GULF STATES UTILITIES

RIVER BEND STATION

ANNUAL OPERATIONS REPORT

FOR

1985

IE24
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1.0 INTRODUCTION

ANNUAL OPERATING REPORT

1.0 Introduction

This River Bend Station Annual Operating Report for 1985 is submitted in accordance with applicable reporting requirements of Technical Specifications 6.9.1.4 and 6.9.1.5, of Appendix A to River Bend Station (RBS) License Number NPF-47. This routine operating report covering operation of the unit during the previous calendar year also complies with applicable sections of USNRC Regulatory Guide 1.16 "Reporting of Operating information - Appendix A Technical Specifications".

2.0 PLANT PERFORMANCE SUMMARY

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I. Summary of Operations

River Bend Station received a low power operating license on August 29, 1985, and began fuel loading shortly thereafter. Initial criticality was achieved on October 31, 1985. Following initial criticality, the plant received a full power license on November 20 and exceeded 5% thermal power on November 25. The main turbine/generator was rolled on November 26 and synchronized to the grid on December 3. The heat up testing phase of the Start-Up Test Program was completed without major complications. Start-up testing in Test Condition 1 continued through the end of 1985. River Bend experienced three reactor scrams during the month of November; one scram was manually initiated per Technical Specifications and was taken credit for as a training scram, and the other two were from automatic action. Three reactor scrams were also experienced in December; one scram was planned as part of the very successful Loss of Offsite Power startup test, and the other two were from automatic actuation. The goal for River Bend was to have reached greater than 200 MWe by year's end. This goal was successfully achieved with the plant operating just greater than 200 MWe for a short period.

II. Duration of Activities

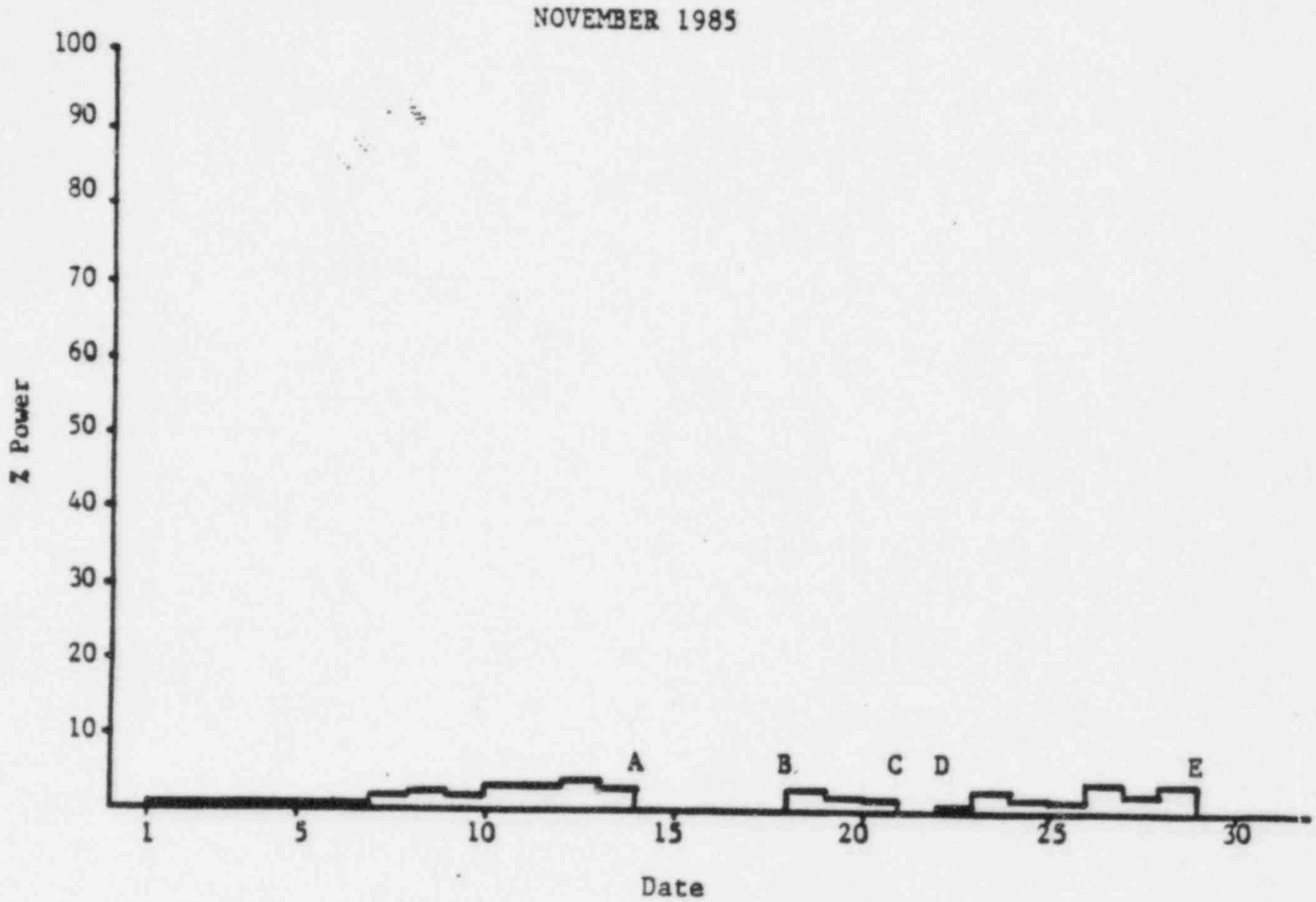
	Date	Duration
Date low power license received	08-29-85	
Fuel loading completed	09-21-85	22 days
Initial criticality	10-31-85	39 days
Full power license received	11-20-85	20 days
Exceeded 5% power	11-25-85	5 days
Synchronized to grid	12-03-85	8 days
200 MWe (20%)	12-28-85	25 days

3.0 OPERATING EVENT SUMMARY

3.0 OPERATING EVENT SUMMARY

River Bend experienced six scrams from August 29, 1985, date the low power license was received, to December 31, 1985. The first forced power reduction was utilized to allow for a training scram, (Reactor Scram 85-01). Reactor Scram 85-04 was planned as part of the Loss of Offsite Power startup test. Four scrams were a result of unplanned automatic actuations.

Enclosed are charts of the Power Profile for River Bend Station since initial criticality was achieved on 10-31-85. A discussion behind the cause for each forced outage is also presented.



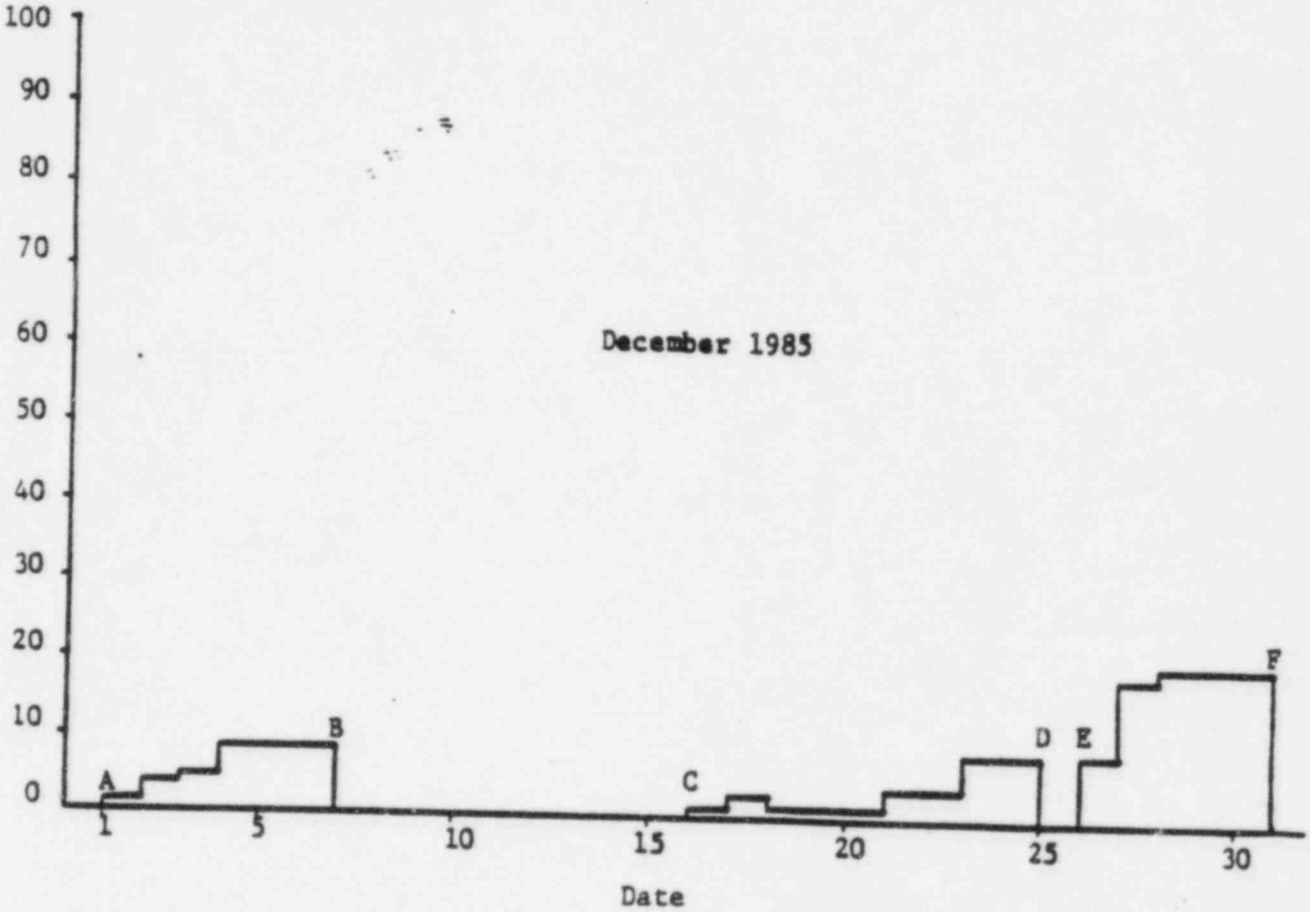
A. Planned training scram on 11-13-85 at 2340 due to Div. I and II Standby Service Water inoperability.

B. Startup; reactor critical on 11-17-85 at 1438.

C. Unplanned scram on 11-21-85 at 0122 due to low level actuation on loss of feedwater.

D. Startup; reactor critical on 11-21-85 at 0943.

E. Unplanned reactor scram on 11-28-85 at 1843 on IRM upscale due to a feedwater transient.



A. Startup; reactor critical on 12-01-85 at 0321.

B. Loss of offsite power test conducted with generator synchronized to the grid and reactor at 12% full power. A 9 day planned outage followed this trip.

C. Startup; reactor critical on 12-16-85 at 0500.

D. Unplanned scram on 12-24-85 at 2123 caused by low reactor water level due to a loss of feedwater transient.

E. Startup; reactor critical on 12-25-85 at 1956.

F. Unplanned scram on 12-31-85 at 1011. Lightning striking a 500KV line tripped the turbine on power to load unbalance causing a reactor scram on high reactor pressure.

3.1 SCRAMS

REACTOR SCRAM 85-01

At 1200 on 11/13/85 with the unit in Operational Condition 2 (Startup) and at approximately 2 percent power, the condition described below was discovered.

A review of the open items on safety related building punch lists revealed that certain Category I structural elements (lateral supports, etc.) had not been installed in piping tunnels G and H (Reference LER 85-033). The affected safety related piping was part of the Standby Service Water (SSW) system, trains A and B.

The final structural load verification program required the addition of supplemental steel members to the existing structural steel in these areas. Engineering had perviously identified the work to be done on Engineering and Design Change Request numbers P-3172A, P-3059D, P-3064 and C-7188 to bring the tunnels into compliance with the design basis.

Immediate corrective action was taken. Trains A and B of SSW were declared inoperable, a power reduction was commenced per Technical Specification 3.0.3. The reactor was held just critical to allow operation personnel to take credit for a training scram should a full shutdown be required. At 0000 hours on 11/14/85 the structural steel support installation had not been completed and the reactor was manually scrammed with operations taking credit for a training scram. (Reference LER 85-034). By 2100 on 11/14/85 the necessary work was complete and SSW System trains A and B were declared operable.

CORRECTIVE ACTION TO PREVENT RECURRENCE

As discussed in LER 85-033, all open items on safety-related building punch lists were reviewed in detail to ensure that all items which impacted the design basis of the unit were identified and appropriate action taken.

There was no single release of radioactivity or single radiation exposure specifically associated with the outage which accounted for more than 10 percent of the allowable annual values.

OPERATING TIME LOSS

1. In terms of generator-off-line hours.
   Time Loss: 0
   There were no generator-off-line hours lost since initial electrical generation occurred after the event.

2. Operating time loss when reactor was "forced" subcritical, all rods in.

   Time Loss: 86 hours 58 minutes

3. The installation of seismic SSW piping support structures was determined to be critical path for the outage.

REACTOR SCRAM 85-02

On 11/21/85 at 0025 with the unit in Operational Condition 2 (Startup), Reactor Feedwater Pump (RFP) B tripped for unknown causes (Reference LER 85-041). The operator restarted the pump and restored reactor water level to normal. In order to investigate the cause, RFP C was started with its discharge Motor Operated Valve (MOV 26C) left shut in the event RFP B tripped again. At 0117 RFP B tripped when its auxiliary oil pump was secured. The operator mistakenly believed that a normal oil supply was available. An attempt to open the discharge MOV for RFP C failed with the breaker tripping on overload. RFP B and its auxiliary oil pump was restarted, but an attempt to open its discharge valve also failed. At 0123 the reactor scrammed when level decreased to the low level scram setpoint. Reactor Core Isolation Cooling (RCIC) was manually started to restore reactor level. The lowest level indicated on narrow range instrumentation was +2 inches (164 inches above TAF). At 0127 the RFP B discharge valve was opened and RCIC secured.

The cause of the initial RFP B trip is believed to have resulted from a main oil pump trip on overload due to low oil temperature. Investigation into the inability to open RFP B and C discharge valves determined that the cause was due to high differential pressure across the discharge valves. This would result when the feedwater to condenser recirculation valve (FV104) is open as it was in this case to aid in reactor level control.

CORRECTIVE ACTION TO PREVENT RECURRENCE

In an effort to prevent recurrence, system operating procedure SOP-0009 "Reactor Plant Feedwater" now has a caution inserted concerning RFP discharge valve operation during low power operations which could result in unusually high differential pressure. The operations staff has also been issued a memorandum reminding personnel of the operation of the RFP oil system. The system is being addressed in the current phase of licensed operator requalification training which began 01/03/86.

There was no single release of radioactivity or single radiation exposure specifically associated with the outage which accounted for more than 10 percent of the allowable annual values.

OPERATING TIME LOSS

1. In terms of generator-off-line hours.
   Time Loss: 0
   There were no generator-off-line hours lost since initial electrical generation occurred after the event.

2. Operating time loss when reactor was "forced" subcritical, all rods in.

   Time Loss: 8 hours 21 minutes

3. There was no major safety related maintenance performed during this outage.

REACTOR SCRAM 85-03

On 11/28/85 at 1845 with the unit in Operational Condition 2 (Startup), the reactor scrammed on Intermediate Range Monitor (IRM) upscale trip (Reference LER 85-047). During a turbine roll the number 3 bearing was showing signs of high vibration forcing the turbine to be manually tripped at 1821 hours on 11/28/85. At this time the number 1 turbine bypass valve was 30 percent open. In anticipation of breaking condenser vacuum reactor power was reduced. At 1833 condenser vacuum was broken to accelerate braking of the turbine. At 1840 the turbine bypass valves were tripped closed on low vacuum and reactor pressure began to increase slowly. As Main Steam Line drains were opened to reduce reactor pressure the level swell caused the feedwater level control valves to shut. Additionally, the increased voids caused power to decrease and the IRMs were down ranged to keep flux levels onscale.

As the pressure reductions slowed and level decreased the feedwater level control valve began to open. With reactor pressure nearly stable and with cold feedwater entering the vessel, reactor power began to increase. The reactor operator, having previously down ranged the IRMs, failed to range up the IRMs in time to prevent a reactor trip.

CORRECTIVE ACTION TO PREVENT RECURRENCE

Since similar transients (i.e. turbine trip, loss of condenser vacuum, level and pressure transients) have occurred without a reactor scram, this is determined to be an isolated event. All operating personnel have been notified of this event and are aware of the conditions which cause reactor power transients. No further corrective action was taken.

There was no single release of radioactivity or single radiation exposure specifically associated with the outage which accounted for more than 10 percent of the allowable annual values.

OPERATING TIME LOSS

1. In terms of generator-off-line hours.
   Time Loss: 0
   There were no generator-off-line hours since initial electrical generation occurred after the event.

2. Operating time loss when reactor was "forced" subcritical, all rods in.
Time Loss: 54 hours 38 minutes

3. Startup was delayed because of blown fuses found on the backup scram valve circuit. Reverse DC polarity at the valve was determined to be root cause of the blown fuse problem. Resolution of this item proved to be critical path for the outage.

REACTOR SCRAM 85-04

On 12/06/85 at 2234, as a result of required startup testing, a preplanned reactor scram occurred due to a turbine trip. The turbine trip at 12 percent of full power with the generator synchronized to the grid was part of the Loss of Offsite Power startup test. Both the Loss of Offsite Power test and scram recovery proceeded without incident.

CORRECTIVE ACTION TO PREVENT RECURRENCE

None required; pre-planned scram to support Start-Up Test Program.

OPERATING TIME LOSS

1. In terms of generator-off-line hours.
   Time Loss: 16 days

2. Operating time loss when reactor was "forced" subcritical, all rods in.
   Time Loss: 9 days

3. This outage was utilized to complete Surveillance Test Procedures required prior to entering Mode 1. Surveillance Test Procedures proved to be critical path for the outage.

   Other Outage Items Worked:

   - Replace Reactor Core Isolation Cooling trip throttle valve because of sticking.

   - Replace five control rod drive position indication probles.

REACTOR SCRAM 85-05

On 12/24/85 at 2123 with the unit at 1 percent rated power in Operational Condition 2 (Startup), the reactor scrammed on vessel low level 3 (+9.7 inches) 172 inches above top of active fuel (Reference LER 85-060). The low level resulted from the trip of the running feedwater pump C. The B feedwater pump was immediately started in an effort to prevent a scram but its discharge valve failed to open. Approximately 5 minutes later with level at +16 inches (178 inches above top of active fuel) the remaining feedwater pump A was started and began injecting water just as the low level 3 scram initiated. Reactor vessel level was restored immediately with the minimum level reached being +9.0 inches (171 inches above top of active fuel).

The initial trip of the C feedwater pump resulted from gear increaser bearing failure due to overheating. Approximately 30 minutes before the scram the C feedwater pump had been started in order to make repairs on the feedwater pump A minimum flow line. Upon investigation it was discovered that there was a lack of sufficient cooling to the lube oil system. Step 4.2.8 of Station Operating Procedures SOP-0009 "Reactor Feedwater System" requires that the Closed Cooling System (CCS) flow to the lube oil coolers be adjusted to maintain oil temperature from the coolers between 80 and 120 degrees F. There is no similar step to be performed after the start of the feedwater pump. Manually operated gate valves are used to control CCS flow through the oil coolers. The subject valve was found throttled nearly closed due to the CCS water temperature being colder than normal (at approximately 50 degrees F). Although this valve position maintained proper temperature prior to the feedwater pump start, it was not adequate to maintain lube oil temperature with the pump running.

CORRECTIVE ACTION TO PREVENT RECURRENCE

A better design and one which would have prevented the above failure is an automatic lube oil temperature control. Modification requests 86-0030, 86-0031 and 86-0032 have been initiated to install automatic temperature control circuits for these gate valves. In the interim, procedure revisions to SOP-0009 are in place to ensure careful monitoring of lube oil temperature and adjustment of CCS flow to maintain lube oil at the proper temperature after the start of the feedwater pump.

The inability to open the B feedwater pump discharge valve against the high differential pressure was attributed to incorrect torque switch settings. The torque switch settings were reset and the valve retested satisfactorily. A similar feedwater pump B discharge valve failure to open occurred on 11/21/85 and was reported in LER 85-041. At that time it was not known that the torque switch settings were incorrect and the failure was attributed to the high differential pressure.

There was no single release of radioactivity or single radiation exposure specifically associated with the outage which accounted for more than 10 percent of the allowable annual values.

OPERATING TIME LOSS

1. Generator-off-line hours - 0

2. Time loss when reactor was "forced" subcritical, all rods in. Time Loss: 22 hours 23 minutes

3. Torque switch settings on the B feedwater pump discharge valve were determined to be incorrect and were reset prior to reactor startup. Resetting of these torque switches proved to be the critical path for this outage.

REACTOR SCRAM 85-06

At 1015 on 12/31/85 with the unit in Operational Condition 1 (Power Operation), with reactor power at 20 percent and the turbine generator synchronized to the grid, the four turbine control valves and four turbine intercept valves were given a fast closure signal. The increased pressure caused the turbine bypass valves to open. Fifty-one seconds later the four turbine stop valves tripped which caused a turbine generator trip. Approximately seven seconds later, the reactor scrammed on high pressure (Reference LER 85-063).

Investigation of the incident revealed that the actuation of the turbine control/intercept valve fast closure was due to a false indication of a turbine generator power to load imbalance. This sensed power to load imbalance was caused by a combination of two separate occurrences. A pressure transducer sensing impulse pressure to the low pressure turbine stage had failed prior to the scram and was documented on a Maintenance Work Request (MWR) to be reworked. At the time of this event the pressure transducer was failed high. Also, a transient was introduced on the Gulf States Utilities (GSU) grid by a lightning strike on a 500 KV transmission line. The power to load imbalance relay requires a power differential between the steam input to the low pressure turbine and the electrical output of the generator of 40 percent differential and a sufficient rate of change of current before it will trip. The failed pressure transducer provided the 40 percent differential and the lightning strike provided a sufficient rate of change in current.

Once the turbine control/intercept valves were given the fast closure signal they tripped and immediately tried to reset. With all eight valves trying to reset simultaneously there was a sufficient loss of hydraulic pressure in the Emergency Trip System to cause the turbine bypass valve to open and the four turbine stop valves to shut. This caused a turbine generator trip and reactor scram on high pressure.

CORRECTIVE ACTION TO PREVENT RECURRENCE

In an effort to prevent recurrence the pressure transducer has been replaced via Maintenance Work Request 6637. A retrofit of the reset logic circuitry for the turbine intercept valve fast closure is being investigated. This retrofit would sequence the reopening of the turbine intercept valves one at a time to minimize the hydraulic pressure reduction. Modification Request 86-0129 has been initiated to request this design change.

There was no single release of radioactivity or single radiation exposure specifically associated with the outage which accounted for more than 10 percent of the allowable annual values.

OPERATING TIME LOSS

1. In terms of generator-off-line hours
   Time Loss: 16 hours 19 minutes

2. The failed pressure transducer which was replaced prior to reactor startup and was the critical path item for restart.

3.2 REDUCTIONS IN POWER

There were no forced reductions in power other than those reported in 3.1 of this report.

4.0 FUEL STATUS

4.0 FUEL STATUS

River Bend began receiving its initial core load of fuel from General Electric on February 1, 1985. Final receipt, channeling, inspection, and initial storage was completed March 8, 1985. The loading of the 624 bundle initial core began on August 31, 1985 and was completed September 21, 1985. Initial criticality was achieved on October 31, 1985 and River Bend continued to perform start-up testing through the end of 1985 in accordance with Chapter 14 of the FSAR.

The first power generated by River Bend occurred December 3, 1985. The station has generated 129,928 MWH thermal which equals 1.90 effective full power days through the end of 1985. The remaining energy for Cycle I is estimated to be 307.8 effective full power days.

River Bend Cycle I contains 5 fuel bundle types described as follows:

76 fuel bundles of 0.7 weight percent of U-235
108 fuel bundles of 0.9 weight percent of U-235
120 fuel bundles of 1.6 weight percent of U-235
280 fuel bundles of 2.5 weight percent of U-235
40 fuel bundles of 2.8 weight percent of U-235

The River Bend design encompasses General Electric's Control Cell Core and Barrier Fuel concept. However, the station did not operate at levels high enough in 1985 to benefit from the expected capacity factor improvements. Both water chemistry and offgas monitoring clearly show that no failure of fuel has occurred during the 1985 operations.

5.0 MAINTENANCE

5.0 MAINTENANCE

This section covers the period from initial criticality (10-31-85) through the end of 1985.

During the first two months of power ascension, problems have arisen as systems are operated as an integrated electrical generating facility. These system interface problems are not as critical as those that result in outages but effect future availability as we increase in power and achieve commercial operation. To address these problems task forces are assembled as the systems problems are identified. These task forces are charged with investigating and determining modifications to be implemented to improve the efficiency of system operations.

This section describes the problems that have occurred and work being performed to correct them.

REACTOR WATER CLEANUP ISOLATIONS

Problems have been experienced with inadvertent isolations of the Reactor Water Cleanup system (RWCU). These problems have been categorized into the areas of hardware deficiencies, personnel errors, and procedural deficiencies. Specific hardware problems include:

1. Water flashing downstream of a flow element causing erratic flow oscillations.

2. Trip setpoints on the differential flow instrument that does not taken into account density changes.

A task force has been assembled to address the RWCU problems, and needed design changes are being implemented. To date the following modifications have been performed:

1. Some orifices in the RWCU system were replaced to eliminate flashing downstream of the flow element.

2. An isolation bypass switch was installed for the purpose of preventing inadvertent isolations of RWCU during performance of related surveillances.

As a result of these modifications RWCU isolations have been reduced.

CONTAINMENT AIRLOCK DOORS

Numerous problems have been encountered with the operation of the containment airlock doors. River Bend has two airlocks consisting of interlocked doors. Each door uses two inflatable seals. Problems encountered result from hardware deficiencies and improper usage by untrained personnel and include:

1. Seal ruptures

2. Seal retaining piece bolt failure

3. Leaking air valves and tubing

A task force has been assembled to review the airlock door design and provide a long-term solution. In the short term, the following has helped to reduce door failures:

- The seal design was changed by the manufacturer to prevent fatigue failure of retaining bolts through repeated inflate/deflate cycles.

- All leaking valves have been fixed and a heavier gauge tubing installed.

- Containment entries are being more closely controlled to minimize use.

- Personnel training on operation of the doors has been conducted.

6.0 OCCUPATIONAL RADIATION SUMMARY REPORT

6.0 OCCUPATIONAL RADIATION SUMMARY REPORT

Enclosed in this section are two reports on Occupational Radiation Exposure for River Bend Station personnel for the year 1985 as required by 10CFR20.407, Reg. Guide 1.16, and Technical Specification section 6.9.1.5a of Appendix A to River Bend Station Operating License Number NPF-47.

Occupational Radiation Summary Report For Calendar Year 1985

Listed below is the report which summarizies the number of personnel who received deep dose equivalent in the various annual dose ranges. This report is required by the requirements of 10CFR20.

Annual Whole Body Exposure Range(1) (rems)	Number of Individuals In Each Range
No measurable exposure	1,518
Measurable exposure less than 0.1	142
0.1 rem but less than 0.25 rem	15
0.25 rem but less than 0.5 rem	1
0.5 rem or greater	None
Total number of individuals reported	1,676

This total, 1,676, is the number of individuals for whom personnel monitoring was provided during calendar year 1985 as per 10CFR20.407, paragraph (a)(2).

Dwight M. Ross /
Dwight M. Ross
Radiological Health Supervisor

(1) Individual values exactly equal to the values separating exposure ranges are reported in the higher range.

1985 Calendar Year Report of Number of
Personnel and Man-Rem by Work and Job Function*

Work & Job Function	Number of Personnel (>100 mrem)			Total Man-Rem		
	Station Employees	Utility Employees	Contract Workers and Others	Station Employees	Utility Employees	Contract Workers and Others
Reactor Operations & Surveillance						
Maintenance Personnel	0	0	1	0	0	0.149
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	10	0	0	0.582	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Routine Maintenance						
Maintenance Personnel	0	0	1	0	0	0.015
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	9	0	1	0.276	0	0.030
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Inservice Inspection						
Maintenance Personnel	0	0	0	0	0	0
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	8	0	0	0.609	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	2	0	0	0.347
Special Maintenance						
Maintenance Personnel	0	0	2	0	0	0.468
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	6	0	1	0.211	0	0.092
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Waste Processing						
Maintenance Personnel	0	0	0	0	0	0
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	0	0	0	0	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
Refueling						
Maintenance Personnel	0	0	0	0	0	0
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	0	0	0	0	0	0
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	0	0	0	0
TOTAL**						
Maintenance Personnel	0	0	3	0	0	0.632
Operating Personnel	0	0	0	0	0	0
Health Physics Personnel	10	0	1	1.678	0	0.122
Supervisory Personnel	0	0	0	0	0	0
Engineering Personnel	0	0	2	0	0	0.347
GRAND TOTAL	10	0	6	1.678	0	1.101

*Only personnel with TLD >100 mrem appear on this report. A worker may be counted in more than one category.
**Figures represent the actual number of personnel exposed.

Wayne C. Hardy / TMP

Wayne C. Hardy, Radiological Engineering Supervisor

7.0 SAFETY RELIEF VALVE HISTORY

7.0 SAFETY RELIEF VALVE HISTORY

Enclosed is documentation of all challenges to safety/relief valves and a summary of their performance as required by Technical Specification Section 6.9.1.5.b of Appendix A to River Bend Station License Number NPF-47.

During 1985, challenges to the Safety Relief Valves (SRVs) were limited to that required as part of the Startup Testing Program at River Bend Station. All SRV operations were preplanned and there were no failures of SRVs to perform as designed. Included in this section is a listing of the SRVs and the time and date of each actuation. In addition the Reporting Operating Information on Main Steam Safety/Relief Valves for Institute of Nuclear Power Operations has been provided.

SRV ACTUATION EVENTS *

SRV	DATE	TIME
1B21*F041A	12-06-85	1058
1B21*F041B	11-19-85 12-06-85	1558 1105
1B21*F041C	11-19-85 12-06-85	1552 1204
1B21*F041D	11-19-85 12-06-85	1542 1110
1B21*F041F	11-19-85 12-06-85	1555 1154
1B21*F041G	12-06-85	1147
1B21*F041L	12-06-85	1159
1B21*F047A	11-19-85 12-06-85	1536 1213
1B21*F047B	12-06-85	1053
1B21*F047C	11-19-85 12-06-85	1549 1040
1B21*F047D	12-06-85	1047
1B21*F047F	12-06-85	0803
1B21*F051B	12-06-85	1033
1B21*F051C	12-06-85	1230
1B21*F051D	12-06-85	0452
1B21*F051G	11-19-85 12-06-85	1546 0813

*All SRV actuations took place as part of the Startup Test Program.

REPORTING OPERATING INFORMATION
ON
MAIN STEAM LINE SAFETY/RELIEF VALVES
FOR
INSTITUTE OF NUCLEAR POWER OPERATIONS

INITIAL RECORDS

SRVS Prepared By: [signature] INITIAL RECORDS 100. PLANT DOCKET # 50- 4 5 8

Approved By: [signature]

NOTE: UP TO FIVE VALVES WITH IDENTICAL MODEL NUMBERS MAY BE REPORTED ON ONE SHEET.

SAME FOR EACH VALVE:

107. S/R VALVE MANUFACTURER
(71) [X] CROSBY VALVE
(91) [ ] DIKKERS
(92) [ ] TARGET ROCK CORP.
[ ] OTHER: ________

109. VALVE SIZE
A. [ ] 6x10, 6x12, ETC.
B. [X] 8x10, 8x12, ETC.
C. [ ] OTHER: ________

108. DATE OF FIRST FUEL LOAD (MO/DA/YR)
8-31-85

110. MANUFACTURER'S MODEL NUMBER (EXACTLY AS GIVEN BY MANUFACTURER)
HB-65-DF

111. VESSEL HYDROTEST PRESSURE AFTER INSTALLATION
1025 PSIG

112. PLANT'S RATED CAPACITY
2894 (MWt)

GIVE THIS INFORMATION FOR EACH VALVE:

	1.	2.	3.	4.	5.
101. MANUFACTURER'S SERIAL NUMBER (EXACTLY AS GIVEN)	N63800-00-0221	N63800-00-0217	N63800-00-0218	N63800-00-0219	
102. INITIAL COMPONENT ID (VALVE LOCATION OR "S" FOR SPARE)	1B21*RVF041A	1B21*RVF041B	1B21*RVF041C	1B21*RVF041D	
103. PART OF ADS? (YES OR NO)	No	Yes	Yes	Yes	
104. SET PRESSURE ON NAMEPLATE (PSIG)	1165	1165	1165	1165	
105. VALVE CAPACITY (LB./HR. STEAM)	895,523	895,523	895,523	895,523	

SRVS Prepared By: [illegible signature]
Approved By: [illegible signature]

INITIAL RECORDS

100. PLANT DOCKET # 50- 4 5 8

NOTE: Up to Five Valves With Identical Model Numbers May Be Reported On One Sheet.

SAME FOR EACH VALVE:

107. S/R VALVE MANUFACTURER
- (71) [X] Crosby Valve
- [illegible] [ ] Dikkers
- [illegible] [ ] Target Rock Corp.
- [illegible] [ ] Other: ____

108. VALVE SIZE
- A. [ ] 6x10, [illegible], Etc.
- B. [X] [illegible], [illegible], Etc.
- C. [ ] Other: ____

109. DATE OF FIRST FUEL LOAD (MO/DA/YR)
8-31-85

110. MANUFACTURER'S MODEL NUMBER (EXACTLY AS GIVEN BY MANUFACTURER)
HB-65-DF

111. VESSEL HYDROTEST PRESSURE AFTER INSTALLATION
1025 PSIG

112. PLANT'S RATED CAPACITY
2894 (MWth)

GIVE THIS INFORMATION FOR EACH VALVE:

	1.	2.	3.	4.	5.
101. Manufacturer's Serial Number (Exactly As Given)	N63800-00-0220	N63800-00-0222	N63800-00-0233	N63800-00-0224	
102. Initial Component ID (Valve Location or "S" for Spare)	1B21*RVF041F	1B21*RVF041G	1B21*RVF041L	1B21*RVF047A	
103. Part of ADS? (Yes or No)	Yes	No	No	Yes	
104. Set Pressure on Nameplate (PSIG)	1165	1165	1165	1180	
105. Valve Capacity (Lb./Hr. Steam)	895,523	895,523	895,523	906,914	

SRVS Prepared By: PATch

Approved By: [illegible signature]

INITIAL RECORDS

100. PLANT DOCKET # 50- 4 5 8

NOTE: UP TO FIVE VALVES WITH IDENTICAL MODEL NUMBERS MAY BE REPORTED ON ONE SHEET.

SAME FOR EACH VALVE:

107. S/R VALVE MANUFACTURER
- (71) [X] CROSBY VALVE
- [illegible] [ ] DIKKERS
- [illegible] [ ] TARGET ROCK CORP.
- [illegible] [ ] OTHER: ________

108. VALVE SIZE
- A. [ ] 6x10, [illegible], ETC.
- B. [X] [illegible], ETC.
- C. [ ] OTHER: ________

109. DATE OF FIRST FUEL LOAD (MO/DA/YR)
8-31-85

110. MANUFACTURER'S MODEL NUMBER (EXACTLY AS GIVEN BY MANUFACTURER)
HB-65-DF

111. VESSEL HYDROTEST PRESSURE AFTER INSTALLATION
1025 PSIG

112. PLANT'S RATED CAPACITY
2894 (MWTH)

GIVE THIS INFORMATION FOR EACH VALVE:

	1.	2.	3.	4.	5.
101. MANUFACTURER'S SERIAL NUMBER (EXACTLY AS GIVEN)	N63800-00-0226	N63800-00-0243	N63800-00-0227	N63800-00-0228	
102. INITIAL COMPONENT ID (VALVE LOCATION OR "S" FOR SPARE)	1B21*RVF047B	1B21*RVF047C	1B21*RVF047D	1B21*RVF047F	
103. PART OF ADS? (YES OR NO)	No	Yes	No	No	
104. SET PRESSURE ON NAMEPLATE (PSIG)	1180	1180	1180	1180	
105. VALVE CAPACITY (LB./HR. STEAM)	906,914	906,914	906,914	906,914	

SRVS Prepared By: [illegible] Approved By: [illegible]

INITIAL RECORDS

100. PLANT DOCKET # 50- 4 5 8

NOTE: UP TO FIVE VALVES WITH IDENTICAL MODEL NUMBERS MAY BE REPORTED ON ONE SHEET.

SAME FOR EACH VALVE:

107. S/R VALVE MANUFACTURER
- (71) [X] CROSBY VALVE
- [illegible] [ ] DIKKERS
- [illegible] [ ] TARGET ROCK CORP.
- [ ] OTHER: ______

108. VALVE SIZE
- A. [ ] 6x10, [illegible], ETC.
- B. [X] [illegible], [illegible], ETC.
- C. [ ] OTHER: ______

109. DATE OF FIRST FUEL LOAD (MO/DA/YR)

8-31-85

110. MANUFACTURER'S MODEL NUMBER (EXACTLY AS GIVEN BY MANUFACTURER)

HB-65-DF

111. VESSEL HYDROTEST PRESSURE AFTER INSTALLATION

1025 PSIG

112. PLANT'S RATED CAPACITY

2894 (MWTH)

GIVE THIS INFORMATION FOR EACH VALVE:

	1.	2.	3.	4.	5.
101. MANUFACTURER'S SERIAL NUMBER (EXACTLY AS GIVEN)	N63800-00-0230	N63800-00-0231	N63800-00-0232	N63800-00-0245	
102. INITIAL COMPONENT ID (VALVE LOCATION OR "S" FOR SPARE)	1B21*RVF051B	1B21*RVF051C	1B21*RVF051D	1B21*RVF051G	
103. PART OF ADS? (YES OR NO)	No	No	No	Yes	
104. SET PRESSURE ON NAMEPLATE (PSIG)	1190	1190	1190	1190	
105. VALVE CAPACITY (LB./HR. STEAM)	914,508	914,508	914,508	914,508	

SRVS Prepared By: [illegible signature] INITIAL RECORDS 100. PLANT DOCKET # 50-458

Approved By: [illegible signature]

NOTE: UP TO FIVE VALVES WITH IDENTICAL MODEL NUMBERS MAY BE REPORTED ON ONE SHEET.

SAME FOR EACH VALVE:

107. S/R VALVE MANUFACTURER
- (71) [X] CROSBY VALVE
- (72) [ ] DIKKERS
- (73) [ ] TARGET ROCK CORP.
- (74) [ ] OTHER: ______

108. VALVE SIZE
- A. [ ] 6x10, [illegible], ETC.
- B. [X] [illegible], [illegible], ETC.
- C. [ ] OTHER: ______

109. DATE OF FIRST FUEL LOAD (MM/DD/YY)
8-31-85

110. MANUFACTURER'S MODEL NUMBER (EXACTLY AS GIVEN BY MANUFACTURER)
HB-65-DF

111. VESSEL HYDROTEST PRESSURE AFTER INSTALLATION
1025 PSIG

112. PLANT'S RATED CAPACITY
2894 (MWTH)

GIVE THIS INFORMATION FOR EACH VALVE:

	1.	2.	3.	4.	5.
101. MANUFACTURER'S SERIAL NUMBER (EXACTLY AS GIVEN)	N63800-00-0223	N63800-00-0234	N63800-00-0235	N63800-00-0236	
102. INITIAL COMPONENT ID (VALVE LOCATION OR "S" FOR SPARE)	1B21*RVF041S	1B21*RVF041S	1B21*RVF041S	1B12*RVF041S	
103. PART OF ADS? (YES OR NO)	NA	NA	NA	NA	
104. SET PRESSURE ON NAMEPLATE (PSIG)	1165	1165	1165	1165	
105. VALVE CAPACITY (LB./HR. STEAM)	895,523	895,523	895,523	895,523	

SRVS Prepared By: [illegible signature]
Approved By: [illegible signature]

INITIAL RECORDS

100. PLANT DOCKET # 50- 4 5 8

NOTE: UP TO FIVE VALVES WITH IDENTICAL MODEL NUMBERS MAY BE REPORTED ON ONE SHEET.

SAME FOR EACH VALVE:

107. S/R VALVE MANUFACTURER
- (71) [X] CROSBY VALVE
- [illegible] [ ] DRESSER
- [illegible] [ ] TARGET ROCK CORP.
- [ ] OTHER: ________

108. VALVE SIZE
- A. [ ] 6X10, [illegible], ETC.
- B. [X] [illegible], [illegible], ETC.
- C. [ ] OTHER: ________

109. DATE OF FIRST FUEL LOAD (MO/DA/YR)
8-31-85

110. MANUFACTURER'S MODEL NUMBER (EXACTLY AS GIVEN BY MANUFACTURER)
HB-65-DF

111. VESSEL HYDROTEST PRESSURE AFTER INSTALLATION
1025 PSIG

112. PLANT'S RATED CAPACITY
2894 (MWt)

GIVE THIS INFORMATION FOR EACH VALVE:

	1.	2.	3.	4.	5.
101. MANUFACTURER'S SERIAL NUMBER (EXACTLY AS GIVEN)	N63800-00-0237	N63800-00-0244	N63800-00-0229		
102. INITIAL COMPONENT ID (VALVE LOCATION OR "S" FOR SPARE)	1B21*RVF041S	1B21*RVF047S	1B21*RVF051S		
103. PART OF ADS? (YES OR NO)	NA	NA	NA		
104. SET PRESSURE ON NAMEPLATE (PSIG)	1165	1180	1190		
105. VALVE CAPACITY (LB./HR. STEAM)	895,523	906,914	914,508		

REPORTING OPERATING INFORMATION
ON
MAIN STEAM LINE SAFETY/RELIEF VALVES
FOR
INSTITUTE OF NUCLEAR POWER OPERATIONS

ACTUATION EVENTS

SRVS   Prepared by: P. L. McCrary   Approved by: [illegible signature]

ACTUATION EVENTS

[illegible] PLANT DOCKET # 50 [illegible] 4 5 8

[illegible]

For each actuation or failure to actuate:	1.	2.	3.	4.	5.
301. S/R Valve Serial Number	N63800-00-0224	N63800-00-0219	N63800-00-0245	N63800-00-0243	N63800-00-0218
302. [illegible]	B21*F047A	B21*F041D	B21*F051G	B21*F047C	B21*F041C
303. [illegible]	11-19-85	11-19-85	11-19-85	11-19-85	11-19-85
304. [illegible]	1536	1542	1546	1549	1552
305. [illegible]	B	B	B	B	B
306. [illegible]	C	C	C	C	C
307. [illegible]	B	B	B	B	B
308. [illegible]	3%	3%	3%	3%	3%
309. [illegible]					
310. [illegible]	A	A	A	A	A
311. [illegible]					
312. [illegible]	900	900	900	900	900
[illegible]					
313. [illegible]	N/A	N/A	N/A	N/A	N/A
314. [illegible]	> 10 Sec	> 10 Sec	> 10 Sec	> 10 Sec	> 10 Sec
315. [illegible]	C	C	C	C	A
316. [illegible]	N/A	N/A	N/A	N/A	N/A
317. [illegible] (yes or no)	Yes	Yes	Yes	Yes	Yes

[illegible] '83

SRVS   Prepared by: P. L. McCrary   Approved by: ______   ACTUATION EVENTS   NUC. PLANT DOCKET # 50- 4 5 8

[illegible]

[illegible]	1.	2.	3.	4.	5.
301. S/N Valve Serial Number	N63800-00-0220	N63800-00-0217			
302. Component ID (Location)	RV B21*F041F	RV B21*F041B			
303. Date of Actuation [illegible]	11-19-85	11-19-85			
304. Time of Day [illegible]	1555	1558			
305. Type of Actuation [illegible]	B	B			
306. [illegible]	C	C			
307. [illegible]	B	B			
308. [illegible]	3%	3%			
309. [illegible]					
310. [illegible]	A	A			
311. [illegible]					
312. [illegible]	900	900			
[illegible]					
313. [illegible]	N/A	N/A			
314. [illegible]	> 10 Sec	> 10 Sec			
315. Failures, Defects [illegible]	C	C			
316. [illegible]	N/A	N/A			
317. [illegible] (yes or no)	Yes	Yes			

[illegible]

SRVS

LEAKAGE LOG OR ACTUATION EVENT COMMENT SHEET

PLANT DOCKET # 50- 4 5 8

NOTE: DATE AND TIME MUST MATCH THOSE ON REPORT SUBMITTED ON.
PLEASE DOUBLE CHECK FOR ACCURACY.

S/R VALVE SERIAL NUMBER: N63800-00-0218

COMMENT PERTAINS TO: LEAKAGE LOG [ ] OR ACTUATION EVENT [X]

DATE OF LEAK OR ACTUATION (MO/DA/YR): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): 1552

COMMENTS: During testing of ADS value 1B21*RVF041C the acoustic monitor for non-ads value 1B21*RVF051C responded. The Ads/SRV tail pipe temperature recorder verified value 1B21*RVF041C operated. Time required for tail pipe temperature to return to normal not available.

S/R VALVE SERIAL NUMBER: N63800-00-0243

COMMENT PERTAINS TO: LEAKAGE LOG [ ] OR ACTUATION EVENT [X]

DATE OF LEAK OR ACTUATION (MO/DA/YR): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): 1549

COMMENTS: Time required for tail pipe temperature to return to normal not available

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S/R VALVE SERIAL NUMBER: N63800-00-0245

COMMENT PERTAINS TO: LEAKAGE LOG [ ] OR ACTUATION EVENT [X]

DATE OF LEAK OR ACTUATION (MO/DA/YR): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): 1546

COMMENTS: Time required for tail pipes temperature to return to normal not available

SRVS

LEAKAGE LOG OR ACTUATION EVENT COMMENT SHEET

PLANT DOCKET # 50- 4 1 5 1 8 1

NOTE: DATE AND TIME MUST MATCH THOSE ON REPORT SUBMITTED ON.
PLEASE DOUBLE CHECK FOR ACCURACY.

S/R VALVE SERIAL NUMBER: N63800-00-0219

COMMENT PERTAINS TO: LEAKAGE LOG [ ] OR ACTUATION EVENT [X]

DATE OF LEAK OR ACTUATION (MO/DA/YR): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): 1542

COMMENTS: Time required for tail pipe temperature to return to normal not available

---

S/R VALVE SERIAL NUMBER: N63800-00-0224

COMMENT PERTAINS TO: LEAKAGE LOG [ ] OR ACTUATION EVENT [X]

DATE OF LEAK OR ACTUATION (MO/DA/YR): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): 1536

COMMENTS: Time required for tail pipe temperature to return to normal not available

---

S/R VALVE SERIAL NUMBER: N63800-00-0220

COMMENT PERTAINS TO: LEAKAGE LOG [ ] OR ACTUATION EVENT [X]

DATE OF LEAK OR ACTUATION (MO/DA/YR): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): 1555

COMMENTS: Time required for tail pipe temperature to return to normal not available

SRVS

LEAKAGE LOG OR ACTUATION EVENT COMMENT SHEET

PLANT DOCKET # 50- 4 1 5 1 8

NOTE: DATE AND TIME MUST MATCH THOSE OF REPORT COMMENTED ON. PLEASE DOUBLE CHECK FOR ACCURACY.

S/R VALVE SERIAL NUMBER: N63800-00-0217

COMMENT PERTAINS TO: LEAKAGE LOG ☐ OR ACTUATION EVENT ☒

DATE OF LEAK OR ACTUATION (MM/DD/YY): 11-19-85

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): ______

COMMENTS: Time required for tail pipe temperature to return to normal not available

S/R VALVE SERIAL NUMBER: ______

COMMENT PERTAINS TO: LEAKAGE LOG ☐ OR ACTUATION EVENT ☐

DATE OF LEAK OR ACTUATION (MM/DD/YY): ______

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): ______

COMMENTS: ______

S/R VALVE SERIAL NUMBER: ______

COMMENT PERTAINS TO: LEAKAGE LOG ☐ OR ACTUATION EVENT ☐

DATE OF LEAK OR ACTUATION (MM/DD/YY): ______

TIME OF LEAK OR ACTUATION (24 HOUR CLOCK): ______

COMMENTS: ______

8.0 REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY ANALYSIS

8.0 REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY ANALYSIS

During 1985 analysis of specific activity of primary coolant indicated that the Limiting Condition of Operation of specification 3.4.5 "Specific Activity" of River Bend Technical Specification was never exceeded.

As delineated by Technical Specification section 6.9.1.5.c of Appendix A to River Bend License Number NPF-47, no further information is required or enclosed.

GULF STATES UTILITIES COMPANY

RIVER BEND STATION POST OFFICE BOX 220 ST. FRANCISVILLE, LOUISIANA 70775

AREA CODE 504 635-6094 346-8651

March 3, 1986
RBG- 23,317
File Nos. G9.5, G9.25.1.5

Mr. Robert D. Martin, Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Dear Mr. Martin:

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

Enclosed is the River Bend Station Annual Operating Report for 1985. This report is submitted in accordance with Technical Specifications 6.9.1.4 and 6.9.1.5 of Appendix A to River Bend Station (RBS) License Number NPF-47.

Sincerely,

William J. Reed for
J.E. Booker
Manager-Engineering
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/DAS/ebm

cc: Director of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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