

Georgia Power Company
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201
Telephone 205 877-7279

J. T. Beckham, Jr.
Vice President—Nuclear
Hatch Project



HL-2356
003816

August 4, 1992

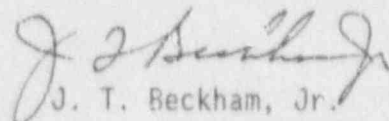
U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

PLANT HATCH - UNIT 2
NRC DOCKET 50-366
OPERATING LICENSE NPF-5
LICENSEE EVENT REPORT
CONDENSATION IN INSTRUMENT SENSING LINES RESULTS
IN NONCOMPLIANCE WITH THE TECHNICAL SPECIFICATIONS

Gentlemen:

In accordance with the requirements of 10 CFR 50.73 (a), Georgia Power Company is submitting the enclosed Licensee Event Report (LER) concerning condensation in instrument sensing lines which resulted in a condition of noncompliance with Technical Specifications requirements. This event occurred at Plant Hatch - Unit 2.

Sincerely,


J. T. Beckham, Jr.

JKB/cr

Enclosure: LER 50-366/1992-010

cc: Georgia Power Company
Mr. H. L. Sumner, General Manager - Nuclear Plant
NORMS

U.S. Nuclear Regulatory Commission, Washington, D.C.
Mr. K. Jabbour, Licensing Project Manager - Hatch

U.S. Nuclear Regulatory Commission, Region II
Mr. S. D. Ebnetter, Regional Administrator
Mr. L. D. Wert, Senior Resident Inspector - Hatch

9208100013 920804
PDR ADOCK 05000366
S PDR

JE22 1/1

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)

PLANT HATCH, UNIT 2

DOCKET NUMBER (2)

05000366

PAGE (3)

1 OF 6

TITLE (4)

CONDENSATION IN INSTRUMENT SENSING LINES RESULTS IN NONCOMPLIANCE WITH TECHNICAL SPECIFICATIONS

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQ NUM	REV	MONTH	DAY	YEAR	FACILITY NAMES	DOCKET NUMBER(S)
05	15	92	92	010	00	08	04	92		05000
THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR (11)										
OPERATING MODE (9)			20.402(b)			20.405(c)			50.73(a)(2)(iv)	
POWER LEVEL			20.405(a)(1)(i)			50.36(c)(1)			50.73(a)(2)(v)	
100			20.405(a)(1)(ii)			50.36(c)(2)			50.73(a)(2)(vii)	
			20.405(a)(1)(iii)			X 50.73(a)(2)(i)			50.73(a)(2)(viii)(A)	
			20.405(a)(1)(iv)			50.73(a)(2)(ii)			50.73(a)(2)(viii)(B)	
			20.405(a)(1)(v)			50.73(a)(2)(i)			50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
STEVEN B. TIPPS, MANAGER NUCLEAR SAFETY AND COMPLIANCE, HATCH	912 367-7851

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

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NRC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORT TO NRC

SUPPLEMENTAL REPORT EXPECTED (14)

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

ABSTRACT (16)

On 5/15/92, at 0115 CDT, Unit 2 was in the Run mode at 2436 CMWT (100 percent rated thermal power). At that time, licensed operators noted that the drywell pressure as indicated on the recorders 2T48-R607A and B was 0.45 psig when the high drywell pressure annunciator alarmed. The annunciator was expected to have alarmed at 0.65 psig. Investigations into the conditions revealed the actual drywell pressure was approximately 0.63 psig and that the instrumentation (including recorders 2T48-R607A and B) served by 3 of 6 drywell pressure instrument sensing lines were reading low by as much as 0.18 psig. The affected instrument lines were inspected and were found to have some amount of improper slope. It was theorized that water had condensed in the instrument lines and had accumulated in the low points produced by the improperly sloped portions of the lines. An accumulation of water would then cause the instruments served by the affected lines to read low. The lines were purged using nitrogen and subsequent readings showed that the pressure differences were corrected. The condition would have caused the instruments to trip at a drywell pressure greater than that allowed by the Technical Specifications. The cause of the event was improper installation of instrument lines. Specifically, the lines should have been sloped upward from the drywell to the instruments. However, in three cases, tubing was improperly sloped creating low points in the line in which water accumulated. Corrective actions include monitoring of the instrument readings until the lines can be rerouted, routing the lines properly during the next refueling outage, and evaluating the Unit 1 drywell pressure instrument line installations for similar conditions.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5)			PAGE (3)		
		YEAR	SEQ NUM	REV			
PLANT HATCH, UNIT 2	05000366	92	01	00	2	OF	6

TEXT

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor
Energy Industry Identification System codes are identified in the text as (EIS Code XX).

DESCRIPTION OF EVENT

On 5/15/92, at 0115 CDT, Unit 2 was in the Run mode at 2436 CMWT (100 percent rated thermal power). At that time, licensed operators noted that the drywell pressure as indicated on the narrow range drywell pressure recorders 2T48-R607A and B was approximately 0.45 psig when the high drywell pressure annunciator alarmed. The annunciator was expected to have alarmed at approximately 0.65 psig. An investigation showed the actual drywell pressure was approximately 0.63 psig. Consequently, at that time, it was surmised that water had accumulated in the instrument line serving drywell pressure switch 2C71-N004, the non-technical specification instrument providing input to the annunciator. As a result, plans were then made to purge the instrument line with nitrogen.

During development of the procedure for the purging operation, it was determined that further investigation of the condition was warranted to confirm that the instrument line serving pressure switch 2C71-N004 was in fact the source of the problem. On 5/20/92, the instrument line was inspected and found to be sloped continuously downward from the pressure switch to the drywell, negating the theory that water had accumulated in the line. Further investigation was initiated and, on 5/21/92, pressure readings were taken on the six instrument lines sensing drywell pressure using a calibrated gage. The readings showed that the pressure was low on 3 of the 6 instrument sensing lines. The maximum difference between the readings was 0.18 psig. Two of the instrument lines with low pressure readings served drywell pressure recorders 2T48-R607A and B, 2T48-R601A and B, 2T48-R608 and 2T48-R609, and some nonsafety related instrumentation. Drywell pressure recorders 2T48-R607A and B, which were assumed to be reading correctly on 5/15/92, were, in retrospect, reading low and the high drywell pressure annunciator was functioning correctly. The other affected instrument line served drywell pressure transmitters 2C71-N050B and 2E11-N094C.

The three instrument lines that had low pressure were inspected and it was noted each line had some amount of improper slope. It was theorized water had potentially condensed in the instrument lines and accumulated in the low points produced by the improperly sloped portions of the lines in effect creating a loop seal type phenomenon.

Procedure 57SP-052692-JC-1-2S, "Drywell Pressure Instrument Purge," was developed and approved in accordance with administrative control procedures for purging the lines. If in fact water had accumulated in the lines, the purging would void the lines of the water. The procedure was performed on 5/29/92 on the three affected instrument lines. Subsequent readings indicated that the pressure differences had been corrected.

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5)			PAGE (3)		
		YEAR	SEQ NUM	REV			
PLANT HATCH, UNIT 2	05000366	92	010	00	3	OF	6

TEXT

This event was initially determined not to represent a reportable condition because of the insignificant affect to plant safety and because the affected instrumentation was determined to be operable. However, during subsequent review of the event, it was determined on 7/7/92 that a reportable condition existed since the intent of the Technical Specifications setpoint was compromised. Specifically, Unit 2 Technical Specifications Tables 2.2.1-1 and 3.3.2-2 require that the affected instruments be set to trip at less than or equal to 1.92 psig. The instruments are actually set to trip at 1.85 psig. However, in this event, the intent of the Technical Specifications requirements was not met in that water accumulation in the associated instrument sensing lines would skew the pressure being sensed by as much as 0.18 psig, in effect, causing the instruments to trip at a drywell pressure higher than the Technical Specifications required setpoint. The potential error of 0.18 psig could have caused the analytical limit to have been exceeded by that amount during a design basis event if all the drifts and inaccuracies in the setpoint calculation had been at their limit.

CAUSE OF EVENT

The cause of the event is incorrect installation of instrument sensing lines. Typically, when the sensing medium is a gas, the instrument lines are routed with an upward slope from the instrument line tap to the pressure measurement to preclude trapping condensation in the line, which could affect the pressure measurement. However, during installation of the affected lines, a short section of instrument line (approximately eight inches in the drywell) in three installations was routed with an improper slope. Consequently, over a period of time, vapor in the sensing medium condensed in the line and accumulated at the low points produced by the improper routing. As pressure in the drywell increases, the water would be forced out of the low point towards the pressure transmitter. The pressure created by the weight of the water would then be acting against the drywell pressure causing the instrument to sense a pressure that was lower than actual drywell pressure.

REPORTABILITY ASSESSMENT AND SAFETY ANALYSIS

This event is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B) because a condition existed which could result in Reactor Protection System (RPS, EIS Code JC) and Engineered Safety Feature instrumentation tripping at an actual drywell pressure above that assumed by the Technical Specifications. Unit 2 Technical Specifications Tables 2.2.1-1 and 3.3.2-2 require that the affected instruments be set to trip at less than or equal to 1.92 psig. The instruments are actually set to trip at 1.85 psig. In this event, the intent of the Technical Specifications requirements was not met in that water accumulation in the associated instrument sensing lines would skew the pressure being sensed by as much as 0.18 psig, in effect, causing the instruments to trip at a drywell pressure higher than the Technical Specifications setpoint. The potential error of 0.18 psig could have caused the analytical limit to have been exceeded by that amount during a design basis event if all the drifts and inaccuracies in the setpoint calculation had been at their limit. Due to uncertainty associated with the magnitude of the affect of the water accumulation on the instrument setpoint, the insignificant affects to nuclear safety, and the determination

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5)			PAGE (3)	
		YEAR	SEQ NUM	REV		
PLANT HATCH, UNIT 2	05000366	92	010	00	4	OF 6

TEXT

that the affected instruments remained operable, this event was initially determined not to be reportable. However, during subsequent review of the event, it was determined on 7/7/92 that a reportable condition existed since the intent of the Technical Specifications setpoint requirements was compromised.

In this event, condensation had accumulated in three drywell pressure instrument sensing lines affecting the pressure measurement by as much as 0.18 psig. Two of the affected instrument sensing lines serve drywell pressure recorders 2T48-R607A and E, 2T48-R601A and B, 2T48-R608, and 2T48-R609, and several nonsafety related instruments. These drywell pressure recorders provide an indication and recording function only, that is they do not provide any trip function. In this application, the amount of error introduced by the water accumulation (0.18 psig) was inconsequential.

The third instrument sensing line serves drywell pressure transmitters 2C71-N050B and 2E11-N094C. Drywell pressure transmitter 2C71-N050B inputs to the RPS, the Primary Containment Isolation System (PCIS, EIIS Code JURE), the Standby Gas Treatment System (SGTS, EIIS Code BH), and the Secondary Containment Isolation System (EIIS Code NG). The transmitter and companion trip unit function to sense drywell (i.e., Primary Containment) pressure and provide a trip signal to one of the four channels of the associated trip systems, which are of the one-out-of-two-taken-twice logic scheme, when the drywell pressure exceeds 1.85 psig. In this type of logic scheme, a trip signal in various combinations of two of the four channels comprising the actuation system is required to effect an actuation.

Drywell pressure transmitter 2E11-N094C provides an input to the Automatic Depressurization System (ADS, EIIS Code SB), the High Pressure Coolant Injection System (HPCI, EIIS Code BJ), the Low Pressure Coolant Injection System (LPCI, EIIS Code BO), the Emergency Diesel Generator System (EDG, EIIS Code EK) and the Core Spray System (CS, EIIS Code BM). The transmitter and companion trip unit function to sense drywell pressure and provide a trip signal in one channel of the associated trip systems when the drywell pressure exceeds 1.7 psig. Each of the trip systems with the exception of ADS and a portion of PCIS are of the one-out-of-two-taken-twice logic design. As before, in this type of logic scheme, a trip signal in various combinations of two of the four channels comprising the trip system is required to effect an actuation. In the ADS logic scheme, a trip in any one of four channels will initiate a high drywell pressure permissive to ADS. In the affected PCIS logic scheme, a trip in either of two channels will result in the initiation of a safety function.

Based on the following information, it is concluded that this condition was of no safety significance. The Technical Specification setpoint for these instruments is 1.92 psig. The analytical limit for drywell pressure is 2.0 psig. Even though the effective instrument setpoint could potentially have been greater than the analytical limit given the water accumulation phenomenon, the resulting increase in instrument response time is of no consequence based on the SAFER/GESTER-LOCA Loss-of-Coolant Accident Analysis. Typically, for BWRs, the Primary Containment pressure response is not explicitly modeled when analyzing for peak clad temperature. For the Design Basis Accident (DBA) Large Break

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (5)			PAGE (3)	
		YEAR	SEQ NUM	REV		
PLANT HATCH, UNIT 2	05000366	92	010	00	5	OF 6

TEXT

LOCA, the accident analysis is insensitive to variations in the drywell pressure analytical limit because the increase in drywell pressure associated with the large break is extremely fast. Consequently, the peak pressure is assumed in the analysis to occur immediately following the break. 2.0 psig was assigned as the analytical limit because it was determined to be the lowest pressure possible that would not result in spurious trips during normal operation. Because the pressure spike is practically instantaneous, the instrument setpoint exceeding the analytical limit by approximately 0.18 psig would amount to a difference in instrument response time of milliseconds for the DBA LOCA, which would be inconsequential. For the small and intermediate break LOCA, drywell pressure and reactor low water level are initiating signals in the analysis of the Emergency Core Cooling Systems (ECCS) response to such an accident. In this event, the reactor low water level measurement was not affected and the system response would remain as presented in the FSAR. Consequently, the increase in the effective drywell setpoint did not adversely affect nuclear safety.

Additionally, as noted previously, these instruments comprise only one channel for each of the affected multiple channel trip systems. These trip systems are designed such that a single failure of one channel would not prevent the initiation of a safety function. Consequently, even total failure of the affected instruments would not have prevented fulfillment of any safety function.

Based on the above analysis, it is concluded that this event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

The affected lines were purged of the water accumulation via performance of procedure 57SP-052692-JC-1-2S on 5/29/92.

Periodic comparisons of the Unit 2 drywell pressure instrumentation readings were made between 5/4/92 and 7/9/92 to determine if the condition had recurred.

Operating Order 00-03-0792S was issued on 7/9/92 requiring a weekly comparison of Unit 2 drywell pressure instrumentation readings to identify any recurrence of the water accumulation condition.

During the upcoming refueling outage on Unit 2 scheduled to begin 9/16/92, the problem with the improperly sloped lines will be corrected.

The Unit 1 instrument lines sensing drywell pressure are currently being evaluated for similar conditions. If conditions exist on Unit 1 that effect compliance with Technical Specifications setpoint requirements, a revision to this report will be submitted.

**LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION**

FACILITY NAME (1) PLANT HATCH, UNIT 2	DOCKET NUMBER (2) 05000366	LER NUMBER (5)			PAGE (3)		
		YEAR	SEQ NUM	REV			
		92	010	00	6	OF	6

TEXT

ADDITIONAL INFORMATION

No systems other than those previously mentioned in the report were affected by this event.

A similar event occurring in the past two years in which an improperly routed instrument line resulted in a reportable event was reported in LER 50-321/91-27, dated 12/4/91. In this event, an improperly routed flow transmitter instrument line in the Unit 1 Reactor Water Cleanup System (RWCU, EIS Code CE) resulted in a spurious automatic actuation of some Group 5 PCIS valves. The corrective actions for this event were to evaluate the line to determine if rerouting it was the optimal action to take and to inspect the Unit 2 counterpart for similar problems. These corrective actions could not have prevented the event addressed in this report since they only involved the RWCU System.

Failed Component Information: No failed components contributed to this event.