Entergy

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Fred Dacimo
Site Vice President
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January 24, 2005
Indian Point Unit No. 2
Docket No. 50-247
NL-05-009

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: Licensee Event Report # 2004-005-00, "Automatic Reactor Trip Due to Turbine Generator Trip Caused by Low Stator Cooling Water Pressure."

Dear Sir:

The attached Licensee Event Report (LER) 2004-005-00 is the follow-up written report submitted in accordance with 10 CFR 50.73. This event is of the type defined in 10 CFR 50.73(a)(2)(iv)(A) for an event recorded in the Entergy corrective action process as Condition Report CR-IP2-2004-06467.

There are no commitments made by the Licensee in the attached LER. Should you or your staff have any questions regarding this matter, please contact Mr. Patric W. Conroy, Manager, Licensing, Indian Point Energy Center at (914) 734-6668.

Sincerely,

Fred R. Dacimo
Site Vice President
Indian Point Energy Center

IE22

Attachment: LER-2004-005-00

cc:

Mr. Samuel J. Collins
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U.S. Nuclear Regulatory Commission
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Resident Inspector's Office
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NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE Automatic Reactor Trip Due to Turbine Generator Trip Caused by Low Stator Cooling Water Pressure

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
11	26	2004	2004 -	05	- 00	01	24	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 092

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(A)(2)(I)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Brian Meek, System Engineer	(914) 734-5814

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
A	TJ	63	M235	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
No				

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

An automatic Reactor Trip occurred on November 26, 2004, at 1322 hours, due to a Main Generator/Turbine trip as a result of low stator cooling water pressure from an erroneous low inlet pressure trip on the Main Generator Stator Cooling Water System (SCWS). All control rods fully inserted and all primary systems functioned properly. The Auxiliary Feed Water (AFW) system automatically started as a result of a Steam Generator low level due to shrink effect. The plant was stabilized in hot standby with decay heat being removed by the main condenser. There was no radiation release. Offsite power remained available therefore the emergency diesel generators did not start. The cause of the event was an incorrect pressure switch setpoint for the SCWS due to inadequate integrated system and post-work testing and setup of the system, inadequate operational procedure guidance for start-up/fill and venting of SCWS and inappropriately attempting to adjust SCWS control valve Y-63. Significant corrective actions included performance of a setup process of the SCWS in accordance with engineering's evaluation and recommendations including setting of the SCWS flow control valve Y-63 hot at 465 gpm (range 454-477) before plant re-start. An integrated SCWS post work test procedure will be prepared including complete development of I&C PM's for the post work test procedure, and improvements in operational procedure guidance. The event had no effect on public health and safety.

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

Note: The Energy Industry Identification System Codes are identified within brackets { }

DESCRIPTION OF EVENT

On November 26, 2004, at approximately 13:22 hours, while holding at 92% reactor power, an automatic Reactor Trip (RT) occurred due to a Main Generator/Turbine trip as a result of an erroneous low inlet pressure trip on the Main Generator {EL} Stator Cooling Water System (SCWS) {TJ}. Prior to the event, at 13:21 hours, the Nuclear Plant Operator (NPO) and the Shift Manager (SM) were investigating a Main Generator Stator Water Cooling high flow condition of 488 gpm as recorded in the conventional logs. This was an increase from 470 gpm as recorded on the previous day's log. While investigating the high flow abnormality, the NPO opened the cover for the Stator Water Cooling Flow Control Valve Y-63 {FCV} Controller YPC-63 and placed his hand on the adjustment knob. The NPO intended to slightly lower the adjustment knob but because it was very stiff, he was unable to move it. An investigation concluded the NPO most likely bumped the bourdon tube in the controller causing a slight system perturbation. The slight system perturbation caused Pressure Switch 63-P79 {63} to toggle in the closed position. Pressure Switch 63-P79 could not reset due to the reset being above SCWS normal operating pressure. The C-1 relay (Generation Protection Circuit Energized) resulted in activation of the Generator Protection Alarm which timed out at 40 seconds and initiated a Main Turbine trip followed by a Reactor trip {JC}. All control rods fully inserted and all primary systems functioned properly. The Auxiliary Feed Water System {BA} automatically started as a result of a Steam Generator low level due to shrink effect. The plant was stabilized in hot standby with decay heat being removed by the main condenser. There was no radiation release. Offsite power remained available therefore the emergency diesel generators {ED} did not start. SCWS flow control valve Y-63 is an air operated Fisher Controls butterfly valve Model 7610. The controller for Y-63 (Controller YPC-63) is a model 4160K Wizard II manufactured by Fisher Controls {F130}. The controller Pressure Switch 63-P79 is a Mercoid {M235} Model DA-33-2 R6.

Control Room (CR) operators observed Alarm 2-5 on Panel FB Stator System Cooling System Generator Protection Circuit Energize at 1321 hours. CR operators observed the rod bottom lights, Reactor Trip (RT) on Turbine Trip First Out Annunciator FAF 1-4 Generator Loss of Coolant Trip at 1322 hours and entered procedure E-0 at 1324 hours. CR Operators entered procedure ES-0.1 at 1327 hours, secured the Main Boiler Feedwater Pumps (MBFP) at 1351 hours and transitioned to procedure POP-3.2. The plant was stabilized in hot standby with decay heat being released to the main condenser via the steam dump valves {V}. At 1429 hours, a 4-hour non-emergency notification was made to the NRC for a reactor trip while critical under 10CFR50.72(b)(2)(iv)(B) and an AFW actuation under 10CFR50.72(b)(3)(iv)(A) (8-hour) (Incident Log No. 41227). Operations recorded the RT event in the corrective action program (CAP) as Condition Report CR-IP2-2004-06467. A post transient evaluation was performed on November 26, 2004.

An extent of condition was performed for similar Mercoid switch issues related to the circumstances behind the inadvertent SCWS trip event. Indian Point Unit 2 uses a SCWS for its main generator. Indian Point 3 has the original

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

Westinghouse main generator that does not use water to cool its windings. There are numerous systems in the plant that use Mercoid type switches, but none where a one-out-of-one automatic unit trip logic is utilized. Several other system configurations such as Main Turbine, MBFPs, and Seal Oil were reviewed. The MBFPs at both units are an example of a system that utilize Mercoid switches for a trip function with a one-out-of-one trip logic but do not directly cause an automatic unit trip. The SCWS was also reviewed for further one-out-of-one trip hazards. Four have been identified. These include Low Inlet Pressure (cause of the trip), Low Stator Cooling Flow, High Stator Water Temperature, and Low Cooling Rectifier Flow. It was determined there are no similar Mercoid switch issues related to the circumstances behind the inadvertent SCWS trip event.

Single Point Failure analyses are planned to be performed on plant systems. These analyses will highlight plant vulnerabilities to one-out-of-one failures. These analyses will produce recommendations to ensure that single point failure risks are reduced to a manageable level.

The Cause of Event

The direct cause of the trip was inadvertent actuation of the SCWS Low Inlet Pressure Trip by Pressure Switch 63-P79. The actuation of switch 63-P79 to toggle to the closed position was due to an unexpected system perturbation during attempted controller adjustment and as a result of a tight operating band around the Normal Operating Pressure (NOP) of the SCWS. Pressure switch 63-P79 reset value was above the NOP of the system and failed to reset resulting in the activation of the Generator Protection Alarm, which timed out (40 seconds) and tripped the turbine. Contributing significantly to this event was over pressurization of the SCWS during system startup one week prior to this event and lack of proper evaluation of the SCWS over pressurization. The system was partially drained as opposed to fully drained as had been in past outages resulting in a water/air pressurization transient during the SCWS startup. The reset point of the switch appears to have been altered by the over pressurization of the SCWS upon start up during the prior week. The minimum flow stops on Y-63 were improperly set which created a condition for the valve to be able to move in the closed direction decreasing flow and pressure and cause a perturbation of the system. Additional maintenance performed on the SCWS system during cycle 16 refueling outage included overhauling Y-63 and other system control valves, installing a new controller and positioner on the system temperature control valve, and changing the system filters and resin beds.

There were two root causes (RC): RC-1 was Work Organization and Planning; Insufficient integrated system and post-work testing after extensive work was performed on the SCWS (i.e., inappropriate use of a Post Work Test used to setup the system in 2002 following the Main Generator Rewind Modification). RC-2 was Written Communication, Omission of Relevant Information; Inadequate operational procedural guidance allowed undesirable system operation, manipulation, and control. Operating procedures did not contain explicit guidance for starting up/fill and venting of the system or adjusting of SCWS control valve Y-63.

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Corrective Actions

The following corrective actions have been or will be performed under the Corrective Action Program (CAP) to address the causes of this event.

- Communicated to all site personnel the event and lessons learned and to reinforce management's expectations on the importance of a robust questioning attitude to address potential concerns and questions that arise. Additionally, the need for improved communications vertically and horizontally when abnormalities are detected with trip risk sensitive equipment. Completed via a Red Memo and reset the Station Event Free Clock on November 30, 2004.
- Instrument and Control checked the calibrations of the Generator Stator Water Trip and Alarm Switches and performed the setup process of the SCWS as per Engineering's direction with Y-63 stops set hot at 465 gpm (range 454-477). Completed on November 27, 2004.
- A plan will be prepared to develop necessary system integrated post work tests in support of start up of the SCWS following refueling outages. Scheduled completion is March 1, 2005.
- An integrated SCWS preventive maintenance and post work test procedure(s) will be prepared to support system start up after each refueling outage as necessary. Scheduled completion is May 20, 2005.
- Operating procedure guidance will be prepared to include information on draining and filling the Generator. Revisions to the system start up procedures are to include opening the knife switch on the pump not being used as the primary start up pump. The procedure revision will also include closing both pump discharge valves fully before starting the system from cold. The Y-63 valve will also be adjusted so it is in the full closed position upon start up. Scheduled completion is May 20, 2005.

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Event Analysis

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply include the reactor protection system (RPS) including reactor scram or reactor trip, and AFWS.

This event meets the reporting criteria because the RPS was actuated by automatic trip of the main turbine and the AFWS actuated on low level due to steam generator level changes in response to the automatic RT, which occurs after a RT from full power as a result of SG shrink.

Past Similar Events

A review of the past two years of Licensee Event Reports (LERs) for events that involved a RT caused by Turbine Generator trip as a result of SCWS malfunctions identified no events.

Safety Significance

This event had no effect on the health and safety of the public. There were no actual safety consequences for the event because the event was an uncomplicated RT with no other transients or accidents. Required safety systems performed as designed when the RT occurred. The AFWS actuation was an expected reaction as a result of decreasing SG water level due to the reduction of SG void fraction (shrink), which occurs after automatic RT/TT from essentially full load. A core damage probability (CDP) for this event was assessed and a CDP of 6.9 E-7 was associated with the turbine trip. This CDP value by itself implies the event has low safety significance, however it should be noted that since all safety equipment operated as designed following the turbine trip, the increase in CDP is actually zero for this event.

For this event rod control was in manual and the reactor scrammed immediately upon a main turbine trip. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.