

**Dennis R. Madison**  
Vice President - Hatch

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Docket No.: 50-366

NL-09-1222

U. S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant  
Licensee Event Report  
Main Generator Runback Due to High Stator  
Water Cooling Water Temperature Results in Reactor Scram

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 a reactor scram resulting from high stator water cooling temperature.

This letter contains no NRC commitments. If you have any questions, please advise.

Sincerely,

A handwritten signature in black ink, appearing to read "Dennis R. Madison".

D. R. Madison  
Vice President – Hatch

DRM/MJK/

Enclosure: LER 2-2009-003

cc: Southern Nuclear Operating Company  
Mr. J. T. Gasser, Executive Vice President  
Ms. P. M. Marino, Vice President – Engineering  
RTYPE: CHA02.004

U. S. Nuclear Regulatory Commission  
Mr. L. A. Reyes, Regional Administrator  
Mr. R. E. Martin, NRR Project Manager – Hatch  
Mr. J. A. Hickey, Senior Resident Inspector – Hatch

# LICENSEE EVENT REPORT (LER)

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

<b>1. FACILITY NAME</b> Edwin I. Hatch Nuclear Plant Unit 2	<b>2. DOCKET NUMBER</b> 05000 366	<b>3. PAGE</b> 1 OF 4
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**4. TITLE**  
Main Generator Runback Due to High Stator Water Cooling Water Temperature Results in Reactor Scram

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	20	2009	2009	- 003 -	0	08	10	2009		05000
									FACILITY NAME	DOCKET NUMBER
										05000

<b>9. OPERATING MODE</b>  1	<b>11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§:</b> <i>(Check all that apply)</i>									
<b>10. POWER LEVEL</b>  100	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)						
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)						
<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)							
<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER							
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A							

**12. LICENSEE CONTACT FOR THIS LER**

FACILITY NAME Edwin I. Hatch / Steve Tipps, Principal Licensing Engineer	TELEPHONE NUMBER (Include Area Code) 912-537-5880
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**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	TJ	TCV	F130	Y	B	TJ	23	F130	Y

<b>14. SUPPLEMENTAL REPORT EXPECTED</b>				<b>15. EXPECTED SUBMISSION DATE</b>		
<input type="checkbox"/> YES (If yes, complete 15. EXPECTED SUBMISSION DATE)				<input checked="" type="checkbox"/> NO		
		MONTH	DAY	YEAR		

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

On June 20, 2009 at 14:17 EDT, Unit 2 was in mode 1 with an approximate reactor power of 2804 CMWTh. At this time protective circuitry initiated a main generator runback due to a high Stator Water Cooling (SWC) outlet temperature. The electro-hydraulic control (EHC) Mark VI cores processor began reducing turbine load by closing the turbine control valves (TCV) and subsequently opening the turbine bypass valves (TBV). Operations personnel began reducing reactor load in response to the main generator runback by reducing recirculation flow. The reactor power was reduced to approximately 66 percent reactor thermal power and could not be reduced any lower without the insertion of control rods. Due to turbine load reducing faster than the manual actions taken to reduce reactor load, reactor pressure began to increase. Pressure had reached the setpoint to actuate the Reactor Protection System, and a reactor scram was initiated due to high reactor pressure.

This event was caused by the improper set-up of the SWC temperature control instrument loop during 2R20 outage.

Corrective actions consist of: set-up of SWC valve 2N43F100 was corrected and the control system was calibrated, review of other work during the refueling outage was performed to ensure a similar condition does not exist on other equipment, and review of the Unit 1 SWC system was performed to confirm proper operation.

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CONTINUATION SHEET**

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

PLANT AND SYSTEM IDENTIFICATION

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

DESCRIPTION OF EVENT

On June 20, 2009 at 14:17 EDT, Unit 2 was in mode 1 with an approximate reactor power of 2804 CMWTh. At this time the control room received a 'GEN INLET TEMP HIGH' alarm followed within minutes by a 'GEN PROTECTION CIRCUIT ENERGIZED' alarm. This protective circuitry initiated a main generator (EISS Code TB) runback due to a high Stator Water Cooling (SWC, EISS Code TJ) outlet temperature. The electro-hydraulic control, Mark VI system (EHC, EISS Code TG) cores processor began reducing turbine load by closing the turbine control valves (TCV, EISS Code TA) and subsequently opening the turbine bypass valves (TBV, EISS Code TA). Operations personnel began reducing reactor load in response to the main generator runback by reducing recirculation flow (EISS Code AD). The reactor power was reduced to approximately 66 percent reactor thermal power and could not be reduced any lower without the insertion of control rods (EISS Code AA). Due to turbine load reducing faster than the manual actions taken to reduce reactor load, reactor pressure began to increase. Reactor pressure increased to a high of 1074 psig and the Reactor protection System (RPS) (EISS Code IG), initiated a reactor scram due to high reactor pressure. Reactor water level decreased resulting in a primary containment valve Group 2 (EISS Code JM) valve isolation per design. Safety Relief Valves (EISS Code SB) did not actuate nor were they required to based on the maximum reactor pressure reached.

CAUSE OF EVENT

The direct cause of this event was improper set-up of a valve controller.

During the investigation of the 'GEN INLET TEMP HIGH' alarm, it was found that the generator SWC heat exchangers were on full bypass when they should have been providing flow through the heat exchanger given the current system operating temperature. When the valve was manually stroked by maintenance, it was noticed that the piping temperatures down stream of the valve immediately dropped. From review of the trend data taken by rounds it was determined that the valve was not maintaining the SWC temperature by design. Therefore, the cause of this event was failure of valve 2N43F100 (SWC Temperature Controller) to properly control the SWC temperature.

The SWC temperature instrument loop consists of (2N43F100) a Fisher model 667-Y AOV three way valve with a model 3582 pneumatic valve positioner. A Fisher 4160B temperature controller feeds the instrument air signal to the above positioner and controls the SWC temperature by its established set points. The function of this instrument loop is to control the inlet SWC temperature within a given operating range. The valve controls this temperature by opening a bypass around the SWC heat exchangers when SWC inlet temperatures begin to drop or vice-a-versa when the temperature increases. This function

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was lost when maintenance was performed on the instrument loop during the 2R20 refuel outage. The positioner and the controller were not set so that the valve would function per design. Both the positioner for valve 2N43F100 and its temperature controller 2N43R310, proportional band were set up as "Direct" acting. In this configuration, the valve responded opposite of its design. Therefore; due to seasonal increases in river temperature, the temperature control set points were reached and the SWC flow was automatically placed on full bypass. Removal of system heat exchangers from the system caused a rapid heat build up in the stator cooling water since the generator was online. When the system went on bypass, the generator stator bar temperatures increased to a point where the generator runback set point was reached.

**REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT**

This event is reportable per 10 CFR 50.73(a)(2)(iv)(A) because unplanned actuations of a safety feature system listed in 10 CFR 50.73 occurred. In this instance, a reactor protection system (RPS) actuation resulting in a reactor scram.

An increase in reactor vessel pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and thermal power transferred to the reactor coolant to increase which could challenge the integrity of the fuel cladding and the reactor coolant pressure boundary. Therefore, the reactor is shut down automatically on high reactor vessel steam dome pressure to limit the neutron flux and thermal power increase. The automatic reactor shutdown on high pressure, along with the SRVs, limits the peak reactor vessel pressure to less than the American Society of Mechanical Engineers Section III Code limits.

In this event, protective circuitry initiated a main generator runback due to a high SWC outlet temperature. EHC, Mark VI, system began reducing turbine load by closing the turbine control valves (TCV) and subsequently opening the turbine bypass valves. Operations personnel began reducing reactor load in response to the main generator runback by reducing recirculation flow. The reactor power was reduced to approximately 66 percent reactor thermal power and could not be reduced any lower without the insertion of control rods. Due to turbine load reducing faster than the manual actions taken to reduce reactor load, reactor pressure began to increase. Pressure increased to a high of 1074 psig and the RPS initiated a reactor scram due to high reactor pressure. Reactor water level decreased resulting in primary containment Group 2 valve isolation per design. No SRVs opened nor were any required to open to limit or reduce reactor vessel pressure. No Emergency Core Cooling Systems actuated nor were any required to actuate to recover or maintain water level during or following this event. All automatic functions operated per design in response to the pressure increase and the automatic reactor shutdown.

Based on this analysis, it is concluded that this event had no adverse impact on nuclear safety.

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CORRECTIVE ACTIONS

The set-up of SWC valve 2N43F100 was corrected and the control system was calibrated.

An instrument loop check on the valve was performed and it was confirmed that the valve will operate as designed on increase and decrease in temperature.

A review of work performed during the recent refuel outage was performed to determine if there was a potential for a similar configuration to exist on another valve. It was determined that no other vulnerabilities of this nature exist.

Plant personnel confirmed that the Unit 1 SWC instrument loop is set up so it will function as designed.

ADDITIONAL INFORMATION

Other Systems Affected: None

Failed Components Information:

Master Parts List Number: 2N43F100  
 Manufacturer: Fisher Controls (F130)  
 Model Number: 3582  
 Type: Valve Positioner

EIIS System Code: TJ  
 Reportable to EPIX: Yes  
 Root Cause Code: B  
 EIIS Component Code: TCV

Failed Components Information:

Master Parts List Number: 2N43R310  
 Manufacturer: Fisher Controls (F130)  
 Model Number: 4160B  
 Type: Temperature Controller

EIIS System Code: TJ  
 Reportable to EPIX: Yes  
 Root Cause Code: B  
 EIIS Component Code: 23

Commitment Information:

This report does not create any new permanent licensing commitments.

Previous Similar Events:

There are no similar events within the past two years in which a valve control system was set-up improperly that resulted in a reactor scram.