NRC Form 366
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO 3150-0104
EXPIRES 8/31/88

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
RIVER BEND STATION	0 5 0 0 0 4 5 8	1 OF 0 6

TITLE (4)
Reactor Vessel Level Instrumentation

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 NAMES	DOCKET NUMBER(S)
										0 5 0 0 0
0 3	2 4	8 9	8 9	0 1 3	0 0	0 4	2 0	8 9		0 5 0 0 0

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

OPERATING MODE (9):
POWER LEVEL (10):

20.402(b)	20.405(c)	50.73(a)(2)(iv)	73.71(b)
20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER (Specify in Abstract below and in Text, NRC Form 366A)
20.405(a)(1)(iii)	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)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER
L. A. England, Director-Nuclear Licensing	AREA CODE 5 0 4 3 8 1 - 4 1 4 5

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)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE)   ☒ NO

EXPECTED SUBMISSION DATE (15): MONTH | DAY | YEAR

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

On 3/24/89 with the unit in Operational Condition 5 (refueling), an investigation into the installation tolerance for the RPV water level reference leg condensing chambers (1B21*TKD004A-D) determined that a number of effects, which could result in non-conservative indicated water level, were not considered in the calibration basis for the reactor water level instrumentation. These effects could result in violation of the allowable values for reactor pressure vessel level 1, 2, 3 and 8 trip points given in Sections 2.2, 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.5, and 3.3.9 of the Technical Specifications. This report is being submitted pursuant to 10CFR50.73(a)(2)(i)(B).

An investigation of possible water level measurement errors due to factors associated with arrangement and temperature of the reference and variable leg instrument piping, determined that the narrow range instrumentation could have a conservative error of about 1.4 inches, while the wide range instrumentation could have a non-conservative error of 0.4 inch at the Level 1 trip setpoint. A safety evaluation demonstrates that there is an insignificant effect on the results of accident analyses from a 1.3 inch error in the Level 1 initiation of protective functions.

There was no impact on the safe operation of the plant or to the health and safety to the public as a result of this condition.

IE22 1/1

REPORTED CONDITION

On 3/24/89 with the unit in Operational Condition 5 (refueling), an investigation into the installation tolerance for the reactor pressure vessel (RPV) water level reference leg condensing chambers (1B21*TKD004A-D), which are part of the Reactor Protection System (*JE*) determined that a number of effects which could result in non-conservative indicated water level, were not considered in the calibration basis for the reactor water level instrumentation. These effects could result in violation of the allowable values for level 1, 2, 3 and 8 trip points given in Sections 2.2, 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.5, and 3.3.9 of the Technical Specifications. This report is being submitted pursuant to 10CFR50.73(a)(i)(B).

No immediate action was necessary since the plant was, and is currently shutdown for a refueling outage. The water level effects addressed in this LER do not apply at cold shutdown conditions.

INVESTIGATION

It has been determined that the methods and assumptions for calculating the present calibration data for wide range and narrow range RPV water level transmitters did not account for a number of factors which could result in non-conservative errors in water level indication. The other reactor water instruments (i.e., fuel zone, upset and refueling level) are calibrated for different conditions and have less stringent accuracy requirements since they do not initiate any automatic protective actions. These instruments are not addressed in this LER. No other instrumentation (e.g., reactor pressure, steam flow) is similarly affected by temperature conditions in drywell and containment, or by thermal expansion of the reactor. The initial investigation determined the following:

1. Differential thermal growth between the reactor vessel and the condensing chamber on the reference leg was not considered. This differential movement arises from the fact that the condensating chamber is supported by the drywell structure which does not expand thermally as much as the reactor vessel. It was assumed in the original design that the vessel and the condensing chamber move together as reactor temperature changes, thus, requiring no correction for elevation changes. The result is a non-conservative water level indication (i.e., indicated water level is higher than the actual level).

2. Higher than assumed reference leg temperatures exist inside the drywell. The level instrumentation calibration assumed that the portion of the reference leg located in the drywell was at the nominal maximum average drywell temperature of 135°F. However, thermocouples mounted on each reference leg in the drywell indicate local temperatures of 230°F. This condition causes the instrumentation to indicate a higher level than actually exists.

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

3. The calibration assumed saturated water conditions in the variable and reference leg instrument piping instead of compressed water. Due to the instrument line arrangement at River Bend, this results in a conservative error, (i.e., this condition causes the instrumentation to indicate a lower level than actually exists).

The net effect of the above factors results in an error of +2.0 inches for the narrow range (NR) and up to +1.3 inches for the wide range (WR). The positive value means that the indicated level would be higher than the actual level. A further evaluation of other systematic (i.e., not random) errors associated with the non-electronic portion of the level instrumentation determined the following additional factors:

4. The temperature of the variable legs temperature inside the drywell is assumed to be 135°F. The portion of the wide range and narrow range variable legs inside the reactor vessel insulation, however, are close to reactor water temperature which is 533°F below the feedwater sparger and 549°F above the sparger at rated conditions. The portions of the variable leg which pass through the insulation and through the biological shield are at a temperature considerably above drywell temperatures. This produces a conservative error as shown on Table 1.

5. The portion of variable leg inside the drywell, but outside the biological shields will be at drywell ambient temperature, assumed to be 135°F. A lower drywell temperature would result in a non-conservative error in level indication. The reference leg temperature in the containment is assumed to be 80°F, while a higher actual temperature would result in a non-conservative error. An increase or decrease in the containment temperature, however, has greater effect on level indication than the same increase or decrease in drywell temperature. A review of the historical temperature data for the containment and drywell indicates that worst case occurred in the summer when the average drywell temperature was at 135°F and containment temperature was about 86.5°F. The variation in drywell temperatures suggests that the local temperature in the vicinity of the variable leg could be 5°F lower than the average. The combination of a higher containment temperature and a lower local drywell temperature results in a non-conservative error as shown on Table 1.

6. The WR calibration assumes 20 Btu/lb subcooling below the NR variable leg nozzle. This is done to account for the effect of feedwater which enters the vessel through the feedwater sparger nozzles approximately 20 inches below the NR variable leg nozzle. The actual relationship of level to the sensed differential pressure has a break point (i.e, a change of slope) at the feedwater sparger elevation, due to the change in water density at that point. Since the indicated wide range level indication is a

linear function with respect to sensed differential pressure at the transmitter, its calibration is a compromise that results in a slight non-conservative error in mid-span, given rated conditions in the reactor. However, this small deviation is more than offset by the large conservative bias due to flow velocity in the reactor annulus at rated conditions (see item 7 below).

Given the likelihood of saturated water conditions in the vessel during a LOCA or loss of feedwater, the assumption of 20 Btu/lb subcooling below the NR nozzle results in a conservative error in level indication as shown on Table 1.

7. The flow velocity in reactor vessel annulus past the wide range variable leg nozzle, produces a conservative bias in wide range level indication of about 13-15 inches at rated core flows. This makes the wide range level instrumentation indicate low relative to the narrow range during normal opreation.

The net effect of all factors discussed above (except item 7) is shown on Table 1.

CORRECTIVE ACTION

Analytical limits will be adjusted downward by June 30, 1989 to account for the additional potential effects on level indication that have been identified. The existing allowable values and nominal trip setpoints as shown in the Technical Specifications will remain the same.

A review of previously submitted LERs from River Bend Station for instrument setpoint errors not previously identified revealed two instances where instrument sensing errors or biases were not included in the original setpoint calculations. The first was LER 86-067 where a design configuration resulted in a water column forming in a sensing line for the RCIC system during operations. The configuration was changed to eliminate the formation of the water column. Because the previous root cause involved water condensation rather than a thermal growth error and was corrected, the previous identified corrective action would not have identified or resolved this item. The second LER was 87-009 in which a pressure correction factor for the reactor vessel safety relief valve functions was omitted. The root cause was determined to be personnel oversight of an identified correction; because the corrective action was incorporation of a known value, the previous corrective action would not have discovered the present condition.

SAFETY ASSESSMENT

GE has analyzed the effect of lowering the analytical limit for Level 3 by 2.0 inches, and the Levels 1 and 2 by 1.3 inches. For transient

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

events, the limiting water-level-decreasing event is the loss of all feedwater flow. The combined effect of the lower analytical limits for Level 1, 2 and 3 trip points would still result in adequate margin for the Level 1 trip avoidance.

For LOCA events, the Level 3 analytical limit is used as the initial water level. The lower analytical limit will result in a slight change in the reactor water level response. Since the decrease in Level 1 and Level 2 analytical limits is less than the decrease of the Level 3 setpoint, the initiation of the ECCS (HPCS, LPCI, LPCS and ADS) will be slightly earlier than the previous Safety Analysis Report (SAR) calculation. The combined effect of the lower anlaytical limits for the most limiting LOCA event (a double-ended break of the recirculation piping) is such that the impact on calculated peak clad temperature is expected to be less than 5°F. It should be noted that high flow velocities present in the reactor vessel annulus during this event would cause a large conservative shift in wide range water level indication, resulting in Level 1 and 2 trips being initiated considerably above the revised analytical values. The USAR shows that River Bend currently has a 56°F margin to the limit of 2200°F in peak clad temperatures. These values were based on older,more conservative calculational methods. Using newer methods (SAFR/GESTR) typically results in a margin of more than 200°F.

The level 8 trip point is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. The identified errors would not affect the reactivity insertion limitation provided by the level 8 trip setpoint since the same amount of cold feedwater would be required to increase from the normal level to the level 8 trip point. In other words, the relative change in reactor water level, and thus the amount of cold feedwater added, is the same as was assumed in the SAR.

Since this condition has been found to maintain the present plant safety analysis there was no adverse impact on the safe operation of the plant or to the health and safety of the public as a result of this event.

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

*U.S.GPO 1986-0-624-538/455

TABLE 1

RESULTS OF REACTOR LEVEL ERROR ANALYSIS

ITEM	NARROW RANGE	WIDE RANGE (3) @ TOP	@ BOTTOM	@ LEVEL 1
1. Hot vs. cold dimensions	+ 2.08	+ 1.83	+ 1.79	+ 1.80
2. Ref. leg in DW @ 230°F	+ 0.68	+ 0.68	+ 0.68	+ 0.68
3. Use of compressed water tables vs. saturated	- 0.74	- 1.20	- 1.36	- 1.35
WR calibration approx. (2)	0.00	- 0.36	0.00	+ 0.14
Subtotal of errors	+ 2.02	+ 0.95	+ 1.11	+ 1.27
4. Variable leg temperature inside bioshield	- 1.20	- 1.17	- 1.14	- 1.14
5. Containment and drywell ambient temp. 5°F higher	+ 0.55	+ 0.74	+ 0.72	+ 0.72
Subtotal of errors	+ 1.37	+ 0.52	+ 0.69	+ 0.85
6. Saturated water below NR Variable leg nozzle	0.00	- 3.39	- 0.07	- 0.50
SUM OF ERRORS	+ 1.37	- 2.87	+ 0.62	+ 0.35

NOTES:

1. A "+" sign indicates that error results in indicated level being higher than actual level while a "-" sign indicates the opposite condition.

2. See discussion in text under Item 6.

3. Wide range level instrumentation measures from (-)160 inches, bottom, to (+)60 inches, top, with Level 1 at (-)143 inches.

GULF STATES UTILITIES COMPANY

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April 20, 1989
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File Nos. G9.5, G9.25.1.3

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

Gentlemen:

River Bend Station - Unit 1
Docket No. 50-458

Please find enclosed Licensee Event Report No. 89-013 for River Bend Station - Unit 1. This report is being submitted pursuant to 10CFR50.73.

Sincerely,

JJ Dietrich

for J. E. Booker
Manager-River Bend Oversight
River Bend Nuclear Group

JEB/TFP/WJB/BMB/ch

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