GULF STATES UTILITIES COMPANY

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RBG-19,889
File Nos. G9.5

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

Attention: A. Schwencer

Dear Gentlemen:

River Bend Station Unit 1
Docket No. 50-458

On July 12, 1984, Gulf States Utilities (GSU) Company met with the ACRS and members of the NRC Staff. Please find attached GSU's position on each of the ten items identified by members of the Committee for which they desired to have additional information.

Sincerely,

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

JEB/BEH/lp

Attachment

cc: D. G. Eisenhut

20 Copies Enclosed

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ATTACHMENT

Item No. 1

The dedicated diesel generator that drives the high pressure core spray (HPCS) pump currently depends on cooling water supplied by pumps powered by the other two diesel generators during loss of offsite power conditions. The ACRS recommends that the merit of removing this dependency be examined.

GSU Position

GSU has examined the merit of a completely independent HPCS cooling water supply. The current design for supplying cooling water to the HPCS diesel (Standby Service Water (SSW) pumps powered by either the Division I or II diesels) meets NRC Staff regulatory requirements and is the same design as was reviewed at the CP stage. The concern expressed by the ACRS requires failure of both Division I and II diesels to operate (i.e. multiple failures which is beyond current NRC design requirements). In this case, the HPCS diesel (Division III) would be denied cooling water and would not be operable.

To remove this dependency a design modification is proposed which involves off loading one of the 50% SSW cooling pump motors and HPCS pump cubicle cooling fan from the Division I diesel-generator and adding this load to the HPCS diesel-generator (i.e. Division III).

This design change will allow 50% of the long term (i.e. greater than 1/2 hour) SSW cooling water to be provided from the Division I diesel while the other 50% will be provided from the HPCS diesel generator (i.e. Division III). Division II remains unaltered and capable of supplying 100% of the required long term SSW cooling water. Single failure criterion for the diesels is satisfied in the following manner:

1. Loss of either the Division I or HPCS (Division III) diesels - 100% SSW cooling water supplied by Division II diesel.

2. Loss of the Division II diesel - 100% SSW cooling is supplied (50% from Division I and 50% from the HPCS Division III)

Operator action would be required after 30 minutes. For example, if the Division I diesel generator fails to function, the Division I powered MOV that normally isolates SSW from Normal Service Water (NSW) will have to be manually closed by the operator. Also if both Division I and Division II diesels fail to function, operator action will have to occur to isolate certain heat exchangers in order to keep the HPCS (i.e. Division III) powered SSW pump from reaching run-out conditions.

GSU has prepared the major portions of the appropriate HPCS modification FSAR change notice. This information has been forwarded to the NRC via

letter (RBG-19576 dated 11/29/84) and will be included, along with the necessary information, in a future FSAR amendment.

Item No. 2

GSU stated that they plan to conduct a limited probabilistic risk assessment (PRA) for the River Bend Station (RBS). The ACRS supports the proposal to perform a plant-specific PRA and recommends that it include seismic-and fire-induced accident scenarios.

GSU Position

GSU has performed a limited PRA analysis for RBS based on previous Grand Gulf-1 analyses (event trees) supplemented by RBS plant specific fault trees and site specific consequence analysis. After consideration of the River Bend design, and the results of its limited PRA compared to more detailed PRA for other plants of similar design, it is concluded that the impact of external initiating events can be predicted adequately and that additional analysis is not warranted. GSU will however, conduct additional studies to support this conclusion for fire induced events since the Limerick plant used for comparative purpose is a BWR-4 with Mark II containment.

Design Criteria

Pertinent criteria contained in 10CFR50 and NRC Regulatory Guides regarding fire, flooding, seismic, high wind, pipe whip, turbine/tornado missiles, chemical hazards, and security have been met or exceeded. Because RBS is designed and constructed pursuant to the latest fire protection requirements, the probability of severe fires at RBS is expected to be equal to or less than the probability of severe fires at any other nuclear power plant of comparable vintage.

RBS Limited PRA Analysis Basis

Grand Gulf-1 (1) and River Bend are of the BWR-6, Mark III containment design. The systems of interest in a limited scope PRA are the Engineered Safety Feature (ESF) and decay heat removal systems which are nearly identical in the two plants. During PRA work, any significant design differences were specifically built into the plant fault tree logic. The containment buildings are both Mark III; however, the RBS containment is free-standing steel while Grand Gulf is reinforced concrete. The design pressure and ultimate strength for both containments are essentially identical.

The Grand Gulf event trees include both the accident event trees (transient and LOCA) and the containment event tree. Since the Grand Gulf-1 accident event trees are general in nature they are applicable to RBS. Sixty-six transient and LOCA event trees derived from Grand Gulf analysis were used and site specific fault trees were constructed at the component level. The primary difference between River Bend and Grand Gulf is in the supply of service water and the electrical power supply and these were included in the RBS plant specific fault trees. Some specifics of the RBS PRA are:

- The INPO Nuclear Plant Reliability Data System (NPRDS) was used.

- Major events requiring operator action were quantified based upon a standard human error data base (Swain's Handbook, NUREG/CR-1278).

- Common cause failures were limited to electrical power supply and service water supplies. Other more indirect common causes of failure, such as environmental conditions, were not considered because they have been shown to be insignificant for other similar plants.

Justification for Exclusion of External Initiators

The RBS site is located in an area of low seismicity well above the Mississippi River flood plain. It is over 100 miles from the nearest seacoast and in an area of low or at most average tornado activity.

GSU is one of 42 utilities participating in the "Seismicity Owner's Group" study entitled "Seismic Hazard Methodology for Eastern U.S.A." (2) and RBS is one of nine reactor sites selected for computations pertaining to earthquakes. Preliminary figures and estimates indicate that the likelihood of an earthquake exceeding the design base would be on the order of 10^{-4} to 10^{-5} per year. This is an order of magnitude or two lower than that anticipated for many sites. Since the frequency of occurrence of these events is extremely low and because of the extreme severity needed to generate severe accidents from these initiators, omission of earthquakes as an external event is justified.

As a way of illustrating the impact of external events on the calculation of risk at a BWR of modern design, the following table provides a summary of the fire and seismic contributions to risk at the Limerick plant. Although Limerick is a BWR-4, Mark II containment design, the factors affecting fires or earthquakes would not be particularly impacted by the difference in nuclear steam supply system or containment designs.

Conclusion

From this table it can be seen from the median that internal intiators contribute approximately 7 times as much to the core melt frequency as fire initiators. Seismic initiators are relatively unimportant, contributing a small amount. RBS would be expected to demonstrate approximately the same risk profile if fire and seismic events were analyzed. RBS is similar in design to Limerick with respect to external events and it is concluded that additional studies to include external events will not yield results significantly different from that already obtained.

Table: Annual Frequency of Core Melt

Initiating Event	5th percentile	Median	95th percentile
Internal Initiators	2.7×10^{-6}	9.2×10^{-6}	6.0×10^{-5}
External Initiators			
Earthquakes	2.2×10^{-9}	3.3×10^{-7}	2.7×10^{-5}
Fires	1.6×10^{-7}	1.3×10^{-6}	1.1×10^{-5}
Total	3.9×10^{-6}	1.7×10^{-5}	7.6×10^{-5}

In addition to the above, GSU has directed General Electric Co. to review its BWR/6 Standard Plant Fire Risk Assessment, compare River Bend to the BWR/6 Standard Plant, and list design differences that could contribute to differences in fire risk.

(1) Grand Gulf PRA (RSSMAP NUREG/CR-1659, Vol. 4)

(2) Report on the groups findings is scheduled for mid-April 1985.

Item No. 3

Although River Bend is in a relatively quiet seismic portion of the country, NRC contractor estimates of the recurrence interval for the safe shutdown earthquake are similar to those for most eastern sites. The ACRS recommends that GSU review, in detail, the seismic capability of the emergency AC power supplies, the DC power supplies, and small components such as actuators, relays, and instrument lines that are part of the decay heat removal system.

GSU Position

GSU is currently reviewing the seismic capability of the emergency AC and DC power supplies. When this review is complete, the results will be provided to the NRC. Completion is expected in February, 1985.

Item No. 4

River Bend employs refrigerated charcoal beds in the offgas processing system for the main condenser. The ACRS requests that GSU provide an estimate of the offsite doses given the complete loss of refrigeration to the beds and the failure to manually isolate the offgas system from the main plant exhaust.

GSU Position

GSU is currently evaluating the offsite dose consequences due to the complete loss of refrigeration to the offgas system charcoal absorber beds and the failure to manually isolate the offgas system from the main plant exhaust. When this analysis is complete, the results will be provided to the NRC. Completion is expected in February, 1985.

Item No. 5

River Bend Station containment personnel and equipment hatches utilize inflatable seals. The ACRS expressed interest in: (1) length of time accumulators would be able to maintain air pressure to the seal in a post-accident situation given a specified leakage from the seals, and (2) recovery plans should one lose air pressure to the seals in a post-accident situation.

GSU Position

The only containment vessel hatches at RBS utilizing inflatable seals are the personnel air lock hatches. GSU has examined the personnel air lock inflatable seal design with respect to maintenance and recovery of air pressure to the seals in a post-accident situation. There are two inflatable seals mounted on the perimeter of each door of the air lock. These seals, when inflated with air, impinge upon stainless steel surfaces in the door jamb.

The air lock is designed to hold the seal inflation pressure for a period of 35 days. In a post loss of coolant accident (LOCA) environment at any time during this 35-day period, it is possible to enter or exit the containment a total of four times. Preliminary calculations indicate that it is possible to perform the above without plant air supply for the aforementioned 35-day period. The loss of plant air supply to the air locks in no way jeopardizes containment integrity for a period of 35 days. To satisfy this criteria an independent auxiliary air supply is self-contained within each door. This auxiliary air supply is designed and built to meet the requirements of a nuclear safety-related Category I air system.

Air is supplied to each containment air lock inside door by the Instrument Air System (IAS). The airlock at elevation 118'-0", azimuth 135° outside door is supplied by the IAS from the fuel building. The air lock at elevation 175'-0", azimuth 315° outside door is supplied by the IAS from the auxiliary building. Investigation of the air lines and isolation valves supplying the inflatable seals indicates that in the event the air locks should lose air pressure and personnel were allowed access to the air lines in a post-accident situation, a secondary source of air such as bottled air could be connected to the doors.

Item No. 6

The ACRS requests that it be provided the qualification program and data for River Bend's containment isolation valves for the 36 inch diameter containment purge and vent lines.

GSU Position

In order to demonstrate the reliability of the 36 inch containment purge valves against expected LOCA conditions, GSU is planning to add actuator unit stops restricting the valves to a 65° opening. This, in addition to changing some bolt material, assures the valves will close against the maximum calculated pressure differential across the valve during a LOCA.

GSU specifically addressed the NRC Staff's Branch Technical Position (BTP) 6-4 (reference RBG-19385 dated 11/08/84) and provided various FSAR revisions to document the findings of Posi-Seal Int. Qualification Reports and SWEC's calculations.

Item No. 7

In the phase II work on the River Bend PRA, GSU plans to modify their PRA to include design considerations for ATWS. The ACRS requests that GSU provide their estimate of the failure rate for the recirculation pump trip logic. It is suggested that the results of the phase II PRA program be provided to the Committee.

GSU Position

GSU has received the preliminary phase II PRA report and currently has this report under review. Several modeling assumptions have been identified which require further review. Pending the outcome of this review, portions of the phase II mini-PRA may have to be revised. GSU plans to provide the NRC staff with the results of this review of the phase II mini-PRA at a later date.

Item No. 8

Unit coolers are to be used at RBS instead of containment sprays to control temperature and pressure following an accident. Containment sprays have previously been cited as being very efficient in the removal of airborne radioiodine. To what degree can a unit cooler system be expected to remove airborne radioiodine or other materials?

GSU Position

The approach to addressing this concern has been (1) to assess the amount of iodine and particulate fission products which escape the pool into the wetwell airspace, (2) to assess the wetwell environment during postulated accidents, and (3) for the pool bypass scenarios, to assess the overall risk associated with the accidents. It is GSU's conclusion that inclusion of the unit coolers would result in negligible impact on the overall plant risk evalution for postulated RBS accidents.

Under postulated accident conditions, essentially all of the non-gasious core fission products released from the reactor pressure vessel would be retained in the suppression pool due to the effectiveness of pool scrubbing. That portion of the fission products escaping the pool would be extremely minute. Furthermore, the wetwell airspace would be saturated with water vapor (100% relative humidity). Any fission products escaping the pool would be plated out on the relatively cooler surface of the structures and walls in the wetwell region (a typical Mark III containment has about 11 million lbm of steel and 0.125 million sq. ft. of surface area in the wetwell). The saturated atmosphere together with condensation would provide an ideal condition to retain essentially all the fission products escaping the pool. Thus, one concludes that the airborne particulate fission products would have a negligible contribution to the overall plant release.

The capability of a unit cooler system to remove the airborne fission products has not been assessed. Even though one would expect a unit cooler or a containment spray to remove the fission products by condensation, the impact on reducing the amount of fission products released is minimal due to the effectiveness of pool scrubbing and condensation of water vapor from the containment atmosphere. Thus, neither the unit coolers nor containment sprays have a significant impact on the accident dose evaluation.

For postulated severe accident scenarios with pool bypass, the probabilistic evaluations show that these types of scenarios have negligiable overall risk. This is based on GE Probabilistic Risk Assessment (PRA) of the GESSAR II Nuclear Island Plant. While it is conceivable that inclusion of containment sprays would reduce the amount of fission products released, the impact on the overall plant risk is minimal, due to the negligibly small probability for significant pool bypass.

It is therefore concluded that inclusion of containment unit coolers rather than containment sprays for RBS would result in minimal impact on the overall plant risk evaluation.

Item No. 9

GSU has proposed to include in the River Bend Emergency Procedures a procedure for venting the containment under certain accident conditions. The bases for the decision to take this action are not yet clear. The NRC Staff has not completed its review of this proposal. The ACRS wishes to be advised when the NRC Staff has reached a position on this matter and to have an opportunity to comment generically or specifically.

GSU Position

GSU has recognized the overall plant risk reductions that can be gained by the use of containment venting during severely degraded plant conditions involving loss of containment cooling as recommended in the generic BWR Emergency Procedure Guidelines (EPG's). GSU is presently evaluating the containment response to the most probable sequence of events that would require containment venting and any methods that would allow containment heat removal by presently existing systems. As this study is currently ongoing, GSU cannot provide detailed procedures for initiation of containment venting or a description of systems that may be used at this time. When the analyses are complete, GSU will inform the NRC of the bases for the operator actions and a description of containment heat removal methods to be used.

Item No. 10

The ACRS has not completed its review of hydrogen control for the River Bend Station, particularly as it may be impacted by differences in containment design features between River Bend and Mark III BWRs previously reviewed. The ACRS will complete its review of the full power operating license when the NRC Staff and GSU have made sufficient progress in resolving the matter of hydrogen control.

GSU Position

GSU is a member of the Hydrogen Control Owner's Group (HCOG) and is actively supporting its programs directed toward resolution of the hydrogen control issue on a generic basis. HCOG has recently reassessed its plan for addressing the hydrogen control issues and has developed an overall program for achieving resolution of this issue. This program plan has been completed and was presented to the NRC on December 14, 1984 for its concurrence. GSU will utilize all applicable generic items from this plan in resolving this issue on RBS. Generic HCOG submittals which are applicable to RBS will be placed on the docket by letter endorsement.

The basic elements of the GSU issue resolution plan include submittal of a CLASIX-3 analysis, containment ultimate pressure retaining capability analysis, a list of equipment required to survive a hydrogen burn, an equipment survivability evaluation for deflagration and diffusion type burns and development of RBS specific Emergency Operating Procedures. As most of these programs are ongoing, additional details are not available at this time. Once completed GSU will provide the NRC with additional detailed results.