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February 16, 2001

U. S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 1
Docket No. 50-413
Licensee Event Report 413/2001-001 Revision 0

Attached please find Licensee Event Report 413/2001-001 Revision 0, entitled "Reactor Trip Caused by a Turbine Trip Due to Incomplete Troubleshooting Analysis". Questions regarding this Licensee Event Report should be directed to J. W. Glenn at 803-831-3051.

The only commitments in this Licensee Event Report are those described in the "Planned Corrective Actions" section.

Sincerely,

G. R. Peterson

G. R. Peterson

Attachment

IE22

xc:

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NRC FORM 366 (1-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001

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)
Reactor Trip Caused by a Turbine Trip Due to Incomplete Troubleshooting Analysis

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									FACILITY NAME	DOCKET NUMBER
01	17	2001	2001 -	001 -	00	02	16	2001		

OPERATING MODE (9)	1
POWER LEVEL (10)	99.8

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (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 Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	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)
J.W. Glenn, Regulatory Compliance	803-831-3051

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
G1h	TD	V	ASCO	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-spaced typewritten lines) **(16)**

On January 17, 2001 at 1618 hours with Catawba Unit 1 operating in Mode 1 at 99.8% power, a reactor trip caused by a turbine trip occurred. The Mechanical Trip Solenoid Valve (MTSV) in the Main Turbine Protection System had been replaced in late November 2000 during refueling outage 1EOC12. Since that time a mechanical "Sequence Halt" was experienced when Turbine Trip Testing was performed. Troubleshooting focused on the limit switch associated with the Mechanical Trip Piston and failed to note that the Mechanical Trip Piston was not fully reset. Replacement of the limit switch required adjustment of the limit switch to the Mechanical Trip Piston. This adjustment produced a false reset signal for the test circuitry while the mechanical trip assembly was not fully reset thus causing a Turbine trip. A root cause investigation determined that the root cause of the event was incomplete troubleshooting analysis. The MTSV was replaced. A planned corrective action is for Engineering to develop "at power" troubleshooting guidelines for components of the Main Turbine Controls and the Main Feedwater Pump Turbine Controls.

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

Background

Catawba Nuclear Station Unit 1 is a four loop Westinghouse Pressurized Water Reactor [EIIS:RCT]. Unit 1 was operating in Mode 1, "Power Operation" at 99.8% power immediately prior to this event. The event is being reported pursuant to 10CFR50.73(a)(2)(iv), (any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature including the Reactor Protection System)(RPS)[EIIS:JC].

Plant conditions immediately prior to the trip were: Reactor Power 99.8%, Turbine Load 1232 MWe, Reactor Coolant System (NC)[EIIS:AB] Tavg 585.0 degrees F., Reactor Coolant System Pressure 2235 psig, Reactor Coolant System Boron Concentration 939 ppm, Cycle Burnup 56.1 Effective Full Power Days.

The Turbine Emergency Trip System (ETS) is a part of the Main Turbine Hydraulic Oil System [EIIS:TD]. A reactor trip signal is generated by the Solid State Protection System (SSPS) [EIIS:JF] when 2 of 4 Turbine [EIIS:TRB] Electro-Hydraulic pressure switches sense pressure dropping below 550 psig when above the P-9 (Power Range Neutron Flux) interlock [EIIS:IEL], or upon 4 of 4 Turbine Stop Valves [EIIS:V] closing when above P-9.

The Turbine Emergency Trip System is designed to allow testing of both the Mechanical Trip Valve circuitry and the Electrical Trip Valve circuitry while on line. The mechanical and electrical portions are tested separately. While one is tested, turbine overspeed protection is provided by the other.

During startup from refueling outage 1EOC12 in late November 2000, while performing the Weekly Turbine Trip Test, a Turbine Control Panel mechanical "Sequence Halt" alarm was received. This alarm prevents completion of the Monthly and Weekly Turbine Trip test. All turbine trip functions had been verified to be fully satisfied. The cause of this problem was suspected to be a faulty limit switch associated with the Main Turbine Hydraulic Oil System mechanical trip piston position. Since the MTSV had just been replaced, it was assumed that it was not the cause of the problem. A work activity was planned to replace the limit switch on line.

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

At the time of this event, no systems, structures, or components were out of service that had any significant effect on the event.

Event Description (dates and approximate times)

1-17-2001	Morning	Maintenance technicians replaced limit switch 1LHLL1253 per work order 98334586-01.
1-17-2001	Afternoon	After replacing the limit switch, Procedure PT/1/B/4250/002A "Main Turbine Weekly Trip Test" was performed as a retest for the limit switch replacement. The "sequence halt" was still present. Troubleshooting determined that the limit switch actuating arm, as installed, could not travel enough to allow the switch to reset and close its contact. Maintenance noted that when the limit switch actuating arm was rotated manually slightly beyond the point where it had stopped, the limit switch reset. Maintenance and Engineering determined that if the limit switch arm were repositioned slightly, the travel would be sufficient to achieve reset.
1-17-2001	Afternoon	The limit switch arm was repositioned and PT/1/B/4250/002A was performed again. The sequence halt was received again. This time it was due to the limit switch not having sufficient travel to reach its trip position. It was determined that repositioning the limit switch actuating arm caused this because the limit switch was repositioned with it being in (what was believed to be) the "reset" position. When the limit switch was actuated and the actuating arm rotated toward its trip position, it did not travel enough to indicate tripped. Engineering and Maintenance determined that the limit switch actuating arm would need to be repositioned to its original position and that it would be necessary to reposition the mechanical trip piston linkage cam to the limit switch arm in order to achieve proper operation.

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

1-17-2001 1530	Engineering and Maintenance decided to "back-out" of the test as allowed by the procedure. The procedure includes a provision for "backing out" of the activity by following steps given in the procedure.
1-17-2001 1618	During the process of backing out of the procedure a Turbine Trip occurred because the limit switch indicated that the Mechanical Trip Piston was in the "reset" position while it was actually in the "tripped" position. The Turbine Trip caused a Reactor Trip.
1-17-2001 1618	Unit 1 entered Mode 3 "Hot Standby".
1-17-2001 1618	Main Feedwater (CF) [EIIS:SJ] isolation upon reactor trip with low Tavg occurred as designed and Main Feedwater Isolation Valves [EIIS:ISV] closed as designed.
1-17-2001 1618	An Auxiliary Feedwater System (CA)[EIIS:BA] automatic start signal was generated on an ATWS Mitigation System Actuation Circuitry (AMSAC) signal and both Motor [EIIS:MO] Driven Auxiliary Feedwater Pumps [EIIS:P] started as designed. There was no Turbine Driven Auxiliary Feedwater automatic start signal.
1-17-2001 1800	A team was formed to investigate the reactor trip.
1-17-2001 2006	Motor Driven Auxiliary Feedwater Pumps were secured.
1-18-2001 0805	During investigation of the event, Engineering noted that the Mechanical Trip Piston shaft was not achieving its full range of motion. Two possible causes were: 1) binding in the mechanical trip piston or 2) a problem with the mechanical trip solenoid [EIIS:SOL] valve.
1-18-2001 1700	A pressure gauge [EIIS:PI] was installed downstream of the mechanical trip solenoid valve and OPS attempted to reset the turbine. Pressure at the gauge was observed to slowly increase to 13 psig

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

	over about 30 seconds. The specification was that it should have increased immediately to 21 psig. This determined that the Mechanical Trip Solenoid Valve was not working properly.
1-18-2001 2140	The Mechanical Trip Solenoid Valve was replaced with a unit that had been rebuilt by a vendor service representative earlier in the day. Limit Switch 1LHLL1253 was adjusted.
1-18-2002 2234	Unit 1 entered Mode 2 "Startup".
1-19-2001 0015	Unit 1 entered Mode 1 "Power Operation".
1-19-2001 0540	PT/1/B/4250/002A "Main Turbine Weekly Trip Test" was performed successfully.
1-19-2001 0624	The Main Turbine was placed on line.

Causal Factors

An investigation by a root cause team determined that the root cause of the event was incomplete troubleshooting analysis. The Mechanical Trip Solenoid Valve (MTSV) in the Main Turbine Protection System had been replaced in late November 2000 during refueling outage 1EOC12. Since that time a mechanical "Sequence Halt" had been experienced each time a Turbine Trip Test was performed. Troubleshooting focused on the limit switch associated with the Mechanical Trip Piston and failed to note that the Mechanical Trip Piston was not fully reset. Replacement of the limit switch required adjustment of the limit switch to the Mechanical Trip Piston. This adjustment produced a false reset signal for the test circuitry while the mechanical trip assembly was not fully reset thus causing a Turbine trip.

There have been three other reactor trips within the previous twenty four months. Two of these were caused by a degraded electrical connector on the Turbine Electrical Trip Solenoid Valve. The other was caused by inadequate oversight of a modification to the Turbine Building roof. Corrective actions for these previous events would not have prevented this event. This event is not a recurring event.

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

Corrective Actions

Subsequent

1. The Mechanical Trip Solenoid Valve was replaced.

Planned

1. Engineering will develop at power troubleshooting guidelines that include independent review for components of the Main Turbine Controls and Main Feedwater Pump Turbine Controls.

Safety Analysis

This event is bounded by the analysis of the turbine trip transient in Section 15.2.3 of the Updated Final Safety Analysis Report. There is an insignificant effect on Core Damage Frequency associated with this event.

After the Reactor Trip, all plant systems functioned as designed except as noted below. Reactor parameters stabilized at normal no-load conditions forty minutes after the trip.

The SSPS functioned as designed upon receipt of a Turbine Trip signal (with Reactor power above 69%) by tripping the Reactor. Reactor Trip Breakers [EIIS:BRK] opened within the required timeframe. All Control Rods [EIIS:ROD] inserted normally.

A Main Feedwater Isolation signal was generated due to Reactor Trip with Low Tavg (</= 564 degrees F.) as designed. All valves associated with Main Feedwater isolation closed within the required time.

Reactor Coolant System Loop A Tavg initially tracked with the other three loops. Loop A began to lag behind the others at approximately 563 degrees F. decreasing. The maximum difference between loop A and the other loops was approximately 11 degrees F. When the plant stabilized at no load conditions, Loop A was approximately 5 degrees F. higher than the other loops. Loop A Tavg remained above the no load Tavg of 557 degrees F. while the other loops decreased to 547 degrees F. This failure affected steam dump performance since the steam dump controller uses auctioneered Hi Tavg. This problem was caused by oxidation on circuit card contacts. The problem was subsequently fixed and did not have a significant effect on

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

the event.

Primary System Pressure Control functioned normally. No Pressurizer [EIIS:PZR] Relief Valves (PORVs) [EIIS:RV] or Pressurizer Safety Valves lifted. Pressurizer Spray Valves and Backup Heaters [EIIS:HTR] controlled pressure as designed.

Secondary System Pressure Control functioned normally. No Steam Generator [EIIS:SG] PORVs or Safety Valves lifted. Condenser [EIIS:COND] Steam Dump Valves functioned as designed, considering the Loop A Tavg problem described above.

Main Feedwater Pump [EIIS:P] response was normal. Both Main Feedwater Pumps went into recirculation after isolation of Main Feedwater.

Auxiliary Feedwater System response was normal. The Turbine Driven Auxiliary Feedwater Pump did not start. Both Motor Driven Auxiliary Feedwater Pumps started automatically on an AMSAC signal as designed. Auxiliary Feedwater System flow to all of the four Steam Generators was within the acceptable range. Steam Generator levels remained in the normal operating band.

Reactor Coolant Pump performance was normal. All seal water leak off flows remained within range.

Pressurizer level control was normal. Level remained above the letdown isolation setpoint of 17%. The Chemical and Volume Control System makeup control functioned properly to maintain Volume Control Tank level.

Condensate System (CM) [EIIS:KA] response was normal.

Two Safety Parameter Display System alarms were received. These were:
1. Subcriticality in Yellow due to Intermediate Range Nuclear Instrumentation startup rate less negative than 0.2 dpm.
2. Subcriticality in Yellow due to Source Range positive startup rate when Source Range Nuclear Instrumentation was energized.

Both alarms cleared within the expected time. Both alarms are considered typical performance during a reactor trip.

All Safety Parameter Display System Critical Safety Functions were green approximately 14 minutes after the start of the event.

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There was no operation prohibited by the Technical Specifications associated with this event. This event is not a Safety System Functional Failure. The health and safety of the public were not affected by this event.