GULF STATES UTILITIES COMPANY

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AREA CODE 409 838-6631

December 30, 1983
RBG - 16,667
File Code No. G9.5
G9.8.6.2

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

Dear Mr. Denton:

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

Enclosed for your review are Gulf States Utilities Company responses to the open items identified in the Draft Safety Evaluation Report by the Auxiliary Systems Branch and responses to the request for additional information identified in part by Staff letters dated August 5, 1981 and December 31, 1981. This letter supplements docketed correspondence from Mr. Booker to Mr. Denton dated December 1, 1983. Attachment 1 summarizes the open items and indicates changes to be made in the River Bend Station FSAR. Attachment 2 provides a brief discussion of each open item, the response and reference material for each item. Where indicated, these responses will be provided in a future amendment to the FSAR.

Sincerely,

William J. Reed Jr.

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

wjr JEP
JEB/WJR/ERG/JEP
Enclosures

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ATTACHMENT 1

ITEM NUMBER	DSER SECTION	SUBJECT	FSAR REVISIONS
2a.	3.5.1.1 Pg 3-13,15 3.5.1.2 Pg 3-17	Valve Bonnets, Thermowells, Nuts Bolts, Studs, Valve Stems, Fan Failures as possible missiles	Enclosure 1
2c.	3.5.1.1 Pg 3-14,15 3.5.1.2 Pg 3-17	Missiles from Rotating Equipment	Enclosure 1
3.	3.6.1 Pg 3-28	High Energy Pipe Break Analysis (Pipe Whip and Jet Impingement Analysis)	January, 1985 (September, 1984)
8.	9.1.2 Pg 9-10,11	SF Racks Criticality Analysis	January, 1984
14a.	9.2.6 Pg 9-58 9.3.3 Pg 9-81,83	Flooding Due to Failure of CST	Enclosure 2
24a.	9.4.5.1 Pg 9-139	Maintenance of Minimum Temp in DGB	12/1/83 Letter (Enclosure 32)
24b.	9.4.5.2 Pg 9-142 12/31/81 Letter	Maintenance of Humidity and Minimum Temp in Switchgear and Pump Rooms in SSWPH	N/A
27.	8/5/81 Letter	Polar crane load drop analysis	Enclosure 3

ATTACHMENT 2

RESPONSES TO DSER OPEN ITEMS

2. 3.5.1.1 3.5.1.2 Incomplete evaluations of missiles - did not include valve bonnets, thermowells, nuts, bolts, studs, valve stems, fan failures, and gravitational missiles. (DSER Pages 3-13 to 3-17).

Response

a. Justification for not viewing components such as valve bonnets, thermowells, nuts, bolts, studs, and valve stems as possible missiles is provided in the response to FSAR Question 410.15 (Enclosure 1). This response will be incorporated into the FSAR in a future amendment.

c. The effects of missiles from rotating equipment, including fans, on safety related equipment both inside and outside of containment has been evaluated and is provided in the response to FSAR Question 410.15 (Enclosure 1). This response will be incorporated into the FSAR in a future amendment.

3. 3.6.1 HELB analysis is not complete; has not provided the results of a compartmental flooding analysis and detailed information on the environmental effects of a HELB and MELB. (DSER Pages 3-28; 3-29).

Response

Updated and revised tables and figures for the pipe whip and jet impingement analysis have been developed and will be provided by the end of September, 1984. Completion of the high energy pipe break analyses is scheduled for completion by the end of January 1985 (see related Questions 410.3, 210.45, and 210.46).

8. 9.1.2 The applicant has not provided a criticality analysis to confirm the criticality limits to be attained in the spent fuel storage facility. (DSER Pages 9-10, 9-11).

Response

The criticality analysis for the fuel building spent fuel storage racks is complete and verifies that k_{eff} is less than 0.95 under all normal and abnormal storage conditions. Because of its proprietary classification, the analysis will be provided, to the Staff by the end of January, 1984.

14. 9.2.6 The applicant should show:

a) that a catastrophic failure of the CST would not cause flooding of any safety-related systems, including electrical power supplies and controls.
b) that a crack in the piping from the CST to the HPCS/RCIC would not adversely affect the operation of these or other safety-related systems. (DSER Pages 9-58, 9-81, 9-83).

c) [See Draft DSER Page 9-81 relative to showing the leakage collection sump and pumps for the CST.]

Response

a. The response to this request is provided in the December 1, 1983 Letter from J. E. Booker to H. R. Denton and supplemented in Enclosure 2.

24. 9.4.5 The applicant should:

a) provide a discussion of the operation of the exhaust fan with respect to accommodating the minimum and maximum outside temperatures for both normal and accident conditions.
b) verify that under no conditions will the temperature, pressure or humidity conditions exceed the environmental qualifications of the safety-related equipment in the diesel generator rooms and the standby service water switchgear and pump rooms. (DSER Pages 9-139, 9-142).

Response

a. Diesel generator room heating is provided to maintain the design minimum temperature and to prevent freezing of fluid systems located in the rooms during the winter when the diesel generators are not operating. See the response to FSAR Question 410.64 (12/1/83 Letter - Enclosure 32).

   Heating is provided for the standby service pump rooms and switchgear rooms to maintain the design minimum temperature during the winter. See the response to FSAR Question 410.64 (12/1/83 Letter - Enclosure 32).

b. The diesel generator building operating rooms' ventilation systems are designed to maintain a maximum temperature of 122°F in the space. This value is based on an outside ambient condition of 96°F DB / 81°F WB. The outside ambient value has been taken from ASHRAE Handbook of Fundamentals 1972 Chapter 24, "Weather Data and Design Conditions" for Baton Rouge, Louisiana. The dry-bulb temperature represents a value which has been equaled or exceeded by 1% (approximately 30 hours) of the total hours during the months of June through September.

The National Oceanic and Atmospheric Administration Climatological Report for Baton Rouge for the years 1949-1979 lists the daily high temperature as well as an hour-by-hour record of the daily temperature. The highest recorded outside air temperature for that period was 103°F on June 30, 1954, which occurred for a period of less than 1 hour.

The maximum outdoor temperature that River Bend can use for ventilation air and still maintain the indoor design temp of the standby diesel generator building is 102°F with the diesel generator running at full load. Although this is 1°F less than the maximum of 103°F, it is unlikely that the diesel generator would be required to operate fully loaded coincident with this once in 30 year maximum outside air temperature. Also, the design air temperature of 122°F is the room bulk air temperature. The entering ventilation air passes over the generator portion of the diesel engine first and is heated up to the design as it exits the building. Consequently a 1°F temperature for less than 1 hour above the maximum outdoor design temperature would have no effect on diesel generator equipment qualification or operation.

The standby service water switchgear and pump rooms will reach a maximum of 115°F with the equipment in this area running/energized in a postulated post-accident lineup coincident with an outside air temp of 103°F. The resulting temperature, pressure, and humidity conditions do not exceed the environmental qualifications of the safety-related equipment located in these areas.

27. The polar crane load drop analysis is provided in the response to FSAR Question 410.8 (Enclosure 3). This response will be incorporated into the FSAR in a future amendment.

ENCLOSURE 1

QUESTION 410.15 (3.5.1.1, 3.5.1.2)

The staff finds that your rationale is insufficient justification for not viewing components such as valve bonnets, thermowells, nuts, bolts, studs and valve stems as possible missiles. Therefore, you must provide satisfactory assurance that these components will damage neither safety-related structures, systems or components (SSC) nor SSC which could cause safety-related SSC to fail to perform their safety functions either by showing that such component missiles will not affect the safety-related or appropriate non-safety-related SSC or that suitable missile barriers have been or will be provided for the protection of the appropriate SSC.

RESPONSE

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

The response to this request is provided in revised Sections 3.5.1.1.1 and 3.5.1.1.2.

6. Orienting the potential missile source in such a manner as to prevent unacceptable consequences due to missile generation.

3.5.1.1 Internally Generated Missiles

Missile protection is provided within the plant structures that are important to safety inside and outside the containment for two general sources of postulated missiles:

1. Rotating component failure
2. Pressurized component failure.

The basic approach is to ensure design adequacy of the equipment components against the generation of missiles, rather than to allow missile formation and try to contain its effects.

3.5.1.1.1 Rotating Component Failure Missiles

Castastrophic failure of rotating equipment~~, namely pumps and turbines, flanged equipment, and piping~~ leading to the generation of missiles is not considered credible. Massive and rapid failure of these components is ~~incredible~~ not credible because of the material characteristics, inspections, quality control during fabrication, erection and operation, conservative design, and prudent operation as applied to the particular component.

Insert A

1) The most substantial piece of NSSS rotating equipment is the recirculation pump and motor. This potential missile source is discussed in detail in Reference 5.

It is concluded in Reference 5 that destructive pump overspeed cannot result in the generation of missiles.

2) It has been further concluded that large, massive rotating components, such as the various ECCS pumps and motors, do not have sufficient energy to move their masses through the housings in which they are contained. The RCIC turbine similarly is concluded not to generate missiles upon failure. Redundant overspeed tripping devices ensure that the RCIC turbine does not reach runaway speed where possible component failure could take place.

Insert B

3.5.1.1.2 Pressurized Component Failure Missiles

The bases for the selection of missiles generated by postulated failures of pressurized components are:

Insert A (for pg. 3.5-3)

Various types of rotating equipment were analyzed (e.g. pumps, fans, and turbines) as to their potential for becoming missile sources. The following was concluded:

Insert B (for pg. 3.5-3)

3) Both axial and centrifugal fans were investigated to determine their potential as missile sources. The rotating components most likely to become missiles are the fan blades and the rotor impellers. A stress analysis was performed to determine the safety factors against the failure of these components. The results are shown in Table 3.5-26. For centrifugal fans the safety factors were found to be about 14-16. For axial fans, the safety factors were found to range from 3 to 36. In addition, if a blade failure were to occur, the failed blade would be moving in a direction tangential to the housing. This would cause the blade to rotate upon impact, due to the oblique angle and blade orientation, and then reimpact the housing. This would substantially reduce the energy to perforate the housing, with the blade being effectively contained due to the high factors of safety involved, and the mode of impact of the blade against the housing should a blade be thrown, it can be concluded that fans are not credible missile sources.

(1) The analysis shows that safety factors incorporated into the design range from 10 to 107 (see Table 3.5-27).

1. Thermometers or other detectors installed on piping or in wells are evaluated. The analysis of the thermowell shows that thermowell ejection is very improbable because of its highly conservative design. (1) Consequently, ~~it is~~ thermowells are not considered ~~a probable~~ as potential missile ~~source~~ sources.

2. Valves of ANSI 900 psig rating and above, constructed in accordance with the ASME Code, Section III, are pressure seal bonnet-type valves. ~~For pressure seal bonnet valves, valve bonnets are prevented from becoming missiles by the retaining ring, which would have to fail in these, and by the yoke, which would capture the bonnet or reduce bonnet energy.~~

Replace with Insert C

~~Because of the highly conservative design of the retaining ring of these valves, bonnet ejection is highly improbable. Hence, bonnets are not considered credible missiles.~~

3. Most valves of ANSI 600 psig rating and below are valves with bolted bonnets. ~~Valve bonnets are prevented from becoming missiles by limiting the stresses in the bonnet-to-body bolting material by the rules set forth in the ASME Code, Section III, and by designing the flanges in accordance with the applicable code requirements. Even if bolt failure were to occur, the likelihood of all bolts experiencing a simultaneous complete severance failure is very remote. The widespread use of valves with bolted bonnets and the low historical incidence of complete severance failure of the valve bonnets confirm that bolted valve bonnets need not be considered as credible missiles.~~

Replace with Insert D

4. ~~Valve stems are not considered potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air- or motor-operated valve stems are effectively restrained by the valve operators. No credible valve stem missiles were identified at River Bend Station.~~

Replace with Insert E

5. ~~Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy and thus are of minimal concern as potential missiles.~~

Insert C (for pg. 3.5-4)

Valve bonnets for pressure seal bonnet type valves were analyzed to evaluate the potential for the bonnets to become missiles. Bonnets could become missiles through one of the following: failure of the bonnet retaining ring, failure of the valve body at the retaining ring interface, or failure of the bonnet critical thickness. All three of these items were investigated, by analyzing a representative group of pressure seal bonnet type valves, and evaluating the safety factors against these types of failures (safety factors are based on the ultimate strengths of the materials). Results (Table 3.5-28) indicate that:

a. The safety factors against retaining ring failure range from 11 to 27 (for shear), and from 7 to 16 (for bearing)

b. The safety factors against failure of the valve body at the retaining ring interfact (for shear) ranged from 21 to 32.

c. The safety factors against failure of the bonnet critical thickness (for shear) ranged from 13 to 29.

Because of the high factors of safety involved against these types of failures, bonnets of pressure seal type valves are not considered potential missile sources.

Insert D (for pg. 3.5-4)

Valve bonnets for bolted bonnet type valves were analyzed to evaluate the potential for the bonnets to become missiles. Bonnets could become missiles through one of the following: failure of the bonnet bolts, or failure of the bonnet critical thickness. Both of these items were investigated by analyzing a representative group of bolted bonnet type valves, and evaluating the safety factors against these types of failures. As before, safety factors are based on the ultimate strengths of the materials. Results (Table 3.5-28) indicate that:

a. Safety factors against bonnet bolt failure range from 6 to 14.

b. Safety factors against failure of bonnet critical thickness range from 51 to 94.

Due to the high factors of safety involved against these types of failures, and the low historical incidence of complete severence failure of the valve bonnets, bolted type valve bonnets are not considered as potential missile sources.

Insert E (for pg. 3.5-4)

4. Valve stems were analyzed by assuming that a failure of the minimum stem thickness would allow the stem to become a missile. Analysis shows that safety factors (based on ultimate material strength) against this type of failure range from 3 to 11 (Table 3.5-28).

   The analysis did not take into account backseats and stem threads, which would further prevent ejection; nor did it take into account valve operators (on air and motor operated valves) which would effectively restrain the valve stem. Based on this conservative approach, valve stems are not considered as potential missile sources.

5. Nuts, bolts, nut and bolt combinations, and nut and stud combinations which were part of bolted bonnet type valves were considered to be the major concern for this type of missile. Analysis of bolted bonnet type valves has already eliminated bolt missiles from this type of source. All other nut and bolt combinations have a minimal amount of stored energy and are not considered further as potential missile sources.

TABLE 3.5-26

ANALYSIS OF FAN BLADES AS POTENTIAL MISSILES

CODE	EQUIP. TAG	TYPE	RPM	FACTOR OF SAFETY (F.S.)		
				SHEAR STRESS	BEARING STRESS	TENSILE STRESS
HVY	*FN1A	Vaneaxial	1,750	N/A	N/A	20.0
	*FN1B	Vaneaxial	1,750	N/A	N/A	20.0
	*FN1C	Vaneaxial	1,750	N/A	N/A	20.0
	*FN1D	Vaneaxial	1,750	N/A	N/A	20.0
SWP	*TWR1	Axivane	580	N/A	N/A	18.5
HVC	*FN8A	Centrifugal	3,500	14.0	16.0	N/A
	*FN8B	Centrifugal	3,500	14.0	16.0	N/A
HVF	*FN7A	Centrifugal	3,500	14.0	16.0	N/A
	*FN7B	Centrifugal	3,500	14.0	16.0	N/A
HVR	-FN1A	Vaneaxial	1,170	N/A	N/A	36.0
	-FN1B	Vaneaxial	1,170	N/A	N/A	36.0
	-FN1C	Vaneaxial	1,170	N/A	N/A	36.0
	-FN1D	Vaneaxial	1,170	N/A	N/A	36.0
HVR	-FN8	Vaneaxial	3,500	N/A	N/A	4.0
DRS	-UC1A	Vaneaxial	3,500	N/A	N/A	3.0
	-UC1B	Vaneaxial	3,500	N/A	N/A	3.0
	-UC1C	Vaneaxial	3,500	N/A	N/A	3.0
	-UC1D	Vaneaxial	3,500	N/A	N/A	3.0
	-UC1E	Vaneaxial	3,500	N/A	N/A	3.0
	-UC1F	Vaneaxial	3,500	N/A	N/A	3.0

N/A = Not Applicable

TABLE 3.5-27

THERMOWELL STRESS ANALYSIS FOR MISSILE CREDIBILITY

PIPING SYSTEM-PIPE SIZE	PIPE THICKNESS (IN)	THERMOWELL TYPE	DIAMETER OF THERMOWELL (IN.)	FACTOR OF SAFETY (FS)
MSS-24	1.218	Welded-In	1.5	83
MSS-16	0.843	Welded-In	1.5	63
MSS-12	0.687	Welded-In	1.5	53
MSS-10	0.593	Welded-In	1.5	50
MSS-6	0.432	Socket Welded	1.66	10
MSS-4	0.337	Socket Welded	1.66	10
MSS-2	0.218	Socket Welded	1.66	10
FWS-30	1.875	Welded-In	1.5	107
FWS-20	1.5	Welded-In	1.5	87
FWS-16	1.218	Welded-In	1.5	73
FWS-12	1	Welded-In	1.5	63
FWS-3	0.437	Socket Welded	1.66	10
FWS-2	0.343	Socket Welded	1.66	10
RHS-20	0.5	Socket Welded	1.66	10
RHS-18	0.937	Welded-In	1.5	67
RHS-16	0.375	Socket-Welded	1.66	10
RHS-14	0.375	Socket-Welded	1.66	10
RHS-10	0.843	Welded-In	1.5	60
RHS-8	0.593	Welded-In	1.5	47
RHS-6	0.280	Socket-Welded	1.66	13
RHS-4	0.337	Socket-Welded	1.66	10
CSH-16	1.031	Welded-In	1.5	63
CSH-14	0.937	Welded-In	1.5	60
CSH-10	0.718	Welded-In	1.5	50
CSH-4	0.437	Socket-Welded	1.66	10
CSH-3	0.300	Socket-Welded	1.66	10
CSL-14	0.437	Socket-Welded	1.66	10
CSL-12	0.406	Socket-Welded	1.66	10
CSL-10	0.593	Welded-In	1.5	50
CSL-4	0.237	Socket-Welded	1.66	13
CSL-3	0.216	Socket-Welded	1.66	13

RBS FSAR

TABLE 3.5-28

ANALYSIS OF VALVE BONNETS & VALVE STEMS AS POTENTIAL MISSILES

				FACTOR OF SAFETY (FS)						
				VALVE BONNET						
					THRUST KING					
VALVE SIZE (IN.)	TYPE	VALVE NUMBER	LINE NUMBER	BONNET CRITICAL THICKNESS	SHEAR	BEARING	BENDING	VALVE BODY AT KING INTERFACE	BOLTS	VALVE STEM
6"	PS-SC	1E51*AOVF066	ICS-6"	37.0	17.0	13.0	NA	NA	NA	NA
20"	PS-SC	1B21*VF010A	1FWS-020-66-1	29.0	14	8.0	10.0	23.	NA	NA
10"	BB-SC	1E12*AOF041B	RHS-10"	51.0	NA	NA	NA	NA	NA	NA
6"	PS-GATE	1G33*MOVF039	1WCS-006-136-2	22.0	27	16.	10.0	32.	NA	11.0
		1G33*MOVF040	1WCS-006-139-2	22.0	27	16.	10.0	32.	NA	11.0
10"	PS-GATE	1E22*VF036	1CSH-010-45-1	18.0	11	8.0	6.0	21.	NA	5.0
		1B21*VF011A	1FWS-020-66-1	18.0	11	8.0	6.0	21.	NA	5.0
20"	PS-GATE	1B21*MOVF065A	1FWS-020-62-2	26.0	14	9.0	10.0	23.	NA	4.0
		1FWS*MOV7A	1FWS-020-62-2	26.0	14	9.0	10.0	23.	NA	4.0
6"	BB-GATE	1G33*MOVF001	1WCS-006-5-1	88.0	NA	NA	NA	NA	9.0	5.0
		1G33*MOVF004	1WCS-006-4-1	88.0	NA	NA	NA	NA	9.0	5.0
8"	BB-GATE	1E51*MOVF063	1ICS-008-1-1	94.0	NA	NA	NA	NA	4.0	4.0
		1E51*MOVF064	1ICS-008-3-1	94.0	NA	NA	NA	NA	4.0	4.0
10"	BB-GATE	1E12*VF039B	1RHS-010-16-1	75.0	NA	NA	NA	NA	6.0	4.0
		1E21*VF007	1CSL-010-43-1	75.0	NA	NA	NA	NA	6.0	4.0
18"	BB-GATE	1E12*VF010	1RHS-018-53-1	60.0	NA	NA	NA	NA	8.0	3.0
		1E12*MOVF009	1RHS-018-53-1	60.0	NA	NA	NA	NA	8.0	3.0
6"	BB-GLOBE	1G33*MOVF102	1WCS-006-5-1	99.0	NA	NA	NA	NA	8.0	5.0

PS = Pressure Sealed Bonnet
BB = Bolted Bonnet
SC = Swing check
NA = Not applicable

ENCLOSURE 2

The condensate storage tank normally supplies the HPCS and RCIC systems. However, automatic shutoff valves are provided to close on low condensate storage tank level and transfer HPCS and RCIC pump suctions to the suppression pool, which is the primary safety design source of core spray water.

The level instruments, together with their power supplies, transmitters, readout equipment, etc, that provide this transfer signal are safety related. The level instruments are connected to the safety-related suction piping leading to the HPCS and RCIC pumps, and are physically located within a Seismic Catagory I structure. 2

That portion of the condensate makeup and drawoff system which penetrates the containment and forms part of the containment boundary (Fig. 6.2-65) is Safety Class 2 and Seismic Category I (Table 3.2-1).

Failure of the condensate storage tank during normal operation would not result in the loss of water supply for the control rod drive hydraulic system since the normal CRD system supply is from the condensate system pump discharge.

INSERT

Failure of the condensate storage tank during accident conditions would not preclude plant safe shutdown or post-accident mitigation processes. The systems which draw water from this tank are all capable of performing their safety function without this water supply. Specifically, the HPCS and RCIC suction would automatically shift to the suppression pool; the CRD accumulators provide enough stored energy to insert control rods; and the fuel pool cooling system receives any required makeup water from the standby service water system.

Level in the condensate storage tank is normally maintained by the demineralized water transfer pumps. Should these pumps fail, level can be maintained directly from the makeup demineralizer forwarding pumps.

Continuous level monitoring of the condensate storage tank provides practical assurance that leakage does not go undetected or uncontrolled and meets the requirements of General Design Criterion 60. Tank overflow and drains are retained by the sump. Water collecting within the sump is pumped to the radioactive liquid waste treatment system by a sump pump. The overflow system is designed for the maximum influent from the largest single source. The tank vent is provided with a screen to prevent the entry of birds or

~~INSERT A (For Page 9.2-35)~~

~~The catastrophic failure of the condensate storage tank (CST) does not flood any safety related equipment. Figure 9.2-25 shows the finish plant grading in the area of the CST, and the approximate area that would be covered by 6 inches of water from CST failure; ie, to an elevation of 95'-0". Since the ground is relatively level, water from CST failure spreads out over a wide area. Exterior openings located closest to the CST are an access door at el. 98'-0" and a truck door at el. 94.75', both in the north wall of the fuel building. These doors are designed to be watertight. In addition, the design of safety related structures is based on a probable maximum flood level of 98'-0" msl. Flood protection of the plant is further discussed in Section 3.4.~~

REPLACE

INSERT A (For Page 9.2-35)

The catastrophic failure of the condensate storage tank (CST) does not flood any safety related equipment. Figure 9.2-25 shows the finish plant grading in the area of the CST, and the approximate area that would be covered by 6 inches of water from CST failure; ie, to an elevation of 95'-0". Since the ground is relatively level, water from CST failure spreads over a wide area. The exterior openings exposed to this flooding in buildings which contain safety-related equipment are an access door at el. 98'-0" and a truck door at el. 94.75', both in the north wall of the fuel building. These doors are designed to be watertight. The design of safety related structures is based on a probable maximum flood level of 98'-0" msl. Flood protection of the plant is further discussed in Section 3.4.

ENCLOSURE 3

QUESTION 410.8 (9.1.4.3)

Provide a schedule for submittal of spent fuel cask drop analysis and the polar crane load drop analysis or provide the analyses.

RESPONSE

The spent fuel cask drop analysis and the polar crane load drop analysis ~~will be provided by June 1983~~. 7
are provided in revised Section 9.1.4.3.

The most serious consequence resulting from an accidental drop of vessel head would be severe plastic deformations of the vessel top flange. This accident does not produce any vessel leaks or result in the release of radioactive material.

A polar crane load drop analysis ~~is currently being~~ was performed on the remainder of containment to verify that consequences of a load drop do not jeopardize the ability to safely shut down the plant or result in the release of INSERT A significant amounts of radioactivity. ~~Detailed results of this analysis will be provided by a later amendment to the FSAR~~. Also safe load paths will be defined per the 3 guidelines of Section 5.1.1(1) of NUREG-0612.

The fuel prep machine removes and installs channels with all parts remaining under water. Mechanical stops prevent the carriage from lifting the fuel bundle or assembly to a height where water shielding is less than 8 ft. Irradiated channels, as well as small parts such as bolts and springs, are stored underwater. The spaces in the channel storage rack have center posts which prevent the loading of fuel bundles into this rack.

There are no nuclear safety problems associated with the handling of new fuel bundles, singly or in pairs. Equipment and procedures prevent an accumulation of more than two bundles in any location.

The refueling platform is designed to prevent it from toppling into the pools during an SSE. Redundant safety interlocks and limit switches are provided to prevent accidentally running the grapple into the pool walls. The grapple utilized for fuel movement is on the end of a telescoping mast. At full retraction of the mast, the grapple is 8 ft below water surface, so there is no chance of raising a fuel assembly to the point where it is inadequately shielded by water. The grapple is hoisted by redundant cables inside the mast and is lowered by gravity. A digital readout is displayed to the operator, showing him the exact coordinates of the grapple over the core.

The mast is suspended and gimbaled from the trolley, near its top, so that the mast can be swung about the axis of platform travel in order to remove the grapple from the water for servicing and for storage.

The grapple has two independent hooks, each operated by an air cylinder. Engagement is indicated to the operator. Interlocks prevent grapple disengagement until a "slack

Insert A (for Pg. 9.1-60)

The Polar Crane Load Drop Analysis evaluated the effects of postulated load drops on both the concrete and the steel framing areas of the refueling floor at 186'-3". The following considerations were applied in performing this analysis:

1. The plant is assumed to be in a stable cold shutdown condition.

2. No credit was taken for the polar crane's dynamic lowering feature, main hoist speed sensing control, or multiple independent braking systems.

3. Drop analyses were based on elastic-plastic curves that represent a true stress-strain relationship.

4. All energy of each postulated drop was assumed to be absorbed by the impacted structure.

5. All loads were assumed to be completely rigid and to experience no deformation during impact.

6. All postulated load drops assumed impact in a manner that would inflict the most damage.

7. Administrative procedures prohibit the travel of any heavy load over fuel stored in the containment fuel pool.

8. Impact loads of the postulated drops included the weight of the load, any lifting apparatus, and the crane load block.

The load drop was divided into two parts. The first part evaluated the effects of postulated drops on to the concrete areas of the refueling floor. The concrete areas consist of the upper containment pools, the RPV head storage position, and a portion of the drywell head storage position. The second part evaluated effects of postulated drops onto the drywell head and RPV head insulation support storage areas which are supported by structural steel members. The results of these analyses are discussed below.

The concrete structures were analyzed for postulated drops of the RPV head, drywell head, steam separator, and steam dryer. Each structure was analyzed to determine its energy-absorbing capacity and structural response within limiting deformation ranges. The postulated load drops were assumed to occur at various points along the load's travel path, and the structures were analyzed in order to verify that the resulting ductility ratios were less than 10. This ensures that the structure has sufficient energy-absorbing capacity to withstand the postulated drop without having unacceptable damage. Administrative controls ensure that each load handling evolution involves only a vertical lift from its origin to its transport height, a horizontal movement of the trolley to the load's destination, and a vertical lowering of the load. The analysis concluded that the evaluation criteria of Section 5.1 of NUREG-0612 were satisfied as explained below:

1. The drywell head storage floor is structurally adequate to withstand a drop of the drywell head from a maximum height of 2 ft. above its concrete slab at El. 186'-3".

2. The RPV head storage floor is structurally adequate to withstand a drop of the RPV head from a maximum height of 6 ft. above its concrete slab at El. 186'-3".

3. The impact of the RPV head on the concrete plugs on the east side of refueling floor could cause these plugs to fail and generate secondary missiles immediately below into the reactor water cleanup (RWCU) filter cubicle. However, this does not degrade the capability to maintain safe shutdown because the RWCU system can be isolated and is not required to operate in order to maintain safe shutdown.

4. A drop fo the RPV head from its maximum lift height of 6 ft. above the refueling floor into the refueling cavity causes structural damage to the cantilever overhang of the drywell roof. This cantilever is located at the interface of the refueling cavity floor at El. 162'-3" with the refueling seal. However, the RPV head may ultimately strike the top flange of the RPV. Neither the impact on the RPV top flange nor the structural failure of the drywell roof cantilever degrades the capability to maintain safe shutdown.

5. The drop of the RPV head from its maximum height into the drained refueling cavity envelopes any postulated drops or the steam dryer or steam seperater. This is due to two factors:

    a. The RPV head with its attached carousel (total Wt. = 84 tons) is the heaviest load handled by the polar crane.

    b. The RPV head drop into the drained refueling cavity is not diminished by the drag effects of water as in the case of the steam dryer and steam separater.

6. None of the postulated drops degraded the plant's ability to maintain safe shutdown or resulted in a release of radioactivity to the environment. A drop into the refueling cavity of any of the loads considered in this analysis may damage the RHR piping and spargers located there, but shutdown cooling could be accomplished using the RHR connection to the feedwater system (refer to Fig. 5.4-13, Sht. 3 of 3) or the alternate shutdown cooling method (refer to Section 5.4.7.1.5).

7. None of the postulated drops result in damage that would cause the vessel core to become uncovered.

8. By inspection, the impact of the limiting load on the pool boundary walls may degrade leak tightness of the upper pool walls. However, leak tightness of the boundaries is not a requirement to maintain the reactor in a safe shutdown condition.

9. A postulated drop of the drywell head from a height of 2 ft. above the fuel pool valve room roof at El. 186'-3" could result in spalling of concrete from the underside of the roof. However, this spalling would not result in damage to any components which are required to maintain safe shutdown.

Calculations were performed to evaluate the ability of the structural steel to withstand drops of the drywell head and the RPV insulation support frame. The calculations demonstrated that the postulated drops could overstress the impacted steel members and collapse the supporting framework. Although a progressive failure of structural steel in the path of the postulated drops is not considered likely, this possibility was evaluated for its effect on the plant's ability to maintain a safe shutdown condition. The evaluation considered a postulated drop resulting in the drywell head or the RPV insulation support frame falling through the refueling floor structural steel and impacting in the suppression pool. The evaluation concluded that the criteria of Section 5.1 of NUREG-0612 were satisfied as explained below:

1. Both RHR loops A and B would remain functional to conduct shutdown cooling operations by taking suction from the recirculation system pipng and returning the coolant to the reactor via the feedwater piping. Neither the common suction line from the recirculation loop nor the separate RHR connections to the feedwater piping are located beneath the travel path of the drywell head or the RPV insulation support frame.

2. RHR loops B and C would remain functional for alternate shutdown cooling operations. RHR loop A could sustain sufficient damage to its suction piping in the suppression pool to preclude its operation in this mode.

3. Due to damage to supply piping that serves the refueling cavity spargers, RHR loops A and B would lose the capability to return shutdown cooling water to the RPV via the spargers in the refueling cavity. However, this loss would not degrade the plant's ability to maintain safe shutdown because the shutdown cooling paths described in 1 and 2 above are still available.

4. None of the postulated drops result in releases of radioactivity to the environment that exceed the NUREG-0612 recommended dose of less than one quarter of the 10CFR100 reference values.

5. None of the postulated drops would result in damage that would cause the vessel core to become uncovered.

In summary, all of the postulated drops could cause damage to the reactor building structures and systems (piping, cable trays, instruments, etc.). However, this analysis indicates that none of the drops will degrade the capability to maintain the reactor in a safe shutdown condition.