems.

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CAUSE OF EVENT

The cause of the initial loss of emergency bus E-2 is attributed to the failure of the emergency bus E-2 undervoltage relay. Due to an inadequate relay contact gap, which existed since initial installation of the relay, the normally closed contacts failed to open as designed.

The 27PK undervoltage relay is a General Electric Type NGV Form 11B relay which uses three telephone type coil/contact assemblies. Contacts from these coils are in two parallel trip circuits which are normally open when the coil is energized and closed when the coil is de-energized (i.e., undervoltage condition). On a sensed undervoltage condition, the 27PK relay is designed to actuate resulting in the separation of the respective emergency bus from its off-site power source.

During inspection of the failed relay, one of the three contact assemblies appeared to have limited contact travel, which resulted in the failure of the normally closed contact to fully open upon relay energization. Based on further inspection of the relay and discussions with General Electric, it was determined that there are two possible adjustments that can be made to affect contact operation. One of these adjustments is the wipe setting adjustment which is made prior to initial installation. The wipe setting adjustment affects the normally closed contacts and is made by adjusting the residual screw to establish a 0.005 inch gap between the residual screw and the pole piece. The as-found condition of this adjustment setting would not have caused this event. The applicable vendor manual does not indicate that this adjustment is recommended nor required on a periodic frequency.

The second adjustment alters the pivot of the contact lever which affects contact gap. This adjustment is made with an adjustment screw on the operating arm/armature assembly. This adjustment is not addressed in the vendor manual, but was identified through discussions with General Electric. During inspection of the failed relay, the operating arm/armature assembly setting on one of the three coils was observed to be different from the same setting on the other two operating arm/armature assemblies. The operating arm/armature assembly was subsequently adjusted to match the other two coils and the relay was retested satisfactorily. This adjustment provided an improved gap between the contacts. There are no records to indicate that an adjustment has ever been made to the operating arm/armature assembly; therefore, it is believed that the as-found condition existed since receipt of the relay from the vendor.

A calibration check of the failed 27PK relay was performed via a relay test device on May 27, 1997. The relay test device cycles the relay in the energized and de-energized states. This testing verifies that the contacts open (normal voltage) and close (undervoltage) within the required values. During the May 27, 1997 testing, the failed 27PK relay actuated per design with no adjustment required. When the relay was re-installed in the emergency bus E-2 switchgear, the contacts did not open, as designed, when voltage was applied to the coil.

The cause of the second loss of emergency bus E-2 is attributed to outer diameter-initiated fatigue of a pipe-nipple installed on the fuel oil line connecting the EDG #2 fuel oil pump to the EDG #2 fuel oil pressure gauge. Initial analysis indicates that a crack initiated at a thread root and propagated through the pipe-nipple due to cyclic stresses, resulting in material fatigue and pipe-nipple fracture. Crack initiation appears to have been caused by outer diameter fatigue as no apparent defects other than the thread root crack were evident at the

pipe-nipple's outer diameter surface. However, due to heavy material damage of the crack surface, the exact cause of crack initiation is indeterminate.

CORRECTIVE ACTIONS

The failed 27PK relay was replaced; the new relay was verified to be operable.

An inspection was performed on the 27PK relays installed in the remaining emergency bus switchgears with no operability concerns identified.

An evaluation was performed to determine if the 27PK relay type is used in any other class 1E applications; no other class 1E applications were identified. Further evaluation determined that a similar non-1E General Electric Type NGV relay is used as a "Loss of Field" relay in the reactor recirculation pump motor generator set; however, due to coil configuration differences, this relay is not interchangeable with those in class 1E applications.

The calibration procedure for class 1E 27PK relays has been reviewed to determine areas for enhancement. This review determined that gap setting for the normally closed contacts should be checked during the calibration process. Procedure revisions to incorporate this change will be implemented by September 30, 1997. Procedures used to calibrate other 1E relay types and non-1E General Electric Type NGV relays will also be reviewed for enhancements.

An evaluation of similar piping configurations was performed to identify additional pipe-nipples which would be subject to fatigue failure; nine similar applications were identified. New pipe-nipples were installed in these applications, and a failure analysis performed on the removed pipe-nipples. Visual microscopic examination of these similar pipe-nipples did not show evidence of surface cracks. Non-destructive test (NDT) analysis of these specimens identified two pipe-nipples as having circumferential indications. A second examination of the nine pipe-nipples using a stereomicroscope was conducted. Based on the results of the stereomicroscopic examination, two additional pipe-nipples were selected for further evaluation. Metallographic examination was performed on the four pipe-nipples identified during NDT and stereomicroscopic examination. The results of the testing revealed that identified indications were associated with rough surfaces on the cut thread surfaces away from the pipe-nipple thread roots. None of these indications appeared to be present at the thread roots. The indications observed in these samples were found to be surface laps associated with the cut threads and there was no evidence of cracking observed in any of these sections. Based on these results, the failure of the failed pipe-nipple appears to be an isolated occurrence.

SAFETY ASSESSMENT

This event is of minimal safety significance. The ESF systems responded, as designed, to the losses of emergency bus E-2.

The failure mode of the 27 PK relay in this event would not have prevented EDG #2 from performing its intended safety function. The 27PK relay provides trip functions only when its associated EDG is operating in the manual

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mode. Therefore, this relay has no affect on the emergency electrical distribution system during normal or emergency plant operations as the EDG is maintained in the automatic mode. The incorrect contact gap setting of the 27 PK relay while the EDG is in the manual mode, will cause the EDG to switch to the automatic mode. In addition, the EDGs are tested each month in the manual mode in accordance with Technical Specifications. Testing of EDG#2 prior to May 27, 1997, had been performed successfully, verifying that this relay was operating satisfactorily. In addition, operation with the loss of a single EDG is bounded by existing accident analysis.

PREVIOUS SIMILAR EVENTS

Previous similar events involving the loss of an emergency bus due to the failure mechanisms identified in these events were not identified.

EIIS COMPONENT IDENTIFICATION

System/Component	EIIS Code
Bus	BU
Pipe Fittings	PSF
Emergency On-site Power Supply System	EK
Undervoltage Relay	27
Reactor Power Control System	JD
Containment Isolation Control System	JM
Emergency Standby Gas Treatment System	BH
Reactor Water Cleanup System	CE
Diesel Fuel Oil System	DC
Containment Environmental Monitoring System	IK

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 1-97-003

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 of any questions regarding this document or any associated regulatory commitments.

Commitment	Committed date or outage
The class 1E 27 PK relay calibration procedure will be revised to require verification of gap settings for the normally closed contacts.	9/30/97
Procedures used to calibrate other 1E relay types and non-1E General Electric Type NGV relays will be reviewed for areas of enhancement.	1/15/98