Docket Nos. 50-416
50-417

SEP 5 1973

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

Gentlemen:

We have reviewed your responses to our request for additional information, dated May 1, 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 the area of system quality group classifications, 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 October 23, 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

OFFICE >						
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nuclear power plants by reference, or you may amend your application by submitting revised pages and supplements.

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

Sincerely,

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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OFFICE	L:BWR-1	L:BWR-1	L:AD/BWR			
SURNAME	GOwsley:sgh	WButler	VAMoore			
DATE	9/5/73	9/5/73	9/5/73			

Docket Nos. 50-416/417

Enclosure 1

Request for Additional Information

GRAND GULF NUCLEAR STATION, UNITS 1 AND 2

Item

4.3.2 In your response to Item 4.3.1, you did not provide sufficient information for the Regulatory Staff to independently evaluate the capability of the Grand Gulf reactor design to achieve and operate within its power objectives. Discuss quantitatively the procedure used by the process computer to determine local values of LHGR and CHFR. Your response should include a discussion of all process variables that are measured and all calculations that are made to provide input for use in this procedure. In addition, describe how each of the uncertainties allowed for in this procedure are obtained, and give the magnitude and confidence limits of each uncertainty.

4.4.2 Although the power distributions presented in Figures 4.4-5 and R4.4.1-3 are for the same exposure of 10,000 MWD/T, the values of the local peaking factors are different. Also, using the data provided in Figure 4.4.1-3 and assuming that 96% of the core power is deposited in the fuel, the calculated total core power is 3920 MW, which is neither the rated or design power. Explain these apparent inconsistencies.

5.11 Your response to Item 5.10 indicated that a portion of the analytical results requested would not be performed..."until about the time of the FSAR analysis." Since the Grand Gulf reactor will produce more thermal power per unit of reactor vessel volume than previous designs, the margin between code allowable pressure and the pressure following a steam line isolation transient may not be the same as in previous designs. In order to confirm that adequate safety and relief valve capacity will be provided, the analyses requested in Question 5.10 are needed. An acceptable analysis result is that the calculated peak pressure in the vessel be at least 25 psi below 110% of the vessel design pressure, following closure of all of the main steam line isolation valves or of all of the turbine stop and bypass valves, assuming one safety valve does not function and assuming that control rod scram is initiated by the high pressure signal.

6.3.8 In your response to Item 6.3.4, you stated that an emissivity of 0.9 has no significance for unwetted surfaces. The grey body view factors calculated assuming an emissivity of 0.9 requested in Question 6.3.4 are necessary in order to make a meaningful comparison of the calculated clad temperatures presented in the PSAR and the measured clad temperatures in the stainless steel clad bundle spray cooling tests, reported in NEDO-10801. Since the emissivity of unwetted stainless steel can be as high as 0.9, view factors calculated using emissivities of 0.67 and 0.9 bracket the range of view factors possible in Zircaloy fuel assemblies and stainless steel test bundles. Experimental confirmation of spray cooling heat transfer in an 8 x 8 rod array is required and can only be obtained at this time from a test with the stainless steel bundle. Therefore, we request that you provide the view factors identified in Item 6.3.4.

6.3.9 The Interim Acceptance Criteria require that an analysis of ECCS performance be made over the spectrum of breaks assuming a single failure. In your response to Item 6.3.5, you indicated that our request for analyses assuming failure of the HPCS system and one of the ADS valves represented the failure of two active components and thus was not included. The analyses requested in Item 6.3.5 represents a break in the HPCS line and a failure of a single active component, an ADS valve. Therefore, you are requested to provide the analyses cited in Item 6.3.5 to demonstrate compliance with the Interim Acceptance Criteria.

6.3.10 It appears that the peak clad temperature for the base case presented in Figure R6.3.6-1 is 1515°F but has been rounded off to "approximately 1500°F" in the text of your response to Item 6.3.6. Confirm that the base case peak temperature is 1515°F or explain why the base case temperature is lower than the temperatures given for cases a.4 (1505 F) and a.5 (1510 F) of Item 6.3.6.

6.3.11 In your response to Item 6.3.6.a.7 you indicated that an emissivity of 0.9 for unwetted surfaces was not realistic, and thus you did not respond to our request. Although a value of 0.67 may be a reasonable estimate of the emissivity of Zircaloy, the value will vary with temperature and surface conditions. An emissivity of 0.9 was selected for the analysis requested in Item 6.3.6.a.7 in order to determine the sensitivity of peak clad temperature to emissivity over any expected range of values of emissivity and so as to be consistent with the view factors requested in Item 6.3.4 (see Item 6.3.8 above). Therefore, provide the analyses requested in Item 6.3.6.a.7.

6.3.12 The analysis that you supplied in your response to Item 6.3.6.b.1 is not the one requested. Provide the peak clad temperatures using the Interim Acceptance Criteria, except that during the period from start of lower plenum flashing until the MCHFR decreases to 1.0, assume the heat transfer coefficient to be 10,000 BTU/hr ft^2 °F. After MCHFR reaches 1.0 and until the core spray reaches rated flow, assume the heat transfer coefficient to be zero.

6.3.13 The analyses that you supplied in your response to Question 6.3.6.b.2 are not the ones requested. Provide the peak clad temperatures using the Interim Acceptance Criteria except that during the period from the start of lower plenum swell until the level falls below the core midplane, assume the heat transfer coefficient to be 50 BTU/hr ft^2 °F. Repeat the analyses for coefficients of 100, 500 and 1,000 BTU/hr ft^{2} °F.

15.18 In your response to Item 15.13 you declined to perform an ATWS analysis for the Grand Gulf units until specific design criteria have been established by the AEC. However, the analyses requested in Item 15.13 are needed so that we may determine if the analyses provided in NEDO-10349 are applicable to the Grand Gulf units. Since the Grand Gulf reactor has a higher power relative to its vessel volume than the reactor analyzed in NFDO-10349, it is not evident that the results presented in the report are applicable. Therefore, provide the analyses requested in Item 15.13.

15.19 In your response to Item 15.14 you declined to provide specific analysis of fuel assembly flow blockage for the Grand Gulf unit core. The analyses requested in Item 15.14 are needed so that the review of the consequences of a flow blockage incident in the Grand Gulf unit core can be completed. Since the fuel assemblies in the Grand Gulf core have a higher power density and higher hydraulic resistance than the designs analyzed in NEDO-10174, the results presented in NEDO-10174 are not clearly applicable to Grand Gulf. Therefore, provide the requested analyses or a justification for your statement in Item 15.14 that "the slight differences in thermal-hydraulic characteristics between the two fuel designs will only slightly modify the analyses..."

Enclosure 2

Statement of Regulatory Requirements

GRAND GULF NUCLEAR STATION, UNITS 1 AND 2

Item

3.2.0 Classification of Structures, Components, and Systems

3.2.1 Your response to Question 3.2.1 (e) as provided in Amendment 8 to the PSAR, indicated that no additional nondestructive testing requirements (other than those specified in API-650) will be required for the Condensate Storage Tank. This is unacceptable to the Regulatory staff.

While this component is not essential for the safe shutdown of the plant, it is the preferred water supply source for the HPCS. Therefore, to be acceptable to the Regulatory staff, the specifications for this tank will have to require (1) 100 percent surface examination of the side wall to bottom joint, and (2) 100 percent volumetric examination of the side wall weld joint.

3.2.2 With regard to your response to Question 3.2.2 (e) as provided in Amendment 8 to the PSAR, your classification of the cooling lines to the Reactor Recirculation Pumps is unacceptable.

The Regulatory Staff considers the Reactor Recirculation Pumps to be safety-related components. Therefore, we require that the classification of the cooling lines to these pumps be Safety Class 3 and Seismic Category 1.

3.2.3 In your response to Question 3.2.2 (f) as provided in Amendment 8 to the PSAR, you indicated that the Standby Gas Treatment System is classified Safety Class 3 because it does not meet the conditions of Quality Group B (Safety Class 2) as defined in Safety Guide 26. This is unacceptable.

The Standby Gas Treatment System is considered by the Regulatory Staff to be an Engineered Safety System. Therefore, to be acceptable to the Staff, the system will have to be classified Safety Class 2.

3.2.4 The classification of the turbine stop valves in Table 3.2.1, Subsection XXXI, Part 1 and in Note i of the same table is unacceptable. To be acceptable to the Regulatory Staff, these valves will have to be classified Safety Class 2, Seismic Category 1 and constructed to the requirements of the ASME Boiler and Pressure Vessel Code Section III, Class 2.

3.2.5 In Table 3.2.1, Subsection XX, Fuel Pool Cooling and Cleanup System, the non-seismic Category 1 Classification of that portion of the system which performs a cooling function is not in agreement with current AEC practice and is unacceptable. To be acceptable to the Regulatory Staff, the cooling loop of the Fuel Pool Cooling and Cleanup System will have to be classified Seismic Category 1.