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February 1, 1985
RBG-20046
File No. G9.5, G9.8.6.2

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 - Unit 1
Docket No. 50-458

Enclosed is Gulf States Utilities Company (GSU) response to the Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI) on the Initial Test Program from Mr. A. Schwencer (NRC) to Mr. W. J. Cahill (GSU) dated November 5, 1984. This response also addresses Safety Evaluation Report (SER) Confirmatory Item #61 - Initial Test Program Revisions (SER Section 14, pg. 14-3) and Generic Letter 83-24. Attachment 1 summarizes the responses to each RAI while revisions to the Final Safety Analysis Report (FSAR), as contained in Enclosures 1 through 4, provide the necessary supporting statements which will be incorporated in a future amendment.

Sincerely,

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

JEB/WJR/JWL/je

Attachments (3)

Enclosures (4)

B001
1/40

Attachment 1

1. Question 640.1, Part 3: The response to this item and FSAR Subsection 14.2.12.1.26 (Normal and Standby Service Water Systems Preoperational Test) should be revised to incorporate tests conducted as stated in the letter from J. E. Booker (GSU) to H. R. Denton (NRC) dated February 27, 1984.

   Response: The response provided in Attachment 1 to the February 27, 1984 letter stated that the preoperational test would include:

   > "testing all four standby service water pumps over the range of the basin level, as well as testing two pumps at a time at the minimum basin level calculated 30 days following a LOCA."

   The testing procedure includes individually testing each of four pumps at the minimum and maximum basin levels and to test pump A for vortexing, then pumps A & C together for vortexing, and finally pump C for vortexing (Division II pumps, B and D, will be tested similarly) all at the minimum basin level 30 days after a LOCA. FSAR Section 9.2.5.2, pg. 9.2-30 identifies this minimum basin level 30 days after a LOCA as 65 feet 0 inches (or 34 inches above pump suction per manufacturer specification.) This value is contained in FSAR Amendment 15. Enclosure 1 contains a revised test abstract for Section 14.2.12.1.26 reflecting the above testing procedures.

2. Question 640.6: The following test abstracts should be modified to reinstate appropriate acceptance criteria which was deleted by FSAR Amendment 13.

   A. FSAR Subsection 14.2.12.1.4 (Nuclear Boiler System Preoperational Test); subparts a-d.

   B. FSAR Subsection 14.2.12.1.42 (Containment Atmosphere Monitoring System Preoperational Test); containment pressure instrumentation.

   C. FSAR Subsection 14.2.12.1.50 (Instrument and Service Air System Acceptance Test); proper functioning of safety-related components on loss of instrument air.

   Response:

   A. Attachment 2 contains the FSAR test abstract acceptance criteria (14.2.12.1.4.4) which was revised by Amendment 13. The following discussion presents the basis for the deletion or revision of each criteria (a through g).

      a. The Containment Isolation System Preoperational Test (14.2.12.1.43) Acceptance Criteria 4.a includes the operation of isolation logic functions. This criteria was therefore deleted from Section 14.2.12.1.4.

      b. The nuclear boiler process instrumentation alarm and actuation set points which provide permissive and prohibit interlocks are tested as part of the system from which they originate. However, the boiler test performed in this abstract does verify some set points and therefore the acceptance criteria was revised accordingly.

c. Actual testing procedures check isolation valve operating times against the test specification, not FSAR Table 6.2-40, although both are the same. The acceptance criteria was revised accordingly.

d. The GE preoperational test specification for the MSIV accumulators indicates the accumulators should operate the MSIVs the required number of times. Therefore, the acceptance criteria was revised accordingly.

e, f, g. Acceptance Criteria e, f, and g are now contained in the Automatic Depressurization System Preoperational Test (14.2.12.1.63) and therefore were deleted from Section 14.2.12.1.4.

Finally, Section 14.2.12.1.4 added Acceptance Criteria 4.d in Amendment 13 to accurately reflect the actual test procedure for the nuclear boiler system.

B. The Amendment 13 revision to the Containment Atmosphere Monitoring System Preoperational Test (14.2.12.1.42) added Test Procedure 3.c (and the reference contained within) without a corresponding acceptance criteria to accurately reflect Startup & Test testing procedures. The first sentence references the Containment Integrated Leak Test (14.2.12.1.53) as the place where pressure instrumentation is functionally verified to within set acceptance criteria. The instrument is actually calibrated during the generic testing phase as a prerequisite to this preoperational test (14.2.12.1.42.2.a). However, the instrument is checked during the testing procedure in this preoperational test (i.e., 14.2.12.1.42.3.c, the second sentence) to assure containment pressure is "tracked". Startup and Test performs this function visually to again assure operability, i.e. the instrument line is free of obstructions, but does not record quantitative data. There is no need to record data as the instrument has already been calibrated and functionally verified. A visual observation is made to assure the gauge follows containment pressure and the performance of the observation is recorded.

C. As indicated in the Amendment 13 revision to Test Procedure 3.d in the Instrument and Service Air Acceptance Test (14.2.12.1.50), a loss of instrument air test is done on an individual component basis during the generic testing phase of the system preoperational tests which utilize instrument air. The systems which have such a test performed are listed and the loss of instrument air test is a prerequisite to the actual system test. Therefore, Acceptance Criteria 4.d was deleted from Section 14.2.12.1.50 in Amendment 13.

3. Question 640.25: The response to this item should address the exception taken to Regulatory Guide 1.68, Initial Test Programs for Water-Cooled Nuclear Power Plants, as contained in FSAR Table 1.8-1, "Regulatory Guide 1.68, Rev. 2 (August 1978)," Section II.2. Appropriate test abstracts should be modified to incorporate any tests conducted in response to this item.

   Response: As indicated in Item 4 of Attachment 1 to the February 27, 1984 letter from J. E. Booker (GSU) to H. R. Denton (NRC), the response provided in Attachment 2 and Enclosure 2 to a February 8, 1984 letter (same parties) was unacceptable to the NRC Staff and revised. The revision, as contained in Enclosure 3 to the 2/27/84 letter, replaced the previous response (Enclosure 2 to the 2/8/84 letter) and was incorporated into FSAR Amendment 13. However, an error in publication of Amendment 13 resulted in the inclusion of the 2/8/84 letter's revision to Regulatory Guide 1.68, Exception II.2 as presently shown in Table 1.8-1, pages 96/96a of 193. It was, and still is, GSU position that the 2/8/84 letter's revision to Regulatory Guide 1.68 be ignored in favor of the original exception, again reiterated in Enclosure 2 of this letter.

4. Question 640.43: The response to this item states that FSAR Subsection 14.2.12.3.12 (RCIC System) will be revised to address the first three tests recommended in Appendix E of BWROG-8120 "BWR Owner's Group Evaluation of NUREG-0737 Requirement I.G.1, Training During Low Power Testing" in June 1984. The response further states that recommendations as outlined in FSAR Subsection 14.2.12.3.28.1 (Loss of Turbine - Generator and Offsite Power, Test Objective) will be forwarded to the NRC by June 1984. This item remains open until the proposed modifications have been received, reviewed, and accepted.

   Response: FSAR Section 14.2.12.1.6, Reactor Core Isolation Cooling System Preoperational Test, lists as Test Procedures 3.g, 3.h, and 3.i a summary of the first three recommended tests in Appendix E to BWROG-8120. Therefore, no plans for revision of Section 14.2.12.3.12 exist and the response to Question 640.43 is modified accordingly in Enclosure 3.

   In regard to LRG-II position 1-HFS, Generic Letter 83-24 explicitly states:

   > "unless the need is identified in the resolution of Generic Issue A-44, 'Station Blackout', the SBO test should not be required at BWR's."

   Gulf States Utilities does not plan to conduct a Station Blackout (SBO) test at River Bend Station (RBS); however, compliance with the training and testing recommended by BWROG-8120 will serve as the RBS position for NUREG-0737, TMI Item I.G.1. Further justification against performing a SBO test is provided in Attachment 3 and in conjunction with BWROG-8120 forms the RBS-specific, LRG-II Position 1-HFS response. At the time Generic Issue A-44 is resolved, GSU will reconsider the Station Blackout event if necessary. Enclosure 3 also contains revised FSAR test abstracts reiterating the aforementioned RBS position.

5. Item 14A: FSAR Figure 14.2-6, "Maximum Acceptance Drive Flow Following Pump Trip," should be provided or an appropriate document should be referenced.

   Response: The General Electric Company's (GE) Transient Safety Analysis Design Report (TSADR) (NEDC-30877) contains the figure(s) required to determine maximum acceptable drive flow following a recirculation pump trip and will be available in March 1985. Therefore FSAR Figure 14.2-6 will be deleted (see Enclosure 4.)

Attachment 2

j. In conjunction with the remote shutdown
5 preoperational test, operation of the three designated safety relief valves from the remote shutdown panel is demonstrated.

4. Acceptance Criteria

a. Operation of isolation logic functions as specified by the system elementary diagrams.

b. Nuclear boiler process instrumentation alarm and actuation set points which provide permissive and prohibit interlocks are specified by the GE design specification, elementary diagrams, and the Technical
5 Specifications.

c. Isolation valve operating times are as specified by FSAR Chapter 6, Table 6.2-40.

d. Capacity of the MSIV accumulators is as specified by the GE Preoperational Test Specification.

e. Safety/relief valves and their actuators function as specified by the GE Preoperational Test Specification.

f. ADS logic functions as specified by the system elementary diagrams.

g. Operation of safety/relief valves from the remote shutdown panel function as specified by the GE Preoperational Test Specification.

14.2.12.1.5 Residual Heat Removal System Preoperational Test

1. Test Objectives

To verify that the residual heat removal (RHR) system provides the following safeguards and operational functions:

a. Low pressure coolant injection (LPCI)

b. Suppression pool cooling

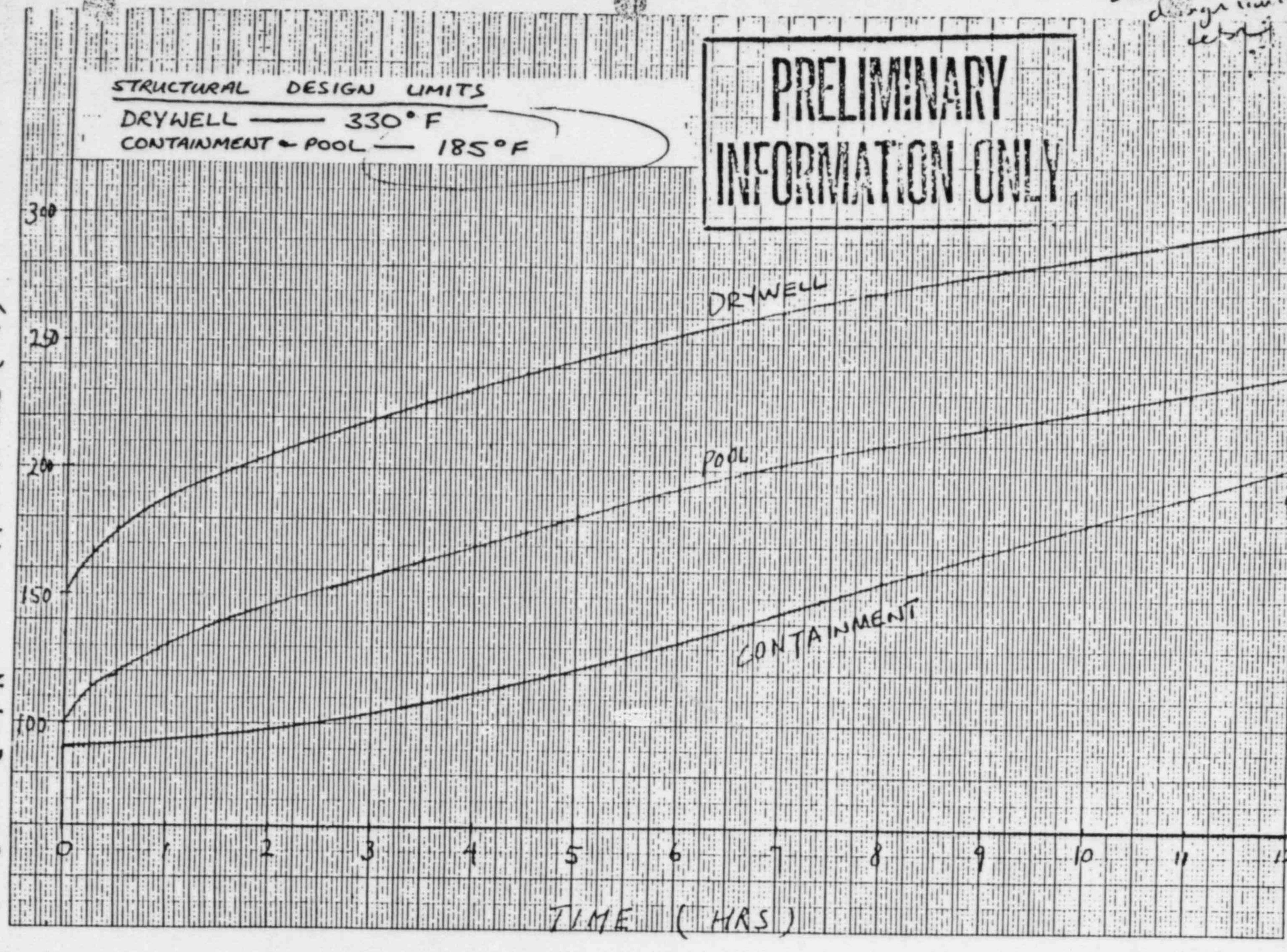
c. Shutdown cooling

Attachment 3

The Station Blackout (SBO) event is defined as the loss of all AC power, which includes both off-site and on-site power, and is therefore more severe than a loss of off-site because ECCS, containment cooling and suppression pool cooling are not available. In fact, the only make-up water to the vessel is from the Reactor Core Isolation Cooling (RCIC) system. An immediate adverse affect of a SBO would be a rise in the drywell ambient temperature. The drywell temperature is restricted to 135° F by Technical Specification 3.6.2.6 while the environmental design criteria established the normal maximum drywell temperature at 140° F. A preliminary scoping study performed by GSU indicated a SBO would create a drywell ambient temperature of approximately 150° F in less than one minute (see the attached figure). Although the safety related equipment in the drywell is qualified for higher temperatures, any increase above normal maximum temperature may compromise qualification of the equipment to survive future transients.

The performance of an actual station blackout test will increase the risk of damage or future malfunction of equipment and may require premature rework or replacement of equipment subjected to the test environment. In addition, failure of non-safety related equipment subjected to elevated temperatures may affect normal plant operation and/or availability. A modified SBO test in which drywell cooling, ECCS or other systems are not disabled such that plant safety is maintained or a SBO test of limited duration would not provide information or training beyond that available from other tests.

For these reasons River Bend Station will not perform a Station Blackout Test; however, in accordance with Generic Letter 83-24, the recommednations contained in BWROG-8120 will be implemented to satisfy NUREG-0737, TMI Item I.G.1.


Structural design limits
STRUCTURAL DESIGN LIMITS
DRYWELL —— 330° F
CONTAINMENT & POOL —— 185° F
PRELIMINARY INFORMATION ONLY
DRYWELL
POOL
CONTAINMENT
TEMPERATURE (°F)
300
250
200
150
100
0
1
2
3
4
5
6
7
8
9
10
11
12
TIME (HRS)
ENCLOSURE

Enclosure 1

2. Prerequisites

13 a. Required preliminary tests are completed and approved.

b. All permanently installed instrumentation properly calibrated and operable

c. All test instrumentation available and properly calibrated

d. Instrument air system available

e. Appropriate ac and dc power sources available

f. Chemically acceptable water source available

3. Test Procedure

a. Measure normal and standby service water pump performance.

b. Check trips, permissives, interlocks, controls, alarm and computer points.

c. Verify flow path between service water pump (SWP) and RPCCW systems.

5 d. In conjunction with the remote shutdown preoperational test, demonstrate operation of the standby service water system through the RHR heat exchangers from the remote shutdown panel.

INSERT

4. Acceptance Criteria

13 a. Normal and standby service water pump performance is as specified in FSAR Chapter 9, Tables 9.2-1 and 9.2-15.

5 b. Trips, permissives, interlocks, and controls function as specified by the system elementary diagrams.

c. Operation of the standby service water system from the remote shutdown panel is as specified by the GE Preoperational Test Specification.

13 d. Rated flow through all heat exchangers is as specified by each component's manufacturer

technical instruction manual and within allowable tolerances of FSAR Tables 9.2-1 and 9.2-15. 13

INSERT

Insert for page 14.2-68

e. Demonstrate combined operation of standby service water pumps A and C (as well as B and D) at the minimum basin level 30 days after a LOCA (i.e. el. 65'-0")

Insert for page 14.2-68a

e. Verify pumps A and C (as well as B and D) in combination operate vortex-free at a basin water level of 65'-0.

Enclosure 2

TABLE 1.8-1 (Cont)

18. Appendix A, paragraph 5.d.d (p. 1.68-18)

Regulatory Guide 1.68.2 is addressed in a separate compliance statement.

19. Appendix A, paragraph 5.o.o (p. 1.68-18)

Vibration on NSSS piping in the drywell is monitored and evaluated using remote sensing devices.

20. Appendix C, paragraph 1.a. (3)(d) (p.1.68-20)

Regulatory Guide 1.37 is addressed in a separate compliance statement.

II. Exceptions

1. Paragraph C.2 (p. 1.68-4)

Surveillance tests necessary to demonstrate proper operation of interlocks, set points, and other protective features, systems, and equipment required by the technical specifications are performed during the initial test program. This does not include the performance of formal surveillance tests required by 10CFR50.36. The actual performance of these tests is not within the scope of the initial test program. Chapter 16 of this FSAR defines the formal surveillance test requirements.

2. Appendix A, paragraph 1.g(2) (p. 1..68-8) 5

Emergency loads are tested at normal voltage.

DELETE AND REPLACE WITH INSERT

Preoperational or preliminary test procedures are currently written to include testing at low voltage on some loads on the 125-V dc safety-related busses, such as inverters and electrical protection relays. For other loads on these busses, it will be verified that the design of the load or component takes into account a minimum battery voltage of 105-V dc and the voltage drop to the load of component. It will be verified on selected worst-case loads that the voltage drop from the battery bus to the component served does not exceed the expected 13

TABLE 1.8-1 (Cont)

~~voltage drop.~~ The preoperational tests demonstrate the proper functioning of the diesel generator voltage regulator under an auto-start and loading condition. 13

3. Appendix A, paragraph 1.h(1)(a) (p. 1.68-9) 5

Expansion tests are not performed during the preoperational testing of ECCS. RCIC turbine steam supply and exhaust lines are visually

Insert for Table 1.8-1, page 96 of 193

Operation at other than rated voltage may be considered as destructive testing.

Enclosure 3

QUESTION 640.43

Modify your FSAR to describe tests and training to comply with NUREG-0737 Item I.G.1. Refer to LRG-II position 1-HFS.

RESPONSE

Revised FSAR Sections 13.2.1.2, 14.2.12.2.2 and 14.2.12.2.3 (Amendment 5, August 1982) commit to the training recommended by BWROG-8120, BWR Owner's Group Evaluation of NUREG-0737 Requirement I.G.1, Training During Low Power Testing. The first three tests recommended in Appendix E of BWROG-8120 are ~~will be~~ incorporated in Section ~~14.2.12.3.12,~~ 14.2.12.1.6 Test Number 14-RCIC System, ~~in June 1984.~~ The fourth recommended test is incorporated as part of
13 Section 14.2.12.1.4, Nuclear Boiler System Preoperational Test, while the fifth additional test recommended by the BWR Owner's Group is incorporated as part of Section 14.2.12.1.53, Containment Structural Integrity Test and Containment Leak Rate Preoperational Tests. Recommended tests 1 through ~~4 and~~ 5 were incorporated in the referenced sections in Amendment 5. Section 14.2.12.3.28, Test Number 31-Loss of Turbine-Generator and Offsite Power, addresses LRG-II position 1-HFS.

TABLE 1B-1 (Cont)

Item	Title	Endorsed	FSAR Section
	A-1 Waterhammer A-9 Anticipated Transients Without Scram A-11 Reactor Vessel Materials Toughness A-17 Systems Interaction in Nuclear Plants A-39 Safety Relief Valve Hydrodynamic Loads A-40 Seismic Design Criteria Short-Term Program A-43 Containment Emergency Sump Reliability A-44 Station Blackout A-45 Shutdown Decay Heat Removal Requirement A-46 Seismic Qualification of Equipment in Operating Plants A-47 Safety Implications of Control Systems A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment		
1HFS	Special Low Power Testing Program	NO ~~Yes~~	14.2.12.3.28 Appendix 1A, Item I.G.1
2HFS	Reactivity Emergency Procedures	Yes	13.5.2 Appendix 1A, Item I.C.1
3HFS	Common Vessel Level Reference	Yes	7.5.1.1.2 Appendix 1A, Item II.K.3.27
1CHEB	Reactor Coolant Sampling	Yes	9.3.2.6 Appendix 1A, Item II.B.3
2CHEB	Suppression Pool Sampling	Yes	9.3.2.6 Appendix 1A, Item II.B.3

3. Test Procedure

Both the jet pumps and the recirculation pumps cavitate at conditions of high flow and low power where NPSH demands are high and little feedwater subcooling occurs. However, the recirculation flow automatically runs back upon sensing a decrease in subcooling to lower the reactor power. The maximum recirculation flow is limited by appropriate limiters which run back the recirculation flow away from the possible cavitation region. It is verified that these limits are sufficient to prevent operation where recirculation pump or jet pump cavitation is predicted to occur.

4. Acceptance Criteria

Level 1

Not applicable.

Level 2

Runback logic is set so operation in areas of potential cavitation is not possible. 5

14.2.12.3.28 Test Number 31 - Loss of Turbine-Generator and Offsite Power 5

1. Test Objective

The purpose of this test is to determine the reactor transient performance during the loss of the main generator and all offsite power, and to demonstrate acceptable performance of the station electrical supply system. Loss of offsite power is maintained for sufficient time to demonstrate that necessary equipment, controls, and indications are available ~~following station blackout~~ to remove decay heat from the core using only standby power supplies and distribution systems.

Presently this test is only a loss of turbine-generator and offsite power. Some plant startup programs also include a simulated loss of onsite ac power. The LRG II position for NUREG-0737, Item I.G.1, is for River Bend Station to review the results preceding simulated loss of all ac power tests, performed at other BWRs, in order to determine the scope of such testing at River Bend 5

DELETE AND REPLACE WITH INSERT

Station. The results of these prior tests will be reviewed, plant-specific analysis performed, and
5 recommendations forwarded to the NRC as to whether or not the test or some portion of it should be
13 repeated at River Bend Station by June 1984.

2. Prerequisites

The appropriate preoperational tests have been completed and the FRC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate. Required electrical systems are aligned for full power operation.

3. Test Procedure

The loss of auxiliary power test is performed at 20 to 30 percent of rated power. The proper response of reactor plant equipment, automatic switching equipment, and the proper sequencing of the diesel generator load are checked. Appropriate reactor parameters are recorded during the resultant
5 transient. Offsite power is not restored for at least 30 min.

4. Acceptance Criteria

Level 1

Reactor pressure is maintained below the set point of the first safety valve, during the transient following the loss of the main generator and all offsite power. Reactor protection systems (RPS) operate to prevent violations of fuel thermal limits as specified in the Technical Specifications.

5 All safety systems such as the RPS, diesel generators, and HPCS automatically operate as specified by the Technical Specifications without manual assistance.

The HPCS and RCIC operate to keep the reactor coolant level above the initiation level of the LPCS, LPCI, and ADS.

Insert for page 14.2-179

However, the performance of a simulated loss of all AC power, or Station Blackout (SBO) test, will not be conducted owing to the hazard presented to plant equipment (Ref. 2). To comply with NUREG-0737, TMI Item I.G.1, RBS complies with the BWR Owner's Group recommendations, as contained in BWROG-8120, "BWR Owner's Group Evaluation of NUREG-0737, Requirement I.G.1, Training During Low Power Testing." This commitment satisfies the intent of LRG-II position 1-HFS as clarified by Generic Letter 83-24 and Reference 2.

References - 14.2

1. ANSI N510-1975, Testing of Nuclear Air Cleaning Systems, American Standards Institute, New York, NY. 5

2. Letter, J.E. Booker, Gulf States Utilities Company, Beaumont, Tx. to H.R. Denton, Nuclear Regulatory Commission, Bethesda, Md., February 1, 1985 (GSU Letter No. RBG-20046).

Enclosure 4

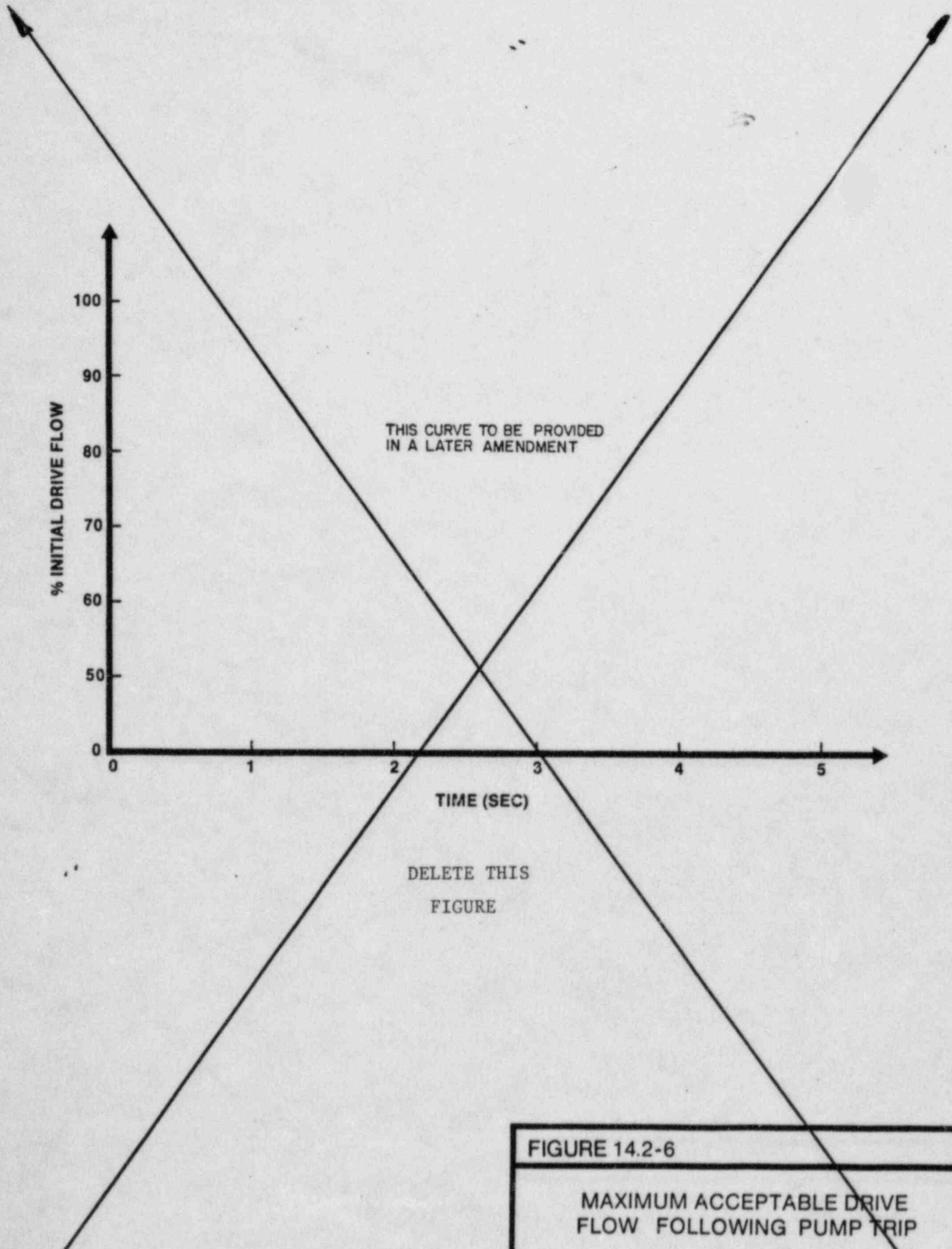


FIGURE 14.2-6

MAXIMUM ACCEPTABLE DRIVE FLOW FOLLOWING PUMP TRIP

RIVER BEND STATION
FINAL SAFETY ANALYSIS REPORT