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

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Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1, AMENDMENT NO. 8 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 65397)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 8 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 May 15, 1987 as supplemented by letters dated June 30, and July 15, 1987. Your June 30, 1987 letter withdrew the TS change request as specified in Attachment 4 to the May 15, 1987 submittal.

This amendment authorizes one-time extensions to the surveillance intervals for certain reactor protection system and isolation actuation instrumentation 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. 8 to License No. NPF-47
2. Safety Evaluation

cc w/enclosures:
See next page

PD4/LA	PD4/PM	ICSB	OGC-Bethesda	PD#4/D
PNoonan	WPaulson:sr			JCalvo
7/17/87	1/20/87	7/21/87	7/24/87	7/30/87
		RLasky 7/21/87		

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P PDR

Mr. James C. Deddens
Gulf States Utilities Company

River Bend Nuclear Plant

cc:
Troy B. Conner, Jr., Esq.
Conner and Wetterhahn
1747 Pennsylvania Avenue, NW
Washington, D.C. 20006

Mr. Eddie Grant
Director - Nuclear Licensing
Gulf States Utilities Company
P. O. Box 220
St. Francisville, LA 70775

Richard M. Troy, Jr., Esq.
Assistant Attorney General in Charge
State of Louisiana Department of Justice
234 Loyola Avenue
New Orleans, Louisiana 70112

Resident Inspector
P. O. Box 1051
St. Francisville, Louisiana 70775

Gretchen R. Rothschild
Louisianians for Safe Energy, Inc.
1659 Glenmore Avenue
Baton Rouge, Louisiana 70775

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
Office of Executive Director
for Operations
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

Philip G. Harris
Cajun Electric Power Coop. Inc.
10719 Airline Highway
P. O. Box 15540
Baton Rouge, LA 70895

Mr. J. E. Booker
Manager-River Bend Oversight
P. O. Box 2951
Beaumont, TX 77704

Mr. William H. Spell, Administrator
Nuclear Energy Division
Office of Environmental Affairs
P. O. Box 14690
Baton Rouge, Louisiana 70898

Mr. J. David McNeill, III
William G. Davis, Esq.
Department of Justice
Attorney General's Office
7434 Perkins Road
Baton Rouge, Louisiana 70808

H. Anne Plettinger
3456 Villa Rose Drive
Baton Rouge, Louisiana 70806

President of West Feliciana
Police Jury
P. O. Box 1921
St. Francisville, Louisiana 70775

Mr. Frank J. Uddo
Uddo & Porter
6305 Elysian Fields Avenue
Suite 400
New Orleans, Louisiana 70122

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. 8
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 May 15, 1987 as supplemented by letters dated June 30, and July 15, 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. 8 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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. 8

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 areas of change. Overleaf pages are provided to maintain document completeness.*

REMOVE	INSERT
3/4 3-1	3/4 3-1
3/4 3-2*	3/4 3-2*
3/4 3-7	3/4 3-7
3/4 3-8*	3/4 3-8*
3/4 3-9	3/4 3-9
3/4 3-10*	3/4 3-10*
3/4 3-11	3/4 3-11
3/4 3-12*	3/4 3-12*
3/4 3-25*	3/4 3-25*
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-28*	3/4 3-28*
3/4 3-29	3/4 3-29
3/4 3-30*	3/4 3-30*

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.***

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition. The requirement to place a trip system in the tripped condition does not apply to Functional Units 6 and 10 of Table 3.3.1-1.

***Logic System Functional Test period may be extended as identified by note 'p' on Table 4.3.1.1-1.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3	3	2
	5(b)4	3	3
b. Inoperative	2	3	1
	3, 4	3	2
	5	3	3
2. Average Power Range Monitor (c):			
a. Neutron Flux - High, Setdown	2	3	1
	3	3	2
	5(b)4	3	3
b. Flow Biased Simulated Thermal Power - High	1	3	4
c. Neutron Flux - High	1	3	4
d. Inoperative	1, 2	3	1
	3, 4	3	2
	5	3	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2(d)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Reactor Vessel Water Level-High, Level 8	1(e)	2	4
6. Main Steam Line Isolation Valve - Closure	1(e)	4	10
7. Main Steam Line Radiation - High	1, 2(d)	2	5
8. Drywell Pressure - High	1, 2(f)	2	1

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b) S	S/U[c], W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U,S,(b) S	S/U[c], W W	SA SA	2 3, 4, 5
b. Flow Biased Simulated Thermal Power - High	S,D[h]	S/U[c], W	W[d][e], SA[o], R[i]	1
c. Neutron Flux - High	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 4, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R[g][p]	1, 2[j]
4. Reactor Vessel Water Level - Low, Level 3	S	M	R[g]	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R[g]	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Main Steam Line Radiation - High	S	M	R	1, 2[j]
8. Drywell Pressure - High	S	M	R[g]	1, 2[l]

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R(g)	1, 2, 5(k)
b. Float Switch	NA	Q	R	1, 2, 5(k)
10. Turbine Stop Valve - Closure	S(m)	M(n)	R(g)(n)	1
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	S(m)	M(n)	R(g)(n)	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.
(g) Calibrate Rosemount trip unit setpoint at least once per 31 days.
(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.
(i) This calibration shall consist of verifying the simulated thermal power time constant to be less than 6.6 seconds.
(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(l) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required per Specification 3.10.1
(m) Verify the Turbine Bypass Valves are closed when THERMAL POWER is greater than or equal to 40% RATED THERMAL POWER.
(n) The CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.
(o) The CHANNEL CALIBRATION shall exclude the flow reference transmitters; these transmitters shall be calibrated at least once per 18 months.
(p) This period may be extended to the first refueling outage, not to exceed 9-15-87.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.2-1.

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each isolation actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.2.1-1.

4.3.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.*

4.3.2.3 The ISOLATION SYSTEM RESPONSE TIME of each isolation trip function shown in Table 3.3.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months, where N is the total number of redundant channels in a specific isolation trip system.

*Logic System Functional Testing period may be extended as identified by notes C and D on Table 4.3.2.1-1.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL***	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	1, 7, 8, 9(b)(j), 15, 16	2	1, 2, 3	20
b. Drywell Pressure - High	1, 3, 8(b)(c)(j)	2	1, 2, 3	20
c. Containment Purge Isolation Radiation - High	8	1	1, 2, 3	21
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level-Low Low Low, Level 1	6(c)	2	1, 2, 3	20
b. Main Steam Line Radiation - High	6, 9(d)	2	1, 2, 3	23
c. Main Steam Line Pressure - Low	6	2	1	24
d. Main Steam Line Flow - High	6	2/MSL	1, 2, 3	23
e. Condenser Vacuum - Low	6	2	1, 2**, 3**	23
f. Main Steam Line Tunnel Temperature - High	6	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	6	2	1, 2, 3	23
h. Main Steam Line Area Temperature High (Turbine Building)	6	2/area	1, 2, 3	23

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
i. Main Steam Line Tunnel Temperature Timer	NA
j. RHR Equipment Room Ambient Temperature - High	NA
k. RHR Equipment Room Δ Temperature - High	NA
l. RHR/RCIC Steam Line Flow - High	NA
m. Drywell Pressure - High	NA
n. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. Reactor Vessel Water Level - Low Level 3	≤ 10[a]
d. Reactor Vessel Water Level - Low Low Low Level 1	≤ 10[a]
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
7. MANUAL INITIATION	NA

(a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Tables 3.6.4-1 and 3.6.5.3-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

##Time delay of 45-47 seconds.

###Time delay of 3-13 seconds.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low Level 2	S	M	R(b)(c)	1, 2, 3
b. Drywell Pressure - High	S	M	R(b)	1, 2, 3
c. Containment Purge Isolation Radiation - High	S	M	R	1, 2, 3
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(b)(c)	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R(b)	1
d. Main Steam Line Flow - High	S	M	R(b)(c)	1, 2, 3
e. Condenser Vacuum - Low	S	M	R(b)	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Area Temperature-High (Turbine Building)	S	M	R(b)	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low Level 2	S	M	R(b)(c)	1, 2, 3
b. Drywell Pressure - High	S	M	R(b)	1, 2, 3
c. Fuel Building Ventilation Exhaust Radiation - High	S	M	R	*
d. Reactor Building Annulus Ventilation Exhaust Radiation - High	S	M	R	1, 2, 3
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	M	R	1, 2, 3
b. Δ Flow Timer	NA	M	Q	1, 2, 3
c. Equipment Area Temperature - High	S	M	R	1, 2, 3
d. Equipment Area Δ Temperature - High	S	M	R	1, 2, 3
e. Reactor Vessel Water Level - Low Low Level 2	S	M	R(b)(c)	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. SLCS Initiation	NA	M(a)	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	M	R(b)	1, 2, 3
b. RCIC Steam Line Flow-High Timer	NA	M	Q	1, 2, 3
c. RCIC Steam Supply Pressure - Low	S	M	R(b)	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	R(b)	1, 2, 3
e. RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
f. RCIC Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
i. Main Steam Line Tunnel Temperature Timer	NA	M	Q	1, 2, 3
j. RHR Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
k. RHR Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
l. RHR/RCIC Steam Line Flow-High	S	M	R(b)	1, 2, 3
m. Drywell Pressure-High	S	M	R(b)	1, 2, 3
n. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	S	M	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	M	R	1, 2, 3
c. Reactor Vessel Water Level - Low Level 3	S	M	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low Low Low Level 1	S	M	R(b)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	M	R(b)(c)	1, 2, 3
f. Drywell Pressure - High	S	M	R(b)	1, 2, 3
7. MANUAL INITIATION	NA	M	NA	1, 2, 3

*When handling irradiated fuel in the Fuel Building.
**When the reactor mode switch is in Run and/or any turbine stop valve is open.
(a) Each train or logic channel shall be tested at least every other 31 days.
(b) Calibrate trip unit setpoint at least once per 31 days.
(c) May be extended to the first refueling outage, not to exceed 9-15-87.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status:

   1. Within 7 days, provided that the HPCS and RCIC systems are OPERABLE, or

   2. Within 72 hours, provided either the HPCS or the RCIC system is inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.3.3 At least once per 18 months, the ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 8 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By letter dated May 15, 1987 as supplemented by letters dated June 30, and July 15, 1987, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would extend the surveillance intervals, on a one-time basis, for certain reactor protection system and isolation actuation instrumentation until the first refueling outage scheduled to begin September 15, 1987. The extensions requested range from about 24 days to 31 days. The licensee's June 30, 1987 letter withdrew the Technical Specification (TS) change request as specified in Attachment 4 to the May 15, 1987 application.

2.0 EVALUATION

Detailed reviews and evaluations of the licensee's May 15, 1987 submittal were performed by EG&G Idaho, Inc., under contract to the NRC, with general supervision by the NRC staff. The licensee's response was reviewed, evaluated, and the results reported by EG&G in their Technical Evaluation Report (TER), "Technical Evaluation Report for Review of River Bend Station Unit 1 Proposed Technical Specification Amendment (RBG-25955)", dated July 1987 (attached). The staff has reviewed this report and concurs with the conclusion that the licensee's one-time request for extension of channel calibration and logic system functional tests is acceptable for the following instruments:

- Reactor Vessel Steam Dome Pressure-High (Reactor Protection System)
- Reactor Vessel (RHR Cut-In Permissive) Pressure-High (RHR System Isolation)
- Main Steam Line Flow-High (Main Steam Line Isolation)
- Reactor Vessel Water Level-Low Low Level 2 (Primary Containment Isolation)

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- Reactor Vessel Water Level-Low Low Low Level 1 (Main Steam Line Isolation)
- Reactor Vessel Water Level-Low Low Level 2 (Secondary Containment Isolation)
- Reactor Vessel Water Level-Low Low Level 2 (Reactor Water Cleanup System Isolation).

The TER notes that the asterisk footnote on TS page 3/4 3-11 includes the words "Isolation System Response Time" but no justification has been provided to support an extension to the surveillance interval for this surveillance. The TER recommends that these words be deleted from the footnote. The licensee's letter dated July 15, 1987 provides the clarification that these words should be deleted. The staff concurs with this change to the footnote.

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.

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.

Principal Contributor: W. A. Paulson

Dated: August 3, 1987

Attachment: Technical Evaluation Report

EGG-NTA-7759
July 1987

INFORMAL REPORT

TECHNICAL EVALUATION REPORT FOR REVIEW OF RIVER BEND STATION UNIT 1 PROPOSED TECHNICAL SPECIFICATION AMENDMENT (RBG-25955)

J. C. Stachew

Prepared for the

U.S. NUCLEAR REGULATORY COMMISSION

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EGG-NTA-7759

TECHNICAL EVALUATION REPORT FOR REVIEW OF RIVER BEND STATION UNIT-1 PROPOSED TECHNICAL SPECIFICATION AMENDMENT (RBG-25955)

Docket No. 50-458

J. C. Stachew

Published July 1987

Idaho National Engineering Laboratory
EG&G Idaho, Inc.

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. D6023

ABSTRACT

This EG&G Idaho, Inc., report evaluates the submittal (RBG-25955) provided by Gulf States Utilities Company (GSU) for River Bend Station (RBS) Unit-1. The evaluation was to determine the acceptability of the proposed Technical Specification (TS) amendment for a one time extension of surveillances for specific Reactor Protection System and Engineered Safety Feature instrumentation.

FOREWORD

This report is supplied as part of the "Technical Assistance for Operating Reactors Licensing Actions" being conducted for the U.S. Nuclear Regulatory Commission, Washington D.C., by EG&G Idaho, Inc., NRR and I&E Support.

The U.S. Nuclear Regulatory Commission funded the work under authorization B&R 20-19-10-11-1, FIN No. D6023.

Docket No 50-458
TAC No. 65396

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TECHNICAL EVALUATION REPORT FOR REVIEW OF RIVER BEND STATION UNIT-1 PROPOSED TECHNICAL SPECIFICATION AMENDMENT (RBG-25955)

1. INTRODUCTION

Gulf States Utilities Company (GSU) has submitted a proposed amendment (RBG-25955) to the Technical Specifications for River Bend Station (RBS) Unit-1. This report provides an evaluation of the proposed one-time extension of surveillances for specific Reactor Protection System and Engineered Safety Feature instrumentation.

2. DISCUSSION AND EVALUATION

The Licensee provided four Attachments to their Technical Specification Amendment application.[1] Each of these Attachments is discussed and evaluated in an individual subsection below. Also, the Licensee's letter of May 11, 1987,[4] requests changes to the Technical Specifications that affect pages that are common to the Licensee's changed pages that are the subject of this report. The individual changes proposed by the Licensee's letter of May 11, 1987, and those in the four attachments to the subject Licensee's letter of this report, have been reviewed for their consistency when all of the individual changes are accumulated on the affected common pages. As the changes which affect common pages only involve adding superscript note designations to specified surveillances for different instrument types, accumulation of the individual changes is acceptable (the different instruments do not have an interaction and the acceptability of the added superscript can be judged on an individual basis). One recurring carryover is the Licensee's reference to Note (d) on Technical Specification page 3/4 3-11. This note (d) does not appear in the subject Table 4.3.2.1-1 in the subject letter of this report, but does appear in the Licensee's letter of May 11, 1987,[4] and is, therefore, acceptable.

In each case, the requested extension of the surveillances already has accounted for the maximum extension of 25% allowed by Technical Specification 4.0.2 to the existing required 18-month surveillance interval.

2.1 Reactor Vessel Steam Dome Pressure-High

The Licensee letter of May 15, 1987,[1] Attachment 1 proposed changes to Technical Specification pages 3/4 3-1, 7, 9, 11, and 29. These changes were for a one-time extension of about 31 days to both the Channel Calibration and Logic System Functional Test (LSFT) for the Reactor Vessel Steam Dome Pressure-High instrument of the Reactor Protection System instrumentation and for the Reactor Vessel (RHR Cut-In Permissive) Pressure-High instrument, and Isolation System Response Time of the RHR

System Isolation Actuation instrumentation. The purpose of this change is to allow the above-listed surveillances to be delayed from their scheduled due date of August 15, 1987, until the first refueling outage scheduled to begin September 15, 1987. The requested extension of the surveillances already has accounted for the maximum extension of 25% allowed by Technical Specification 4.0.2 to the existing required 18-month surveillance interval.

Evaluation:

The Licensee provided the following technical justification for requesting these surveillance extensions.

1. For the Channel Calibration extension, the present allowance of 15 psig for channel drift between the Technical Specification "allowable valves" of ≤1079.7 psig and ≤150 psig and the respective "trip setpoints" of ≤1064.7 psig and ≤135 psig for the reactor vessel pressure-high of the Reactor Protection System and RHR Isolation System, respectively, more than compensates for any additional drift that may occur during the requested extension period. Licensee calculations indicate that the calculated drift for the requested extended surveillance period was 12.7 psig. Thus, drift is adequately specified and neither the Technical Specification "trip setpoints" nor the "allowable valves" require any change from their present values for the requested surveillance period extension. Channel Checks and Channel Functional Tests are specified and executed for the Reactor Vessel Steam Dome Pressure-High and Reactor Vessel (RHR Cut-In Permissive) Pressure-High instruments in Technical Specification Tables 4.3.1.1-1 and 4.3.2.1-1, respectively.

2. For the LSFT, Licensee results of a search of the industry Nuclear Plant Reliability Data System (NPRDS) indicated a very low probability for equipment failures with regard to manual switches and auxiliary relays for the subject surveillance period and requested extension. Channel Checks and Channel Functional Tests are specified and executed per Technical Specification Tables 4.3.1.1-1 and 4.3.2.1-1. In

addition to the automatic isolation feature, administrative controls do not allow placing the RHR system in the cooling mode until reactor pressure has been reduced to less than 135 psig.

3. Maintenance or calibration of instruments on the reactor pressure vessel sensing legs has resulted in spurious Engineered Safety Features actuations.

Based on the above justification provided by the Licensee and the relatively short extension requested (about 31 days) compared to the required 18 month + 25% extension and the fact that the subject instrumentation does not exhibit sudden catastrophic failure or out-of-specification performance in these time frames, it is judged that the requested extensions for the Channel Calibration and LSFT are acceptable. Also, per Footnote (g) of Technical Specification Table 4.3.1.1-1 and Footnote (b) of Table 4.3.2.1-1, trip unit setpoints are calibrated at least once per 31 days.

However, it is noted that although the Licensee has also requested the same time extension for the RHR System Isolation, Reactor Vessel (RHR Cut-In Permissive) Pressure-High, Isolation System Response Time, no justification for this change was presented. Table 3.3.2-3, Isolation System Instrumentation Response Time of the Technical Specifications indicates that the response time for the Reactor Vessel (RHR Cut-In Permissive) Pressure-High instrument of the RHR System Isolation is "NA". It is recommended that the words "Isolation System Response Time and" be deleted from the asterisk footnote at the bottom of proposed Technical Specification page 3/4 3-11.

2.2 Main Steam Line Flow-High

The Licensee letter of May 15, 1987,[1] Attachment 2 proposed changes to Technical Specification pages 3/4 3-11, 26, and 29. These changes were for a one-time extension from August 22, 1987, to no later than September 15, 1987, to the Channel Calibration, Logic System Function Test

(LSFT), and Isolation System Response Time surveillances of the Main Steam Line Flow-High instrument of the Main Steam Line Isolation actuation instrumentation. The purpose of this change is to allow the above-listed surveillances to be delayed from their scheduled due date of August 22, 1987, until the first refueling outage scheduled to begin September 15, 1987.

Evaluation:

The Licensee provided the following technical justification for requesting the surveillance extensions.

1. For the Channel Calibration extension, the present allowance of 5 psid for channel drift between the Technical Specification "allowable value" of 178 psid and the "trip setpoint" of 173 psid more than compensates for any additional drift that may occur during the extension period. Licensee calculations using standard setpoint methodology (GE Topical Report NEDC-31336) indicate a channel drift of 1.41 psid for a 30-month surveillance interval. The 173 psid trip setpoint converts to a 140% of rated mass flow setpoint [using flow element calibration data to convert from differential pressure units (psid) to main steam line flow]. The 140 % analytical limit is to protect a safety limit of 170 % of rated mass flow in the event of a postulated guillotine break of a main steam line. Channel Checks and Channel Functional Tests are specified and executed for the Main Steam Line Flow-High instrument per Technical Specification surveillance Table 4.3.2.1-1. Also, per footnote (b) to Technical Specification Table 4.3.2.1-1, calibration of the Main Steam Line Flow-High trip unit setpoint is required at least once per 31 days.

2. For the Logic System Functional Test, Licensee calculations performed using a fault tree model of the Main Steam Isolation Valve (MSIV) isolation logic with basic event failure rates based on conservative WASH-1400 data,[2] indicates MSIV isolation logic failure at

1.06×10^{-3}/demand and 1.41×10^{-3}/demand for 18- and 24-month surveillance intervals, respectively. (The Licensee's letter had these failures per demand interchanged. A correction will be issued.)

3. GSU has determined, based on the group isolation logic of the isolation actuation instrumentation, that the subject surveillances should be performed while in cold shutdown due to the high risk of placing the plant in a scram condition.

Based on the above justification provided by the Licensee and the relatively short extension requested (about 25 days) compared to the required 18 month + 25% extension and the fact that the subject instrumentation does not exhibit sudden catastrophic failure or out of specification performance in these time frames, it is judged that the requested extensions for the Channel Calibration and LSFT are acceptable.

However, it is noted that although the Licensee has also requested the same time extension for the Main Steam Line Flow-High instrument for the Isolation System Response Time in Technical Specification 4.3.2.3, no justification for this change was presented. The RBS FSAR, Section 15.6.4, Steam System Piping Break Outside Containment, takes credit for isolation valve closure time and a value is specified for the subject instrument in Technical Specification Table 3.3.2-3, Isolation System Instrumentation Response Time. Although the short time extension requested compared to the required 18 month + 25% extension surveillance interval is such that significant changes to the Isolation System Response Time would not be anticipated; without any Licensee justification, it is recommended that the words "Isolation System Response Time and" be deleted from the asterisk footnote at the bottom of proposed Technical Specification page 3/4 3-11.

2.3 Low Reactor Vessel Water Level

The Licensee letter of May 15, 1987, Attachment 3 proposed changes to Technical Specification pages 3/4 3-11, 26, 27, and 29. These changes were

for a one-time extension of about 30 days to the Channel Calibration, Logic System Functional Test (LSFT), and Isolation System Response Time surveillances of the following instruments:

Reactor Vessel Water Level-Low Low Level 2 (Primary Containment Isolation)

Reactor Vessel Water Level-Low Low Low Level 1 (Main Steam Line Isolation)

Reactor Vessel Water Level-Low Low Level 2 (Secondary Containment Isolation)

Reactor Vessel Water Level-Low Low Level 2 (Reactor Water Cleanup System Isolation).

The purpose of this change is to allow the above stated surveillances to be delayed from their scheduled due date of August 16, 1987, until the first refueling outage scheduled to begin September 15, 1987.

Evaluation:

The Licensee provided the following technical justification for requesting the surveillance extensions.

1. For Channel Calibration extension, the present allowance of 4 inches of water for channel drift between the Technical Specification "allowable values," either -47 or -147 inches, and the respective "trip setpoints" of either -43 or -143 inches, more than compensates for any additional drift that may occur during the extension period. Licensee calculations, concurred in by the vendor, indicate that the calculated drift for 24 months between surveillances is 3.76 inches. The existing surveillance interval of 18 months plus the 25% extension allowed by Technical Specification 4.0.2, plus the additional 30 days requested by the Licensee, accumulates to 23.5 months. Thus, drift is

adequately specified and the Technical Specification "trip setpoints" and "allowable values" for the subject instrumentation do not require any change from their present values for the requested surveillance period extension.

2. For the LSFT, Licensee calculations performed using a fault tree model of the Main Steam Isolation Valve (MSIV) isolation logic with basic event failure rates based on conservative WASH-1400 data,[2] indicates MSIV isolation logic failure at 1.06×10^{-3}/demand and 1.41×10^{-3}/demand for 18- and 24-month surveillance intervals, respectively. (The Licensee's letter had these failures per demand interchanged. A correction will be issued.)

   A review of the Nuclear Plant Reliability Data System (NPRDS) was performed by the Licensee to gather data on manual switches and auxiliary relays. This review found no reports of manual switch failures and a mean time between failure (MTBF) of 239, 278 hours (about 27 years) for auxiliary relays. These data show that the basic design of the logic is highly reliable and is unlikely to fail due to the delay in testing.

   The logic for Reactor Vessel Water Level-Low Low Level 2 operated successfully including all Level 2 isolations during a loss of feedwater event on March 1, 1986 (LER 86-021).

3. GSU has determined, based on the group isolation logic of the isolation actuation instrumentation, that the subject surveillances should be performed while in cold shutdown due to the high risk of placing the plant in a scram condition.

Based on the above justification provided by the Licensee and the relatively short extension requested (about 30 days) compared to the required 18 month + 25% extension and the fact that the subject instrumentation does not exhibit sudden catastrophic failure or out-of-specification performance in these time frames, it is judged that

the requested extensions for the Channel Calibrations and LSFTs are acceptable. Further, as noted in Section 2.2 of this report, Channel Checks and Channel Functional Tests are specified and executed for Low Reactor Vessel Water Level for the subject isolation actuation instrumentation per Technical Specification surveillance Table 4.3.2.1-1. Also, per footnote (b) of Technical Specification Table 4.3.2.1-1, calibration of the Low Reactor Vessel Water Level involved trip unit setpoints are required at least once per 31 days.

However, it is noted that although the Licensee has also requested the same time extension for the involved Low Reactor Vessel Water Level, Isolation System Response Times in Technical Specification 4.3.2.3, no justification for this change was presented. Response time values are specified per the subject instruments in Technical Specification Table 3.3.2-3, Isolation System Instrumentation Response Time. Although the short time extension requested compared to the required 18 month + 25% extension surveillance interval is such that significant changes to the Isolation System Response Time would not be anticipated; without any Licensee justification, it is recommended that the words "Isolation System Response Time and" be deleted from the asterisk footnote at the bottom of proposed Technical Specification page 3/4 3-11.

2.4 Reactor Protection Instrumentation System

The Licensee letter of May 15, 1987,[1] Attachment 4 proposed changes to Technical Specification pages 3/4 3-30, 41, 42, 43, 54, 58 and 3/4 5-5. These changes were for a one-time extension of surveillances from their due date of August 25, 1987, until the first refueling outage scheduled to begin September 15, 1987. The surveillances were on several Emergency Core Cooling System Actuation instruments and Reactor Core Isolation Cooling System Actuation Instrumentation. Subsequently, the Licensee performed these surveillances prior to their due date of August 25, 1987, during an unexpected short outage. The Licensee, by letter dated June 30, 1987, withdrew the request for the changes in Attachment 4.[3] Therefore, Attachment 4 of the Licensee's subject letter,[1] is not further addressed in this report.

3. CONCLUSIONS

Based on the evaluations presented, GSU's requested one time extension of surveillances was found acceptable for the Channel Calibration and Logic System Functional Test, but not for any Isolation System Response Time for the following instruments.

Reactor Vessel Steam Dome Pressure-High (Reactor Protection System)
Reactor Vessel (RHR Cut-In Permissive) Pressure-High (RHR System Isolation)
Main Steam Line Flow-High (Main Steam Line Isolation)
Reactor Vessel Water Level-Low Low Level 2 (Primary Containment Isolation)
Reactor Vessel Water Level-Low Low Low Level 1 (Main Steam Line Isolation)
Reactor Vessel Water Level-Low Low Level 2 (Secondary Containment Isolation)
Reactor Vessel Water Level-Low Low Level 2 (Reactor Water Cleanup System Isolation).

The requested Extension of the required 18-month surveillance interval accounted for the maximum 25% extension allowed by Technical Specification 4.0.2. The purpose of the change is to allow the listed instrument surveillances to be delayed from their scheduled due dates (beginning August 15, 1987), until the first refueling outage scheduled to begin September 15, 1987. As the Licensee did not provide any justification for the involved Isolation System Response Times, the request for extension of their surveillances is not acceptable and it is recommended that this part of the request be deleted from Technical Specification page 3/4 3-11 in the asterisk in 4.3.2.2 and in the footnote.

It is also noted that although the proposed asterisk footnote on Technical Specification page 3/4 3-11 refers to a Note d on Table 4.3.2.1-1 (in Attachments 1, 2, and 3 of the Licensee's letter[1]), no Note d appears in that table. However, the Licensee's letter of May 11, 1987,[4]

Attachment 3, does provide a Note (d) to the subject Table. As both Licensee letters[1,4] provide changes to the same Table 4.3.2.1-1 and as the individual changes were to be accumulated, reference to Note (d) is acceptable.

Finally, Attachment 4 of the Licensee's submittal[1] was withdrawn[3] as the involved surveillances were completed during an unexpected outage. This report, therefore, does not provide an evaluation of Attachment 4.

4. REFERENCES

1. J. C. Deddens letter to NRC, "River Bend Station Unit-1 Docket No. 50-458," RBG-25955, May 15, 1987, Gulf States Utilities Company.

2. NUREG 75/014 (WASH-1400), Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, October 1975.

3. J. C. Deddens letter to NRC, "River Bend Station-Unit 1 Docket No. 50-458," RBG-26186, Gulf States Utilities Company, June 30, 1987.

4. J. C. Deddens letter to NRC, "River Bend Station-Unit 1, Docket No. 50-458," RBG-25917, Gulf States Utilities Company, May 11, 1987.

NRC FORM 335
(2-84)
NRCM 1102
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1 REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)

EGG-NTA-7759

2 TITLE AND SUBTITLE

TECHNICAL EVALUATION REPORT FOR REVIEW OF RIVER BEND STATION UNIT 1 PROPOSED TECHNICAL SPECIFICATION AMENDMENT (RBG-25955)

3 LEAVE BLANK

4 DATE REPORT COMPLETED

MONTH	YEAR
July	1987

5 AUTHOR(S)

J. C. Stachew

6 DATE REPORT ISSUED

MONTH	YEAR
July	1987

7 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

NRR and I&E Support
EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

8 PROJECT/TASK/WORK UNIT NUMBER

9 FIN OR GRANT NUMBER

D6023

10 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a TYPE OF REPORT

Final Technical Evaluation Report

b PERIOD COVERED (Inclusive dates)

12 SUPPLEMENTARY NOTES

13 ABSTRACT (200 words or less)

Technical evaluation report on the review of Gulf States Utilities Company amendment (GSU letter of May 15, 1987, RBG-25955) for the Technical Specifications for the River Bend Station Unit 1.

14 DOCUMENT ANALYSIS a KEYWORDS/DESCRIPTORS

b IDENTIFIERS/OPEN-ENDED TERMS

15 AVAILABILITY STATEMENT

Unlimited

16 SECURITY CLASSIFICATION

(This page)

Unclassified

(This report)

Unclassified

17 NUMBER OF PAGES

18 PRICE