TVA

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John A. Scalice
Site Vice President, Watts Bar Nuclear Plant

APR 04 1997

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

Gentlemen:

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

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 FACILITY OPERATING LICENSE NPF-90 - LICENSEE EVENT REPORT (LER) 50-390/97006 - REACTOR/TURBINE TRIP DUE TO LOW-LOW STEAM GENERATOR LEVEL

The purpose of this letter is to provide the subject report. The enclosed report provides details concerning the reactor trip which occurred on March 6, 1997.

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

Sincerely,

J. A. Scalice

Enclosure
cc: See page 2

IE22 1/1

cc (Enclosure):

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NRC FORM 366 (4-95)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 04/30/98
	LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS MANDATORY INFORMATION COLLECTION REQUEST: 50.0 HRS. REPORTED LESSONS LEARNED ARE INCORPORATED INTO THE LICENSING PROCESS AND FED BACK TO INDUSTRY. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (T-6 F33), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20603.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Watts Bar Nuclear Plant - Unit 1	05000390	1 OF 10

TITLE (4)

REACTOR/TURBINE TRIP DUE TO LOW-LOW STEAM GENERATOR LEVEL

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	06	97	97	006	00	04	07	97	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1	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)
POWER LEVEL (10)	100		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, 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
B	SJ	P	B580	YES					

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)

On March 6, 1997, at approximately 0517 EST, while Unit 1 was operating at 100 percent RTP, the reactor automatically tripped due to low-low water level in steam generator No. 4. Three conditions led to the low-low level; (1) the main feed pump B (MFP B) shaft fractured and broke in half at approximately 0513, (2) the standby main feed pump was temporarily out-of-service for recirculation line repairs, and (3) flow was partially restricted to steam generator No. 4 at the inlet nozzle. When the shaft broke, the resulting damage caused the pump turbine to trip resulting in an automatic balance of plant runback to 85 percent. Operators further reduced the load to approximately 54 percent, as directed by procedure, to arrest decreasing steam generator water level, since the SBMFP was unavailable. However, steam generator No. 4 was experiencing high inlet pressure due to foreign material obstruction at the feedwater inlet. It decreased to the low-low level trip setpoint at 0517 causing an automatic reactor/turbine trip and subsequent Auxiliary Feedwater System (AFW) actuation at 0518. Operators responding to the trip monitored and observed steam generator No. 4 level off with feed flow approaching steam flow and believed the transient was beginning to reverse, therefore did not initiate a manual reactor trip prior to the automatic trip. AFW responded as designed to control steam generator water levels. The plant was then stabilized in Mode 3. Corrective actions consist of inspecting and repairing hardware, restoring equipment to service, evaluating component failures and material applications, and improving personnel awareness.

I. PLANT CONDITIONS

Watts Bar Nuclear Plant Unit 1 was operating in Mode 1 at approximately 100 percent rated thermal power (RPT) in steady state generating 1203 MWe. The Reactor Coolant System (Energy Industry Identification System (EIIS) code AB) pressure and temperature were at 2235 psig and 588.2 °F, respectively.

II. DESCRIPTION OF EVENT

On March 6, 1997, at approximately 0517 EST, while Unit 1 was operating at 100 percent RTP, the reactor (EIIS code RCT) automatically tripped due to low-low water level in steam generator No. 4 (EIIS code SG). Three conditions led to the low-low level; (1) the main feed pump B (MFP B) (1-PMP-003-B) (EIIS code SJ/P) shaft fractured and broke in half at approximately 0513, (2) the (motor driven) standby main feed pump (SBMFP) (EIIS code SJ/P) (1-PMP-003-0200) was temporarily out-of-service for recirculation line repairs, and (3) flow was partially restricted to steam generator No. 4 at the inlet nozzle. When the shaft broke, the resulting damage caused the pump turbine to trip on "Thrust Bearing Wear." This condition resulted in an automatic balance of plant (BOP) runback to 85 percent. Operators further reduced the load to approximately 54 percent, as directed by procedure, to arrest decreasing steam generator water level, since the SBMFP was unavailable. However, steam generator No. 4 level was experiencing high inlet pressure due to foreign material obstruction at the feedwater inlet. It decreased to the low-low level trip setpoint at 0517 causing an automatic reactor/turbine (EIIS codes RCT/TG) trip and subsequent Auxiliary Feedwater System (AFW) (EIIS code BA) actuation at 0518. Reactor Operators responding to the trip monitored and observed steam generator No. 4 level off with feed flow approaching steam flow and believed the transient was beginning to reverse, therefore did not initiate a manual reactor trip prior to the automatic trip. AFW responded as designed to control steam generator water levels. The plant was then stabilized in Mode 3. All safety systems functioned as designed. All control rods fully inserted as required. There were no challenges to the pressurizer or steam generator safety valves. There were no abnormal radiological conditions throughout the event.

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

1. MFP B (1-PMP-003-B) (EIIS code SJ/P)
2. SBMFP (1-PMP-003-0200) (EIIS code SJ/P)

C. Dates and Approximate Times of Major Occurrences

Time (EST)	Major Occurrences
05:13:30	MFP B turbine vibration high
05:13:30	MFP B turbine abnormal
05:13:37	Steam generator No. 4 steam/feedwater flow mismatch
05:13:37	MFP B turbine oil tank level high-low
05:13:37	MFP A vibration high
05:13:37	MFP A turbine abnormal

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
	05000	YEAR	SEQUENTIAL NUMBER	REVISION	3 OF 10
Watts Bar Nuclear Plant, Unit 1	05000390	97 -	006 -	00	

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

II. DESCRIPTION OF EVENT (continued)

C. Dates and Approximate Times of Major Occurrences (continued)

Time (EST)	Major Occurrences
05:13:38	Steam/feedwater flow mismatch - normal
05:13:42	MFP B turbine "Thrust Bearing Wear"
05:13:43	MFP B trip with automatic turbine runback initiated to 85 percent load followed by manual runback to 800 MW_E due to SBMFP out of service
05:14:00	MFP B bearing oil pressure low
05:16	Decreased turbine load to 700 MW_E to arrest decreasing steam generator levels
05:17	Personnel reported in-board bearing on MFP B blowing oil
05:17	Manually decreased turbine load to 600 MW_E then proceeded to 550 MW_E
05:17:30	Steam generator No. 4 level low-low
05:17:34	Turbine trip/reactor trip breakers open
05:17:35	Rods at bottom
05:18	AFW system initiated to recover steam generator level
05:20	Closed suction and discharge to MFP B
05:27:28	Steam generator No. 4 level low - normal
05:28	Secured MFP B turbine oil pumps

D. Other Systems or Secondary Functions Affected

Other secondary plant equipment discrepancies were identified during the event evaluation that did not impact the reportable event. These and some system enhancements have been documented in the Corrective Action Program for disposition.

E. Method of Discovery

The event was immediately monitored through control room indication as it occurred.

F. Operator Actions

Following the automatic runback, the operating crew manually reduced turbine load to approximately 54 percent in an attempt to prevent a plant trip on low-low steam generator levels. Reactor Operators responding to the trip monitored and observed steam generator No. 4 level off with feed flow approaching steam flow and believed the transient was beginning to reverse, therefore did not initiate a manual reactor trip. Upon automatic trip of the plant, emergency procedure E-0, "Reactor Trip or Safety Injection," was entered, and the plant stabilized in Mode 3 at hot zero power (HZP) conditions. The Operations crew closed the MFP B suction and discharge valves, secured the associated oil pumps to stop sprays and leaks, dispatched clean up crews, and locked out the turbine building (station) sump to prevent oil discharges.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
	05000	YEAR	SEQUENTIAL NUMBER	REVISION	4 OF 10
Watts Bar Nuclear Plant, Unit 1	05000390	97 -	006 -	00	

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

II. DESCRIPTION OF EVENT (continued)

G. Automatic and manual safety system responses

Response of the plant to the MFP B trip was as designed considering two pre-existing conditions. The SBMFP, which normally starts for the loss of a turbine driven MFP, had been tagged out for repairs to its recirculation line. Also, steam generator No. 4 had a partial flow restriction at its inlet nozzle which limited the ability to maintain water level during the event. These items had been reviewed and discussed prior to the event under the Corrective Action Program with no impact to safe plant operation found; however, it was recognized that an increased reactor/turbine trip risk existed. Plant Abnormal Operating Instructions (AOIs) contained adequate mitigation actions in the event of a feedwater transient.

The response of the plant to the turbine/reactor trip was as designed. With the resulting Main Feedwater System (EIIS code SJ) isolation signal, the AFW automatically started and restored level to all steam generators. Overall the responses of the plant systems and components to the automatic and operator commands were correct allowing the plant to be stabilized in Mode 3.

III. CAUSE OF EVENT

Because plant design does not consider avoiding a reactor trip with the loss of a MFP while the SBMFP is tagged out-of-service, the root cause of this event is considered to be the loss of MFP B. Contributing causes were the unavailability of the SBMFP (tagged for repairs to its recirculation line) and the flow blockage in the inlet nozzle of steam generator No. 4.

Main Feed Pump

Preliminary laboratory analysis indicates that the MFP B shaft failed as a result of fatigue. A crack likely propagated from the corner of a step machined in the shaft and was either initiated from a pre-existing flaw or the sharp radius of the corner. The shaft is stainless steel (type 410) and chrome plated in the area of failure. The crack initiated at the outside diameter surface and propagated from that location toward the opposite side until there was no longer enough material to sustain the load at which time the shaft failed. The shaft material hardness met the requirements for 410 stainless steel (equivalent to Class 1). The chrome plating contained numerous cracks generating from the outer surface inclusions/voids and was not tightly adhered. Also, the final tear was intergranular which indicates very low impact properties. There are also preliminary indications of improper heat treating.

Motor Driven Standby Main Feed Pump

At the time of the event, the SBMFP was tagged out of service. This pump had been secured due to erosion repairs on its recirculation line.

III. CAUSE OF EVENT (continued)

Steam Generator No. 4 Flow Restriction

A boroscopic examination of the inlet nozzle to the steam generator revealed Flexitallic gasket material partially blocking several holes in the first inlet plate. The blockage had been identified by a pressure increase on the feedwater inlet while the plant was operating. The increase occurred at the same time as a Furmanite temporary leak seal repair on a flange in the same line. During flange disassembly, TVA determined that the gasket material had been pushed into the flow stream by the leak sealing compound.

Four of the feedwater flow venturi flanges had leaks repaired using the same Furmanite procedure. All flanges have been disassembled and inspected during this outage. Two of the flanges had no evidence of Furmanite compound intruding into the gasket seating area. One flange had a small amount of compound between the gasket backing ring and flex material in two places. The flange in question had approximately the top half of the flex material pushed out by the Furmanite compound.

The gaskets installed in the flow venturi flanges were determined to be the incorrect type. A records search indicated that the gasket installation occurred during final construction prior to initial licensing of Unit 1. Flexitallic type CG gaskets were found to be installed instead of type CGI. Type CGI has the same ratings as the type CG; however, it includes an internal metal backing ring. Had the correct gaskets been installed, it is unlikely the gasket material could have shifted and entered the flow stream. A warehouse stock number search was performed to determine if the gasket had been misapplied in other locations. No other problems were found.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES

A. Evaluation of Plant Systems/Components

The main feedwater system was operating with MFP A and B in service prior to the event. For the event, the plant responded as designed except for automatic start of the motor driven SBMFP. This pump had been secured for standby line-up due to erosion repairs on its recirculation line. Since the maximum capability of the remaining feedwater pump is approximately 67 percent power, the runback to 85 percent power would have had limited success for preventing loss of level in the steam generators and automatic reactor trip. In addition, one steam generator was experiencing high inlet pressure due to a foreign material partial obstruction at the feedwater inlet (not connected to the MFP B failure). This resulted in reduced feedwater flow which caused one steam generator (No. 4) to lose water level to the low-low level trip setpoint. At the time of trip, Operations was attempting to rapidly reduce load to the capability of one MFP and to achieve a match of steam flow to feedwater flow for each of the steam generators as specified by procedure when the SBMFP is unavailable. This would arrest water level decrease if successful. Transient data indicates that a power reduction to less than 67 percent had been achieved prior to trip. Just prior to the trip, reactor power was approximately 54 percent, steam generator No. 1 level was 32 percent and stable, steam generator No. 2 level was 30 percent and stable, steam generator No. 3

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IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES (continued)

level was 30 percent and stable, and steam generator No. 4 level was approximately 18 percent and slowly trending downward. Because of the slight decrease of this level, the plant trip point of 17 percent was reached for low-low level.

B. Evaluation of Personnel Performance

Control room operator action during and following the initiating event was fully adequate. Three way communication was timely and precise. Appropriate procedure response was demonstrated with proper transition based on plant parameters. Control room notification was immediately conveyed to the plant and appropriate personnel upon MFP B trip and the resulting turbine/reactor trip. During the automatic and manual decrease in unit load, key plant parameters were monitored to ensure adequate plant response. Steam generator levels were monitored closely. Upon control room notification of the MFP B trip and the resulting turbine/reactor trip, assistant unit operators (AUOs) provided timely plant response. Other plant personnel (i.e., maintenance, laborers, fire operators, radiological control, and security) demonstrated timely effective support and good teamwork in containing and cleaning up of the oil spill following the MFP B failure. Timely actions were taken to remove the station sump pumps from service due to the oil spill, thereby preventing a challenge to the environment. Teamwork was consistently demonstrated throughout the event among members of the control room staff and resulting interactions with the plant support personnel. During the load reduction, good verbal communication among the control room staff was maintained which insured awareness of each upcoming evolution and expected equipment response.

C. Safety Significance

The partial loss of feedwater transient experienced at Watts Bar on March 6, 1997, is bounded by FSAR described in Section 15.2.8, "Loss of Normal Feedwater." The loss of feedwater results in a reduced ability to remove heat generated in the reactor core. Protection systems are provided in the plant design to prevent the loss of heat sink from allowing core heat up and damage. These protective systems include:

1. Reactor trip on low-low water level in any steam generator.

2. Automatic start of AFW motor driven pumps (1-PMP- 003-0118-A and 1-PMP-003-0128-B) (EIIS code BA/P) on low-low level in any one generator or trip of both turbine driven MFPs.

3. Automatic start of the turbine driven AFW pump (1-PMP-003-0001A-S) (EIIS code BA/P) on low-low level in any two steam generators or trip of both turbine driven MFPs.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES (continued)

These features restore the ability of the system to provide adequate core cooling by reducing the heat produced in the reactor core (reactor trip) to decay heat values and providing a guaranteed minimum safety grade auxiliary feedwater flow for the steam generators to remove heat. Analyses were performed for this event demonstrating the protective features are adequate to prevent over pressurization of the Reactor Coolant System (RCS) (EIIS code AB) and loss of water from the reactor core. Conservatism in the FSAR analysis include:

1. Steam generator levels at initially low values.
2. Plant operating at 102 percent power.
3. Steam generator heat transfer coefficients consistent with natural circulation.
4. Failure of the turbine driven AFW supply.
5. Maximum AFW supply temperature.
6. Steam generator steam relief via safety valves.
7. High initial RCS average temperature and pressurizer level.
8. Steam generator low-low level trip assumed to be zero percent narrow range (58 percent wide range).

The actual plant transient experienced is less severe in that steam generator initial water levels were normal, plant power was approximately 100 percent, offsite power was not lost during the event, steam generator heat transfer was consistent with normal RCS flow, AFW started and operated normally, AFW temperature was not at the high limit, steam dumps to the condenser were available and operated, RCS initial conditions were nominal, and reactor trip on low-low level occurred at a level greater than zero percent narrow range. Response of the plant is therefore bounded by the FSAR event. The loss of feedwater event did not challenge the pressurizer safety valves by water discharge and did not result in core cooling problems. The Watts Bar plant design also contains features to accommodate partial loss of feedwater events. These features are described in FSAR section 10.4. Normally a partial loss of feedwater due to MFP trip is accommodated by design features intended to mitigate the partial loss of feedwater and keep the plant in operation. These features include:

1. A turbine runback from 100 percent to approximately 85 percent power.
2. Acceleration of the remaining feed pump to its high speed stop.
3. Isolation of the tripped feed pump turbine condenser.
4. Automatic start of the motor driven SBMFP.

Note: The fact that the SBMFP was unavailable during the subject event is not in conflict with the FSAR.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

A team was assembled consisting of members of various plant organizations divided into day and night shifts to investigate this event.

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

V. CORRECTIVE ACTIONS (continued)

B. Corrective Actions to Prevent Recurrence

Main Feed Pump

The section of failed pump shaft was sent to the TVA lab for analysis. The preliminary results indicated that the shaft broke as a result of fatigue. The crack initiated at one location and propagated from that location toward the opposite side until there was no longer enough material to sustain the load at which time it failed.

As a result of these findings, the replacement shaft and the shaft on MFP A were ultrasonically and dye-penetrant tested to determine if a similar flaw existed. Both the new shaft and the MFP A shaft exhibited no indications of the type found on the broken shaft. Sequoyah Nuclear Plant (SQNP) was contacted and it was determined that no failure of this type had occurred with their MFPs. Also, the pump manufacturer (Byron-Jackson) was contacted and they stated that no similar failures had occurred with any pumps of similar design currently installed at other facilities.

An investigator from FPI (TVA's cause analysis consultant) reviewed all the data collected and performed his own investigation into the failure. FPI's preliminary findings agreed with TVA's findings. The FPI investigation is still on-going at this time to ensure all possible failure modes are investigated.

The MFP B damage repairs have been completed and the pump has been restored to service. The MFP B turbine has been inspected for damage from the event and no damage was found.

Standby Main Feed Pump

Repairs have been completed on the SBMFP recirculation lines.

Steam Generator Inlets

The gasket material that had been restricting flow to steam generator No. 4 has been removed. The incorrect gaskets have been replaced.

A warehouse stock history search was performed to determine if the gasket had been misapplied in other applications. No other problems were found.

The feedwater piping has been inspected from the flow venturi to the steam generator inlet nozzles for foreign material for steam generator No. 4.

The Furmanite compound has been removed and flanges have been repaired for each feedwater flow venturi where Furmanite had been used.

V. CORRECTIVE ACTIONS (continued)

A training memorandum has been issued to planners and craft supervision discussing this event and the need to properly verify parts installed are correct per design documents.

VI. ADDITIONAL INFORMATION

A. Failed Components

1. Safety Train Inoperability

None.

2. Other Component/System Failure Information

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

Failure of MFP B was indicated and monitored by control room annunciation.

Flow restriction in the No. 4 steam generator was also indicated by control room displays.

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

Preliminary results indicate that the shaft of the MFP B broke as a result of fatigue. Refer to Section IV for additional details.

Flexitallic gasket material partially blocked the first inlet plate to steam generator No. 4. The gasket material originated from the top half of the inlet feedwater flow venturi flange. It was pushed out by the injection of Furmanite compound during leak repairs. The gasket installed in the flow venturi flange was determined to be the incorrect type. Had the correct gasket been installed, it is unlikely the gasket material could have shifted and entered the flow stream.

c. Root Cause of Failure:

Refer to details in Section III.

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

No failures of this type occurred.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
	05000	YEAR	SEQUENTIAL NUMBER	REVISION	10 OF 10
Watts Bar Nuclear Plant, Unit 1	05000390	97 –	006 –	00	

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

VI. ADDITIONAL INFORMATION (continued)

e. Manufacturer and Model Number of Each Failed Component:

Byron Jackson MFP pump model No. 20X20X18B HDR, shaft reference No. 167

Flexitallic gasket model CG (incorrect application)

B. Previous Similar Events

No other similar events have occurred at Watts Bar.