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

FEB 1 1 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) 50-390/97001 - NONCOMPLIANCE WITH TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT (SR) 3.3.2.11

The purpose of this letter is to provide the subject report. The enclosed report provides details concerning the noncompliance with TS SR 3.3.2.11 which requires the verification of the P-4 interlock functions.

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

Sincerely,

J. A. Scalice

Enclosure
cc: See page 2

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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
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.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Watts Bar Nuclear Plant - Unit 1	05000390	1 OF 6

TITLE (4)

NONCOMPLIANCE WITH TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 3.3.2.11

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
1	13	97	97	001	00	2	10	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)			
POWER LEVEL (10)	100	20.2201(b)	20.2203(a)(2)(v)	X 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)	50.73(a)(2)(iv)	OTHER - Specify in Abstract below or in NRC Form 366A
		20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	
		20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vii)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Rickey Stockton, Licensing Engineer	(423)-365-1818

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)

On January 13, 1997, with Unit 1 operating in Mode 1 at 100 percent rated thermal power (RTP), it was determined that a misinterpretation of Technical Specification (TS) Surveillance Requirement (SR) 3.3.2.11 had occurred resulting in the SR not being performed for two of the five P-4 interlock functions. The P-4 interlock as described in the TS basis performs the following five functions: 1) turbine trip, 2) isolate main feedwater coincident with low Tavg, 3) prevent re-actuation of safety injection (SI) after manual reset, 4) transfer the steam dump controller from load rejection to unit trip controller, and 5) prevent opening of the main feedwater valves if closed due to SI, high-high steam generator water level, or high main steam valve vault water level. Since the P-4 interlock is described in TS table 3.3.2-1 and in the basis as an Engineered Safety Feature Actuation System (ESFAS) interlock, this was interpreted to mean that the only functions requiring SR verification were the functions that established the ESFAS interlock. Two of the above functions (Items 1 and 4) are not ESFAS actuations and were considered not to require the SR verification. However, based on re-examination of this interpretation, it was determined that SR verification of these functions was required.

Corrective action required revisions of SIs 1-SI-99-4-A, -4-B, -10-A, and -10-B to incorporate the SR verifications for the two functions, revisions of 1-SI-85-1, 1-SI-85-4, 1-SI-99-201-A and 1-SI-99-201-B to be made prior to refueling outage performances, and feedback provided to the individuals involved in initial interpretation of the SR.

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

I. PLANT CONDITIONS

Watts Bar Nuclear Plant Unit 1 was operating in Mode 1 at 100 percent RTP.

II. DESCRIPTION OF EVENT

A. Event

On January 13, 1997, with Unit 1 operating in Mode 1 at 100 percent RTP, it was determined that a misinterpretation of TS SR 3.3.2.11 had occurred resulting in the SR not being performed for two of the five P-4 interlock functions. The P-4 interlock as described in the TS basis performs the following five functions: 1) turbine trip, 2) isolate main feedwater coincident with low Tavg, 3) prevent re-actuation of safety injection (SI) after manual reset, 4) transfer the steam dump controller from load rejection to unit trip controller, and 5) prevent opening of the main feedwater valves if closed due to SI, high high steam generator water level, or high main steam valve vault water level. Since the P-4 interlock is described in TS table 3.3.2-1 and in the basis section 3.3.2.8 as an ESFAS [Energy Industry Identification System (EIIS) Code JE] interlock, this was interpreted by plant and engineering personnel to mean that the only functions requiring SR verification were the functions input through to the solid state protection system (EIIS Code JC/JG) to establish the ESFAS interlock. Two of the above functions (Items 1 and 4) are not ESFAS actuations and were considered not to require the SR verification. Accordingly, the surveillance instructions that were written and performed to satisfy this SR used this interpretation. However, based on re-examination of this interpretation, it was determined that the SR verification should also include functions 1 and 4.

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

There were no inoperable structures, components or systems that contributed to this event.

C. Dates and Approximate Times of Major Occurrences

DATE	TIME	EVENT
1/13/97		Re-examination of the P-4 interlock functions identified the need for SR verification of functions 1 and 4.
1/13/96	1645	Shift Manager identified Surveillance Requirement 3.0.3 as applicable to condition. Tracked SR completion within 24 hours for the two remaining P-4 functions.

1/13/97	1645	Work Request C138822 (Work Order 97-00532-00) was initiated to provide the required SR verifications.
1/13/97	2051	Entered LCO 3.3.1 Q on Reactor trip breaker A and bypass to perform SR verifications.
1/13/97	2102	Exited LCO 3.3.1.Q on Reactor trip breaker A and bypass upon completion of SR verifications.
1/13/97	2114	Entered LCO 3.3.1.Q on Reactor trip breaker B and bypass to perform SR verifications.
1/13/97	2122	Exited LCO 3.3.1.Q on Reactor trip breaker B and bypass upon completion of SR verifications.
1/13/97	2142	SRs 3.0.3 and 3.3.2.11 requirements were deemed satisfied.

D. Other Systems or Secondary Functions Affected

No other systems or secondary functions were affected.

E. Method of Discovery

Upon re-examination resulting from questions asked by a Sequoyah Nuclear Plant (SQN) system engineer and a subsequent review of a draft SQN Problem Evaluation Report, the previous interpretation of the P-4 interlock function verification was determined to be incorrect.

F. Operator Actions

Shift Manager identified Surveillance Requirement 3.0.3 as applicable to this condition and tracked SR completion within 24 hours for the two remaining P-4 functions.

G. Automatic and Manual Safety System Response

There were no automatic or manual safety system responses due to this condition.

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

III. CAUSE OF EVENT

The root cause of this event was determined to be a misinterpretation of the Technical Specification as to which P-4 interlock functions were to be verified by SR 3.3.2.11. This misinterpretation led to the SI being written and performed using this false assumption.

IV. ANALYSIS OF EVENT - ASSESSMENT OF SAFETY CONSEQUENCES

A. Evaluation of Plant Systems/Components

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker are open. Once the P-4 interlock is enabled, automatic SI initiation may be blocked after a 90 second time delay. This function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed. The functions as previously stated are: 1) trip the main turbine, 2) isolate main feedwater coincident with low Tavg, 3) prevent re-actuation of safety injection (SI) after manual reset, 4) transfer the steam dump controller from load rejection to unit trip controller, and 5) prevent opening of the main feedwater valves if closed due to SI, high high steam generator water level, or high main steam valve vault water level.

B. Evaluation of Personnel Performance

Once recognized that a misinterpretation had occurred, the Shift Manager initiated actions to satisfy the requirements of Surveillance Requirement 3.0.3 and tracked SR completion within 24 hours for the two remaining P-4 functions.

C. Safety Significance

Each of the five functions is interlocked with P-4 to avert or reduce the continued cooldown of the reactor coolant system (EIIS Code AB) following a reactor (EIIS Code RCT) trip. An excessive cooldown of the reactor coolant system following a reactor trip could cause an insertion of positive reactivity with a subsequent increase in generated power. To avoid such a situation, the noted functions have been interlocked with P-4 as part of the design of the unit control and protection system.

As a result of the successful completion of the SR verification of the remaining two P-4 functions, it can be concluded that these functions would have performed their intended function. However, no credit is taken in the plant safety analysis for these two functions to mitigate a design basis event. Therefore, there was no safety significance associated with this condition.

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

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Work order 97-00532-00 was performed to verify the P-4 functions not previously verified.

B. Corrective Actions to Prevent Recurrence

Surveillance Instructions 1-SI-99-4-A, "Trip Actuating Device Operations Test of Reactor Trip P-4 ESFAS Interlock Train A," 1-SI-99-4-B, "Trip Actuating Device Operations Test of Reactor Trip P-4 ESFAS Interlock Train B," 1-SI-99-10-A, "31 Day Functional Test of SSPS Train A and Reactor Trip Breaker A," 1-SI-99-10-B, "31 Day Functional Test of SSPS Train B and Reactor Trip Breaker B," were placed on administrative hold until revisions were made to incorporate the SR verifications for the two non-ESFAS functions. Surveillance Instructions 1-SI-85-1, "Rod Drop Time Measurement By Simultaneously Dropping All Rod Banks," 1-SI-85-4, "Rod Drop Time Measurement By Dropping Individual Rod Banks," 1-SI-99-201-A, "Response Time Test of Reactor Trip Train A," and 1-SI-99-201-B, "Response Time Test of Reactor Trip Train B," have also been placed on administrative hold and will be revised prior to the next scheduled performance which will occur at the first refueling outage. Feedback concerning this condition has been provided to the individuals involved in initial interpretation of the SR. Since it has been determined through the reviews of other interlocks that this condition is limited to the reactor trip breakers and the P-4 interlock, no further recurrence control actions are deemed necessary.

VI. ADDITIONAL INFORMATION

A. Failed Components

1. Safety Train Inoperability

Although no component failure as a result of this condition occurred, the two functions of P-4 were considered inoperable due to not having been previously verified.

2. Component/System Failure Information

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

There was no component failure as a result of this condition.

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

There was no component failure as a result of this condition.

c. **Root Cause of Failure:**

There was no component failure as a result of this condition.

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

There was no component failure as a result of this condition.

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

There was no component failure as a result of this condition.

B. **Previous Similar Events**

A review of previous WBN LERs identified three others that were similar in nature. These are 390/95001, 390/96018, and 390/96022. LERs 390/95001 and 390/96018 involved a misinterpretation of the implementing procedure. LER 390/96022 involved a misinterpretation of SR 3.1.7.3. Corrective actions were taken to address these conditions as described in each LER.

VII. COMMITMENTS

Surveillance Instructions 1-SI-85-1, 1-SI-85-4, 1-SI-99-201-A, and 1-SI-99-201-B will be revised prior to the next scheduled performance which will occur at the first refueling outage.