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James J. Fisicaro
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April 10, 1996

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

Subject: River Bend Station - Unit 1
Docket No. 50-458
License No. NPF-47
Licensee Event Report 50-458/96-008-01

File Nos.: G9.5, G9.25.1.3

RBG-42782
RBF1-96-0092

Gentlemen:

In accordance with 10CFR50.73, enclosed is Revision 1 to Licensee Event Report 50-458/96-008. This revision clarifies the description of corrective actions taken as a result of the event. This revision has been discussed with Mr. Ward Smith, Senior Resident Inspector at River Bend Station.

Sincerely,

George Ben Zinke for

JJF/JPO/jr
enclosure

150084

IE22 1/1

xc: U. S. Nuclear Regulatory Commission
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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 20503.

LICENSEE EVENT REPORT (LER)

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

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
River Bend Station	**05000-458**	**1 OF 5**

TITLE (4)

Mispositioned Drywell Pressure Transmitter Isolation Valve Causing a Condition Prohibited by Technical Specifications

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
02	14	96	96	-- 008 --	01	04	10	96	N/A	05000
									N/A	05000

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

OPERATING MODE (9)	1								
POWER LEVEL (10)	59		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
			20.2203(a)(2)(iii)		50.36(c)(1)		50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			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)
David Lorfing, Licensing Supervisor	**(504) 381-4157**

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	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 February 14, 1996, with the plant in Mode 1 (Power Operation) at 59% power, one trip unit channel for Drywell Pressure failed its channel check. Subsequent investigation revealed that a drywell pressure transmitter isolation valve, A4-B21*N094E, was closed. This condition is reportable as an operation prohibited by Technical Specifications.

The cause of this event is indeterminate. A Significant Event Response Team (SERT) performed an investigation including a search of maintenance, testing, and operational history. No work documents or procedures that would have involved repositioning this valve were identified. Interviews were conducted with plant personnel, however, these interviews failed to identify a cause for the valve being out of position that could be verified and validated. The SERT believes the most probable cause is that the valve was intentionally closed by someone who was directed by procedure to close another valve (right action on the wrong component/failure to self-check).

Corrective actions included system restoration and position verification of other safety related instrumentation valves. The condition was determined not to have significant impact on safe plant operation.

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

REPORTED CONDITION

On February 14, 1996, with the plant in Mode 1 (Power Operation) at 59% and raising power from Refuel Outage Six (RF-06), one trip unit channel of indication for Drywell Pressure failed its channel check during the performance of the Daily Operating Logs. Subsequent investigation revealed that a drywell pressure transmitter isolation valve(*ISV*), A4-B21*N094E, was closed.

Technical Specifications (TS) 3.3.5.1 requires two channels per function of Drywell Pressure-High instrumentation to be operable in Modes 1, 2, and 3. TS 3.3.5.1 requires the affected channel to be placed in the trip condition within 24 hours. TS 3.5.1 allows 72 hours to restore the inoperable system prior to commencing shutdown. Since this problem was not identified until after plant start-up, the mode change restriction per Limiting Condition for Operation (LCO) 3.0.4 was not met. On February 12, 1996 at 1900 the plant was placed in Mode 1 (from Mode 2) with the transmitter inoperable. This is a violation of TS 3.0.4 and is reportable as a condition prohibited by Technical Specifications pursuant to 10CFR50.73 (a)(2)(i)(B).

INVESTIGATION

On February 14, 1996, at 1303, an operator performing Surveillance Test Procedure (STP) 000-0001 "Daily Operating Logs," noticed that the trip unit, B21-N694E High Drywell Pressure , was reading 0.4 psi with all other channels reading 0 psi. Acceptance criteria for this channel check is for all channels to be within 0.3 psi. Therefore this instrument failed its channel check. A Limiting Condition for Operation (LCO) was initiated at 1303. The LCO allows 24 hours before placing the channel in a tripped condition. LCO 3.0.6 was entered for this condition and a Loss of Safety Function Evaluation was completed. The operator contacted the Instrumentation and Controls group (I & C), and initiated a troubleshooting Maintenance Action Item (MAI) to investigate the cause of this deviation. Condition Report (CR) 96-0499 was also initiated to identify the deficiency.

While performing the MAI, an I & C technician recognized that the normally open drywell pressure transmitter isolation valve, A4-B21*N094E, was closed. Condition Report 96-0503 was initiated.

At approximately 1630 on February 14, 1996, the I & C technician contacted his supervisor in the main control room and obtained permission to open the valve and return the transmitter to service. The LCO was cleared at 2103 on February 14, 1996. An instrumentation valve lineup was initiated to ensure all other valves in the Nuclear Boiler Instrumentation system were properly aligned. No other Nuclear Boiler Instrumentation system valves were found out of position.

A Significant Event Response Team (SERT) was formed on February 15, 1996, to investigate this event.

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TEXT CONTINUATION

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

The team determined that the valve was most likely in the proper position January 31, 1996. The valve had been checked and independently verified open on January 16, 1996, and again on January 31, 1996. These checks were performed by four different I&C technicians. In addition, an Engineering evaluation of December 1995's operating logs determined that the transmitter was not isolated in December since all the drywell pressure instruments tracked similarly.

A review of the clearance (tagging) system, manipulated device log, main control room logs, maintenance action items (MAIs), modification requests (MRs), surveillance test procedures, and outage records failed to reveal any reason for this valve to be intentionally closed in the time period between January 31, 1996, and February 14, 1996. In addition a computer search was done and no permanent procedure was found that ever directs this valve to be closed at any time. Interviews with Operators, I&C Technicians and Engineers failed to reveal any reason that this valve would have been intentionally manipulated.

A system line-up verification was directed to be performed on the High Pressure Core Spray (HPCS) System and one instrument valve on a non-safety related, out-of-service transmitter was found out of position. This valve was restored to its proper position. As a result of this valve being found out of position, the position of all accessible safety-related instrumentation valves was verified, as well as two complete System Operating Procedures (SOP) lineups including electrical breakers and non-instrumentation valves. No other components were found out of position.

To aid in the investigation of this event the causes and corrective actions of five Licensee Event Reports (LERs) were reviewed including: LER 87-017 "Technical Specification Violation due to Incorrectly Positioned Instrument Root Valve," LER 87-027 "Mispositioned instrument Valve Renders Leak Detection System Inoperable," LER-89-030 "Mispositioned Pressure Transmitter Isolation Valve Found Misaligned Causing Inability to Sense Drywell Pressure, a Condition Prohibited by Technical Specification 3.0.4," LER 92-018 "Trip System for the "A" Automatic Depressurization System Inoperable due to Mispositioned Root Valve," and LER 93-024 "Reactor Scram During Turbine Testing Due to Failure of Relay Contacts Open."

CAUSE(S)

The cause of this event is indeterminate. The SERT performed an investigation including a search of maintenance, testing, and operational history. No work documents or procedures that would have involved repositioning this valve were discovered. Interviews were conducted with plant personnel, however, these interviews failed to identify a cause for the valve being out of position that could be verified and validated. The SERT believes the most probable cause is that the valve was intentionally closed by someone who was directed by procedure to close another valve (right action on the wrong component/failure to self-check).

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

CORRECTIVE ACTIONS

Immediate corrective actions were performed including restoring the transmitter to service, verifying the Nuclear Boiler Instrumentation line-up in accordance with System Operating Procedure (SOP)-0001, performing a HPCS instrumentation valve line-up in accordance with SOP-0030, and performing a verification of accessible safety related instrumentation valves (a total of approx. 2,650 valves). Additionally, verification was performed of accessible non-instrument valve line-ups for two SOPs, and of the electrical line-ups for two SOPs.

A natural work team will review mispositioning events and line-ups for improvement areas. The mark number and noun name description used for labeling instrument valves and in procedures directing operation of instrument valves will be evaluated. In addition other program enhancements are ongoing as part of River Bend's Corrective Action Program.

SAFETY SIGNIFICANCE

This investigation revealed that a drywell pressure transmitter isolation valve, A4-B21-N094E, which feeds Drywell Pressure Transmitter B21-PTN094E was closed. This pressure transmitter provides input for the following Division I systems: Low Pressure Core Spray (LPCS) Initiation, Low Pressure Core Injection (LPCI) A Initiation, Division 1 Emergency Diesel Generator (EDG) start, Containment Unit Cooler 1A (HVR-UC1A) Start, Containment Sampling Valves SSR-SOV131, 133, 139, and 140 close, Hydrogen Mixing Valves CPM-MOV1A, 2A, 3A, 4A close, and Automatic Depressurization System Channel E High Drywell pressure permissive.

Since this transmitter affects the availability of these safety systems, a safety assessment was performed to determine if the loss of this pressure transmitter affects any accident scenario. The isolation valve to pressure transmitter B21-PTN094E was last verified open on January 31, 1996. Between the last known verification and the time of discovery (February 14, 1996), there were several maintenance activities on Division II equipment, most notably the Division II Emergency Diesel Generator. Therefore, the safety assessment assumed both a loss of function of the high drywell pressure transmitter, B21-PTN094E , and a loss of the Division II EDG.

Several Accident scenarios were evaluated to determine if the loss of B21-PTN094E impacted the results of these accidents. Based on the results of the design basis accident scenarios of interest, only one scenario would be affected by the closure of the isolation valve to B21-PTN094E. This scenario is the small break LOCA concurrent with a loss of offsite power. The only change to the design basis for this scenario is the use of a manual action to align SSW to containment unit cooler HVR-UC1A instead of an automatic actuation. Adequate procedures and guidance exist to perform this action within the 10 minute start time in the Updated Safety Analysis Report (USAR). The condition was determined not to have significant impact on safe plant operation.

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Note: Energy Industry Identification System Codes are identified in the text as (*XX*)