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Serial No: MNS-18-024
April 17, 2018

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
ATTENTION: Document Control Desk

10 CFR 50.73

Subject: Duke Energy Carolinas, LLC
McGuire Nuclear Station, Unit 1
Docket No. 50-369
Renewed License No. NPF-9
Licensee Event Report 369/2018-01, Revision 0
Nuclear Condition Report Number 02185409

Pursuant to 10 CFR 50.73 Sections (a) (1) and (d), attached is Licensee Event Report (LER) 369/2018-01, Revision 0, regarding valid actuations of Unit 1 Reactor Protection System and Auxiliary Feedwater System.

This report is being submitted for Unit 1 in accordance with 10 CFR 50.73 (a) (2) (iv) (A), "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a) (2) (iv) (B)." The 10 CFR 50.73 (a) (2) (iv) (B) systems to which the requirements of paragraph (a) (2) (iv) (A) applied was the Reactor Protection System and the Auxiliary Feedwater System.

This LER is preliminary and will be supplemented upon completion of the cause analysis. Duke Energy will provide a supplement to this LER within 30 days.

This event is considered to have no significance with respect to the health and safety of the public. There are no regulatory commitments contained in this LER.

If questions arise regarding this LER, contact Joseph F. Hussey at 980-875-5045.

Sincerely,

Thomas D. Ray

Thomas D. Ray
Site Vice President
McGuire Nuclear Station

Attachment

IE22
NRR

cc:

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NRC FORM 366 (04-2017) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block) (See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)	Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Information Services Branch (T-2 F43), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@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 an 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.

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4. TITLE
Valid Actuation of the Unit 1 Reactor Protection System and Auxiliary Feedwater System

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
02	16	18	2018	- 01	- 00	04	17	2018	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
		☐ 50.73(a)(2)(i)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A	

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Joseph F. Hussey	(980) 875-5045

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

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☒ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☐ NO		05	17	2018

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

16. ABSTRACT

At 1014 [EST] hours on February 16, 2018, with Unit 1 in Mode 1 at approximately 100 percent power, the reactor was tripped when the 1B Reactor Auxiliary Trip Breaker was actuated during Train B Solid State Protection System (SSPS) testing. The trip was uncomplicated with all systems responding normally post-trip. Operations manually started the motor driven auxiliary feedwater pumps. The turbine driven auxiliary feedwater pump auto-started on low low steam generator level in two out of four steam generators. A Feedwater Isolation occurred as designed due to the Reactor Trip and Lo Tave condition.

The preliminary cause is: Instrument and Electrical Technician took an improper action to perform hands on verification of the position of the incorrect reactor trip breaker.

This LER will be supplemented upon finalization of the cause analysis.

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NARRATIVE

BACKGROUND

The following information is provided to assist readers in understanding the event described in this LER. Applicable Energy Industry Identification [EIIS] system and component codes are enclosed within brackets. McGuire Nuclear Station unique system and component identifiers are contained within parentheses.

Reactor Protection System [JC] (IPE):

The Reactor Protection System keeps the Reactor operating within a safe operating range by automatically shutting down the Reactor whenever the limits of the operating range are approached by monitoring process variables. Whenever a direct or calculated process variable exceeds a setpoint the Reactor is automatically tripped to protect against fuel cladding damage or loss of Reactor Coolant System (NC) integrity. Station operators may elect to manually actuate the reactor trip switchgear (manual reactor trip) using either of two control board switches.

Reactor Main Trip Breakers and Reactor Auxiliary Trip Breakers [AA] (IRE)

The reactor trip breaker system consists of four circuit breakers. The breakers are arranged in a duplicate, two train configuration with a main reactor trip (RT) and a bypass (BY) breaker associated with each train (A or B). The Reactor Main Trip Breakers, Train A (RTA) and Train B (RTB), are located in a series arrangement so that only one of these train related breakers needs to open to remove power from the Control Rod Drive Mechanisms , thus causing a reactor trip.

There is one bypass breaker in parallel with each reactor trip breaker, BYA for Train A and BYB for Train B" When the bypass breaker is closed in parallel with the reactor main trip breaker, the reactor main trip breaker can be opened without causing a loss of power to the Rod Control System. This arrangement allows testing of the Reactor Protection System and reactor trip breaker operation without having to shutdown the unit.

Engineering Safety Feature Actuation System [JE] (ISE):

The Engineering Safety Feature Actuation System is a functionally defined system that consists of all components from the field-mounted process instrumentation to the output of the device that actuates an engineered safety feature when required. The ESFAS includes portions of:

- System EIA - NSSS Process Control System; and
- System EYA - SSPS Test Cabinets
- System ISE - ESF Actuation (SSPS).

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BACKGROUND (continued)

Feedwater System [SJ] (CF):

The CF System takes treated Condensate (CM) System water, heats it further to improve the plant's thermal cycle efficiency, and delivers it at the required flow rate, pressure and temperature to the steam generators. The CF System is designed to maintain proper vessel water levels with respect to reactor power output and turbine steam requirements.

Auxiliary Feedwater System [BA] (CA):

The CA System automatically supplies feedwater to the steam generators (S/Gs) to remove decay heat from the NC System upon the loss of normal feedwater supply. The CA System mitigates the consequences of any event with loss of normal feedwater. The design basis of the CA System is to supply water to the S/Gs to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators.

The CA system is designed to start automatically for any event requiring emergency feedwater.

The CA System Motor Driven Pumps will automatically provide feedwater when initiated on any of the following conditions:

1. Trip of both main feedwater pumps
2. AMSAC Actuation (AMSAC - Anticipated Transient Without Scram (ATWS) Mitigation System Activation Circuitry)
3. Two out of four (2/4) low-low level alarms in any one steam generator
4. Initiation of a safety injection signal
5. Loss of power to the 4160V essential bus (Blackout)

The CA System Turbine Driven Pump (TDCA) will automatically provide feedwater when initiated on any of the following conditions:

1. Two out of four (2/4) low-low level alarms in any two steam generator
2. Loss of power to the 4160V essential bus (Blackout)
3. 1/1 detector from SSF SG Wide Range Low-Low Level on 2/4 SGs (72%)

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EVENT DESCRIPTION

On February 16, 2018, while in Mode 1, an Instrument and Electrical Technician performing portions of PT/0/A/4601/008 B, "SSPS Train B Periodic Test NC Pressure Greater Than 1955 psig", actuated 1B Reactor Auxiliary Trip Breaker (BYB) tripping Unit 1.

The purpose of SSPS testing is as follows:

- Verify Reactor Trip System Instrumentation surveillance requirements for automatic trip and interlock logic.
- Verify Engineered Safety Features Actuation System Instrumentation surveillance requirements for automatic trip and interlock logic.
- Provide Reactor Trip breaker monthly time response surveillance. This provides trendable information necessary to predict possible breaker degradation.

At 0946 Operations logged SSPS Train B inoperable to allow performance of PT/0/A/4601/008 B to begin. Instrument and Electrical Technician 1 (Technician 1) and Instrument and Electrical Technician 2 (Technician 2) began executing procedure steps and encountered no issues until the procedure step following actuation of the Reactor Trip Breaker under test (1RTB). When Technician 2 approached the back of breaker CABINET-1 (RTB/BYB), he failed to perform correct component verification (CCV) and opened bypass breaker (1BYB) compartment instead of the reactor trip (1RTB) breaker compartment. Upon opening the 1BYB compartment, Technician 2 saw the 1BYB breaker position to be closed which was not what the Technician expected. At this point Technician 2 depressed the 1BYB TRIP button to lower the shutter to look inside. This intrusive hands on action was not directed by the procedure. Since 1BYB was closed and the circuit energized, a reactor trip signal was generated, and Unit 1 tripped. At this point, work was suspended and management notified.

The relevant sequence of events was taken from the Control Room Logs (eSOMS) and Operator Aid Computer (OAC) alarms and is as follows (all times approximate, where time is the same the event or action is occurring within fractions of seconds):

09:46:10 SSPS TRAIN B, SSPS REACTOR TRIP BREAKER SURVEILLANCE TESTING (eSOMS)
09:46:35 1B REACTOR AUXILIARY TRIP BREAKER (Closed)
10:14:08 U1 PZR HI PRESS RX TRIP (SSPS Test Signal)
10:14:08 U1 RX TRIP- SWGR B TRIPPED (1RTB open)
10:14:08 1B REACTOR MAIN TRIP BREAKER (1RTB open)
10:14:12 U1 PZR HI PRESS RX TRIP (SSPS Test Signal Reset)
10:14:23 1B REACTOR AUXILIARY TRIP BREAKER (1BYB Manually Opened)
10:14:23 U1 TURBINE TRIP (Reactor Trip causes Turbine Trip)
10:17:53 1A CA PUMP, (Manual Start logged)
10:17:57 1B CA PUMP (Manual Start logged)
10:19:43 1B S/G LO-LO LEVEL RX TRIP, (OAC alarm)
10:19:45 1D S/G LO-LO LEVEL RX TRIP, (OAC alarm)
10:20:07 TDCA PUMP, (Auto Start)

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REPORTABILITY DETERMINATION

The Unit 1 Reactor Protection System actuation while critical was a valid actuation and initially reported, as required, under 10 CFR 50.72 (b)(2)(iv)(B), "Any event or condition that results in actuation of the Reactor Protection System (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation." The event also resulted in the valid actuation of the TDCA pump and MDCA pumps and was initially reported, as required, under 10 CFR 50.72(b)(3)(iv)(A), and this LER will satisfy the corresponding reporting criteria 10 CFR 50.73 (a)(2)(iv)(A), "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)." The applicable 10 CFR 50.73(a)(2)(iv)(B) systems include the Reactor Protection System and the Auxiliary Feedwater System.

CAUSAL FACTORS

A cause evaluation is currently in progress. This LER will be supplemented upon finalization of the cause analysis. The preliminary cause is: The technician took an improper action to perform hands on verification of the position of the incorrect reactor trip breaker.

CORRECTIVE ACTIONS

Immediate Actions:

1. Technicians 1 and 2 removed from duty.
2. A stand down was held for supervisors to cover this event with their teams at the beginning of the next shift.
3. Additional qualified Maintenance technicians were called in to support recovery from 1B Train SSPS testing.
4. A Prompt Investigation Response Team (PIRT) was formed and investigation performed.

Subsequent Actions:

1. Revise PT/0/A/4601/008 A and PT/0/A/4601/008 B "SSPS Train A Periodic Test NC Pressure Greater Than 1955 psig and SSPS Train B Periodic Test NC Pressure Greater Than 1955 psig" to include the following guidance:
   - Place Unit Trip Potential Signage on Reactor Auxiliary Trip Breaker compartment door while breaker in service during the performance of the surveillance test. (Complete)
   - Establish a Critical Step per AD-HU-ALL-0004, " Procedure and Work Instruction Use and Adherence section 5.4" in both train SSPS test procedures in the step or series of steps where the Reactor Main Trip Breaker trips open. (Complete)
2. Coach and counsel individuals involved in the event. (Complete)

NARRATIVE

Planned Actions:

1. The LER supplement will address any necessary planned actions.

SAFETY ANALYSIS

At 1014 [EST] hours on February 16, 2018, with Unit 1 in Mode 1 at approximately 100 percent power, the reactor tripped when the 1B Reactor Auxiliary Trip Breaker was manually opened during Train B Solid State Protection System (SSPS) testing. The trip was uncomplicated with all systems responding normally post-trip. Operations manually started the motor driven auxiliary feedwater pumps. The turbine driven auxiliary feedwater pump (TDCAP) auto-started on low steam generator level. A Feedwater Isolation occurred as designed due to the Reactor Trip and Lo Tave condition. Operations stabilized the plant. Unit 2 was not affected.

The reactor tripped due to the opening of a 1B Reactor Auxiliary Trip Breaker and the plant safety systems responded as designed. There is no safety consequence to this aspect of the event and resulted in no significant impact to the health and safety of the public.

ADDITIONAL INFORMATION

A search of the Corrective Action Program (NCR) database was conducted, based on the preliminary root cause, to determine if this event was recurring at McGuire, i.e., similar significant event with the same cause code. No NCR of similar significance with the same cause code was identified. Currently this is not considered a recurring event.