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November 23, 1990

Docket No. 50-458

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Gulf States Utilities
ATTN: Mr. James C. Deddens
Senior Vice President (RBNG)
Post Office Box 220
St. Francisville, Louisiana 70775

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 77503)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 52 to Facility Operating License No. NPF-47 for the River Bend Station (RBS), Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated August 22, 1990, and supplemented by letter dated October 17, 1990.

The amendment revises the TSs regarding operation in the steam condensing mode (SCM) of the residual heat removal (RHR) system. Currently, License Condition C.5.a to Facility Operating License No. NPF-47 requires NRC written approval prior to use of the SCM of RHR. Gulf States Utilities (GSU) has determined that the SCM of RHR is not required for safe operation of RBS and plans on permanently disabling the SCM of RHR. Therefore, License Condition C.5.a is no longer applicable. The amendment deletes maintenance and surveillance requirements for three valves associated with the SCM and establishes a final trip setpoint for the High RHR/Reactor Coolant Isolation Cooling (RCIC) steam line flow for RCIC isolation.

A copy of our Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original Signed By

Claudia M. Abbate, Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 52 to NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page

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*SEE PREVIOUS CONCURRENCES

OFC	PDIV-2/LA	PDIV-2/PM	OGC	SRXB/BC	PDIV-2/(A)D
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DATE	10/18/90*	10/18/90*	10/26/90*	10/18/90*	~~10~~ 11/19/90

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

GULF STATES UTILITIES COMPANY

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 52
License No. NPF-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Gulf States Utilities Company (the licensee) dated August 22, 1990, and supplemented by letter dated October 17, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 52 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. GSU shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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Eugene V. Imbro, Acting Director
Project Directorate IV-2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 52

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3/4 3-22	3/4 3-22
3/4 3-23	3/4 3-23
3/4 3-79	3/4 3-79

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
4. REACTOR WATER CLEANUP SYSTEM ISOLATION (Cont'd)		
d. Equipment Area Δ Temperature - High		
1. Heat Exchanger Room	$\leq$ 39°F	$\leq$ 42.5°F
2. Pump Rooms A and B	$\leq$ 78°F	$\leq$ 82°F
3. Valve Nest Room	$\leq$ 46°F	$\leq$ 49.5°F
4. Demineralizer Rooms 1 and 2	$\leq$ 46°F	$\leq$ 49.5°F
5. Receiving Tank Room	$\leq$ 46°F	$\leq$ 49.5°F
e. Reactor Vessel Water Level - Low Low Level 2	$\geq$ - 43 inches*	$\geq$ - 47 inches
f. Main Steam Line Tunnel Ambient Temperature - High	$\leq$ 141°F	$\leq$ 148.5°F
g. Main Steam Line Tunnel Δ Temperature - High	$\leq$ 57°F	$\leq$ 61°F
h. SLCS Initiation	NA	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION		
a. RCIC Steam Line Flow - High	$\leq$ 127" H_2O	$\leq$ 135.5" H_2O
b. RCIC Steam Line Flow - High Timer	$\geq$ 3 seconds	$\leq$ 13 seconds
c. RCIC Steam Supply Pressure - Low	$\geq$ 60 psig	$\geq$ 55 psig
d. RCIC Turbine Exhaust Diaphragm Pressure - High	$\leq$ 10 psig	$\leq$ 20 psig
e. RCIC Equipment Room Ambient Temperature - High	$\leq$ 182°F	$\leq$ 186.4°F
f. RCIC Equipment Room Δ Temperature - High	$\leq$ 96°F	$\leq$ 99°F

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Cont'd)		
g. Main Steam Line Tunnel Ambient Temperature - High	≤ 141°F	≤ 148.5°F
h. Main Steam Line Tunnel Δ Temperature - High	≤ 57°F	≤ 61°F
i. Main Steam Line Tunnel Temperature Timer	0 seconds	NA
j. RHR Equipment Room Ambient Temperature - High	≤ 117°F	≤ 121.1°F
k. RHR Equipment Room Δ Temperature - High	≤ 29°F	≤ 33.6°F
l. RHR/RCIC Steam Line Flow - High	≤ 60.7" H_2O	≤ 64.2" H_2O
m. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
n. Manual Initiation	NA	NA
6. RHR SYSTEM ISOLATION		
a. RHR Equipment Area Ambient Temperature - High	≤ 117°F	≤ 121.1°F
b. RHR Equipment Area Δ Temperature - High	≤ 29°F	≤ 33.6°F
c. Reactor Vessel Water Level - Low Level 3	≥ 9.7 inches*	≥ 8.7 inches
d. Reactor Vessel Water Level - Low Low Low Level 1	≥ - 143 inches*	≥ - 147 inches

TABLE 3.3.2-2 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
6. RHR SYSTEM ISOLATION (Cont'd)		
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	≤ 135 psig	≤ 150 psig
f. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
7. MANUAL INITIATION	NA	NA

* See Bases Figure B 3/4 3-1.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$
b. Drywell Pressure - High	$\leq 10^{(a)}$
c. Containment Purge Isolation Radiation - High(b)	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low Level 1	≤ 1.0 */$\leq 10^{(a)}$**
b. Main Steam Line Radiation - High(b)	≤ 1.0 */$\leq 10^{(a)}$**
c. Main Steam Line Pressure - Low	≤ 1.0 */$\leq 10^{(a)}$**
d. Main Steam Line Flow - High	≤ 0.5 */$\leq 10^{(a)}$**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Area Temperature - High (Turbine Bldg)	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$
b. Drywell Pressure - High	$\leq 10^{(a)}$
c. Fuel Building Ventilation Exhaust Radiation - High(b)	NA
d. Reactor Building Annulus Ventilation Exhaust Radiation - High(b)	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	$\leq 10^{(a)}$##
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low Low Level 2	$\leq 10^{(a)}$
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	$\leq 10^{(a)}$###
b. RCIC Steam Line Flow-High Timer	NA
c. RCIC Steam Supply Pressure - Low	$\leq 10^{(a)}$
d. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
e. RCIC Equipment Room Ambient Temperature - High	NA
f. RCIC Equipment Room Δ Temperature - High	NA
g. Main Steam Line Tunnel Ambient Temperature - High	NA
h. Main Steam Line Tunnel Δ Temperature - High	NA

TABLE 3.3.7.4-2 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

	MINIMUM CHANNELS OPERABLE RSP1	RSP2
22. RHR Shutdown Cooling MOV (1E12*MOVF006A, 6B)	2[a]	NA
23. RHR Outboard Shutdown Isolation MOV (1E12*MOVF008)	1	NA
24. RHR Inboard Shutdown Isolation MOV (1E12*MOVF009)	1	NA
25. RHR Hx Flow to Suppression Pool MOV (1E12*MOVF011A, B)	1	1
26. RHR Reactor Head Spray MOV (1E12*MOVF023)	1	NA
27. RHR Test Line MOV (1E12*MOVF024A, B)	1	1
28. Deleted		
29. RHR Injection Shutoff MOV (1E12*MOVF027A, B)	1	1
30. RHR Upper Pool Cooling Shutoff MOV (1E12*MOVF037A, B)	1	1
31. RHR Injection MOV (1E12*MOVF042A, B, C)	1	2[a]
32. RHR Hx Shell Side Inlet MOV (1E12*MOVF047A, B)	1	1
33. RHR Hx Shell Side Bypass MOV (1E12*MOVF048A, B)	1	1
34. RHR Discharge to Radwaste MOV (1E12*MOVF040)	1	NA
35. Deleted		
36. RHR Injection MOV (1E12*MOVF053A, B)	1	1
37. RHR Pump Minimum Flow MOV (1E12*MOVF064A, B, C)	1	2[a]
38. RHR Hx Water Discharge MOV (1E12*MOVF068A, B)	1	1
39. Safety Relief Valves (1B21*RVF051C, G, D)	3[a]	3[a]
40. SSW Pump (1SWP*P2A, 2C,[b] 2B, 2D)	1	2[a]
41. Normal Service Water Isolation MOV (1SWP*MOV96A, B)	1	1
42. SSW Cooling Tower Inlet MOV (1SWP*MOV55A, B)	1	1
43. SSW Component Cooling Water Inlet MOV (1SWP*MOV510A, B)	1	1
44. SSW Component Cooling Water Outlet MOV (1SWP*MOV504A, B)	1	1

(a) One per control equipment.
(b) SSW pump 1SWP*P2C is provided on panel 1EGS*PNL4C.

TABLE 4.3.7.4-1

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Vessel Pressure		M	R
2. Reactor Vessel Water Level		M	R
3. Safety/Relief Valve Demand Position		M	NA
4. Suppression Pool Water Level		M	R
5. Suppression Pool Water Temperature		M	R
6. Drywell Pressure		M	R
7. Drywell Temperature		M	R
8. RHR System Flow:	Loop A	M	R
	Loop B	M	R
	Loop C	M	R
9. RHR Hx Cooling Water System Flow:	Loop A	M	R
	Loop B	M	R
10. RCIC System Flow		M	R
11. RCIC Turbine Speed		M	R

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 52 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

INTRODUCTION

By letters dated August 22 and October 17, 1990, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station (RBS), Unit 1. The proposed amendment would revise the Technical Specifications (TSs) regarding operation in the steam condensing mode (SCM) of the residual heat removal (RHR) system. Currently, License Condition C.5.a to Facility Operating License No. NPF-47 requires NRC written approval prior to use of the SCM of RHR. This license condition was agreed upon as a result of concerns which were raised regarding loads on the suppression pool wall from operation of an RHR heat exchanger relief valve when in the SCM of operation. GSU has determined the SCM of RHR is not, however, required for the safe operation of the RBS and plans on permanently disabling the SCM of RHR. Therefore, License Condition C.5.a is no longer applicable. As a result of disabling the SCM, GSU has requested changes to TSs 3/4.3.7.4 and 3/4.3.2 to delete maintenance and surveillance requirements for three disabled valves located on the remote shutdown panel and to establish a final trip setpoint and allowable value for the High RHR/Reactor Coolant Isolation Cooling (RCIC) steam line flow for RCIC isolation.

The SCM of RHR is used when the reactor is isolated from its primary heat sink, the main condenser. The SCM of RHR is used in conjunction with the RCIC system to remove decay heat and minimize the makeup water requirements. The SCM of RHR draws reactor steam through the combined RCIC turbine/RHR heat exchanger steam supply line to the RHR heat exchangers which condense the steam. The condensate from the heat exchangers is forced by heat exchanger pressure to the suction of the RCIC pump which then returns the condensate to the reactor vessel via the RCIC system. The SCM is designed to be placed in service by the operator. Other decay heat removal systems which could be used in place of the SCM of RHR include the main safety/relief valves (SRVs) and the suppression pool with the shutdown cooling mode of RHR or the suppression pool cooling mode of RHR.

EVALUATION

In support of disabling the SCM of RHR, GSU had planned to weld one plug in each of the two steam supply lines to the RHR heat exchangers. However, GSU identified a radiation source in the area of the proposed welding which would yield a significant exposure to the workers. The licensee re-examined the

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modification and determined that removing the normally closed RHR steam supply valves, 1E12*MOVF052A and B, and installing a bolted blind flange in each valve would accomplish the same result, blocking the steam supply lines, but with significantly less exposure to the workers. The new approach to the modification was discussed in the October 17, 1990, letter.

The blind flanges will be ASME III, Division I qualified and will meet the same quality requirements as the piping in which they are installed. The flanges will be installed in the RHR steam supply lines in the auxiliary building. The licensee reviewed steam line pipe supports and the seismic analysis to ensure the decrease in weight due to the removal of the valves and the addition of the blind flanges would not impact the seismic response of the system. The licensee determined the seismic analysis remains unchanged. Additionally, the flanges are located on negatively sloped lines which will ensure the pipes, which are designed for steam, will not fill with water. The proposed installation of the flanges does not impact the high energy line break analysis in the Safety Analysis Report (SAR). A high energy line break in the steam tunnel and the auxiliary building is the only accident involving the SCM of RHR that was analyzed in the SAR.

In addition to removing the two steam supply line valves, GSU plans on electrically or pneumatically disabling six valves in the SCM flow path and removing the associated control switches from the panels in the control room. Three of the valves (1E12*MOVF052A and B and 1E12*MOVF026A) form a high/low pressure interface and must not spuriously reposition during a fire event. Spurious repositioning of the valves could result in an interfacing intersystem loss-of-coolant accident (LOCA). With the removal of the valves and the installation of the blind flanges, the high/low pressure interface is now the blind flange, and cannot cause an interfacing system LOCA. Valve 1E12*MOVF026A will be closed and electrically disconnected thus eliminating spurious repositioning during a fire event. Therefore, these three valves no longer require controls on the remote shutdown panel. Table 3.3.7.4-2, "Remote Shutdown System Controls" includes these three valves. With the controls to these valves disconnected, the valves may be deleted from the table.

TS Table 3.3.2-2, Item 5.1 lists the trip setpoint and allowable value for the RHR/RCIC High Steam Line Flow RCIC isolation. A footnote to that item indicates the values are initial and that final values will be determined during testing prior to operation of the SCM. The licensee proposes deletion of the footnote due to the disabling of the SCM and proposes the existing values of 60.7 inches water for the trip setpoint and 64.2 inches water for the allowable value become the final values. The difference between the allowable value and the setpoint allows instrument drift and instrument and calibration inaccuracies.

The licensee reevaluated the initial trip setpoint value to determine whether the value needed to be changed. The analytical limit for the setpoint is 125 percent of the maximum normal steam flow through the steam supply line when the SCM of RHR is in operation. This value is approximately 183,200 pounds mass per hour (lbsm/hr). The initial trip setpoint of 60.7 inches water corresponds to a steam flow rate of approximately 216,080 lbsm/hr.

A review of the mass and energy release calculations show that the initial setpoint will be exceeded approximately 0.1 second after a break in the 4-inch steam line leading to the RCIC turbine and the line will be isolated by closure of the containment isolation valves approximately 12 seconds after the break. Lowering the setpoint would not significantly decrease the amount of the time before the setpoint is exceeded nor would a lower setpoint significantly decrease the amount of time before the break is isolated by closure of the containment isolation valves. Additionally, all equipment is qualified based on the initial setpoint and no increase in radiological consequences would result by allowing the initial setpoint to become the final setpoint. Therefore, the initial setpoint and allowable value are adequate and should become the final values.

A review of the accident analyses in the USAR was also performed. This was done to ensure that no credit was taken for the SCM of RHR to prevent or mitigate the consequences of an accident. Two of the analyses, closure of one main steam isolation valve (MSIV) and loss of normal and preferred station service transformers, assume the operator places the SCM into operation, but no credit for this action is taken. If the SCM is disabled, reactor pressure would increase after an MSIV isolation and the number of main steam SRV cycles may be affected. The licensee reperformed the analysis assuming the SCM is unavailable and found the number of cycles would be 15. The current number of cycles used as input to the containment fatigue analysis is also 15. Therefore, the containment fatigue analysis is unaffected by disabling the SCM of RHR. Additionally, the licensee reviewed the radiological consequences of an MSIV isolation. The radiological consequences calculation did not take credit for the SCM of RHR; therefore, elimination of the SCM will not affect the results.

Based on the staff's review of the licensee's submittal, the installation of the blind flanges in the steam supply lines to the RHR heat exchanger, the disabling of the associated SCM valves, and the isolation actuation instrumentation values proposed by the licensee do not affect current piping analyses or accident analyses. Therefore, the proposed plant modifications and TS changes are acceptable.

ENVIRONMENTAL CONSIDERATION

The amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposures. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff therefore concludes that the proposed changes are acceptable.

Dated: November 23, 1990

Principal Contributor: Claudia M. Abbate, PDIV-2