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

Subject: Oconee Nuclear Station
Docket Nos. 50-269,-270, -287
Licensee Event Report 287/2002-01, Revision 0
Problem Investigation Process No.: O-02-06562

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report 287/2002-01, Revision 0, concerning the Unit 3 Reactor trip due to Moisture Separator Reheater level.

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

R. A. Jones

Attachment

IE22

cc: Mr. Luis A. Reyes
Administrator, Region II
U.S. Nuclear Regulatory Commission
61 Forsyth Street, S. W., Suite 23T85
Atlanta, GA 30303

Mr. L. N. Olshan
Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Mr. M. C. Shannon
NRC Senior Resident Inspector
Oconee Nuclear Station

INPO (via E-mail)

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
Oconee Nuclear Station, Unit Three	050- 0287	1 OF 7

4. TITLE

Moisture Separator Reheater Level Results in Reactor Trip

5. EVENT DATE MO	DAY	YEAR	6. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO	7. REPORT DATE MO	DAY	YEAR	8. OTHER FACILITIES INVOLVED FACILITY NAME	DOCKET NUMBER
11	14	02	2002	- 01	- 00	1	13	2003	None	
									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)
L.E. Nicholson, Regulatory Compliance Manager	(864) 885-3292

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
B	SM	valve	F130	Y	B	SM	valve	M040	Y

14. SUPPLEMENTAL REPORT EXPECTED			15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	x	NO		01	13	2003

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

On November 14, 2002, Oconee Unit 3 was operating in Mode 1 at 100% power. At approximately 0419 hours, Unit 3 experienced a high level in a Moisture Separator Reheater (MSRH). The turbine tripped as a protective feature to prevent water from backing up into the turbine. This caused an anticipatory reactor trip. The high level was caused by three separate equipment failures which resulted in discharge flow paths from the 3A Moisture Separator Drain Tank (MSDT) becoming isolated and water backing up into the MSDT and MSRH.

In this event, the plant systems and operators responded as expected. Automatic system responses and operator actions in accordance with procedures were adequate to safely control the reactor following this trip.

The corrective action was to repair the three Unit 3 A MSDT level control equipment problems.

The health and safety of the public was not compromised by this event.

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

EVALUATION:

BACKGROUND

This event is reportable per 10CFR50.73(a)(2)(iv) as "any event or condition that resulted in automatic actuation of the Reactor Protection System."

Each Oconee unit contains four Moisture Separator/Reheater (MSR) shell and tube heat exchangers [EIIS:HX] which reduce the water content and raise the temperature of the steam from the high pressure turbine main steam discharge before it is passed to the low pressure turbines [EIIS:TRB]. Removing the water and reheating the steam reduces erosion and improves the thermal efficiency of the low pressure turbines.

The water removed from the steam is passed into an MSR drain tank [EIIS:SN]. In the Dump to Condenser mode, drain tank level sensors [EIIS:XT] open valves to route the contents of this drain tank to the condenser. In Feed Forward (FF) mode, flow is routed to the feedwater [EIIS:SJ] system to improve unit efficiency.

However, even in FF mode, a high level in the drain tank will result in an alarm and the dump valves will open. If the drain tank controls were to malfunction such that the water level was allowed to rise into the MSR, the water could be entrained by the steam flow and could be passed on to the low pressure turbine. In severe cases, the low pressure turbines could be significantly damaged. Therefore, the control system will trip the turbine if an emergency high water level is sensed in any of the four moisture separators.

Prior to this event Unit 3 was operating in Mode 1 at 100% power with no safety systems or components out of service that would have contributed to this event. Unit 2 was in a refueling outage (2EOC19) and Unit 1 was operating at 100% power.

EVENT DESCRIPTION

Unit 3 tripped on November 14, 2002, at 0419 due to high level in 3A2 Moisture Separator Reheater (MSRH). This trip was caused by three separate equipment failures which resulted in both the Feed

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

Forward and the Dump to Condenser discharge flow paths from the 3A Moisture Separator Drain Tank (MSDT) becoming isolated and water backing up into the MSDT and the MSRH.

The sequence of events is as follows: with Unit 3 at 100% power, water was being removed from 3A MSDT to the 3C1 Flash Tank in the Feed Forward (FF) mode. In the FF mode of operation, valve 3HD-37 will normally regulate flow from the MSDT based on tank level. If the FF flow path becomes isolated or water enters the tank at a rate that exceeds the discharge capability of the FF flow path AND tank level begins to rise, a separate level control circuit will begin to regulate valve 3HD-27 to allow excess flow to dump to the condenser. If the normal level control circuit for valve 3HD-27 fails, a separate 3A MSDT emergency high level switch will energize a solenoid to dump air off the 3HD-27 valve actuator, fail the valve full open, and open the Dump to Condenser flow path.

The root cause investigation found that 3HD-37 failed closed due to a failure of the Digital Valve Controller (DVC) of valve 3HD-37. Also, it was found that an electrical component failure in the DVC of valve 3HD-27 resulted in the valve becoming unable to respond to the increasing 3A MSDT high level signal and the valve remaining closed isolating the Dump to Condenser flow path. The third equipment failure was the 3A MSDT emergency high level switch became stuck and would not actuate the 3HD-27 solenoid valve to dump air off the actuator to open the valve. The Dump to Condenser flow path therefore remained isolated.

With the FF flow path isolated due to valve 3HD-37 closure and the Dump to Condenser flow path isolated due to valve 3HD-27 remaining closed, water backed up in the tanks until the Unit trip setpoint was reached.

Following the reactor trip, the Reactor Coolant System (RCS) [EIIS:AB] parameters remained within established norms. Operators opened 3HP-26 (HPI TO LOOP A REACTOR INLET VALVE) and started a second High Pressure Injection (HPI) [EIIS:CB] pump for approximately ten minutes to increase make-up due to normal inventory cooldown/shrinkage. This manual start of a second HPI pump for normal RCS make-up is typical after a trip and is not associated with the Engineered Safeguards [EIIS:JE] role of HPI. The 3B HPI pump, which was on prior to the trip, was stopped at 04:30.

The High Pressure Injection (HPI) [EIIS:CB] system performs both normal makeup and Emergency Core Cooling functions. By design, the RCS temperature cools down immediately following a trip, which results in expected inventory shrinkage and affects the pressurizer level. Therefore, post-trip response includes actions to increase make-up for several minutes to compensate for this normal shrinkage.

In this event, initial Steam Generator level on Unit 3 was slightly higher than in prior events, because of some known fouling and because of heat transfer characteristics of the 3A Steam Generator. Therefore, the RCS shrinkage was slightly greater than typical, such that the Pressurizer level dropped to 35 inches rather than a more typical value of 40 inches.

Operations personnel notified the NRC of the trip at 0524 hours using the Emergency Notification System. The NRC assigned event number 39369.

CAUSAL FACTORS

The cause of the Reactor Trip was three independent equipment failures. The initiating transient was caused by the following Equipment Failures:

1. **3HD-37 (3A MSDT Level Control) failed closed** due to failure of its digital valve controller (DVC). Failure of the DVC occurred due to wear and overuse because of excessive operation of 3HD-37 to adjust to system parameter changes. The potentiometer inside the DVC exceeded its design number of cycles. A procedure error led to excessive operation of 3HD-37. When Nuclear Station Modification ON-32941 was implemented in 1997, calibration procedure IP/0/B/0275/011B was changed to incorporate the new equipment. However, the procedure change was technically incorrect, in that the DVC for 3HD-37 was set up to a 7.2-15.2 mA signal, instead of 4-20 mA as specified on design drawings. The DVC is replaced on a 3R (every 3rd refueling outage) frequency, but the PM frequency was not adequate to prevent failure, in light of the excessive number of cycles of operation.
2. **3HD-27 (3A MSDT Dump to Condenser) failed to open fully** due in part to failure of its digital valve controller (DVC). A wear spot of 1/64" was discovered inside the potentiometer that caused an

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

open circuit when the potentiometer moved to the wear spot. This resulted in the valve becoming unable to respond to the increasing 3A MSDT high level signal and the valve remaining closed isolating the dump-to-condenser flow path. The cause of the wear spot is unknown.

3. **3LS-38 (MSDT 3A Level High) was not energizing** 3SV-192 (MSDT 'A' Dump Solenoid Valve) because the spring was weak. This prevented 3HD-27 from opening and dumping water to the Condenser. The cause for the weak spring is unknown.

System water hammer concerns led to a decision not to physically inspect the DVC associated with 3B MSDT Dump to Condenser valve 3HD-28. System and equipment operating data indicated that the valve is currently operating as expected.

Because Unit 2 was in refueling outage 2EOC19, the DVCs associated with the 2A and 2B MSDT level control valves were inspected. No problems were found with the DVCs associated with valves 2HD-27, -28, and -52. The feedback linkage in the DVC of valve 2HD-37 had signs of wear and did not move smoothly. The DVC for valve 2HD-37 was replaced. A review of work performed during 2EOC19 indicated that the 2A and 2B MSDT emergency high level switches had been checked for proper operation.

CORRECTIVE ACTIONS

- **Immediate:**
  - Replaced DVC associated with 3HD-37 (Root cause #1).
  - Replaced DVC associated with 3HD-27 (Root cause #2).
  - Replaced 3A MSDT emergency high level switch (Root cause #3).
- **Subsequent:**
  - Replaced DVC associated with 2HD-37.
  - Replaced DVC associated with 3HD-52.
- **Planned:**
  - Change IP/0/B/0275/011B to calibrate 1,2,3HD-37 and 1,2,3HD-52 to 4-20 mA instead of 7.2-15.2 mA.
  - Recalibrate 1,2,3HD-37 and 1,2,3HD-52 per the revised procedure.
  - Inspect the 1A and 1B MSDT level control equipment during 1EOC21.
  - Initiate long term corrective actions as appropriate based on final Root Cause Evaluation recommendations.

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

- Replace DVC for 1,2,3HD-27 and 1,2,3HD-28 at the next refueling outage.
- Evaluate adequacy of 3R PM frequency for 1,2,3HD-27 and 1,2,3HD-28. Assign to MCE.

There are no NRC Commitment items contained in this LER.

SAFETY ANALYSIS

This event did not include a Safety System Functional Failure.

The MSRH emergency high level trip is a turbine protective trip for economic reasons: high MSRH level does not affect nuclear safety.

Post-reactor trip response, as discussed in the Event Description section of this report was within acceptable limits as defined by the Babcock and Wilcox Owners Group Transient Assessment Program.

There were no Emergency Feedwater actuations required as a result of this event.

ADDITIONAL INFORMATION

A review of reportable events indicated that no reactor trip events have occurred within the past two years due to the root cause identified in this event.

The trip of the Reactor constitutes a Maintenance Rule functional failure and is considered reportable under the Equipment Performance and Information Exchange (EPIX) program. The NRC cause code is B.

There were no releases of radioactive materials, radiation exposures or personnel injuries associated with this event. Therefore, there was no actual impact on the health and safety of the public due to this event.

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FAILURE CONTINUATION

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SM	IBSSW	M040	Y					