J. R. Buchanan, IC
T. B. Abernathy, TIC
J. Yore, ASLBP
ACRS (16)

JUN 27 1978

Distribution
Docket File
NRC PDR
Local PDR
NRR Reading
LWR 1 File
R. S. Boyd
R. C. DeYoung
D. B. Vassallo w/o encl
F. J. Williams w/o encl
R. Mattson
J. Stolz
C. Thomas
E. Hylton
J. Rutberg w/o encl
J. Saltzman w/o encl
C. Miles w/o encl
N. Dube
G. Williams
K. Singer
B. Grimes
R. Tedesco
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IE (3)
H. Denton
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B. Youngblood
J. C. Stepp
L. G. Hulman
ELD
EP PM
EP LA
EP BC
J. P. Knight
W. Ross w/o encl
R. Clark w/o encl

Docket Nos. 50-416
and 50-417

POOR ORIGINAL

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

Gentlemen:

SUBJECT: ACCEPTANCE REVIEW
(GRAND GULF NUCLEAR STATION, UNIT NOS. 1 AND 2)

On April 28, 1978, you tendered an application for licenses to operate the Grand Gulf Nuclear Station, Unit Nos. 1 and 2. Your application included a Final Safety Analysis Report, an Environmental Report, and a document containing general information.

We have completed our review of your tendered application and have concluded that it is acceptable for docketing. Your application should be provided to us as soon as possible. Your filing of the application should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen (15) copies of that portion of the application containing the general information, forty (40) copies of the Final Safety Analysis Report, and forty-one (41) copies of the Environmental Report. Your filing of any amendments to the application should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization and, in addition, fifteen (15) copies of that portion of the application containing the general information, sixty (60) copies of the Final Safety Analysis Report, and forty-one (41) copies of the Environmental Report as appropriate. You should retain an additional ten (10) copies of the general information, thirty (30) copies of the Final Safety Analysis Report, and one-hundred nine (109) copies f the Environmental Report for direct distribution in accordance th Enclosure 1 to this letter and further instructions which might provided later. Within ten days after docketing, you must provide ffidavit that distribution in accordance with Enclosure 1 has completed. These requirements also apply to all subsequent ents to your application.

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1033 120

POOR ORIGINAL

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The Environmental Report is considered sufficiently complete for us to begin our detailed environmental assessment. However, additional information will be required to enable us to complete our detailed environmental assessment and to prepare our Environmental Statement. This information will be requested separately by the Division of Site Safety and Environmental Analysis.

If, during the course of our review, you believe there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that an appropriate meeting can be arranged on a timely basis. A written request is not necessary and all such requests should be initiated through our staff project manager assigned to the review of your application. This procedure is an informal one, designed to allow opportunity for applicants to discuss, with management, areas of disagreement in the case review.

Sincerely,

Original Signed By
Roger S. Boyd

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Application Distribution List
2. Acceptance Review Request for Additional Information.

cc: See page 3

EP/BC*	EP/AD*
RSC/RLB	VAM
6/21/78	6/21/78

SEE PREVIOUS YELLOW FOR PREVIOUS CONCURRENCES*

OFFICE	LWR 1	LWR 1	LWR/AD	DPM	DPM	
SURNAME	CThomas/red	JStolz	DBVassallo	RCDeYoung	RSBoyd	
DATE	6/26/78	6/27/78	6/27/78	6/ /78	6/27/78	

JUN 27 1978

POOR ORIGINAL

cc: Mr. Robert C. Travis, Attorney
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

Bechtel Power Corporation
ATTN: T. W. Habernas, Project Engineer
Grand Gulf Nuclear Station
Gaithersburg, Maryland 20760

OFFICE ➤						
SURNAME ➤						
DATE ➤						

bcc: J. R. Buchanan, ASIC
T. B. Abernathy, TIC
J. Yore, ASLBP
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and 50-417

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ATTN: Mr. N. L. Stampley
Vice President - Production
P. O. Box 1640
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We have completed our review of your tendered application and have concluded that it is acceptable for docketing provided that when you docket your application, it reflects that you are applying for licenses to operate the Grand Gulf Nuclear Station, Unit Nos. 1 and 2 at steady state reactor core power levels not in excess of 3800 megawatts thermal. This stipulation is in accordance with the Commission's policy referenced in Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," which provides that applications should not be submitted for reactor core power levels greater than 3800 megawatts thermal before January 1, 1979 at the earliest and that we will issue a notice of intent to consider applications for reactor core power levels greater than 3800 megawatts thermal at least two years prior to accepting such applications. It is acceptable, however, for your application to reflect that the reactors have been designed to operate at core power levels of 3833 megawatts thermal.

Your application should be provided to us as soon as possible. Your filing of the application should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen (15) copies of that portion of the application containing the general information, forty (40) copies of the Final Safety Analysis Report, and forty-one (41) copies of the Environmental Report. Your filing of any amendments to the application should include three (3) originals

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RSC/m RLB
EP/BC
6/21/78

EP/AD
6/21/78

bcc: J. P. Buchanan, NSIC
T. B. Abernathy, TIC
J. Yore, ASLBP
ACRS (16)

OFFICE ➤	LWR 1	LWR 1	LWR/AD	ELD	DPM	DPM
SURNAME ➤	CThomas/red	JSTOLZ JFS	DBVassallo		BCDeYoung	RSBoyd
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Sincerely,

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

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OFFICE	LWR 1	LWR 1	LWR/AD	ELD	DPM	DPM
SURNAME	GThomas/red	JStolz	DBVassallo		RCDeYoung	RSBoyd
DATE	6/22/78	6/22/78	6/23/78	6/ /78	6/ /78	6/ /78

ENCLOSURE 1

APPLICATION DISTRIBUTION LIST

GRAND GULF NUCLEAR STATION

UNIT NOS. 1 AND 2

DOCKET NOS. 50-416 AND 50-417

DISTRIBUTION LIST

APPLICATION, SAFETY ANALYSIS REPORT AND AMENDMENTS

STATE OFFICIALS(S)

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

Attorney General
Gartin Building
Jackson, Mississippi 39205

cc: Coordinator (w/o encl)
Suite 400, Watkins Building
510 George Street
Jackson, Mississippi 39201

LOCAL OFFICIAL(S)

Mr. William Matt Ross, President
Claiborne County Board of
Supervisors
Port Gibson, Mississippi 39150

U. S. ENVIRONMENTAL PROTECTION AGENCY
REGIONAL OFFICE (SAR & AMENDMENTS ONLY)

U. S. Environmental Protection Agency
ATTN: EIS Coordinator
Region IV Office
345 Courtland Street
Atlanta, Goergia 30308

NATIONAL LABORATORY

H. E. Zittel
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37830

DISTRIBUTION LIST

ENVIRONMENTAL REPORT, SUPPLEMENTS AND AMENDMENTS

The updated distribution list enclosed in our letter to you dated April 25, 1978 should be used for the distribution of the Environmental Report, supplements and amendments.

ENCLOSURE 2

ACCEPTANCE REVIEW

REQUEST FOR ADDITIONAL INFORMATION

GRAND GULF NUCLEAR STATION

UNIT NOS. 1 AND 2

DOCKET NOS. 50-416 AND 50-417

010.0 AUXILIARY SYSTEMS

010.1 (3.6) (a) Your summary in Tables 3.6A-18 through 3.6A-27 on the type of protection provided and the type of hazard (i.e., whipping, jet impingement, spraying, and flooding) is incomplete. You describe some of the types of protection that will be provided but you do not state for which types of hazards these will be provided or the consequences of the pipe failure. Complete these tables and verify that the essential items will be qualified for the environment (i.e., temperature, pressure, and humidity) resulting from a pipe failure.

(b) We require that the compartment between the containment and the reactor building which houses the main steam lines and feedwater lines and their isolation valves, be designed to consider the environmental effects (pressure, temperature, humidity) and potential flooding consequences from an assumed crack, equivalent to the flow area of a single ended pipe rupture in these lines. We require that essential equipment located within the compartment, including the main steam isolation and feedwater valves and their operators be capable of operating in the environment resulting from the above crack. We also require that if

010.1 this assumed crack could cause the structural failure of this compartment, then the failure should not jeopardize the safe shutdown of the plant. In addition, the remaining portion of the pipe in the tunnel between the reactor building and the turbine building should meet the guidelines of Branch Technical Position ASB 3-1.

Submit a subcompartment pressure analysis to confirm that the design of both areas of the pipe tunnel conforms to our position as outlined above.

Evaluate the design against this staff position, and advise us as to the outcome of your review, including any design changes which may be required. The evaluation should include a verification that the methods used to calculate the pressure build-up in the subcompartments outside of the containment for postulated breaks are the same as those used for subcompartments inside the containment. Also, the allowance for structural design margins (pressure) should be the same. If different methods are used, justify the margins that are available. In your submittal, identify the computer codes used, the mass and energy release rates assumed, and provide sufficient design data so that we may perform independent calculations.

010.2 (9.1.4) Provide information describing how the cask travel is limited or prevented from being transported over the spent fuel racks, including a detailed drawing showing the limits to the path of travel.

021.00 CONTAINMENT SYSTEMS

021.01 General In our SER and following supplements for the Grand Gulf Nuclear Station CP application several commitments were made referencing specifically pool dynamic test data and the GESSAR docket. Verify, through schedule comparisons with GESSAR, GGNS understanding for the submittal of test data and substantiating reports on your plant. Further, discuss provisions made to implement design changes which may become necessary as a result of review of the data.

021.02 (9.4.7) The design and proposed operation of the containment purge system is not in complete conformance with our Branch Technical Position CSB 6-4 "Containment Purging During Normal Operation." We note that the FSAR does not present information addressing several items in Section B of the BTP. Therefore, provide information in accordance with the attached position and address the following areas:

(a) B.1.b - number of purge and vent lines

(b) B.1.c - size of purge and vent lines

(c) B.1.f - isolation valve closure time

(d) B.1.g - valve closure will not be affected by debris

(e) B.2. - temperature and humidity control

(f) B.5.a - radiological analysis

(g) B.5.c - reduction in containment pressure

(h) B.5.d - allowable leak rates

021.03 (6.2.1) In the unlikely event of a pipe rupture inside a major component subcompartment the initial blowdown transient would lead to nonuniform pressure loadings on both the structure and the enclosed component(s). To assure the integrity of these design features, we request that you provide the following information:

a) Provide a schematic drawing showing the compartment nodalization for the determination of structural loads, and for the component supports evaluation. Provide sufficiently detailed plan and section drawings for several views, including principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas to permit verification of the subcompartment nodalization and vent locations.

b) Describe the nodalization sensitivity study performed to determine the minimum number of volume nodes required to conservatively predict the maximum pressure load acting on the compartment structure. The nodalization sensitivity study should include consideration of spatial pressure variation; e.g., pressure variation circumferentially, axially, and radially within the compartment. Describe and justify the nodalization sensitivity study performed for the major component supports evaluation, where transient forces and moments acting on the components are of concern.

c) Discuss the manner in which movable obstructions to vent flow (such as insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification that vent areas will not be partially or completely plugged by displaced objects. Discuss how insulation for piping and components was considered in determining volumes and vent areas.

d) Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

021.04 (6.2.1) Identify the types of insulation used within the containment (e.g., reflective metal insulation, mass insulation, and encapsulated (sheathed) mass insulation) and discuss the methods of attachment to piping and components. Estimate, for a representative break location, the amount of insulation material that would be removed from the pipes by a LOCA. On the basis of the properties and characteristics of this material determine the locations it would accumulate and in what form. Discuss the potential for loose insulation and other debris to clog drains leading to the sump and the sump screening.

021.05 (6.2.1) With regard to all Class 1E equipment located inside the containment building such as CRD hydraulic system, reactor vessel supports and all incore instrumentation leads, we require that the environment is maintained within the temperature range for which the equipment is qualified to operate.

Indicate if the Reactor Building Ventilation System (RBVS) is required to assist in the maintaining of an acceptable temperature range. If it is, provide the following information on the RBVS:

1. Justification for not treating this system as an ESF system.

2. The results of an analysis that the RBVS will not be a potential source for missiles and meets our pipe whip criteria.

3. A discussion on the operating procedures to be initiated should the RBVS be unavailable.

4. The location of all temperature sensors associated with the operation of the RBVS.

5. The requirements imposed on this system in order to perform all Appendix J testing.

021.06 (6.2.1) With respect to purging of the drywell, present a discussion as to the interlocks which exist on the system as well as the specific control and state not only the purpose for purging but also during what specific reactor modes the purging system will operate.

021.07 (6.2.1) Provide the maximum calculated negative pressure following a design basis accident of the containment spray system and the external design pressure. Include a description of the analytical model and justify that the assumptions used to determine the internal containment pressure response are conservative.

021.08 (6.2.5) Identify (1) the location of the hydrogen sample points in the drywell and suppression chamber and (2) location of CGCS suction and discharge points, with respect to local structures and equipment.

021.09 (6.2.6) Discuss and schematically show the design provisions that will permit the personnel air-lock door seals and the entire air lock door seals to be tested. Discuss the design capability of the door seals to be leak tested at a pressure of Pa; i.e., the calculated peak containment internal pressure. If it will be necessary to exert a force on the doors to prevent them from being unseated during leak testing, describe the provisions for doing this and discuss whether or not the mechanism can be operated from within the air lock. Also, discuss how the force on the door will be monitored.

031.00 INSTRUMENTATION AND CONTROL SYSTEMS

031.01 (3.11) Table 3.11-7 of the FSAR shows in several instances that the temperature profile or maximum temperature is listed as 100°F and in some cases 104°F. We have a concern if this is intended to indicate the maximum temperature to which the equipment was qualified. Table 2.3-2 indicates that the temperature extreme for the area in which the plant site is located has been as high as 107°F.

Also Table 2.3-2 indicates that temperatures in the area have reached a low of minus 5°F. There is no reference to the low temperature environments in the qualification tables.

We require that the environmental qualification program for all safety related equipment include the complete environmental envelope. That is both maximum and minimum values expected to occur during plant shutdown, normal operation abnormal operation and during any design basis event and post design basis event. Provide this information and include any necessary changes in the environmental qualification program of affected equipment.

031.02 (3.11) We require that the qualification program be provided for at least one item in each of the following groups of Class 1E equipment. (Both NSSS supplied and B.O.P. equipment).

a) Switchgear

b) Motor control centers

c) Valve operators (in containment)

d) Motors

e) Logic equipment

f) Cables

g) Diesel generator control equipment

h) Sensors

i) Limit switches

j) Heaters

k) Fans

l) Control Boards

m) Instrument racks and panels

n) Connectors

o) Penetrations - including design provisions for the over-current protection circuits

p) Splices

q) Terminal blocks and

r) Terminal cabinets

The qualification program should include:

a) Identification of Equipment including,

   1) Manufacturer

   2) Manufacturer's type number

   3) Manufacturer's model number

b) Equipment design specification requirements, including,

1) The system function requirements

2) An environmental envelope which includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown, normal operation, abnormal operation, and any design basis event.

3) Time required to fulfill its function when subjected to any of the extremes of the environmental envelope specified above.

c) Test plan,

d) Test set-up,

e) Test procedures,

f) Acceptability Goals and requirements,

g) Test results

h) Identification of the documents which include and describe the above items

i) Justification must be provided when analyses is used to quality equipment.

In accordance with the requirements of Appendix B of 10 CFR 50 the staff requires a statement verifying: 1) that all remaining Class 1E equipment will be qualified to the program described above and 2) that the qualification information will be available for an NRC audit.

Provide the information requested above.

031.03 (3.11,7.2 8.0) Section 8.3.1.1.5.1 of the FSAR states that the RPS (Reactor Protection System) power is classified as non-essential, because failure of the power supply causes a reactor scram. Also Figure 8.3-11 shows that this RPS power is supplied to (1) the RPS, (2) the radiation and neutron monitors and (3) the nuclear steam supply shut off systems.

Since the power supplies to this system are not Class 1E we assumed that the voltages and frequencies may exceed or fall below the values used to qualify the safety related equipment connected to the bus. High and low voltages and low frequencies may degrade the safety related equipment to the point when they may not perform their function within the required time frame. This may be the case even though the failure mode may be for the equipment to return to its deenergized mechanical restored position and perform its function, when power is removed. Equipment will be subjected to high currents as a result of high voltage, low frequencies and low voltages which fail to actuate components.

Describe how the Grand Gulf design prevents this type of degradation to safety related equipment.

031.04 (7.4) Section 7.4.1.4 of the FSAR describes the remote shut down system. Describe the features of this design which demonstrate that the system satisfies the requirements of GDC 19 and 34. Also, describe the provisions included in this design that assure redundant features will not be degraded by a single failure.

031.05 (7.5) Information provided in FSAR section 7.5.2.5.5 indicates that the recommendations of Regulatory Guide 1.47 will not be satisfied. We require that a bypass indication system be provided for this plant to assure that the operator is aware of the plant status at all times. A commitment to conform to the provisions of Regulatory Guide 1.47 during the CP review (included in the CP - SER) was provided. Therefore we require that this final design satisfy this commitment. Provide a modified design to include a bypass indication system which satisfies the recommendations of Regulatory Guide 1.47.

031.06 (7.5) Describe how this design satisfies position No. 23 listed in Appendix 7A in the Standard Review Plan.

031.07 (7.1.2.4 3.11) Section 7.1.2.4 of the FSAR does not indicate conformance to IEEE Std 323-1971. However, Section 3.11.2.5.1 states that non NSSS supplied equipment purchased before November 1974 has been qualified in accordance with IEEE Std 323-1971 and equipment purchased after November 1974 has been qualified in accordance with IEEE Std 323-1974. Correct this inconsistancy.

031.08 (3.11) Section 3.11 of the FSAR does not indicate that the NSSS Class 1E equipment will satisfy the requirements of IEEE-323-1971 nor the 1974 version. Identify and justify all exceptions to this standard for qualifying all NSSS supplied Class 1E equipment.

031.09 (3.10) In view of the natural phenomenon (tornado) which recently occurred at your site, identify all portions of the IE equipment that were damaged or degraded during this event. In addition provide a description which justifies that each of those equipment still satisfy the requirements of the present design and the commissions requirements identified in Table 7.1 of the Standard Review Plan.

031.10 (7.2) (7.3) Justify use of RPS inputs from devices mounted on non-seismically qualified equipment and/or located in non-seismically qualified enclosures. Specific examples are turbine trip inputs and generator load rejection inputs. Also justify use of these inputs for recirculation pump trip (RPT). Identify and justify any other trip inputs that may be included in this category.

031.11 (7) Identify and justify all exceptions taken to all branch technical positions listed in appendix 7A of the Standard Review Plan. Where these positions are discussed in the FSAR, identify all sections where they appear.

031.12 (7) Identify and justify all deviations to the acceptance criteria listed on Table 7.1 of the Standard Review Plan. If this information is already included in the FSAR list each section where it is discussed.

031.13 (7) Several previously reviewed BWR installations included a start up transient monitoring system to provide recordings for selected parameter during the start up and warranty testing. There has been no information in the FSAR which describes this system. If this system or any similar system is intended for use in the Grand Gulf Units provide the following information.

1. Provide separation requirements for all permanently installed cables and equipment.

2. Provide separation requirements for all temporary cables and equipment.

3. Identify all safety related parameters which will be monitored.

4. Describe the isolation devices, with acceptable qualification, used to isolate the safety related parameters from the monitoring system.

5. Describe all other provisions in this design to prevent failures of this system from degrading all safety related systems.

040.0 POWER SYSTEMS

040.1 (8.3) Diesel generator alarms in the control room: A review of malfunction reports of diesel generators at operating nuclear plants has uncovered that in some cases the information available to the control room operator to indicate the operational status of the diesel generator may be imprecise and could lead to misinterpretation. This can be caused by the sharing of a single annunciator station to alarm conditions that render a diesel generator unable to respond to an automatic emergency start signal and to also alarm abnormal, but not disabling, conditions. Another cause can be the use of wording of an annunciator window that does not specifically say that a diesel generator is inoperable (i.e., unable at the time to respond to an automatic emergency start signal) when in fact it is inoperable for that purpose.

Review and evaluate the alarm and control circuitry logic for the diesel generators at your facility to determine how each condition that renders a diesel generator unable to respond to an automatic emergency start signal is alarmed in the control room. These conditions include not only the trips that lock out the diesel generator start and require manual reset, but also control switch or mode switch positions that block automatic start, loss of control voltage, insufficient starting air pressure or battery voltage, etc. This

review should consider all aspects of possible diesel generator operational conditions for example, test conditions and operation from local control stations. One area of particular concern is the unreset condition following a manual stop at the local station which terminates a diesel generator test and prior to reseting the diesel generator controls for enabling subsequent automatic operation.

Provide the details of your evaluation, the results and conclusions, and a tabulation of the following information:

(a) all conditions that render the diesel generator incapable of responding to an automatic emergency start signal for each operating mode as discussed above;

(b) the wording on the annunciator window in the control room that is alarmed for each of the conditions identified in (a);

(c) any other alarm signals not included in (a) above that also cause the same annunciator to alarm;

(d) any condition that renders the diesel generator incapable of responding to an automatic emergency start signal which is not alarmed in the control room; and

(e) any proposed modifications resulting from this evaluation.

040.2 (8.2) In regard to the physical separation between the preferred power sources from the service transformers and ESF Transformer to the onsite Class 1E power system, sufficient information has not been provided in the FSAR to demonstrate compliance with NRC General Design Criteria 1, 3, 4, 17, and 18. Provide this information.

040.3 (3.11) (8.0) The staff requires that the following qualification test program information be provided for all Class 1E equipment:

1. Identification of Equipment including,
   a) Manufacturer
   b) Manufacturer's type model
   c) Manufacturer's model number
   d) All Class 1E equipment should be identified including the following, as applicable:
      1) Switchgear
      2) Motor control centers,
      3) Valve operators
      4) Motors
      5) Logic equipment
      6) Cable
      7) Diesel generator control equipment
      8) Sensors
      9) Limit Switches
      10) Heaters

11) Fans

12) Control Boards

13) Instrument racks and panels

14) Connectors

15) Electrical penetrations

16) Splices

17) Terminal blocks

2. Equipment design specification requirements, including,

   a) The system safety function requirements

   b) An environmental envelope as a function of time which includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown, normal operation, abnormal operation, and any design basis event including LOCA and MSLB.

   c) Time required to fulfill its safety function when subjected to any of the extremes of the environmental envelope specified above.

3. The qualification test plan, test set-up, test procedures, and acceptability goals and requirements.

4. For equipment subject to a design basis accident environment, the actual qualification envelope simulated during testing (defining the duration of the hostile environment and the margin in excess of the design requirements).

5. A summary of test results (or schedule for submission) that demonstrates the adequacy of the qualification program. If analysis is used for qualification justification must be provided.

6. Identification of the qualification documents which contain detailed supporting information, including test data, for items 3, 4 and 5.

The information requested in items 1, 2, 4, 5 and 6 shall be provided for all items of Class 1E equipment. The information in item 3 shall be provided for at least one of each group of equipment of item 1d (as applicable) which is subject to a design basis accident environment. The information in item 3 shall also be provided for representative major equipment of item 1d which is not subject to a design basis accident environment.

In addition, in accordance with the requirements of Appendix B of 10 CFR 50, the staff requires a statement verifying: 1) that <u>all</u> Class 1E equipment has been (OL) or will be (CP) qualified to the program described above, and 2) that the detailed qualification information is (or will be) available for an NRC audit.

040.4 (8.3) (3.11) Qualification of Safety Related Cable

The Regulatory staff is currently requesting, of all plants in OL review, information on the use of poleythelen type cable in safety systems. These type cables were found to have degraded considerably after many years of installed operation at the Savannah fuel processing plant.

Identify all safety related cable used in your design that has polyethelene in its construction. Provide the following information for each type of cable identified:

a) Type of cable by name and Cat. No.

b) Manufacturer

c) Type of polyethelene used

d) How is the polyethelene used in the cables construction, i.e., insulation and/or jacket.

e) Results of environmental qualification tests performed.

040.5 (8.3) Qualification of Penetrations

Describe how your design meets the recommendations of Regulatory Guide 1.63, Revision 1.

Identify each type of electrical circuit that penetrates containment. Describe the primary and backup over current protection systems provided for each type of circuit. Describe the fault-current-versus-time for which the primary and backup protection systems and the penetrations are designed and qualified.

Provide coordinated curves which demonstrate, for each circuit identified, that the maximum fault-current-versus-time condition to which the penetration and cable were qualified will not be exceeded.

Describe the provision for periodic testing under simulated fault conditions.

040.6 (3.11) (8.3) Potential Problem with Containment Electrical Penetration Assemblies

Recent operating experience at Millstone Unit No. 2 has shown that the deterioration of the epoxy insulation between splices has caused electrical shorts between conductors within a containment electrical penetration assembly. Indicate what tests and/or analysis that have been performed to demonstrate the acceptability of the design in this regard. Provide whatever information is required to perform an independent evaluation of this aspect of the electrical penetration design.

040.7 (8.2) Recent operating experience has shown that adverse effects on the safety-related power system and safety related equipment and loads can be caused by sustained low or high grid voltage conditions. We therefore require that your design of the safety related electrical system meet the following staff positions. Supplement the description of your design in the FSAR to show how it meets these positions or provide appropriate analyses to justify non-conformance with these positions.

1. We require that an additional level of voltage protection for the onsite power system be provided and that this additional level of voltage protection shall satisfy the following criteria:

   a) The selection of voltage and time set points shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;

   b) The voltage protection shall include coincidence logic on a per bus basis to preclude spurious trips of the offsite power source;

   c) The time delay selected shall be based on the following conditions:

      (1) The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the PSAR accident analyses;

(2) The time delay shall minimize the effect of short duration disturbances from reducing the availability of the offsite power source(s); and

(3) The allowable time duration of a degraded voltage condition at all distribution system levels shall not result in failure of safety systems or components;

d) The voltage sensors shall automatically initiate the disconnection of offsite power sources whehenver the voltage set point and time delay limits have been exceeded;

e) The voltage sensors shall be designed to satisfy the applicable requirements of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"; and

f) The Technical Specifications shall include limiting condition for operation, surveillance requirements, trip set points with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.

2. We require that the current system designs automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency buses. The design shall also include the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstatement feature shall be verified during the periodic testing identified in Position 3.

   In the event an adequate basis can be provided for retaining the load shed feature when loads are energized by the onsite power system, we will require that the setpoint value in the Technical Specifications, which is currently specific as "...equal to or greater than..." be amended to specify a value having maximum and minimum limits. The licensee' bases for the setpoints and limits selected must be documented.

3. We require that the Technical Specifications include a test requirement to demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during shutdown. The Technical Specifications shall include a requirement for tests: (1) simulating loss of offsite power; (2) simulating loss of offsite power in conjunction with a safety feature actuation signal; and (3) simulating interruption and subsequent reconnection of onsite power sources to their respective buses. Proper operation shall be determined by:

a). Verifying that on loss of offsite power the emergency buses have been de-energized and that the loads have been shed from the emergency buses in accordance with design requirements.

b) Verifying that on loss of offsite power the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected shutdown loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the shutdown loads.

c) Verifying that on a safety features actuation signal (without loss of offsite power) the diesel generators start on the autostart signal and operate on standby for five minutes.

d) Verifying that on loss of offsite power in conjunction with a safety features actuation signal the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected emergency (accident) loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the emergency loads.

e) Verifying that on interruption of the onsite sources the loads are shed from the emergency buses in accordance with design requirements, and that subsequent loading of the onsite sources is through the load sequencer.

4. The voltage levels at the safety-related buses should be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Provide a description of the method for making this verification; before initial reactor power operation, provide the documentation required to establish that this verification has been accomplished.

040.8 (9.5.4) Provide in Section 9.5.4 the means for indicating, controlling and monitoring the emergency diesel engine fuel oil temperature (SRP 9.5.4, Part III, Item 1).

040.9 (9.5.5) The diesel engine generator sets should be capable of operation at less than full load for extended periods without degradation of performance or reliability. Provide a discussion of your diesel engine operating parameters, including minimum load requirements, and relate this to anticipate minimum loads under accident recovery conditions and during accident standby operation when offsite power is available (SRP 9.5.5, Part III, Item 7).

040.10 (9.5.8) Section 9.5.8 states that the diesel generator combustion air intake and exhaust system is missile protected. Provide further description (with the aid of drawings) explaining how the openings in the diesel generator building for the air exhaust are protected from tornado borne missiles.

040.11 (10.2) Discuss what protection will be provided the turbine overspeed control system equipment and associated electrical wiring and hydraulic lines from the effects of a high or moderate energy pipe failure so that the turbine overspeed protection system will not be damaged to preclude its safety function. (SRP 10.2, Part III, Item 8).

040.12 (10.2) Describe with the aid of drawings, the bulk hydrogen storage facility including its location and distribution system. Include the protective measures considered in the design to prevent fires and explosions during operations such as filling and purging the generator, as well as during normal operations.

040.13 (10.2) The FSAR discusses the main steam stop and control, and reheat stop and intercept valves. Show that a single failure of any of the above valves cannot disable the turbine overspeed trip functions. (SRP 10.2, Part III, Item 3).

110.0 MECHANICAL ENGINEERING

110.1 (3.9.1) (3.9.3) Recent analyses have shown that certain reactor system components and their supports may be subjected to previously underestimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations. It is therefore necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. For the purpose of this request for additional information the reactor system components that require reassessment shall include:

a. Reactor Pressure Vessel
b. Fuel Assemblies, including Grid Structures
c. Control Rod Drives
d. ECCS Piping that is attached to the Primary Coolant Piping
e. Primary Coolant Piping
f. Reactor Vessel and Pump Supports
g. Reactor Internals
h. Biological Shield Wall
i. Pump Compartment Wall

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above-mentioned reactor system components and the reactor cavity structure.

1. Provide arrangement drawings of the reactor vessel and pump support systems to show the geometry of all principal elements and materials of construction.

2. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

   (a) Steam line nozzles to piping terminal ends.
   (b) Feedwater nozzle to piping terminal ends.
   (c) Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials[1/] on these systems/components in combination with all external loadings including safe shutdown earthquake loads. Note that we require that responses from these loads be combined by the Absolute Sum method unless acceptable

[1/] Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the vessel and the shield, and transient differential pressures across the core barrel within the reactor vessel.

110.1 (3.9.1) (3.9.3) (Cont'd.)

justification is provided for use of an alternative method. This assessment may utilize the following mechanistic effects as applicable:

a. limited displacement break areas
b. fluid-structure interaction
c. actual time-dependent forcing function
d. reactor support stiffness
e. break opening times

3. If the results of the assessment required by 2. above indicate loads leading to inelastic action in these systems or displacement exceeding previous design limits provide an evaluation of the following:

   a. Inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

4. For all analysis performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

5. Demonstrate that active components will perform their safety function when subjected to the postulated loads resulting from a pipe break in the reactor coolant system.

6. Demonstrate the functional capability of any essential piping when using service level C or D limits. Guidance on acceptable methods for proceeding with the demonstration are attached.

110.2

Most of the operating BWR plants have reported finding radial cracks on the reactor vessel feedwater nozzle and the CRD return line. Describe what design modifications will be made to eliminate this problem. In addition, provide a description of the analyses that will be performed to demonstrate the adequacy of the reactor vessel feedwater nozzle and CRD return line to withstand the imposed service condition without the cracking experienced in the operating plants.

110.3 (3.9.3)

Subparagraph NCA-1130(b) of the ASME B & PV Code Section III requires non-code mechanical or electromechanical devices such as valve operators to be covered by the code when these devices act as component supports. Provide a commitment to insure that the design of devices which become attachment points for component supports, thus providing component support load path, will adequately consider these support loadings.

110.4 (3.9.3) Describe the allowable buckling loads for Class 1, 2, and 3 component supports subjected to normal, upset, emergency, and faulted load combinations.

110.5 (3.9.3) Provide the basis for selecting the location, required load capacity, and structural and mechanical performance parameters of safety related hydraulic snubbers in order to achieve a high level of operability assurance, including:

(a) A description of the analytical and design methodology utilized to develop the required snubber locations and characteristics.

(b) A discussion of design specification requirements to assure that required structural and mechanical performance characteristics and product quality are achieved.

(c) Procedures, controls to assure correct installation of snubbers and checking the hot and cold settings during plant start-up tests.

(d) Provisions for accessibility for inspection, testing, and repair or replacement of snubbers.

121.0 MATERIALS ENGINEERING - MATERIALS INTEGRITY

121.1 (5.3.1.6) You state that the surveillance program will comply with Appendix H to 10 CFR Part 50. However, any deviations from the requirements of American Society for Testing Materials Specifications (ASTM) E 185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," should be justified.

121.2 (5.3.2) The maximum anticipated change in RT_{NDT} and the resultant pressure-temperature limit curves should be estimated according to Regulatory Guide 1.99, Revision 1 (April 1977).

212.0 REACTOR SYSTEMS

212.1 (3.5.1) Per Reg. Guide 1.70, Revision 2 provide elevation and plan drawings clearly denoting those barriers protecting structures, systems and components listed in Section 3.5.1.2. When identifying a system to be protected and possible missiles which might affect it, reference these elevation and plan drawings to show how the system is protected.

212.2 (3.5.1) Modify Table 3.2-1 to include a listing of those sections of the SAR which describe the saftey-related structures, systems and components inside containment required for safe shutdown.

212.3 (3.5.1) Provide a discussion of the ability of the structures, systems and components described in Table 3.2-1 to withstand the effects of selected internally generated missiles.

212.4 (5.2.2) Under Section 5.2.2.1 on overpressure protection, provide a discussion of and identify the postulated events or transients on which the design requirements are based. Include in your discussion, the assumptions of plant initial conditions and parameters.

212.5 (5.2.2) Acceptance Criterion II.2.b of SRP 5.2.2 states that, "All system and core parameters are at the values within the normal operating range, including uncertainties and technical specification limits, which would result in the highest transient pressure." Insufficient information is presented in the FSAR to determine that this acceptance criterion will be met. The applicant should confirm that the overpressure analysis will be based on an initial operating pressure (up to the Technical Specification limit) which will result in the most limiting peak pressure. The applicant should also confirm that the overpressure analysis will include the effects of the ATWS reactor recirculation pump trip on high reactor pressure.

Acceptance Criterion II.2.c of SRP 5.2.2 states that, "The reactor scram is initiated either by the high pressure signal or by the second signal from the reactor protection system, whichever is later." The applicant has stated that the safety valve sizing analyses can take credit for the first indirect scram, which is the high neutron flux scram. The neutron flux scram occurs before the high pressure scram and results in a lower calculated peak pressure. The applicant should confirm that the safety valve sizing analyses will be based on the SRP acceptance criterion for reactor scram initiation.

212.6 (5.2.2) Provide a reference for your studies on overpressurization which examine the sensitivity of the performance of the system to variations in system and equipment conditions, parameters, and performance.

212.7 (5.2.2) Under Section 5.2.2.4.2 provide a discussion of the number and type of operating cycles for which each component in the overpressure protection system is designed.

212.8 (5.2.5) Per Reg. Guide 1.70, Revision 2, provide the following discussions in Section 5.2.5:

1. Discuss the reliance placed on the proper functioning of systems employed to detect leakage,

2. Describe how signals from various leakage detection systems are correlated to provide information to the plant operators on conditions of quantatitive leakage flow rate,

3. Clearly identify those systems which are not alarmed and which are backups to the alarmed systems,

4. Provide the sensitivity of each detection system. Justify the ability of these systems to achieve such sensitivity given their normal operating environments,

5. Discuss full compliance with Reg. Guide 1.45.

6. Identify fluid systems connected to the primary coolout system and discuss detection and control of intersystem leakage.

212.9 (5.4.7) Under Section 5.4.7.1.3 discuss pressure relief capacity in the RHR system with respect to operator errors during plant startup and shutdown when the RHR system is not isolated from the RCS.

212.10 (5.4.7) Under Section 5.4.7.2.1 provide a complete description of RHR system interlocks.

212.11 (5.4.7) Under Section 5.4.7.2 state the RHR system relief valve capacity, settings, and state the method of collection of fluids discharged through the relief valve.

212.12 (6.3.2) Under Section 6.3.2 provide a description of the significant design parameters (including pressure and temperature with explanation of bases for their selection) and design requirements for ECC delivery lag times for each system.

212.13 (6.3) Provide a failure modes and effect analysis for the ECCS.

212.14 (6.3) Identify all ECCS valves which may be potentially submerged following a LOCA. Discuss the consequences of their spurrious movement or inability to perform as required if submerged.

212.15 (3.2) It is not clear whether portions of the recirculation pump seal cooling water are or are not seismic Category I (Reg. Guide 1.29). The staff requires additional information to show that a complete loss of pump seal cooling water would not lead to unacceptable consequences.

212.16 (4.6) The description or reference to the Standby Liquid Control System should be presented in Section 4.6. Address the requirements of Standard Review Plan (SRP) 4.6.

212.17 (5.2.2) Section 5.2.2.10 which addresses safety/relief valve inspection and testing does not provide an adequate discussion of quality assurance programs to assure that S/R valves will meet specifications. The safety/relief valves must be subject to a Q.A. program which meets Appendix B to 10 CFR 50.

212.18 All P & I Diagrams which have their cross hatches deleted are unacceptable for review. Either provide the cross hatches or provide a suitable method of referencing inter-connections between diagrams such that the staff can distinguish interfaces easily.

212.19 (5.4.6) The acceptance criteria of SRP 5.4.6 (page 5.4.6-3) state that, "As a system which must respond to certain abnormal events, the RCIC system must be designed to seismic Category I standards, as defined in Regulatory Guide 1.29." The condensate storage tank which is the normal suction supply for the RCIC is not seismic Category I. The suppression pool provides a seismic Category I backup source of water, but the switchover requires operator action.

The applicant should confirm that Grand Gulf will conform to the above acceptance criterion. Either of the following alternatives would be acceptable approaches for meeting the acceptance criteria: (1) seismic Category I supply, or (2) safety-grade switchover to a seismic Category I supply, or (3) manual switchover to a seismic Category I supply if appropriately justified. The applicant should discuss the approach to be used for Grand Gulf.

212.20 (5.4.7) The SRP 5.4.7 states the residual heat removal system (RHRS) should meet the requirements of General Design Criterion (GDC) 34 of Appendix A to 10 CFR Part 50. The RHR by itself cannot accomplish the heat removal functions as required by GDC 34. To comply with the single failure criterion the FSAR describes an alternate method of achieving cold shutdown in Section 5.4.7.1.5. Insufficient information is provided to allow an adequate evaluation of this alternate method. In particular, the staff has recently approved Revision 2 to SRP 5.4.7 (containing Branch Technical Position RSB 5-1) which delineates acceptable methods for meeting the single failure criterion. This Branch Technical Position requires testing to demonstrate the expected performance of the alternate method for achieving cold shutdown. The applicant should describe plans to meet this requirement. In addition, we require that all components of the alternate system be safety grade (seismic Category I and IEEE-279). As a result of this requirement, the air supply to the automatic depressurization system (ADS) valves, including the system upstream of the accumulators, must be safety grade. This air supply must be sufficient to account for air consumption necessary for valve operation plus air loss due to system leakage over a prolonged period with loss of off-site power.

212.21 (6.3) The SRP 6.3 does not allow credit for operator action for 20 minutes following a loss-of-coolant accident (LOCA). The FSAR states no operator action is required for at least 10 minutes. The applicant should confirm that no operator action is required until 20 minutes after the LOCA, or provide technical justification and an associated data base to support a time less than 20 minutes. The applicant should identify the manual actions which must be performed to prevent safety criteria from being exceeded following a LOCA over the break spectrum, including single failures. It should also be shown that adequate alarms, instrumentation, and time will be availabe to the operator to perform manual actions necessary to prevent safety criteria from being exceeded.

212.22 (6.3) Review procedure III.20 of SRP 6.3 requires that long-term cooling capacity following a LOCA should be adequate in the event of failure of any single active or passive component of the ECCS. Insufficient information is presented in the FSAR to determine that this requirement will be satisfied with regard to passive failures. The ECCS should retain its capability to cool the core in the event of a passive failure during the long-term recirculation cooling phase following an accident. We will require Grand Gulf to address the following:

Detection and alarms must be provided to alert the operator to passive ECCS failures during long-term cooling which allow sufficient time to identify and isolate the faulted ECCS line. The leak detection system should meet the following requirements:

1. Identification and justification of maximum leak rate should be provided.

2. Maximum allowable time for operator action should be provided and justified.

3. Demonstration should be provided that the leak detection system will be sensitive enough to initiate (by alarm) operator action, permit identification of the faulted line, and isolation of the line prior to the leak creating undesirable consequences such as flooding of redundant equipment. The minimum time following initiation of an alarm before operator action is permitted is 30 minutes.

4. It should be shown that the leak detection system can identify the faulted ECCS train and that the leak is isolable.

5. The leak detection system must meet the following standards:

   a. Control Room Alarm.

   b. IEEE-279, except single failure requirements.

In addition, Grand Gulf should determine the effects on ECCS of passive failures such as pump seals, valve seals, and measurement devices. This analysis should address the potential for ECCS flooding and ECCS inoperability that could result from a depletion of suppression pool water inventory. The analysis should include consideration of (1) the flow paths of the radioactive fluid through floor drains, sump pump discharge piping, and the auxiliary building; (2) the operation of the auxiliary systems that would receive this radioactive fluid; (3) the ability of the leakage detection system to detect the passive failure; and (4) the ability of the operator to isolate the ECCS passive failure, including the case of an ECCS suction valve seal failure.

212.23 (6.3) Review procedure III.5 of SRP 6.3 requires that prior to installation, representative active components used in the ECCS will be proof-tested under environmental conditions and for time periods representative of the most severe operating conditions to which they may be subjected.

Insufficient information is presented in the FSAR to determine that proof testing has been performed for ECCS pumps which must function during the long term following a loss-of-coolant accident. Grand Gulf should demonstrate that the design of the ECCS pumps which must function during the long term following a loss-of-coolant accident have been qualified by representative testing.

212.24 (6.3) Provide analyses to show that diversion of ECCS to containment cooling at or less than 10 minutes after a LOCA will not result in exceeding any safety criteria for the entire break spectrum with consideration of single failure.

212.25 (6.3) Address the inadvertent closure of the reactor recirculation system line suction valve as a single failure in the LOCA analysis, for the break size most affected by this failure.

212.26 (15.0) Provide an analysis of "The Loss of Instrument Air" transient.

212.27 (5.5.7) Provide an evaluation which assesses whether the consequences of a single valve malfunction or operator error could result in possible damage to a heat exchanger of the RHR system while in the steam condensing mode. The evaluation should examine the consequences from two aspects: (1) overpressurization and, (2) hydrodynamic forces. Show that the systems will respond in an acceptable manner.

212.28 (15.0) Modify NSOA drawings to include benefits of non-safety grade equipment which mitigate transients and accidents. Such equipment includes relief valves, rod block monitors, and vessel level (high) trip.

212.29 (15.2) In the analyses for the generator load rejection and turbine trip transients, credit is taken for immediate reactor scram and recirculation pump trip obtained from a valve closure signal (turbine control valve for load rejection and turbine stop valve for turbine trip). Analyze these transients without taking credit for immediate reactor scram and recirculation pump trip. Take credit only for safety-grade, seismic Category I equipment and assume loss of offsite power. What is the effect of the failure of a single safety-grade component?

Present curves similar to those of Figures 15.2-2 and 15.2-4 and give values of maximum vessel pressure and minimum MCPR with the times at which these values occur. Evaluate the percent of fuel rods which reach boiling transition. Since this event is not an anticipated transient, limited fuel failure can be allowed if dose consequences are acceptable.

212.30 (15.0) Identify the limiting transient for each category in Section 15.0.2. For MCPR limiting transients, provide the MCPR versus time plots. Large scale time plots of these parameters presented in Chapter 15.0 should be presented for the limiting transient in each category.

212.31 (15.0) The applicant must provide assurance that the pressure-time plots in Chapter 15 are consistent with the initiation logic for the safety-relief valves. For example, modifications may have been made to the safety/relief system to prevent subsequent reopening of these valves during pressure increase transients to meet containment design base loadings.

212.32 (15.0) Provide assurance that the limiting pump trip is assumed in analyzing decrease in reactor coolant system flow rate transients. The trip initiated from a loss of power may be different than a trip initiated from the recirculation pump trip (RPT) system since the location of the electrical breakers may be different and, thereby, cause different coastdown characteristics.

311.0 ACCIDENT ANALYSIS

311.1 The FSAR implies that the applicant owns all property within the exclusion area, but this should be explicitly stated, if true.

311.2 In Section 3.11.2.2 it was not clear what the sequence would be for the testing of the essential equipment to the more severe conditions associated with the DBA. Please clarify.

311.3 There are no radiological units associated with the dose rate values for the Design Basis Accident column in Table 3.11-2. Please specify.

311.4 The dose rate values for the Design Basis Accident column of Table 3.11-2 appear to be instantaneous value (i.e., t=0). Please state if this is the case. If so, provide a simple figure giving the DBA dose rate as a function of time post-LOCA so that the accuracy of the total integrated dose over 6 months can be verified. If possible identify the major radioactive isotopes which contribute substantially to the dose at the end of 6 months.

311.5 Specify the beta particle radiation dose rate field in the drywell associated with DBA conditions. While it is accepted that the conduit will be of sufficient thickness to stop the poorly penetrating beta particles, describe any qualification testing that has been performed in the postulated high beta dose rate fields associated with DBA's to verify

that there are no adverse effects from; (1) surface heating resulting from the energy deposition of the stopped particles; or (2) induced conductivity or secondary emission and charge transfer which can compromise component operation due to spurious false signal generation. Provide appropriate details and references of such testing in the text.

311.6 The Figure 3.11-1 is not currently referenced in the text of Section 3.11. Please reference and discuss in the appropriate SAR section the purpose of this figure, the appropriate dose rate associated with zones 1, 2 and 3 for both normal and accident conditions and how these dose rates were used in calculating the integrated dose values of Table 3.11-2. Indicate on Figure 3.11-1 (if possible) the approximate location of the reference points identified in Table 3.11-2 for the drywell and containment.

311.7 Figure 3.11-2 is not currently referenced in the text of SAR Section 3.11. Please reference and discuss the purpose of this figure in the text of the appropriate SAR section. Also provide the normalizing value for t=720 hours and the justification for that value in light of the fact that the total integrated doses for the DBA conditions are supposedly calculated using a time period of 6 months (180 days).

311.8 For a continuous containment purge system, such as proposed for the Grand Gulf Mark III containment, analyze the radiological consequences from a postulated LOCA during a purge and include this analyses and results in the appropriate SAR section. Provide in the analysis the assumptions with regard to the size of the purge lines and flow rate through the system, isolation valve closure time, amount of steam release prior to valve closure and any credit taken for removal of fission products prior to release of any radioactivity.

311.9 The text describes Table 3.11-4 as listing integrated dose consequences, although none appear in this table. These data should be added to the table.

311.10 Provide a table listing all safety systems/components necessary to achieve and maintain a safe shutdown and needed to mitigate the radiological consequences of design basis accidents. Include in the table, the method of tornado missile protection provided for each system/component and significant protection parameters (i.e., wall and roof thickness, concrete strength (psi) and curing time on which concrete strength is based etc.).

321.0 EFFLUENT TREATMENT SYSTEMS

321.1 (11.2, 11.3) Provide the following additional source term information:

(a) The inputs to the liquid radwaste system are different from those provided in NUREG-0016. Indicate whether these inputs are based on operating experience or engineering judgement. In particular indicate the basis for the URC waste inputs;

(b) Indicate the liquid waste subsystem in which the URC wastes are processed;

(c) Indicate whether deflectors or diffusers are used for each release point in Table 11.3-10;

(d) Indicate the additional flow rate resulting from the intermittent operation of the drywell purge and the mechanical vacuum pump in Table 11.3-10.

321.2 Provide the following additional information concerning equipment design:

(a) The building housing the gaseous radwaste system charcoal adsorber is designed to the seismic criteria of BTP-ETSB 11-1 (Rev. 1). Indicate whether the charcoal adsorber tank support elements are designed to the seismic criteria of BTP-ETSB 11-1 (Rev. 1);

(b) Indicate whether plastic pipe is used in the gaseous radwaste system;

(c) Indicate whether the quality assurance program for the liquid and solid radwaste system discussed in the note (r) of Table 3.2-1 is the same as that discussed for the gaseous radwaste system in Section 11.3.2.2.1.3. Also indicate whether the inspection and testing provisions for the liquid and solid radwaste system meet the criteria of BTP-ETSB 11-1 (Rev. 1).

321.3 (11.2) In addition to the information provided in Subsection 11.2.2.6 provide a table listing tanks outside reactor containment which contain potentially radioactive liquids. The table should include tanks both inside and outside plant buildings and should not be restricted to radwaste system components. For each tank, indicate the provisions incorporated to monitor tank levels, to annunciate potential overflow conditions, and to collect and process liquids in the event of an overflow. Acceptable provisions are given in Branch Technical Position - ETSB 11-1 (Rev. 1).

321.4 (11.3, 9.4) Provide an analysis with respect to each position in the Branch Technical Position, ETSB No. 11-2, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Reactor Plants," for each atmosphere cleanup system designed to collect airborne radioactive materials during normal plant operation including anticipated operational occurrences. Only the items of noncompliance need to be listed with the justification for noncompliance.

321.5 Indicate whether wet solid wastes are stored and shipped in 50 ft^3 containers or in 55 gallon drums. Provide the storage capacity of the solid waste system for packaged solid waste in terms of the maximum number of drums or containers that can be accomodated at one time, i.e., the maximum quantity of solidified waste in cubic feet that can be stored. Also indicate the basis for the quantity of solid waste generated given in Section 11.4.6.

331.0 RADIOLOGICAL ASSESSMENT

331.1 (12.1.1) Describe the applicable activities performed by the individual(s) in your utility management having responsibility for radiation protection. Describe the individual(s) with specific responsibility for design review to assure that occupational dose will be maintained As Low As is Reasonably Achievable (ALARA) by title and general qualifications.

331.2 (12.1.1) Your description of your compliance with the guidance of Regulatory Guide 8.8, (Revision 2), states that the considerations were not implemented:

1) C.2.e(3), use of bright hydrogen-annealed tubing and piping in primary coolant and feedwater systems,

2) C.2.e(6), provision of laminar flow in the primary system,

3) C.2.i(7), use of canned pumps to reduce leakage,

4) C.2.i(9), use of spare connections on tanks and other components located in higher radiation zones.

Explain why these considerations were not implemented in light of your commitment to maintain doses ALARA.

331.3 (12.1.2) Describe how you have used your dose assessment and the resultant man-rem doses to evaluate the facility design to assure that occupational doses will be ALARA. Also describe how you have factored experience from operating power reactors into your radiation protection design and procedures. Provide examples of improvements you have made in your design and procedures as a result of your use of 1) dose assessment, 2) operational experience, and 3) ALARA design review.

331.4 (12.3.4) (RSP) It is our position that the in-plant accident radiation monitoring systems should provide personnel with the capability to assess the radiation hazard in areas which may be accessed during the course of an accident. The accident monitoring systems may include the normal area radiation monitors, airborne radioactivity monitors, and portable radiation monitoring equipment

(item 331.8 deals with the portable equipment). Emergency power should be provided for installed accident monitoring systems. The accident monitoring systems should have usable ranges which include the maximum calculated accident levels, and they should be designed to operate properly in the environment caused by the accident. Describe your accident monitoring systems, and describe how your systems will meet this position.

331.5 (12.3.4) Provide the frequency for calibration of the area radiation monitors.

331.6 (12.3.4) Describe how your continuous airborne radioactivity systems will provide adequate coverage of general areas, rooms, and corridors which have a possibility of containing airborne radioactivity and which may be occupied by personnel. In order to provide adequate coverage, the systems must be capable of detecting ten MPC_a-hours of airborne particulate and iodine radioactivity.

331.7 (12.4) Your dose assessment requires two additional elements in order to be complete. First, provide sufficient illustrative detail to explain how the radiation dose assessment process was performed. Table 12.4-3 provides adequate summary information for all categories except special maintenance; however, you should provide table(s) showing the activity or job, average dose rate, exposure time, number of workers, frequency, and dose for several jobs to demonstrate the detailed method which is summarized in Table 12.4-3. The details for every activity are not necessary, only several illustrative [illegible]s. Second, provide a breakdown of the activities which are included in the total of 150 man-rems/unit for special maintenance. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-rem Estimates", (attached) which has been published for comment will provide further guidance.

331.8 (12.5.2) Describe what radiation protection equipment, including portable monitoring devices and respiratory protection devices, will be available to personnel responding to an accident. Describe where this equipment will be stored.

331.9
(12.5.3)

Provide the minimum frequency of whole body counting for personnel who enter radiation areas.

361.0 GEOLOGY - SEISMOLOGY

361.1 Page 2.5-22, line 26, states "South of Memphis from the Mississippi-Tennessee border to the buried Ouachita Tectonic belt, the top of Precambrian consists of a relatively smooth, uniformly southwest-dipping surface which is unbroken by major faulting (Figure 2.5-3)." Recent information developed by the Mississippi Geological Survey indicates faulting along the Mississippi River in Tunica County, Mississippi and probable additional faulting to the east of Tunica County.

a) Provide copies of the cross-section number 2 from Tishomingo County to Tunica County, Mississippi by Alvin R. Bicker Jr., 1974.

b) Discuss the significance of the recent information developed by the Mississippi Geological Survey to the Grand Gulf sites.

361.2 Discuss in detail the paper presented at the 3rd International Conference on Basement Tectonics entitled "Geological Age and Significance of Lineament and Aeromagnetic Patterns in the Mississippi Embayment by Dennis W. O'Leary and Thomas G. Hildenbrand, both with the U. S. Geological Survey, Denver Federal Center, Denver, Colorado.

a) What are the implications of this paper to the southern boundary of the New Madrid Seismic Zone as presently defined, and

b) What significance does the information presented in the paper have to the seismic design of the Grand Gulf Nuclear Plant?

361.3 In Figures 2.5-4 and 2.5-58 clearly show the boundaries of the Gulf Coast Basin and Mississippi Embayment tectonic provinces and the New Madrid Fault Zone.

361.4 Explain the relationships of the October 22, 1882, Paris, Texas; January 9, 1891, Rusk, Texas; and December 17, 1931, Northern Mississippi earthquakes to tectonic provinces or geologic structures.

361.5 In the FSAR the New Madrid Fault Zone continues southwestward to near Memphis, Tennessee, and a New Madrid earthquake of Intensity XI-XIII (MM) is postulated to occur 220 miles north of the site (2.5.2.4).

The staff position in previous licensing actions has been that we do not accept the acceleration levels predicted by Nuttli (1973) from New Madrid earthquakes to scale Regulatory Guide 1.60 spectra at large distances as used in the FSAR. Instead, determine the effect of the New Madrid events at the site by scaling appropriate accelerograms of long duration for similar large earthquakes. To determine if the New Madrid event is the controlling event at the site, compute response spectra for these accelerograms and compare them to the Regulatory Guide 1.60 spectrum scaled to 0.15g. For examples of this approach see the PSAR for Clinton (Docket No. 50-461/462), Marble Hill (Docket No. 50-545/547), and Callaway (Docket No. 50-483/486).

361.6 Compare the conservatism in the design spectra for the SSE (Figure 3.7-1) and the OBE (Figure 3.7-2) to that of Regulatory Guide 1.60 by comparing the amplification factors in these response spectra with both the mean and mean-plus-one-standard-deviation of the amplification factors used in developing Regulatory Guide 1.60.

362.0 GEOTECHNICAL ENGINEERING

362.1 (2.5.4.2.1) Show the justification that the Catahoula Formation has an overconsolidation ratio (OCR) in excess of 2 (pages 2.5-54) and, thus, that consolidation settlement does not apply (page 2.5-72). In the justification, identify those consolidation test results which are used, and the OCR and compressibility for each.

362.2 (2.5.4.5.5) State the quantities of borrow material obtained from each of the three sources which were used under and around Category I structures.

362.3 (2.5.4.5.5) Provide a summary of field density and moisture tests obtained for quality control during construction of fill under and around Category I structures. Present the results as a statistical distribution plot or by other convenient method(s). All data sheets need not be provided; however, low, high and average values should be shown so that it can be verified that suitable compaction of structural fill has been attained.

371.0 HYDROLOGIC ENGINEERING

371.1 Provide detailed drawings and plans of the site drainage system. These drawings of the drainage channels, dikes, culverts, stilling basins, etc. should show (in both plan and profile) important features of the system. The data should include sizes, slopes, composition, elevations, cross-sections, detailed drainage-area outlines, erosion protection (where applicable), and the location of stream and drainage channels, For each drainage channel and/or culvert, provide your estimates (and bases, therefore) of time of concentration, rainfall intensity, PMF discharge, and PMF water level.

371.2 Provide the details of the roof drainage system, including drawings of parapet walls, overflow scuppers, and drain pipes.

371.3 Document that the design maximum groundwater level of Elev. 109.00 ft MSL is conservative. Since you have indicated that levels exceeding the design basis have occurred in the past, we cannot conclude that the level constitutes an acceptable design basis. Therefore, document that the level cannot be exceeded by any of the following postulated events:

a) normal or excessive rainfall (e.g., wet season) at the site and recharge areas,

b) rupture of the circulating water system pipeline inside or or outside the turbine building caused by an event such as an earthquake (SSE),

c) rupture of service water system pipe due to a single failure,

d) local PMF at the site (if adequate time is available to establish seepage to the plant),

e) seepage from onsite reservoirs (if applicable),

371.4 State the maximum groundwater levels that safety-related buildings can withstand under (1) static conditions and (2) SSE loading conditions.

371.5 Provide a groundwater contour map of the site area indicating expected groundwater conditions during the operating life of the plant.

372.00 METEOROLOGY

372.01 (2.3.1) Discuss the tornado that passed through the site area on April 17, 1978. Include meteorological information (e.g., peak gusts, wind shifts, pressure drop) and copies of analog traces obtained from all levels of the onsite tower along with estimates of tornado path width, length and intensity.

372.02 (2.3.1) Estimate the probability of a lightning strike and recurrence interval on the plant structures. (See for example, "Electrical Protection Guide for Land-Based Radio Facilities" by D. Bodle, 1971, (JES-159-3-3M 3/76), Joslyn Electronic Systems, or, "Lightning Protection" by J. L. Marshall, 1973, John Wiley & Sons, Inc.)

372.03 (2.3.2) The three years of onsite data (8/72-7/74, 1/76-12/76) at the Grand Gulf site indicate a much lower percentage (.02-.1%) of calm windspeeds (which should be defined as wind speeds less than the starting speed of the anemometer) than would be expected for this region. Discuss the reasons for this low percentage of calm winds.

372.04 (2.3.3) Provide monthly summaries of the meteorological data obtained from the two temporary meteorological towers and compare these summaries with the data from the 162-ft permanent tower. Discuss also the "effects of the hills along the eastern shoreline" (FSAR, p. 2.3-32) through analysis of data from the temporary river tower.

372.05 (2.3.3) Provide a description of the area where the 162-ft meteorological tower is located. Include in the discussion such information as distances to the nearest bluffs and trees, their heights above the base of the tower, and a description of the ground surrounding the tower (i.e., is it grass, soil, etc.).

372.06 (2.3.3) Describe the inspection, maintenance and calibration procedures for the onsite meteorological instrumentation and their frequencies.

372.07 (2.3.3) Discuss the methodology by which calms are determined from the onsite wind data.

372.08 (2.3.4) NRC has developed a new short-term (accident) diffusion model that takes into consideration horizontal plume meander, the direction dependency of wind, and the actual exclusion area boundaries. Enclosed is draft Regulatory Guide 1.XXX "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," (9/23/77) which explains this new model.

The staff considers that this new model will provide a more realistic evaluation of atmospheric diffusion conditions than the approach found in Standard Review Plan Section 2.3.4, and have determined that the new model would be appropriate for use in evaluating the Grand Gulf site. The model was approved for interim use by the Regulatory Requirements Review Committee on May 2, 1978. A copy of this interim branch technical position has been enclosed. In either case, provide the 16 exclusion area boundary distances for the Gran Gulf site.

372.09 (2.3.5) Discuss why terrain recirculation factors were not considered for the long-term diffusion analysis.

ATTACHMENT

INTERIM BRANCH TECHNICAL POSITION
HYDROLOGY-METEOROLOGY BRANCH
ACCIDENT METEOROLOGY MODEL

It is our position that either the draft Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (dated September 28, 1977), or the procedures described in Standard Review Plan Section 2.3.4 may be used to evaluate atmospheric transport conditions for analysis of accidents with the following amendments to the draft regulatory guide model: (a) the accumulated frequency of the limiting sector X/Q value in all sectors may not exceed 5% for the site, and; (b) normalization of individual sector probability distributions is not used.

422.0 CONDUCT OF OPERATIONS

422.1 (13.1.2.1) You indicate in subsection 13.1.2.1 that "The functional positions in Figure 13.1-2 will be filled by the time of initial fuel loading of Unit 1." Please revise Figure 13.1-2 (or provide the additional information by other means) to show the number of persons to be assigned to these functional positions by the time of initial fuel loading of Unit 2.

422.2 (13.1.3) It is not possible to draw a clear comparison between several of your plant staff positions and the functional positions described in ANSI N18.1. Therefore, please provide a list of your plant staff positions vs. the corresponding positions identified in Section 4 of ANSI N18.1 or describe in detail your proposed qualifications for each position.

423.0 INITIAL TEST PROGRAM

423.1 (14.2.2) Expand subsection 14.2.2 to discuss how the plant engineering staff will be utilized in conducting your initial test program.

423.2 (14.2.3) Modify subsection 14.2.3 to describe the review and approval process for preoperational test procedures and startup test procedures. This information should be consistent with the responsibilities of personnel and the Test Working Group stated in subsection 14.2.2 for the preoperational test phase.

423.3 (14.2.4) Expand subsection 14.2.4 to describe your controls provided for plant modifications and repairs. These controls should assure that (1) required repairs or modifications will be made; (2) retesting is done, as necessary, following the modifications or repairs; and (3) any proposed facility modifications will be reviewed by the original design organization or other designated design organizations.

423.4 (14.2.4) Provide a description of your method for changing test procedures when the scope or intent is changed. Note any differences in this method between preoperational and startup tests.

423.5 (14.2.4) Clarify subsection 14.2.4 to verify that the "completion of the required preoperational testing" that is required prior to fuel loading includes review and approval of test results. If portions of any preoperational tests are intended to be conducted, or their results approved,after fuel loading: (1) list each test; (2) state what portions of each test will be delayed until after fuel loading; (3) provide technical justification for delaying these portions; and (4) state when each test will be completed (key to operating modes defined in your technical specifications, or to test conditions defined in Chapter 14). Note that any test that you do not intend to begin prior to fuel loading should be included in your startup test phase instead of the preoperational test phase.

423.6 (14.2.11) In order to facilitate the staff's review of your individual test descriptions, provide an index of preoperational test descriptions.