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December 30, 1983
RBG- 16,669
File Code G9.5, G9.8.6.1

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

Dear Mr. Denton:

River Bend Station Units 1 and 2
Docket Nos. 50-458/50-459

Enclosed for your review are Gulf States Utilities Company responses to Draft Safety Evaluation Report (DSER) open items identified by the Nuclear Regulatory Commission's Core Performance Branch (CPB). This letter supplements information contained in docketed correspondence from J.E. Booker to H.R. Denton dated November 11, 1983. Also enclosed is the response to the request for additional information identified in a letter from A. Schwencer to W.J. Cahill, Jr. dated December 6, 1983. Attachment 1 for this letter summarizes the open items and indicates changes to be made in the River Bend Station FSAR. Attachment 2 provides the response and reference material for each item. Where indicated, these responses will be provided in a future amendment to the FSAR.

Sincerely,

William J. Reed Jr.

for J.E. Booker
Manager-Engineering
Nuclear Fuels & Licensing
River Bend Nuclear Group

JEB/WJR/ERG/JEP erg JEP

Enclosures

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ATTACHMENT 1

	DSER SECTION	SUBJECT	FSAR REVISIONS
4.	4.2.4.3 pg. 4-29	Post Irradiation Monitoring Program	N/A
9.	4.4.6 pg. 4-44,47	Loose Parts Monitoring System	Enclosure 1 and 11/11/83 Letter
10.	4.4.7 pg. 4-44,47	Inadequate Core Cooling (LRG II 6-CPB; TMI Item II.F.2)	Enclosure 2
12.	12/6/83 Letter	Fuel and Poison Rod Pressures	N/A

ATTACHMENT 2

RESPONSES TO DSER OPEN ITEMS

4. DSER (page 4-29) - Post Irradiation Monitoring Program

Response

River Bend Station meets the criterion of Paragraph II.D.3 of Section 4.2 of the Standard Review Plan (NUREG-0800) for post-irradiation fuel surveillance program as documented in NEDE-24343-P and additionally described in a letter from J.S. Charnley (General Electric) to C.H. Berlinger (NRC) dated November 23, 1983.

River Bend Station is currently working with the Licensing Review Group - II to develope a generic program to address the specific monitoring applications requested by the Staff.

9. DSER (page 4-44, 4-47) - Loose Parts Monitoring System

Response

The response to this request is provided in the November 11, 1983 letter from J.E. Booker to H.R. Denton and supplemented in Enclosure 1.

10. DSER (page 4-44, 4-47) - Inadequate Core Cooling (II.F.2)

Response

The response to this request is provided in the response to Question 421.12 and revised FSAR Table 1A-1, Table 7.5-1 and Table 7.5-2 (Enclosure 2). The response to Question 421.12 was previously submitted to the Staff (Instrumentation and Control Systems Branch) in a letter dated December 9, 1983 (RBG-16535).

12. 12/6/83 Letter - Fuel and Poison Rod Pressures

Response

The River Bend Station fuel rod internal pressure does not exceed nominal system pressure during the first cycle of plant operation. Prior to the start of the second cycle of plant operation the fuel rod pressure for all burnups considered will be evaluated to assure that River Bend Station meets the criterion of Paragraph II.A.1.f of Section 4.2 of the Standard Review Plan (NUREG-0800).

This response will be incorporated into the FSAR in a future amendment.

Enclosure 1

the following reactor vessel parameters is provided in the main control room and is discussed in Chapter 7.

1. Reactor vessel water level 2
2. Reactor vessel differential pressures
3. Reactor vessel internal pressure
4. Neutron monitoring system.

Insert new Section 4.4.6.1 here

INSERT (For Page 4.4-27)

4.4.6.1 Loose Parts Monitoring System

The Loose Parts Monitoring System (LPMS) is designed to detect loose parts in the reactor pressure vessel and to provide early warning to the operator so that damage to or malfunctions of safety-related primary system components may be avoided or mitigated. An alarm from the LPMS is provided in the main control room. The LPMS is not considered to be a safety-related system and is designed to require minimum operator interfacing during normal operation. Provisions are made to properly analyze and investigate potential loose parts. Since the analysis of potential loose parts signals requires comparison with baseline signals, the LPMS will be operational and capable of recording vibration signals for signature analysis during initial reactor start-up testing. The LPMS is designed to meet the requirements of Regulatory Guide 1.133.

Insert (Page 4.4-27)

4.4.6.1 Loose Parts Monitoring System

The Loose Parts Monitoring System (LPMS) is designed to detect loose parts in the reactor pressure vessel and to provide early warning to the operator so that damage to or malfunctions of safety-related primary system components may be avoided or mitigated. The LPMS is not considered to be a safety-related system and is designed to require minimum operator interfacing during normal operation. Provisions are made to properly analyze and investigate potential loose parts. Since the analysis of potential loose part signals requires comparison with baseline signals, the LPMS will be operational and capable of recording vibration signals for signature analysis during initial reactor start-up testing. The LPMS is designed to meet the requirements of Regulatory Guide 1.133.

REPLACE

c. Functionally demonstrate the ability of the systems to properly collect and dispose of drainage.

4. Acceptance Criteria

a. Interlocks, controls, and alarms performances are as specified by the system elementary diagrams.

5 b. Pump performance is comparable to that shown in the manufacturer's technical instruction manual.

INSERT

14.2.12.2 Initial Startup Test Phase Discussion

1. Startup Test Procedure

All those required tests comprising the initial startup test phase are discussed in Section 14.2.12.3. For each test a description is provided for test objective, test prerequisites, 5 test procedure, and a statement of test acceptance criteria, where applicable.

The operating power-flow map is presented as Fig. 14.2-4. The test conditions are marked on Fig. 14.2-4, and each test described in Section 14.2.12.3 is accomplished at the test conditions stated in Fig. 14.2-5.

The acceptance criteria section of each test has one or two sections. The following two paragraphs describe the degree of each kind of test criterion 5 and the actions to be taken after an individual criterion is not satisfied.

a. Level 1

If a Level 1 test criterion is not satisfied, the plant is placed in a hold condition that is judged to be satisfactory and safe, based upon prior testing. Plant operating or test procedures or the Technical Specifications may guide the decision on the direction taken. Startup tests consistent with this hold condition may be continued. Resolution of the problem is immediately pursued by appropriate equipment adjustments or through engineering support by offsite personnel if needed.

INSERT (For Page 14.2-124)

14.2.12.1.69 Loose Parts Monitoring System

TEST OBJECTIVES

1. To demonstrate proper operation of the Loose Parts Monitoring Equipment.

2. To collect data to use as baseline information during subsequent operation.

3. To verify alert level.

PREREQUISITES AND INITIAL CONDITIONS

The Loose Parts Monitoring System is connected to the reactor and instrumentation and control testing is complete.

TEST PROCEDURE

1. Indication and alarm functions will be demonstrated.

2. The reactor recirculation pumps will be operational for portions of the test.

3. Test results will be reviewed and corresponding alert and alarm setpoints established.

ACCEPTANCE CRITERIA

Alert and alarm functions will perform within design tolerances.

REPLACE

INSERT (For Page 14.2-124)

14.2.12.1.69 Loose Parts Monitoring System

TEST OBJECTIVES

1. To demonstrate proper operation of the Loose Parts Monitoring Equipment.

2. To collect data to use as baseline information during subsequent operation.

3. To verify alert level.

PREREQUISITES AND INITIAL CONDITIONS

The Loose Parts Monitoring System is connected to the reactor and instrumentation and control testing is complete.

The Loose Parts Monitoring System channel check and functional tests have been completed with acceptable results. Reactor recirculation system is operable.

TEST PROCEDURE

1. With the reactor recirculation system operational, record baseline data.

2. With the reactor recirculation system and the loose parts simulator operational, record simulated loose parts data.

3. Review test results and establish corresponding alert and alarm setpoints.

4. Perform Step 2 again to verify alert function is operational and data acquisition equipment is automatically activated.

ACCEPTANCE CRITERIA

Alert, alarm, and automatic data acquisition equipment function as required upon receipt of a loose part signal reaching the alert level.

Enclosure 2

QUESTION 421.014 (7.2, 7.3, 7.4, 7.5)

Provide an evaluation of the effects of high temperatures on reference legs of water level measuring instruments subsequent to high-energy line breaks, including the potential for reference leg flashing/boil off, the indication/annunciation available to alert the control room operator of erroneously high vessel level indications resulting from high temperatures, and the effects on safety systems actuation (e.g., delays).

RESPONSE

~~The response to this request will be provided by September 1983~~.

REPLACE WITH INSERT

INSERT (for Pg. Q&R 7.2-2)

A comprehensive report discussing the effects of high temperatures on water level reference legs for BWR water level instrumentation has been submitted to the NRC Staff for review by the BWR Owners' Group. The report is entitled "Review of BWR Reactor Vessel Water Level Measurement Systems" and is identified as S. Levy Report No. SLI-8211. River Bend Station endorses the content and findings of this report where applicable to the design of RBS Unit 1.

The following design features are implemented at River Bend Station to improve control room operator and safety system response where accurate reactor vessel water level measurements are required:

1) The vertical drop of the water level reference leg instrument lines does not exceed eighteen inches where the lines are subject to temperature excursions capable of causing erroneous readings. The area of primary concern for this design improvement is the drywell.

   RBS procedures delineate for operator information the maximum expected errors for water level measurements given the unlikely event of drywell heatup beyond normal ambient conditions.

2) Annunciation is provided in the main control room to alert the operator to potential or actual water level measurement anomolies owing to high reference/variable leg temperatures. The annunciator is synthesized from two, redundant, Class 1E instrument channels which monitor drywell temperature.

3) The control room operator is furnished with redundant, Class 1E reactor vessel water level instrument channels for two overlapping regions of the vessel. The first region covers water level over a wide range from the dryer down to near the top of the fuel zone. The second region overlaps the first but extends down to the bottom of the core region. This safety-related display instrumentation and other water level measurement readings are deemed sufficient to provide the operator with an accurate appraisal of reactor vessel water level.

4) ECCS initiating signals are generated from analog circuitry which provides a switching function. Water level instruments used for ECCS actuation are grouped according to range (narrow, wide, fuel zone, and high level-upset) and electrical channel separation. This allows individual instrument channels to be observed for proper operation. This design feature greatly reduces the possibility of either a failed channel being unnoticed or erroneous channel information being used for system actuation.

5) Reactor water level information from several sources within the main control room is monitored by the Safety Parameter Display System (SPDS). The SPDS alerts control room operators to water level measurement anomolies should the situation arise. The

SPDS performs this function by performing a channel check by comparing two or more channels for equivalence within a given error margin.

6. River Bend Station utilizes restriction orifices in the variable and reference leg sensing lines for RPV water level measurements. The orifices are located in close proximity to the drywell instrument penetrations. This design feature effectively obviates oscillatory water level readings when flashing occurs in the drywell portion of sensing lines.

Additional information is provided in revised Section 7.5.1.1.2 and Appendix 1A, Item II.F.2.

channels to verify operability and variable level. All trip units display trip status, using an indicator light located on the trip unit.

7.5.1.1.2 Reactor Water Level

REPLACE WITH INSERT

~~Two wide range water level signals are transmitted from two independent differential pressure transmitters and are recorded on two, two pen recorders. One pen in each recorder records the wide range level, and the other pen records the reactor pressure. The range of the recorded level is from the top of the feedwater control range (just above the high level turbine trip point) down to a point near the top of the active fuel~~.

7.5.1.1.3 Reactor Pressure

Two reactor pressure signals are transmitted from two independent differential pressure transmitters and are recorded on two, two-pen recorders. One pen records pressure and the other pen records the wide range level. The range of recorded pressure is from 0 to 1,500 psig.

7.5.1.2 Reactor Shutdown Indication

The following information is provided to the main control room operator to monitor reactor shutdown.

1. Control rod status lights indicate each rod fully inserted. Control rod scram pilot valve position status lights indicate open valves.

2. Neutron monitoring power range channels and recorders downscale. The power sources are from RPS MG sets. A loss of offsite power would result in all scram valve solenoids being deenergized and reactor scram. 7

3. Annunciators and indicators for RPS variables and trip logic in the tripped state.

4. The process computer provides logging of trips and control rod position and provides thermal-hydraulic information to the operator which he uses to keep the plant operating within technical specification limits. Redundant capability exists in case of process computer failure. The power source for the process computer is a Normal UPS.

INSERT (for Pg. 7.5-2)

Reactor water level information is obtained from physically and electrically separated differential pressure (dp) instrumentation. A cold reference leg design is utilized for RBS with a minimum amount of elevation change inside the drywell to minimize instrument channel error. The dp instruments operate an analog current loop which transmits level information to the main control room. Table 7.5-1 identifies reactor water level displays.

TABLE 1A-1 (Cont)

Item and Title	Position	FSAR Reference*	
II.E.4.2 Containment isolation dependability	A design review will be conducted and any recommended design changes will be evaluated at a later date for possible incorporation into the River Bend Station design.	6.2.4	
II.F.1 Accident-monitoring instrumentation	A review of accident monitoring instrumentation in comparison with the guidance of Regulatory Guide 1.97, Revision 2, has been conducted. The results of this review are provided in revised Section 7.5.	7.5	7
II.F.2 Instrumentation for detection of inadequate core cooling	~~Procedures will be written that will allow the operator to operate and interpret main control room indications installed to detect inadequate core cooling.~~ REPLACE WITH INSERT	7.5	
II.K.1.5 Review ESF valves	GSU will review all safety-related valve positions, positioning requirements, and positive controls to assure that valves remain positioned (open or closed) in a manner that ensures the proper operation of engineered safety features. This review will cover procedures for control of maintenance and testing of safety-related systems, system operating and general plant startup procedures, and shift turnover and general rounds procedures. Manually operated valves in the main flow paths for safety-related systems will be locked in position and verified by valve lineup. Valve lineups for safety-related systems will be performed by two qualified operations personnel, one person doing the initial positioning and a second person verifying the position. The maintenance work request procedure requires the Shift Supervisor or Control Operations Foreman to ensure that appropriate functional tests are assigned and/or the system has been properly restored to its normal configuration after maintenance has been performed. Surveillance test procedures will have data sheets requiring operator signoff to verify that each system is returned to its normal configuration after performing a test. Shift turnover procedures will be as described in GSU response to Item I.C.2 and will include review of system status.	13.5	7

INSERT (for Table 1A-1 Item II.F.2)

River Bend Station has participated in a BWR Owner's Group study analyzing inadequate core cooling (ICC) in boiling water reactors. In conjunction with the submittal of Sol Levy Report Nos. SLI-8211 and SLI-8218 to the NRC, River Bend Station has reviewed and evaluated the reports against its plant design and have concluded the following:

1. The results of Sol Levy Report No. SLI-8218 affirm that reactor pressure vessel (RPV) water level is a reliable, responsive indicator of ICC. RPV water level instrumentation measures the trend toward ICC, indicates its existence, and indicates the return of adequate core cooling.

2. The RPV water level measurement enhancements, as modified by specific changes identified in SLI-8211 and discussed in the response to NRC Question 421.014, provide suitable means for detection of adequacy of core cooling.

TABLE 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

System	Parameter	Type of Readout	Range	Readout Location	RG 1.97 Item No.	Class 1E Power	
Rod Control and Information	Control rod position	Lights	NA	MCR	B2	No	
	Control rod scram Valves	Lights	NA	MCR	NA	No	
Neutron Monitoring	Power range neutron flux	Recorder	0-125%	MCR	B1	No	
	Source range count rate	Meter	10^{-1} to 10^{6} CPS	MCR	B1	No	
Nuclear Boiler	Reactor vessel pressure	Recorder	0-1,500 psig	MCR	A2, B6, C4, C9	Yes	
	Reactor vessel water level	Recorder/ meter	Bottom of active ~~-160" +60"~~ fuel to steam dryer	MCR	B4	Yes	
RCIC	RCIC Flow	Meter	0-1,000 gpm	MCR	D13	Yes	
	RCIC discharge pressure	Meter	0-1,500 psig	MCR	NA	Yes	7
Emergency Core Cooling	HPCS flow	Meter	0-8,000 gpm	MCR	D14	Yes	
	HPCS discharge pressure	Meter	0-1,500 psig	MCR	NA	Yes	
	LPCS flow	Meter	0-8,000 gpm	MCR	D15	Yes	
	RHR flow (LPCI and shutdown cooling)	Meter	0-8,000 gpm	MCR	D16	Yes	
	RHR service water flow	Meter	0-10,000 gpm	MCR	NA	Yes	
	ECCS Pumps	Status lights	NA	MCR	NA	Yes	
	ECCS Valves	Position lights	NA	MCR	NA	Yes	

TABLE 7.5-2

REGULATORY GUIDE 1.97, REVISION 2, INSTRUMENTATION PROVISIONS

Item	Parameter	Regulatory Guide Type	Regulatory Guide Category	Plant Type	Plant Category	Range	Notes
A1	Containment and drywell hydrogen concentration	C	1	A	1	0-10%	1
A2	Reactor vessel pressure	C	1	A	1	0-1,500 psig	
A3	Suppression pool water temperature	D	2	A	1	0°F-200°F	2
B1	Neutron flux	B	1	B	2	3.57×10^{-10} to 125%	3
B2	Control rod position	B	3	B	3		4
B3	RCS soluble boron concentration (sample)	B	3	B	3	8-1,000 ppm	5
B4	Coolant level in reactor	B	1	B	1	~~Top~~ Bottom of active fuel to ~~below main steam line~~ steam dryer	6
B5	BWR core thermocouples	B	1				7
B6	RCS pressure	B	1	A	1	0-1,500 psig	
B7	Drywell pressure	B	1	B	1	0-75 psia	
B8	Drywell sump level	B	1	B	3	Near bottom to top	8
B9	Primary containment pressure	B	1	B	1	0-75 psia	
B10	Primary containment isolation valves	B	1	B	1	Open-closed	9
C1	Radioactivity concentration or radiation level in circulating primary coolant	C	1				10
C2	Analysis of primary coolant (gamma spectrum)	C	3	C	3	Isotopic analysis	5
C3	BWR core thermocouples	C	1				7