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February 13, 2006

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

Subject: McGuire Nuclear Station, Unit 1
Docket No. 50-369
Licensee Event Report 369/2005-06, Revision 0
Problem Investigation Process (PIP) M-05-05989

Pursuant to 10 CFR 50.73, Sections (a)(1) and (d), attached is Licensee Event Report (LER) 369/05-06, Revision 0, concerning an automatic trip of the McGuire Nuclear Station Unit 1 reactor and automatic actuation of the Unit 1 Auxiliary Feedwater (CA) System.

This LER is being submitted as per the requirements of 10 CFR 50.73 (a)(2)(iv)(A). This event is considered to be of no significance to the health and safety of the public. There are no regulatory commitments contained in this LER.

Very truly yours,

JPHarrall / for

Gary R. Peterson

Attachment

IE22

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NRC FORM 366 (6-2004)

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 06-30-2007**

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 and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects@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
Automatic Reactor Trip and Auxiliary Feedwater System Actuation Due To Steam Generator Hi-Hi Water Level

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
12	17	2005	2005	- 006 -	00	02	13	2006	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)
Reza Djali-Regulatory Compliance	704-875-4228

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).	X	NO				

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

Unit Status: At the time of the event, both Unit 1 and Unit 2 were in Mode 1 (Power Operation) at 100 percent power.

Event Description: On December 17, 2005, the 1A Steam Generator experienced High-High water level when its controlling channel for main feedwater flow failed low. A main feedwater isolation occurred along with a turbine trip, trip of both CF pumps, automatic start of the motor driven CA pumps, and reactor trip. All safety systems needed to respond to this event operated as designed. This event is considered to be of no significance to the health and safety of the public.

Event Cause: The cause of this event was intermittent degraded voltage to transmitter 1A SG Channel 1 flow loop (1CFFT5000).

Corrective Action: Selected components were replaced. Lessons learned from this event will be incorporated in appropriate training to further strengthen operator response time to steam generator level deviations.

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

BACKGROUND

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

Steam Generator [SG](SG) Water Level Hi-Hi Signal (P-14):

This signal prevents damage to the turbine [TG] due to water in the steam lines and provides protection against excessive feedwater [SJ](CF) flow. The nominal setpoint for this signal is 83.9 percent SG water level, and the allowable value is less than or equal to 85.6 percent SG water level. Three channels are required by technical specifications and P-14 signal is initiated by two-out-of-three logic on any SG. This signal trips the main turbine, trips the CF pumps, and initiates CF isolation.

SG Level Control:

During high CF flow conditions when reactor power is greater than or equal to 15 percent, CF flow control valves [FCV](FCV) 1CF17, 1CF20, 1CF23 and 1CF32 control CF flow to maintain acceptable water level in SGs 1D, 1C, 1B, and 1A, respectively. These valves are capable of both automatic and manual operation. The mode of operation is selected at the "Auto/Manual" control station located on control board MC2 in the control room. There are two CF flow channels for each SG. CF flow is indicated on the control board for both channels of each SG. The channel to be used for level control is selected using a select switch. There is one switch per SG, with Channel 1 normally selected for control. CF flow for the channel selected is also recorded on the main control board. The 7300 Process Control System [JF](PCS) provides two separate control circuits for each FCV (Normal and Alternate). Only one of the two control circuits will be controlling a FCV at any time. The desired control circuit is selected via the "Norm/Alt" select switch located on Main Control Board MC2 in the Control Room.

Auxiliary Feedwater [BA](CA) System:

The CA System is designed for operation during plant startup, plant shutdown, and emergency conditions where the CF System is not available. The Unit 1 CA System contains one turbine driven pump and two motor driven pumps. The 1A and 1B Motor Driven CA (MDCA) Pumps [P] start automatically and provide flow to the Unit 1 SG's upon tripping of both CF pumps to act as the heat sink for the reactor.

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Reactor Protection [JC](IPE) System:

The function of the RPS System circuits associated with turbine trip is to minimize the pressure/temperature transient on the reactor. A turbine trip from a power level below the P-8 setpoint, approximately 48% power, will not actuate a reactor trip. Above the P-8 setpoint, a turbine trip will cause a reactor trip to minimize the transient on the reactor.

EVENT DESCRIPTION

On December 17, 2005, with Unit 1 at 100% power, the 1A SG experienced Hi-Hi water level when its controlling channel (Channel 1) for CF flow failed low. This caused the associated CF control valve 1CF-32 to open and SG level to increase. In response, 1CF-32 was placed in manual control per Abnormal Procedure AP-06, SG Feedwater Malfunction, to reduce 1A SG water level. Notwithstanding, 1A SG water level reached the Hi-Hi level and a P-14 signal was actuated. A main feedwater isolation occurred along with a turbine trip, trip of both CF pumps, automatic start of the motor driven CA pumps, and reactor trip.

The relevant sequence of events is as follows (all times approximate):

03:10:28 1A SG Channel 1 CF flow started trending down.

03:10:34 1A SG Channel 1 CF flow failed low. 1A SG Channel 2 CF flow was spiking up.

03:10:37 1CF-32 position indicated full open.

03:10:40 CF pump speed and flow were increasing. CF pump flow to B, C And D SG's were increasing.

03:10:58 1CF-32 position indicated intermediate (not full open) due to manual operator actions.

03:11:26 1A SG Narrow Range Level II indicated Hi-Hi.

03:11:29 1A SG Narrow Range Level II and Level IV indicated Hi-Hi.

03:11:29 1A SG Hi-Hi Level Turbine Trip and Reactor Trip occurred.

03:11:30 Both CF pumps tripped and both motor driven CA pumps automatically started as designed.

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The operators responded to the reactor trip properly using the appropriate plant procedures, and all safety systems needed to respond to the event operated as designed.

This event was initially reported on December 17, 2005 as an actuation of the Unit 1 Reactor Protection and Auxiliary Feedwater Systems in accordance with the requirement of 10 CFR 50.72 (b)(2)(iv)(B). This LER is being submitted as per the requirements of 10 CFR 50.73 (a)(2)(iv)(A).

CAUSAL FACTORS

The event started when the flow signal of 1A SG Channel 1 flow control loop (1CFLP5000) dropped abruptly. In addition to the sudden signal drop, operator aid computer (OAC) data showed small cyclic signal activity in the loop from the time the signal dropped until (and coincident with) CF isolation. After that, the loop signal returned to normal output corresponding to the redundant flow Channel 2.

The transmitter, manifold valves and power supply (NLP) card were removed from the plant and taken to a testing facility at McGuire. Testing was unable to replicate the failure, however, it identified that the transmitter output exhibited cyclic activity similar to the one observed on the OAC when it was subjected to a starved or degraded voltage condition. The tested starved or degraded voltage condition was well below the stated equipment voltage specification for Rosemount 3051C transmitters. Normal operating voltages are above tested degraded voltage, and transmitters operating at normal voltage levels do not exhibit this cyclic activity.

The cause of this event was intermittent degraded voltage to 1A SG Channel 1 flow transmitter (1CFFT5000). Momentary failure of either the 7300 PCS system NLP loop power supply card or internal electronics of transmitter 1CFFT5000, along with cable termination issues could have caused this condition. As found calibration checks, along with in-depth diagnostic testing of the actual components in a simulation mockup loop, have been unable to narrow down a specific failure mechanism that would have caused this degraded voltage condition.

The operators responded to the reactor trip properly using the appropriate plant procedures. A post trip review identified future enhancements to operators' response time to steam generator level deviations. The lessons learned from this review will be incorporated in training to further strengthen operators' response which should reduce challenging protection system actuations.

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CORRECTIVE ACTIONS

Immediate:

- Performed troubleshooting of the 1A SG Channel 1 flow control loop (1CFLP5000).
- Replaced 1A SG Channel 1 flow transmitter (1CFFT5000) and its associated manifold valves and 7300 PCS loop power supply card.
- Inspected 1CFFT 5000 transmitter impulse lines for clogging/debris. No items were found.
- Satisfactorily Performed Channel 1 calibration with new components.

Planned:

- Lessons learned from this event will be incorporated in training to further strengthen operator response to steam generator level deviations.

SAFETY ANALYSIS

At no time was the safety or health of the public or plant personnel affected as a result of the event.

Reactor trips and turbine trips are analyzed in Chapter 15 of the McGuire Nuclear Station Final Safety Analysis Report. Those analyses demonstrate that, given the plant conditions and sequence of events associated with the December 17, 2005 event, the plant design and response was adequate. Therefore, this event presented no hazard to the integrity of the Reactor Coolant System or the reactor fuel/cladding.

The Conditional Core Damage Probability (CCDP) associated with this event is evaluated to be < 1E-6, and the Conditional Large Early Release Probability (CLERP) associated with this event is evaluated to be < 1E-7.

Given the above, this event is considered to be of no significance with respect to the health and safety of the public.

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ADDITIONAL INFORMATION

A review of previous events at McGuire for the past three years did not identify any previous events with the same underlying concern or reason for reporting. However, one event was identified that involved failure of CF FCV and increasing SG level on unit 2 which did not result in a turbine or reactor trip.

On May 13, 2005, with unit 2 at 100% power, SG 2A Channel 1 feed flow failed low causing SG level to increase. The operator's manual control of 2CF-32 restored normal flow and level conditions just below turbine trip setpoint. The cause of this event was failure in the associated 7300 PCS isolator card. The failed component was replaced.