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**Rick J. King**  
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July 30, 2003

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

Subject: River Bend Station  
Docket No. 50-458  
License No. NPF-47  
Licensee Event Report 50-458 / 03-005-01

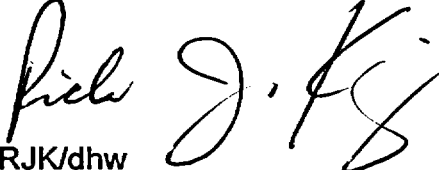
File Nos. G9.5, G9.25.1.3

RBG-46151  
RBF1-03-0137

Ladies and Gentlemen:

In accordance with 10CFR50.73, enclosed is the subject Licensee Event Report. This is a supplement to LER 50-458 / 03-005-00, submitted on June 9, 2003. Revisions are indicated by change bars in the right margin. Commitments are summarized on the Commitment Identification Form.

Sincerely,

  
RJK/dhw  
enclosure

JE22

Licensee Event Report 50-458 / 03-005-01  
July 30, 2003  
RBG-46151  
Page 2 of 3

cc: U. S. Nuclear Regulatory Commission  
Region IV  
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Baton Rouge, LA 70821-4312

Commitment Identification Form

COMMITMENT	ONE-TIME ACTION*	CONTINUING COMPLIANCE*
Computer points will be developed for the best statistical estimate of core thermal power and the percent difference between that estimate and actual core thermal power. Limits for this variable will be developed for use in prescribing operator actions.	X	

\*Check one only

## 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 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to [bs1@nrc.gov](mailto:bs1@nrc.gov), and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME River Bend Station Unit 1						2. DOCKET NUMBER 05000 458				3. PAGE 1 OF 6	
4. TITLE Operation Greater Than Maximum Licensed Power Due to Erroneous Feedwater Flow Measurement											
5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED		
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER	
05	10	2003	2003	- 005 -	01	07	30	2003		05000	
9. OPERATING MODE			1						FACILITY NAME		
10. POWER LEVEL			098						DOCKET NUMBER		
									05000		
11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)											
			20.2201(b)			20.2203(a)(3)(ii)			50.73(a)(2)(ii)(B)		
			20.2201(d)			20.2203(a)(4)			50.73(a)(2)(iii)		
			20.2203(a)(1)			50.36(c)(1)(i)(A)			50.73(a)(2)(iv)(A)		
			20.2203(a)(2)(i)			50.36(c)(1)(ii)(A)			50.73(a)(2)(v)(A)		
			20.2203(a)(2)(ii)			50.36(c)(2)			50.73(a)(2)(v)(B) X		
			20.2203(a)(2)(iii)			50.46(a)(3)(ii)			50.73(a)(2)(v)(C)		
			20.2203(a)(2)(iv)			50.73(a)(2)(i)(A)			50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)			50.73(a)(2)(i)(B)			50.73(a)(2)(vii)		
			20.2203(a)(2)(vi)			50.73(a)(2)(i)(C)			50.73(a)(2)(viii)(A)		
			20.2203(a)(3)(i)			50.73(a)(2)(ii)(A)			50.73(a)(2)(viii)(B)		
12. LICENSEE CONTACT FOR THIS LER											
NAME J.W. Leavines, Manager - Licensing						TELEPHONE NUMBER (Include Area Code) 225-381-4642					
13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT											
CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX		
B	SJ	PE	CO23	YES							
14. SUPPLEMENTAL REPORT EXPECTED											
YES (If yes, complete EXPECTED SUBMISSION DATE).					X NO		15. EXPECTED SUBMISSION DATE		MONTH	DAY	YEAR

## 16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On May 10, 2003, with the plant operating at 98 percent power, information was received from the manufacturer of the station's reactor feedwater flow measurement device indicating the station had been operated in excess of maximum licensed reactor thermal power in the past. This condition is being reported in accordance with the Facility Operating License (NPF-47) as a violation of License Condition 2.C(1), "Maximum Licensed Power." The investigation determined that changes to plant parameters outside the design assumptions of the instrument resulted in erroneous feed flow indication. This component is no longer installed in the plant. An evaluation has determined that adequate design margin existed to accommodate the overpower condition.

## LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
River Bend Station Unit 1	05000-458	2003	- 005	- 01	2 OF 6

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

## REPORTED CONDITION

On May 10, 2003, with the plant operating at 98 percent power, information was received from the manufacturer of the station's reactor feedwater flow measurement device (\*\*PE\*\*) indicating the station had been operated in excess of maximum licensed reactor thermal power. This condition is being reported in accordance with the Facility Operating License (NPF-47) as a violation of License Condition 2.C(1), "Maximum Licensed Power."

## INVESTIGATION AND CORRECTIVE ACTIONS

Reactor thermal power is calculated to assure compliance with licensed power limits and safety analysis assumptions. At the time this condition existed, River Bend was licensed at 3039 megawatts thermal (MWth), and was analyzed at 102 percent of licensed thermal power. The primary factor affecting this calculation is feedwater flow. From initial plant operation to 1996, reactor feedwater flow measurement was performed using a calibrated venturi in each of two feedwater headers supplying the reactor.

A new feedwater flow measurement device, the Leading Edge Flow Meter (LEFM) 8300, manufactured by Caldon, Inc., was installed in February 1996. This device was to be used to correct feedwater venturi flow readings in the plant process computer for the purpose of eliminating reactor heat balance conservatism caused by allowances built in to account for venturi fouling. This correction would result in maximizing plant efficiency and electrical output. The first correction factors were used on February 27, 1996, and core thermal power was adjusted accordingly, resulting in an increased output of approximately 1.5 percent of rated thermal power.

During plant operation, the correction factors were checked monthly to compare feedwater venturi flow indication and LEFM flow indication. New correction factors were calculated and installed if the difference between venturi flow and LEFM flow reached a predetermined limit or each time the plant was shut down and restarted.

During a refueling outage in March 2003, a more accurate LEFM CheckPlus™ unit was installed to replace the existing external LEFM 8300. Due to the increased accuracy of the flow input to the reactor heat balance provided by the LEFM CheckPlus™, an increase in core thermal power of an additional 1.7 percent was expected. During the subsequent startup and power ascent, the indicated feed flow data from the LEFM CheckPlus™ was reviewed against feedwater venturi data. A comparison of these flows with the external LEFM 8300 data indicates that the station has, during certain periods, been operated in excess of its licensed power limit by a maximum of 2.7 percent.

## LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
River Bend Station Unit 1	05000-458	2003	- 005	- 01	3 OF 6

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

A detailed analysis was performed to determine the magnitude of the overpower condition. From the installation of the external LEFM 8300 in February 1996, until the high pressure (HP) main turbine rotor replacement in 1999, there is no indication that the 102 percent accident analysis limit was exceeded. To obtain the best estimate of reactor thermal power, main turbine first stage pressure was normalized for the period being evaluated. This estimate for the period of time from the replacement of the HP turbine rotor in 1999 to 2003 indicates the following:

- Between the HP turbine rotor replacement in 1999 and the 5 percent power uprate in October 2000, the licensed power limit was exceeded. However, there is no indication that the 102 percent accident analysis limit was exceeded during this time.
- The period from October 2000, (when new correction factors were installed following the 5 percent uprate) until April 2001, (when new correction factors were installed following a forced outage) represents the bounding case for the overpower condition. During this time, reactor power level was as apparently as high as 102.7 percent.
- From April 2001, when correction factors were changed due to a forced outage, to refueling outage number 10 (RF10) in October 2001, reactor power level was as high as 102.5 percent.
- Following RF10, the 102 percent analyzed limit was not exceeded again until January 2002, when a scheduled derate for control rod sequence exchange and turbine valve testing was performed. This condition lasted until May 2002, when correction factors were revised due to a planned outage, at which time power level decreased to less than 102 percent, but remained above the licensed power limit until January 2003, with the coastdown leading into RF11.

Based on this analysis, the reactor was operated in excess of the 102 percent accident analysis limit for approximately 15 months.

The LEFM 8300 is no longer installed in the plant. Since the installation of the more accurate LEFM CheckPlus™ device, the station has operated within the licensed power limit.

## CAUSAL ANALYSIS

An analysis of data for the operating history in question was performed. The period immediately following the 5 percent power uprate in October 2000 was determined to be the bounding case for the overpower condition. The specified uncertainty factor for the

## LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
River Bend Station Unit 1	05000-458	2003	- 005	- 01	4 OF 6

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

LEFM 8300 is plus or minus 1.1 percent of rated system flow. The analysis determined that the actual uncertainty factor for the least conservative case was plus or minus 2.69 percent.

The investigation determined that three contributory factors caused the erroneous feed flow indication by the LEFM 8300.

1. Certain assumptions about the flow velocity through the instrumented pipe section are made in the manufacturer's design. Variations in pipe wall smoothness following installation caused changes in velocity profiles beyond the LEFM's calibration basis.
2. The change in feedwater temperature related to the 5 percent power uprate in October 2000 caused changes in the baseline acoustic path such that its position moved outside the design range.
3. A small error was made in the measurement of the outside pipe diameter in the "B" feedwater header at the time of the installation of the instrument. This adversely affected the assumed cross-sectional flow area of the instrument.

It should be noted that, while the biases due to factors 2 and 3 above contributed to the overall error of the LEFM 8300, they would not, in and of themselves, have caused the instrument to operate outside its design basis. The dominant source of error was factor 1. Although the instrument manufacturer included an allowance in this uncertainty analysis for profile changes, the actual changes in pipe wall characteristics led to errors greater than the allowance by more than a factor of two.

Following the installation of the LEFM 8300, modifications were made to the plant that changed power-dependent parameters which inhibited earlier detection of the overpower condition:

1. the high pressure turbine rotor replacement in July 1999,
2. the steam flowpath adjustment in the HP turbine in April 2000,
3. the flow-only 5 percent power uprate in October 2000, and,
4. the reactor pressure uprate in October 2001.

## LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
River Bend Station Unit 1	05000-458	2003	- 005	- 01	5 OF 6

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

During the development of these modifications, the potential effect on LEFM performance was not adequately evaluated. Each time a large parameter change occurred due to an equipment or plant upgrade, the relatively small change made to the same parameter due to LEFM correction factor changes was difficult to detect. When slight changes were noticed, the LEFM vendor certified the external LEFM 8300 flow reading to be within specified accuracy further validating the belief that the revised parameter was correct.

## CORRECTIVE ACTION TO PREVENT RECURRENCE

The LEFM 8300 was removed from the feedwater system in April 2003. The LEFM CheckPlus™ has been verified to be performing within specifications.

Computer points will be developed for the best statistical estimate of core thermal power and the percent difference between that estimate and actual core thermal power. Limits for this variable will be developed for use in prescribing operator actions.

## SAFETY SIGNIFICANCE

An analysis of this condition was performed to assess the postulated effects on fuel cladding integrity, reactor vessel integrity, primary containment integrity, and post-accident activity releases.

Fuel cladding integrity is protected by operating to a number of parameters collectively referred to as thermal limits. For River Bend, these thermal limits are average planar heat generation rate, minimum critical power ratio, and linear heat generation rate. The two percent power measurement uncertainty per 10CFR50.46 and Regulatory Guide 1.49 are already imbedded within these thermal limits so that the incremental increase over this limit of 0.7 percent of licensed power level needs to be addressed. Each thermal limit is linearly dependent upon core thermal power. As such, with the power measurement bias in place, the measure of the margin to a particular thermal limit would have been 0.7 percent less than calculated by the core monitoring system. Operating logs were reviewed for the period of operation of most concern (i.e., from the implementation of the five percent power uprate until RF11) in order to assess the impact on fuel cladding integrity. Based upon this review, it was concluded that margin to the operating limits was sufficient to accommodate the 0.7 percent overpower.

Vessel integrity is evaluated every cycle by analyzing the MSIV closure event with failure to scram on MSIV position. This event was analyzed at 102 percent thermal power, thus, the impact of an additional 0.7 percent in initial thermal power was evaluated. Operating at the slightly higher power limit would not significantly change the peak pressure attained as operation of the main steam safety / relief valves following the reactor scram is the primary means of limiting the pressure rise in the vessel. Further,



## LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET	6. LER NUMBER			3. PAGE
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
River Bend Station Unit 1	05000-458	2003	- 005	- 01	6 OF 6

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

the Cycle 10 and 11 reload analyses were reviewed to identify the margin to the acceptance criteria for the postulated overpressure event. There is margin in these analyses sufficient to absorb the 0.7 percent overpower condition. It is concluded that reactor vessel integrity was not challenged during the period in which the external LEFM 8300 correction factors were in-service.

The current containment analysis was performed for the 5 percent power uprate project. The 5 percent power uprate project resulted in an increase in licensed reactor power from 2894 to 3039 MWth and an increase in reactor pressure of 30 PSI. As required, the analyses were performed assuming an initial reactor power of 102 percent, thus the impact of an additional 0.7 percent power was assessed. As part of the power uprate containment analysis, evaluations at 102 percent of the pre- and post-power uprate initial conditions were performed. These evaluations indicate a small sensitivity in peak containment parameters relative to the initial conditions assumed. The effect of the 0.7 percent overpower would much smaller than the effect of the 5 percent power uprate as the latter included an increase in reactor dome pressure. Therefore, based upon the small sensitivity, and the existing margin to containment design limits, it is concluded that containment integrity would not have been challenged during the period in question.

The post-accident radiological dose consequence evaluation is performed to assess dose consequences to individuals at the exclusion area boundary, the low population zone, and the main control room. River Bend has recently completed upgrading these evaluations to include the Alternative Source Term per 10CFR50.67 and Regulatory Guide 1.183. The associated analyses were based upon a reactor operating at 3100 megawatts thermal, which is 102 percent of the previous licensed thermal power of 3039 megawatts thermal. Therefore, the impact of an additional 0.7 percent power was evaluated. In reviewing the subject calculations, it is concluded that the source term, and therefore the dose consequences, are linear with respect to reactor power. As such, the dose consequences would be expected to increase 0.7 percent. A review of the calculations indicates that there is more than enough margin in the 10CFR50.67 acceptance criteria to accommodate the slight overpower.

(NOTE: Energy Industry Component Identification codes are annotated as (\*\*XX\*\*).)