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HL-6156

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

Edwin I. Hatch Nuclear Plant - Unit 2
Licensee Event Report
Reactor Recirculation Pump Flow Rate Changes
Cause Reactor Scram on APRM High Flux

Ladies and Gentlemen:

In accordance with the requirements of 10 CFR 50.73(a)(2)(iv)(A), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning reactor recirculation pump flow rate changes that cause a reactor scram on APRM high flux.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

CLT/eb

Enclosure: LER 50-366/2001-002

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

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NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7/31/2004

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Edwin I. Hatch Nuclear Plant - Unit 2	05000-366	1 OF 5

4. TITLE
Reactor Recirculation Pump Flow Rate Changes Cause Reactor Scram on APRM High Flux

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER(S)
									Plant Hatch Unit 1	05000321
10	26	2001	2001	002	0	12	14	2001	FACILITY NAME	DOCKET NUMBER(S) 05000

9. OPERATING MODE	1
10. POWER LEVEL	100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR § : (Check all that apply)

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)	

12. LICENSEE CONTACT FOR THIS LER

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

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	AD	ECBD	B042	Yes					

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

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

On 10/26/2001 at 1917 EDT, Unit 2 was in the Run mode. At that time, the reactor scrammed on Average Power Range Monitor high neutron flux caused by a rapid increase, after an unexpected decrease, in the flow rate of the 2B reactor recirculation pump. Following the scram, water level decreased due to void collapse from the rapid reduction in power resulting in closure of Group 2 and Group 5 primary containment isolation valves and automatic initiation of the Reactor Core Isolation Cooling and High Pressure Coolant Injection systems. The low level initiation signal cleared before either system could inject water to the vessel. The secondary containment dampers automatically isolated and all trains of the Unit 1 and Unit 2 Standby Gas Treatment systems automatically started on low water level. Level reached a minimum of 40 inches below instrument zero. The Reactor Feedwater Pumps restored level to its pre-event value of approximately 35 inches above instrument zero within 35 seconds of the scram. Reactor vessel pressure did not exceed its normal value of approximately 1035 psig; therefore, no safety/relief valves lifted nor were any required to lift to reduce pressure.

The cause of this event is not known conclusively, although the available evidence indicates a problem with the amplifier board in the 2B reactor recirculation pump scoop tube positioner logic circuit is the most likely cause. Corrective actions include replacing the amplifier board; temporarily requiring the 2B recirculation pump scoop tube to be locked when pump speed is not being changed; monitoring selected parameters in the pump speed control system; and performing a failure analysis of the amplifier board.

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 10/26/2001 at 1917 EDT, Unit 2 was in the Run mode. At that time, the reactor scrammed on Average Power Range Monitor (APRM, EIIS Code IG) high neutron flux after reactor power had increased to approximately 116.5 percent rated thermal power when, after an unexpected decrease, the flow rate of the 2B reactor recirculation pump (EIIS Code AD) increased rapidly. More specifically, the flow rate of the 2B reactor recirculation pump, unexpectedly and for no apparent reason, decreased from approximately 17.3 mlb_m/hr to 5.4 mlb_m/hr over 44 seconds (a rate of approximately 1.6 percent rated pump flow per second). Pump flow rate then returned to normal, that is, 17.3 mlb_m/hr, over six seconds (a rate of approximately 11.3 percent rated pump flow per second). The rapid pump flow rate increase caused reactor thermal power to increase to about 116.5 percent rated, with power reaching this point three seconds after the recirculation pump flow rate increase began, causing a reactor scram on APRM high neutron flux per design. The flow rate of the 2B reactor recirculation pump continued to increase after the scram, reaching its pre-event flow rate approximately 50 seconds after the unexpected flow rate decrease began and three seconds after the scram. The 2B recirculation pump remained at this flow rate until it received the expected runback signals following the scram and, along with the 2A reactor recirculation pump, automatically decreased flow rate to the #1 speed limiter (22 percent rated pump speed) as designed.

Following the automatic reactor scram, vessel water level decreased due to void collapse from the rapid reduction in power. Water level reached a minimum of approximately 40 inches below instrument zero (approximately 118 inches above the top of the active fuel) resulting in closure of the Group 2 and Group 5 primary containment isolation valves (EIIS Code JM) and automatic initiation of the Reactor Core Isolation Cooling (RCIC, EIIS Code BN) and High Pressure Coolant Injection (HPCI, EIIS Code BJ) systems. The secondary containment isolation dampers automatically closed and all four trains of the Unit 1 and Unit 2 Standby Gas Treatment (EIIS Code BH) systems (SGTS) automatically started.

The Reactor Feedwater Pumps (EIIS Code SJ) rapidly recovered reactor vessel water level, restoring level to its pre-event valve of approximately 35 inches above instrument zero within 35 seconds of the scram. As a result, the HPCI and RCIC system low water level initiation signals cleared before either system could inject makeup water to the reactor vessel.

Reactor vessel pressure did not exceed its normal value of approximately 1035 psig; therefore, no safety/relief valves lifted nor were any required to lift to reduce pressure.

CAUSE OF EVENT

The cause of this event is not known conclusively. In-situ testing and inspection of the 2B reactor recirculation pump flow controller and scoop tube positioner circuits and components revealed no problems or failures. However, the amplifier board in the 2B reactor recirculation pump scoop tube positioner logic circuit was sent to a vendor for additional testing and failure analysis, which revealed anomalies in board output caused by a failed transistor. Therefore, this and other available evidence indicates a problem with the amplifier board is the most likely cause of this event.

Investigating personnel developed a fault-tree to identify potential failure modes that could result in reactor recirculation pump flow rate changes. In parallel with this effort, maintenance personnel checked the condition and operation of pump flow controller and scoop tube positioner circuit components per Maintenance Work Order 2-01-03423 and procedures 57IT-B31-001-2S, "Reactor Recirc Scoop Tube Positioner Inspection and Test," and 57CP-CAL-278-2S, "Acromag Series 700 Instruments." No problems were noted; this was not unexpected since the anomaly cleared approximately 44 seconds into the event, allowing the 2B reactor recirculation pump to return to its pre-event flow rate. The rate at which the pump returned to its pre-event setting, however, served to eliminate one item on the fault tree: the pump flow rate controller. This controller contains a rate limiter that prevents pump speed from increasing at a rate faster than one percent per second. Since the pump's speed increased at a much greater rate, the problem could not have been in the controller unless the rate limiter coincidentally failed. This failure was eliminated because it was an unlikely coincidence and because the rate limiter was tested successfully per procedure 57CP-CAL-278-2S following the event.

A failure of the mechanical coupling between the motor and pump was considered as part of the fault-tree analysis. It, too, was eliminated as a cause because the connection between the pump and motor is bolted. While a failure of this type of connection could allow the motor to spin freely and the pump to coast down in speed, as was seen at the beginning of the event, no credible failure mechanism would result in the motor becoming re-coupled to the pump and the pump returning to its normal flow rate, as was seen in the latter stages of the event. It therefore was eliminated as a cause.

The fault-tree analysis also included consideration of a failure in the reactor recirculation pump motor-generator fluid drive coupler, that is, the hydraulic coupling. A momentary or intermittent failure of one of the two operating lube oil pumps could have resulted in a brief decrease in the hydraulic coupling between the drive motor and generator and a corresponding decrease in generator output and recirculation pump flow rate. Possible failure modes include a trip of one of the two pumps or foreign material temporarily clogging the pump suction. Investigating personnel eliminated the former failure mode when they found that none of the pump trip (amber) lights indicated a trip had occurred. The latter failure mode was considered improbable as the lube oil reservoir had not been opened recently, making it unlikely foreign material could have been introduced into the pump suction.

Another possible failure included the scoop tube positioner circuit. Available evidence indicated this was the most probable failure area and that an amplifier board in this circuit was the most likely failed component. The positioner circuit receives a signal from the recirculation pump flow controller and thus has no rate limiter. An intermittent failure in this circuit therefore could result in rapid changes in pump flow rate. In point of fact, this board failed on Unit 1 in 1990 and at another nuclear plant in 1985, resulting in similar flow rate and core thermal power changes. Moreover, the positioner circuit amplifier board had been replaced recently with a refurbished board, the only component to be replaced of late. The history of refurbished amplifier boards has been problematical with two other refurbished amplifier boards having failed, the first on Unit 1 in 2000 and the second on Unit 2 immediately after being installed during the 2001 refueling outage. Indeed, the refurbished amplifier board in the scoop tube positioner circuit at the time of this event replaced the refurbished board that failed upon installation. Finally, maintenance and engineering personnel inspecting the amplifier board after its removal found that three of its transistors showed indications of physical damage.

Maintenance personnel on 11/07/2001 sent the amplifier board to a vendor for failure analysis. Test results, reported on 11/13/2001 and 11/15/2001, indicated a problem with a transistor sufficient to affect adversely the output of the board. Specifically, a transistor was causing the amplifier board output "to drive in the downscale

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL YEAR	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 2	05000-366	2001	-- 002 --	00	4 OF 5

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

direction." However, testing of the amplifier board failed to duplicate completely the event: testing disclosed a decreasing output signal, but failed to reproduce a subsequent increasing output signal. Nevertheless, the investigation team members concluded, based upon a preponderance of the available evidence, that a problem with the amplifier board in the scoop tube positioner circuit was the most likely cause of this event.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv)(A) because of the unplanned actuation of reportable systems. Specifically, the reactor protection system (EIIS Code JC) actuated on APRM high neutron flux. Group 2 and Group 5 primary containment isolation valves closed and the RCIC and HPCI systems initiated as a result of the expected reactor vessel water level decrease following the scram.

Reactor power level changes can be made through changes in reactor recirculation pump speed (flow rate), which result in changes in core flow. An increase in recirculation pump flow rate temporarily reduces the void content of the moderator by increasing the flow of coolant through the core. The additional neutron moderation increases the reactivity of the core, which causes the reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new, higher steady-state power level is established. When recirculation pump flow rate is reduced, the core thermal power level is reduced.

Each recirculation pump motor has its own motor-generator set for a power supply. A variable speed converter is provided between the motor-generator set drive motor and generator. To change the speed of the recirculation pump, the variable speed converter, that is, the scoop tube positioner, varies the generator speed which changes the frequency and magnitude of the voltage supplied to the pump motor to give the desired pump speed. A manually set signal from the master, or individual pump, controller adjusts the setting of the speed control system for each motor-generator set converter.

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (EIIS Code IG) within the reactor core to provide an indication of the power distribution and local power changes. The APRM channels average these local power range monitor signals to provide a continuous indication of average reactor power from a few percent to greater than rated thermal power. The APRM high neutron flux function is capable of generating a reactor protection system trip signal in sufficient time to prevent fuel damage or excessive reactor coolant system pressure.

In this event, the reactor scrammed on Average Power Range Monitor high neutron flux caused by a rapid increase, after an unexpected decrease, in the flow rate of the 2B reactor recirculation pump. All systems functioned as expected and per their design given the core thermal power, water level, and pressure transients caused by this event. Fuel cladding integrity was not jeopardized because of the rapid response of the APRMs to the neutron flux increase. This response resulted in a reactor scram before the increased energy from the fuel pellets could be transferred fully to the metal cladding. Additionally, reactor vessel water level was maintained well above the top of the active fuel throughout the event.

Based upon the preceding analysis, it is concluded this event had no adverse impact on nuclear safety. The analysis is applicable to all power levels.

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 2	05000-366	2001	-- 002 --	00	5 OF 5

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

CORRECTIVE ACTIONS

Maintenance personnel replaced the scoop tube positioner circuit amplifier board on 10/27/2001 per Maintenance Work Order 2-01-03423.

Operations management issued Operating Order OO-02-1001S on 10/27/2001 temporarily requiring either the 2B recirculation pump scoop tube to be locked when pump flow rate is not being changed or an operator to be stationed at the flow controller to observe the pump for proper operation.

Maintenance personnel, based upon work instructions developed by engineering personnel and per Maintenance Work Order 2-01-03430, connected recorders to monitor selected parameters in the 2B recirculation pump speed control and scoop tube positioner systems. Personnel connected the recorders on 10/28/2001 and, to date, no similar anomalies have been noted in any monitored parameter.

The plant will participate with the BWR Owners Group committee formed to address recirculation system reliability.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

This LER does not contain any permanent licensing commitments.

Failed Component Information:

Master Parts List Number: 2B31-S001B
Manufacturer: Bailey
Model Number: 6619848A1
Type: Board, Electrical Control
Manufacturer Code: B042

EIIS System Code: AD
Reportable to EPIX: Yes
Root Cause Code: X
EIIS Component Code: ECBD

Previous similar events in the last two years in which the reactor scrammed automatically while critical were reported in the following Licensee Event Reports:

50-321/2000-002, dated 2/25/2000
50-321/2000-004, dated 8/4/2000
50-321/2001-002, dated 5/21/2001

Corrective actions for these previous similar events could not have prevented this event because they involved different components and were the result of different causes.