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William R. Lagergren, Jr.
Site Vice President, Watts Bar Nuclear Plant

10 CFR 50.73

FEB 2 5 2004

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

Gentlemen:

In the Matter of)
Tennessee Valley Authority) Docket No.50-390

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - DOCKET NO. 50-390 - FACILITY OPERATING LICENSE NPF-90 - LICENSEE EVENT REPORT (LER) 50-390/2004-001

This submittal provides Licensee Event Report 390/2004-001. This LER addresses an event that occurred on January 16, 2004, which resulted in automatic actuation of engineered safety features, which included the Reactor Protection and Auxiliary Feedwater systems. This event is being reported under 10CFR 50.73(a)(2)(iv)(A).

If you have any questions about this report, please contact Paul Pace at (423) 365-1824.

Sincerely,

W.R.Lagergren

W. R. Lagergren

Enclosure
cc: See page 2

IE22

cc (Enclosure):

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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)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
Watts Bar Nuclear Plant, Unit 1	05000 - 390	1 OF 6

4. TITLE Automatic Reactor Trip Due to a Invalid Turbine Trip Signal (P-4)

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
										05000
01	16	2004	2004	- 001 -	00	02	25	2004	FACILITY NAME	DOCKET NUMBER 05000

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)
Rickey Stockton, Licensing Engineer	(423) 365-1818

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 EXPECTED SUBMISSION DATE)	x	NO				

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

On January 16, 2004, with Watts Bar Nuclear Plant Unit 1 at 100 percent power, an automatic turbine trip occurred in response to a invalid trip signal (P-4), which then caused an automatic reactor trip because reactor power was above 50 percent power (P-9). The auxiliary feedwater system started as designed.

Surveillance Instruction 1-SI-99-10-B, "31 Day Functional Test of SSPS Train B and Reactor Trip Breaker B," Revision 22, was in progress when an instrument mechanic inserted test leads to take a voltage reading across the P-4 contacts without realizing that the multi-test meter was in the ohms reading position. The result of the volt-ohm meter being in the ohm position was to create a current path equivalent to P-4 contact closure which energized Train B turbine trip bus.

The root cause of this event was determined to be a failure of the involved individuals to follow expectations to "stop" when unexpected conditions occurred. A contributor to the event was improper use of test leads for connection to multi-test meters.

Corrective actions included: 1) appropriate personnel action, 2) requiring an additional management observer to be present during the future test performances, 3) providing lessons learned and reinforcing expectations to site personnel on the use of human performance error reduction tools and appropriate test equipment, 4) and placing affected procedures on administrative hold until precautions are added.

I. PLANT CONDITION(S)

On January 16, 2004, at approximately 1138 Eastern Standard Time, Unit 1 was in Mode 1, steady state operation at 100 percent power. The Reactor Coolant System (RCS) (Energy Industry Identification System (EIIS) Code AB) pressure was 2235 psig and RCS Tavg was 588 degrees F.

II. DESCRIPTION OF EVENT

A. Event

On January 16, 2004, with Watts Bar Nuclear Plant Unit 1 at 100 percent power, an automatic turbine (EIIS Code TRB) trip occurred in response to a invalid turbine trip signal (P-4), which then caused an automatic reactor (EIIS Code RCT) trip because reactor power was above 50 percent power (P-9). The auxiliary feedwater system (EIIS Code BA) started as designed. The invalid trip signal resulted from the introduction of a external circuit which created a current path equivalent to P-4 contact closure.

Surveillance Instruction 1-SI-99-10-B, "31 Day Functional Test of SSPS Train B and Reactor Trip Breaker B," Revision 22, was in progress. The test had progressed through Section 7.0, "Post Performance Activities," Step 24. The test director instructed the Instrument Mechanics (IMs) located in the Reactor Protection System (RPS) (EIIS Code JC) motor generator (MG) Set (EIIS Code MG) Room to perform Steps 24a and 24b, then call him back. The IMs proceeded at Step 24a which is to verify the position of the RPS Trip Breaker B, P-4 contact, using DC voltage measurements. One IM held the volt ohm meter (Triplett) while the second IM plugged the test leads into test points TB4, Terminals 1 and 2. The IMs did not obtain the expected 240 to 290 volts DC. At that point, they changed the volt-ohm meter to ohms. The IM using the test leads then realized one of the test leads had fallen out of the volt-ohm meter. The IM reinserted the test lead and again attempted to take a voltage reading. The volt-ohm meter was most probably in ohms since neither IM recalls switching the volt-ohm meter to volts prior to inserting the test lead. The result of the volt-ohm meter being in the ohm position was to create a current path equivalent to P-4 contact closure which energized Train B turbine trip bus.

B. Inoperable Structures, Components, or Systems that Contributed to the Event

There were no inoperable systems that contributed to this event.

C. Dates and Approximate Times of Major Occurrences:

Time	Event
0903	Authorized performance of 1-SI-99-10-B, "31 Day Functional Test SSPS Train B and Reactor Trip Breaker (RTB) B."
0954	Entered the following Limiting Conditions for Operations (LCOs): LCO 3.3.1, Condition P; LCO 3.3.2, Condition C; LCO 3.3.6, Condition B; LCO 3.3.7, Condition A; LCO 3.3.8, Condition A.
1004	Entered LCO 3.3.1, Condition Q - one RTB maybe bypassed for 2 hour.
1138	Reactor Trip/Turbine Trip.
1140	Transition from E0-0, "Reactor Trip or Safety Injection," to ES-0.1, "Reactor Trip Response."

D. Other Systems or Secondary Functions Affected:

There were no other systems affected other than equipment required for plant shutdown.

E. Method of Discovery:

The operators were first alerted of the event by the annunciation in the control room.

F. Operator Actions:

Operations crew performance for this Reactor/Turbine Trip was satisfactory. At the time of the trip, the Shift Manager, Unit Supervisor, and three Board Operators were in the control room. The operating crew commenced implementation of E-0, "Reactor Trip or Safety Injection." Progress through E-0 and transition to ES-0.1, "Reactor Trip Response" was as expected.

Progress through ES-0.1 was as expected. Auxiliary feedwater system throttling was required to limit RCS cooldown and low pressurizer level. AUOs in the field implemented AOI-17, "Turbine Trip," in a timely manner.

G. Safety System Responses:

Plant safety systems operated as designed.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the trip was the placement of multi-meter test leads across the P-4 contacts with the meter set to read ohms instead of volts.

B. Root Cause:

The root cause of this event was determined to be a failure of the involved individuals to follow expectations to "stop" when unexpected conditions occurred. Without realizing the test lead had pulled loose, the IMs immediately went into the "troubleshooting mode" when the expected voltage was not obtained, by repositioning the meter to read ohms in a effort to determine if the contact was closed. When they discovered that the lead was disconnected, one of the IMs reinserted the test lead and again attempted to take a voltage reading apparently without repositioning the meter back to read voltage. The result of the meter being in the ohms position was to create a current path equivalent to P-4 contact closure which energized Train B turbine trip bus.

C. Contributing Factors

Contributing factor was an accepted improper use of test leads for connection to multi-test meters.

IV. ANALYSIS OF THE EVENT

The principal plant safety systems operated as designed. The investigation of the cause of the trip focused on the performance of the SSPS Reactor Trip Breaker Surveillance Testing and the voltmeter application. With WBN Unit 1 at 100 percent power, the monthly functional test of the B-train Solid State Protection System (SSPS) (EIIS Code JE) and Reactor Trip Breaker B (RTB) was in progress when an automatic plant trip occurred due to an invalid turbine trip signal. At the time of the trip, two instrument mechanics (IMs) were attempting to verify that RTB's P-4 auxiliary contacts were open, as indicated by 240-290 VDC across the contacts. The P-4 contacts of each reactor trip (scram) breaker are closed when the breaker is closed to generate a turbine trip signal.

To accomplish this task, IM-A held the multi-meter (Triplett), while IM-B plugged the test leads into the specified test points in the back of RTB's cubicle. Based on interview, when the multi-meter indicated 0.0 VDC, the IMs "instinctively" went into the troubleshooting mode, wherein they changed the multi-meter to measure ohms to see if the 0.0 VDC was due to the P-4 contacts actually being closed. After changing the multi-meter to read ohms, IM-B then noticed one of the test leads had fallen out of the multi-meter. The test lead was reinserted, and IM-B again attempted to take a voltage reading. At this time, the multi-meter was most probably selected for measuring ohms, since neither IM recalls switching the multi-meter to volts prior to reinserting the test lead. The result of the multi-meter being in the ohm position would be to create a current path equivalent to P-4 contact closure, thereby energizing the B-train turbine trip bus and tripping the turbine. An automatic reactor trip occurred since power was above 50 percent (P-9). This is supported by the absence of "first out" alarms other than those indicating turbine and reactor trips had occurred, and there were no other alarms or indications of any equipment problems.

This event is compared to the LOSS OF EXTERNAL ELECTRICAL LOAD AND/OR TURBINE TRIP as described in Final Safety Analysis Report (FSAR) Section 15.2.7. The complete loss of load/turbine trip from full power is examined to show the adequacy of the pressure relieving devices and also to demonstrate protection from the departure from nucleate boiling (DNB). This plant trip was less challenging than and bounded by the event described in the FSAR. The following plant conditions were bounded by the event described in the FSAR:

1. Reactor power was at 100% and less than the analysis value of 100.6%.

2. The anticipatory reactor trip occurred on turbine trip versus the reactor protection system trip setpoints.

3. Reactor control was in automatic versus manual assumed in the FSAR.

4. Steam dumps operated as designed. The FSAR design basis does not credit the operation of the steam dump system or steam generator power operated relief valves (SG-PORVs) (EIIS Code SG/V).

5. Station Power was not lost during the event.

The reactor trip occurred as designed from a turbine trip. The plant response remained within the FSAR boundary analysis. The main condenser steam dump valves opened per design and as a result it was not necessary for the SG-PORVs to operate. Pressurizer level and pressure did not increase to challenge the pressurizer PORVs and safeties to limit RCS pressure. RCS pressure and loop average temperatures decreased during the transient rather than increasing as predicted by the conservative FSAR assumptions and the DNBR was not challenged. The differences between the FSAR and the plant event are associated with the conservatism assumed in the FSAR analysis and the benign nature of the actual plant event which was quickly brought to a stable condition.

Therefore, there was no safety significance to this event. The plant responded as designed to the initiating condition.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the discussion in Section IV above, there was no safety significance to this event.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

The following actions and those being evaluated in Item B below are tracked under TVA's Corrective Action Program and therefore are not considered to be regulatory commitments. The immediate corrective measures developed to address the above cause of the trip included:

1. Appropriate personnel action was taken for the individuals involved.

2. Standdown meetings were conducted with the appropriate plant personnel on the use of correct Measuring and Testing Equipment (M&TE) leads and self checking practices.

3. Lessons learned from the event were provided to site personnel describing the use of appropriate test equipment, self checking and peer checking expectations.

4. Management observers were required to be present when using volt ohmmeters during the performance of 1-SI-99-10A, "31 Day functional Test of SSPS Train A and Reactor Trip Breaker A," and 1-SI-99-10-B, "31 Day Functional Test of SSPS Train B and Reactor Trip breaker B," and 1-SI-99-4-A, "Trip Actuating Device Operation Test of Reactor Trip P-4 ESFAS Interlock Train A," and 1-SI-99-4-B, "Trip Actuating Device Operation Test of Reactor Trip P-4 ESFAS Interlock Train B."

5. Procedures, 1-SI-99-10-A&B were placed on Administrative Hold until the revisions can be made to provide additional guidance/precautions.

B. Corrective Actions to Prevent Recurrence

Long term items that are being evaluated include:

1. Revising 1-SI-99-10A and -10B to place appropriate precautions at the affected steps.
2. Inspecting shops, toolrooms, and training center to identify and correct similar tool/equipment issues.
3. Developing and conducting training on management observer expectations.
4. Reviewing lessons learned from this event with all WBN Curriculum Review Committees.

VII. ADDITIONAL INFORMATION

A. Failed Components:

There were no failed components which caused this event.

B. Previous LERs on Similar Events:

A review of previous WBN LERs indicated that there had been a number of plant trips but none attributed to the placement of test leads across P-4 contacts that caused a plant trip.

C. Additional Information:

None.

D. Safety System Functional Failure Consideration:

This event is not considered a safety system functional failure in accordance with NEI 99-02 in that the principal plant safety systems operated as designed. Therefore, the functional capability of the overall system was not jeopardized.

E. Loss Of Normal Heat Removal Consideration:

This event is not considered a scram with loss of normal heat removal.

VIII. COMMITMENTS

None.