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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Poor Work Practice Results in Trip of
Emergency 600-Volt Bus "2C" and Unplanned System Actuations

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a poor work practice which resulted in a trip of emergency 600-volt bus "2C" and unplanned system actuations.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

OCV/eb

Enclosure: LER 50-366/2001-001

cc: Southern Nuclear Operating Company
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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)
Poor Work Practice Results in Trip of Emergency 600-Volt Bus "2C" and Unplanned System Actuations

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: Plant Hatch Unit 1	DOCKET NUMBER(S): 05000321
03	08	2001	2001	001	00	04	30	2001		DOCKET NUMBER(S): 05000

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)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20-2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20-2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Steven B. Tipps, Nuclear Safety and Compliance Manager, Hatch	(912) 367-7851

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX

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-space typewritten lines) (16)

On 03/08/2001 at 0306 EST, Unit 2 was in the Run mode at a power level of 2763 CMWT (100 percent rated thermal power). At that time, the supply breaker to Unit 2 emergency 600-volt bus "2C" opened on a false over-current signal. As a result, Reactor Protection System (RPS) bus "2A" lost power. Logic systems powered by the RPS bus include the RPS trip logic, the Main Control Room Environmental Control System (MCRECS) initiation logic, Primary Containment Isolation System (PCIS) trip logic, and Steam Packing Exhauster trip logic. All affected systems responded per design on the loss of power, producing a half scram signal, several PCIS valve isolations, MCRECS pressurization mode initiation, and other actuations. In addition, the "A" subsystems of the Core Spray, Residual Heat Removal (RHR) containment spray, RHR suppression pool spray, and RHR service water systems were rendered inoperable. By 0540 EST, personnel had re-energized 600-volt bus "2C" and returned affected safety-related systems to their normal lineups.

The cause of the supply breaker trip was a poor work practice causing personnel calibrating an over-current relay to short two trip contacts in the relay case, generate a false over-current signal, and trip the bus supply breaker. Corrective actions included revising calibration procedures to eliminate the poor work practice.

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
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TEXT *(If more space is required, use additional copies of NRC Form 366A)* (17)

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes appear in the text as (EIIS Code XX).

DESCRIPTION OF EVENT

On 03/08/2001 at 0306 EST, Unit 2 was in the Run mode at a power level of 2763 CMWT (100 percent rated thermal power). At that time, the supply breaker to Unit 2 emergency 600-volt bus (EIIS Code EB) "2C" opened on a false over-current signal. An Instrument and Control technician had been preparing to calibrate 600-volt bus "2C" over-current relay 2S32-K837-2 per plant procedure 57CP-CAL-108-2S, "Westinghouse CO Overcurrent Relay." After removing the relay from its case, he inadvertently shorted two contacts in the relay case when he stored a relay connecting plug in the case. This action caused a false over-current signal to be generated, tripping the supply breaker to the 600-volt bus.

As a result of the loss of power to emergency 600-volt bus "2C," Reactor Protection System (RPS, EIIS Code JC) bus "2A" lost power causing some Group 2, Group 5, and outboard small-bore Group 1 Primary Containment Isolation System (PCIS, EIIS Code JM) valves to receive an automatic isolation signal. Those valves open at the time of the event closed per design. Also, the Main Control Room Environmental Control System (MCRECS, EIIS Code VI) entered the pressurization mode; the Primary Containment Hydrogen and Oxygen Analyzers (EIIS Code IK) isolated; and the Reactor Water Cleanup (RWCU, EIIS Code CE) system, the Fission Product Monitoring (FPM, EIIS Code IJ) system, and the operating steam packing exhauster tripped.

The loss of power to emergency 600-volt bus "2C" also rendered inoperable the "A" subsystems of the Core Spray (EIIS Code BM), Residual Heat Removal (RHR) containment spray (EIIS Code BO), RHR suppression pool spray (EIIS Code BO), and RHR service water (EIIS Code BI) systems because of a loss of power to motor-operated valves in the major flow paths of the respective systems. The "A" subsystem of the Unit 1 Low Pressure Coolant Injection system (EIIS Code BO) was rendered inoperable when Motor Control Center (EIIS Code EB) 1R24-S018A lost its normal power supply, that is, Unit 2 emergency 600-volt bus "2C." The "2A" station service DC bus (EIIS Code EJ) also was rendered inoperable upon loss of power to its one standby and two in-service battery chargers. However, the station service batteries (EIIS Code EJ) supplied power to and maintained operable the DC loads powered by the "2A" DC bus until the battery chargers were restored to service approximately 40 minutes after the loss of power to the 600-volt bus.

Loss of power to Standby Gas Treatment (SGT, EIIS Code BH) system logic components resulted in isolation of the secondary containment and actuation of the SGT system trains. Loss of power to the reactor recirculation system (EIIS Code AD) master controller caused its output to fail downscale and, per its design, send a forty-four percent speed signal to the individual reactor recirculation system pump controllers. The "2B" reactor recirculation system pump decreased to forty-four percent speed. However,

power also was lost to the "2A" reactor recirculation system pump controller, causing the pump scoop tube to lock at its pre-event position per design and therefore the "2A" pump did not run back. Runback of the "2B" pump caused power to decrease to approximately 70 percent rated thermal power.

Immediately upon hearing relays actuating and the supply breaker opening unexpectedly, the Instrument and Control technician contacted Operations personnel in the Main Control Room (EIIS Code NA) to inform them he likely had caused the breaker to open. Given this information, and with no indication of any abnormal condition, Operations personnel re-closed the 600-volt bus supply breaker and re-energized the bus approximately three minutes after the supply breaker opened. Restoration of power to the 600-volt bus restored the "A" subsystems of the Core Spray, RHR containment spray, RHR suppression pool spray, RHR service water, and Unit 1 Low Pressure Coolant Injection systems to an operable status. However, restoration of power also resulted in an unexpected increase in the speed of the "2B" reactor recirculation system pump and a corresponding increase in core thermal power as the pump controller attempted to return pump speed to the pre-event speed demand setting on the master controller. Because this reaction of the pump controller upon restoration of power was not expected, Operations personnel locked manually the pump scoop tube and terminated the pump speed, and core thermal power, increase within one minute. Between the time 600-volt bus power was restored and the pump scoop tube was locked, power increased about 15 percent.

By 0540 EST, personnel completed resetting the affected logic systems, restoring power to affected components, and returning affected safety-related systems to their normal configurations.

CAUSE OF EVENT

The cause of the supply breaker trip was a poor work practice. While performing a routine calibration of an over-current relay per plant procedure 57CP-CAL-108-2S, an Instrument and Control technician shorted two contacts in the over-current relay case, causing a false over-current signal to be generated and the supply breaker to emergency 600-volt bus "2C" to trip. He shorted the two contacts when he placed a connecting plug back in the case for storage after removing it and the relay from the case.

A connecting plug is used to provide electrical continuity between the over-current relay and the corresponding contacts located in the relay case. The plug is not required to perform the bench calibration of the relay. Therefore, placing the connecting plug back in the case for storage while the relay was being calibrated was a standard practice for this and some of the other technicians. This technician had performed the same action without incident during the calibration of the first of three over-current relays associated with emergency 600-volt bus "2C" completed earlier in the shift, and during many prior relay calibrations as well. However, in this event one of the metal strips on the plug simultaneously touched two contacts in the case, creating a circuit between the contacts. This occurred probably from an unusual combination of the two contacts protruding above the floor of the case and the plug being placed into the case at a slight angle. This circuit generated a false over-current signal and tripped the 600-volt bus supply breaker.

The cause of the "2B" reactor recirculation system pump unexpectedly increasing speed following the restoration of power was an inadequate design. Personnel installed new reactor recirculation system master and pump controllers in October 1992 per Design Change Request 90-163. The design of the new controllers, as embodied in their new programming data sheets, incorrectly required the controllers to be set in the "hot start" mode. The "hot start" mode will automatically increase pump speed to the demand speed setting of the master controller upon restoration of power to that controller. Reactor recirculation pump speed increases occurring without operator intervention are undesirable. Therefore, the design should have required the controllers to be set in the "cold start" mode in which operator intervention is required to increase pump speed following restoration of power to the master controller.

Compounding this error in the controller design was an error in incorporating this design into the simulator. The controllers in the simulator were set in the "cold start" mode so that, upon restoration of controller power, the reactor recirculation system pumps stayed at their runback speeds until manually reset by the operators. Simulator training created an expectation that the actual reactor recirculation system pumps would not increase speed upon restoration of power to the master controller. Furthermore, plant procedures reflected this expectation in that they contained no instructions to prevent automatic pump speed increase upon restoration of power. Therefore, Operations personnel did not take action to lock the "2B" pump scoop tube prior to re-energizing the 600-volt bus, leading to an unexpected increase in pump speed and core thermal power.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This event is reportable per 10 CFR 50.73(a)(2)(iv) because unplanned actuations of safety feature systems listed in 10 CFR 50.73 occurred. Specifically, several safety systems actuated as a result of a trip of the supply breaker to emergency 600-volt bus "2C."

The Plant Hatch Class 1E electrical power distribution system is divided into redundant and independent electrical power distribution subsystems. The primary AC distribution system consists of three 4.16 kV engineered safety feature buses (EIIS Code EB) each having an offsite source of power as well as a dedicated onsite emergency diesel generator (EIIS Code EK) source. The secondary plant distribution system includes 600-volt emergency buses "2C" and "2D" and associated load centers and transformers. The Class 1E distribution system is divided into redundant load groups so that loss of any one bus does not prevent the minimum safety functions from being performed.

In this event, one of the two 600-volt emergency buses in the secondary plant distribution system lost power when its supply breaker opened on a false over-current signal. Because the secondary plant distribution system is divided into redundant load groups, the minimum equipment necessary to mitigate the consequences of analyzed transients and accidents remained powered by the second, and unaffected, 600-volt emergency bus. Moreover, emergency 600-volt bus "2C" was re-energized quickly, that is, in approximately three minutes, restoring all but the station service battery chargers to service and returning the major safety systems to an operable status. Per their design, the station service batteries supplied DC

power to emergency loads, maintaining them operable, until the chargers were restored to service within 40 minutes of the loss of power to the 600-volt bus. Finally, all logic systems responded per their design upon loss of power, placing the affected systems, such as the Primary Containment Isolation and Standby Gas Treatment systems, into their fail-safe or accident conditions.

The unexpected increase in recirculation pump speed, and the corresponding increase in core flow and thermal power, that occurred when power was restored to the recirculation system master controller is an analyzed operational occurrence (AOO). Specifically, a reactor recirculation system controller failure resulting in increasing recirculation pump speed and core flow is analyzed in section 15.2.5.1 of the Unit 2 Final Safety Analysis Report. The Final Safety Analysis Report AOO analysis, which demonstrates that no fuel or cladding damage will occur as a result of a rapid increase in recirculation pump speed, envelopes this event. That is, the AOO analysis assumes that the speed of both reactor recirculation system pumps is increasing whereas in this event the speed of only one pump increased. Moreover, the AOO analysis assumes recirculation pump speed increases at a much greater rate (twenty-five percent per second) than is allowed by each pump controller's rate limiter (one percent per second) and than occurred in this event. Section 7.7.2.2 of the Unit 2 Final Safety Analysis Report describes the rate limiter, indicating that its purpose is to allow "the control signal to slowly ramp up to the speed setpoint, if a runback is reset, without reducing the setpoint" on the controller. The rate limiter functioned as designed to limit the ramp back to the speed demand setpoint following a restoration of power to, and a resetting of the forty-four percent speed signal from, the recirculation system master controller. This limited the resulting transient, ensuring it was much less severe than the analyzed operational occurrence.

Based on this analysis, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Plant calibration procedures 57CP-CAL-108-1S, "GE IAC and Westinghouse CO Overcurrent Relay," and 57CP-CAL-108-2S were revised to prohibit the storing or placement of the connecting plug in a relay case while the relay is removed. These revisions were effective 03/12/2001.

The elimination of the practice of storing connecting plugs in the relay cases while the relay is removed also has been communicated verbally to the appropriate Maintenance Department personnel, including electricians and Instrument and Control technicians.

Temporary modifications were implemented on the Unit 1 and Unit 2 reactor recirculation system master controllers to set them to the "cold start" mode. These modifications will become permanent as part of the routine design change process.

Unit 1 and Unit 2 controllers of the type used in the reactor recirculation system will be reviewed to ensure that their settings result in the desired behavior upon loss of power and upon restoration of power and that the settings of the controllers in the plant match those of the controllers in the simulator.

ADDITIONAL INFORMATION

1. Other Systems Affected: No systems other than those mentioned in this report were affected by this event.

2. Failed Components Information: No failed components caused or resulted from this event.

3. Commitments: No permanent commitments are created as a result of this report.

4. Previous Similar Events: There has been only one previous similar event reported in the last two years in which a trip of a power supply breaker resulted in unplanned safety system actuations. This event, reported in Licensee Event Report 50-366/2000-008, dated 09/27/2000, occurred when an electrician inadvertently bumped a breaker control switch causing the 600-volt supply breaker to the "2A" RPS motor-generator set to open. The motor-generator set coasted down until its own output breakers tripped on under-frequency per design. The trip of the output breakers caused some Group 2, Group 5, and outboard small-bore Group 1 PCIS valves to receive an automatic isolation signal. Those valves open at the time of the event closed per design. Also, the MCRECS entered the pressurization mode; the Primary Containment Hydrogen and Oxygen Analyzers isolated; and the RWCU system, the FPM system, and the operating steam packing exhauster tripped. However, the causes and circumstances of, and the personnel involved in, the two events were different. Therefore, the corrective actions for the previous similar event could not have prevented this event.