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

NRC PDR

JAN 17 1979

Docket Nos: 50-416
and 50-417

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

Dear Mr. Stampley:

SUBJECT: FIRST-ROUND REQUESTS 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 first-round requests for additional information. As suggested by our review schedule, a copy of which was forwarded to you by our letter dated December 8, 1978, additional first-round requests are being developed by other review branches. We will forward these additional requests as they become available.

In order to maintain our current review schedule, we request that you amend your Final Safety Analysis Report to reflect your responses to the enclosed requests by May 4, 1979. If you cannot meet this date, please advise us as soon as possible so that we may consider the need to revise our review schedule.

Please contact us if you desire any discussion or clarification of the enclosed requests.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Enclosure:
Requests for Additional
Information

cc:
See next page

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ENCLOSURE

FIRST-ROUND

REQUESTS FOR ADDITIONAL INFORMATION

GRAND GULF NUCLEAR STATION

UNITS 1 AND 2

DOCKET NOS. 50-416 AND 50-417

121.0 MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.1 (5.2.3) Provide a sketch of the Grand Gulf reactor vessels (including dimensions) showing all longitudinal and circumferential welds, and all forgings and/or plates. Welds should be identified by a shop control number (such as a procedure qualification number), the heat of filler metal, type and batch of flux, and the welding process. Each forging and/or plate should be identified by a heat number and material specification.

121.2 (5.2.3) Supply the following information for each of the ferritic materials of the pressure retaining components in the reactor coolant pressure boundary of the Grand Gulf plant:

(1) The unirradiated mechanical properties as required by the testing programs in Section III of the ASME Code and Appendix G of 10 CFR Part 50 (test results to be presented should include Charpy V-notch, dropweight, lateral expansion, tensile, upper shelf energy, T_{NDT} and RT_{NDT}). If any of these properties have not been determined by a test method required by Appendix G of 10 CFR Part 50, state the actual test procedure used and/or the method used to estimate the test result together with a complete technical justification of the procedure used.

(2) Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves at the beginning-of-life.

For each reactor vessel beltline weld, plate or forging, provide the following additional information:

(3) The chemical composition (particularly the Cu, P and S content) and the maximum end-of-life fluence.

(4) The relationship used to predict the shift in the RT_{NDT} and percent decrease in upper shelf energy as a function of neutron fluence.

(5) Identify the material(s) in the reactor coolant pressure boundary that will limit the pressure-temperature operating curves at the end-of-life.

121.3 (5.3) Paragraph II.C.2 of Appendix H, 10 CFR Part 50 states: "Surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region,...." FSAR Section 5.3 indicates that the capsule holder brackets were welded to the reactor pressure vessel inner wall. Present sufficient design and fabrication detail to demonstrate that the capsule attachments were designed and constructed in accordance with accepted standards, such as the ASME Code Section III rules for attachments to vessels.

121.4 (1.6) (5.3) In FSAR Sections 1.6 and 5.3 General Electric Report NEDO-20631, "Mechanical Property Surveillance of Reactor Pressure Vessels for General Electric BWR-6 Plants," dated March 1975, is referenced. At present this report is not in the OL docket file. In order to make an evaluation of compliance with Appendix H, of 10 CFR Part 50, for this plant, the information referenced by this GE report must be submitted for review.

121.5 (Hatch-2) (Reference HNP-2 response to 121.15 and 121.18) The following information is necessary to demonstrate that the Grand Gulf Unit Nos. 1 and 2 feedwater inlet nozzle thermal sleeve/sparger design has been evaluated with due consideration to nozzle cracking due to thermal cycling and that a program of scheduled augmented inservice inspections, with a sensitive method that will assure detection, has been developed:

(1) The technical basis to assure the structural integrity of both the feedwater inlet nozzle and the sparger.

(2) An evaluation of the feasibility of automated ultrasonic testing (UT) fixtures installed on all feedwater inlet nozzles with particular attention on examination of the nozzle bore region.

(3) An evaluation of the feasibility of performing the internal surface examination by magnetic particle methods.

Your response should contain:

a description of the nozzle and sparger design including dimensions, materials of construction and weld locations.

description of analyses and test data, referencing if necessary data previously submitted to the staff where directly appropriate for this plant.

projected crack growth rates, stress levels and usage factors for both the nozzle and the sparger should be described in detail.

any plant modifications that are planned to reduce the feedwater to reactor water temperature differential during low power operation.

any instrumentation that will be installed in the reactor to verify the conclusions of the design analysis should be identified.

Several ultrasonic testing concepts and procedures have been used to examine the feedwater inlet nozzle regions in operating plants. Define the specific ultrasonic testing procedure that will be used on Grand Gulf Unit Nos. 1 and 2. Discuss the influence of local grindouts on crack detection on your ultrasonic testing method.

In addition, provide a description of the augmented inservice inspection (ISI) program to be implemented including scheduled surface examination, ultrasonic testing and verification of the leak tight integrity of the thermal sleeve to safe end joint on all nozzles. The essential elements of an acceptable program are given below:

Augumented Inservice Inspection Plan

(1) Preservice Examination - Preservice UT examination should nozzle inner radius, bore, and safe end regions. In addition, a preservice surface examination should be performed on the accessible regions of all nozzle inner radii.

(2) Inservice Examination - To confirm the continuing structural integrity, the following examinations should be performed:

(a) At each scheduled refueling outage, an external UT examination of all feedwater nozzle inner radii, bore and safe end regions.

(b) After 50 startup/shutdown cycles but prior to 70 cycles a surface examination of the accessible regions of all nozzle inner radii. The definition of startup/shutdown cycles and the procedure for liquid penetrant examination is contained in report NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking."

(c) Subsequent surface examinations of the accessible region of all nozzle inner radii should be performed at the earlier of (i) every other scheduled refueling outage, or (ii) at the scheduled refueling outage after 20 but prior to 40 startup/shutdown cycles after the last surface examination.

(3) Thermal Sleeve to Safe End Joint - An examination method, such as a leak test should be developed to confirm the continuing structural and leak tight integrity of the thermal sleeve to safe end joint.

Acceptance Standards

(1) All UT indications evaluated to be cracks should be verified by appropriate surface examination and removed by local grinding.

(2) All surface indications evaluated to be service induced cracks should be removed by local grinding.

(3) The UT inspection personnel should be required to demonstrate supplemental qualifications by either (i) past sucessful experience in locating and identifying cracks in BWR feedwater inlet nozzles or (ii) performing a qualification test on a full size unclad nozzle mockup.

Recording and Reporting Standards

Requirements for recording of indications and reporting of inspection results are contained in report NUREG-0312.

121.6 Considering the recent BWR service experience of cracking of the vessel nozzle and wall associated with the control rod drive return line, we require a description of any proposed plant modifications (such as changes in material, location of the CRD return line, delection of the CRD return line, etc.) that will preclude such cracking and a complete technical justification for the proposed modifications.

121.7 We will require that your inspection program for Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g) published in the February 12, 1976 issue of the FEDERAL REGISTER.

To evaluate your inspection program, the following minimum information is necessary for our review:

(1) A preservice inspection plan to consist of the applicable ASME Code Edition and the exceptions to the Code requirements.

(2) An inservice inspection plan submitted within six months of anticipated commercial operation.

The preservice inspection plan will be reviewed to support the safety evaluation report finding on compliance with preservice and inservice inspection requirements. The basis for the determination will be compliance with:

(1) The Edition of Section XI of the ASME Code stated in your PSAR or later Editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply.

(2) All augmented examinations established by the Commission when added assurance of structural reliability was deemed necessary. Examples of augmented examination requirements can be found in NRC positions on (a) high energy fluid systems in SRP Section 3.2, (b) turbine disk integrity in SRP Section 10.2.3, and (c) feedwater inlet nozzle inner radii.

Your response should define the applicable Section XI Edition(s) and subsections. If any examination requirements of the Edition of Section XI in your PSAR can not be met, a relief request including complete technical justification to support your conclusion must be provided.

The inservice inspection plan should be submitted for review within six months of anticipated commercial operation to demonstrate compliance with 10 CFR Part 50, Section 50.55a, paragraph (g). This plan will be evaluated in a safety evaluation report supplement. The objective is to incorporate into the inservice inspection program Section XI requirements in effect six months prior to commercial operation and any augmented examination requirements established by the Commission. Your response should define all examination requirements that you determine are not practical within the limitations of design, geometry, and materials of construction of the components.

Attached are detailed guidelines for the preparation and content of the inspection programs and relief requests to be submitted for staff review.

GUIDANCE FOR PREPARING PRESERVICE AND INSERVICE INSPECTION PROGRAMS AND RELIEF REQUEST PURSUANT TO 10 CFR 50.55a(g)

A. Preservice/Inservice Inspection Program Description

This program covers the requirements set forth in 10 CFR 50.55a(g) and the ASME Boiler and Pressure Vessel Code Section XI, Subsections IWA, IWB, IWC and IWD.

The guidance provided in this enclosure is intended to illustrate the type and extent of information that should be provided for NRC review. It also describes the information necessary for "request for relief" of items that cannot be fully inspected to the requirements of ASME Section XI. By utilizing these guidelines, licensees can significantly reduce the need for having to respond to additional information requests from the NRC staff.

B. Contents of the Submittal

The information listed below should be included in the submittal:

1. For each facility, include the applicable ASME B & P V Code date and appropriate addendum date.

2. The period and interval for which this program is applicable.

3. Include the proposed codes and addenda to be used for repairs, modifications, additions or alternations to the facility that might occur during this inspection period.

4. Identify the examinations that you have exempted under the rules of ASME Section XI. A reference to the applicable paragraph of the code that grants the exemption is satisfactory. The inspection requirements for exempt components should be shown; i.e., visual inspection during a pressure test.

5. Identify the inspection and pressure testing requirements of the applicable Section XI requirements that are deemed impractical because of the limitations of design, geometry and material of construction of the components. Provide the information requested in paragraph C for the inspections and pressure tests identified.

C. Request for Relief from Certain Inspection and Testing Requirements

It has been the staff's experience that many requests for relief from testing requirements submitted by licensees have not been supported by adequate descriptive and detailed technical information. This detailed information is necessary to document the impracticality of the ASME Code requirements within the limitations of design, geometry and materials of construction of components and to determine whether the use of alternatives will provide an acceptable level of quality and safety.

Relief requests submitted with a justification such as "impractical," "inaccessible," or any other categorical basis, require additional information to permit the staff to make an evaluation of that relief request. The objective of the guidance set forth below is to illustrate the extent of the information that is required by the NRC staff to make a proper evaluation and to adequately document the basis for granting the relief in the safety evaluation report. The NRC staff believes subsequent requests for additional information and delays in completing the review can be considerably reduced, if this information is provided initially in the licensee's submittal.

For each relief request submitted, the following information should be included:

1. Identification the component(s) and/or the examination requirement for which relief is requested.

2. Number of items associated with the requested relief.

3. ASME Code class.

4. Identification of the specific ASME Code requirement that has been determined to be impractical.

5. Information to support the determination that the requirement is impractical; i.e., state and explain the basis for requesting relief.

6. Identification of the alternative examinations that are proposed in lieu of Section XI requirements or to supplement partially performed Section XI examinations.

7. Description and justification of any changes expected in the overall level of plant safety by performing the proposed alternative examinations in lieu of the ASME Section XI examination. If it is not possible to perform alternate examinations, discuss the impact on the overall level of plant quality and safety.

For inservice inspection provide the following additional information regarding the inspection frequency:

8. State when the relief request would apply during the inspection period or interval; i.e., is the request to defer an examination.

9. State when the proposed alternative examinations will be implemented and performed.

10. State the time period for which the requested relief is needed.

Technical justification or data must be submitted to support the relief request. Opinions without substantiation that a change will not affect the quality level are unsatisfactory. If the relief is requested for inaccessibility, a detailed description or drawing which depicts the inaccessibility must accompany the request. A relief request is not required for tests prescribed in Section XI that do not apply to your facility. A statement of N/A (not applicable) or none will suffice.

D. Request for Relief for Radiation Considerations

Exposures of test personnel to radiation to accomplish the examinations prescribed in ASME Section XI can be an important factor in determining whether or under what condition an examination must be performed. A request for relief must be submitted and approved similar to that required for inaccessibility.

We recognize that some of the radiation considerations will only be known at the time of the test. However, the licensee generally is aware, from experience at operating facilities, of those areas where relief is necessary and should submit as a minimum the following information with the request for relief:

1. Total estimated man-rem exposure involved in the examination.

2. Radiation levels at the test area.

3. Flushing or shielding capabilities which might reduce radiation levels.

4. Alternate inspection techniques proposed.

5. Remote inspections considerations.

6. Redundant systems or similar welds which can be inspected.

7. Preservice and any inservice results of welds involved.

8. Consequences if the weld failed.

220.0 ANALYSIS BRANCH, REACTOR ANALYSIS SECTION

221.1 (4.4.3.3) The discussion on minimum power limit at low core flow states "Therefore, the system is provided with an interlock to trip off the 60 Hz power source and close the 15 Hz power source if the feedwater flow falls below a preset level (typically 23 percent of rated)...". What is the actual setpoint for Grand Gulf?

221.2 (4.4.3.3) What interlocks are available to prevent a pump speed change from 15 Hz source to 60HZ source without closure of the flow control valve(s)?

221.3 (4.4.4.1) The GEXL data base (for the approved correlation) is for 7x7 and 8x8 one water rod bundles. No substantial data base has been provided to support the 8x8, two water rod design. The GEXL correlation must be demonstrated to be applicable to the new 8x8 design, by comparison to applicable data, prior to issuance of an operating license for Grand Gulf. Alternatively, the MCPR limit may be increased by 0.05 to accommodate GEXL uncertainties.

221.4 (4.4.4.5) Page 4.4-16 of the FSAR states that "the nominal expected bypass flow fraction is approximately 10 percent". What is the expected bypass flow fraction and what is its uncertainty?

221.5 (4.4.4.5) What fraction of the fuel bundle flow is "water rod flow"?

221.6 (4.4.4.6) An operational design guide used by GE is that the decay ratios, X_2, X_0, are bounded by:

Channel hydrodynamic performance, $X_2/X_0 \leq .5$

Reactor core performance, $X_2/X_0 \leq .25$

Total system performance, $X_2/X_0 \leq .25$.

The power and flow conditions which analytically satisfy the above limits are referred to as the operational boundary for normal manual or automatic control. Identify this operational boundary on a power-flow map.

221.7 (4.4.4.6) The reactor core decay ratio for natural circulation, 105% rod pattern is give as 0.97. It is unlikely that operation in a region of the power-flow map with such a high decay ratio will be permitted. Discuss the uncertainty in the calculation of the decay ratio and discuss possible means of preventing operation in that region of the power flow map, e.g., adjustment of rod block limits and APRM power-flow scram setpoints to preclude operation in that region of the power-flow map.

221.8 (4.4.4.6) (Page 4.4-23 states that the decay ratio of the plant becomes 0.25 at a point along the 105 percent rated flow control line which corresponds to 50% flow and 70% power. Figure 4.4-6 shows this point to be on the rod block line. Please clarify.

221.9 (4.4.4.6) The decay ratio for channel hydrodynamic performance is also given as 0.97 at the natural circulation, 105% rod pattern point. As with the reactor core decay ratio, this value is unacceptably high. If measures for preventing operation with such high decay ratios differ from those requested in 221.7, please provide a discussion, similar to that requested in 221.7, for prevention of operation with such high hydrodynamic performance decay ratios.

221.10P (4.4.6.1) No substantitive discussion of a loose parts monitoring system is given. A design description and the LPMS manufacturer's sensitivity specifications shall be provided. The LPMS must be operational and capable of recording vibration signals for signature analysis at the time of initial startup testing.

A description of the monitoring equipment including location and basis for alarm settings shall be provided in the FSAR. Anticipated major sources of internal and external noise will be provided along with plans to minimize these sources. A description of precautions taken to insure the operability of the LPMS after operational basis earthquakes should be discussed. A detailed discussion of the operator training program for operation of the LPMS, planned operating procedures, and record keeping procedures should be provided.

We require a minimum of two LPMS sensors at each natural collection region. The LPMS is required to function after any seismic event for which plant shutdown is not required. An exception is that recorders are not required to function within their specified accuracy during or after seismic events without maintenance. However, monitoring (alarm and/or indication) capability must remain available for that channel at all times during and after the seismic event. The system should also be shown to be adequate by analysis and/or test for the normal operating radiation, vibration, temperature, and humidity environment.

221.11 (Table 4.4-1) Table 4.4-1 shows steam flow rate and feedwater flow rate to be equal. Please adjust one of these values to account for control rod drive flow.

221.12 (Figure 4.4-6) The core reactivity stability plot shows only natural circulation points and 105% rod line points. On the same figure, provide plots of decay ratio versus % power for the 100%, 95%, 90%, 80%

70% and 60% rod lines.

221.13 (Figure 4.4-5) Provide plots for 95%, 90%, 80%, 70% and 60% rod line on the power flow map in addition to the lines already shown on the map.

221.14 Provide the relative bundle power histogram for the power distribution used in the statistical analysis and compare with the most adverse power distribution anticipated. Indicate the respective burnups corresponding to the reference relative bundle power distribution.

221.15 Provide the assumptions used for the amount of crud buildup in the design calculations and the sensitivity of CPR and core pressure drop to variations in the amount of crud present. Also, provide data supporting the assumption on crud thickness and discuss how crud buildup in the core would be detected.

311.0 ACCIDENT ANALYSIS

311.11 (6.1.2) Table 6.1-2 lists approximately 5 tons of organic materials within the drywell and 80 tons of organic materials elsewhere in the containment. The bulk of these materials are noted to be "designed to withstand radiation dose" or "contained within an enclosure or vessel". It is a safety concern that these materials may emit hydrogen or simple alkane gases by radiolysis during the course of a postulated LOCA. Please indicate the protection against radiolysis for these materials and the amounts of hydrogen and alkanes likely to be formed by radiolysis.

311.12 (6.4.2) Figure 6.4.6 indicates the shielding provided to maintain control room habitability during postulated accidents. Provide a description of this control room structure boundary (wall, ceiling and floor materials and thickness). Verify that radiation streaming through penetrations will not occur.

311.13 (6.4.4) Describe the self-contained breathing apparatus that will be supplied for an emergency team. On-hand supplies should be sufficient for at least 5 people with a six-hour on-site bottled air supply with unlimited off-site replenishment capability from nearby locations.

311.14 (15.6.5) Insufficient information has been provided to determine if the potential consequences from engineered safety feature (ECCS) component leakage is indeed negligible. Therefore, identify each source of potential leakage and the maximum operational leakage from each source. The maximum operational leakage is defined as the sum of the leakage for all recirculating systems (1) which are detectable during test and (2) above which the technical specifications would require declaring a system out of service.

311.15 (15.7.1) FSAR Section 15.7.1 currently does not indicate that a technical specification limiting the release rate of activity from the SJAE will be incorporated in the pl technical specification. Provide such indication in FSAR Section 15.7.1. Further guidance on determining the technical specification value can be obtained from NUREG-0133 "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants" (1978). If the proposed technical specification for the activity release rate from the SJAE's will exceed the 350,000 μCi/sec at 30 minutes decay assumed in the analysis, calculate the potential consequences assuming that release at the proposed larger tech. spec. rate has occurred for one month before the accident occurs.

311.16 (15.7.4) Provide the supporting rationale in FSAR Section 15.7.6 for the assumption that 130 fuel rods experience cladding damage for a fuel handling accident (FHA) inside containment. In FSAR Section 15.7.4 only 101 fuel rods were calculated to experience cladding damage for a fuel handling accident in the spent fuel pool. Identify all assumptions (i.e., max. potential fall height, etc.) which are different between these two potential accidents and which would directly affect the determination of the number of fuel rods experiencing cladding perforation. Also provide indication in FSAR Sections 15.7.4 and 15.7.6 that appropriate technical specifications will be incorporated into the unit technical specifications which will support the basis for the assumptions used in the fuel handling accident analysis.

311.17 (15.7.4) The staff typically uses a minimum value of 0.25" water gauge negative pressure to permit credit for the ESF charcoal filtration of the released iodine from a fuel handling accident. In FSAR Section 15.7.4 the applicant has not indicated that the fuel handling area will be maintained at this negative pressure level during fuel handling operations. Therefore provide the degree of negative pressure to be maintained in the fuel handling area during fuel transferring operations. Provide the appropriate justification for any negative pressure value which is lower than the current staff value. Also state that the value of negative pressure will be incorporated in the Grand-Gulf plant technical specifications.

311.18 (15.7.4) In the description of the fuel handling area ventilation system in FSAR Section 9.4.2, the fuel handling area is presently described as being maintained at a slight negative pressure during normal operation. Clarify if it is your intention to maintain the fuel handling area at a slight negative pressure even when fuel handling operations are not being performed. State in quantitative terms the definition of "slight negative pressure" (e.g., negative 1/16" water gauge, etc.).

313.0 ACCIDENT ANALYSIS BRANCH (EMERGENCY PLANNING)

313.1 (13.3.1) In order that we may better understand the planning and scoping criteria used in the formulation of your emergency plan, please provide additional definition to the phrase "very low probability events" as used in Section 13.3.1.

313.2 (13.3.1.1, 13.3.5, Fig. 13.3-1) We consider that your definition and use of the term "Emergency Action Levels" should be changed in order to avoid confusion with that generally accepted in the nuclear industry. Your definition encompasses that normally associated with Protective Action Guides. Subsequently the distinction is blurred between the concept of projected dose and observed parameters. We request that you revise Section 13.3.1.1 of your plan to include the definitions for Protective Action Guides and Emergency Action Levels as set forth in Section 1 of Annex A to Regulatory Guide 1.101. Also revise the applicable portions of your plan for consistency with these definitions. For example, the statement in Section 13.3.5 beginning "Specific Emergency Action Levels, including..." should be revised, substituting the term "Protective Action Guides" for "Emergency Action Levels". Likewise the title of Figure 13.3-1 should be changed. Note that your plan should also specify the observed parameter values (Emergency Action Levels) and/or other observable criteria which correspond to Protective Action Guide values. See question 313.3 for the required information.

313.3 (13.3.3.1) Amend Section 13.3.3.1 to include the specific criteria to be used for declaring a plant emergency, site emergency, and general emergency. Your plan should identify the Emergency Action Levels, including specific parameter values, as well as other criteria such as alarm annunciators, system status indicators, etc., which provide the basis for declaring each emergency category. Note in the case of a "general emergency" that the information should be readily available in the control room and the activation criteria based on predetermined conditions of the plant which would likely lead to serious releases of radioactive fission products into the atmosphere. Guidance with respect to the criteria to be used is discussed in Sections 4.1.3, 4.1.4, and 4.1.5 of Annex A to Regulatory Guide 1.101.

313.4 (13.3.3.2.2) The last paragraph in Section 13.3.3.2.2 can be interpreted such that, in the event of a serious accident (general emergency), there is a residual question regarding the methodology to be employed in the time period leading up to initiating a recommendation that protective measures be taken offsite. In particular we are concerned with the words "detection and evaluation of accidental releases which are classed as ... a General Emergency will normally be confirmed... by field methods". The use of field survey results for estimating offsite dose consequences and determining whether protective actions should be recommended for offsite populations is acceptable for lesser accident categories but would not be acceptable for a "General Emergency". Accident consequences which would produce offsite doses in excess of the upper limit of EPA's Protective Action Guides would almost certainly involve some degree of core melting. Therefore, under such conditions, we consider that the declaration of this emergency class be based on information readily available in the control room, and the criteria for recommending offsite protective measures be based on predetermined plant conditions which could likely lead to serious releases of radioactive fission products into the atmosphere. Rapid evaluation (on the order of 15 minutes) of plant parameters, containment and possibly effluent radiation levels, and meteorological conditions should dictate the early response actions regarding recommendations for early warning of the public and prompt initiation of protective actions within the LPZ. Field surveys, which require some time to conduct and evaluate, may be used as they become available to modify or expand the protective actions. Discuss the consistency of your intended methodology with respect to the above. If appropriate, revise your plan accordingly.

313.5 (13.3.4.4) In accordance with Section 5.4 of Annex A to Regulatory Guide 1.101, your plan should address the coordination with the appropriate Louisiana State and local agencies having jurisdiction over that area within a four mile radius of the proposed plant. In the event there are no residents within that area of Tensas Parish, we would expect the coordinated effort to be focused on the control of transients which may utilize the land and water areas for recreational purposes. The nature and extent of information required from the appropriate State and local government agencies is detailed in question 313.6

313.6 (13.3.4.4, App. 13.3A) Your plan indicates that the letters of agreement and understanding, and copies of State and local response plans will not be provided until approximately 90 days prior to fuel loading. In order for us to complete our review of your coordination with participating government agencies, such information should be submitted well in advance of the scheduled Q2 date for this case. However, in lieu of submitting State and local agencies' radiological response plans as evidence of reasonable assurance that appropriate and timely response measures can and will be taken in behalf of the population-at-risk in the plant environs, you may address the applicable elements of the list below for <u>each</u> State and local agency having a response role in support of the Grand Gulf Emergency Plan. If you choose to submit any State or local plans in lieu of the above, ensure that the plans are reviewed for completeness with respect to the applicable elements listed below. If necessary request the State and/or local agencies to include such information in their plans, or you may supplement their plans with the necessary information in Section 13.3.4.4.1 of your plan. Note that in the absence of a State or local agency plan, the written agreement with that agency should reflect their concurrence with your docketed description of the applicable elements from the following list as related to that agency's role in support of the emergency response plans developed for the Grand Gulf facility.

1. The identity of the agency.

2. A description of the authority and responsibility for emergency response functions.

3. A description of the concept of operations including the operational interrelationships of all organizations having emergency response roles.

4. The designation and location of the Emergency Operations Center for the direction and/or coordination of emergency support activities.

5. The established relationship and interface with State and/or local government emergency response plans.

6. The provisions established with the Department of Energy Regional Coordinating Office for radiological assistance under the RAP and IRAP programs.

7. A description of the communication plan for emergencies including titles and alternates for both ends of the communication links, and primary and backup means of communication. Where consistent with the agency function, include the following:

   a. Provision for 24-hour/day manning of communication link.

   b. Provision for administrative control methods for ensuring the effective coordination and control of the emergency support activities.

   c. Provision for communications arrangements with contiguous local governments where applicable.

   d. Provision for communications arrangements with Federal emergency response organizations.

   e. Provision for communications with the nuclear facility, State and/or local emergency operations centers, and field assessment teams.

8. A description of the communications methods for issuing emergency instructions to the public in the potentially affected environs of the nuclear facility.

9. A description of the methods and equipment to be employed in determining the magnitude and locations of any radiological hazards following liquid or gaseous radioactivity releases.

10. The designation of protective action guides and/or other criteria to be used for implementing specific protective actions and the information needs (e.g., dose rates, projected dose levels, contamination levels, airborne or waterborne activity levels) for implementing such actions.

11. A description of the methods for protecting the public from consumption of contaminated foodstuffs.

12. A description of the evacuation plans for the Low Population Zone (LPZ) including survey maps for the facility environs showing evacuation routes as well as relocation and shelter areas. The plans may extend to areas beyond the LPZ and should include the following:

    a. Population and their distribution around the nuclear facility.

    b. Means for notification of the potentially affected population.

    c. Disabilities, institutional confinement, or other factors which may impair mobility of parts of the population.

    d. Means of effecting relocation.

    e. Potential egress routes and their traffic capacities.

    f. Potential impediments to use of egress routes.

13. The provisions for maintaining dose records of all potentially exposed emergency workers involved in response activities.

14. The provisions for emergency drills and exercises to test and evaluate the response role of the agency, including provisions for critique by qualified observers.

15. A description of the training program established for those individuals having an emergency response assignment.

16. The provisions for periodic review and updating of the emergency response plans of the agency.

313.7
(13.3.6.2,
Table 13.3-2)

The staff considers, in the event of a serious accident which would be categorized as a General Emergency, that your plan contain provisions for assured and direct notification from the site to the local authorities responsible for implementing protective actions in both the land and water areas within the LPZ. Identify the local authorities having jurisdiction over these areas, and identify the primary and backup communications systems that will be immediately available onsite for contacting these organizations.

Note that although Section 9.5.2 of the FSAR indicates that two-way radio communication is available between the control room and offsite, this link is not identified in Table 13.3-1 or 13.3-2.

313.8 (13.3.5.4) In accordance with Section 6.4 of Annex A to Regulatory Guide 1.101, amend your plan to include the provisions to make available on request to occupants in the LPZ, information concerning how the emergency plans provide for notification to them and how they can expect to be advised what to do.

313.9 (13.3.6, 13.3.7) Your plan appears silent with respect to the recommendations contained in Sections 7.6 and 8.1.2 of Annex A to Regulatory Guide 1.101 regarding damage control equipment, supplies, training, and drills. Discuss your position with respect to those recommendations set forth in the guide.

313.10 (13.3.7.1.2) In accordance with the position set forth in Section 8.1.2 of Annex A to Regulatory Guide 1.101, amend your plan to include provisions for:

(a) An initial exercise prior to fuel loading using scenarios appropriate to the Site Emergency or General Emergency classifications.

(b) An annual fire drill incorporating participation by an offsite fire department.

(c) Mandatory tests of the communication links and notification procedures with offsite agencies as part of the annual coordinated drill.

313.11 (13.3.7.2) In accordance with the position set forth in Section 8.2 of Annex A to Regulatory Guide 1.101, amend your plan to include provisions for:

(a) Maintaining all coordinate elements of the total emergency organization informed of the plan and revisions to the plan or relevant procedures.

(b) Reviewing and updating all written agreements at least every two years.

313.12 (Figure 13.3.3-2) In accordance with Section 10-2 of Annex A to Regulatory Guide 1.101, and Sections 13.3-1.a, b, and c of Regulatory Guide 1.70, amend Figure 13.3B-2 to include the thyroid dose curves for 5 rem and 25 rem. Note that in the event these curves intersect the LPZ boundary before reaching a point corresponding to an ordinate value of 8 hours, the curves should be extended to that ordinate value. This was not done for the 1 rem and 5 rem whole body dose curves in Figure 13.3B-1, which should be corrected also.

313.13 (13.3) In accordance with Section 10-3 of Annex A to Regulatory Guide 1.101 and Sections 13.3-6.a and b of Regulatory Guide 1.70, provide the information requested. In particular, ensure that the road network information is complete and that the transient population is presented in the specified format.

313.14 (2.1.1.3, 13.3) Section 2.1.1.3 of the FSAR states that "MP&L will allow access to the restricted area for recreational purposes and will also allow use of the county roads in the restricted area. The Emergency Plan provides for control of the roads in the event of emergency conditions at the plant."

(a) Describe how you plan to control access to these areas and how you plan to monitor the activities of these individuals.

(b) Identify the nature of the recreational activities anticipated.

(c) Estimate the daily maximum number of persons expected to engage in such recreational activities.

(d) Under emergency conditions, describe the means for notification and removal of persons from the restricted area, and the organization(s) responsible for same.

(e) Describe the provisions which you intend to include in your Emergency Plan for the control of the roads referred to in the above FSAR statement.

(f) Resolve the apparent discrepancy between the "restricted area" definitions given in Sections 2.1.1.3 and 13.3.1.1 of the FSAR.

313.15 (2.1.3.4, 13.3) Section 2.1.3.4 of the FSAR states "Table 2.1-9 lists the facilities and institutions within five miles of the LPZ which may require special consideration in evaluating emergency plans". Discuss what special provisions, if any, have been or will be made in your plan or in the State and local plans developed in support of the Grand Gulf facility for each of these facilities and institutions. If none, discuss the reasoning leading to such a determination.

313.16 (13.3) Because of the proximity of the Grand Gulf Military Park to your facility and the relatively large number of visitors particularly on a typical summer weekend, it seems reasonable that some preplanning effort is prudent with respect to handling these transients in the unlikely event of a serious accident. Discuss the provisions that have been or will be made in the appropriate emergency plans for protecting the public health and safety in this area contiguous to your site property.

320.0 EFFLUENT TREATMENT SYSTEMS

320.6 (11.2) You indicate in your response to Question 320.1 that the URC waste flow is approximately 1800 gal/day and that it is sent to the spent resin tank for processing. NUREG-0016 indicates that the expected average daily input to the radwaste system from the ultrasonic resin cleaner is approximately 15,000 gal/day. Indicate how your radwaste system will handle wastes of this magnitude and what method of processing in the liquid radwaste system will be used to handle these wastes.

320.7 (11.2, 11.3, 11.4) Provide the following additional information concerning the radwaste systems capabilities to meet the requirements of Regulatory Guide 1.143 (formerly Branch Technical Position ETSB 11-1, Revision 1).

a) Correct Table 3.2-1, note (q).2 to read "Screwed connections backed-up by seal welding or mechanical joints are permitted only on lines greater than 3/4 inch nominal pipe size and under 2-1/2 inches";

b) Table 11.2-15 provides a listing of tanks outside containment which contain potentially radioactive fluid.

(1) Indicate why certain of the tanks do not have tank level monitoring and high level annunciation;

(2) Indicate why certain of the tanks do not have high level annunciation;

(3) Indicate disposition of overflow from fuel pool drain tank and laundry waste monitoring tank;

(4) Indicate whether the outdoor tanks have a dike or retention pond capable of preventing run-off in the event of a tank overflow, have provisions for sampling collected liquids and provision for processing these wastes in the liquid radwaste systems.

320.8 Provide the following additional information concerning Table 9.4-14 with regard to your exceptions to Regulatory Guide 1.140 (formerly Branch Technical Position ETSB 11-2):

a) Indicate how you will be able to generate sufficient uniform quantities of DOP for leak testing on HEPA filter banks larger than 30,000 cfm;

b) Indicate how much access space you provide around components in support of your exception to Regulatory Position B.4.c;

c) You indicate that you take exception to a number of regulatory positions as not applicable because you do not have charcoal filters on the radwaste building vent as indicated in your exception to B.2.d. However, a number of those positions also discuss criteria for HEPA filters. Indicate whether you are also taking exception to those portions of the regulatory positions.

320.9 (11.3) For the gaseous radwaste systems:

a) Indicate whether the hydrogen analyzers annunciate locally, as well as in the control room, and are set to alarm at 4% hydrogen;

b) Indicate whether all hydrogen analyzers are nonsparking and whether these analyzers are able to withstand a hydrogen explosion.

320.10 (11.4) Provide the following additional information concerning your solid radwaste system (SRS):

a) The procesing capacity of the SRS, i.e., the number of drums per day which the system is capable of processing, filling, and storing, and compare with the expected total waste input to the system and the bases for the values used;

b) Indicate the amount of storage area for dry wastes and contaminated equipment;

c) Indicate the disposition of flush water which does not go into the shipping container, i.e., indicate whether this water is routed to the liquid radwaste system;

d) Provide more detail concerning the process control program including the following:

(1) Process Control Parameters

(a) Depending on the type of waste (bead resins, powdered resins, diatomaceous earth, evaporator bottoms, sodium sulfate solutions, etc.) to be solidified, indicate the process parameters which will be established to provide boundary conditions within which it is reasonably assumed that complete solidification will occur. These process parameters should include, but not be limited to, for each type of influent waste: pH, water content, temperature, effect of various contaminants such as oil content, detergent content, lab chemicals and non-depleted ion exchange resins, ratio of cement to influent waste, type of cement to be used in each case, and the ratio of cement to sodium hydroxide additive;

(b) The basis upon which the process control parameters are chosen should be given. The process control program should be based on tests performed with simulated waste forumlations based on the expected inputs and the parameters listed in (a) above;

(c) The requirements for sampling of the waste input to the solid waste system as it relates to the process control program prior to processing to assure a satisfactory solidified product. Where will the waste be sampled? Discuss how the results of the sampling will be analyzed and used as operational considerations in the process control program;

(2) System Performance

(a) Identify equipment (interlocks, alarms, monitors, etc.) which are required to be functional before processing can begin;

(b) Identify administrative controls, instrumentation, or equipment features to assure that operating procedures will be followed;

(c) Identify the processing steps to assure that the solid waste system is being operated within the boundary conditions established by the process parameters;

(d) Indicate that the plant operator will provide assurance that the process is run within the process parameters boundary conditions. Appropriate records should be maintained for individual batches showing conformance with the established parameters.

320.11 (11.5) Provide the following additional information concerning the process and effluent radiological monitoring system.

a) In addition to the information provided in Subsection 11.5.2, you should provide a table showing that the gaseous and liquid process streams and effluent release points are monitored and sampled according to Tables 1A and 1B of SRP 11.5, Rev. 1;

b) Describe the provisions made for reducing plateout in sample line of process stream samples, including liquid sample lines;

c) Describe what provisions are made to replace or decontaminate monitors without opening the process system or losing the capability to isolate the effluent stream;

d) Indicate whether isolation valves with automatic control features fail in the closed position;

e) Indicate what provisions for postulated accidents are made in the process and effluent radiological monitoring and sampling system. Acceptable provisions include supplemental monitoring equipment on the normal gaseous effluent paths capable of monitoring postulated accident releases in accordance with ANSI Draft Standard N13/42 WG6;

f) For liquid and gaseous effluents that cannot be practicably sampled batchwise; indicate whether you use one of the sampling methods listed in SRP 11.5.II.1.b, Rev. 1.

320.12 (5.4.8) For the reactor water cleanup system, indicate what provisions were made for monitoring resin transfers in the Reactor Water Cleanup System (RWCS). Also indicate whether the resin transfer lines of the RWCS were designed to avoid resins collecting in valves, low points, or stagnant areas.

320.13 (6.5.1) For the engineered safety feature system, according to Regulatory Guide 1.52, Table 2, Test Nos. 5a, b, and d, you should test the carbon at the relative humidity indicated in the regulatory position to demonstrate the quality of the carbon and you should test the carbon to the criteria of the Savannah River Laboratory test.

320.14 (9.3.2) For the process sampling system, state what provisions were made to assure that isolation valves leading to the Process Sampling System fail in the closed position.

320.15 (10.4.2) For the main condenser evacuation system (MCES):

a) Indicate whether the MCES is designed to withstand the effects of an explosion;

b) Indicate what design provisions in the MCES are made to stop continuous leakage paths after an explosion.

320.16 (11.2) You indicate that emergency laundry facilities have been provided in the control building and the wastes from those facilities will be released through the sanitary waste treatment system. Provide the following information concerning this system:

a) P&ID's for the system, as well as interface information between this system and the liquid and gaseous or plant ventilation system;

b) An estimate of the volume (in gal/yr) of liquid waste released by this system and the activity (in curies/yr) released by this system:

c) A discussion of the capacity of the sanitary waste system to handle this input. Also discuss the provisions taken to prevent radioactive contamination of non-radioactive portions of the sanitary waste system;

d) Indicate whether the design of the emergency laundry system meets the requirements of Regulatory Guide 1.143 (formerly Branch Technical Position ETSB 11-1, Revision 1);

e) Indicate your provisions for monitoring the effluents from the system.

331.0
12.0 RADIOLOGICAL ASSESSMENT

331.10 (12.1.1.3) Describe your criteria for deciding when exposure time required to complete tasks and the estimated doses anticipated from the exposure shall be entered in the RWP as recommended by revision 3 to Regulatory Guide 8.8 (June 1978), Paragraph C.3a(8)(e).

331.11 (12.2.2) Describe the airborne radioactive material sources resulting from reactor vessel removal, relief valve venting, and movement of spent fuel. Include a tabulation of the calculated concentrations of airborne radioactive material by nuclides expected during normal operation and anticipated operational occurrences for equipment cubicles, corridors, and operating areas normally occupied by operating personnel.

331.12 (12.2.2.1) Provide estimated man-hours of occupancy and estimated inhalation exposures to personnel for the following areas: 1) refueling water storage tank (RWST) and the condensate storage tank (CST) on page 12.2-8, 2) pump rooms on page 12.2-10. Include the calculations that were based on your assumption that five percent of the releases from the radwaste building ventilation system are from the pump room ventilation system. Also justify your conclusion that no noble gas effluents are released in the pump room ventilation system during normal and anticipated operational occurrences.

331.13 (12.3.2.2.2) a) Describe types of structural barriers used to prevent inadvertent access to the high radiation levels near the fuel transfer tube, and shielding provided to assure acceptable radiation levels in adjacent occupied areas (page 12.3-13). b) Provide a plan and elevation layout drawing of the areas through which the spent fuel transfer tube passes. c) Discuss your procedures for positive access control and radiation monitoring to the areas where the spent fuel transfer tube may be exposed.

331.14 (12.1.1) Describe the radiation protection aspects of decommissioning that you have included in your design to insure that occupational dose will be "ALARA."

331.15 (12.1.1) Describe the features that you have incorporated into your design to maintain occupational radiation exposure ALARA by minimizing and controlling the buildup, transport and deposition of activated corrosion products in reactor coolant and auxiliary systems. Include information on the following steps taken to minimize Co-58 and CO-60, including:

(1) The use of reduced nickel content in systems in contact with reactor coolant.

(2) The low cobalt impurity specification in systems in contact with reactor coolant.

(3) The minimization of high cobalt, hard facing wear materials in the systems in contact with reactor coolant.

(4) The use of high flow rate/hign temperature filtrations for systems in contact with reactor coolant.

(5) The selection of valves and packings materials to minimize crud buildup and maintenance.

(6) Provisions for decontamination of components and systems contaminated with activated corrosion products.

(7) The types of cleanup systems for removal of crud from primary coolant during operation.

331.16 (331.3) Your response to item 331.3, part b is incomplete. In the FSAR, include any changes or additions in the operational considerations resulting from these studies.

361.0 GEOSCIENCES BRANCH - SEISMOLOGY

361.7 In Section 2.5.2.4 you state the peak acceleration at the site due to a New Madrid event is slightly less than 0.02g. Discuss the method and input parameters used to derive this value.

361.8 For the strong motion records described in section 2.5.2.6, state (1) the magnitudes of these earthquakes, (2) the distances from the recording sites to the sources, and (3) the duration of the motions.

361.9 On Figures 2.5-59a and 3.7-68 show the frequency range of primary response modes for Seismic Category I structures, systems, and components. Primary response modes are defined as a subgroup of modes in a dynamic analysis model whose sum of modal responses either equals or exceeds 80% of the total response.

362.0 GEOTECHNICAL ENGINEERING

362.4 (2.5.4.6) The surface of the sand backfill will be covered with a clay seal layer. Provide details of the clay seal layer, including material properties, layer thickness, method of construction and acceptance criteria.

362.5 (2.5.4.1.31) Provide graphical data which show settlements recorded to date versus time. Compare and discuss the results of measured with predicted values. What frequency of monitoring of settlements is planned and what measured values of total and differential settlement would be of concern and require notification of NRC?

422.0 Conduct of Operations

422.3 (13.1.1.1) Provide the number of professional persons reporting to both the Engineering Section Leader and the Licensing Administration Section Leader of the Grand Gulf Project Engineering Organization.

422.4 (13.1.1.1) Identify the organizational group that will provide technical support for the operation of the Grand Gulf facility in the areas of nuclear, mechanical, structural, electrical, thermal-hydraulic, instrumentation and controls engineering, and health physics. In addition, provide resumes of these individuals employed to fulfill the above identified functions.

422.5 (13.1.1.3) Provide the resume of the person filling the position of Licensing Administration Section Leader.

422.6 (13.1.2.2) In Section 13.1.2.2 you state and show in Figures 13.1-3 and 13.1-4 that the Operations Supervisor and others report to the Assistant Superintendent. Figure 13.1-2 shows them reporting to the Operations and Maintenance Supervisor. Please correct this inconsistency.

422.7 (13.1.2.2) Describe the functions and responsibilities of the Lead Maintenance Engineer and his staff.

422.8 (13.1.3) In your response to question 422.2, you listed some Grand Gulf positions as meeting more than one ANSI N18.1-1971 classification, e.g., the positions of Operations Supervisor, Maintenance Supervisor, Technical Supervisor, Senior Reactor Engineer, Instrument Supervisor, and Radiation Protection Supervisor. Please clarify this apparent inconsistency.

422.9 (RSP) (13.1.3) Your plant staff organization does not contain a line position under the Plant Superintendent which has overall responsibility for operations, maintenance and technical support. Therefore, for the purpose of reducing the qualification requirements of the Plant Superintendent, we do not consider that you have a principal alternate for the Plant Superintendent. It is the staff's position that your Plant Superintendent should acquire the experience and training normally required for examination by the NRC for a Senior Reactor Operators License whether or not the examination is taken.

422.10 (RSP) (13.1.3)	Your response to question 422.2 indicates that the qualification requirements for the positions of Technicians and Mechanics correspond to the position of Technician as described in Section 4.5.2 of ANSI N18.1-1971. It is the staff's position that your Electricians and Mechanics should meet the qualification requirements described in Section 4.5.3 of ANSI N18.1 for the position of Repairman. Amend your response to question 422.2 to address this position.
422.11 (RSP) (13.1.3)	Your response to question 422.2 indicates that the Quality Supervisor will meet the qualification requirements corresponding to the position of Supervisor, not requiring NRC licenses as described in Section 4.3.2 of ANSI N18.1-1971. It is the staff's position that the qualification requirements for the position of Quality Supervisor should be, at the time of initial core loading or assignment to the active position, six years experience in the field of quality assurance, preferably at an operating nuclear plant, or operations supervisory experience. At least one year of this six years experience shall be nuclear power plant experience in the overall implementation of the quality assurance program. (This experience shall be obtained within the quality assurance organization.) A minimum of one year of this six years experience shall be related technical or academic training. A maximum of four years of this six years experience may be fulfilled by related technical or academic training. (Note Section 4.4.5 - Quality Assurance of ANSI N18.1-1976.) Amend your response to question 422.2 to address this position.
422.12 (13.4.1)	Identify the position title of each member of your Safety Review Committee or describe the qualification requirements for assignment to the Safety Review Committee.
422.13 (13.4.2)	Identify the position title of each member of the Plant Safety Review Committee and include the specific provisions for limiting alternates.

422.14 (13.4.2) Delineate the specific review responsibilities of the Plant Safety Committee. (Note Section 6.5 of the NRC Standard Technical Specifications.)

422.15 (13.4.2) Describe the authority of the Plant Safety Review Committee.