RAS 12261

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SOUTHERN COMPANY
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Edwin I. Hatch Nuclear Plant - Unit 1
Licensee Event Report
Component Failure Causes Turbine Trip and Reactor Scram

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 component failure which resulted in a turbine trip and reactor scram.

Respectfully submitted,

H. L. Stunner, Jr.

OCV/eb

Enclosure: LER 50-321/2000-004

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

U.S. Nuclear Regulatory Commission, Washington D.C.
Mr. L. N. Olshan, Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

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RULEMAKINGS AND
ADJUDICATIONS STAFF

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee LLC
Docket No. 50-271 Official Exhibit No. Entergy 11
OFFERED by: Applicant/Licensee Intervenor ______
NRC Staff Other ______
IDENTIFIED on 9/13/06 Witness/Panel Nichols/Casillas
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk MAC

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SECY-02

NRC FORM 366 (06-1998) U.S. NUCLEAR REGULATORY COMMISSION LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	APPROVED BY OMB NO. 3150-0104 EXPIRES 06/30/2001 Estimated burden per response to comply with this mandatory information collection request: 50 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. If a document 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.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	1 OF 6

TITLE (4)
Component Failure Causes Turbine Trip and Reactor Scram

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(S) 05000
07	10	2000	2000	004	00	08	04	2000		DOCKET NUMBER(S) 05000

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)			
POWER LEVEL (10)	99.7	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
		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)
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
X	TA	VT	G080	Yes					

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 07/10/2000 at 1050 EDT, Unit 1 was in the Run mode at a power level of 2754 CMWT (99.7 percent rated thermal power). At that time, the reactor scrammed and the reactor recirculation pumps tripped automatically on turbiie stop valve fast closure caused by a turbine trip. The turbine tripped when the vibration instrument on the #10 bearing failed causing a false high vibration trip signal to be generated. Following the reactor scram, water level decreased due to void collapse from the rapid reduction in power. However, the reactor feedwater pumps maintained water level higher than seventeen inches above instrument zero. Consequently, no safety system actuations on low level were received nor were any required. Pressure reached a maximum value of 1128 psig; nine of eleven safety/relief valves lifted to reduce reactor pressure. Pressure did not reach the nominal actuation setpoints for the remaining two safety/relief valves. The temperature in the vessel bottom head region decreased by more than the Technical Specification allowed 100°F in one hour before a recirculation pump could be re-started.

This event was caused by component failure. The vibration instrument on the #10 bearing failed, generating a false high vibration signal. The high vibration signal caused the main turbine to trip, producing a reactor scram on turbine stop valve fast closure per design. The failed vibration instrument was replaced. The vibration instruments on the remaining bearings were checked resulting in the replacement of the shaft rider probe on the #6 bearing. No'other instrument problems were found.

FACILITY NAME (1)	DOCKET	LER NUMBER (6) YEAR	SEQUENTIAL YEAR	REVISION NUMBER	PAGE (3)
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2000	004	00	2 OF 6

EXT (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 07/10/2000 at 1050 EDT, Unit 1 was in the Run mode at a power level of 2754 CMWT (99.7 percent rated thermal power). At that time, the reactor automatically scrammed and the reactor recirculation pumps (EIIS Code AD) automatically tripped on turbine stop valve (EIIS Code TA) fast closure caused by a main turbine (EIIS Code TA) trip. The main turbine tripped when the vibration instrument on the #10 bearing, the main generator exciter (EIIS Code TB) outboard bearing, failed. The instrument failure produced a false high bearing vibration signal, causing the main turbine to trip automatically on high bearing vibration. The turbine trip resulted in fast closure of the turbine stop valves. Turbine stop valve fast closure is a direct input to the reactor protection system (EIIS Code JC) logic system.

Following the automatic reactor scram, vessel water level decreased due to void collapse from the rapid reduction in power. However, the reactor feedwater pumps (EIIS Code SJ) continued to operate limiting the drop in water level. The minimum water level reached during this event was eighteen inches above instrument zero (176.44 inches above the top of the active fuel), a decrease of approximately 19 inches from a normal level of 37 inches above instrument zero. Vessel water level did not decrease to the actuation setpoint of three inches above instrument zero. Thus, no safety system, including emergency core cooling system, actuations on low water level were received nor were any required.

Vessel pressure reached a maximum value of 1128 psig after receipt of the scram. Nine of the eleven safety/relief valves actuated to reduce reactor pressure. Vessel pressure did not reach the nominal actuation setpoint of 1140 psig for safety/relief valves 1B21-F013E and 1B21-F013J; therefore, they did not actuate nor were they required to actuate. (Although safety/relief valve 1B21-F013B has a nominal setpoint of 1140 psig, it actuated during this event. The maximum vessel pressure of 1128 psig was within its Technical Specification-allowed setpoint tolerance of 1115.5 psig to 1184.5 psig. Therefore, the safety/relief valve functioned properly during the event.) As vessel pressure was reduced below its pre-event value of 1034 psig, all but the four low-low set safety/relief valves closed. The low-low set safety/relief valves closed as vessel pressure decreased to 883 psig, 874 psig, 859 psig, and 843 psig, respectively.

Non-emergency 4160-volt bus 1B failed to trader automatically from its normal to its alternate supply as expected when the main turbine tripped. Operations personnel manually energized the bus, which provides power to the 1B reactor recirculation pump, from its alternate supply at 1115 EDT.

The reactor coolant temperature in the vessel bottom head region, as measured by the vessel bottom head drain line temperature, decreased by 180°F in one hour. Unit 1 Technical Specification Limiting Condition for Operation 3.4.9 limits the reactor coolant system cooldown rate to a maximum of 100°F in one hour.

Because the temperature difference between the bottom head coolant temperature and the reactor coolant temperature in the steam dome exceeded the maximum allowed by Unit 1 Technical Specifications Surveillance Requirement SR 3.4.9.3, the reactor recirculation pumps could not be restarted. Therefore, the bottom head coolant temperature continued to decrease as expected, albeit at a rate within the 100°F per hour limit.

CAUSE OF EVENT

This event was caused by component failure. The vibration instrument on the #10 bearing, the main generator exciter outboard bearing, failed when a solder connection inside the shaft rider probe came apart. This created a loose wire that made intermittent contact with a coil within the probe. The loose wire contacted the coil such that a false high vibration signal was generated. The high vibration signal caused the main turbine to trip automatically, producing a reactor scram on turbine stop valve fast closure per design.

Non-emergency 4160-volt bus 1B failed to transfer automatically because its normal supply breaker was slow in opening. The automatic transfer logic requires the normal supply breaker to open within ten cycles (166.7 milliseconds). If the normal supply breaker does not open within the required time, the transfer logic prevents the alternate supply breaker from closing. The first test of the normal supply breaker performed after it had opened during the event revealed that the breaker opened in 124 milliseconds, nearly three times the procedural acceptance criterion of 45 milliseconds. Subsequent tests of the breaker indicated it would open faster the more it was exercised. For example, the breaker opened in 114 milliseconds during the third test and 91.6 milliseconds during the fourth test, a 26 percent improvement from the time recorded in the first test. Finally, testing revealed that actuation of the logic necessary to indicate that the normal supply breaker was open added 33 to 50 milliseconds to the transfer logic signal. Considering this additional time and the likelihood that the opening time of the normal supply breaker was greater than 124 milliseconds, investigating personnel concluded that the breaker opened too slowly, preventing transfer to the alternate power supply.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv) because of the unplanned actuation of Engineered Safety Feature systems. The reactor protection system, an Engineered Safety Feature system, actuated on turbine stop valve fast closure when the main turbine tripped on a false high bearing vibration signal. Both reactor recirculation pumps tripped also on turbine stop valve fast closure. Nine of eleven safety/relief valves opened on high vessel pressure; four of the valves continued to operate in the low-low set mode until pressure decreased to their respective closure setpoints.

Fast closure of the turbme stop valves is initiated whenever the main turbine tips. The turbme stop valves close as rapidly as possible to prevent overspeed of the turbine-generator rotor. Valve closing causes a sudden reduction in steam flow that, in turn results in a reactor vessel pressure increase. If the pressure increases to the pressure

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL YEAR	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2000	004	00	4 OF 6

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

relief setpoints, some or all of the safety/relief valves will briefly discharge steam to the suppression pool (EIIS Code BL).

Reactor scram and recirculation pump trip initiation by turbine stop valve fast closure prevent the core from exceeding thermal hydraulic safety limits following a main turbiie trip. Closure of the turbiie stop valves results in the loss of the normal heat sink (main condenser) thereby producing reactor pressure, neutron flux, and heat flux transients that must be limited. A reactor scram is initiated on turbine stop valve fast closure in anticipation of these transients. The scram, along with the reactor recirculation pump trip system, ensures that the minimum critical power ratio safety limit is not exceeded.

The recirculation pump trip system, upon sensing a turbiie stop valve fast closure, trips the reactor recirculation pumps, resulting in a decrease in core flow. The rapid core flow reduction increases void content and reduces reactivity in conjunction with the reactor scram to reduce the severity of the transients caused by the turbine trip.

In this event, the main turbine tripped on a false high bearing vibration trip signal. The turbine trip actuated the reactor protection system and scrammed the reactor. All systems functioned as expected and per their design given the water level and pressure transients caused by the turbiie trip and reactor scram. Vessel water level was maintained well above the top of the active fuel throughout the transient and indeed never decreased to the Level 3 actuation setpoint. Because the water level decrease was mild, no safety system actuations on low water level were received nor were any required.

Typically, the bottom head region of the pressure vessel experiences rapid cooling following a scram coincident with a trip of the reactor recirculation pumps. This cooling is the result of the loss of effective water mixing due to the trip of the recirculation pumps and increased cold water flow from the control rod drive (EIIS Code AA) system following a scram. In this event, the temperature in the vessel bottom head region decreased by 180°F in one hour. However, a bounding analysis indicated cooldown up to 397.7°F in one hour will not place unacceptable stress on components of the reactor coolant system.

Based upon the preceding analysis, 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 1	05000-321	2000	004	00	5 OF 6

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

CORRECTIVE ACTIONS

The vibration instrument for the #10 bearing was replaced on 7/12/2000 per Maintenance Work Order 1-00-02145. Additionally, the remaining vibration instruments were checked on 7/12/2000 per Maintenance Work Order 1-00-02159. As a result of this inspection, the shaft rider probe of the vibration instrument for the #6 bearing was replaced. No problems were found with any of the other bearing vibration instruments.

The high bearing vibration trip from the #9 and #10 bearings, with the concurrence of the turbine vendor, has been temporarily disabled. The final disposition of the main turbine high bearing vibration trips will be determined through the corrective action program.

Personnel assessed the effects of the excessive cooldown rate on the reactor coolant system. An evaluation performed by General Electric in May 1994 (NEDC-323 19P) was used in assessing the effects of this event. The May 1994 evaluation, intended to eliminate the need to perform an evaluation for each specific event, demonstrated that reactor pressure vessel cooldown rates up to 397.7°F per hour were acceptable provided certain bounding conditions were met. General Electric and Southern Nuclear personnel reviewed the May 1994 evaluation and concluded that the cooldown of 180% in one hour experienced during this event was bounded by the generic evaluation. Therefore, personnel determined that the Unit 1 reactor coolant system was acceptable for operation.

The normal supply breaker for non-emergency 4160-volt bus 1B was removed and replaced with a refurbished breaker on 7/12/2000 per Maintenance Work Order 1-99-04564. A fast transfer functional test of the newly installed normal supply breaker was completed successfully.

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: 1N3 1-N892
Manufacturer: General Electric
Model Number: 3S7700VB100A1
Type: Vibration Transmitter
Manufacturer Code: GO80

EIIS System Code: TA
Reportable to EPIC: Yes
Root Cause Code: X
EIIS Component Code: VT

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL YEAR	REVISION NUMBER	
Edwin I. Hatch Nuclear Plant - Unit 1	05000-321	2000	– 004	– 00	6 OF 6

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

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

50-321/1999-003 dated 6/1/1999
50-321/2000-002 dated 2/25/2000
50-366/1999-005 dated 5/27/1999
50-366/1999-007 dated 7/27/1999

Corrective actions for these previous similar events could not have prevented this event because their causes were different. Specifically, none of the other previous similar events was the result of an instrument failure. Indeed, only one of the previous four events was caused by a main turbine trip. In that event, reported in Licensee Event Report 50-366/1999-005, the main turbiie tripped when the main generator tripped on an actual ground fault. Therefore, any corrective actions taken for the previous events would not have addressed turbiie bearing vibration instruments.