FILE
GRAND GULF

OCT 2 1973

Docket Nos. 50-416
50-417

Mississippi Power & Light Co.
ATTN: Mr. N. L. Stampley
Vice President - Production
P. O. Box 1640
Jackson, Mississippi 39205

Gentlemen:

We have reviewed your responses to our requests for additional information, dated May 12, 1973 and July 19, 1973. Based on our review of your responses, we find that some responses will have to be supplemented with additional information. In addition, we have established certain Regulatory requirements in regard to the design of the containment which will have to be resolved prior to concluding our review.

Identification of this need for additional information and a Statement of Regulatory requirements are set forth in Enclosures 1 and 2, respectively.

To maintain our licensing review schedule, we will need a completely adequate response to the request for additional information and to the Statement of Regulatory Requirements by November 12, 1973. Please inform us within 7 days after receipt of this letter of your confirmation of the above schedule or the date you will be able to meet. If you cannot meet our specified date or if your reply is not fully responsive to our requests, it is highly likely that the overall schedule for completing the licensing review for this project will have to be extended. Since reassignment of the staff's efforts will require completion of the new assignment prior to returning to this project, the amount of extension will most likely be greater than the extent of delay in your response.

The identifying numbers used in the enclosed Request for Additional Information follow the pattern established in our previous requests for additional information. Where appropriate, reference is made to the original question number to which your response was made. Your response to these additional questions and requirements may be made either by incorporating the information provided for other nuclear power plants by reference, or you may amend your application by submitting revised pages and supplements.

OFFICE ▶						
SURNAME ▶						
DATE ▶						

B/44

Please contact us if you desire additional discussion or clarification of the material requested.

Sincerely,

Original signed by
Voss A. Moore

Voss A. Moore, Assistant Director
for Boiling Water Reactors
Directorate of Licensing

Enclosures:
1. Request for Additional Information
2. Statement of Regulatory Requirements

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Distribution:
Docket Files (50-416/417)
AEC PDR
Local PDR
BWR-1 File
V. A. Moore
J. Hendrie
D. Eisenhut
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RO (3)
G. Owsley
M. Maigret (w/5 extras)
R. Cudlin
TR Branch Chiefs
BWR Branch Chiefs
ACRS (16)

OFFICE ▶	L:BWR-1	L:BWR-1	L:BWR/AD			
SURNAME ▶	GOwsley:dlb	WRButler	VAMoore			
DATE ▶	10/1/73	10/1/73	10/1/73			

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION
GRAND GULF NUCLEAR STATION, UNITS 1 AND 2
DOCKET NOS. 50-416/417

6.2.43 As you indicated in our meeting concerning containment design matters on August 10, 1973, the Mark III vent clearing model has been revised. The new model is described in the second Mark III test program progress report, NEDM 10976, transmitted by letter from MP&L on July 31, 1973, and is shown to correlate with small scale Mark III test data. However, in the first progress report, NEDM 10848, the originsl vent clearing model was also shown to agree with small scale Mark III test data. Considering the above:

a. Describe the differences between the vent clearing models;

b. Discuss the effect that each variation in the model has on drywell differential pressure;

c. Explain the significance of each vent clearing model being able to adequately predict vent clearing phenomenon as seen in the small scale Mark III tests;

d. Specify the values of the loss coefficients and describe how they are determined. (The control volume equations used to model the horizontal vents indicate that "turning" loss coefficients are applied to the vent velocity terms.)

e. Specify the value of the effective length, L*, used in the control volume equations.

6.2.44 Specify the values of the individual loss coefficients used in the vent flow model and describe how these coefficients will be experimentally verified on the full scale Mark III test facility.

(Ref.: Item 6.2.3)

6.2.45 Provide sensitivity curves, similar to your response to Item 6.2.3, of drywell differential pressure as a function of reactor vessel level swell time following a main steam line break.

6.2.46 Provide an updated response to Item 6.2.6 concerning the potential input of feedwater energy to the containment following a loss-of-coolant accident. Specify the total amount of feedwater energy which could be added to the containment and justify your statement that this additional energy would not result in higher long term containment pressures.

6.2.47 As requested by Item 6.2.9 (c), provide a detailed description of the modeling of the drywell depressurization which occurs at about 600 seconds post-LOCA (PSAR Figure 6.2-9). Specifically discuss and provide analyses of the manner in which the spray effectiveness of the break flow is determined and provide a table of break flow and enthalpy as a function of time.

6.2.48 Supplement your response to Item 6.2.20, regarding Category II air lines within primary containment, as follows:

a. Specify the amount of service air which could be released to primary containment following failure of the Category II service air lines;

b. Discuss the principles of operation of the alarm system which would indicate to the operator than an instrument air line had broken within primary containment;

c. Specify the amount of instrument air which could be released to primary containment following failure of the Category II lines and before manual isolation of the system; and

d. Discuss the effect that the volumes of air in (a) and (c) would have on containment peak pressure.

6.2.49 Provide the design criteria which have been applied for postulated breaks in high energy, unguarded pipe lines located within primary containment but outside the drywell. Also, provide details of the analyses performed for each break, e.g., break sizes and location, blowdown rate, lape times for detection systems, containment pressure, etc., which demonstrate that the amount of blowdown fluid released to the containment is within acceptable limits.

6.2.50 As requested in Item 6.2.29 (b), specify the number of hydrogen recirculation valves which will operate as automatic vacuum relief valves.

6.2.51 Your response to Item 6.2.30 (d) does not consider positive drywell to containment differential pressures. Discuss the capability of the recirculation valves to open if a differential pressure, corresponding to the submergence of the first row of vents, existed across the valve disk.

6.2.52 Your response to Item 6.2.31 states that none of the pipes within the drywell are in enclosed subcompartments. However, from PSAR figures 5.5.10a and 5.1-4, it appears that the RHR head spray line may be located in a restricted volume within the drywell. Therefore, provide a drawing which illustrates the routing of the RHR head spray line and discuss the consequences of a rupture in the line between the reactor vessel and the inboard check valve.

6.2.53 Your response to Item 6.2.34, concerning post-accident suppression pool water levels, does not appear accurate in the following respects and should be revised:

a. Figure R.6.2.34-1 indicates that the containment pool overflows into the drywell following drywell depressurization. If the vacuum breakers are operable this should not occur.

b. Figure R.6.2.34-1 shown the water level in the drywell above the top of the weir wall (at about 5 minutes) and the water level in the annulus at the top of the wall. This does not appear possible.

c. The final level of water in the containment and vent annulus is indicated to be at the first row of vents. However, your response to Item 6.2.32 states that the vents should be submerged two feet following pool drawdown.

6.2.54 With respect to your analyses of subcompartments, describe the analytical methods that were used to determine the jet impingement forces.

6.2.55 The containment external design differential pressure is indicated to be 3 psi in the PSAR (p. R6.2.38-1). Describe and provide details of the analyses of the types of transients which could result in negative pressures within primary containment. Justify the margins which are allowed between the maximum calculated negative pressures and the design value. From this evaluation and other factors deemed pertinent, discuss the need for a containment vacuum breaker system.

6.2.56 Our review of the isolation valve arrangements described in PSAR Section 6.2.4 and Table 6.2.7 indicates that a number of primary containment penetrations do not explicity conform to General Design Criteria (GDC) 55-57. Briefly, GDC 55-57 require that two isolation valves be provided, one inside and one outside the primary containment; that these valves close automatically; and that the valves fail, on loss of power, in the position of greater safety. Considering the above, provide a list of all primary containment penetrations which do not meet these criteria and in each case provide a discussion and an analysis which demonstrate that exception to the GDC is justified. If in some cases this justification is already provided in the PSAR, then a specific page reference will be an acceptable response.

6.2.57 Provide the following information with respect to the Standby Gas Treatment System (SGTS) and secondary containment:

a. Elevation drawings which clearly indicate the boundaries of the secondary containment;

b. A list of any high energy lines which are within secondary containment;

c. The failure modes of valves in the SGTS; and

d. Justification for assuming that both trains of the SGTS are available for drawdown of the secondary containment pressure to -1/4" w.g., following an accident (Ref.: PSAR p.6.4-3).

6.2.58 Section 6.2.1.3.5 of the PSAR references Appendix A of NEDO 10329, "Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors", for the reactor vessel level swell model used in the main steam line break analysis. However, since the containment design basis break is indicated to be the main steam line, the manner in which the topical report model is applied to containment analysis requires clarification as follows:

a. Reference the specific parts of Appendix A which are applicable to the level swell associated with a main steam line blowdown analysis;

b. List any conservative assumptions which were incorporated in the model for containment analysis purposes;

c. Provide results of the level swell times that are calculated by the model considering various reactor operating conditions such as full power, hot standby etc.; and

d. List the input parameters used in the above analyses.

6.2.59 Specify the pump heat rate input (btu/sec) to the containment following a loss-of-coolant accident and justify its exclusion from the long-term containment response analysis described in PSAR Section 6.2.1.3.7.

6.2.60 Your response to Item 6.2.9 and discussions in the Regulatory staff's meeting with you on August 10, 1973, indicate that for containment mass and energy input rates, the primary system is modeled as a single volume at the average primary system enthalpy. Justify that this represents a conservative assumption for containment analysis purposes considering that the recirculation loop conditions display a significant (20 Btu/lbm) degree of sub-cooling. Provide a blowdown table (mass and energy input rates versus time) and containment response profiles for a recirculation line break assuming that the primary system is modeled as one volume, with the total mass inventory at operating pressure and at the average enthalpy of the recirculation loops.

6.2.61 Provide an analysis of containment and drywell pressure response to a main steam line or recirculation line break (whichever is the worst case) assuming that the reactor is at hot standby and the suppression pool is at an elevated temperature corresponding to the hot standby condition at the time of the break.

6.2.62 As indicated in response to Item 6.2.33, the vent flow model as described in GE Topical Report, NEDM-10320, "The General Electric Pressure Suppression Containment Analytical Model" was used to preduct the critical flow threshold for the Mark III vent system. This method is based on the principle that only the vapor region determines the sonic characteristics of the two-phase mixture. Using this assumption, ideal gas relationships have been used to predict sonic flow. There are, however, experimental and analytical data within the literature that indicate the liquid phase does lower the sonic characteristics of the mixture below the value that would be calculated for only the vapor phase. Regulatory staff members discussed with General Electric specific references indicating this trend during a meeting at San Jose on September 20 and 20, 1973. As a result, justify the applicability of the ideal gas-choked flow model in light of the contradictory data within the field, which includes the RELAP prediction comparisons of both the semiscale and Battelle blowdown experiments.

ENCLOSURE 2

STATEMENT OF REGULATORY REQUIREMENTS
GRAND GULF NUCLEAR STATION, UNITS 1 AND 2
DOCKET NOS. 50-416 AND 50-417

Item

6.2.1 The Grand Gulf Nuclear Station Mark III Containment should have the capability to operate recirculation mixing of hydrogen at any time after 10 minutes following a loss-of-coolant accident. This assures a reasonable design basis to accommodate the metal-water reaction extent indicated in Regulatory Guide 1.7. The applicant has provided an analysis in Amendment 11 which acceptably demonstrates this capability for Grand Gulf.

6.2.2 You were requested, in Item 6.2.9(c), to justify the post-blowdown drywell depressurization phase of the containment response. Your response was not adequate. Item 6.2.47 in Enclosure 1 of this letter, also addresses this concern. If this phenomenon cannot be adequately demonstrated, some positive means will need to be provided to reduce the drywell pressure and ensure the return of air from the containment to the drywell.

6.2.3 You are proposing a drywell design differential pressure which is 15% greater than the peak calculated differential pressure. We believe that a minimum of a 30% margin should be applied and that the final margin will be a function of the conservatism of the pressure response analysis which is under review.