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A Duke Energy Company

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October 21, 2003

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 413/03-005 Revision 0
Reactor Trip due to Pressurizer Pressure Channel Failure

Attached please find Licensee Event Report 413/03-005 Revision 0, entitled "Reactor Trip due to Pressurizer Pressure Channel Failure."

This Licensee Event Report does not contain any regulatory commitments.

Questions regarding this Licensee Event Report should be directed to G. K Strickland at (803) 831-3858.

Sincerely,

D. M. Jamil

Attachment

IE22

xc:

L. A. Reyes
U. S. Nuclear Regulatory Commission
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700 Galleria Place
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NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

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 1	05000 413	1 OF 7

4. TITLE

Reactor Trip due to Pressurizer Pressure Channel Failure

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: None	DOCKET NUMBER
08	29	2003	2003	- 005 -	00	10	21	2003	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	95%

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)
G. K. Strickland, Regulatory Compliance	803-831-3585

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	EIA	EIACB2521	W120	Y					

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 August 29, 2003 at 0203 hours with Catawba Unit 1 operating in Mode 1 at 95% power, an automatic reactor trip occurred due to an instrument failure. The reactor trip was caused by the pressurizer pressure channel 2 failing low resulting in the reduction of the loop 1B overtemperature delta T (OTdT) setpoint to approximately 50%. When the OTdT setpoint reached the operating value of the loop 1B conditions (approximately 95%), an automatic reactor trip was initiated on 2/4 OTdT trip bistables. The loop 1A OTdT trip bistable was previously tripped due to a reactor coolant system hot leg temperature detector failure.

This event was caused by the equipment failure of the pressurizer pressure loop power supply card. The plant response to the reactor trip remained within the limits of the Updated Final Safety Analysis Report.

Corrective actions for this event included repairs to the pressurizer pressure channel and reactor coolant system hot leg temperature detectors.

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 Unit 1 is a Westinghouse Pressurized Water Reactor [EIIS: RCT]. Unit 1 was operating in Mode 1, "Power Operation" at 95% power immediately prior to this event. Reactor power was being maintained at 95% as a precautionary measure against possible spurious temperature spikes while loop 1A reactor coolant system temperature bistables were placed in the tripped condition. The bistables were tripped due to a failure of the hot leg temperature detectors.

The Reactor Protection System [EIIS: JC] is designed to open the reactor trip breakers to shutdown the reactor whenever a plant operating condition exceeds a specific setpoint. The OTdT reactor trip setpoint is designed to protect the reactor from departure from nucleate boiling (DNBR) conditions by calculating a trip setpoint based on pressurizer pressure, reactor coolant system temperature, and axial power distribution. When two of the four OTdT channels actuate, a reactor trip signal is generated.

With the exception of the loop 1A OTdT channel being in the tripped condition, there were no other systems, structures, or components out of service that had any significant effect on the event.

EVENT DESCRIPTION
(Dates and times are approximate)

Date/Time	Event Description
08/06/03	Loop 1A OTdT channel placed in tripped condition due to failure of the hot leg temperature detectors. The detectors were inadvertently damaged during surveillance testing.
08/29/03 ~0203	Pressurizer pressure channel 2 failed low resulting in an automatic reactor trip from OTdT.

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

	Main condenser dump valves operated as designed to reduce the reactor coolant system temperature to the no load value of 557 degrees F.
08/29/03 ~0222	During the transfer of the main condenser dump valve controls [EIIS: JI] to the steam pressure control mode, the valves opened for approximately 1 minute causing a plant cooldown to 548 degrees F.
08/29/03 ~0222	The plant cooldown to 548 degrees F caused the pressurizer level to decrease to approximately 16%. Letdown isolation occurred when the pressurizer level decreased below the isolation setpoint of 17%.
	During the letdown isolation, one of the letdown isolation valves, 1NV1A, did not close. The redundant series valves, 1NV10A and 1NV2A, closed and terminated letdown flow automatically.
08/29/03 ~0223	Pressurizer level restored above 17%.
	Condenser dump valves placed in the steam pressure control mode and operated as designed to control reactor coolant system temperature.
08/29/03 ~0247	Excess letdown placed in service.
	All other plant control systems operated as designed. Plant conditions stabilized at no load conditions.
08/29/03 ~0527	Required 4-hour notification to the NRC completed.
08/29/03 ~1904	Pressurizer pressure channel 2 repaired.
08/30/03 ~0104	Valve 1NV1A repaired and normal letdown placed in service. Excess letdown isolated.

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

09/02/03	Plant cooldown to Mode 5 and reactor coolant loop 1A hot leg temperature detectors replaced.
09/09/03~0612	Unit 1 placed on line.
09/10/03 ~1228	Unit 1 returned to 100% power.

CAUSAL FACTORS

The cause of the reactor trip was failure of the pressurizer pressure loop power supply card.

The cause of the condenser dump valves opening was the failure of the control room operator to adjust the controller correctly prior to transferring to the steam pressure control mode.

Engineering concluded that letdown isolation valve 1NV1A did not receive an automatic close signal because pressurizer level channel 2 remained slightly above the letdown isolation setpoint. Pressurizer level channel 1 decreased slightly below the isolation setpoint thereby generating a close signal for valves 1NV10A and 1NV2A and abating the decrease in pressurizer level. During troubleshooting of 1NV1A controls, a relay was discovered failed that prevented the operators from manually closing 1NV1A. The relay was repaired prior to plant startup.

CORRECTIVE ACTIONS

Immediate:

1. Plant conditions were stabilized at no-load conditions.

Subsequent:

1. Pressurizer pressure channel 2 was repaired and returned to operable status.

2. Reactor coolant system loop 1A hot leg temperature detectors were repaired and returned to operable status.

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

3. Letdown isolation valve 1NV1A relay was repaired.

4. Operator training was completed on transferring the condenser dump valves to the steam pressure control mode.

Planned:

1. Thermography measurements of the reactor protection cabinets will be scheduled on a six month frequency to detect suspect cards prior to failure. This frequency will be evaluated and adjusted based on results. Thermography readings were previously scheduled on a two-year frequency.

The planned corrective actions, as well as any future corrective actions, will be addressed through the Catawba Corrective Action Program. There are no NRC commitments contained in this LER.

SAFETY ANALYSIS

The Solid State Protection System functioned as designed upon receipt of the reactor trip signal. The reactor trip breakers opened within 150 milliseconds of the reactor trip signal as designed. All control rods and shutdown rods inserted as designed. The rod drop time was within the Technical Specification requirement. Nuclear instrumentation response was normal following the trip.

Reactor coolant system pressure control functioned as expected. No pressurizer power operated relief valves or code safety valves lifted. The pressurizer spray valves and pressurizer heaters controlled primary system pressure as designed. Pressurizer heaters were unavailable for a brief period of time when pressurizer level decreased below 17%. During the event, reactor coolant system pressure ranged from 2096 psig to 2232 psig.

Pressurizer level control functioned as expected with the exception of the failure of letdown isolation valve 1NV1A to close. This failure did not impact plant operations due to the operation of valves 1NV10A and 1NV2A. Pressurizer level control was maintained by normal charging, normal letdown initially, and excess letdown. Pressurizer level was maintained between 16% and 46%.

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

Reactor coolant temperature control was by the condenser dump valves. No steam generator power operated relief valves or code safety valves lifted. The plant cooldown rate was less than the Technical Specification limit of 100 degrees F per hour. The plant cooldown rate was less than 36 degrees F per hour.

Steam generator level control functioned as expected with makeup by the motor driven auxiliary feedwater pumps. The main feedwater isolation system operated as designed. Steam generator levels were maintained between 23% and 40%.

In summary, the post-trip transient response remained within the bounds of the Updated Final Safety Analysis Report.

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

Reactor trip initiating event
Actual plant configuration and maintenance activities at the time of the trip
Credit for recent Nuclear Service Water to Auxiliary Feedwater System modifications to minimize clam intrusion

The conditional core damage probability for the event being evaluated is estimated to be 8.3E-08, which is less than the accident sequence precursor threshold of 1.0E-06.

The dominant core damage sequences associated with this event involve loss of secondary side heat removal and failure of feed and bleed cooling. In contrast, the dominant Large Early Release Frequency (LERF) sequences for Catawba involve Inter-system Loss of Coolant Accidents (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 judged not to be significant with respect to the LERF.

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

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

ADDITIONAL INFORMATION

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

LER 413/01-001 described a Unit 1 reactor trip caused by a turbine trip. The root cause of this event was determined to be an incomplete troubleshooting analysis associated with the main turbine protection system mechanical trip solenoid valve.

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

LER 413/03-001 described a Unit 1 reactor trip resulting from a turbine trip. The turbine trip was due to a 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.

The corrective actions taken in response to these events would not have prevented this latest event from occurring. 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]. This event is reportable to the Equipment Performance and Information Exchange (EPIX) program.

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