Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

July 28, 2011

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Subject: **Licensee Event Report 390/2011-004, Reactor/Turbine Trip due to Loss of Main Generator Excitation**

This submittal provides Licensee Event Report (LER) 390/2010-004. This LER documents an incident where the reactor tripped automatically from a turbine trip above 50% rated thermal power due to loss of main generator excitation. The condition is reported as an LER in accordance with 10 CFR 50.73(a)(2)(iv).

There are no regulatory commitments in this letter. Please direct any questions concerning this matter to Robert Clark, WBN Site Senior Licensing Engineer, at (423) 365-1818.

Respectfully,

D. E. Grissette
Site Vice President
Watts Bar Nuclear Plant

Enclosure
cc: See Page 2

Enclosure
cc (Enclosure):

NRC Regional Administrator - Region II

NRC Senior Resident Inspector - Watts Bar Nuclear Plant

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 08/31/2010

Estimated burden per response to comply with this mandatory 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.

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	05000390	1 OF 7

4. TITLE

Reactor / Turbine Trip due to Loss of Main Generator Excitation

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
									N/A	N/A
05	29	2011	2011	- 004 -	0	07	28	2011	FACILITY NAME: N/A	DOCKET NUMBER: N/A

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)(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

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Robert Clark, Senior 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
B	**EL**	**EXC**	**ABB**						

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

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

On May 29, 2011, 01:55 Eastern Daylight Saving Time (EDT) with Watts Bar Nuclear (WBN) Unit 1 at 100% rated thermal power, the reactor tripped automatically on a Main Generator/Turbine trip due to loss of main generator excitation, which resulted from a spurious fault signal from the Extended Gate Controller (EGC) in the Automatic Voltage Regulator (AVR). The most probable cause of the spurious signal was an intermittent fault in a ribbon cable to the EGC. Following cooldown to no-load equilibrium conditions, main feedwater isolation occurred due to Reactor Trip coincident with low Reactor Coolant System T_{AVG}. Main feedwater isolation resulted in a Main Feed Pump Turbine (MFPT) trip signal which resulted in the automatic startup of the AFW System. No overcooling transient occurred and no safety injection signal was initiated for this event.

Plant personnel immediately entered appropriate trip response procedures and stabilized the plant.

The EGC circuit has been bypassed to prevent further spurious trips. The ribbon cable will be replaced at the next outage, and EGC output will be monitored for approximately one cycle to confirm no spurious trip signals before placing the EGC back in service.

This event is reported as an LER in accordance with 10 CFR 50.73(a)(2)(iv) for the valid actuation of the reactor protection system and the auxiliary feedwater system.

I. PLANT CONDITIONS:

Unit 1 in Mode 1 at 100% RTP.

II. DESCRIPTION OF EVENT:

A. Event:

On May 29, 2011 at 01:55 EDT with WBN Unit 1 at 100% rated thermal power the field circuit breakers (FCBs) for the Main Generator Excitation System [EIIS[1] Code EL] received a spurious opening signal without a valid fault command. The opening of the FCBs caused a loss of main generator excitation which was sensed by the generator "Loss of Field" protective relay. The Loss of Field Relay tripped the main generator and turbine. With the reactor operating above the P9 interlock (50% rated thermal power), a Reactor Trip signal was generated by the Solid State Protection System (SSPS). Immediately following the reactor trip, the Steam Dump Control System [EIIS Code J] automatically opened the condenser bypass valves [EIIS Code SB] to remove stored energy and residual heat from the reactor coolant system (RCS) [EIIS Code AB]. During plant cooldown to no-load equilibrium conditions, a main feedwater isolation signal was generated by the SSPS due to Reactor Trip coincident with low RCS T_{AVG}. The purpose of this signal is to prevent a potential RCS overcooling transient following a turbine trip. The closure of the main feedwater isolation valves [EIIS Code SJ] generated a Main Feed Pump Turbine (MFPT) [EIIS Code JK] trip signal which then resulted in the automatic startup of the Auxiliary Feedwater (AFW) System [EIIS Code BA].

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

None.

C. Dates and Approximate Times of Major Occurrences

Date	Time (EDT)	Event
May 29, 2011	01:55:08	EGC Fault
May 29, 2011	01:55:44	SSPS Alarm - Reactor Trip From Turbine Trip Above P9
May 29, 2011	01:56:26	Plant Personnel Entered E-0, Reactor Trip or Safety Injection
May 29, 2011	01:58:54	Plant Personnel Entered ES-01, Reactor Trip Response
May 29, 2011	02:08:20	Plant Personnel Entered AOI-17, BOP Realignment
May 29, 2011	02:24:51	Plant Personnel Entered GO-5, Unit Shutdown from 30% Power to Hot Standby
May 29, 2011	06:27:00	Plant Personnel Exited AOI-17

[1] Energy Industry Identification System

II. DESCRIPTION OF EVENT (continued):

D. Other Systems or Secondary Functions Affected

None.

E. Method of Discovery

SSPS Alarm - Reactor Trip From Turbine Trip Above P9.

F. Operator Actions

Following Reactor/Turbine trip, the operators entered the following procedures to stabilize the plant at Hot Standby Conditions:

E-0 - Reactor Trip or Safety Injection
ES 0.1 - Reactor Trip Response
AOI-17 - BOP Realignment
GO-5 - Unit Shutdown from 30% Power to Hot Standby

G. Safety System Responses

The reactor tripped in response to a Main Turbine trip above the P9 interlock. The turbine trip was due to loss of main generator excitation. Following cooldown to no-load equilibrium conditions, a main feedwater isolation signal was generated by the SSPS due to a reactor trip coincident with low RCS T_{AVG}. The closure of the main feedwater isolation valves generated a Main Feed Pump Turbine (MFPT) trip signal which then resulted in the automatic startup of the AFW System. During plant cooldown, automatic RCS pressure control was not available, however manual control was available. All safety systems functioned as designed. No overcooling transient occurred and no safety injection signal was initiated.

III. CAUSE OF EVENT

Background

The major electrical components that make up the main generator excitation system are the AVR, exciter field windings, exciter armature windings, rotating rectifier assembly, and the main generator field windings (see Figure 1). The AVR is a dual-channel digital control system designed by ABB that was installed during U1 Refueling Outage 10, which ended May 21, 2010. Each AVR channel has one Control Board (COB) module and one Extended Gate Controller (EGC).

The COB module has two operating modes, “manual” and “auto.” In the automatic mode, main generator terminal voltage is maintained at the desired level by automatically adjusting the excitation current within acceptable limits in response to varying grid conditions. In the manual mode, the operator adjusts the excitation current to control both reactive power and main generator terminal voltage. Normally, manual control is avoided because the excitation system will not automatically respond to changes in grid conditions that affect main generator terminal voltage. In both operating modes, maximum and minimum excitation limiters are used to prevent the AVR from imposing unacceptable conditions on the generator.

III. CAUSE OF EVENT (continued)

The Extended Gate Controller (EGC) provides an emergency back-up to the COB. The EGC board is independent from the COB. In case of a failure of both COB channels, excitation control is switched over automatically to the EGC on the active channel. The EGC controller offers manual field current regulation with an AVR over current relay but without excitation limiters. Control may be transferred back to the COB when the COB failure has been corrected. Because of the lack of excitation limiters, the EGC is considered an emergency back-up, and the vendor manual recommends that the plant begin manual shutdown procedures when the AVR is on the EGC.

Fault Analysis:

Testing at ABB Switzerland and ABB Canada facilities indicates that the most probable cause of the loss of excitation was an intermittent fault in the "X5" multi conductor cable that provides signal communication between the EGC and the COB.

During normal COB channel operation in either automatic or manual, the EGC firing pulses are blocked. An intermittent fault in the "X5" multi conductor ribbon cable could cause a spurious signal to be sent to the EGC signaling the EGC to turn on. With the EGC on and no output pulses (pulses blocked), the AVR software will interpret this condition as an EGC fault and open the exciter field breakers. During testing at ABB, a simulated "X5" cable fault replicated the faults and alarms that occurred at WBN on May 29. Other failures replicated some of the alarms or faults experienced, but failure of the"X5" cable was the only tested configuration that produced all the alarms and signals experienced on May 29, 2011.

IV. ANALYSIS OF THE EVENT

Spurious signals generated in the AVR caused the exciter field breakers to open causing a loss of main generator excitation which was sensed by the Loss of Field Relay. The Loss of Field Relay tripped the main generator and turbine. With the reactor operating above the P9 interlock, a reactor trip signal was generated by the SSPS. Immediately following reactor trip, the Steam Dump Control System automatically opened the condenser bypass valves to remove stored energy and residual heat from the RCS. During plant cooldown to no-load equilibrium conditions, a main feedwater isolation signal was generated by the SSPS due to reactor trip coincident with low RCS T_{AVG}. The closure of the main feedwater isolation valves generated a Main Feed Pump Turbine (MFPT) trip signal which then resulted in the automatic startup of the Auxiliary Feedwater (AFW) System.

The steam dump and the pressurizer control systems functioned properly in response to the trip to minimize RCS and steam generator (SG) [EIIS Code SB] temperature and pressure transients. No pressurizer [EIIS Code AB] or SG Power Operated Relief Valves (PORVs) or Safety Valves lifted during the transient. Main Feedwater isolation occurred as designed following plant cool down to prevent a potential RCS overcooling transient. AFW system auto-start on trip of all MFPTs following main feedwater isolation functioned as designed.

V. ASSESSMENT OF SAFETY CONSEQUENCES

The safety significance of this event (Reactor/Turbine Trip) is low because there was no loss of safety function. This event is categorized in the WBN Updated Final Safety Analysis Report (UFSAR) Chapter 15 Accident Analyses as a Condition II event, a fault of moderate frequency. The event is specifically discussed in UFSAR subsection 15.2.7, Loss of External Electrical Load and/or Turbine Trip, which also encompasses loss of main feedwater flow with subsequent startup of the AFW System.

V. ASSESSMENT OF SAFETY CONSEQUENCES (continued)

The UFSAR 15.2.7 analysis contains several analysis conservatisms which were not characteristic of the actual event. The UFSAR analysis assumes, among other things, a loss of offsite power to the station auxiliaries; while offsite power was available for this event. The analysis also assumed that the plant is initially operating at 100.2% of rated thermal power while this automatic reactor trip occurred at 100% power. In addition, the analysis assumes there is a complete loss of steam load from full power without a direct reactor trip, primarily to show the adequacy of the primary and secondary system pressure relieving devices and also to demonstrate core protection margins. In the actual event, direct reactor trip occurred on turbine trip. The analysis also assumed a loss of the Chemical and Volume Control System [EIIS Code CB] and that the water level in the SGs is at a conservatively low level. None of these abnormal conditions existed at the time of the event. Therefore, the actual event was bounded by the UFSAR safety analysis assumptions.

VI. CORRECTIVE ACTIONS

This event was documented within TVA's Corrective Action Program as PER 379346.

A. Immediate Corrective Actions

The EGC circuit was bypassed using TVA's Temporary Alteration (TA) process to prevent a spurious signal from tripping the exciter field circuit breakers.

B. Corrective Actions to Prevent Recurrence

Pursuant to TVA and ABB Root Cause Analysis, the following actions will be taken in accordance with TVA's corrective action program.

1. Replace existing "X5" ribbon cable in the next planned or forced outage.
2. Send suspect "X5" ribbon cable to TVA Central Laboratory for failure analysis.
3. Revise TA to place the EGC in service with the field circuit breaker trip function disabled and instrumented for approximately one operating cycle. At the end of the cycle, evaluate for no spurious trip signals and return EGC to full service.

VII. ADDITIONAL INFORMATION

A. Failed Components

Suspect "X5" ribbon cable and/or EGC module.

B. Previous LERs on Similar Events

An LER search for turbine trips at Watts Bar found no previous events similar to that which occurred on 5/29/11. The existing voltage regulator was replaced with the ABB UNITROL 5000 AVR during WBN U1R10. The replacement AVR had been operating for approximately 10 days prior to this event. Industry OE was reviewed. No similar events were identified with either the EGC or "X5" cable as the root cause. Industry experience dealt with replacement ARCNET card incompatibility and Electromagnetic Interference/Radio Frequency Interferences (EMI/RFI). In one experience corrective action was taken to test all ribbon cables. No problems with cable were identified.

VII. ADDITIONAL INFORMATION

C. Additional Information:

None.

D. Safety System Functional Failure

This event did not involve a safety system functional failure as defined in NEI 99-02, Revision 5.

E. Loss of Normal Heat Removal Consideration

Main Feedwater isolation following reactor/turbine trip is a design feature used to preclude potential RCS overcooling transients. Loss of normal heat removal during this event was expected and was properly mitigated by the automatic startup of the AFW System including operator actions to stabilize the plant at Hot Standby Conditions.

VIII. COMMITMENTS

None.

Figure 1

Simplified Brushless Excitation System Block Diagram