October 9, 1986

Docket No. 50-458

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Mr. William J. Cahill, Jr.
Senior Vice President
River Bend Nuclear Group
Gulf States Utilities Company
Post Office Box 220
St. Francisville, Louisiana 70775
ATTN: Nuclear Licensing Department-MA-Z

Dear Mr. Cahill:

SUBJECT: CHANGE TO CONTAINMENT ISOLATION VALVE TABLE

RE: RIVER BEND STATION, UNIT 1

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 2 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment is in response to your letter dated September 17, 1986 and supplemented by letter dated September 19, 1986. Facility Operating License NPF-47 is amended by changing Table 3.6.4-1, Item a-1 of River Bend Technical Specification 3.6.4 by the addition of a footnote, applicable to isolation valve MOVF076, that would exempt the operability requirements for this valve until October 4, 1986.

This amendment was authorized by telephone on September 19, 1986 and confirmed by letter on September 19, 1986.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's bi-weekly <u>Federal Register</u> notice.

Sincerely,

Original signed by
Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

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

cc: w/enclosures
See next page

PD#4/LA	PD#4/PM	FOB/D	OGC-B	PD#4/D
MO'Brien	SStern:ca	DVassallo	~~Goldberg~~	WButler
9/26/86	9/26/86	9/26/86	9/30/86	9/26/86

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

October 9, 1986

Docket No. 50-458

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Senior Vice President
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Mr. William J. Cahill, Jr.
Gulf States Utilities Company

River Bend Nuclear Plant

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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. 2
License No. NPF-47

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

   A. The application for amendment filed by Gulf States Utilities Company, dated September 17, 1986 and supplemented by letter dated September 19, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the 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 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.

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. 2 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. This license amendment became effective September 19, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 9, 1986

No concurrence
see Note 10/1/86

PD#4/LA	PD#4/PM	~~OGC~~	PD#4/D
MO'Brien	SStern	~~Goldberg~~	WButler
9/26/86	9/26/86	~~9/30/86~~	9/26/86

3. This license amendment became effective September 19, 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
BWR Project Directorate No. 4
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 9, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 2
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. Overleaf pages provided to maintain document completeness.*

REMOVE	INSERT
3/4 6-35	3/4 6-35
3/4 6-36	3/4 6-36*
3/4 6-47	3/4 6-47
3/4 6-48	3/4 6-48*

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP[l]	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. **Automatic Isolation Valves**					
1. **Primary Containment**[a] (Continued)					
RHR & RCIC Steam Sup.	1E51*MOVF063[b]	1KJB*Z15	2	9.9	No
RHR & RCIC Steam Sup.	1E51*MOVF076[b](m)	1KJB*Z15	2	13.4	No
RHR & RCIC Steam Sup.	1E51*MOVF064	1KJB*Z15	2	9.9	No
RCIC Pump Suc.-Supp. Pool	1E51*MOVF031[j]	1KJB*Z16	2	30.5	No
RCIC Turbine Exh.-Supp. Pool	1E51*MOVF077	1KJB*Z17	3	14.2	No
RCIC Turbine Exh. Vac. Bkrs.	1E51*MOVF078	1KJB*Z18B,C	3	16.5	No
Cont./Drywell Purge Sup.	1HVR*AOV165	1KJB*Z31	8	3	No
Cont./Drywell Purge Sup.	1HVR*AOV123	1KJB*Z31	8	3	No
Cont./Drywell Purge Outlet	1HVR*AOV128	1KJB*Z33	8	3	No
Cont./Drywell Purge Outlet	1HVR*AOV166	1KJB*Z33	8	3	No
Post-Accident Samp. Sup.	1SSR*SOV130	1KJB*Z601B	10	3	No
Post-Accident Samp. Sup.	1SSR*SOV131	1KJB*Z601B	10	3	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM	VALVE NUMBER	PENETRATION NUMBER	VALVE GROUP(l)	MAXIMUM ISOLATION TIME (Seconds)	SECONDARY CONTAINMENT BYPASS PATH (Yes/No)
a. Automatic Isolation Valves					
2. Drywell(k)					
Cont./Drywell Purge Sup.	1HVR*AOV147	1DRB*Z32	1	3	No
RPCCW Supply	1CCP*MOV142	1DRB*Z50	1	30	No
RPCCW Return	1CCP*MOV144	1DRB*Z51	1	30	No
RPCCW Return	1CCP*MOV143	1DRB*Z51	1	30	No
Service Water Supply	1SWP*MOV4A	1DRB*Z54	1	52.8	No
Service Water Supply	1SWP*MOV4B	1DRB*Z54	1	51.7	No
Service Water Return	1SWP*MOV5A	1DRB*Z55	1	50.6	No
Service Water Return	1SWP*MOV5B	1DRB*Z55	1	53.9	No
Recirc. Flow Control	1RCS*MOV58A	1DRB*Z152	1	11.0	No
Recirc. Flow Control	1RCS*MOV59A	1DRB*Z153	1	10.6	No
Recirc. Flow Control	1RCS*MOV60A	1DRB*Z154	1	6.3	No
Recirc. Flow Control	1RCS*MOV61A	1DRB*Z155	1	8.6	No
Recirc. Flow Control	1RCS*MOV58B	1DRB*Z156	1	10.6	No
Recirc. Flow Control	1RCS*MOV59B	1DRB*Z157	1	10.8	No
Recirc. Flow Control	1RCS*MOV60B	1DRB*Z158	1	6.38	No
Recirc. Flow Control	1RCS*MOV61B	1DRB*Z159	1	8.9	No
Cont./Drywell Purge Sup.	1HVR*AOV125	1DRB*Z32	1	3	No
Cont./Drywell Purge Rtn.	1HVR*AOV126	1DRB*Z34	1	3	No
Cont./Drywell Purge Rtn.	1HVR*AOV148	1DRB*Z34	1	3	No

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

NOTES

(a) Subject to a Type C leak rate test at a test pressure of 7.6 psig except as otherwise noted.

(b) Also isolates the drywell.

(c) Testable check valve.

(d) Isolates on MS-PLCS air line high flow or MS-PLCS air line header to Main Steam Line low differential pressure.

(e) Receives a remote manual isolation signal.

(f) This line is sealed by the penetration valve leakage control system (PVLCS). The combined leakage from valves sealed by the PVLCS is not included in 0.60 La Type B and C test total.

(g) This valve sealed by the main steam positive leakage control system (MS-PLCS). Valves sealed by the MS-PLCS are tested in accordance with Surveillance Requirement 4.6.1.3.f to verify that leakage does not exceed the limit specified in Specification 3.6.1.3.c. This leakage is not included in the 0.60 La Type B and C test total.

(h) Not subject to Type C leakage tests. Valve(s) will be included in the Type A test.

(j) Valve is hydrostatically leak tested at a test pressure of 8.36 psig (1.1 Pa). The leakage from hydrostatically tested valves is not included in the 0.60 La Type B and C test total.

(k) Not subject to a Type A, B, or C leak rate test.

(l) Valve groups listed are designated in Table 3.3.2-1.

(m) Valve 1E51*MOVF076 is not required to be OPERABLE through October 4, 1986.

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY - OPERATING restore SECONDARY CONTAINMENT INTEGRITY - OPERATING within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY - OPERATING shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressures within the Shield Building annulus, the Auxiliary Building and the Fuel Building are less than or equal to 3.0, 0.00, and 0.00 inches of vacuum water gauge, respectively.

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatch covers are installed.

2. The door in each access to the secondary containment is closed, except during normal entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers/valves secured in position.

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

SAFETY EVALUATION REPORT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 2 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

In River Bend Technical Specification 3.6.4, Table 3.6.4-1, Item a.1 identifies the reactor core isolation cooling (RCIC) steam supply valve 1E51*MOVF076 (MOVF076) as a primary containment automatic isolation valve. By letters dated September 17, 1986 and September 19, 1986, the Gulf States Utilities Company (GSU) requested a one-time change to the Table by adding a footnote, applicable to isolation valve MOVF076, that would exempt the operability requirements for this valve from September 22, 1986 through October 4, 1986.

While the reactor was at 100% power on September 8, 1986 the MOVF076 valve was undergoing a functional test of the annunciator when the MOV76 breaker tripped due to a ground in the system. A subsequent attempt to stroke the valve indicated the valve to be fully open. As required by the River Bend Technical Specification 3.6.4 Action Statements, the outboard isolation valve 1E51*MOVF064 (MOVF064) was closed. Consequently, the RCIC system was rendered inoperable and by Specification 3.7.3, the plant was placed in a 14-day Action Statement that requires the plant to initiate a shutdown on September 22, 1986.

The GSU proposed technical specification revision will exempt valve MOVF076 from technical specification operability requirements, thereby allowing isolation valve F064 to be opened and returning the RCIC system to operable status. This request would be a one-time modification to the Technical Specification effective through October 4, 1986, at which time GSU has scheduled a shutdown and will repair the inoperable isolation valve (MOVF076).

2.0 EVALUATION

The RCIC steam turbine supply line is an 8-inch line that branches off the main steamline "A". Inside the River Bend drywell the 8-inch line has an inboard isolation valve 1E51*MOVF063 (MOVF063) and circumventing this valve is a 3/4 inch bypass line consisting of a manual isolation valve and the motor-operated isolation valve MOVF076. The 3/4-inch bypass line is parallel to the 8-inch inboard isolation valve; its functions during start-up are to equalize the pressure around MOV63 and to warmup the RCIC steamline. The common out-board isolation valve (8-inch) is MOVF064. During power operation, the normal RCIC turbine steam supply line valve lineup is MOVF076 closed and inboard and outboard isolation valves MOVF063 and MOVF064 open.

The RCIC is a high pressure system primarily designed to maintain sufficient water in the reactor pressure vessel to cool the core and to maintain the reactor in the standby condition in the event the vessel becomes isolated and feedwater is not available. During a LOCA, RCIC (which augments other core injection systems) initiates on low vessel water level and delivers rated flow to the vessel through a connection in the vessel head.

Even though RCIC is not a part of the ECCS network, the RCIC system is designated as a safety-related system. During limiting conditions of operation (LCO) (i.e., when HPCS is inoperable), power operation is allowed to continue for a period of time provided RCIC is operable.

Valve MOVF076 is only used for RCIC turbine steam supply line warm-up before start-up of the system and the ability to close the valve is not required for the RCIC system to perform its intended design function. Therefore, in the open position MOVF076 will not affect system operation.

The RCIC steam line isolation valves MOVF063 and MOVF064 are normally kept open and should be kept open in order to keep the steam inlet lines warmed-up and to minimize the potential for water hammer upon system startup. Accordingly, the RCIC steam supply line does not receive the standard containment isolation signals (e.g., high drywell pressure and low reactor water level). The isolation signals that serve the inboard and outboard isolation valve monitor conditions to reveal a break in the RCIC system.

The applicant in a letter dated September 19, 1986, also submitted a radiation dose assessment. In the event of a pipe break downstream of the outboard isolation valve (MOVF064), the leakage will be detected by the leakage detection systems. The leakage detection systems are the area temperature monitoring system and the RCIC steam line flow monitoring system. Either leakage detection system will provide an isolation signal to MOVF063 and MOVF064. Once a signal is received, a total of a 20-second (a 10-second delay for diesel start plus a 10-second delay for valve closure) delay is assumed for closing of the valve. Full flow is assumed through the 8-inch line for the first 20 seconds. At the end of 20 seconds, the inboard isolation valve F063 is assumed to close, but the outboard isolation valve fails to close (single failure). With the inoperable valve, F076, in the failed open position and the outboard F064 assumed to be failed open, there will be an unisolated leakage path (3/4-inch line) from the reactor to the auxiliary building. The maximum expected flow through the bypass (F076) is 30 GPM. Feedwater flow could provide make-up for this fluid loss. If feedwater is not available, the

High Pressure Core Spray (HPCS) system could be used to provide cooling water to the vessel, thereby preventing uncovering of the reactor fuel. Fuel failure would, therefore, be precluded. The only radioactivity available for release from the break is that which is present in the reactor coolant prior to the break. The bypass flow is assumed to be constant for the first 30 minutes of the transient and then is ramped down to zero in the next 3.5 hours as the reactor is brought down to a cold shutdown condition. The total mass released from the RCIC steam line break is estimated to be about 50,000 lbs compared with the 140,000 lbs estimated by the staff for Main Steam Line Break Accident Analysis. In section 15.6.4 of the River Bend SER, we concluded that the doses for Main Steam Line Break are only a small fraction of 10 CFR Part 100 Guideline values. Since the RCIC piping break with unisolated bypass flow through the F076 valve is bounded by the Main Steam Line Break Analysis, the expected doses for the RCIC break also will be a small fraction of 10 CFR Part 100 guideline values, and hence within our acceptance criteria.

The staff concurs with GSU's assessment that there is no significant reduction in the margin of safety relating to the temporary period (i.e., about 12 days) of having the MOVF076 valve in an inoperable condition. The alternative approach would require reactor shutdown in order to enter the drywell and subject plant personnel to exposures from the harsh drywell environment. Therefore, such an action would induce unwarranted and inherent challenges to various plant systems and to the affected plant personnel.

3.0 BASIS FOR EMERGENCY CIRCUMSTANCES

The Commission published guidance concerning emergency circumstances in the Federal Register on March 6, 1986 (51 FR 7762) which states, in part: "The staff may receive an amendment request and find an emergency situation, where failure to act in a timely way would result in derating or shutdown of a nuclear power plant...(NRC) may proceed to issue the license amendment, if it determines, among other things, that no significant hazards considerations are involved." In addition, the Commission further states in 51 FR 7762, "...that it expects its licensees to apply for license amendments in a timely fashion..."

On September 8, 1986, while the licensee was performing a functional test on annunciator 1E51*MOVF076, the MOV76 breaker tripped with the valve open. MOV76 is a 3/4" bypass line valve used for prewarming the RCIC steam line. With this valve open RCIC was declared inoperable and the licensee entered into a 14-day LCO within which time the licensee had to restore RCIC to operable status or shutdown.

The licensee determined that the breaker malfunction was due to an unintended electrical ground, apparently due to a wet valve motor. Since the valve is inside the containment, the licensee attempted to close the valve by remote means. By September 15, 1986, the licensee concluded that the valve could not be closed remotely.

The licensee had three basic options:

(1) Derate to a very low power level, enter containment and open/repair the valve
(2) Shutdown
(3) Request an amendment to their operating license to allow continued operations.

The staff finds that denial of the licensee's request for an amendment to their operating license would have placed the reactor in a shutdown or derated condition. Moreover, the licensee could not have foreseen this need for a license amendment in advance because it involved a component failure. Furthermore, even had the licensee requested an amendment of the license on the date the event occurred (September 8, 1986), the 14-day LCO would have required NRC to issue the amendment on an emergency basis.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The licensee has provided an analysis of significant hazards considerations in its September 17, 1986 request which was amplified by their letter of September 19, 1986 for a license amendment. The licensee has concluded, with appropriate bases, that the proposed amendment meets the three standards in 10 CFR 50.92 and, therefore, involves no significant hazards considerations. The Commission has also provided guidance concerning the application of these standards by providing examples of amendments considered likely, and not likely, to involve a significant hazards consideration. These were published in the Federal Register on March 6, 1986 (51 FR 7744). The NRC staff has made a preliminary review of the licensee's submittal. A discussion of these examples as they relate to the proposed amendment follows.

One of the examples of actions involving no significant hazards consideration (vi) is a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the standard review plan.

The proposed amendment to the Technical Specifications may involve some increase, but in our judgment not a significant increase in the probability or consequences of an accident previously evaluated because the potentially unisolated penetration is limited to a 3/4" line discharging to a closed system that returns to the containment or assuming a break in the RCIC steam line between valve MOV64 and MOV45, 10CFR Part 100 limits will not be exceeded. This conclusion is based upon the fact that the fuel failure is not expected because the make-up water from the fluid loss can be provided by the Feedwater System or the High Pressure Core Spray system. The only activity available for release from the break is that which is present in the reactor coolant and steam lines prior to the break. Moreover, large break LOCA (Double-Ended Recirculation Line Break) with fuel failure assumes a closed loop in the RCIC system and filtration by the Standby Gas Treatment System for any valve packing leaks and meets the current accident analysis and 10CFR Part 100 limits.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because line breaks such as may be postulated for the RCIC system have been considered in the analysis and design basis with many of the previous analyses bounding all the short-term consequences of the plant for this event. There may, however, be longer term environmental effects which would only impact the overall lifespan of the equipment and would be evaluated if such an event should occur. This change introduces no new mode of operation and no change in the plant design is being made.

The proposed amendment does not involve a significant reduction in the margin of safety because this event is bounded by a previous analysis and design basis. Therefore the margin of safety remains the same.

5.0 STATE CONSULTATIONS

In accordance with the Commission's regulations, consultation was held with officials of the State of Louisiana by telephone on September 19, 1986. The Manager, Nuclear Projects of the Nuclear Energy Division, Office of Air Quality and Nuclear Energy, Department of Environmental Quality of the State of Louisiana, had no objections to the proposed action.

6.0 PUBLIC COMMENTS

None

7.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to 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. 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 exposure. The Commission has made a final no significant hazards consideration finding with respect to this amendment. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 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 this amendment.

8.0 CONCLUSION

The staff has concluded, based on the considerations discussed above that: (1) the amendment does not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) create the possibility of a new or different kind of accident from any previously evaluated or (c) significantly reduce a safety margin and, therefore, the amendment does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) 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 the security or to the health and safety of the public.

Principal Contributors: A. Notafrancesco
C. Schulten
S. Stern
G. Thomas

Dated: October 9, 1986