GULF STATES UTILITIES COMPANY

RIVER BEND STATION POST OFFICE BOX 220 ST. FRANCISVILLE, LOUISIANA 70775

AREA CODE 504 635-6094 346-8651

July 8, 1985
RBG- 21484
File Nos. G9.5

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 for your review (Enclosure 2) are revisions to the River Bend Station Final Safety Analysis Report (FSAR) Section 6.2 and Appendix 6A. Enclosure 1 provides a discussion of each of these revisions including their effect, if any, on the Safety Evaluation Report. No revisions of the proposed Technical Specifications are expected to be required. These FSAR revisions will be included in a future amendment.

Sincerely,

J. E. Booker

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

JEB/ERG/je

Enclosures (2)

3001
1/1

ENCLOSURE 1

A. Page 6.2-54 has been revised to indicate that some relief valves in the containment heat removal system may be tested in place, as allowed by the ASME Code. No affect on the SER or the proposed Technical Specifications is expected.

B. Pages 6.2-58 and 6.7-9 have been revised to reflect RBS design. Individual MSIV leakage is not a concern because dose consequence analyses are based on leakage rates for a group of valves served by the main steam positive leakage control system; therefore, valve group leakage rates are specified. No effect is expected on the SER or the proposed Technical Specifications.

C. Page 6.2-94 has been revised to accurately reflect the drywell bypass leakage test parameters specified in the proposed Technical Specifications. No revisions are expected to be required for the SER.

D. Page 6.2-60a has been revised to clarify the term "constant security surveillance". The SER discussion on Page 6-23 may require similar clarification. No revisions are expected to be required for the proposed Technical Specifications.

E. Pages 6.2-58 and 59 have been revised to delete the reference to a separate shield building inleakage text. The required inleakage is demonstrated through the surveillance requirements associated with proposed Technical Specification 3/4.6.5. No revisions are expected to be required for the SER.

F. Page 6.2-74a and the notes for Table 6.2-40 have been revised to provide consistency of terminology with the design documents. No effect is expected on either the SER or the proposed Technical Specifications.

G. Information designated as "later" in Tables A.6A.7-1 and A.6A.7-2 and Figure A.6A.7-3 has been provided. This information is not expected to affect either the SER or the proposed Technical Specifications.

H. Attachment L has to Appendix 6A, Page L.6A-2, has been revised to show applicability of GESSAR-II vent clearing velocities to RBS. No effect is expected to either the SER or the proposed Technical Specifications.

I. The responses to Questions 440.40 and 440.44 have been updated to reflect previous transmittals. No impact is expected on either the SER or the proposed Technical Specifications.

components is indicated by installed instrumentation.
21 Relief valves are either removed for bench tests and setting adjustments or tested in place for setpoint verification, as scheduled, at refueling outages.

Operational sequencing of the LPCI made of the RHR system is tested after the reactor is shut down and the RHR system has been drained and flushed.

Further information on the testing and inspection of the components of the containment heat removal systems may be found in Sections 5.4.7.4, and 9.4.6.4 and in Chapter 14.

6.2.2.5 Instrumentation Requirements

The details of the instrumentation are provided in Sections 7.3.1.1.6, 7.3.1.1.7, and 9.4.6.5.1. The suppression pool cooling mode of the RHR system is manually initiated from the main control room.

6.2.3 Secondary Containment Functional Design

The secondary containment consists of the shield building, the fuel building, and the auxiliary building. The secondary containment is provided with leakage and filtration systems and is subjected to individual tests in accordance with the procedures specified in the Technical Specification. The secondary containment structures house the refueling equipment, safety-related equipment required for safe shutdown of the plant, and equipment, components, piping, cables, and instrumentation necessary for power generation. The secondary containment is of Seismic Category I design. Refer to Section 3.8.4 for a discussion of secondary containment structural design. The containment/drywell purge system (Section 9.4.6) is provided for purging the containment and drywell volumes during normal operation and shutdown. The annulus (shield building primary containment annulus) pressure control system (Section 9.4.6) maintains the annulus at negative pressure during normal operation. The standby gas treatment system (SGTS) (Section 6.5) maintains the annulus and auxiliary building at a negative pressure and provides cleanup of the potentially contaminated annulus and auxiliary building volumes during high radiation conditions and following a design basis accident (DBA). The fuel building charcoal filtration system (Section 9.4.2) maintains the fuel building at negative pressure and filters the effluent following a DBA and during high radiation conditions. In addition, the annulus mixing system is provided to thoroughly mix any leakage of iodine and noble gases (from the primary containment) to the annulus during a DBA and

If the SGTS filter trains are not treating the annulus
atmosphere or the exhaust air of the shielded compartments
in the auxiliary building, the containment and drywell purge
can be manually diverted through both SGTS filter trains.
By utilizing both SGTS filter trains, a maximum of 25,000
cfm of containment/drywell purge air can be processed by the
filter trains (Section 9.4.6).

The SGTS is designed to maintain a negative pressure of at
least 0.50 in W.G. in the annulus during post-LOCA
operation. With the annulus at a negative pressure, any
potential leakage is directed inward (away from the shield
building). Therefore, if a primary containment DBA occurs,
7 airborne radioactivity which exfiltrates the steel primary
containment is collected and passed through a filter train
of the SGTS before being released.

Potential paths exist for bypass of the annulus mixing
system and the SGTS. Potential leakage paths and measures
taken to mitigate their consequences are discussed in
18 Sections 6.7 and 9.3.6. The inleakage rates for the MS-PLCS
21 13 valves are provided in the Technical Specifications, and
discussed in Section 6.7.5.

Since the annulus and the auxiliary building are exhausted
via the SGTS filter train following a LOCA, the only leakage
considered to bypass the SGTS is that which leaks to areas
other than those previously mentioned. This would be from
systems listed in Table 9.3-3. The PVLCS is designed to
minimize this potential leakage in compliance with the
guidelines of Regulatory Guide 1.96 as described in
Section 9.3.6.

The primary containment and penetration valve leakage are
monitored during the periodic tests of the containment and
during the tests to measure local leakage, as discussed in
Section 9.3.6.

The integral welded design of the guard pipes precludes
leakage from the drywell into the containment portion of the
main steam tunnel (see Fig. 3.8-4 for the sleeved
penetration design). The electrical penetrations in the
primary containment can leak only into the annulus and this
leakage is treated by the SGTS.

The maximum inleakage rate across the shield building
barrier when the annulus is at a pressure of -3 in W.G. is
21 2,000 cfm. During normal operation, the annulus inleakage
approximately equals the exhaust capability of the annulus
pressure control system. The exhaust air is not diverted

through the SGTS unless it is radioactive. If the leak rate is actually less than 2,000 cfm, the initial pressure is at a value lower than -3 in W.G. (e.g., -4 in W.G.). 16

Two full-capacity SGTS exhaust fans are provided, each powered by a separate standby diesel generator. The DBA is assumed to occur with the annulus at its maximum normal operating conditions, namely, -3 in W.G. and 2,000 cfm inleakage. If a DBA occurs along with loss of offsite power and if a standby diesel generator also fails to start, the other standby diesel generator is available approximately 30 16 sec later (i.e., when the generator is up to speed). The SGTS fan then receives power from this standby diesel generator and is available within 38 sec after the DBA (i.e., 30 sec plus 8 sec for the fan to get up to speed). 16 The design flow rate of the SGTS in the post-accident mode is 12,500 cfm, which is equal to the maximum estimated flow rate being exhausted from the annulus and the shielded compartments in the auxiliary building during a DBA.

6.2.3.2.2 Annulus Mixing System

The annulus mixing system is provided for a thorough mixing of any leakage from the primary containment to the annulus, while the annulus is being maintained at a pressure of -0.50 in W.G. by the SGTS. Upon receipt of a LOCA or high radiation signal from the radiation monitor(s), the annulus mixing system is automatically actuated by starting the annulus mixing fans. For a detailed description of this system and its components, see Section 9.4.6.

6.2.3.2.3 Fuel Building Charcoal Filtration System

The fuel building charcoal filtration system is designed to limit the release of airborne radioactivity to the environment and maintain the building at a pressure of at 19 least -0.25 in W.G. following an accident. Regulatory Guide 1.52 is used as a basis of design for the fuel building charcoal filtration system. See Table 6.5-1 for a compliance summary. For a detailed description of the fuel building charcoal filtration system and its components, see Section 9.4.2.

post-LOCA leakage paths to the outside atmosphere, see Table 6.2-39. 15

6.2.3.2.7 Openings to Secondary Containment

All openings leading to the secondary containment are equipped with alarms, except as noted below:

1. Shield building personnel and equipment hatch at elevation 95 ft 0 in. 11

2. Personnel access door (A97/2) in the exterior wall of the auxiliary building at elevation 97 ft 0 in.

Except for these two doors, doors leading to the secondary containment have alarm capability to the security areas in the normal switchgear buildings and primary access point (PAP) facility, which are manned 24 hr a day by security personnel.

The shield building equipment hatch requires more than one person to open. Because of its location, the hatch is within view of all personnel entering and leaving the 21 protected area, and is viewed by the protected area security motor patrol at least once every 2 hr. The hatch is equipped with a balanced magnetic switch.

The personnel access door (A97/2) in the south wall of the auxiliary building leads to miscellaneous platforms above and below elevation 97 ft 0 in. All other access openings 11 leading into this area from outside are equipped with alarms, thus negating the need for alarms for this door.

6.2.3.3 Design Evaluation

The SGTS and fuel building charcoal filtration system maintain the secondary containment at a negative pressure with respect to environment following a design basis LOCA, ensuring that any leakage from the primary containment does not escape to the environment.

The SGTS and fuel building charcoal filtration system incorporate two 100 percent capacity exhaust fans. The redundant fans and dampers are connected to separate standby buses which are capable of being supplied from either normal or standby power sources. The two redundant, full capacity filter trains of the SGTS and fuel building charcoal filtration system ensure that no fission products are released directly to the environment. Temperature monitors and differential pressure gauges warn of filter failure.

All equipment is designed to Seismic Category I, Safety Class 2 and 3 criteria and is housed in Seismic Category I structures. The equipment is located in areas accessible to maintenance personnel.

Radiation monitors are provided in the annulus exhaust duct to continuously monitor the radiation level of the discharged air. During normal operation, the annulus is exhausted through the annulus pressure control system. However, the annulus exhaust is automatically lined up to the standby gas treatment system, upon the following contingencies:

Group No.	Isolation Signal Code	Description	
			19
1	B	Reactor Vessel Low Water Level 2	19
	K	High Drywell Pressure	
2	W	Reactor Pressure Low	19
	X(Note)	RCIC Isolation Signals	
3	W	Reactor Pressure Low	19
	K	High Drywell Pressure	
4	Y	MS-PLCS Air Line Header Flow High	21
	Z	MS-PLCS Air Line Header & Steamline Differential Pressure Low	19
5	C	Reactor Vessel Low Water Level 3	
	L	High Reactor Pressure	18 19
	R	RHR System Equipment Area High Ambient Temperature	
	S	RHR System Equipment Area High Differential Temperature	19 11
6	A	Reactor Vessel Low Water Level 1	19
	D	High Main Steam Line Flow	
	E	High Main Steam Line Radiation	
	F	Main Steam Line Low Pressure (Reactor Mode Switch in Run Only)	19
	G	Low Main Condenser Vacuum	
	H	High Main Steam Line Tunnel Ambient Temperature	
	I	High Main Steam Line Tunnel Differential Temperature	
	J	High Main Steam Line Area Temperature (Turbine Building)	19
7	B	Reactor Vessel Low Water Level 2	19
	H	High Main Steam Line Tunnel Ambient Temperature	
	I	High Main Steam Line Tunnel Differential Temperature	
	M	Standby Liquid Control System Actuated	19
	N	High Nonregenerative Heat Exchanger Outlet Temperature (RWCU System)	
	O	RWCU System High Differential Flow	
	P	RWCU System Equipment Area High Ambient Temperature	11 19
	Q	RWCU System Equipment Area High Differential Temperature	

reduced periodic test pressure is less than that required to cause drywell air to flow through the horizontal vents to the wetwell. The drywell atmosphere is allowed to stabilize for a period of 1 hr after attaining test pressure. Leakage rate tests commence after the stabilization period. The bypass leakage tests are conducted for a minimum of 1 hr to determine actual leakage rates. If at the end of 1 hr,
17 leakage rates are well established and less than allowable values, the test is terminated; otherwise, leakage testing is continued for a minimum of 4 hr or until reaching atmospheric pressure, whichever occurs first.

The test method is based on drywell atmosphere pressure observations and the known drywell free air volume. The leakage rate is calculated from pressure data, drywell free air volume, and elapsed time.

21 The periodic drywell bypass leakage test pressures and acceptance criteria are specified in the technical specifications. Periodic drywell structural inspections are
17 performed at intervals specified in the technical specifications.

The preoperational drywell leakage is required to be no greater than the maximum allowable leakage rate of
16 9,660 scfm at drywell design pressure (25 psig) test and maximum allowable leakage rate of 4,011 scfm at drywell reduced pressure (3 psig) test. Preoperational drywell leakage tests are performed as late as is practical in the construction sequence, but before initial plant operation.

17

Preoperational tests of the main steam positive leakage control system and the penetration valve leakage control system shall be performed to ensure that these systems meet the requirements of 10CFR50, Appendix J. The basis for the acceptable fluid leakage rates is established in the Technical Specifications. The main steam positive leakage control system and the penetration valve leakage control system can deliver seal fluid sufficient to assure the sealing function for at least 30 days at a pressure of 1.10 P_a.

during extended reactor shutdown or plant refueling. This precludes inadvertent steam discharge.

Since the MS-PLCS establishes a containment leakage barrier, direct MSIV out-leakage to the environment is eliminated. Therefore, the only concern regarding MSIV leak rate is possible containment pressurization when the MS-PLCS is in
operation. The MS-PLCS allowable leak rate is provided in 21
the Technical Specifications and established such that the
total air inleakage from valves served by the MS-PLCS and 21
from valves served by the penetration valve leakage control 13
system, PVLCS, (Section 9.3.6) does not contribute greater than 50 percent of the containment design pressure in a 30-day period.

TABLE 6.2-40 (CONT)

UMULATOR AIR PRESSURE PLUS SPRING FORCE ACT TOGETHER TO CLOSE VALVES WHEN BOTH PILOTS ARE

VALVE SEAT. THE VALVES CLOSE ON REVERSE FLOW EVEN THOUGH THE TEST
EVEN THOUGH TEST SWITCH MAY BE POSITIONED FOR CLOSE.

IES THAT EXTEND OUTSIDE THE DRYWELL ARE SMALL AND TERMINATE IN A SYSTEM THAT IS DESIGNED
JRING REACTOR SCRAM.

ISOLATION VALVES ARE POWERED FROM THE DESIGNATED PLANT BATTERY.

AIR-OPERATED VALVES CLOSE ON MOTIVE AIR FAILURE. ALL AIR OPERATED VALVES, EXCEPT MAIN STEAM

F THE REACTOR (SEE "POSITION - NORMAL" COLUMN).

IATOR ACTION IS REQUIRED TO ISOLATE THESE LINES.

'O RPV.

NMENT POOL.

ELOW DRYWELL DESIGN PRESSURE.

ONS SEALED BY PVLCS IS NOT INCLUDED IN THE 0.60 L_a TYPE B AND C TEST TOTALS.
RACTERISTICS ARE THE SAME IN BOTH DIRECTIONS.

TC UNSEAT THE DISC.

N POOL MINIMUM WATER LEVEL AND ARE THUS PROVIDED WITH A WATER SEAL.
AY BE PLACED RELATIVE TO A PRESSURE RELIEF DEVICE WHICH COULD REDUCE
C TESTS.
L NOT BE TYPE C TESTED.
E OF SEALING AIR INTO THE CONTAINMENT DOES
INCE :

W - WATER
N/A - NOT APPLICABLE
A-N - AIR OR NITROGEN
N - NON-ESSENTIAL
E - ESSENTIAL
SGTS - STANDBY GAS TREATMENT SYSTEM
FBCFS- FUEL BUILDING CHARCOAL FILTRATION SYSTEM
RV - RELIEF VALVE
AOV - AIR-OPERATED VALVE
CV - CHECK VALVE
MOV - MOTOR-OPERATED VALVE
MV - MANUAL VALVE
TCV - TESTABLE CHECK VALVE
SOV - SOLENOID OPERATED VALVE
ESF - ENGINEERED SAFETY FEATURE
F.C. - FAIL CLOSED
F.A.I. - FAIL AS IS
F.O. - FAIL OPEN
L.C. - LOCKED CLOSED
L.O. - LOCKED OPEN
HYV - HYDRAULIC-OPERATED VALVE

ISOLATION SIGNAL CODES

VESSEL LOW WATER LEVEL 1
VESSEL LOW WATER LEVEL 2
VESSEL LOW WATER LEVEL 3
N STEAM LINE FLOW
N STEAM LINE RADIATION
AM LINE LOW PRESSURE (REACTOR MODE SWITCH IN RUN ONLY)
CONDENSER VACUUM
N STEAM LINE TUNNEL AMBIENT TEMPERATURE
N STEAM LINE TUNNEL DIFFERENTIAL TEMPERATURE
N STEAM LINE AREA TEMPERATURE (TURBINE BUILDING)
WELL PRESSURE
CTOR PRESSURE
LIQUID CONTROL SYSTEM ACTUATED
-REGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE (RWCU SYSTEM)

O. RWCU SYSTEM HIGH DIFFERENTIAL FLOW
P. RWCU SYSTEM EQUIPMENT AREA HIGH AMBIENT TEMPERATURE
Q. RWCU SYSTEM EQUIPMENT AREA HIGH DIFFERENTIAL TEMPERATURE
R. RHR SYSTEM EQUIPMENT AREA HIGH AMBIENT TEMPERATURE
S. RHR SYSTEM EQUIPMENT AREA HIGH DIFFERENTIAL TEMPERATURE
T. CONTAINMENT HIGH GASEOUS RADIATION
U. REACTOR BUILDING ANNULUS VENTILATION RADIATION HIGH
V. FUEL BUILDING VENTILATION EXHAUST RADIATION HIGH
W. REACTOR PRESSURE LOW
X. RCIC ISOLATION SIGNALS:
(a) PIPE ROUTE AREA HIGH TEMPERATURE
(b) EQUIPMENT AREA HIGH TEMPERATURE
(c) TURBINE EXHAUST DIAPHRAGM HIGH PRESSURE
(d) STEAM LINE HIGH DIFFERENTIAL PRESSURE OR INSTRUMENT LINE BREAK

MSPLCS
Y. ~~MSIV~~ AIR LINE HEADER FLOW HIGH
Z. ~~MSIV~~ AIR LINE HEADER AND ~~DRYWELL~~ STEAMLINE DIFFERENTIAL PRESSURE LOW

RM. REMOTE MANUAL OPERATION AS APPROPRIATE

TABLE A.6A.7-1

DIRECT LOADS ON QUENCHER STRUTS
(Reference Fig. A.6A.7-1)

		Max Load (lb)	
F	Drag in any direction normal to strut axis		
	SRV bubble drag - active or inactive quencher	2481.0	21
	Chug (SBA/IBA, DBA)	1200.4	
	CO (SBA/IBA, DBA)	214.2	15 21
	LOCA bubble (DBA)	2727.0	
	Vent clearing water jet (DBA)	2828.0	
	Sloshing	73.0	
F	Vertical drag load - active or inactive quencher		
	Pool swell (DBA)	5302.0	21
	Fall back (DBA)	5302.0	

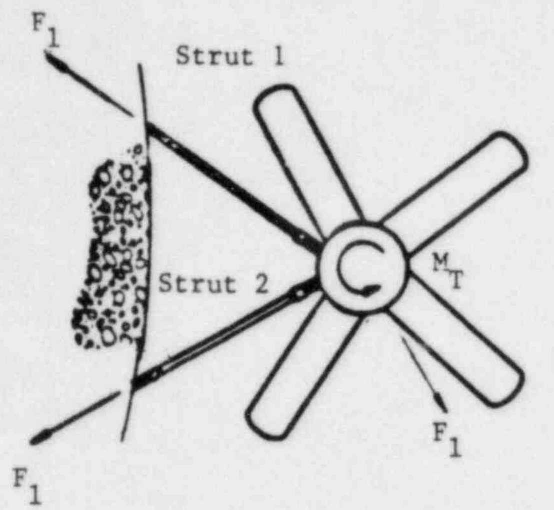
TABLE A.6A.7-2

MAXIMUM QUENCHER ASSEMBLY REACTION LOADS AND MAXIMUM QUENCHER/SRVDL INTERFACE LOADS (Reference Fig. A.6A.7-3) 15 21

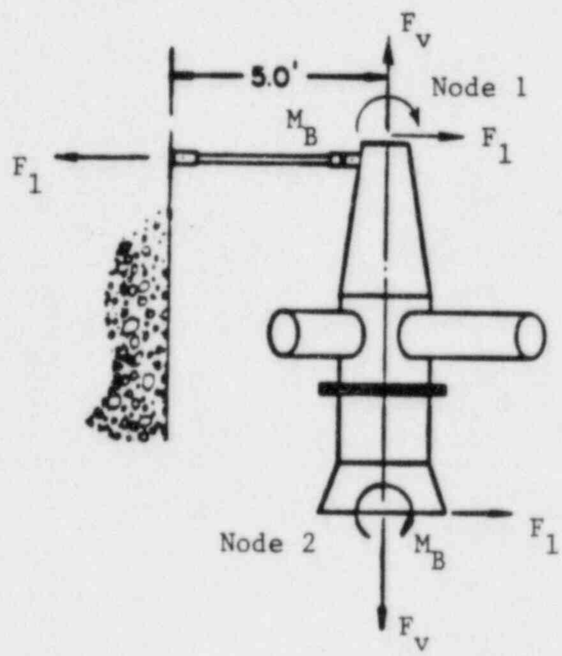
	Interface Loads (Node 1)	Reaction Loads (Node 2)	Strut 1	Strut 2
THERMAL				
F_l	6902	38736	28850	32241
F_V	10428	10428		
M_B	22323	324162		
M_T	8101	12793		
SRSS (OBET, OCCU)				
F_l	45286	63288	36778	40679
F_V	46561	181839		
M_B	17862	372324		
M_T	5019	105758		
SRSS (OBEI, OCCE)				
F_l	45346	77923	35045	39586
F_V	46454	181856		
M_B	19448	389493		
M_T	5738	107556		
SRSS (SSEI, OCCF)				
F_l	45357	77944	35063	39613
F_V	46460	181858		
M_B	19468	389852		
M_T	5768	107568		

21

NOTE: Forces in lb, moments in ft lb.



PLAN VIEW



ELEVATION

FIGURE A.6A.7-3

QUENCHER AND STRUT
ANCHOR LOADS

RIVER BEND STATION
FINAL SAFETY ANALYSIS REPORT

These air bubbles are smaller in size than the LOCA air bubbles, reside longer in the pool, and oscillate as they rise to the pool surface. The load created by these bubbles is discussed in Section L.6A.3.2.

The material in this attachment is organized as follows:

1. The specific analytical model, and

2. A load calculation procedure that summarizes the engineering process.

L.6A.2 SUBMERGED STRUCTURE LOADS DUE TO LOCA

L.6A.2.1 Compressive Wave Loading

As discussed in Section 6A.6.1.1, the very rapid compression of the drywell air theoretically generates a compressive wave; but, as pointed out in Sections 6A.6.1.1 and 6A.6.1.2, there were no loads recorded on the containment wall in PSTF for this phenomena. Therefore, it can be concluded that compression wave loads on structures in the suppression pool are significantly smaller than loads caused by the water jet for structures close to drywell. For structures near the containment, neither compressive nor jet loads are significant.

L.6A.2.2 LOCA Water Jet Load

During the initial phase of the DBA, the drywell air space is pressurized and the water in the weir annulus vents is expelled to the pool and induces a flow field throughout the suppression pool. This induced flow field is not limited to direct jet contact and creates a dynamic load on structures submerged in the pool.

GESSAR-II gives the zone of influence of these jets as a cylindrical region whose diameter is 1.5 times the diameter of the weir vent and extends 5 times the vent diameter into the pool. X-quencher struts are located in this zone (see Figure L.6A-1), and jet impact and drag loads are calculated for these struts. Impact loads are based on the empirical methods of Reference 1, and drag loads include both standard and acceleration drags.

Jet velocities, accelerations, and frontal displacements needed for the impact and drag load analysis are calculated based on one-dimensional mass and momentum equations and assume inviscid flow[2]. The GESSAR II vent clearing water
21 jet velocities were evaluated and determined to be applicable for RBS based on sensitivity studies performed with the SWEC LOCTVS containment analysis code. For RBS

load definition, the GESSAR II vent clearing velocity time history for the middle vent (GESSAR II, Figure 3BL-3) is multiplied by the factor 1.2 and used as input to the jet load analysis. 21

A typical load time history is shown in Fig. L.6A-2.

L.6A.2.3 LOCA Bubble Loads

During the initial phase of the DBA, pressurized drywell air is purged into the suppression pool through the submerged vents. After vent clearing, a single bubble is formed around each top vent. It is during the bubble growth period that unsteady fluid motion is created within the suppression pool. During this period, all submerged structures below the pool surface will be exposed to transient hydrodynamic loads.

The bases of the flow model and load evaluation for the LOCA bubble-induced submerged structure load definition are derived from the model of Reference 3. The following procedure is recommended for calculating loads on submerged structures:

1. Pool Dimensions and Bubble Data

   Specific data that must be obtained are:

   R_i - Initial bubble radius, assumed to be the same as the vent radius (1.146 ft)

   P_o - Drywell transient pressure obtained from Fig. 6.2-5 (psia) (from plant unique analysis of Section 6.2.1.1.3.1)

   ρ_o - Air density corresponding to drywell conditions when the drywell pressure is P_o (lb_m/ft^3)

   ρ - Pool liquid densi[illegible] 2.4 lb_m/ft^3)

   P_c - Containment air space pressure assumed to be constant at 14.7 psia

   P_∞ - Initial pool pressure at the top vent centerline submergence (psia)

   H - Pool depth (20 ft)

   L - Pool length (20.5 ft)

   D - Pool width (Fig. L.6A-3) (312.6 ft)

QUESTION 440.44 (6.3)

Response to the following TMI action items are required to complete the review of section 6.3.

(a) II.K.1. IE bulletins on measures to mitigate small break LOCA's and Loss-of Feedwater Accidents. | 21

(i) II.K.1.5. Review of E S F Valves

Response given in Appendix 1A is not acceptable. Staff position is given below:

NRC Position

Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Confirm that GSU performed or will perform the review described above.

(ii) II.K.1.10 Operability Status

Response given in Appendix-1A is not acceptable. Staff position is given below.

NRC Position

Review and modify as necessary your maintenance and test procedures to ensure that they require:

a. Verification, by test or inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

c. Explicit notification of involved reactor operational personnel whenever a safety-related system is removed from and returned to service.

Confirm that GSU performed or will perform the review described above.

(iii) <u>II.K.1.22 Auxiliary Heat Removal</u>
Response given in Appendix-1A is not acceptable. The applicant may refer to Grand Gulf response on this item which we found to be acceptable. (Ref: Attachment 1 to AECM-81/325, GG FSAR 18.1.29.3).

(b) <u>II.K.3 Final Recommendations of Bulletins and Orders Task Force</u>

(i) <u>II.K.3.17</u> We require a commitment from the applicant that they report ECCS outages via LERs and report summary of outages via annual reports.

(ii) II.K.3.18, II.K.3.21, II.K.3.16, II.K.3.25, II.K.3.30, II.K.3.31, II.K.3.45.

Since a generic position is already developed by the BWR Owners' Group, of which GSU is a member, we require detailed and more specific response. Refer to BWR Owners' Group position and submit plant specific response in detail as given by other BWR owners.

RESPONSE

7 The response to this request is provided in revised Appendix 1A, Table 1A-1, under the items identified above.

21

QUESTION 440.40 (6.3.3.7.3)

Justify selection of a lead plant for the LOCA break spectrum analysis. River Bend is committed to submit a plant specific LOCA analysis. We require a schedule for submittal of the plant specific LOCA analysis.

RESPONSE

A River Bend Station unique LOCA analysis has been submitted. 21