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December 15, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control
Washington D.C. 20555-0001

Subject: Duke Energy Corporation
Catawba Nuclear Station Unit 1
Docket No. 50-414
License Event Report 414/2004-002, Revision 0
Manual Reactor Trip Initiated due to Control Rods
From Shutdown Bank D Dropping into the Core

Attached please find License Event Report 414/2004-002, Revision 0 entitled, "Manual Reactor Trip Initiated due to Control Rods From Shutdown Bank D Dropping into the Core."

This Licensee Event Report does not contain any regulatory commitments. This event is considered to be of no significance with respect to the health and safety of the public. Questions regarding this License Event Report should be directed to A. Jones-Young at (803) 831-3051.

Sincerely,

D. M. Jamil

Attachment

IE22

xc:

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NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

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

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Catawba Nuclear Station, Unit 2	050- 00414	1 OF 6

4. TITLE

Manual Reactor Trip Initiated Due to Control Rods from Shutdown Bank D Dropping into the Core

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME: NA	DOCKET NUMBER
10	28	2004	2004	– 002	– 00	12	15	2004	FACILITY NAME	DOCKET NUMBER

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 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)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
A. Jones-Young, Regulatory Compliance	(803)831-3051

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
X6	IRE	2IRECM0046	WEST	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 October 28, 2004, at 0052 hours, with Catawba Unit 2 operating in Mode 1 at 100% power, all four control rods from Shutdown Bank D inserted into the core. Consequently, the Operators responded by manually tripping the reactor.

The plant response to the reactor trip remained within the limits of the Updated Final Safety Analysis Report. All plant equipment operated as expected.

The most probable cause of this event was an intermittent failure of a circuit card in the rod control system. Corrective actions for this event included replacing the suspect circuit card and sending the card to the vendor for further analysis.

This event was of no significance with respect to the health and safety of the public.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Catawba Nuclear Station, Unit 2	05000414	2004	- 002	- 00	2 OF 6

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

BACKGROUND

This event is being reported under 10 CFR 50.73 (a)(2)(iv)(A), Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature including the Reactor Protection System.

Catawba Nuclear Station (CNS) Unit 2 is a Westinghouse four-loop pressurized water reactor [EIIS: RCT]. Unit 2 was operating in Mode 1 (Power Operation) at 100% power prior to this event.

The CNS Unit 2 core contains 53 control rods [EIIS: ROD]. The control rods [EIIS: ROD] are grouped into four control banks and five shutdown banks. During Mode 1 operation, the shutdown banks are fully withdrawn. The shutdown banks are held in position by energizing the stationary gripper coils from the control rod power supply cabinets. Manual movement of the shutdown banks are based on input signals from the rod control logic cabinet.

There were no systems, structures or components out of service during this event that contributed to this event.

EVENT DESCRIPTION
(Dates and times are approximate)

10/28/04/0052:22	The Operator at the Controls (OATC) simultaneously received the following Annunciator alarms, "Pressurizer Low Pressure Control"; "Rod Control Urgent Failure"; "RPI at Rod Bottom Drop"; and "RPI Two or more Rods at Bottom".
10/28/04/0052	The OATC announced that all four rods of Shutdown Bank D had dropped and that he was manually tripping the reactor.
10/28/04/0052:38	The reactor was manually tripped per procedure.

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

10/28/04/0122	All plant control systems functioned as designed. Plant conditions stabilized at no load conditions.
10/28/04/0312	Required four hour notification to the NRC completed.
10/29/04/2037	Unit 2 Turbine Generator placed on line.
10/31/04/1447	Unit 2 returned to 100% power.

CAUSAL FACTORS

The reactor was manually tripped because all four control rods from Shutdown Bank D unexpectedly inserted into the core. The most probable cause of this event was an intermittent failure of a circuit card in the rod control system. Extensive testing of the circuit card was performed and the failure mode could not be duplicated. Technology is not currently available to detect intermittent failures. The consensus of Westinghouse and the failure investigation process (FIP) team was that the testing was inconclusive as far as ruling out the card as the cause of the failure because the card could be exhibiting a sporadic problem. Industry operating experience for Westinghouse Rod Control indicates that intermittent failures do occur. Based on the extensive testing that was performed on the card and the industry operating experience, the FIP team had reasonable assurance that the circuit card was the most likely cause and agreed to accept the Westinghouse recommendation to replace the circuit card prior to restart. The suspect circuit card was sent to Westinghouse for further analysis of the failure. Investigations into the failure of the rod control system with respect to the circuit card are inconclusive.

The circuit card has been installed since Unit 2 Startup and has not experienced any major problems. The card was last tested in the year 2000; however, checks are made on the rod control logic cabinet each refueling outage.

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

CORRECTIVE ACTIONS

Immediate:

1. Operations entered the reactor trip response procedure and plant conditions were stabilized at no-load conditions.

Subsequent:

1. The circuit card was replaced and the unit was returned to service.

2. The suspect circuit card was sent to Westinghouse for further analysis.

Planned:

1. Complete the root cause evaluation of the suspect circuit card.

2. Based on the results of the evaluation, Engineering will evaluate and initiate corrective actions, as appropriate, for the rod control circuit cards.

The planned corrective actions are being addressed via the Catawba Corrective Action Program. There are no NRC commitments contained in this LER.

SAFETY ANALYSIS

Safety Analysis Overview

The Solid State Protection System functioned as designed upon receipt of the manual reactor trip signal. The reactor trip breakers opened and all control rods and shutdown rods inserted as designed. Main turbine trip was initiated by a reactor trip signal as designed. Nuclear instrumentation response was normal following the trip.

Reactor coolant system pressure control functioned as expected. Pressurizer power operated relief valves and code safety valves were not challenged and did not lift during the event. The pressurizer spray valves and pressurizer heaters controlled primary system pressure as designed. Pressurizer level control functioned as expected.

Reactor coolant temperature control was by the condenser dump valves. Steam generator power operated relief valves and code safety valves were not challenged and did not lift during the event. The plant cooldown rate was less than the Technical Specification limit of 100 degrees F per hour.

Steam generator level control functioned as expected with makeup being provided by the motor driven auxiliary feedwater pumps. The Main Feedwater isolation system operated as designed.

In summary, the transient remained within the bounds of the Updated Final Safety Analysis Report and the post trip transient response was as expected.

The core damage significance of this event has been evaluated quantitatively by considering the following:

- A reactor trip initiating event
- Diesel Generator 1A in maintenance on the opposite unit at the time of the trip

The conditional core damage probability for the event is estimated to be 3E-7, which is less than the accident sequence precursor threshold of 1E-6.

The dominant base case Large Early Release Frequency (LERF) sequences for Catawba involve steam generator tube rupture, inter-system Loss of Coolant Accident (ISLOCA) and seismic-initiated sequences. In addition, the reliability of the important containment safeguards systems (containment spray and hydrogen mitigation) was not impacted by the reactor trip. Therefore, this event is not significant with respect to the LERF for Catawba.

This event was of no significance with respect to the health and safety of the public.

ADDITIONAL INFORMATION

Within the last three years, four other reactor trip events occurred from power operation at Catawba. They were as follows:

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

LER 414/01-003 described a Unit 2 reactor trip that resulted from low reactor coolant flow when the 2D reactor coolant pump 6900 VAC feeder breaker opened in response to a protective relay actuation caused by an electrical fault internal to the pump motor.

LER 413/03-001 described a Unit 1 reactor trip that resulted from a turbine trip. The turbine trip was due to steam generator high level. The root cause of this event was determined to be an inadequate understanding of the digital feedwater control system response to a common impulse line hydraulic interaction.

LER 413/03-005 described a Unit 1 trip that resulted when the two out of four trip logic for OTdT was satisfied. One channel of OTdT was previously tripped because of a reactor coolant system hot leg temperature detector failure. The second channel trip was caused by the failure of the pressurizer pressure loop power supply card.

LER 413/04-002 described a manual reactor trip of Unit 1 that resulted when the hydraulic system for the actuator on 1B Steam Generator Main Feedwater Isolation valve (1CF-42) failed and the valve closed.

The specific causes of the four events were unrelated. Therefore, this event was determined to be non-recurring in nature.

Energy Industry Identification System (EIIS) codes are identified in the text as [EIIS: XX]. The rod control system circuit card failure is an EPIX program reportable equipment failure.

This event does not reflect a manual reactor trip with a loss of secondary heat removal capability as monitored by the NRC performance indicator. Main Feedwater and Auxiliary Feedwater systems remained available. Condenser vacuum was maintained and condenser steam dump valves controlled reactor coolant system temperature throughout the event.

This event did not involve a Safety System Functional Failure. There were no releases of radioactive materials, radiation exposures, or personnel injuries associated with this event.