GULF STATES UTILITIES COMPANY

RIVER BEND STATION POST OFFICE BOX 220 ST. FRANCISVILLE, LOUISIANA 70775
AREA CODE 504 635-6094 346-8651

December 17 , 1990
RBG- 34174
File No. G9.5, G9.25.1

Director, Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Gentlemen:

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

Pursuant to 10CFR55.45(b)(5)(i), Gulf States Utilities Company is submitting NRC Form 474, "Simulation Facility Certification." The River Bend Station Simulator is a plant-referenced simulator as described in 10CFR55.45(b)(1)(ii) and ANSI/ANS3.5. Testing results and supporting documentation are on file at the River Bend Station Training Center.

If further information is required, please contact Mr. James W. Cook at (504) 381-4151.

Sincerely,

J. C. Deddens
Senior Vice President
River Bend Nuclear Group

WHO/LAE/DNL/JWC/JCM/pg

Attachments

cc: Mr. John Pellet
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

A005
1/1

NRC FORM 474 (1-90)
10 CFR 55.45(b), 55.4 and 55.5

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0138
EXPIRES: 9-30-92

ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 120 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0138), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

SIMULATION FACILITY CERTIFICATION

INSTRUCTIONS. This form is to be filed for initial certification, recertification (if required), and for any change to a simulation facility performance testing plan made after initial submittal of such a plan. Provide the following information, and check the appropriate box to indicate reason for submittal.

FACILITY: River Bend Station

DOCKET NUMBER: 50-458

LICENSEE: Gulf States Utilities Company

DATE: November 26, 1990

This is to certify that:

1. The above named facility licensee is using a simulation facility consisting solely of a plant-referenced simulator that meets the requirements of 10 CFR 55.45.
2. Documentation is available for NRC review in accordance with 10 CFR 55.45(b).
3. This simulation facility meets the guidance contained in ANSI/ANS 3.5, 1985, as endorsed by NRC Regulatory Guide 1.149.
   If there are any exceptions to the certification of this item, check here [X] and describe fully on additional pages as necessary. (See Attachment 1)

NAME *(or other identification)* AND LOCATION OF SIMULATION FACILITY

River Bend Simulator located at the
River Bend Training Center, St. Francisville, Louisiana

[X] SIMULATION FACILITY PERFORMANCE TEST ABSTRACTS ATTACHED. *(For performance tests conducted in the period ending with the date of this certification)*

DESCRIPTION OF PERFORMANCE TESTING COMPLETED *(Attach additional page(s) as necessary, and identify the item description being continued)*

See Attachment 2 - Performance Testing Completed

[X] SIMULATION FACILITY PERFORMANCE TESTING SCHEDULE ATTACHED. *(For the conduct of approximately 25% of performance tests per year for the four year period commencing with the date of this certification.)*

DESCRIPTION OF PERFORMANCE TESTING TO BE CONDUCTED. *(Attach additional page(s) as necessary, and identify the item description being continued)*

See Attachment 3 - Simulator Performance Testing to be Conducted

[ ] PERFORMANCE TESTING PLAN CHANGE. *(For any modification to a performance testing plan submitted on a previous certification)*

DESCRIPTION OF PERFORMANCE TESTING PLAN CHANGE *(Attach additional page(s) as necessary, and identify the item description being continued)*

N/A

[ ] RECERTIFICATION *(Describe corrective actions taken, attach results of completed performance testing in accordance with 10 CFR § 55.45(b)(5)(iv). Attach additional page(s) as necessary, and identify the item description being continued.)*

N/A

Any false statement or omission in this document, including attachments, may be subject to civil and criminal sanctions. I certify under penalty of perjury that the information in this document and attachments is true and correct.

SIGNATURE – AUTHORIZED REPRESENTATIVE: J C Deddens
James C. Deddens

TITLE: Sr. Vice President-RBNG

DATE: 12/17/90

In accordance with 10 CFR § 55.5, Communications, this form shall be submitted to the NRC as follows:

BY MAIL ADDRESSED TO: Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

BY DELIVERY IN PERSON TO THE NRC OFFICE AT: One White Flint North
11555 Rockville Pike
Rockville, MD

ATTACHMENT 1 - EXCEPTIONS TAKEN TO ANSI/ANS 3.5, 1985

Section 3.1.2, item (20) of ANS 3.5 identifies feedwater and steam line breaks inside and outside containment as required malfunctions. The River Bend simulator does not have the capability to simulate feedwater line breaks. The River Bend simulator was procured prior to the feedwater line break being added to ANS 3.5. The transient response and operator response is bounded by recirculation line breaks and steam line breaks. Feedwater line breaks have not been identified as being required by the operator training program nor is it a required control manipulation identified in 10CFR55.59(c)(3)(i).

Section 3.1.2, item (12) of ANS 3.5 identifies misaligned control rods as a required malfunction. The River Bend simulator does include malfunctions to stick, drift, uncouple, and drop control rods which could result in misaligned rods but no specific malfunction exists to just misalign rods in a group. The present capability meets the needs of the licensed operator training programs.

Section 3.2.2 of ANS 3.5 specifies requirements for panel controls to be simulated. Two deviations have been identified between the reference plant and the simulator resulting from modifications to the plant control room more than 18 months ago. The first deviation is a Bypass/Inoperability status indicator grid on panel P863 insert 74B. The plant grid now consists of 20 grid sections while the simulator grid only has 10 sections. Materials are on order to correct this deviation. The second deviation is the Harris circulating water and normal service water control panel on panel P808 insert 85C. The simulator controls are arranged in a different order than those in the reference plant control room and 4 indicating switches are missing in the simulator. Neither of the deviations has a significant detrimental impact on operator training; however, modifications to correct both deviations are in progress. This work is projected to be completed by November 1, 1991.

ATTACHMENT 2 - PERFORMANCE TESTING COMPLETED

1. Computer Real Time Test

   A counter was initialized in a simulation module that cycles at a frequency of twice a second. At the end of at least 30 minutes, the simulator was frozen and the actual program cycle frequency was compared to real time as determined by clock time.

2. Steady State and Normal Operations Tests

   Data was obtained from the plant process computer at power levels of approximately 100%, 70%, and 50%. The simulator was manipulated to match the plant power and recirculation flow rate for each condition and the critical and non-critical parameters of ANS 3.5, Appendix B were compared.

   Verification was obtained that the simulator could be operated with the reference plant operating procedures through the licensed operator training program. Simulator scenarios covering plant startup and heatup, plant heatup and power escalation, and plant shutdown and cooldown require utilization of controlled distribution reference plant procedures in the performance of those operations.

   A simulator stability test was performed by placing the simulator at an initial condition of 100% power and then, utilizing a computer program, determining the variation of each critical parameter at one minute intervals for a period of sixty minutes. There were no test failures during the steady state and normal operations tests.

3. Transient Tests

   The following transient tests were performed as specified by ANS 3.5, Appendix B:

   a. Manual Scram
   b. Simultaneous Trip of all Feedwater Pumps
   c. Simultaneous Closure of all MSIV's
   d. Simultaneous Trip of Both Reactor Recirculation Pumps
   e. A Single Recirculation Pump Trip
   f. A Main Turbine Trip From 37% Reactor Power
   g. A Maximum Rate Power Ramp of 100 to 75 to 100% Power
   h. A Recirculation Loop Rupture With a Loss of all Offsite Power
   i. An Unisolable Main Steam Line Rupture Within the Drywell

j. A Simultaneous Isolation of All MSIV's With One Stuck Open Safety Relief Valve and No High Pressure ECCS

Each transient (except f) was initiated from 100% power. Data identified in Appendix B to ANS 3.5 was recorded using a simulator computer at 1/2 second intervals until each transient stabilized. Plots of the data were compared to the USAR predictions or to the curves in the River Bend Station Control Systems Design Report produced by General Electric.

One test failure occurred during the transient testing. Test d), Simultaneous Trip of Both Reactor Recirculation Pumps, required satisfactory compliance with a Startup Test 1-ST-30B which specified a minimum and maximum loop flow coastdown time. The observed simulator flow during coastdown was 10-15% above the allowed values over the period of 1 to 5 seconds after the dual pump trip. Other data taken during the transient was satisfactory. The flow coastdown problem will be corrected during 1991 prior to the next annual set of certification transient tests. All other transient tests were satisfactory.

4. <u>Malfunction Tests</u>

The following malfunction tests were performed corresponding to the malfunctions listed in Section 3.1.2 of ANS 3.5:

<u>1 - Loss of Coolant</u>

1) Steam Leak Inside the Drywell

- A 3000 gpm unisolable leak is inserted inside the drywell from 100% reactor power.

2) Steam Leak Outside the Containment

- A 10% steam line rupture is introduced in a steam line going through the steam tunnel in the Auxiliary Building from 100% reactor power.

3) Relief Valve B21-F041A Fails Open

- One of the 16 safety relief valves is failed open from 100% reactor power.

2 - Loss of Instrument Air

1) Instrument Air Line Rupture

- An instrument air header is ruptured with the reactor at 100% power.

3 - Loss or Degraded Electrical Power

1) Loss of 1NPS-SWG1A

- A ground short is inserted in the 13.8 KV normal bus A from 100% reactor power.

2) Loss of 1NPS-SWG1B

- A ground short is inserted in the 13.8 KV normal bus B from 100% reactor power.

3) Loss of 1NPS-SWG1C

- A ground short is inserted in the 13.8 KV normal bus C from 100% reactor power.

4) Loss of 1NNS-SWG1A

- A ground short is inserted in the 4160 V normal bus A from 100% reactor power.

5) Loss of 1NNS-SWG1B

- A ground short is inserted in the 4160 V normal bus B from 100% reactor power.

6) Loss of 1NNS-SWG1C

- A ground short is inserted in the 4160 V normal bus C from 100% reactor power.

7) Loss of 1ENS*SWG1A

- A ground short is inserted in the 4160 V emergency bus A from 100% reactor power.

8) Loss of 1ENS*SWG1B

- A ground short is inserted in the 4160 V emergency bus B from 100% reactor power.

9) Loss of 1E22*S004

- A ground short is inserted in the 4160 V emergency bus C from 100% reactor power.

10) Loss of 1EJS*SWG1A

- A ground short is inserted in the 480 V emergency bus EJS-1A from 100% power.

11) Loss of 1EJS*SWG1B

- A ground short is inserted in the 480 V emergency bus EJS-1B from 100% power.

12) Loss of 1EJS*SWG2A

- A ground short is inserted in the 480 V emergency bus EJS-2A from 100% power.

13) Loss of 1EJS*SWG2B

- A ground short is inserted in the 480 V emergency bus EJS-2B from 100% power.

14) Loss of 1E22*S002

- A ground short is inserted in the 480 V division 3 emergency bus from 100% power.

15) Loss of 1NJS-LDC1A

- A ground short is inserted in the 480 V normal motor control center (MCC) 1A from 100% power.

16) Loss of 1NJS-LDC1B

- A ground short is inserted in the 480 V normal MCC 1B from 100% power.

17) Loss of 1NJS-LDC1C

- A ground short is inserted in the 480 V normal MCC 1C from 100% power.

18) Loss of 1NJS-LDC1D

- A ground short is inserted in the 480 V normal MCC 1D from 100% power.

19) Loss of 1NJS-LDC1E

- A ground short is inserted in the 480 V normal MCC 1E from 100% power.

20) Loss of 1NJS-LDC1F

- A ground short is inserted in the 480 V normal MCC 1F from 100% power.

21) Loss of 1NJS-LDC1G

- A ground short is inserted in the 480 V normal MCC 1G from 100% power.

22) Loss of 1NJS-LDC1H

- A ground short is inserted in the 480 V normal MCC 1H from 100% power.

23) Loss of 1NJS-LDC1J

- A ground short is inserted in the 480 V normal MCC 1J from 100% power.

24) Loss of 1NJS-LDC1K

- A ground short is inserted in the 480 V normal MCC 1K from 100% power.

25) Loss of 1NJS-LDC1L

- A ground short is inserted in the 480 V normal MCC 1L from 100% power.

26) Loss of 1NJS-LDC1M

- A ground short is inserted in the 480 V normal MCC 1M from 100% power.

27) Loss of 1BYS-SWG1A

- A ground short is inserted in the 125 V normal DC bus 1A.

28) Loss of 1BYS-SWG1B

- A ground short is inserted in the 125 V normal DC bus 1B.

29) Loss of 1ENB*SWG1A

- A ground short is inserted in the 125 V emergency DC bus 1A.

30) Loss of 1ENB*SWG1B

- A ground short is inserted in the 125 V emergency DC bus 1B.

31) Loss of 1E22*S001

- A ground short is inserted in the 125 V division 3 DC emergency bus.

32) Loss of 1SCM-PNL01A

- A ground short is inserted in the 120 V AC Station Control and Monitoring Panel 1A.

33) Loss of 1SCM-PNL01B

- A ground short is inserted in the 120 V AC Station Control and Monitoring Panel 1B.

34) Loss of 1VBS*PNL01A

- A ground short is inserted in the 120 V AC Division 1 instrument bus.

35) Loss of 1VBS*PNL01B

- A ground short is inserted in the 120 V AC Division 2 instrument bus.

36) Loss of 1VBN-PNL01A1

- A ground short is inserted in the normal 120 V AC instrument bus 1A1.

37) Loss of 1VBN-PNL01B1

- A ground short is inserted in the normal 120 V AC instrument bus 1B1.

38) Loss of RPS-A

- A ground short is inserted in the 120 V AC Reactor Protection power bus A.

39) Loss of RPS-B

- A ground short is inserted in the 120 V AC Reactor Protection power bus B.

40) Loss of Offsite Power

- A fault is simulated in all offsite power feeds to River Bend Station.

4 - Loss of Forced Coolant Flow

Both single and dual recirculation pump trips are tested in the transient tests.

5 - Loss of Condenser Vacuum and Level Control

1) 1CNS-H/A103 Fails Low

- The automatic output signal of the condenser hotwell makeup controller is failed low at 100% power.

2) 1CNS-H/A104 Fails Low

- The automatic output signal of the condenser hotwell emergency makeup controller is failed low at 100% power.

3) 1CNS-H/A103 Fails High

- The automatic output signal of the condenser hotwell makeup controller is failed high at 100% power.

4) 1CNS-H/A104 Fails High

- The automatic output signal of the condenser hotwell emergency makeup controller is failed high.

5) Main Condenser Vacuum Loss

- A 50% severity air inleakage to the main condenser is inserted at 100% power.

6 - Loss of Service Water

1) Loss of NSW Pump/Loss of NSW

- One normal service water pump is failed to verify system response to a single pump trip. Then, the remaining normal service water pumps are tripped - all from an initial 100% power.

7 - Loss of Shutdown Cooling

1) RHR Pump Trip

- The simulator is aligned in a shutdown condition with one RHR pump running in the shutdown cooling mode. The RHR pump is then tripped.

8 - Loss of Component Cooling Water

1) Loss of CCP Pump/Loss of CCP

- One Reactor Plant Component Cooling (CCP) water pump is failed to verify the system response to a single pump trip. Then the remaining CCP pumps are tripped.

2) Loss of CCS Pump/Loss of CCS

- One Turbine Plant Component Cooling Water (CCS) pump is failed to verify the system response to a single pump trip. Then the remaining CCS pumps are tripped.

9 - Normal Feedwater Failure

1) Feedwater Regulating Valve Failure

From 100% power, one of the three reactor feedwater regulating valves is failed to a 10% open position. The simulator is then reset to 100% power, and one valve is failed to the 100% open position.

2) Feedwater Pump Trip

One of the three reactor feedwater pumps is tripped from 100% power.

10 - Loss of all Feedwater

Loss of all feedwater is tested in the transient tests.

11 - Loss of Protective System Channel

1) RPS Failure to SCRAM on Low Water Level

- The RPS low reactor water level scram is disabled from 100% power. The reactor feedwater flow is then decreased to slowly decrease vessel level below the RPS actuation setpoint.

12 - Control Rod Failures

1) Control Rod Drifts In

- One control rod is failed such that it slowly drifts into the core from 100% power.

2) Control Rod Drifts Out

- One control rod is failed such that it slowly drifts out of the core from 100% power.

3) Control Rod Accumulator Failure

- One control rod scram accumulator is depressurized from 100% power.

4) Stuck Control Rod

- One control rod is stuck at an intermediate core position from 100% power.

5) Control Rod Uncoupled

- One control rod is uncoupled from the control rod drive spud. The control rod is then fully withdrawn from the core.

13 - Inability to Drive Control Rods

1) CRD Pump Trip

- The operating control rod drive hydraulic pump is tripped from 100% power.

14 - Fuel Cladding Failure

1) Fuel Cladding Failure

- A small cladding failure is inserted at 100% power to verify initial indications. The severity is then increased to obtain a Main Steam Isolation Valve (MSIV) isolation on high main steam line radiation levels.

15 - Turbine Trip

1) Turbine Trip

- A main turbine trip signal is inserted at 100% power.

16 - Generator Trip

1) Generator Trip

- A main generator trip signal is inserted at 100% power.

17 - Automatic Control Systems Affecting Reactivity and Core Heat Removal

1) EHC Pressure Regulator Fails Low

- The steam header pressure input to the turbine control and turbine bypass valve control regulation system is failed low from 100% power.

2) EHC Pressure Regulator Fails High

- The steam header pressure input to the turbine control and turbine bypass valve control regulation system is failed high from 100% power.

18 - Failure of Reactor Coolant Pressure and Volume Control Systems (PWR)

Not Applicable to BWR's.

19 - Reactor Trip

The reactor trip is tested in the transient tests.

20 - Main Steam Line Breaks

Steam line breaks are tested in Section 1 (Loss of Coolant Malfunctions).

21 - Nuclear Instrumentation

1) SRM Failure - Inoperative

One Source Range Monitor channel is failed as inoperative (tripped) with the reactor in a startup mode and power indicating in the source range.

2) IRM Failure - Inoperative

One Intermediate Range Monitor channel is failed as inoperative (tripped) with the reactor in a startup mode and power indicating in the intermediate range.

3) APRM Failure - Inoperative

One Average Power Range Monitor channel is failed as inoperative (tripped) with the reactor at 100% power.

22 - Process System Control Failures

1) Feedwater Master Level Controller Fails Low

- The feedwater system master level controller automatic output signal to the feedwater regulating valve controllers is failed low at 100% power.

2) MSR Level Controller Fails

- One of the moisture separator-reheater (MSR) shell side level inputs to the respective level controller is failed low with the MSR on line at 100% reactor power.

23 - Passive Safety System Failures

1) RCIC Fails to Auto Start

- A failure is inserted into the automatic initiation logic for the Reactor Core Isolation Cooling (RCIC) system at 100% power. Reactor feedwater flow is then manually decreased such that reactor water level drops below the RCIC initiation setpoint.

2) HPCS Fails to Auto Start

- A failure is inserted into the automatic initiation logic for the High Pressure Core Spray (HPCS) system at 100% power. Reactor feedwater flow is then manually decreased such that reactor water level drops below the HPCS initiation setpoint.

24 - Failure of the Automatic Reactor Trip System

1) RPS Fails to SCRAM

- A failure is inserted into the Reactor Protection System (RPS) that prevents an automatic reactor SCRAM signal from being processed. A steam leak is then introduced into the drywell building raising the drywell to containment building diferential pressure to above the RPS trip setpoint.

25 - Reactor Pressure Control System Failure

1) Turbine Bypass Valves Fail Closed

- A close signal is sent to the turbine bypass valves with the reactor at approximately 10% power and the turbine is off-line.

2) Turbine Bypass Valves Fail Open

- An open signal is sent to the turbine bypass valves with the reactor at 100% power.

There were no test failures of the malfunction tests.

ATTACHMENT 3 - SIMULATOR PERFORMANCE TESTING TO BE CONDUCTED

1 - Tests to be performed each year

1 - Computer Real Time Test
2 - Steady State and Normal Operations Tests
3 - Transient Tests

2 - Tests to be performed during calendar year 1991

1 - Steam Leak Inside the Drywell
2 - Steam Leak Outside the Containment
3 - Relief Valve B21-F041A Fails Open
4 - Instrument Air Line Rupture
5 - Loss of 1NPS-SWG1A
6 - Loss of 1NPS-SWG1B
7 - Loss of 1NPS-SWG1C
8 - Loss of 1NNS-SWG1A
9 - Loss of 1NNS-SWG1B
10 - Loss of 1NNS-SWG1C
11 - Loss of 1ENS*SWG1A
12 - Loss of 1ENS*SWG1B
13 - Loss of 1E22*S004
14 - Loss of 1EJS*SWG1A
15 - Loss of 1EJS*SWG1B
16 - Loss of 1EJS*SWG2A
17 - Loss of 1EJS*SWG2B
18 - Loss of 1E22*S002
19 - Loss of 1NJS-LDC1A

3 - Tests to be performed during calendar year 1992

1 - Loss of 1NJS-LDC1B
2 - Loss of 1NJS-LDC1C
3 - Loss of 1NJS-LDC1D
4 - Loss of 1NJS-LDC1E
5 - Loss of 1NJS-LDC1F
6 - Loss of 1NJS-LDC1G
7 - Loss of 1NJS-LDC1H
8 - Loss of 1NJS-LDC1J
9 - Loss of 1NJS-LDC1K
10 - Loss of 1NJS-LDC1L
11 - Loss of 1NJS-LDC1M
12 - Loss of 1BYS-SWG1A
13 - Loss of 1BYS-SWG1B
14 - Loss of 1ENB*SWG1A
15 - Loss of 1ENB*SWG1B
16 - Loss of 1E22*S001
17 - Loss of 1SCM-PNL01A

18 - Loss of 1SCM-PNL01B
19 - Loss of 1VBS*PNL01A

4 - Tests to be performed during calendar year 1993

1 - Loss of 1VBS*PNL01B
2 - Loss of 1VBN-PNL01A1
3 - Loss of 1VBN-PNL01B1
4 - Loss of RPS-A
5 - Loss of RPS-B
6 - Loss of Offsite Power
7 - 1CNS-H/A103 Fails Low
8 - 1CNS-H/A104 Fails Low
9 - 1CNS-H/A103 Fails High
10 - 1CNS-H/A104 Fails High
11 - Main Condenser Vacuum Loss
12 - Loss of NSW Pump/Loss of NSW
13 - RHR Pump Trip
14 - Loss of CCP Pump/Loss of CCP
15 - Loss of CCS Pump/Loss of CCS
16 - Feedwater Regulating Valve Failure
17 - Feedwater Pump Trip
18 - RPS Failure to SCRAM on Low Water Level
19 - Control Rod Drifts In

5 - Tests to be performed during calendar year 1994

1 - Control Rod Drifts Out
2 - Control Rod Accumulator Failure
3 - Stuck Control Rod
4 - Control Rod Uncoupled
5 - CRD Pump Trip
6 - Fuel Cladding Failure
7 - Turbine Trip
8 - Generator Trip
9 - EHC Pressure Regulator Fails Low
10 - EHC Pressure Regulator Fails High
11 - SRM Failure - Inoperative
12 - IRM Failure - Inoperative
13 - APRM Failure - Inoperative
14 - Feedwater Master Level Controller Fails Low
15 - MSR Level Controller Fails
16 - RCIC Fails to Auto Start
17 - HPCS Fails to Auto Start
18 - RPS Fails to SCRAM
19 - Turbine Bypass Valves Fail Closed
20 - Turbine Bypass Valves Fail Open