UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos. 50-416
and 50-417

JUN 8 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. J. P. McGaughy, Jr.
Assistant Vice President, Nuclear Production
Mississippi Power and Light Company
Post Office Box 1640
Jackson, Mississippi 39205

Dear Mr. McGaughy:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - GRAND GULF NUCLEAR STATION, UNITS 1 AND 2

As a result of our review of the information contained in the Final Safety Analysis Report for the Grand Gulf Nuclear Station, Units 1 and 2, we have developed the enclosed request for additional information. Included are questions from the Auxiliary Systems Branch and Reactor Systems Branch.

We request that you amend your Final Safety Analysis Report to reflect your responses to the enclosed requests as soon as possible and to inform the Project Manager, Joseph A. Martore, of the date by which you intend to respond.

Sincerely,

R. Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
Request for Additional Information

cc w/enclosure:
See next page

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Mr. J. P. McGaughy
Assistant Vice President - Nuclear Production
Mississippi Power & Light Company
P. O. Box 1640
Jackson, Mississippi 39205

cc: Robert B. McGehee, Esq.
Wise, Carter, Child, Steen & Caraway
P. O. Box 651
Jackson, Mississippi 39205

Troy B. Conner, Jr., Esq.
Conner, Moore & Corber
1747 Pennsylvania Avenue, N. W.
Washington, D. C. 20006

Mr. Adrian Zaccaria, Project Engineer
Grand Gulf Nuclear Station
Bechtel Power Corporation
Gaithersburg, Maryland 20760

Mr. Alan G. Wagner, Resident Inspector
Route 2 Box 150
Port Gibson, Mississippi 39150

Mr. N. L. Stampley, Sr. Vice President
Engineering, Production & Construction
P.O. Box 1640
Jackson, Mississippi 39205

Mr. L. F. Dale
Nuclear Project Manager
P.O. Box 1640
Jackson, Mississippi 39205

Mr. John Richardson
P. O. Box 1640
Jackson, Mississippi 39205

POOR ORIGINAL

REQUEST FOR ADDITIONAL INFORMATION
GRAND GULF NUCLEAR STATION UNITS 1&2
AUXILIARY SYSTEMS BRANCH

010.28 (4.6) Describe the means provided in the design of the scram discharge system and verify that it meets the criteria enumerated in the Generic Safety Evaluation Report BWR Scram Discharge System, dated December 1, 1980, and transmitted to you by NRC letter dated December 22, 1980.

010.29 (4.6) RSP Demonstrate that a slow or partial loss of air pressure to the scram discharge valves will not result in the following:

1) Rapid filling of both the scram discharge volume and the instrument volume due to the lifting of most or all scram discharge valves, with consequent loss of adequate scram discharge volume.

2) Loss of reactor coolant due to the combination of lifting of most or all scram discharge valves, without compensating closure of the vent and drain valves, with consequent environmental effects inside containment.

Unless it can be demonstrated that no adverse effects can result, a system shall be provided and described in this section to protect against these two conditions.

010.30 (4.6) Describe the effects on the safety and operability of the control rod drive hydraulic system if the drive/cooling water pressure control valve (F003) fails either closed or open.

POOR ORIGINAL

010.31 (4.6) Describe the means provided in the control rod drive system design to meet the criteria enumerated in NUREG-0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking and verify that this design is in full compliance with this document.

REACTOR SYSTEMS BRANCH

211.211 Calculations of NPSH available to ECCS pumps in BWRs are normally provided with reference to the pump suction. We are concerned that under certain post accident conditions the potential may exist for damage to ECCS pumps from cavitation because of local flashing in the system suction lines. The potential can result for example from local elevation changes in the piping runs. Calculations of NPSH available at the pump suction may erroneously assume liquid continuity up to the point of pump suction. We require therefore that the applicants provide calculations demonstrating that all points in all safety related suction piping, the NPSH available is adequate to preclude local flashing under the worst postulated conditions.