RAS 12262

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SOUTHERN COMPANY
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Docket No. 50-321

HL-6088

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

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)(A), Southern Nuclear Operating Company is submitting the enclosed Licensee Event Report (LER) concerning a component failure which caused a turbine trip and reactor scram.

Respectfully submitted,

H. L. Sumner, Jr.

DMC/eb

Enclosure: LER 50-321/2001-002

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.
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OFFICE OF THE SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

U.S. NUCLEAR REGULATORY COMMISSION
In the Matter of Entergy Nuclear Vermont Yankee L.L.C
Docket No. 50-271 Official Exhibit No. Entergy 12
OFFERED by: Applicant/Licensee Intervenor
NRC Staff Other
IDENTIFIED on 9/13/06 Witness/Panel Nichols/Casillas
Action Taken: ADMITTED REJECTED WITHDRAWN
Reporter/Clerk HAC

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NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

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

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

EVENT DATE (5) MONTH	DAY	YEAR	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	REPORT DATE (7) MONTH	DAY	YEAR	OTHER FACILITIES INVOLVED (8) FACILITY NAME	DOCKET NUMBER(S)
03	28	2001	2001	002	00	05	21	2001		05000
										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
X	EA	XFMR	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 03/28/2001 at 1853 EST, Unit 1 was in the Run mode at a power level of 2763 CMWT (100 percent rated thermal power). At that time, the reactor scrammed on turbine control valve fast closure caused by a turbine trip. The turbine tripped when actuation of phase 2 and 3 differential relays for unit auxiliary transformer 1B resulted in actuation of a lockout relay, generating a direct turbine trip signal. Following the scram, water level decreased due to void collapse from the rapid reduction in power resulting in closure of Group 2 and the outboard 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 outboard 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 37 inches below instrument zero. The Reactor Feedwater Pumps restored level to its pre-event value of approximately 35 inches above instrument zero within 30 seconds of the scram. Pressure reached a maximum value of 1127 psig; five of eleven safety/relief valves lifted to reduce pressure. Pressure did not reach the nominal actuation setpoints for the remaining safety/relief valves.

This event was caused by an internal fault in unit auxiliary transformer 1B. The fault occurred on the high side winding of transformer phase 3. The transformer was removed from service; its loads will continue to be supplied from their alternate supply until a new transformer can be procured and installed.

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 03/28/2001 at 1853 EST, Unit 1 was in the Run mode at a power level of 2763 CMWT (100 percent rated thermal power). At that time, the reactor automatically scrammed on turbine control valve (EIIS Code TA) fast closure caused by a main turbine (EIIS Code TA) trip. The main turbine tripped when actuation of phase 2 and phase 3 differential relays monitoring unit auxiliary transformer 1B (EIIS Code EA) resulted in actuation of lockout relay 87T1BX. Actuation of this lockout relay generated a direct turbine trip signal and the main turbine tripped per design. The turbine trip resulted in fast closure of the turbine control valves. Turbine control valve fast closure is a direct input to the reactor protection system (EIIS Code JC).

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 37 inches below instrument zero (approximately 121 inches above the top of the active fuel) resulting in closure of the Group 2 and outboard 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 outboard 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 30 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. Also, the inboard Group 5 primary containment isolation valve and the inboard secondary containment isolation dampers did not close because water level increased before all of the logic necessary to isolate the inboard valve and dampers sensed, and could actuate on, a low, water level condition.

Vessel pressure reached a maximum value of 1127 psig after receipt of the scram. Five of the eleven safety/relief valves actuated to reduce reactor pressure. Vessel pressure did not reach the nominal actuation setpoints of the remaining safety/relief valves; 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 1127 psig, however, 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, the low-low set safety/relief valves closed at 887 psig, 877 psig, 862 psig, and 847 psig, respectively. The main turbine bypass valves functioned to control vessel pressure thereafter, maintaining pressure below 975 psig.

CAUSE OF EVENT

This event was caused by an internal fault in unit auxiliary transformer 1B. An inspection revealed a turn-to-turn failure caused extensive damage to the high side winding of transformer phase 3. Although an Event Review Team investigated this event, the root causes of the transformer internal fault were not determined.

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

Some evidence gathered by the Event Review Team, that is, transformer winding temperatures from Main Control Room recorder 1N41-R900, six-month load voltage readings, and transformer operating history, appeared to indicate the possibility of a load-induced or cooling-related problem as the direct cause of the transformer fault. However, other evidence, such as the periodic recording of local transformer winding and oil temperature gauge readings, which indicated temperatures significantly lower than the recorder readings, and a successful check of transformer temperature switch operation, was inconsistent with this conclusion.

An internal transformer fault might have developed if contamination had been introduced in 1999 when part of phase 3 was re-wound as a result of a problem discovered during routine- testing of the transformer. However, the damage from the fault destroyed any evidence that might have existed. Therefore, it is impossible to confirm the presence, or lack, of contamination and to prove, or disprove, contamination as the direct cause of the internal fault in unit auxiliary transformer 1B. It should be noted that internal contamination almost certainly was not the cause of failures of the high side winding of transformer phase 3 in 1984 and 1999 due to the many years of in-service time between those failures, making it less likely to be the cause for this most recent similar failure.

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 actuated on turbine control valve fast closure when the main turbine tripped following the detection of a fault in unit auxiliary transformer 1B. Group 2 and outboard Group 5 primary containment isolation valves closed and the RCIC and HPCI systems initiated. Five 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 turbine control valves is initiated whenever the main turbine trips. The turbine control 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 relief setpoints, some or all of the safety/relief valves will briefly discharge steam to the suppression pool (EIIS Code BL).

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

In this event, the main turbine tripped when the unit auxiliary transformer lockout relay actuated on signals from the phase 2 and phase 3 differential current relays. 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 turbine trip and reactor scram. Vessel water level was maintained well above the top of the active fuel throughout the transient.

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.

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

CORRECTIVE ACTIONS

The unit auxiliary transformer was removed from service and taken to an off-site facility for further inspection. This inspection revealed extensive damage to the high side windings of phase 3 caused by a turn-to-turn fault. The transformer loads will continue to be supplied from their alternate power supply, startup transformer 1C (EIIS Code EA), until a new transformer can be procured and installed.

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: 1S11-S003
Manufacturer: General Electric
Model Number: NP 167B5 180
Type: Transformer
Manufacturer Code: GO80

EIIS System Code: EA
Reportable to EPIX: Yes
Root Cause Code: X
EIIS Component Code: XFMR

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
50-32 1/2000-004, dated 8/4/2000
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 they involved different components and were the result of different direct causes.

Similar failures of unit auxiliary transformer 1B occurred in 1984 and 1999. Specifically, the high side windings of phase 3 of the unit auxiliary transformer failed in August 1984 after approximately ten years of service; this event resulted in an unplanned automatic reactor scram while critical (Licensee Event Report 50-321/1984-015, dated 8/30/1984). The high side windings of this phase also failed a routine doble test in March 1999 after almost fifteen years of service; this problem was discovered before the windings had deteriorated to the point of causing an internal transformer fault. The transformer was completely rebuilt as a result of the former event. Part of the high side windings of phase 3 was rebuilt as a result of the latter event. In neither event were the root causes of the failure determined; therefore, the corrective action of repairing the transformer was not intended to address the causes of the failure and to prevent subsequent failures.