TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

APR 26 1996

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

Gentlemen:

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

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - FACILITY OPERATING LICENSE NPF-90 - LICENSEE EVENT REPORT (LER) 50-390/96011

The enclosure provides LER 50-390/96011 concerning a reportable event involving a manual start of the Auxiliary Feedwater System to mitigate plant conditions that would have eventually led to an automatic actuation. This LER is provided in accordance with 10 CFR 50.73 (a)(2)(iv).

If you should have any questions, please contact P. L. Pace at (423) 365-1824.

Sincerely,

D. V. Kehoe
Nuclear Assurance
and Licensing Manager

Enclosure
cc: See page 2

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PDR ADOCK 05000390
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Enclosure
cc (Enclosure):

NRC Resident Inspector
Watts Bar Nuclear Plant
1260 Nuclear Plant Road
Spring City, Tennessee 37381

Mr. P. S. Tam, Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland 20852

U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

ENCLOSURE

NRC FORM 366 (4-95)

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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TITLE (4)

MANUAL ACTUATION OF AUXILIARY FEEDWATER

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	27	96	96	011	00	04	26	96	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	2
POWER LEVEL (10)	002

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more) (11)

	20.2201(b)		20.2203(a)(2)(v)		50.73(a)(2)(i)		50.73(a)(2)(viii)
	20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
	20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
	20.2203(a)(2)(ii)		20.2203(a)(4)	X	50.73(a)(2)(iv)		OTHER
	20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below
	20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		or in NRC Form 366A

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
R. M. Brown, Compliance Licensing Engineer	(423)-365-8195

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

The motor driven auxiliary feedwater pumps were manually started by the unit operator in response to decreasing steam generator levels and isolation of the intermediate pressure feedwater heater strings while the plant was in Mode 2 just after the unit under went rapid load reduction from 30 percent power and a turbine trip. The load reduction was initiated in response to loss of flow in loop 3 of the Reactor Coolant System due to a failure of the power supply bus during transfer of power from the alternate to normal bus. Approximately two minutes after the motor driven auxiliary feedwater pumps were initiated, the intermediate pressure heaters and main feedwater flow were restored. The motor driven auxiliary feedwater pumps were secured nine minutes later. The cause of the low feedwater flow has been attributed to the isolation of intermediate pressure feedwater heater strings at low power levels after rapid load reduction. Under these conditions, the number 2 heaters do not adequately drain to the number 3 heater drain tank. Intermediate heater isolations occurred because high water level setpoints were reached within the heaters. The manual start of the Auxiliary Feedwater System (engineered safety feature system) was actuated in response to anticipated loss of feedwater. Corrective measures include installing a number 2 feedwater heater string bypass drain to the condenser, revising procedures to move up the steps for placing the number 3 heater drain tanks in full bypass and removing the number 2 heaters from service during the load reduction process.

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

I. PLANT CONDITIONS:

Westinghouse - Pressurized Water Reactor

Pre-Trip Conditions

On March 27, 1996, Watts Bar Nuclear Plant Unit 1 pre-trip status was in Mode 1 at 30 percent reactor power, producing 280 MWe in steady state, preparing for power ascension testing. At approximately 0903 hours EST, the power to reactor coolant pump (RCP) (Energy Industry Identification System (EIIS) code P) number 3 (1-PMP-068-0050) failed to transfer from its alternate power supply bus to its normal power supply bus (EIIS code BU). Coolant flow was lost in Reactor Coolant System (RCS) (EIIS code AB) loop 3. The average temperature in RCS loop 3 briefly dropped below 551 degrees Fahrenheit on two occasions. Rapid load reduction was initiated by the licensed unit operating crew in accordance with the appropriate abnormal operating instruction in preparation to place the unit in Mode 3 as specified in the plant Technical Requirements Manual. At 0930 hours, Unit 1 entered Mode 2 at approximately 2 percent reactor power just prior to the start of the reportable event described below.

II. DESCRIPTION OF EVENT

A. Event

After entry into Mode 2, the water levels in all four steam generators began to decrease. At approximately 0935 hours EST, steam generator number 1 (EIIS code SG) (1-SGEN-068-SG1) reached its "Lo-Lo" level setpoint (17 percent). The trip time delays (EIIS code RLY-2) automatically started as the setpoint was reached (Note: If the timers had timed out, the reactor would have tripped and the Auxiliary Feedwater System (AFW) (EIIS code BA) would have automatically started). At approximately 0939 hours, the motor driven auxiliary feedwater pumps (MDAFWPs) (EIIS code P) (1-PMP-003-0118-A and -0128-B) were manually started by the unit operator in response to decreasing steam generator levels and isolations occurring on the intermediate pressure heaters (EIIS code HX) (1-FWHT-002-A2, -B2, and -B3) in anticipation of loss of feedwater. Approximately two minutes later, the intermediate pressure heaters and Main Feedwater System (MF) (EIIS code SJ) flow were restored. The MDAFWPs were secured after operating eleven minutes.

Problem Evaluation Report WBPER960210 was initiated to document the event in the corrective action program.

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

Refer to Section VI.A.2, "Additional Information."

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C. Dates and Approximate Times of Major Occurrences

09:00	RCP number 3 failed to manually transfer power from the alternate supply (start bus) to the normal supply (unit station service transformer). The RCP number 3 oil pump contact interlock failed to makeup in the normal supply circuit.
09:01:47	RCS loop number 3 average temperature dropped to ≤ 551 degrees Fahrenheit.
09:03	RCS flow stopped in loop 3 and the unit operating crew entered the appropriate abnormal operating instruction.
0903:07	RCS loop number 3 average temperature returned to ≥ 551 degrees Fahrenheit.
09:11	Load reduction was commenced per the appropriate abnormal operating instruction.
09:23:13	RCP loop number 3 average temperature dropped to ≤ 551 degrees Fahrenheit (second occurrence).
09:26	The Steam Dump Control System (EIIS code JI) was placed in the pressure mode.
09:26	The group 1 steam dump valves (EIIS code FCV) (1-FCV-001-103, -107, and -111) opened.
09:26:42	RCS loop 3 average temperature returned to ≥ to 551 degrees Fahrenheit.
09:28	The turbine was manually tripped at 6 percent reactor power.
09:30	Unit 1 entered Mode 2.
09:33	Reactor power held at approximately 3 percent.
09:33:40	Steam generator number 1 power operated relief valve (PORV) (EIIS code PCV) (1-PCV-001-0005-T) lifted.
09:33:42	Steam generator number 2 PORV (1-PCV-001-0012-T) lifted.
09:34:56	Steam generator number 3 PORV (1-PCV-001-0023-T) lifted.
09:35	The trip time delays started when the water level dropped to ≤ 17 percent in steam generator number 1.

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

(continued)

09:39 The intermediate pressure heaters began to isolate (all three heaters were not isolated at the same time).

09:39 The unit operator manually started the MDAFWPs in anticipation of loss of feedwater flow.

09:41 Intermediate heaters returned to service (unisolated).

09:50 Both MDAFWPs were secured.

10:00 Steam generator number 3 PORV closed.

10:07 Steam generator number 2 PORV closed.

10:17 Steam generator number 1 PORV closed.

D. Other Systems or Secondary Functions Affected

The minimum water level for steam generator number 1 reached 5.3 percent during the transient. The minimum levels in steam generator numbers 2, 3, and 4 reached 19.4 percent, 36.9 percent, and 20.8 percent, respectively. Only steam generator number 1 dropped below the "Lo-Lo" setpoint level (17 percent).

E. Method of Discovery

AFW was initiated by the operating crew in anticipation of loss of feedwater flow based upon control room indication.

F. Operator Actions

Operating crew actions are listed within Section C, above. The operating crew actions were conservative and safe.

G. Automatic and Manual Safety System Responses

The trip time delays started as designed when steam generator number 1 water level reached $\leq$ 17 percent.

The AFW performed as designed to restore feedwater inventory and mitigate decreasing levels in the steam generators.

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III. CAUSE OF EVENT

A. Immediate Cause

The Main Feedwater System failed to provide adequate feedwater flow to the steam generators following rapid load reduction from 30 percent power due to isolation of the intermediate heaters at low power.

B. Root Cause

The "Hi-Hi" level setpoint was reached in the number 2 feedwater heater string because of inadequate drainage to the number 3 heater drain tank (EIIS code TK) (1-TANK-006-0104) at the lower reactor power levels. This occurs because the number 2 feedwater heaters are located at a lower elevation than the number 3 heater drain tank. Lower extraction steam pressure at low power does not force drainage up to the number 3 heater drain tank. This phenomenon is especially prone to occur during rapid load reductions when time is not available to ease the number 2 feedwater heaters out of service after placing the number 3 heater drain tank in full bypass operation to the condenser.

Note: At higher power levels, adequate pressure exists in the number 2 feedwater heaters to force drainage up to the number 3 heater drain tank.

Contributing Factors

A contributing cause to the isolations within the intermediate pressure heater string was that procedures were not timely in directing removal of the number 1 and number 2 feedwater heaters from service before low power levels were reached during load reduction.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES

The personnel actions by the unit operating crew were conservative and safe. The MDAFWPs were manually started before the automatic trip time delays timed out. AFW compensated for further loss of feedwater and no anomalies were noted in the actions performed or the AFW behavior in recovering from low feedwater conditions. The number 2 feedwater heaters were restored in two minutes following the AFW actuation. Plant conditions including flow in RCS loop 3 were restored within Technical Specification limits and a reactor trip was not required.

There were no adverse safety implications to the public related to the event.

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V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

General Operating Instruction (GO)-5, "Unit Shutdown From 30 Percent Reactor Power to Hot Standby," has been revised to move up the steps for placing the number 3 heater drain tanks in full bypass operation and removing the number 2 feedwater heaters from service. This will isolate drainage from the moisture separator reheaters low pressure and high pressure drains into the number 2 feedwater heaters and bypass number 3 heater drain tank water to the condenser earlier in the load reduction process.

B. Corrective Actions to Prevent Recurrence

A bypass drain from the number 2 feedwater heaters to the condenser will be installed by the end of the mid-cycle outage.

VI. ADDITIONAL INFORMATION

A. Failed Components

1. Safety Train Inoperability

Normal feedwater flow was temporarily interrupted (low flow) due to the intermediate feedwater heater isolations.

2. Component/System Failure Information

There were no component or system failures which were associated with the reportable event (manual actuation of AFW). AFW responded as designed to restore feedwater flow. Refer to section III.B for a discussion on the intermediate feedwater heater isolations.

Trouble shooting associated with the RCP power transfer failure discovered a problem with the 62RX relay contact interlock (EIIS code RLY-3) in the normal power feed closing circuit bus (EIIS code BU). This contact circuit interlock is closed for normal RCP oil pressure. The relay has been replaced by work order (WO) 960519500 and the RCS average temperature and flow in loop number 3 were restored before applicable Technical Specification Licensee Conditions for Operation (LCOs) expired. The relay contact interlock failure caused the loss of flow in RCP loop 3; however, it was considered unrelated to the reportable event.

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

a. **Method of Discovery of Each Component or System Failure:**

The plant conditions that led to manual actuation of AFW (ESF) were acknowledged through control room indication.

b. **Failure Mode, Mechanism, and Effect of Each Failed Component:**

There were no component failures associated with the reportable event.

c. **Root Cause of Failure:**

There were no component failures associated with the reportable event.

d. **For Failed Components With Multiple Functions, List of Systems or Secondary Functions Affected:**

There were no component failures associated with the reportable event.

e. **Manufacturer and Model Number of Each Failed Component:**

There were no component failures associated with the reportable event.

B. Previous Similar Events

LER 390/96-009 - The lack of drainage from the number 2 heaters to the number 3 drain tank also occurred on the March 13, 1996, turbine trip at low power.

VII. COMMITMENTS

A bypass drain from the number 2 feedwater heaters to the condenser will be installed by the end of the mid-cycle outage.