NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Virgil C. Summer Nuclear Station	0 5 0 0 0 3 9 5	1 OF 0 3

TITLE (4)
Residual Heat Removal Suction Isolation

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 NAMES	DOCKET NUMBER(S)
1 0	1 8	8 4	8 4	0 4 5	0 0	1 1	1 4	8 4		0 5 0 0 0
										0 5 0 0 0

OPERATING MODE (9)	5	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)			
POWER LEVEL (10)	0 0 0	20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
		20.405(a)(1)(i)	50.36(c)(1)	XX 50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
		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
	AREA CODE
A. R. Koon, Jr., Assoc. Mgr., Regulatory Compliance	8 0 3 — 3 4 5 - 5 2 0 9

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	
A	B P			N							

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE)	XXX NO				

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

On October 18, 1984, the Plant was in Mode 5 for the First Refueling Outage with Train "A" of the Residual Heat Removal (RHR) System in service, RHR Train "B" out-of-service for routine maintenance, and the Reactor Coolant System (RCS) vented at a temperature of approximately 110°F. At 1605 hours a power loss to 120 VAC Distribution Panel APN-5901 de-energized Solid State Protection System (SSPS) Channel I and caused the instrument channel for RCS Wide Range Pressure (PT-403) to initiate an auto-closure of the operable RHR train's Suction Isolation Valve XVG-8701A. Following determination that the power loss had been caused by personnel error during the performance of a Plant modification, Operations personnel restored power to APN-5901.

XVG-8701A was opened and Train "A" of the RHR system returned to operable status at 1630 hours (total time of RHR isolation was approximately 25 minutes). RCS temperature increased from 110°F to 130°F during the event. The loss of RHR met the conditions of an Alert, and the proper notifications were made in accordance with the Emergency Plan.

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FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Virgil C. Summer Nuclear Station	0 5 0 0 0 3 9 5	8 4	— 0 4 5	— 0 0	0 2 OF 0 3

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

On October 18, 1984, the Plant was in Mode 5 for the First Refueling Outage with Train "A" of the Residual Heat Removal (RHR) System in service, RHR Train "B" out-of-service for routine maintenance, and the Reactor Coolant System (RCS) vented at a temperature of approximately 110°F. At 1605 hours, a power loss to 120 VAC Distribution Panel APN-5901 de-energized Solid State Protection System (SSPS) Channel I and caused the instrument channel for RCS Wide Range Pressure (PT-403) to initiate an auto-closure of the operable RHR train's Suction Isolation Valve XVG-8701A. Following determination that the power loss had been caused by personnel error during the performance of a Plant modification, Operations personnel restored power to APN-5901. XVG-8701A was then reopened and Train "A" of the RHR System returned to service at 1630 hours (total time of RHR isolation was approximately 25 minutes). The consequences of this event were negligible due to the availability of two (2) steam generators for heat removal and the short duration of the isolation.

The cause of the power loss and subsequent RHR isolation has been attributed to personnel error. This event occurred during implementation of a modification to upgrade the incore thermocouple circuitry in accordance with the conditions set forth in the Operating License Section C.23.d, "Inadequate Core Cooling Instruments (II.F.2, SSER4)." A section of this design package required the removal of a previously installed power cable (ICV2A) and the deletion of its corresponding termination sheet. The termination sheet incorrectly indicated that ICV2A was routed from Breaker #19 of APN-5901 to an Isolation Fuse Panel.

ICV2A had been initially installed but not terminated during the construction program as a part of preliminary work to upgrade the incore thermocouple system. A subsequent modification installed a power feed to panel APN-5907 using Breaker #19 on APN-5901. The termination sheet for ICV2A was not revised to show this modification; however, the Plant one-line drawings were revised to reflect that Breaker #19 on APN-5901 actually provided power to APN-5907.

The instructions contained in the modification package to remove ICV2A failed to show that ICV2A was not actually terminated to Breaker #19. Thus when the electrician implementing this section of the modification saw Breaker #19 closed, he obtained authorization from the Control Room to open the breaker. When the breaker on APN-5901 was opened, the loss of power to APN-5907 caused three (3) process cabinets being fed from the panel to automatically transfer to an alternate source of power. Annunciator alarms immediately made the Control Room aware of the power loss even though no loss of equipment had occurred at this time.

At the request of the Control Room, an Instrument and Control Supervisor went to the relay room and identified the cause of the alarms. At approximately 1605 hours, the electrician reclosed the breaker without first obtaining clearance from the Control Room. Reconnecting the load

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Virgil C. Summer Nuclear Station	0 5 0 0 0 3 9 5	8 4	— 0 4 5	— 0 0	0 3 OF 0 3

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generated an overcurrent condition on the inverter feeding APN-5901 causing a reduction in the output voltage. When an operator reopened Breaker #19, the inverter tripped off line and initiated the RHR suction isolation valve closure. Operations restored power to APN-5901 from an alternate source approximately 25 minutes later and re-established RHR. The loss of RHR met the conditions of an alert, and the proper notifications were made in accordance with the Emergency Plan.

The Licensee has initiated the following actions to prevent a potential recurrence:

1) Electrical Maintenance personnel attended training sessions on October 19 and 22, 1984, which emphasized:

   A. the need to verify circuits prior to de-energizing by either the circuit number or the electrical feeder list to insure accuracy of drawings and,

   B. if a circuit is incorrectly de-energized, notify the Control Room to restore the circuit per the appropriate Plant procedure.

2) The event will be reviewed by the appropriate engineering personnel by December 31, 1984 This review will emphasize the need to include sufficient instructions in modification packages to reduce the probability of inadvertent equipment isolations during the implementation of Plant modifications.

3) The Licensee has initiated an investigation into the cause of the inverter trip. This investigation is designed to evaluate the response of the inverter during the event.

SOUTH CAROLINA ELECTRIC & GAS COMPANY
POST OFFICE 764
COLUMBIA, SOUTH CAROLINA 29218

O. W. DIXON, JR.
VICE PRESIDENT
NUCLEAR OPERATIONS

November 14, 1984

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

SUBJECT: Virgil C. Summer Nuclear Station
Docket No. 50/395
Operating License No. NPF-12
LER 84-045

Dear Sir:

Attached is Licensee Event Report #84-045 for the Virgil C. Summer Nuclear Station. This Report is submitted pursuant to the requirements of 10CFR50.73(a)(2)(v).

Should there be any questions, please call us at your convenience.

Very truly yours,

O. W. Dixon, Jr.

CJM:OWD/lcd
Attachment

cc: V. C. Summer
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IE22
1/1