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April 20, 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. 50-413
License Event Report 413/2004-002, Revision 0
Manual Reactor Trip Initiated due to the Closure of a Main Feedwater Isolation Valve

Attached please find License Event Report 413/2004-002, Revision 0 entitled, "Manual Reactor Trip Initiated Due to the Closure of a Main Feedwater Isolation Valve."

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 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Catawba Nuclear Station, Unit 1	05000 413	1 OF 8

4. TITLE
Manual Reactor Trip Initiated Due to the Closure of a Main Feedwater Isolation Valve

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	DOCKET NUMBER
02	22	2004	2004 -	002 -	00	04	22	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	CF	CFGA0042	B350	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 February 22, 2004, at 1729 hours, with Catawba Unit 1 operating in Mode 1 at 100% power, the hydraulic system for the actuator on 1B Steam Generator Main Feedwater Isolation valve (1CF-42) failed and the valve closed. As a result, the Operators manually tripped the reactor.

This event was caused by the equipment failure of the valve actuator for 1CF-42. The plant response to the reactor trip remained within the limits of the Updated Final Safety Analysis Report.

Corrective actions for this event included replacing the actuator for valve 1CF-42.

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

The purposes of the main feedwater [EIIS:SJ] isolation valves [EIIS:ISV] are to restrict reactor coolant [EIIS:AB] system thermal transients following a secondary system high energy line break inside or outside containment, isolate forward feedwater flow to the steam generators [EIIS:SG] on a high-high level to prevent overfill, and prevent flooding of the main steam/feedwater penetration room and the consequential disablement of safety related equipment due to a pipe rupture in the main steam/feedwater penetration room. The main feedwater isolation valves are arranged with 1CF-33 aligned to steam generator A, 1CF-42 aligned to steam generator B, 1CF-51 aligned to steam generator C, and 1CF-60 aligned to steam generator D.

The main feedwater isolation valves are flex wedge gate valves with pneumatic-hydraulic piston cylinder operators. Hydraulic fluid is pumped into the opening side of the piston cylinder to open the main feedwater isolation valves. At the same time, nitrogen from the closing side of the piston cylinder is compressed into a nitrogen accumulator as the piston retracts into the cylinder. Hydraulic solenoid valves control application of hydraulic pressure to the piston. The solenoids are de-energized to block fluid flow to the reservoir and allow the hydraulic pump to force the main feedwater isolation valve open. The hydraulic solenoid valves are energized to allow hydraulic fluid flow to the reservoir. Nitrogen from the accumulators works against the piston to force the hydraulic fluid to the reservoir and close the main feedwater isolation valve.

During normal operation, the hydraulic system is maintained pressurized by a pressure switch in the hydraulic pump motor circuit that turns the pump on at the low pressure setpoint and shuts the pump off at the high pressure setpoint.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	2004	- 002 -	00	3 OF 8

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

The main feedwater isolation valves are included in the Catawba preventive maintenance program. The EQ replacement requirement for these valves is 15 years (approximately every ninth refueling outage). However, the actuators are replaced every sixth refueling outage to ensure plant reliability and preclude inadvertent equipment failures.

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

EVENT DESCRIPTION
(Certain event times are approximate)

Date/Time	Event Description
02/22/04/0613	The Operator at the Controls (OATC) received an Operator-Aid-Computer (OAC) alarm, "Valve CF042 CF ISOL HYDRAULIC PUMP". Alarm response instruction on the OAC included dispatching an operator to check for oil leaks and monitoring for further motor starts. The Control Room Senior Reactor Operator (CRSRO) was notified. A Non-Licensed Operator (NLO) was dispatched to the valve. The NLO reported that there were no leaks and the pump was cycling on and off approximately every 5 seconds.
02/22/04/0638	The OATC received an OAC alarm "1CF42 ISOL HYD Pump Current Run Time," and the CRSRO was notified. The motor for 1CF-42 was determined to be running continuously. Alarm response on the OAC specified the following: 1) "Immediately perform an evaluation to determine if the nitrogen system should be isolated to avoid an inadvertent closure of the CFIV" and 2) "Issue work request to investigate and repair the valve." Work request 98305320 was written to inspect/repair the continuous operation of valve 1CF42 hydraulic pump.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	2004	- 002 -	00	4 OF 8

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

02/22/04/1045	A conference call was held with Engineering, Maintenance, Operations, and Plant Management to determine the appropriate course of action. Inspections of the actuator had determined that there was no sign of external fluid leakage and the fluid level in the hydraulic system was not decreasing. A decision was made to formulate a troubleshooting plan.
	As an interim mitigating measure, Engineering and Maintenance personnel arrived on site. Engineering recommended that nitrogen pressure be reduced to just above the operability limit to provide margin if the problem was a hydraulic leak that could reduce hydraulic pressure if the leak rate increased.
02/22/04/1244	The nitrogen pressure for 1CF-42 was decreased at the direction of Operations Shift Manager to just above the low alarm set point.
02/22/04/1312	Unit threat team assembled to develop troubleshooting plans for the problem with 1CF-42.
02/22/04/1559	The OATC received an OAC alarm,"VALVE 1CF42 CF ISOL HYDRAULIC PUMP". The CRSRO was notified and an NLO was dispatched to the valve. The NLO reported that there were no leaks and that the pump motor was again cycling on and off approximately every five seconds.
02/22/04/1729	While the troubleshooting plan was being reviewed by the Unit threat team, 1CF-42 failed closed and the OATC manually tripped the Unit 1 Reactor.
02/22/04/1759	All plant control systems functioned as designed. Plant conditions stabilized at no load conditions.

FACILITY NAME (1)	DOCKET (2) NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	2004	- 002 -	00	5 OF 8

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

02/22/04/1855	Required four hour notification to the NRC completed.
02/27/04/1130	Unit 1 placed on line and maintained at 16% power.
03/02/04/0930	The actuator for valve 1CF-42 was replaced with a spare actuator. OATC began increasing power.
03/03/04/1511	Unit 1 returned to 100% power.

CAUSAL FACTORS

The reactor was manually tripped because valve 1CF-42 closed. Investigations into the failure of valve 1CF-42 revealed that the hydraulic system pressure could not be maintained when a plug in the bottom of the hydraulic reservoir ejected. It is believed that the plug o-ring started leaking early in the morning on February 22 and caused the pump cycling and continuous operation that was seen throughout the day. Eventually, the plug threads failed and the plug ejected from the reservoir. This rendered the hydraulic system incapable of maintaining pressure in the piston cylinder and allowed nitrogen pressure to close the valve.

The valve failure is an Equipment Performance and Information Exchange (EPIX) reportable equipment failure.

CORRECTIVE ACTIONS

Immediate:

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

Subsequent:

1. The actuator for valve 1CF-42 was replaced with a rebuilt spare.

2. The remaining main feedwater isolation valves for both units were inspected and motor start data from the OAC was reviewed to ensure that no abnormal conditions were present. No abnormal conditions existed.

3. The alarm response information was changed to provide additional guidance in the event similar conditions are seen in the future.

Planned:

1. Complete the root cause evaluation of the plug o-ring leak and thread failure on valve 1CF-42.

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.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6) YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	PAGE (3)
Catawba Nuclear Station, Unit 1	05000413	2004	- 002 -	00	7 OF 8

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

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:

- Reactor trip initiating event (reactor trip with no complications),
- 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 is estimated to be 8.3E-8, which is less than the accident sequence precursor threshold of 1.0E-6.

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 dominate Large Early Release Frequency (LERF) sequences for Catawba involve 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:

LER 413/01-001 described a Unit 1 reactor trip that resulted from 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.

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.

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 valve 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 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.