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A Duke Energy Company

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August 15, 2000

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

SUBJECT: Duke Energy Corporation
Catawba Nuclear Station Unit 2
Docket No. 50-414
Licensee Event Report 414/2000-003 Revision 1

Attached please find Licensee Event Report 414/2000-003 Revision 1, entitled "Reactor Trip Caused by Moisture Intrusion into Main Feedwater Pump 2B Speed Control Circuitry, and Operation Prohibited by Technical Specification 3.3.2". Questions regarding this Licensee Event Report should be directed to J. W. Glenn at 803-831-3051.

The only commitments in this report are those described in the "Planned Corrective Actions" section.

Sincerely,

G. R. Peterson

Attachment

IE22

xc:

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NRC FORM 366 (6-1998) 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

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TITLE (4)

Reactor Trip Caused by Moisture Intrusion into Main Feedwater Pump 2B Speed Control Circuitry,and Operation Prohibited by Technical Specification 3.3.2.

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
06	05	2000	2000	003	01	08	15	2000	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)							
POWER LEVEL (10)	100	20.2201(b)		20.2203(a)(2)(v)	X	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)			
		20.2203(a)(2)(iv)	X	50.36(c)(2)		50.73(a)(2)(vii)			

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	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
H11	SJ	SC		Yes		H11	NM	DRN	NA	Yes

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED	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 June 5, 2000 at 1237 hours with Catawba Unit 2 operating in Mode 1 "Power Operation" at 100% power, a reactor trip occurred. Rainwater entered the Turbine Building and affected control of Main Feedwater Pump 2B. A feedwater transient occurred which resulted in a reactor trip. The root cause of this aspect of the event is inadequate oversight of a Turbine Building roof modification. On June 8, 2000 from 0914 to 1414 hours with Unit 2 in Mode 1 at power levels ranging from 17% to 31%, there were eight separate occurrences prohibited by Technical Specification (TS) 3.3.2. This involved working on Main Feedwater Pump 2B without having it in the "tripped" condition as required by the TS. The root cause of this aspect of the event is inadequate communications between station groups. Corrective actions include measures to ensure that excessive rainfall can drain to the outside of the Turbine Building, instituting housekeeping controls for roofing work, defining and documenting oversight responsibilities of the job sponsor for roofing modifications, procedure revisions to clarify requirements for installation of jumpers and development of measures to address inadequate communications between station groups.

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Background

Catawba Nuclear Station Unit 2 is a four loop Westinghouse Pressurized Water Reactor [EIIS:RCT]. Unit 2 was operating in Mode 1, "Power Operation" at 100% 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 (ESF) [EIIS:JE], including the Reactor Protection System (RPS) [EIIS:JC]]; 10CFR50.73 (a)(2)(i)(B), (any operation or condition prohibited by the plant Technical Specifications; and 10CFR50.36(c)(2)(i) (limiting condition for operation of a nuclear power reactor not met).

Plant conditions immediately prior to the event were: Reactor Power 100%, Turbine Load 1219 MWe, Reactor Coolant System (NC)[EIIS:AB] Tavg 587.5 degrees F., Reactor Coolant System Pressure 2235 psig, Reactor Coolant System Boron Concentration 1187 ppm, Cycle Burnup 54.3 Effective Full Power Days.

The Main Feedwater System (CF)[EIIS:SJ] supplies feedwater to the four steam generators [EIIS:SG] at the temperature, pressure, and flow required to maintain proper steam generator water level. The system utilizes two steam turbine [EIIS:TRB] driven pumps [EIIS:P] which normally carry approximately equal shares of the feedwater flow requirement. The Feedwater Pumps are variable speed pumps, which can be controlled manually or automatically by the Feedwater Pump Turbine Speed Control System.

The Digital Feedwater Control System (DFCS) [EIIS:JB] controls main feedwater flow to the Steam Generators. The DFCS monitors several Main Feedwater System parameters in order to perform its function. When any of these parameters are out of their specified range the system switches control of the component from automatic to manual and generates a "DFCS Not in Auto" annunciator.

The current design of the Turbine Building [EIIS:NM] roof utilizes an interior gutter system to collect rainfall from the roof. Gutters that are integral to the building run along the edges of the Turbine Building roofs and have drain [EIIS:DRN] piping that extends from the gutters and runs down through the interior of the Turbine Building. A station modification is underway, during roof replacement, which will cover the gutter system and abandon it.

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

Surface mounted roof drains will be installed which will be connected to the existing gutter drain piping. The modification will also add scuppers (openings in the wall of the building through which water can drain from the roof) to drain any water that might collect in the abandoned guttering system until the modification is completed. Permanent scuppers will be added to the final roof configuration. Overflow water passing through the scuppers would drain to the outside of the Turbine Building.

The Auxiliary Feedwater System (CA)[EIIS:BA] is provided to supply feedwater to the Steam Generators to remove decay heat from the primary system whenever there is a loss of Main Feedwater. The Auxiliary Feedwater System has two motor [EIIS:MO] driven pumps and one steam turbine driven pump. The Motor Driven Auxiliary Feedwater Pumps will start automatically upon trip of both Main Feedwater Pumps.

Technical Specification 3.3.2 "Engineered Safety Features Actuation System (ESFAS) Instrumentation" Table 3.3.2-1 Function 5f. (Turbine Trip and Feedwater Isolation - Trip of all Main Feedwater Pumps) and Function 6e. (Auxiliary Feedwater - Trip of all Main Feedwater Pumps) refer to Condition K which requires that when one Main Feedwater Pump's trip channel [EIIS:CHA] is inoperable that the channel be placed in trip within one hour or the Unit must be placed in Mode 3 with seven hours. Since the TS does not specify a case for more than one inoperable channel, TS 3.0.3 would apply. TS 3.0.3 requires that action be initiated within one hour to place the Unit in Mode 3 within seven hours, Mode 4 within 13 hours, and Mode 5 within 37 hours.

ESFAS logic generates an Auxiliary Feedwater System Automatic Start and Turbine Trip whenever both Main Feedwater Pumps are tripped. During work on Main Feedwater Pumps, Procedure IP/1,2/B/3226/001A,B; Main Feedwater Pump Turbine Governor Calibration, directs Maintenance and Operations to determine if jumpers need to be installed to simulate a Main Feedwater Pump Trip.

At the time of the reactor trip, the A Train of the Auxiliary Feedwater System was inoperable due to the Nuclear Service Water System (RN)[EIIS:BI] to Auxiliary Feedwater System Suction Piping Flow Test. This test is performed periodically to ensure that the piping is able to pass the required amount of Nuclear Service Water, the assured source, to the suction of the Auxiliary Feedwater Pumps. In this alignment the normal condensate grade auxiliary feedwater sources are available although the system is declared inoperable because the assured source is not available. This inoperability had no actual effect on the event since an adequate source of condensate grade feedwater was available.

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Event Description (dates and approximate times)

6-5-2000 Modification CN-61511 was in progress on the Unit 2 Turbine Building roof. The modification changes the Turbine Building roof drainage system to utilize roof drains rather than gutters.

6-5-2000 Morning A very heavy rainfall occurred at the Catawba Site. Some foreign material, which had accumulated on the roof, was swept into the Turbine Building roof gutters. The foreign material partially blocked the gutter drains. The rainwater overflowed the gutter and entered the interior of the Turbine Building.

6-5-2000 1235 Water leaked onto the Main Feedwater Pump 2B area and affected the pump speed control circuitry. Feedwater Pump 2B speed increased and Feedwater Pump 2A speed decreased to compensate. Shortly thereafter Pump 2B speed decreased and the Pump 2A speed increased. Control room operators placed the Feedwater Pump Turbine Master Speed Controller in manual in an attempt to recover from the transient. This action stabilized pump speed.

6-5-2000 1236 The transient caused the DFCS to place the Feedwater Regulating valves in manual control. Levels in Steam Generators B, C and D began to rise.

6-5-2000 1237 A Hi-Hi Steam Generator Level (P-14) occurred in the 2B Steam Generator. The P-14 signal tripped both Main Feedwater Pumps, generated a feedwater isolation, and tripped the turbine. The Unit 2 Reactor tripped upon turbine trip. Auxiliary Feedwater System automatic starts occurred for Motor Driven Pumps A and B as well as the Turbine Driven Auxiliary Feedwater Pump (all of these responses were as designed). Operations entered Procedure EP/2/A/5000/E-0 "Reactor Trip or Safety Injection". Post trip conditions were normal. Unit 2 entered Mode 3 "Hot Standby".

6-5-2000	1248	The Turbine Driven Auxiliary Feedwater Pump was secured.
6-5-2000	After Noon	A plan was developed for recovery from the trip and for post trip maintenance. Part of the plan determined that procedure IP/2/B/3226/001B (Main Feedwater Pump 2B Turbine Governor Calibration) could be used to support post maintenance testing on Main Feedwater Pump 2B.
6-6-2000	Morning	"Recovery Work for Main Feedwater Pump 2B Action Plan" was issued by Engineering and work began per Work Orders 98282822 and 98275003. At this point work did not require tripping and resetting Main Feedwater Pump 2B.
6-6-2000	1800	Main Feedwater Pump 2A was placed in service.
6-6-2000	2136	Motor Driven Auxiliary Feedwater Pumps A and B were secured.
6-7-2000	0800 to 0830	Maintenance and the Control Room SRO discussed the planned work. At this time the Unit was in Mode 3 with Main Feedwater Pump 2A in service. Procedure IP/2/B/3226/001B Step 10.18.4 was discussed but not referred to in concluding that jumper installation would prevent an automatic start of the Auxiliary Feedwater System. The focus of this discussion was protecting the running Main Feedwater Pump 2A and preventing an inadvertent automatic start of the Auxiliary Feedwater System. After this discussion Step 10.18.4 (the step that would have required jumper installation was marked as "N/A" (not applicable)).
6-7-2000	0900	Maintenance work began again on Main Feedwater Pump 2B. Power supplies and circuit cards were replaced per the recovery plan and work continued through the nightshift. Checkout of the Main Feedwater Pump 2B governor began per IP/2/B/3226/001B. This work did not require tripping and resetting Main Feedwater Pump Turbine 2B.

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6-7-2000 2200 Unit 2 entered Mode 2 (Startup)

6-8-2000 0214 Unit 2 entered Mode 1 (Power Operation)

6-8-2000 0542 Unit 2 Generator Breakers were closed

6-8-2000 0914 to 1414 Unit 2 was operating in Mode 1 (Power Operation) with Main Feedwater Pump 2A in operation. Power levels were in the 17% to 31% range during this time. Main Feedwater Pump 2B was being calibrated per Procedure IP/2/B/3226/001B. During this work, there were eight separate occurrences in which Main Feedwater Pump 2B was placed in the "reset" condition (while the pump was uncoupled from the turbine) according to the procedure. The duration of these occurrences varied from one second to 34 minutes 26 seconds. Total time in reset was about 84 minutes. During the periods when the pump was reset, ESFAS Functions 5f. and 6e. of TS Table 3.3.2-1 would not have actuated had a trip of the operating Main Feedwater Pump 2A occurred. This resulted in eight separate violations of Technical Specification 3.3.2 and the corresponding unrecognized entries into Technical Specification 3.0.3.

6-8-2000 1330 to 1500 An engineer was in the Main Feedwater Pump 2B area and noticed that oil pressure indicator lights were on that should have been off if jumpers were in place per procedure IP/2/B/3226/001B. A discussion between the Engineer and Maintenance concluded that the jumpers should be installed. Maintenance and Operations personnel discussed the problem and concluded that the jumpers should be installed. The affected ESFAS circuitry was declared inoperable and logged in the Technical Specification Action Item Log.

6-8-2000 1500 Jumpers were installed per procedure IP/2/B/3226/001B, step 10.18.4. This restored the Auxiliary Feedwater System automatic start function upon loss of both Main Feedwater Pumps.

6-9-2000 1441 Main Feedwater Pump 2B repairs and testing were completed and the pump was placed in service.

6-10-2000 0130 Unit 2 returned to 100% power.

Causal Factors

The reactor trip was caused by rainwater entering the Unit 2 Turbine Building and affecting the speed control circuitry for the 2B Main Feedwater Pump Turbine. This caused a transient that resulted in a reactor trip on Hi-Hi level in Steam Generator 2B. A root cause analysis determined that the root cause of this event was inadequate oversight of the Turbine Building roof modification during implementation. Proper oversight of the roofing modification should have ensured adequate housekeeping so the gutters would not have become blocked during the rainfall. It is estimated that this rainfall was not much more intense than a "two year" storm. Therefore it is likely that many storms of similar intensity have occurred without causing a trip of the unit. Blockage of the drains resulted in much more water entering the Turbine Building than has occurred in previous storms. The amount of activity on the roof resulted in more loose foreign material on the roof that was transported to the roof drainage system.

In the past twenty-four months there have been two other automatic reactor trips. Those two trips were caused by degraded insulation in an electrical connector on the Turbine Electrical Trip Solenoid valve. This reactor trip is not similar to those two trips.

The root cause of the operation prohibited by TS 3.3.2 was determined to be inadequate communication between Engineering, Operations and Maintenance during planning and execution of recovery work of the Main Feedwater Pump 2B Action Plan. This led to not installing jumpers in the Main Feedwater Pump 2B control circuitry to maintain operability of the Auxiliary Feedwater System automatic start function and the Turbine Trip function for loss of two Main Feedwater Pumps, while testing was in progress on Main Feedwater Pump 2B.

In the past twenty-four months there has been one other event involving operation prohibited by technical specifications caused by inadequate communications between Engineering, Operations, and Maintenance. This event was LER 414/00-002 concerning disabling an Operator Aid Computer

Point that was required to be in service. Corrective actions from that event were directed at improving the Compensatory Action Program and would not have prevented this event.

Recent analyses from a Management Error Common Cause and an LER Assessment have identified communication between station groups as an area which needs improvement. Corrective actions are being developed that will address the recurring nature of this event.

An EPIX report will be submitted addressing the reactor trip and the functional failure of function CF-08 (Feedwater System) and function TR-19 (Turbine Building).

Corrective Actions

Subsequent

1. Foreign material was removed from the Unit 2 Turbine Building roof and gutters.

2. A temporary gutter cover of one half-inch square mesh plastic was installed to prevent foreign materials from migrating into the gutter.

3. Temporary scuppers were installed to direct any gutter overflow to the outside of the Turbine Building.

4. A "Stop Work" order on roofing work was issued until foreign material exclusion corrective actions could be taken.

5. Jumpers were installed to restore the Auxiliary Feedwater System automatic start function upon loss of both Main Feedwater Pumps.

Planned

1. Housekeeping controls for roofing work will be instituted to ensure that the roof remains free of debris that could clog gutters.

2. Oversight responsibilities of the roofing modification job sponsor will be defined and documented.

3. Procedures IP/1/B/3226/001 A, IP/2/B/3226/001 A, IP/1/B/3226/001 B, IP/2/B/3226/001 B (Calibration Procedures For Main Feedwater Pump Turbine Governors) and procedures IP/1/B/3226/003 A, IP/2/B/3226/003 A, IP/1/B/3226/003 B, and IP/2/B/3226/003 B (Calibration Procedures for IWE Stop Valve Interlock Limit Switch will be revised to ensure that requirements for installation of jumpers are clearly stated.

4. Based on the results of the Management Error Common Cause and the LER Assessment, corrective actions will be developed to address the problem of inadequate communications between station groups.

Safety Analysis

The reactor trip portion of the event is bounded by the analysis of the turbine trip transient in Section 15.2.3 of the Updated Final Safety Analysis Report. The core damage probability significance of this event has been evaluated by considering:

1. A Loss of Main Feedwater Initiating Event
2. No flow from the Nuclear Service Water System Train A to Motor Driven Auxiliary Feedwater Pump 2A if a swap occurred to the assured Nuclear Service Water source due to an orifice installed in the Nuclear Service Water to Auxiliary Feedwater line for flow testing.
3. Moderator Temperature Coefficient value at the time of the trip.
4. Actual plant configuration and maintenance activities at the time of the trip.

This event is little different from any loss of Main Feedwater Event. The inability to supply Nuclear Service Water to Auxiliary Feedwater Train A and therefore Motor Driven Auxiliary Feedwater Pump 2A has little effect on the overall sequence quantification. The conditional core damage probability for the event being evaluated is 9.6E-07, and is less than the accident sequence precursor threshold of 1.0E-06.

This event is not considered to be a "SCRAM with Loss of Normal Heat Removal" event because Main Feedwater Isolation is a normal expected plant response to a Reactor Trip.

The dominant core damage sequences associated with this event have the significant containment safeguards system available. These include the Containment Spray Systems and Hydrogen Mitigation Systems. Furthermore,

most have low to moderate reactor coolant system pressures at reactor vessel failure. Sequences of this nature contribute insignificantly to the large early release frequency (LERF), which is dominated by the ISLOCA and seismic initiating events. Therefore, this event is judged to have no significance with respect to LERF.

After the Reactor Trip, all plant systems functioned as designed. Reactor parameters stabilized at normal no-load conditions thirty minutes after the trip. The SSPS functioned as designed upon the 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 a P-14 signal. All Main Feedwater isolation valves closed within the required timeframe after receipt of this signal. A pressure transient occurred on the Main Feedwater System during the event. Feedwater heater [EIIS:HTR] relief valves [EIIS:RV] 2CF-64, 2CF-68, 2CF- 71, and 2CF-74 were damaged during the event. Valves 2CF-64 and 2CF-68 were replaced and valves 2CF-71 and 2CF-74 were repaired before the unit was restarted.

Primary System Pressure Control functioned normally. No Pressurizer [EIIS:PZR] Relief Valves (PORVs) or Pressurizer Safety Valves lifted. Pressurizer Spray Valves and Backup Heaters [EIIS:HTR] controlled pressure as designed. Pressurizer spray valves opened before the nomimal setpoint but during a rapid increase in Pressurizer pressure.

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.

Both Main Feedwater Pumps tripped on the P-14 signal.

Auxiliary Feedwater System response was normal. Both Motor Driven Auxiliary Pumps started automatically on loss of both Main Feedwater Pumps. Auxiliary Feedwater Pump Turbine #2 started on low level in two Steam Generators following Main Feedwater isolation. Auxiliary Feedwater flow to the four Steam Generators was within the acceptable range.

Condensate System response was normal.

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There is no safety significance to the TS Violation aspect of the event. Although the Engineered Safety Features Actuation System functions of tripping the Main Turbine and starting the Motor Driven Auxiliary Feedwater Pumps would not have functioned during the time periods when the Main Feedwater Pump 2B was in the "reset" condition, these ESFAS functions are anticipatory in nature and are not assumed to function in any accident analysis. Had an actual trip of the Main Feedwater Pump 2A occurred while the Main Feedwater Pump 2B was reset, the loss of Main Feedwater flow would have resulted in a Lo-Lo level condition in the Steam Generators. A reactor trip would have occurred as required by the Solid State Protection System on a Lo-Lo narrow range level condition in any Steam Generator, as described in UFSAR Section 15.2.7, Loss of Normal Feedwater Flow. In addition the Motor Driven and Turbine Driven Auxiliary Feedwater Pumps would have been automatically started by the ESFAS as a result of the Lo-Lo level condition. The Motor Driven Auxiliary Feedwater Pumps are automatically started on a Lo-Lo level condition in any Steam Generator, and the Turbine Driven Auxiliary Feedwater Pump is automatically started on a Lo-Lo level condition in any two Steam Generators. Following the reactor trip, the turbine would have been automatically tripped by the ESFAS as designed. (TS Table 3.3.2-1, Function 5d. is a turbine trip on a Tavg-low condition coincident with a reactor trip). The automatic turbine trip would have prevented any excessive primary system cooldown following the reactor trip. No single active failure would have prevented the fulfillment of any of these safely functions. The UFSAR Analysis for Section 15.2.7 demonstrates that a loss of normal feedwater will not result in the occurrence of a DNB condition.

The health and safety of the public were not affected by this event.