August 3, 1987

Docket No. 50-458

Mr. James C. Deddens
Senior Vice President, (RBNG)
Gulf States Utilities
P. O. Box 220
St. Francisville, LA 70775
ATTN: Nuclear Licensing

DISTRIBUTION
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NRC PDR
LPDR
PD4 Rdg.
FSchroeder
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EJordan
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EButcher
ACRS (10)
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GPA/PA
Plant File

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1, AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 65129)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 7 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated March 10, 1987 as supplemented June 9, June 30, July 8, 23, and 27, 1987. Your June 30, 1987 letter withdrew the TS change requests as specified in Attachments 1 and 3 to the March 10, 1987 submittal.

This amendment authorizes one-time extensions to the surveillance intervals for drywell bypass leakage testing and leakage testing of 52 isolation valves and two air systems of the drywell airlock/equipment hatch until the first refueling outage scheduled to begin September 15, 1987.

In connection with this amendment, the Commission has granted an exemption from certain requirements of Section III.D.3 of Appendix J to 10 CFR Part 50 to the extent that the 24-month interval for performing Type C tests of five containment isolation valves may be extended until the first refueling outage scheduled to begin September 15, 1987.

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

Sincerely,

/s/

Walter A. Paulson, Project Manager
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 7 to License No. NPF-47
2. Exemption
3. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

*PD4/LA	*PD4/PM	*OGC-Bethesda	PD#4/D	DR4A
PNoonan	WPaulson:sr		JCalvo	FSchroeder
7/23/87	7/23/87	7/24/87	7/30/87	7/31/87

DRSP
DCrutchfield
7/31/87

Docket No. 50-458

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7/23/87	7/23/87	7/24/87	/ /87	/ /87

DRSP:D
DCrutchfield
/ /87

Mr. James C. Deddens
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. 7
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 March 10, 1987 as supplemented June 9, June 30, July 8, 23, and 27, 1987, 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 I;

   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.7 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 is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 7

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 the captioned Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

REMOVE	INSERT
3/4 4-12	3/4 4-12
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-18	3/4 6-18
3/4 6-19	3/4 6-19
3/4 6-21	3/4 6-21

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage (averaged over any 24-hour period).

d. 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm leakage at a reactor coolant system pressure of 1025 ± 15 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual, deactivated automatic or check* valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours. The provisions of Specification 3.0.4 are not applicable.

* Which have been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate radioactivity at least once per 12 hours,

b. Monitoring the sump flow rates at least once per 12 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 including paragraph IWV-3427(B) of the ASME Code and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months*, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

---

*This test may be performed during the refueling outage following the first cycle, scheduled to begin September 15, 1987.

b. If any periodic Type A test fails to meet 0.75 La, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 La, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 La, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental test data and the Type A test data is within 0.25 La. The formula to be used is: $[Lo + Lam - 0.25\ La] \leq Lc \leq [Lo + Lam + 0.25\ La]$ where Lc = supplemental test results; Lo = superimposed leakage; Lam = measured Type A leakage.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas, injected into the primary containment or bled from the primary containment during the supplemental test, to be between 0.75 La and 1.25 La.

d. Type B and C tests shall be conducted with gas at Pa, 7.6 psig*, at intervals no greater than 24 months** except for tests involving:

1. Air locks,

2. Main steam positive leakage control system (MS-PLCS) valves and PVLCS valves,

3. Penetrations using continuous leakage monitoring systems,

4. Primary containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, and

5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.4.

f. Total sealing air leakage into the primary containment, at a test pressure of 11.5 psid for MS-PLCS valves and 33 psid for penetration leakage control system sealed valves, shall be determined by test at least once per 18 months.** This leakage may be excluded when determining the combined leakage rate, 0.6 La.

* Unless a hydrostatic test is required per Table 3.6.4-1.

**This test may be performed during the refueling outage following the first cycle, scheduled to begin September 15, 1987.

SURVEILLANCE REQUIREMENTS (Continued)

g. Type B tests for electrical penetrations employing a continuous leakage monitoring system shall be conducted at Pa, 7.6 psig, at intervals no greater than once per 3 years.

h. Leakage from isolation valves that are sealed with the PVLCS shall be tested once per 24 months with the valves pressurized to at least Pa, 7.6 psig. This leakage may be excluded when determining the combined leakage rate, 0.6 La.

i. Primary containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment shall be leak tested at least once per 18 months*.

j. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.9.3.

k. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.3.a, 4.6.1.3.d, 4.6.1.3.g, and 4.6.1.3.h.

*This test may be performed during the refueling outage following the first cycle, scheduled to begin September 15, 1987.

CONTAINMENT SYSTEMS

3/4.6.2 DRYWELL

DRYWELL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.1 DRYWELL INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

Without DRYWELL INTEGRITY, restore DRYWELL INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 DRYWELL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all drywell penetrations**, not capable of being closed by OPERABLE drywell automatic isolation valves and required to be closed during accident conditions, are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Specification 3.6.4.

b. By verifying the drywell air lock is in compliance with the requirements of Specification 3.6.2.3.

c. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

d. By verifying the drywell bypass leakage is in compliance with the requirements of Specification 3.6.2.2.

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the drywell or primary containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN, except such verification need not be performed more often than once per 92 days.

SURVEILLANCE REQUIREMENTS (Continued)

e. By verifying the personnel door inflatable seal system OPERABLE by:

1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 75 psig.

2. At least once per 18 months* conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 0.67 psig from 75 psig within 24 hours.

*This test may be performed during the refueling outage following the first cycle, scheduled to begin September 15, 1987.

CONTAINMENT SYSTEMS

DRYWELL BYPASS LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.2.2 Drywell bypass leakage shall be less than or equal to 10% of the acceptable $A/\sqrt{k}$ design value of 1.0 ft^2.

APPLICABILITY: When DRYWELL INTEGRITY is required per Specification 3.6.2.1.

ACTION:

With the drywell bypass leakage greater than 10% of the acceptable $A/\sqrt{k}$ design value of 1.0 ft^2, restore the drywell bypass leakage to within the limit prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.2.2 At least once per 18 months*, the drywell bypass leakage rate test shall be conducted at an initial differential pressure of 3.0 psid and the $A/\sqrt{k}$ shall be calculated from the measured leakage. One drywell airlock door shall remain open during the drywell leakage test such that each drywell door is leak tested during at least every other leakage rate test.

a. If any drywell bypass leakage test fails to meet the specified limit, the schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the limit, a test shall be performed at least every 9 months until two consecutive tests meet the limit, at which time the 18 month test schedule may be resumed.

*For the first cycle only, this may be extended to coincide with the refueling outage, scheduled to begin September 15, 1987.

DRYWELL AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.2.3 The drywell air lock shall be OPERABLE with:

a. Both doors closed except that, when the air lock is being used for normal transit entry and exit through the drywell, at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 11.85 scf per hour at 3.0 psid, and

c. The inflatable seal system air flask pressure $\geq$ 75 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2*, and 3.

ACTION:

a. With one drywell air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. The provisions of Specification 3.0.4 are not applicable.

b. With the drywell air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one inoperable drywell air lock door inflatable seal system air flask pressure instrumentation channel, restore the inoperable channel to OPERABLE status within 7 days or verify air flask pressure to be $\geq$ 75 psig at least once per 12 hours.

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS

4.6.2.3 The drywell air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 4.05 scf per hour when the gap between the door seals is pressurized to 3.0 psid.

b.* By conducting an overall air lock leakage test at 3.0 psid and verifying that the overall air lock leakage rate is within its limit:

1. Each cold shutdown if not performed within the previous 6 months#.

2. Prior to establishing DRYWELL INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.

c. By verifying, prior to drywell entry if not performed within the past 18 months, that only one door in the air lock can be opened at a time.

d. By verifying the door inflatable seal system OPERABLE by:

1. At least once per 7 days verifying seal air flask pressure to be greater than or equal to 75 psig.

2. At least once per 18 months* conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 0.67 psig from 75 psig within 24 hours.

#At least once per 18 months, the air lock will be pressurized to 19.2 psid prior to conducting the overall air lock test.

*This test may be performed during the refueling outage following the first cycle scheduled to begin September 15, 1987.

CONTAINMENT SYSTEMS

DRYWELL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.2.4 The structural integrity of the drywell shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.2.4.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the drywell not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.4.1 The structural integrity of the exposed accessible interior and exterior surfaces of the drywell shall be determined, during the shutdown for each Type A containment leakage rate test, by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of Gulf States Utilities Company Cajun Electric Power Cooperative (River Bend Station, Unit 1)	))))))	Docket No. 50-458

EXEMPTION

I.

The Gulf States Utilities Company, et al. (the licensee), is the holder of Facility Operating License No. NPF-47 which authorizes operation of the River Bend Station, Unit 1, at reactor core power levels not in excess of 2894 megawatts thermal. The facility is a boiling water reactor located at the licensee's site in West Feliciana Parish, Louisiana. The license provides, among other things, that the facility is subject to all rules, regulations and orders of the Commission now or hereafter in effect.

II.

Paragraphs III.C.3 and III.D.3 of Appendix J to 10 CFR Part 50, require that containment isolation valves, which may provide a pathway for leakage of containment atmosphere, be tested on at least a 24-month frequency for comparison with the limiting value of 0.6 L_a for Type B and Type C tests.

The Gulf States Utilities Company proposed a one-time extension to this 24-month surveillance interval for conducting Type C tests on 5 containment isolation valves. The current testing interval is to be extended until the first refueling outage, which is scheduled to begin on September 15, 1987. The extensions requested for leak testing these valves vary from 5 days to 29 days as shown in the following table:

VALVE	EXTENSION DAYS	DESCRIPTION/SYSTEM	(inch) SIZE
G33*MOVF004	29	RWCU Pump Suction	6
G33*MOVF054	29	RWCU Pump Discharge	4
C11*VF122	16	CRD Supply to Containment	2
SWP*MOV507A	05	SW Supply	12
SWP*V174	05	SW Supply	12

The staff has found that approval of the proposed extension is warranted and that the proposed extension should be authorized by the granting of this one-time exemption so that the River Bend Station, Unit 1, may continue to operate until shutdown for the first refueling outage.

III.

The NRC has evaluated the licensee's basis for requesting the extension in the surveillance interval and finds that not granting this exemption would require the licensee to shut down the plant on or about August 16, 1987 to conduct the testing. The granting of this exemption is likely to result in a negligible reduction in containment integrity during the 5 to 29-day extension period. In evaluating the changes to the Technical Specifications and the associated exemption, the staff reviewed the licensee's technical justifications for the requested extension. The staff reviewed the licensee's position that a shutdown would be required to perform these tests. The staff reviewed the previous leakage test results on the specific valves subject to the request for exemption and has found that there is ample margin between the leak rate values previously measured and the limiting values in Appendix J to accommodate any additional degradation likely to occur during the period of the extension. The details of the above described review are discussed in the enclosed Safety Evaluation. Based on the above information provided by the licensee and the staff's evaluation of the licensee's submittals, the NRC staff concludes that the licensee has provided an adequate basis for the conclusion that postponing the subject local leak rate tests until the first refueling outage is likely to have little effect on containment integrity.

The Commission's regulations in 10 CFR 50.12 state that the Commission will not consider granting an exemption unless special circumstances are present.

In its letter of March 10, 1987 as supplemented June 9 and July 8, 15, and 30, 1987, the licensee addressed one of those special circumstances which is applicable to this request for exemption.

The licensee states that the special circumstances of 10 CFR 50.12 (a) (2) (v) are present in that the exemption would provide only temporary relief from the applicable regulation and became necessary because the preoperational testing was scheduled to be consistent with a projected fuel load date of April 1985. The intent of the scheduling was to allow adequate time for the first cycle of operation so as to satisfy the 24-month Type B and C testing requirements of Technical Specification 4.6.3.1.d at the first refueling outage. However, the low power license was not issued until August 29, 1985 and commercial operation occurred in June 1986. Only five valves require a surveillance extension out of a total population of 166 valves at River Bend Station that are associated with Technical Specification 4.6.3.1.d and Type C outleakage testing. The exemption is temporary because these five valves will be tested during the refueling outage scheduled to begin September 15, 1987. In addition, the licensee has committed to make a good faith effort to conduct these surveillances within the current frequency if an outage of sufficient length occurs. On June 18, 1987, the River Bend Station entered an unscheduled outage. The licensee stated that five surveillance tests, for which surveillance interval extensions had been requested, had been performed during this outage. However, the surveillance tests of the five valves for which an exemption was requested, were not performed because of considerations of ALARA, equipment availability, and test duration which would have added a significant length of time to the outage. The licensee stated that the summer months are a time of high system demand and other sources of power in the licensee's system were not available such that the River Bend Station could be maintained out of service for the extended period. Therefore, the staff concluded that special circumstances of 10 CFR 50.12(a)(2)(v) associated with this request for an exemption, have been demonstrated by the licensee. Accordingly, the NRC staff finds that operation of River Bend Station, Unit 1, during the proposed extension period is acceptable.

Therefore, the staff finds that the proposed temporary exemption from 10 CFR 50, Appendix J, Paragraph III.D.3 is acceptable.

IV.

Accordingly, the Commission has determined that, pursuant to 10 CFR 50.12, the proposed exemption is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest. Therefore, the Commission hereby grants the exemption as follows:

> "An exemption is granted from the requirement to conduct Type C testing on containment isolation valves at an interval no greater than 24 months as stated in 10 CFR 50, Appendix J, Paragraph III.D.3.. This exemption is granted for the period specified in the licensee's request for exemption dated March 10, 1987, as supplemented July 8, 1987 (from current test deadline dates which begin August 16, 1987 until the first refueling outage which is scheduled to begin on September 15, 1987) and is only applicable to five valves in the River Bend Station as listed in Section II. of this exemption."

Pursuant to 10 CFR 51.32, the Commission has determined that the issuance of the exemption will have no significant impact on the environment (52 FR 28054).

A copy of the Commission's Safety Evaluation dated July 31, 1987 related to this action is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W. Washington, DC, and at the Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803.

This Exemption is effective on August 16, 1987 and is to expire at the start of the first refueling outage.

Dated at Bethesda, Maryland, this 31day of July 1987.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Director
Division of Reactor Projects - III, IV,
V and Special Projects

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 7 TO FACILITY OPERATING LICENSE NO. NPF-47

AND EXEMPTION FROM APPENDIX J OF 10 CFR PART 50

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated March 10, 1987, the licensee (Gulf States Utilities Company) requested a temporary amendment to the Facility Operating License NPF-47 for River Bend Station, Unit 1, in the form of Technical Specifications (TS) changes. The proposed TS changes would permit the licensee, on a one-time basis, to extend the surveillance test intervals for: (1) the drywell bypass leakage testing, (2) the hydrogen mixing system flow rate surveillance; and (3) the leakrate testing for certain isolation valves and air systems of the drywell airlock/equipment hatch beyond the test criteria specified in the TS. The licensee also requested an exemption from the test interval requirements of Appendix J to 10 CFR Part 50 (Appendix J) for certain Type C tested isolation valves.

The River Bend TS require surveillance tests to be performed every 18 months to allow power operation from one refueling outage to the next. For flexibility in operations scheduling, TS 4.0.2 also allows the 18-month test interval to be extended by 25 percent (%) when necessary. Since the start-up testing and unscheduled outages have extended the first fuel cycle of power operation, a surveillance postponement is necessary to coincide with the projected first refueling outage. The licensee stated that a forced plant shutdown solely for performing tests on certain components would cause an unnecessary thermal transient on the plant and also exposure to the workers. In accordance with 10 CFR 50.12, the licensee requested a temporary change to the TS to defer the subject tests until the first refueling outage which is scheduled to begin on September 15, 1987.

By letters dated June 6, July 8, 23, and 27, 1987, the licensee submitted additional information to support its TS change requests and by letter dated June 30, 1987 withdrew two surveillance extension requests as specified in Attachments 1 and 3 of the March 10 submittal. The licensee has committed to conduct these surveillance tests if an unscheduled outage of sufficient length occurs. The licensee's TS changes and Appendix J exemptions are itemized below with the staff's evaluation and conclusions of each item.

2.0 EVALUATION

2.1 Drywell Bypass Leakage

The surveillance requirement of TS 4.6.2.2 requires that the drywell bypass leakrate test shall be performed at least once per 18 months at a differential pressure of 3 psid between the drywell and the containment. This test is to determine the $A/\sqrt{K}$ value from the measured leakage, where K is the resistance factor of the actual flow area, A. The licensee requested a temporary amendment to TS 4.6.2.2 to extend the 18-month (plus 25% allowance permitted by TS 4.0.2) test interval by up to 3 days to coincide with the first refueling outage.

The licensee has provided information on preoperational and periodic drywell integrated leakrate test results to confirm drywell leaktightness. The acceptance criterion for these tests is that the measured leakages shall be less than or equal to 10% of the allowable drywell leakage capability ($A/\sqrt{K}$) of 1.0 square feet (sq. ft.). The preoperational test of the drywell was performed on February 10 and 11, 1985. The test results have shown that, at a design pressure of 25 psig, the measured leakage was 2425 standard cubic feet per minute (SCFM) compared to the allowable limit of 9660 SCFM (about 25%). At a test pressure of 3 psid, the measured leakage was 647 SCFM compared to the allowable limit of 4011 SCFM (about 16%). The periodic low pressure drywell bypass leakage test was performed on October 25, 1985. During the test, the $A/\sqrt{K}$ value was computed at 0.014 sq. ft. (corresponding to 14% of the allowable leakage of 0.1 sq. ft.).

Bypass leakage paths between the drywell and the containment would produce pressurization of the containment and increase containment pressure during the blowdown of loss-of-coolant accident (LOCA). The licensee has previously evaluated the allowable drywell bypass leakage for a complete spectrum of credible primary system ruptures to determine the amount of steam which could bypass the suppression pool without exceeding the containment design pressure. The staff has reviewed its previous leakrate test histories and finds the bypass leakage would have to increase several times before effecting containment design pressure. Furthermore, the volume of the MARK III containment of River Bend is about five times that of the drywell; the compression of drywell air into the containment during vent flow results in only a small rise in containment pressure.

Based on the foregoing, the staff finds that the possibility of increasing containment design pressure due to a short delay in drywell bypass leakage test should not be significant. Therefore, the staff concludes that the proposed one-time extension of the leakrate test interval for the drywell is acceptable.

2.2 Local Leak Rate Tests

The licensee requested a temporary amendment to the TS to extend the leakrate test intervals for a total of 52 isolation valves and 2 air systems in the drywell airlock/equipment hatch personnel door by up to a maximum of 41 days and also requested an exemption from Appendix J for five of these valves. The licensee has proposed to postpone the testing dates temporarily on five kinds of the local leak rates tests beyond their scheduled testing dates. These tests are itemized below.

2.2.1 Reactor Coolant System (RCS) Boundary Leakage

The licensee requested a temporary amendment to TS 4.4.3.2.2a to extend the 18-month (plus 25% allowance) test interval for the 13 valves in the reactor coolant system (RCS) boundary by 23 to 41 days. These valves cannot be tested at power.

The 13 valves are the pressure isolation valves (PIV's) in seven RCS lines. These systems are reactor core isolation cooling (RCIC) head and steam supplies, reactor water cleanup (RWCU) head supply, low pressure coolant injection (LPCI) 'A' to reactor, residual heat removal (RHR) shutdown coolant supply, LPCI 'B' and 'C' to reactor and high pressure core spray (HPCS) to reactor. The licensee has provided information on the latest leakrate tests on these valves which were performed in late 1985. The as-left leakages were 0.01 gallons-per-minute (GPM) for 2 of the 13 valves and zero for all the others. The allowable leakage for PIV tests is 0.5 GPM per nominal inch of valve size up to 5 GPM per valve at RCS pressure of 1025 ± 15 psig as addressed in the TS. In reviewing the valves test results, the staff finds that the measured leakage of these valves is well below the allowable leakage.

The General Design Criterion (GDC) 14 requires that the integrity of the reactor coolant pressure boundary (RCPB) shall be maintained at high-to-low pressure interfaces. The PIV's are any two valves in series for each interface within the RCPB which separate the high pressure RCS from the attached low pressure systems. These valves are normally closed during power operation. Periodic tests of all PIV's are required in accordance with GDC 14, 30 and 32 of 10 CFR Part 50, Appendix A. The NRC has been requiring all licensed plants to leak test PIV's individually, and address their test criteria in the TS. However, current NRC regulations or Appendix A do not define a test frequency for the PIV's.

The staff has reviewed the available information on the PIV testing. Since the licensee has shown that the PIV's have demonstrated to be a very low leakage (near zero) pathway, the staff concludes that the proposed test interval extension for the PIV's is acceptable.

2.2.2 Containment Outleakage

The licensee requested a one-time exemption from Appendix J, paragraph III.D.3, and a temporary amendment to TS 4.6.1.3d to extend the 24-month test interval for five containment isolation valves by 5 to 29 days. The five valves out of a total of 166 Type C tested valves are in the system lines of RWCU pump suction and discharge, control rod drive (CRD) supply to containment, and service water (SW) supply. The licensee has provided the latest leakrate test results for these valves. Total leakage of the 5 valves was found to be 307.5 standard cubic feet per day (SCFD). The as-left leakage for the entire scope of Type C tested valves was 2114.64 SCFD, while the allowable leakage is 3412 SCFD (0.6 La). Currently, the licensee has performed Type C tests on all those valves which can be tested at power except these five, which cannot be tested at power.

The staff has reviewed these previous leakrate test results. Based on the review of the provided information, the staff has made the following findings:

(1) The leakage for the five valves constitutes only 9% of the allowable leakage. Also, these valves do not have a significant amount of operating time since receiving a low-power license on August 29, 1985. The probability of exceeding the allowable leakage limit due to the short period extension is, therefore, believed to be low.

(2) The ample margin between the measured leakage and the allowable leakage should accommodate any degradation likely to be experienced for these valves during the extended period.

Consequently, the staff finds that the deferred testing has little safety significance. The staff, therefore, concludes that the proposed TS changes and the Appendix J exemption request are acceptable.

2.2.3 Containment Inleakage

The surveillance requirement of TS 4.6.1.3f requires that the sealing air leakage into the primary containment shall be determined by tests at least once per 18 months. The TS has required testing on all process line isolation valves sealed by main steam positive leakage control system (MS-PLCS) and penetration valve leakage control system (PVLCS). The design and operation of these leakage control systems also require testing to ensure that leakage from the associated pressurized systems into containment be limited to 50% of the containment design pressure for the 30-day period following an onsite LOCA. There are 25 valves out of a total of 39 valves which are included in the inleakage criterion which need to extend their test interval. The licensee has requested a temporary amendment to TS 4.6.1.3f to extend the 18-month (plus 25% allowance) test interval for the 25 valves by 6 to 33 days. The 25 valves do not include any main steam isolation valves (MSIV's). The MSIV's were previously tested during the October-November 1986 outage.

The MS-PLCS and PVLCS can deliver seal fluid sufficiently to assure the sealing function on the valve at a pressure of 1.1Pa for 30 days. The licensee has previously analyzed the maximum acceptable containment inleakage rate for the MS-PLCS and PVLCS to ensure these systems do not repressurize the containment to more than 50% of the containment design pressure during the 30-day period following a LOCA. The analysis has determined that the maximum acceptable containment inleakage rate from both the MS-PLCS and PVLCS is 425 standard cubic feet per hour (SCFH). As a safety margin, the TS has set up an allowable inleakage rate at 80% of the maximum acceptable inleakage rate, i.e., 340 SCFH. The licensee stated that the as-left leakage of the 25 valves from the latest test is less than one-third of the allowable inleakage of 340 SCFH. The contributing inleakage from the 25 valves to the allowable inleakage rate for the extended period is expected to be minimal. Among these valves, the majority are exposed to conditions of clean water, low flow or station filtered air systems, so minimal degradation is expected. The remaining valves are in the high pressure closed systems that account for an inleakage rate of less than 3% of the allowable leakage.

In reviewing the licensee's information, the staff finds that the possible increasing inleakage rate due to deferred testing on the 25 valves would not represent a significant threat to over-pressurize the containment. Also, sufficient diverse containment pressure control methods and emergency operating procedures enable the plant operator to prevent containment overpressurization. The staff, therefore, concludes that the proposed TS change to extend the test interval for the 25 valves is acceptable.

2.2.4 Hydro-Tested Isolation Valves

The licensee requested a temporary amendment to TS 4.6.1.3i to extend the 18-month (plus 25% allowance) test interval for nine hydro-tested valves by 32 to 41 days. The TS requires hydrostatically testing certain containment isolation valves at a pressure of 1.1Pa (8.36 psig) if sufficient fluid inventory is available to provide a water seal for 30 days. The allowable leak rate for the hydro-tested valve is 1-gallon-per-minute-(GPM) per-valve. There are 24 hydro-tested valves, 9 of them require extending their test interval. The licensee stated that all these valves have been previously hydrostatically tested with several valves having zero leakage. Total measured leakage for the nine valves was 0.319 GPM compared to 9 GPM allowable leakage (about 6% of the allowable leakage).

In view of the licensee's information, the staff finds that: (1) extending the test interval for these valves is not expected to increase the overall leakage rate in excess of the allowable limit due to their low leakage rate from previous testing; (2) these valves are a minority of the total containment isolation valves; (3) these valves do not have a significant amount of operating time on them and previous tests show no failure or necessary rework on the valves. Based on these considerations, the staff concludes that the deferred testing on these valves has little safety significance. Therefore, the staff finds the proposed TS change to be acceptable. Since the extended test interval is still within the 24-month test interval required by Appendix J for those valves with a 30-day filled water seal, exemption from Appendix J for these valves is not required.

2.2.5 Airlock and Equipment Hatch Air System Testing

The surveillance requirements of TS 4.6.2.3d.2 and TS 4.6.2.1e.2 require that the inflatable seal system of the drywell airlock and the personnel door in the drywell equipment hatch be demonstrated operable by conducting a seal pneumatic system leak test at least once per 18 months, and verifying that the system pressure does not decay more than 0.67 psig from 75 psig within 24 hours.
The licensee requested a relief from the TS requirement to extend the 18-month test interval for the two air systems by 21 days to coincide with the first refueling outage. The air systems supply air to the inflatable door seals. The

inflatable seals provide a barrier to bypass leakage from the drywell to the containment to ensure all steam resulting from a LOCA is routed through the suppression pool. Each door seal has an individual air system with a dedicated accumulator and check valve. The latest test data for the pneumatic systems of the airlock and the equipment hatch personnel door show leakage of approximately 53% and 56%, respectively, of the TS-allowable limit. Seal leakage by pressurizing the area between the door seals shows a total of 0.4068 SCFD for all the three doors, versus an allowable leakage of 97.2 SCFD.

The licensee has provided the following justification to support its TS changes:

(1) The drywell airlock and the personnel door in the drywell equipment hatch have had little wear to their mechanical components or inflatable seals due to operation of the door.

(2) Historically, testing has shown low leakage either from the air systems or from the pressure tests conducted between the door seals. Adequate margin between the measured leakages and the allowable leakages will ensure the drywell integrity during the extended period.

(3) The air systems are operated at a relatively low pressure (75 psig) and the incoming air is regulated at the door which lessens the effects of any instrument air system pressure fluctuations or spikes.

During the blowdown period of an onsite LOCA, the potential leaking steam through airlock and equipment hatch personnel door seals would produce pressurization of the containment and increase containment pressure. The staff has evaluated drywell bypass leakage consequences considering the above and concludes that the possibility of increasing containment pressure beyond design due to bypass leakage is very small. Consequently, the chance of steam leakage through these door seals to cause containment pressurization should not be significant. Therefore, the staff concludes that the potential impact to containment integrity as a result of the requested testing delay for 21 days is negligible.

Based on the above discussion, the staff finds the licensee proposed amendment to TS 4.6.2.3d.2 and TS 4.6.2.1e.2 is acceptable.

Based on the above, the staff concludes that the proposed extension of the test intervals for the components identified in the submittal are acceptable. A one-time exemption from Appendix J for the five (outleakage) containment isolation valves is found to be acceptable. These approvals are based on the assumption that all other tests will be conducted in accordance with the requirements of the TS and Appendix J.

3.0 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 and/or changes to the 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 exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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 nor environmental assessment need be prepared in connection with the issuance of this amendment.

In connection with issuance of the Exemption, the Commission has determined that the granting of the exemption will have no significant impact on the environment pursuant to 10 CFR 51.32 (52 FR 28054).

4.0 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 this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

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