NUREG-0989
Supplement No. 4

Safety Evaluation Report
related to the operation of River Bend Station

Docket No. 50-458

Gulf States Utilities Company
Cajun Electric Power Cooperative

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1985

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

ABSTRACT

Supplement No. 4 to the Safety Evaluation Report on the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners, for a license to operate River Bend Station has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in West Feliciana Parish, near St. Francisville, Louisiana. This supplement reports the status of certain items that had not been resolved at the time the Safety Evaluation Report and its first three supplements were published.

CONTENTS

APPENDICES

FIGURES

TABLES

1 INTRODUCTION AND GENERAL DESCRIPTION

1.1 Introduction

In May 1984, the Nuclear Regulatory Commission (NRC) staff issued its Safety Evaluation Report (SER) (NUREG-0989) on the application filed by Gulf States Utilities Company (applicant or GSU), acting on behalf of itself and for Cajun Electric Power Cooperative (CEPCO), for a license to operate the River Bend Station, Docket No. 50-458. On August 29, 1985, the staff issued a low-power operating license for River Bend Station.

In the SER, the staff identified items that were not yet resolved with the applicant. Supplement No. 1 (SSER 1) was issued in October 1984 to provide the staff evaluation of open items that had been resolved and to report on the status of all open items. Supplement No. 2 (SSER 2) and Supplement No. 3 (SSER 3) were both issued in August 1985 to update the previous evaluation and status. Supplement No. 4 (SSER 4) is being issued to provide more recent information regarding resolution of other open and confirmatory items and license conditions identified in the SER and its supplements. SSER 4, in general, reflects staff assessment of the applicant's hydrogen control system to handle hydrogen generated from postulated degraded-core accidents.

Each of the following sections and appendices is numbered the same as the corresponding SER section or appendix that is being updated. Appendix B lists the references cited in this report.* Appendix D is a list of acronyms and initialisms used herein, and Appendix E is a list of the principal staff members who contributed to this supplement.

Copies of this SER suppplement are available for inspection at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the Government Documents Department, Louisiana State University, Baton Rouge, Louisiana. Copies are also available for purchase from the sources indicated on the inside front cover.

The NRC Project Manager assigned to the operating license application for River Bend is Stephen M. Stern. Mr. Stern may be contacted by telephoning him at (301)492-8349 or by writing to the following address:

Mr. Stephen M. Stern
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

---

*Letters between the staff and the applicant are not included in Appendix B; they are, however, available in the NRC Public Document Room. Availability of all material cited is described on the inside front cover of this report.

1.5 Outstanding Issues

In the SER, the staff identified certain outstanding issues that had not been resolved with the applicant. The status of these issues is listed in an updated version of Table 1.3, which follows. The sections of this supplement where these issues are discussed are indicated.

SSER 3 closed all outstanding issues for the low-power license. Outstanding Issue 23 had been closed in SSER 2 only for the low-power license. Since the low-power license was issued on August 29, 1985, this issue is now open. Outstanding Issue 4 was closed in SSER 2 for 2 years from the issuance of an operating license. Since the low-power operating license was issued on August 29, 1985, Table 1.3 reflects that this issue has been closed until August 29, 1987.

1.6 Confirmatory Items

In the SER, the staff identified confirmatory items for which additional information was required to confirm preliminary conclusions. Table 1.4 lists the status of these issues and the sections of this supplement in which they are discussed. Confirmatory Item 75 was closed in SSER 3 for the low-power license only. Since the low-power license was issued on August 29, 1985, this issue has been reopened and the staff is awaiting information from the licensee.

1.7 License Conditions

The current status of license conditions has not changed since SSER 3 was issued.

Table 1.3 Listing of outstanding issues (revised from SER)

Issue	Status*	SSER 4 Section(s)
(1) Hydrostatic loading	Closed (SSER 1)	
(2) Moderate-energy line break	Closed (SSER 3)	
(3) High-energy line break	Closed (SSER 3)	
(4) Inservice test program (including RCS pressure boundary valve leakage)	Closed until August 29, 1987 (SSER 2)	
(5) Equipment qualification		
(a) Seismic and dynamic qualification	Closed (SSER 3); see Lic. Cond. 12	
(b) Environmental qualification of equipment	Closed (SSER 3); see Lic. Cond. 13	
(6) Preservice inspection program	Closed (SSER 3)	
(7) Containment loads	Closed (SSER 2)	
(8) ECCS LOCA analysis (II.K.3.31)	Closed (SSER 2)	
(9) Bypassed and inoperable status	Closed (SSER 3); see Lic. Cond. 15	
(10) Emergency diesel generators		
(a) Electrical loads	Closed (SSER 3)	
(b) Qualification of TDI diesel generators	Closed (SSER 3); see Lic. Cond. 16	
(c) Auxiliary support systems	Closed (SSER 2)	
(11) Submergence of electrical equipment	Closed (SSER 2)	
(12) Heavy-load handling system	Closed (SSER 2)	
(13) Safe/alternate shutdown	Closed (SSER 3); see Lic. Cond. 19	
(14) Communications systems	Closed (SSER 2)	
(15) Lighting systems	Closed (SSER 2)	

See footnotes at end of table.

Table 1.3 (Continued)

Issue	Status*	SSER 4 Section(s)
(16) HPCS diesel generator	Closed (SSER 2)	
(17) Fuel oil storage	Closed (SSER 2)	
(18) Emergency preparedness	Closed (SSER 3); see Lic. Cond. 11	
(19) Separation of electric circuits	Closed (SSER 2)	
(20) Human factors issues		
(a) Safety parameter display system	Closed (SSER 3); see Lic. Cond. 17	
(b) Control room survey	Closed (SSER 3)	
(c) Resolution of HEDs	Closed (SSER 3); see Lic. Cond. 17	
(21) Auxiliary systems		
(a) Standby service water system	Closed (SSER 3)	
(b) Standby liquid control systems	Closed (SSER 3)	
(c) Low-pressure interface leakage	Closed (SSER 3)	
†(d) Equipment and floor drains	Closed (SSER 3)	
†(e) Control building ventilation	Closed (SSER 3)	
†(f) Miscellaneous HVAC systems	Closed (SSER 3)	
(22) Starting voltage for Class 1E motors	Closed (SSER 3)	
(23) Hydrogen control - degraded core accident	Review continuing**	6.2.5

*License condition references are numbered as listed in Table 1.5.

**License subject to provision of 10 CFR 50.44.

Table 1.4 Listing of confirmatory items (revised from SER)

Issue	Status*	SSER 4 Section(s)
(1) West Creek sediment removal	Closed (SSER 2)	
(2) Ultimate heat sink	Closed (SSER 1)	
(3) Slope stability	Closed (SSER 2)	
(4) Pipe failure modes and check valve stress analysis	Closed (SSER 3)	
(5) Annulus pressurization	Closed (SSER 2)	
(6) Minimum wall thickness	Closed (SSER 1)	
(7) Thermal and anchor displacement loads	Closed (SSER 2)	
(8) Fuel rod mechanical fracturing	Closed (SSER 2)	
(9) Fuel assembly structural damage	Closed (SSER 2)	
(10) Postirradiation surveillance	Closed (SSER 1)	
(11) LOCTVS/CONTEMPT-LT 28 computer codes	Closed (SSER 2)	
(12) Reactor vessel cooldown rate	Closed (SSER 2)	
(13) SRV discharge testing	Closed (SSER 3)	
(14) Mark III-related issues	Closed (SSER 2); see Lic. Cond. 9	
(15) Containment repressurization	Closed (SSER 2)	
(16) Inleakage limit	Closed (SSER 1)	
(17) ECCS test return line design	Closed (SSER 1)	
(18) Containment purge valves	Closed (SSER 2)	
(19) Hydrogen control	Closed (SSER 2)**	
(20) PVLCS leakage	Closed (SSER 2)	

See footnotes at end of table.

Table 1.4 (Continued)

Issue	Status*	SSER 4 Section(s)
(21) Electrical and instrumentation and control diagrams	Closed (SSER 3)	
(22) Routing of circuits and sensors	Closed (SSER 2)	
(23) Instrumentation setpoints	Closed (SSER 3)	
(24) RPS power supply protection	Closed (SSER 3)	
(25) RPS and ESF channel separation	Closed (SSER 3)	
(26) Isolation devices	Closed (SSER 3)	
(27) Reactor mode switch	Closed (SSER 2)	
(28) ADS actuation	Closed (SSER 2)	
(29) ESF reset controls	Closed (SSER 3)	
(30) Initiation of ESF support systems	Closed (SSER 3)	
(31) Instrumentation and control power bus loss	Closed (SSER 3)	
(32) RCIC system	Closed (SSER 3)	
(33) Standby liquid control system (SCLC)	Closed (SSER 2)	
(34) Postaccident monitoring instrumentation	Closed (SSER 3); see Lic. Cond. 17	
(35) Temperature effects on level measurements	Closed (SSER 2)	
(36) High/low pressure interlocks	Closed (SSER 3)	
(37) End of cycle recirculation pump trip	Closed (SSER 2)	
(38) NMS and RCIS isolation	Closed (SSER 3)	
(39) Rod pattern control system microprocessors	Closed (SSER 3)	
(40) DRMS	Closed (SSER 3)	

See footnotes at end of table.

Table 1.4 (Continued)

Issue	Status*	SSER 4 Section(s)
(41) High-energy line break control system failures	Closed (SSER 3)	
(42) Multiple control system failures	Closed (SSER 3)	
(43) Emergency Response Information System (ERIS)	Closed (SSER 2)	
(44) LPCS/RHR A pump procedures	Closed (SSER 3)	
(45) EPA/RPS motor generator set interconnection	Closed (SSER 2)	
(46) Second level undervoltage protection relay setpoint	Closed (SSER 3)	
(47) Verification of test results for station electric distribution system voltage	Closed†	
(48) Safety cable identification	Closed (SSER 2)	
(49) Non-Class 1E loads - powered from Class 1E power supplies	Closed (SSER 3)	
(50) Postaccident sampling system	Closed (SSER 2)	
(51) Diesel generators - mechanical issues	Closed (SSER 2)	
(52) TMI Item II.F.1, Attachment 2	Closed (SSER 2)	
(53) Spent fuel transfer canal	Closed (SSER 1)	
(54) TMI Item II.B.2	Closed (SSER 2)	
(55) Backup RPM designate	Closed (SSER 2)	
(56) Personnel résumés	Closed (SSER 2)	
(57) Licensed operator review	Closed (SSER 2)	
(58) Offsite fire department training	Closed (SSER 2)	
(59) Emergency planning	Closed (SSER 3)	

See footnotes at end of table.

Table 1.4 (Continued)

Issue	Status*	SSER 4 Section(s)
(60) TMI Item I.C.1	Closed (SSER3)	
(61) Initial test program revisions	Closed (SSER 3)	
(62) Proper ESF function (II.K.1.5)	Closed (SSER 2)	
(63) Safety system operability status (II.K.1.10)	Closed (SSER 2)	
(64) QA organization	Closed (SSER 1)	
(65) Ultimate heat sink with delayed fan start	Closed (SSER 3); see Lic. Cond. 20	
(66) Participation of human factors specialists in detailed control room design review	Closed (SSER 3); see Lic. Cond. 17	
(67) Task analysis documentation	Closed (SSER 3)	
(68) Control room modifications	Closed (SSER 3); see Lic. Cond. 17	
(69) Containment venting procedures	Closed (SSER 3); see Lic. Cond. 17	
(70) Monitoring instruments for HPCS 125-V ac system	Closed (SSER 3)	
(71) Protection for lighting penetration circuits	Closed (SSER 3)	
(72) Process Control Program	Interim approval until first refueling outage (SSER 2)	
(73) Subcompartment pressure analysis	Closed (SSER 2)	
(74) Cable derating	Closed (SSER 2)	
(75) Equipment qualification - audit	Awaiting information (SSER 3)	1.6

*License condition references are as numbered in Table 1.5.

**Reclassified as Outstanding Issue 23.

†Assigned to Region IV by the low-power license.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.5 Combustible Gas Control in Containment

NUREG-0737 Item II.B.7 - Analysis of Hydrogen Control, and Item II.B.8 - Rulemaking Proceeding on Degraded Core Accidents

In Section 6.2.5 of the SER, the staff noted that the licensee needed to provide a system to control hydrogen generated from a postulated degraded-core accident. It was reported that the licensee intended to install a hydrogen ignition system (HIS) in the plant for this purpose, based on the Grand Gulf Nuclear Station (Grand Gulf) system design. Also, the licensee planned to submit plant-specific information, including relevant generic information developed by the Mark III containment Hydrogen Control Owners Group (HCOG) in which the licensee [Gulf States Utilities Company (GSU)] is a participant, in support of the River Bend design.

The Nuclear Regulatory Commission (NRC) published an amendment to the hydrogen rule (10 CFR 50.44) on January 25, 1985 (50 FR 3498). This amendment, which affects the River Bend plant, became effective on February 25, 1985. The amended rule requires that a hydrogen control system be provided and that the system be capable of accommodating, without loss of containment structural integrity, the amount of hydrogen generated from a metal-water reaction involving up to 75% of the active fuel cladding. Since the licensee has elected to install a hydrogen ignition system, similar to the system installed at Grand Gulf, the licensee is also required to demonstrate that systems and components necessary to establish and maintain safe shutdown are capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen.

Pursuant to and in accordance with the provisions of the rule, the licensee has submitted a preliminary analysis of the River Bend system design, and a schedule for meeting the requirements of the rule (see Table 6.3). This schedule indicates that the final analysis of the River Bend system design is to be based on the generic findings of the Mark III containment HCOG, supplemented by plant-specific design considerations.

The River Bend plant is one of four plants operating or under construction with a Mark III type of containment and which relies on an HIS as a means of controlling hydrogen during postulated degraded-core accident events. In its evaluation of the HIS for Grand Gulf, the staff concluded, subject to a related license condition, that the ignition system is effective in controlling hydrogen releases from the more likely degraded-core accident scenarios with acceptable consequences. The staff's evaluation is documented in Supplements 3 and 5 to the Grand Gulf Safety Evaluation Report (NUREG-0831). In this fourth supplement to the River Bend SER, the staff provides its evaluations of the preliminary analysis of the HIS for the River Bend Station. Although there are

geometric configuration similarities between the Grand Gulf and River Bend containment designs, there are dimensional and other differences that are reflected in the analysis. (Table 6.4 compares the Mark III containment characteristics with those of Grand Gulf, Perry, and River Bend.) A principal difference is that the Grand Gulf analysis of hydrogen control was influenced by assumptions regarding the plant's containment spray system which reduced containment atmosphere pressures and temperatures. The River Bend Station, however, does not include sprays but relies on fan coolers.

A significant part of the remaining work to be accomplished by the HCOG is to determine the effects of hydrogen burning in a continuous fashion such as a diffusion flame extending upwards from the surface of the suppression pool. Since there are no analysis codes available at this time that can be relied on to represent such a mode of hydrogen burning, the HCOG has developed a test program to be carried out in a quarter-scale test facility. Plant-specific configurations for each of the four Mark III plants are to be simulated in these tests. The test results are expected to establish a valid basis for the final analyses required by the rule.

(1) Hydrogen Ignition System

The River Bend hydrogen ignition system is designed to burn hydrogen at low concentrations, preventing the accumulation of high concentrations of hydrogen, which would threaten containment integrity. The HIS is a system of thermal igniters and ancillary equipment. The licensee has indicated that the system will be operable before plant operation exceeds 5% of full reactor power. This complies with the specific requirement of the rule [10 CFR 50.44(c)(3)(vii)(B)].

The igniters are designed to ensure burning of hydrogen in the unlikely event that substantial quantities of hydrogen are generated as a result of a postulated degraded-core accident (involving up to 75% equivalent metal-water reaction, as required by the rule).

The HIS consists of 104 igniter assemblies distributed throughout the drywell structure and the containment vessel. There are 18 igniters in the drywell structure and 86 igniters in the containment vessel that are distributed as follows: 12 igniters in the wetwell region [i.e., below the hydraulic control unit (HCU) floor at plant elevation 114 feet]; 54 igniters in the region above the HCU floor and below the refueling floor at plant elevation 186 feet; and the remaining 20 igniters in the upper dome region.

The igniter is a glow plug (commonly used in diesel engines) that is manufactured as Model 7G by General Motors AC Division. The River Bend igniter assembly design is identical to that installed in the Grand Gulf plant. The igniter is powered directly from a 120/12-V stepdown transformer that has multitap capability. Each igniter assembly consists of a 1/8-inch-thick stainless steel box that contains the transformer and all electrical connections. The igniter assembly is manufactured by the Power Systems Division of Morrison Knudsen. The igniter assemblies are classified as Class 1E and are seismic Category I.

The 104 igniters in the system are equally divided into two redundant groups; each group is powered from a separate Class 1E division power supply. There are ten circuits in each divisional power supply. The HIS is to be manually

activated from the reactor control room in accordance with emergency operating procedures.

The River Bend HIS is designed to promote the burning of hydrogen at lean concentrations and in a reliable manner. At least two igniters are located in each enclosed volume or area within the containment that are subject to possible hydrogen pocketing; each of the two igniters is powered from a separate power division. In open areas within the containment and drywell regions, igniter assemblies at the same elevation are alternated with respect to their Class 1E division power source.

The licensee has implemented preoperational and surveillance testing programs similar to the programs established and approved by the staff for Grand Gulf. These programs ensure the operability and functionality of the HIS, verifying the proper functioning of the controls, wiring, instrumentation, and critical components, along with adequacy of the operating igniters' surface temperature.

On the basis of its review of the design information provided, the staff finds that the design arrangement described above ensures that hydrogen will burn reliably and effectively and has an acceptable number and distribution of potential ignition points. Therefore, the staff concludes that the River Bend HIS is acceptable, subject to resolution of certain issues that are to be addressed in the licensee's final analysis, as provided for in the rule, and discussed in the sections that follow.

(2) Structural Capacity

The River Bend containment vessel is a pressure-retaining structure consisting of a freestanding steel cylinder, with a torispherical dome, secured to a steel-lined reinforced-concrete foundation mat. The steel containment structure ensures a high degree of leaktightness during normal operating and accident conditions. It is designed for a maximum internal pressure of 15 psig and a maximum external pressure differential of 0.6 psig. The containment cylinder has five external stiffening rings at various elevations (four in the concrete-filled annulus area between the steel shell and the shield building in the lower 20 feet and one about midheight on the steel shell), two airlocks for personnel access and one equipment hatch.

The licensee reported the ultimate capacity of the containment vessel in a letter from J. Booker to R. Tedesco, dated June 22, 1983. This report was supplemented by a letter from J. Booker to T. Novak, dated September 30, 1983, a letter from J. Booker to H. Denton, dated August 6, 1985, and a letter from J. Booker to H. Denton, dated August 8, 1985. The licensee also provided the engineering calculations (FAE-1095, dated December 15, 1982) for the staff review and responded to staff questions on August 7, 1985. The ultimate pressure capacity of the containment shell has been evaluated according to the requirements of ASME Code, Section III, Division I, Subarticle NE-3220, Service Level C stress limits using the Code-specified material values for the containment shell and personnel airlock door. Actual material values were used for the equipment hatch and personnel airlock bulkhead.

The steel shell of the containment vessel has a calculated internal pressure capacity of 53 psig, limited by the buckling in the knuckle area between the

shell and the dome. The pressure capacity of the limiting containment penetration is the personnel airlock door. The door capacity is 53 psig using the ASME Code, Service Level C stress limits. Because the licensee has demonstrated that the results of the River Bend steel containment structural integrity analysis meet the requirements of ASME Code, Section III, Division I, Service Level C stress limits, the staff concludes that the 53-psig pressure can be considered as a lower bound value and is a conservative value for the ultimate pressure capacity for the River Bend containment.

The licensee provided the negative ultimate internal pressure capacity by letter from J. Booker to H. Denton, dated June 25, 1984. The capacity is based on the provisions of the ASME Code Case N-284 and is limited to -4.8 psig by buckling of the steel shell. The negative pressure capacity of other components is larger than the capacity of the shell.

The drywell structure is a vertical reinforced-concrete cylinder that is topped by a flat circular slab. The top slab has a hole in the center over the reactor vessel that is closed with a steel torispherical shell called the drywell head. The cylinder is anchored to the containment reinforced-concrete basemat. Access to the drywell structure is through a combination equipment hatch and personnel door or a personnel airlock. The design pressure for the drywell structure is 25 psig internal and 20 psig external. The licensee stated in a letter from J. Booker to T. Novak, dated September 30, 1983, that the calculated static internal positive pressure-retaining capacity of the drywell structure is greater than 80 psig. The criteria used for defining failure was the yield strength of the reinforcing steel and a compression strain in the concrete of 0.003 inch per inch. The critical area is the junction of the top slab and the drywell cylinder.

The staff concludes the ultimate internal pressure for the River Bend containment steel shell is 53 psig for positive internal pressure and 4.8 psig for negative internal pressure. The ultimate internal positive pressure for the reinforced-concrete drywell structure is 80 psig.

(3) Hydrogen Generation for Postulated Degraded-Core Accidents

This section addresses the suitability of hydrogen generation rates employed by the licensee in its preliminary analysis of the River Bend containment pressure and temperature response to hydrogen combustion.

(a) Sequences Analyzed

The licensee has analyzed two degraded-core accident sequences. The first is a transient with an initial loss of all makeup water sources and a concurrent stuck open relief valve (SORV) and subsequent recovery of an injection source greater than or equal to 5000 gallons per minute (gpm) after a substantial time delay following reactor trip. The staff notes that this event is a surrogate for all transients with loss of makeup water. The SORV leads to faster core uncovery (three-fourths uncovered at about 40 minutes following reactor trip) than a transient with operable cycling relief valves (three-fourths uncovered at about an hour following reactor trip). The SORV case is somewhat more conservative than the transient with cycling relief valves since the decay heat

is somewhat higher at the earlier uncovery. The second sequence is a small-break loss-of-coolant accident (LOCA) inside the drywell structure with concurrent loss of all makeup water sources.

The SORV transient sequence is believed to be much more probable than the LOCA sequence. However, in the drywell LOCA sequence, some of the steam and hydrogen would be released directly into the drywell structure, rather than into the suppression pool. This, therefore, represents a potentially different challenge to the containment vessel.

(b) Hydrogen Generation

The boiling-water reactor (BWR) emergency procedures guidelines (EPG) with loss of all makeup water, independent of the sequence, i.e., independent of whether the event is a transient, transient with SORV, or LOCA, call for the operator to manually depressurize the plant using the automatic depressurization system (ADS). Hence these sequences all degenerate to a depressurized reactor with the core significantly uncovered. The core would heat up, with ultimately sufficiently high temperatures to cause Zircaloy reaction with passing steam yielding zirconium dioxide and hydrogen.

Following the BWR-EPG, the operator would depressurize the plant from a water level of about three-quarters of the core uncovered. The licensee, however, has conservatively assumed an earlier depressurization and blowdown to the three-quarters of the core-uncovered level at 2000 seconds. At that point, the core would be steam cooled. Subsequent boiloff of the remaining water in the core is the source of steam and hydrogen. Maximum hydrogen generation rates are achieved with the core 1 or 2 feet covered. Hydrogen generation decreases below about 1 foot covered because of a lack of steam. During the boiloff from the three-quarters uncovered (3 feet covered) level with no injection, hydrogen production is to a large degree limited by what is termed steam starvation. Subsequent massive reflood of the hot core causes a spike in the hydrogen production, but is rapidly terminated by the cooling associated with the reflood. These aspects have been reasonably modeled by the licensee.

The River Bend Station containment analysis is based on hydrogen generation for the transient sequence obtained from HCOG calculations which employed the BWR Core HEATUP Code, and on hydrogen generation for the drywell LOCA sequence obtained using the MAAP Code. The staff has made extensive comparisons of the HEATUP Code results to independent calculations of the transient sequence. These comparisons show good agreement and provide the basis for acceptance of the hydrogen generation predictions. The staff has not performed the same in-depth comparison to MAAP results and makes no finding at this time on the acceptability of these calculations for the drywell LOCA sequence. Because this sequence is considerably less probable than the transient sequence, it is the staff's judgment that there is no undue risk involved in deferring consideration of the LOCA sequence to the final analysis.

The evolution of the sequence described above is dependent on successful manual depressurization. Failure to depressurize to a constant pressure (about 2 atmospheres) in a timely manner would result in a slow continuing depressurization over the time frame of interest (2000 to 4000 seconds). This slow depressurization would result in the continual flashing of water in the lower elevations of

the core and lower plenum and a continued source of steam for hydrogen production. The staff accepts the licensee's scenario, however, which models rapid depressurization on the basis of the perceived reliability of the BWR 6 depressurization systems (ADS and the remaining complement of safety/relief valves (SRVs) which are manually operable from the control room) and the guidance provided by the EPGs.

The final hydrogen rule requires that the hydrogen control system can handle the hydrogen produced by oxidation of up to 75% of the equivalent active cladding. HEATUP Code calculations and independent staff calculations show that this degree of oxidation is improbable in an intact geometric core configuration that would exist if injection flow is restored early enough and not interrupted further. The licensee has referenced HCOG work which is based on a quasi-static debris bed coolability model. This model predicts a hydrogen generation rate of 0.1 lb/sec. This rate was added as a tail to the mechanistic HEATUP Code predictions. The integral of the HEATUP Code predicted hydrogen generation rate and the 0.1 lb/sec tail is set to yield 75% equivalent zirconium-water reaction. The staff accepts the 0.1 lb/sec rate for use in the preliminary containment analysis. Independent staff/consultant calculations of a repeated cyclic core uncovery, heatup, and quenching show a similar average rate over several cycles.

(4) Containment Response Analysis

The licensee has provided a preliminary analysis to support interim operation at full power until the final analysis is completed. Using the hydrogen and steam releases obtained from the BWR core HEATUP Code (see section above on hydrogen generation for degraded-core accidents), the licensee calculated the containment atmosphere transient using the CLASIX-3 Code. The CLASIX-3 Code is a multivolume containment code that is used to calculate the containment pressure and temperature response in the separate compartments. This Code has the capability to model features that are unique to Mark III containments while tracking the distribution of the atmosphere constituents, i.e., oxygen, nitrogen, hydrogen, steam. It should be noted that CLASIX-3 models deflagration (discrete-type) burning in the containment; diffusion-type (continuous) burning will be addressed in the licensee's final analysis, on the basis of results of the quarter-scale test program as noted earlier.

The licensee considered two types of accidents in its preliminary analysis of the HIS: a stuck-open relief valve (SORV) transient and a small-break LOCA in the drywell. The BWR core HEATUP Code steam and hydrogen releases are fed into the CLASIX-3 Code. For the SORV case, all the mass and energy releases are directed into the suppression pool. For the drywell break case, the mass and energy releases are directed to the drywell structure and into the suppression pool as determined by the MAAP Code.

The CLASIX-3 model used in this analysis simulates four compartments of the River Bend Station: the drywell volume, the wetwell volume (bounded by the HCU floor and the top of the suppression pool), the intermediate volume (bounded by the HCU floor and the refueling floor), and the containment volume (above the refueling floor). A River Bend containment unit cooler is modeled to only remove heat from the intermediate volume. With regard to hydrogen combustion, ignition was calculated to occur at 8% hydrogen concentration with 85% combustion completeness; propagation of burning into other compartments also was

assumed to occur at 8% hydrogen concentration with a flame speed of 6 feet per second. The River Bend SORV base case produces a transient in which the hydrogen is ignited in a series of burns in the wetwell volume. The CLASIX-3 Code predicts a sequence of 42 wetwell burns (approximately 40 pounds of hydrogen per burn). Figures 6.5 and 6.6 show the wetwell temperature and pressure profiles. Referring to the wetwell temperature transient, the third burn resulted in the highest wetwell temperature (2135°F) for the entire transient; this burn is a consequence of the peak hydrogen release. Otherwise, the remaining wetwell burns produce a peak wetwell temperature of about 1300°F.

At the end of the hydrogen release period, the hydrogen concentration in the containment volume does not reach the ignition setpoint of 8%; the average concentration of hydrogen in the containment volume is about 6.5%. The licensee induced the only containment burn at the lower concentration and this resulted in the most severe pressure excursion to approximately 24 psig.

For the base case analysis of a small pipe break in the drywell, the CLASIX-3 containment model used is similar to the SORV case except for the hydrogen/steam source terms. The results of the drywell break CLASIX-3 analysis produces a similar wetwell temperature profile (36 burns) as compared with the SORV case (42 burns). Also, at the end of the hydrogen release period, the hydrogen concentration in the containment volume does not reach the ignition setpoint of 8%; the concentration of hydrogen in the containment volume is about 7%. The licensee induced the only hydrogen burn in the containment volume at the lower concentration and this resulted in the most severe pressure excursion (to approximately 35 psig).

Previously submitted preliminary analyses performed by Mark III owners, i.e., Grand Gulf and Perry, used the CLASIX-3 wetwell thermal profiles (SORV) case as the atmospheric boundary condition for the equipment survivability analysis. This wetwell temperature profile was also applied to essential equipment outside the wetwell region, because it represents the bounding profile and those plants were able to accommodate this most severe environment. Containment sprays were assumed to be available for the Grand Gulf and Perry preliminary analysis. Water sprays, if available, are an effective way to reduce atmospheric temperatures. However, the licensee does not use containment sprays in the River Bend plant. As a result, the licensee found it necessary to refine its CLASIX-3 analysis and has provided a revised SORV case, which is discussed below.

The following enhancements were incorporated into the revised SORV River Bend CLASIX-3 analysis.

(a) The intermediate volume is divided into an upper and lower volume in which the HCU floor and its ceiling bound the lower intermediate volume, i.e., a five-volume model as compared with the previous four-volume model.

(b) Heat removal from operation of the containment unit cooler is assumed in the upper intermediate volume only.

(c) The hydrogen ignition control used for the wetwell volume remained at 8% with 85% completeness. However, the hydrogen ignition criterion for all volumes above the wetwell was set at 6% with 65% completeness. This is

predicated on the upward propagation behavior of hydrogen and the igniter located at the mid-plane of the annular floors of the containment structure.

(d) The non-mechanistic hydrogen release rate tail of 0.1 lb/sec was reduced to 0.078 lb/sec to reflect the lesser amount of zirconium in the core, using the ratio of River Bend fuel rods to the Grand Gulf core.

The revised River Bend SORV base case produces a transient in which the hydrogen is ignited in a series of burns confined to the wetwell volume (39 burns) and the lower intermediate volume (29 burns). Figures 6.7 and 6.8 show the respective volume temperature profiles.

In reviewing the overall CLASIX-3 Code containment modeling used in the preliminary analysis, certain assumptions need to be justified. The licensee will address these concerns in the final analysis and will submit the justifications to the staff for review. If the final containment environmental profiles are found to be more severe than the present analysis, the licensee will have to, and has committed to, confirm equipment survivability with the modified profiles or initiate appropriate equipment modifications as necessary. The staff concludes that the licensee's preliminary analysis of the containment constitutes a reasonable methodology for demonstrating the performance of the HIS on the basis of the above findings.

Testing of Wetwell Thermal Environment

In Grand Gulf SSER 5, the staff described some of the results of an earlier twentieth-scale HCOG test program. The objective of the twentieth-scale test was to provide a visual record of hydrogen combustion behavior in a Mark III containment enclosure. A principal finding was that when hydrogen flow rates of 0.4 lb/sec or greater are injected in the suppression pool, continuous burning of hydrogen in the form of a steady diffusion flame occurred above the suppression pool. However, the test facility was inadequate to permit scaleup of the thermal environment to an actual Mark III wetwell volume. Therefore, HCOG proposed a quarter-scale research test facility, and this is also described in the Grand Gulf SSER 5.

Preliminary results from the quarter-scale scoping tests reveal that the diffusion flame threshold is significantly lower, about 0.14 lb/sec, and steady flames can exist as low as 0.07 lb/sec under certain background conditions (HCOG Letter No. HGN-053, dated August 1, 1985). In these initial tests, the only deflagration burns observed have been the initial "lightoff" before the formation of diffusion flames. Also, the bulk hydrogen concentration was maintained below 5%, thus demonstrating effective performance of the igniters to burn at lean hydrogen mixtures. These results have a significant bearing on the extent to which the CLASIX-3 Code can be relied upon to predict containment temperature environments and further emphasize the importance of the test program.

(5) Equipment Survivability

By letters dated July 1 and 5 and August 5, 7, 16, and 19, 1985, the licensee submitted an evaluation of survivability of the essential equipment exposed to the thermal environment postulated in the containment vessel during hydrogen burn initiated by the hydrogen ignition system (HIS). Although this system was

designed to prevent high hydrogen concentration buildup by deliberate ignition of relatively low concentration of hydrogen/air/steam mixtures, the resulting release of thermal energy may still be sufficient to significantly increase the temperature of the equipment located in the containment vessel. Since some of this equipment is needed to ensure safe shutdown and containment integrity, the licensee is required to demonstrate that the essential equipment located inside the containment will survive the hydrogen burn environment resulting from operation of the HIS.

The licensee limited its preliminary evaluation to determining analytically thermal responses of selected pieces of essential equipment exposed to deflagration burn only. Thermal responses of the equipment exposed to diffusion flames on the surface of the suppression pool were not considered. The licensee, by letter dated July 1, 1985, has committed to provide this evaluation after the results from the HCOG experimental program become available. Survivability of the equipment was demonstrated by comparing temperatures obtained in thermal response studies with the corresponding qualification temperatures for the same equipment.

(a) Safety-Related Equipment

The selection of the equipment that has to survive a hydrogen burn was based on its function during and after an accident. In general, all the equipment located in the containment vessel which is included in the following four categories of systems is considered to be essential to plant safety:

- equipment and systems required to mitigate the consequences of the event
- equipment and structures required to maintain the integrity of the containment pressure boundary
- systems and components required to re-cover the core
- instrumentation and systems required to monitor the course of the event

However, the licensee excluded from the list certain equipment on the basis of its failure mode or active safety function before exposure to the hydrogen burn environment.

Specific exclusion criteria are listed below:

- components that have performed their active safety function before a hydrogen burn
- isolation valves that remain in the closed position, i.e., fail closed or "as is"
- isolation valves that are open during post-LOCA, fail in the "as is" position, and have a redundant motor-operated isolation valve outside containment for functional backup
- check valves that are qualified for reactor pressure and temperature with no safety-related instrumentation or electrical function and are assumed to survive a hydrogen burn mechanically

- equipment and/or components that fail in a safe condition with no subsequent functional requirement

- manually operated valves or dampers that remain in the "as is" position, i.e., normally open or normally closed

Using these two sets of criteria, the licensee has prepared a list of equipment for which survivability needs to be evaluated. Each piece of equipment included in the list consisted of several individual components each of which had to meet survivability criteria. Since many of these components were similar in different systems, the licensee has restricted the survivability evaluation to the components that are limiting to the survivability of equipment and that are most sensitive to temperature changes. This reduced considerably the number of thermal response analyses that had to be performed. The six components of essential equipment for which thermal response analyses were made are listed below.

- hydrogen igniter
- Limitorque valve operator 0-13 HP Reliance motor
- Target Rock solenoid valve
- Crosby pilot air solenoid valve (ADS)
- Rosemount pressure/level transmitter
- power cable

The staff has reviewed the licensee's criteria for selecting the essential equipment and has determined that the criteria should be modified to also include equipment whose failure could adversely affect the function of essential equipment required for the hydrogen burn event or could mislead the operator, and the equipment required by the EPGs. The licensee has committed to resolve this item in the final analysis through the HCOG program. The staff's view is that these new criteria are not likely to result in additional equipment being identified, or, these new criteria will not result in any new equipment type being identified for which equipment operability has not been analyzed. The staff has also asked the licensee to include the check valves in the list of essential equipment and demonstrate their survivability. The licensee, in its letter of August 16, 1985, has provided the basis to demonstrate the survivability of check valves in a hydrogen burn event. Therefore, on these bases, the staff finds the essential equipment list for River Bend acceptable for the preliminary analysis. The staff also reviewed the list of limiting components for which the licensee has developed computer models and performed analyses. Determination of survivability of these components will suffice for establishing the survivability of the equipment listed by the licensee for the preliminary analysis purposes.

(b) Thermal Environment

In the preliminary analysis, the licensee limited evaluation of equipment survival to the thermal environments produced by deflagration hydrogen burning only. No environments created by diffusion burning were considered. The temperatures in different locations in the containment were determined by the CLASIX-3 Code as described earlier in this evaluation.

(c) Thermal Response Analyses

Thermal response analyses were performed analytically for all six components of essential equipment identified previously. The results of the thermal environment calculations identified above were used for these analyses. Thermal energy transfer between hot gases and equipment was assumed to occur by convection and radiation. The licensee has conservatively assumed a gas velocity of 12 feet per second in calculating convection heat transfer coefficients. The analysis also assumed thermal energy transfer to heat sinks consisting of structural elements of the containment vessel. Heat transfer through the air spaces inside the equipment was assumed to occur by radiation and conduction with enhanced thermal conductivities of gas in order to account for some convection transfer which may be significant in larger gas spaces.

The temperatures of different components were calculated using the HEATING-6 Code. This is the latest version of the Code incorporating certain modifications which improve its efficiency. With one exception, two-dimensional modeling was employed; therefore, the components have to be represented by relatively simple geometries. The licensee developed such models which, despite their simplicity, included all significant heat transfer characteristics. A one-dimensional model was used in one case (Crosby pilot valve) where the symmetry of the object permitted the use of this approach.

The analyses included several conservative assumptions: orientation of the component inside the containment was chosen to maximize convection heat transfer from hot gases, the emissivities and absorptivities of different surfaces were set at conservatively high values, and each piece of equipment was assumed to be surrounded on its exposed sides by hot gas with sufficient depth to maximize its emissivity. In addition, special care was taken to ensure that the highest temperature reached by the sensitive component was calculated. Internal heat generation was assumed only in the case of the igniter transformer where it has some effect on the calculated temperature. The staff has reviewed the methodology used by the licensee and finds it to be a realistic representation of the thermal phenomena which take place during a postulated hydrogen burn. The methodology, though simplified, has sufficient conservatism that analytical results should encompass all the temperatures which may be reached by the equipment during a postulated deflagration burn of hydrogen.

The staff has also performed an independent evaluation of the thermal response of igniters using the HYBER Code developed by Sandia. Two cases were considered. In one, the igniter was located in the drywell structure and in the other, in the wetwell structure. In both these cases, the temperatures calculated by the HYBER Code were less than the temperatures determined by the licensee using the HEATING-6 Code.

(d) Survivability of Essential Equipment

The acceptance criterion used for evaluating survivability of essential equipment is based on its qualification temperature and the duration for which the temperature is maintained. The equipment located in the containment will survive hydrogen burning if the temperature reached by its most sensitive component will not exceed the temperature reached by this component during qualification

tests. During these tests, the actual temperature reached by the tested equipment was not measured and qualification temperature was the temperature of thermal environment to which the tested equipment was exposed; therefore, there is no direct way to determine the actual qualification temperature reached by the limiting components. However, environmental qualification tests are typically conducted for extended periods of time and the equilibrium surface temperature should achieve thermal equilibrium with the test chamber during the tests; thus, the temperature of the test environment can be used as a close approximation to the temperature reached by the sensitive components. The staff finds this assumption acceptable. Because of several conservative assumptions existing in the thermal response analysis, the staff is of the opinion that use of the qualification temperature by the licensee as a criterion for evaluating the survivability of limiting components is acceptable.

In order to evaluate the survivability of all six components of essential equipment identified previously, the licensee compared analytically calculated temperatures to the corresponding qualification temperatures. For the equipment located in the drywell structure, it was shown that in all cases the qualification temperatures exceeded the temperatures reached by the most sensitive components of the equipment. For the equipment located in the wetwell volume which consisted of igniters and power cables, the temperatures determined analytically exceeded the qualification temperatures. Survivability of this equipment for the full (75% metal-water reaction) hydrogen burning could not, therefore, be demonstrated. In its submittal of August 16, 1985, the licensee proposed several techniques by which survivability of this equipment could be ensured. The proposed methods consisted of insulating the equipment and providing thermal shielding, or of relocating the equipment. Quantitative evaluations demonstrating the feasibility of some of the methods were given. Although failure of the wetwell igniters themselves during a degraded-core accident would not prevent safe shutdown, it would alter the thermal environment. The licensee has not analyzed the possible impact this might have on the survivability of essential equipment located on the HCU floor.

At this time, therefore, the licensee has not demonstrated that the wetwell igniters and power cables can survive the complete CLASIX-3 wetwell thermal environment when using the HEATING-6 thermal response code. It should be noted that computer codes, in general, involve the use of simplifying assumptions to mathematically represent complex physical phenomena. In particular, the CLASIX-3 Code may not realistically model the hydrogen burn phenomena in a Mark III wetwell. This conclusion is based on the preliminary results of the quarter-scale test program, which is described above. The HCOG program plan contains provisions for evaluating potential physical modifications to their respective containments to ensure equipment survivability if this should be necessary.

In the staff's judgment, it would be inappropriate and premature to require immediate equipment enhancements to protect the igniters in the wetwell region, based solely on the CLASIX-3 analysis for River Bend.

The equipment located in the intermediate volume above the wetwell structure was first evaluated using the wetwell environment. In all cases, qualification temperatures were reached after numbers of burns that were smaller than the total number of burns predicted to occur in the wetwell structure. However, this environment was more severe than the environment expected to exist in the intermediate volume; therefore, no direct conclusion about survivability could

be reached. Hence, the licensee, in its August 19, 1985, submittal, performed a revised CLASIX-3 analysis to determine the realistic thermal environment in the lower intermediate volume (LIV).

The licensee performed thermal response analysis of the Rosemount transmitter and Reliance motor, using the thermal environment obtained in the LIV. By showing their survival in this environment, the licensee demonstrated survivability of all other equipment located in the intermediate volume.

Although in this analysis, the licensee used temperature profiles that may result from postulated events, there is a location in the containment above the suppression pool where, under certain circumstances, the equipment could be exposed to a diffusion flame environment produced by a continuous burn of hydrogen released from the suppression pool. The licensee will address this problem in its final analysis after the results of the HCOG test dealing with this phenomenon become available.

(e) Pressure Effects on Essential Equipment

The licensee has performed a preliminary analysis to show that the effect of pressures generated during the hydrogen burn would not affect the survivability of the essential equipment. This preliminary analysis indicated that with two exceptions the pressure during qualification bounds the calculated pressures the equipment could experience during hydrogen combustion. The two exceptions are the containment unit cooler motor and the fan motor in the hydrogen-mixing system. The licensee has used the similarity to other motors which are qualified to higher pressure as the basis to demonstrate that these motors will survive the pressure created by the hydrogen burn event. It should also be noted that the 35-psig pressure spike occurs because of the forced burn at the end of the CLASIX-3 run to burn off all remaining hydrogen at the same time. The staff has reviewed the licensee's evaluation and finds that there is reasonable assurance that the essential equipment will survive the pressure generated during a hydrogen burn event.

(f) Conclusions on Equipment Survivability

After reviewing the licensee's analysis of equipment survivability, the staff concludes that the licensee has provided sufficient information necessary for preliminary analysis purposes to show that the equipment required to ensure safe shutdown conditions and containment integrity will survive the environment created by the burn of hydrogen generated during the more likely events in which substantial quantities of hydrogen are generated. This conclusion is based on the following findings:

- The list of equipment provided in the submittal included all relevant essential equipment.
- The limiting components selected for the analysis characterized adequately the essential equipment on the list.
- The analytical methods used by the licensee adequately calculate thermal response of equipment, based on the postulated thermal environment.

- The comparison of analytically determined thermal responses to the corresponding qualification temperatures for some sample components has indicated that these temperatures will not be exceeded during a hydrogen burn.

- It was satisfactorily demonstrated that the essential equipment will survive the pressure spike predicted to occur during hydrogen burn.

The licensee has yet to demonstrate the survivability of igniters and power cables located in the wetwell volume for a full (75% metal-water reaction) hydrogen burn. On the basis of the evaluation described above, it is the judgment of the staff that the effect of this source of thermal energy on equipment thermal response is significantly reduced by the shielding provided by different structural components and, therefore, there is no undue risk to the public health and safety to defer resolution of this matter pending the outcome of the HCOG test program and the final analysis.

(6) Overall Conclusions

The preliminary analysis performed by the licensee provides a satisfactory basis to support interim operation at full reactor power until the final analysis has been completed. There is reasonable assurance that the River Bend hydrogen ignition system will act to control the burning of hydrogen so that there is adequate protection against containment failure. On the basis of the preliminary analysis performed by the licensee, the staff finds that the peak pressures as a result of igniter-induced burns will be less than the containment pressure capacity (53 psig).

As a result of that preliminary analysis, there are residual staff concerns that need to be resolved in the final analysis. The licensee has indicated that it will rely on the ongoing experimental and analytical program, as set forth by the HCOG, to resolve the staff's residual concerns in a manner that is expected to resolve them for the River Bend plant. The following are the key staff concerns that the licensee has committed to address in the final analysis of the hydrogen control system design for River Bend.

(a) hydrogen-steam release histories that are representative of the degraded core accident sequences chosen by the licensee and discussed above

(b) containment diffusion flame environment; reliance on data from the quarter-scale test facility

(c) containment environmental analysis as related to deflagration-type burning, e.g., effects on transients when considering operational drywell mixers and buoyancy effects

(d) drywell environmental analysis, deflagration combustion, and the assessment of possible inverted diffusion flames

(e) confirmation of equipment survivability determined from the preliminary analysis discussed above, using modified thermal profiles as applicable

(f) development of combustible gas control emergency procedures guidelines

The staff will continue to work with the licensee through the HCOG and will report its findings following receipt of the final analysis to be submitted by the licensee at the conclusion of the HCOG program.

The staff concludes that the licensee satisfactorily meets the requirements of 10 CFR 50.44, applicable at this stage, and permits operation in excess of 5% of rated power.

Pursuant to 10 CFR 50.44, the hydrogen control rule, the licensee provided a schedule for revolving long-term hydrogen control issues. The staff will report on this schedule in a future SER supplement.

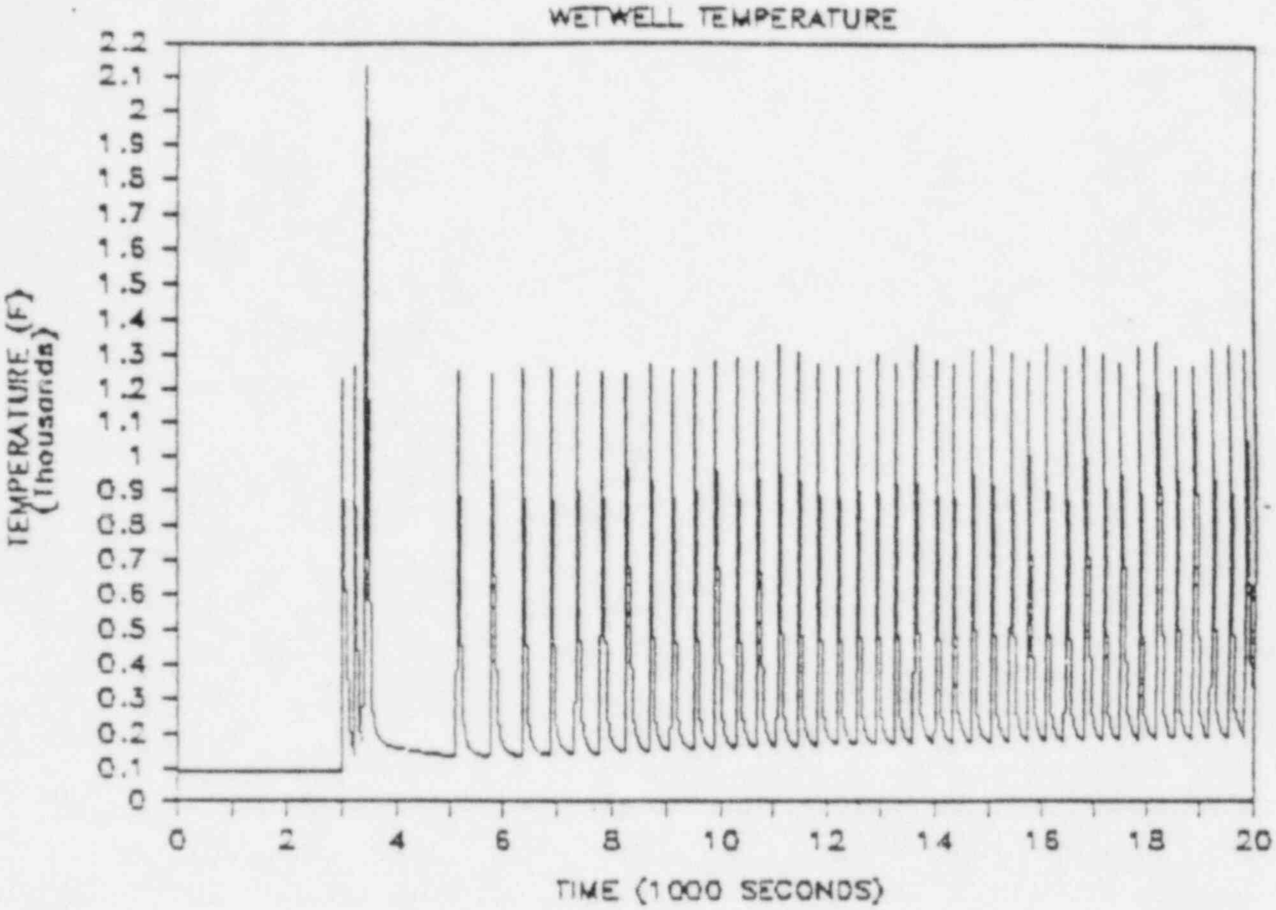

Figure 6.5 Wetwell temperature: Stuck-open relief valve, base case
Source: Licensee's letter of June 7, 1983, Figure 3

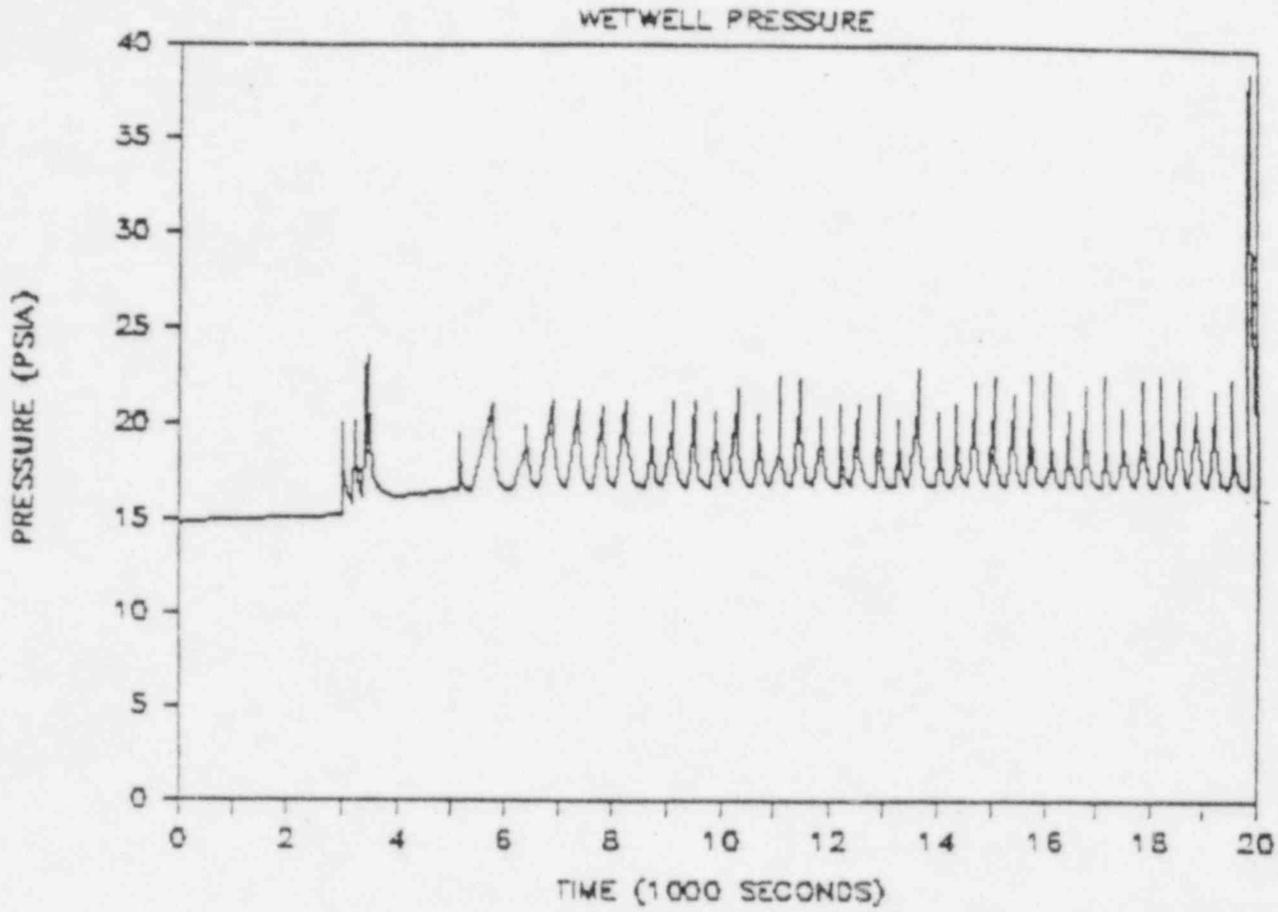

Figure 6.6 Wetwell pressure: Stuck-open relief valve, base case
Source: Licensee's letter of June 7, 1985, Figure 7

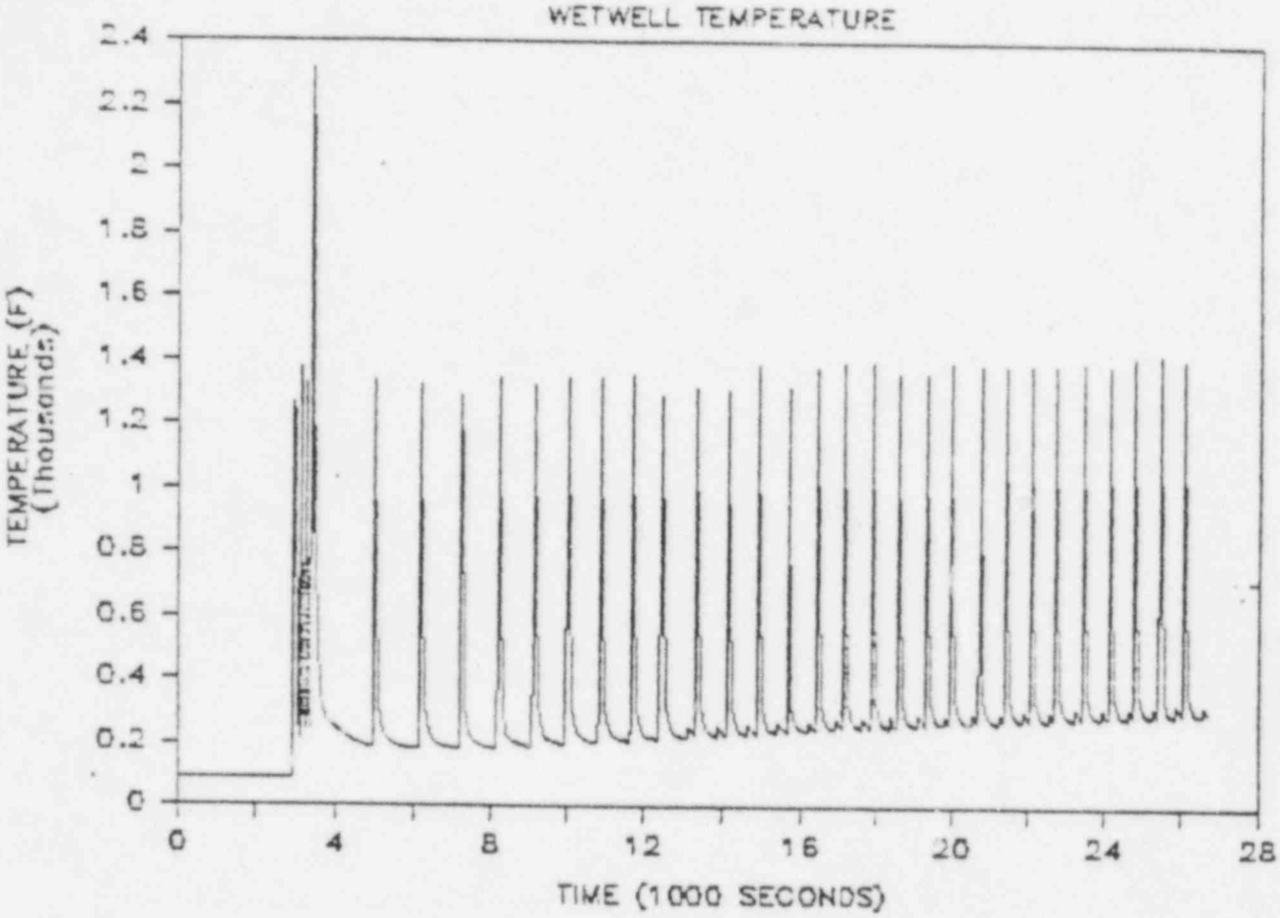

Figure 6.7 Wetwell temperature: Stuck-open relief valve, revised base case
Source: Licensee's letter of August 19, 1985, Figure 3

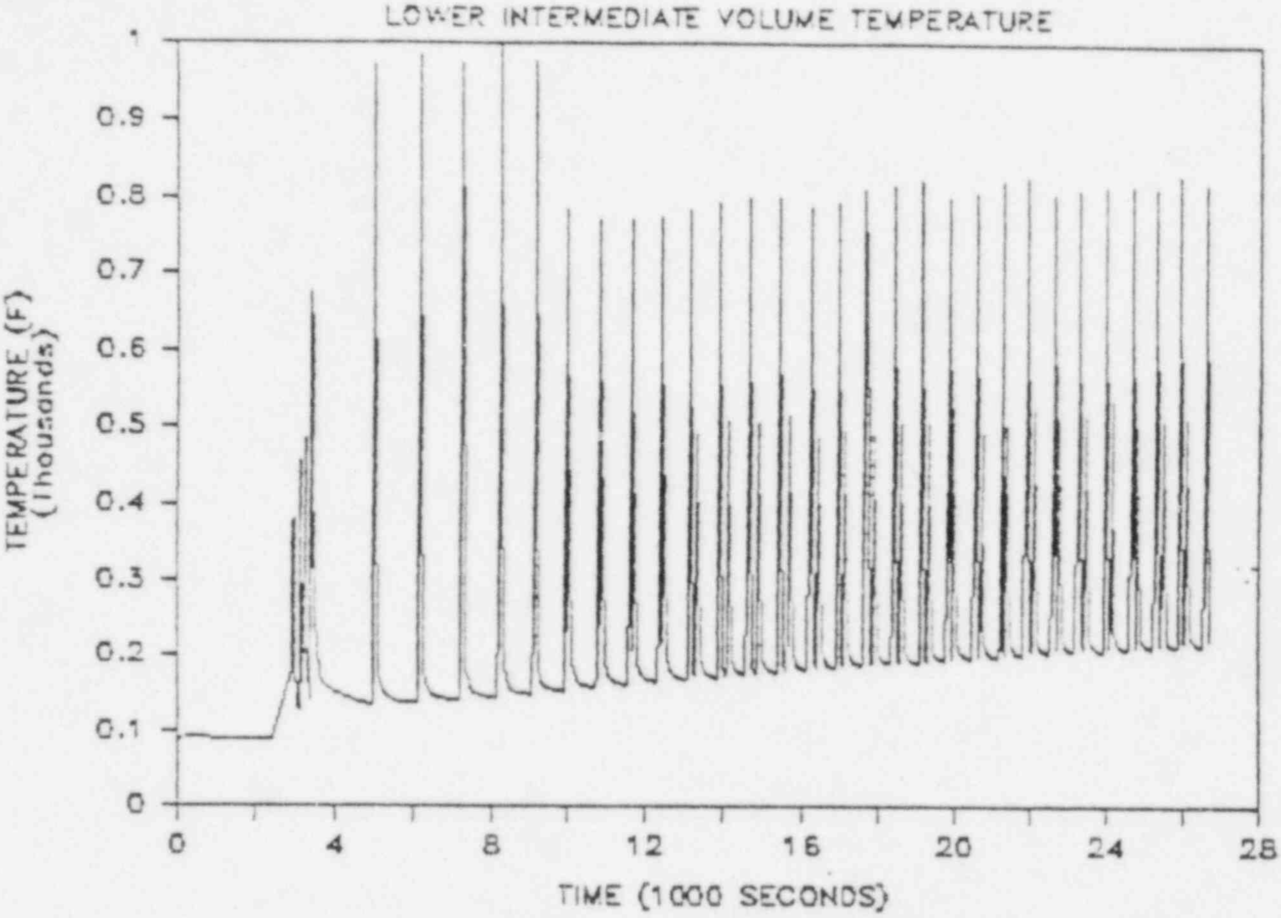

Figure 6.8 Lower intermediate volume temperature, revised base case
Source: Licensee's letter of August 19, 1985, Figure 4

Table 6.3 GSU submittals containing a preliminary analysis of the River Bend system design and a schedule for meeting the requirements of 10 CFR 50.44 as amended on January 25, 1985

Date	Description of submittal
June 22, 1983	Partial responses to NRC's 3/30/81 letter concerning "Ultimate Capacity Analysis of Mark III Containments"
September 30, 1983	Final responses to NRC's 3/30/81 letter concerning "Ultimate Capacity Analysis of Mark III Containments"
June 25, 1984	Responses to requests for additional information on combustible gas control system concerning containment negative pressure capacity
February 1, 1985	Responses to RAIs on combustible gas control system
June 4, 1985	Program for demonstrating compliance with final rule
June 7, 1985	H_2 burn response (CLASIX-3) analysis
June 11, 1985	H_2 control system activation criteria
June 26, 1985	Schedule for meeting the requirements of the final rule on H_2 control
July 1, 1985	"Preliminary Essential Equipment Survivability Report"
July 5, 1985	Revision of drywell break case CLASIX-3 analysis as requested during GSU/NRC staff meeting on 6/13/85
July 5, 1985	GSU's report, "Hydrogen Deflagration Pressure Effects on Equipment," requested during GSU/NRC staff meeting on 6/13/85
August 5, 1985	Additional information as requested during GSU/NRC staff meeting on 7/24/85
August 7, 1985	Responses to requests for additional information on essential equipment survivability report
August 16, 1985	Feasibility of equipment enhancements for survivability of hydrogen burn event
August 19, 1985	Revisions of SORV CLASIX-3 base case and equipment survivability analyses

Table 6.4 Comparison of BWR Mark III containment characteristics

Characteristic	Grand Gulf	Perry	River Bend
Rated thermal output, MWt	3,833	3,579	2,894
Number of fuel bundles	800	748	624
<u>Drywell structures</u>			
Design pressure, psig	30	30	25
External design pressure, psid	21	21	20
Air volume, ft^3	270,000	277,685	236,196
Suppression pool volume (includes vents), ft^3	1.3E4	1.12E4	1.3E4
Suppression pool surface area, ft^2	553	482	522
Holdup volume, ft^3	50,000	40,564	20,353
Holdup surface area, ft^2	3,145	2,617	2,564
<u>Containment vessel</u>			
Design pressure, psig	15	15	15
Ultimate pressure capacity, psig	56	50	53
External design pressure, psid	3	0.8	0.6
Total air volume, ft^3	1.4E6	1.141E6	1.192E6
Air volume below hydraulic control unit floor, ft^3	151,644	181,626	153,792
Suppression pool volume, ft^3	1.24E5	1.06E5	1.28E5
Suppression pool surface area, ft^2	6,667	5,900	6,408
Upper pool makeup volume, ft^3	36,380	32,830	0
Containment spray flow rate (1 train), gpm	5,650	5,250	0
Number of loss-of-coolant-accident vents	135	120	129

APPENDIX B

BIBLIOGRAPHY

Hydrogen Control Owners Group, Letter No. HGN-053 from S. Hobbs to R. Bernero (NRC), Subject: Hydrogen Control, August 1, 1985.

U.S. Nuclear Regulatory Commission, NUREG-0831, "Safety Evaluation Report Related to the Operation of Grand Gulf Nuclear Station, Units 1 and 2," Supplement No. 3, July 1982; Supplement No. 5, August 1984.

APPENDIX D

ACRONYMS AND INITIALISMS

ADS	automatic depressurization system
ATWS	anticipated transient without scram
BWR	boiling-water reactor
CEPCO	Cajun Electric Power Cooperative
CFR	Code of Federal Regulations
DRMS	digital radiation monitoring system
ECCS	emergency core cooling system
EPA	electrical protection assembly
EPG	emergency procedures guidelines
ERIS	emergency response and information system
ESF	engineered safety feature
FR	Federal Register
GSU	Gulf States Utilities Company
HCOG	Hydrogen Control Owners Group
HCU	hydraulic control unit
HED	human engineering discrepancy
HIS	hydrogen ignition system
HPCS	high-pressure core spray
HVAC	heating, ventilation, and air conditioning
LIV	lower intermediate volume
LOCA	loss-of-coolant accident
LPCS	low-pressure core spray
NMS	neutron monitoring system
NRC	Nuclear Regulatory Commission
PVLCS	penetration valve leakage control system
QA	quality assurance
RCIC	reactor core isolation cooling
RCIS	rod control and isolation system
RCS	reactor coolant system
RHR	residual heat removal
RPM	radiation protection manager
RPS	reactor protection system

SCLC	standby liquid control system
SER	Safety Evaluation Report
SORV	stuck-open relief valve
SRV	safety/relief valve
SSER	Supplement to the Safety Evaluation Report
TDI	Transamerica Delaval, Inc.
TMI	Three Mile Island Nuclear Station

APPENDIX E

PRINCIPAL STAFF CONTRIBUTORS

Name	Title	Review Branch
H. Garg	Electrical Engineer	Equipment Qualification
A. Notafrancesco	Containment Systems Engineer	Containment Systems
K. Parczewski	Senior Chemical Engineer	Chemical Engineering
H. Polk	Structural Engineer	Structural and Geotechnical Engineering
J. Rosenthal	Section Leader	Reactor Systems

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)
NUREG-0989
Supplement No. 4

2. TITLE AND SUBTITLE
Safety Evaluation Report related to the operation of River Bend Station

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
September	1985

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
September	1985

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Licensing
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Same as 7. above

11a. TYPE OF REPORT
Safety Evaluation Report

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES
Pertains to Docket No. 50-458

13. ABSTRACT (200 words or less)

Supplement No. 4 to the Safety Evaluation Report for the application filed by Gulf States Utilities Company as applicant and for itself and Cajun Electric Power Cooperative, as owners for a license to operate River Bend Station has been prepared by the Office of the Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in West Feliciana Parish, near St. Francisville, Louisiana.

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