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

FEB 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.

We note that in your response to Question 021.01, you plan to provide us with information concerning suppression pool dynamic loads in January 1980. We further note that in response to Question 021.03, you plan to provide us with information concerning subcompartment pressure loadings during the first quarter of 1980. These dates are inconsistent with our current review schedule and are expected to have a significant impact on our ability to complete our review of these matters in a timely manner. Therefore, we encourage you to reassess these dates and advise us of any changes thereto as soon as possible so that we may consider the need to revise our review schedule.

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Please note that Question 110.29 is an updated version of Acceptance Review Question 110.1. Therefore, a response to Acceptance Review Question 110.1 is no longer required. You are requested to respond to Question 110.29 instead.

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 w/enclosure:
See page 3

Mr. N. L. Stampley

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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

021.0 CONTAINMENT SYSTEMS

021.1 (6.2.1) Recent efforts in the review of Mark II containments have provided new methodologies for calculating the response of structures to pool dynamic loads. These advances are detailed in NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." In our view, some of the methods presented in NUREG-0487 are applicable to plants with Mark III containments. In this regard, provide the following information:

1. Provide the analytical methods and experimental results used to determine the drag loads on gratings above the suppression pool. Compare these loads with loads calculated using the methods outlined in NUREG-0487. Show that the present design of the gratings is conservative when compared to the methodology presented in NUREG-0487.

2. Provide the analytical methods and experimental results used to determine the loads on submerged structures in the suppression pool. Compare these loads with loads generated using the methodology presented in NUREG-0487. Show that the present design can accommodate the loads calculated using the methodology presented in NUREG-0487.

021.2 (6.2.1) As a result of additional test and further analytical studies, changes have been made in the pool dynamic design loads detailed in the GE report NEDO-11314-08. These changes are reflected in GE report 22A4365, "Interim Containment Loads Report - Mark III Containment," Rev. 2, which is currently under review by the staff. Evaluate the pool dynamic design loads for the Grand Gulf containment design, in particular the condensation loads, in light of the test information obtained since the Construction Permit stage of our review. Provide a detailed comparison of the design loads for Grand Gulf with the revised and refined loads contained in the GE report 22A4365. Show that the present design of the containment and structures for the Grand Gulf Station is conservative with respect to the new loads in GE report 22A4365.

021.3 (6.2.1) Provide detailed plan and section drawings of the TIP station, equipment hatch, personnel hatch, and any other structure located within 20 feet of the suppression pool surface whose width is greater than 20 inches. Show on these drawings your plan to extend these structures into the suppression pool, thus eliminating impact loads due to pool swell.

021.4 (6.2.1) As a result of our review of the Grand Gulf FSAR we find that the Grand Gulf applicant has not committed to adoption of the acceptance criteria regarding bubble phasing for multiple SRV actuation. Specifically, we require the applicant to consider, as part of his analyses to determine the maximum loads on containment structures and equipment, that all bubbles from each SRV line enter the suppression pool simultaneously

and oscillate inphase. Provide the commitment to this criterion or a justification for assuming the bubbles to be out-of-phase.

021.5 (6.2.1) Provide a detailed description of the in-plant test program for the SRV quencher device including the following:

a. Description of the test matrix which should include single and multiple valves tests, first and subsequent SRV actuation and leaking valve tests. The number of tests for each test series should be sufficient for determining the repeatability of the test.

b. Instrumentation set-up with schematic diagrams showing the locations of sensors.

021.6 (6.2.1) Appendix 6A of the Grand Gulf FSAR specifies that only a single SRV could undergo subsequent actuation. However, GE notified the Commission on October 6, 1977, pursuant to the requirement of 10 CFR Part 21, that new analyses showed more than one SRV could undergo subsequent actuations following certain plant transients. In late 1977, GE submitted a low-low setpoint SRV control logic by which the current criterion for allowing a single SRV actuating subsequently can be maintained. This logic is currently being reviewed. In view of the above, provide justification for your position.

021.7 (6.2.1) With respect to the suppression pool temperature limit for SRV operation, our criterion specifies 200°F local temperature and 185°F bulk temperature. To demonstrate the adequacy of the Grand Gulf design, provide the following additional information:

A. In-plant test data or a proposed test plan showing the difference between the local and bulk temperature. The local temperature is referred to as the average pool temperature in the vicinity of the SRV discharge while the bulk temperature is referred to as the calculated average pool temperature by considering the entire pool as a uniform heat sink.

B. Based on the Grand Gulf plant parameters, provide figures showing reactor pressure, mass flux through the quencher and suppression pool temperature versus time for the following events:

(a) Stuck-open SRV during power operation, assuming reactor scram at 10 minutes after pool temperature reaches 110°F and all RHR systems are operable.

(b) Same as event (a) above except that only one RHR train is available.

(c) Stuck-open SRV during hot standby condition assuming 120°F pool temperature initially and only one RHR train is available.

(d) Automatic Depressurization System (ADS) activated following a small line break assuming an initial pool temperature of 120°F and only one RHR train is available.

(e) Primary system is isolated and depressurized at a rate of 100°F per hour with an initial pool temperature of 120°F and only one RHR train is available.

021.8 (6.2.1) Appendix 6A of the Grand Gulf FSAR states that the load specifications described in the GE Topical Report NEDO-11314-08 are needed for the Grand Gulf design. However, it should be noted that Section 8.3.1 of NEDO-11314-08 dealing with load combinations for the main steam line break is not acceptable. As a minimum, we will require consideration of a single, spuriously actuated SRV concurrent with the pool swell loads associated with a DBA. (This position is also reflected in Question 110.34 where SRV_x should be assumed to be one SRV). We will also require analysis to show that SRV's cannot mechanistically actuate concurrent with a DBA. This analysis should include consideration of the uncertainties associated with the calculated pressure response and the uncertainty associated with the pressure required for SRV actuation (e.g., set point drift). Should the confirmatory reactor system response analyses indicate that there is a potential for mechanistic actuation of the SRV's following a DBA, we will reassess our premise that only one SRV will actuate following a DBA.

021.9 (6.2.1) In Section 6.2.1.1.3.3.1.1 of the FSAR, it is stated that reactor feedwater is assumed to stop instantaneously at time zero for a recirculation line break. Provide a scoping analysis of the effect on the long term pressure response of assuming that feedwater continues to flow into the reactor pressure vessel (RPV). The calculation of the amount of feedwater assumed to enter the RPV should include the effects of pump coastdown and the possible pressure difference across the pump aiding the pump in pumping the feedwater.

021.10 (6.2.1) Provide the maximum calculated negative pressure following an inadvertent actuation of the containment spray system. Include a description of the analytical model and justify that the assumptions used to determine the internal containment pressure response are conservative.

021.11 (6.2.1) In Section 6.2.1.1.2.6, it is stated that a total failure of the containment cooling system will not result in exceeding design conditions inside containment. Provide analytical justification for this statement including the calculated rate of temperature increase inside the drywell and containment, the time it takes to effect repairs, and the final temperature of safety-related equipment inside the drywell and containment. Compare this calculated final temperature with the design temperature of the equipment.

021.12 (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:

1. Provide and justify the pipe break type, area, and location for each analysis. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design, or both. Consideration should be given for breaks in the main steam lines, feedwater lines and recirculation lines in the analysis.

2. For each compartment, provide a table of blowdown mass flow rate and energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component supports evaluation.

3. Provide a schematic drawing showing the compartment nodalization for the determination of maximum 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.

4. Provide a tabulation of the nodal net-free volumes and interconnecting flow path areas. For each flow path provide an L/A (ft^{-1}) ratio, where L is the average distance the fluid flows in that flow path and A is the effective cross sectional area. Provide and justify values of vent loss coefficients and/or friction factors used to calculate flow between nodal volumes. When a loss coefficient consists of more than one component, identify each component, its value, and the flow area at which the loss coefficient applies.

5. 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.

6. Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for a selected number of nodes. Discuss the basis establishing the differential pressure on structures and components.

7. Provide justification for the initial atmospheric conditions assumed in the analysis. An acceptable approach would be to assume air at maximum allowable temperature, minimum absolute pressure, and minimum relative humidity.

8. For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is

uniformally applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed.

9. Provide the peak and transient loading on the major components used to establish the adequacy of the supports design. This should include the load forcing functions [e.g., $f_x(t)$, $f_y(t)$, $f_z(t)$] and transient moments [e.g., $M_x(t)$, $M_y(t)$, $M_z(t)$] as resolved about a specific, identified coordinate system.

021.13 (6.2.1) 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.

021.14 (6.2.1) In Section 6.2.1.2.3 of the FSAR, it is stated that the subcompartment code used is described in Bechtel Topical Report, BN-TOP-4, Rev. 0, which does not account for inertial effects. We feel that the methods used in calculating the pressure response in the various subcompartments should include terms that account for the inertial effects present. Provide a subcompartment analysis using techniques that account for inertial effects.

021.15 (6.2.2) Provide a detailed analysis of the available net positive suction head for the RHR pumps to demonstrate compliance with the provisions of Regulatory Guide 1.1, "NPSH for Emergency Core Cooling and Containment Heat Removal Systems Pumps." Specify the required NPSH of the pumps.

021.16 (6.2.2) Describe the sizing analysis performed for the RHR suction screens. Provide a drawing that shows the suction screen assembly.

021.17 (6.2.3) Attached is our Branch Technical Position (BTP) CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containments." The criteria stated in BTP CSB 6-3 should be applied to any line penetrating primary containment that is not entirely contained in the secondary containment. Note that to eliminate air filled lines as potential bypass leak paths that are not entirely enclosed within the secondary containment, an important consideration is that the pressure inside these lines must be shown to be greater than the pressure inside containment at all times, unless it can be shown that for the section of line outside secondary containment there exists no path for leakage (e.g., a continuous section of pipe with no connections).

021.18 (6.2.3) Provide the following additional information related to potential bypass leakage paths.

a) For each air or water seal, perform an analysis that will demonstrate that a sufficient inventory of the fluid is available to maintain the seal for 30 days, and describe the testing program and proposed

entries for the Technical Specifications that will verify the assumptions used in the analysis. Provide the basis for the valve fluid leakage used in the analysis;

b) For each of these paths where water seals eliminate the potential for bypass leakage, provide a sketch to show the location of the water seal relative to the system isolation valves.

021.19 (6.2.3) For those lines that are considered potential bypass leak paths, provide the calculated leak rates and the analytical method by which this bypass leakage was calculated.

021.20 (6.2.3) The purpose of the secondary containment is to collect and process all leakage from the primary containment by using the Standby Gas Treatment System to maintain a negative pressure in the secondary containment. Discuss the preoperational and periodic test programs that will be carried out to verify the depressurization time stated in the FSAR and also that the proscribed negative pressure can be uniformly maintained throughout the secondary containment. The effect of open doors or hatches on the functional capability of the depressurization and filtration systems should be evaluated and included in the test programs.

021.21 (6.2.3) Identify all openings provided for gaining access to the secondary containment, and discuss the administrative controls that will be exercised over them. Discuss the instrumentation to be provided to monitor the status of the openings and whether or not position indicators and alarms will have readout and alarm capability in the main control room.

021.22 (6.2.3) Heat loads from equipment inside the auxiliary building (e.g., the ECCS pumps) were not considered in the drawdown analysis for the secondary containment. Demonstrate that these heat loads will have no effect on the drawdown time in the secondary system or redo your analysis for the drawdown time considering all possible heat loads.

021.23 (6.2.4) Figures 6.2-76 through 6.2-80 do not include all lines penetrating containment and do not provide sufficient detail to allow for a review of the arrangement of lines penetrating containment against the criteria set forth in GDC 55 and 56. Provide schematics for each line penetrating containment showing all connections to these lines, such as test, vent, and drain (TVD) connections and branch lines, between the inside and outside isolation barriers. In particular, Figures 6.2-76 through 6.2-80 do not include the following penetrations:

1) the 1-1/2" lines through penetrations 5, 6, 7 and 8;
2) the 1" line through penetration 17;
3) the 1-1/2" lines through penetration 48; and
4) Penetration No. 75.

021.24 (6.2.4) On page 6.2-64 of the FSAR it is stated that the outside isolation valve for the RHR head spray line (penetration number 18) receives an automatic isolation signal and that the outside isolation valve for the RCIC line is remote-manually actuated to provide long-term leakage control. Table 6.2-44 however, indicates that the RHR valve is opened during post-accident conditions and that the RCIC valve is closed during post-accident conditions. For these lines, clarify the post-accident conditions.

021.25 (6.2.4) The leak tight ECCS pump rooms are to be sized such that if a failure of a seal or gasket on a line from the suppression pool occurred inside the pump room, the volume of suppression pool water needed to fill the ECCS pump room would not reduce the suppression pool level below the minimum drawdown line. Verify that this is the case for the Grand Gulf Station and provide the details of your analysis, including the distance below the pool surface of the suction lines at the minimum drawdown level.

021.26 (6.2.4) Discuss the capability available to detect leakage and to take appropriate action in those lines needed for safe shutdown of the plant, or which are part of the engineered safety features, and that have a remote-manual valve or check valve inside containment and a remote-manual valve outside containment.

021.27 (6.2.4) For those lines that are required to be open following an accident which have either two or more isolation valves outside containment and no isolation valves inside containment, or a single remote-manual isolation valve outside containment in a closed system and no isolation valves inside containment, show that:

1) The remote-manual isolation valve nearest the containment and the piping between the containment and the valve is enclosed in a leak-tight or controlled leakage housing, or;

2) The design of the piping up to and including the first remote-manual isolation valve conforms to the provisions of SRP Section 3.6.2.

In either case the design of the valve and/or the piping compartment should provide the capability to detect leakage from the valve shaft and/or bonnet seals and to terminate the leakage.

021.28 (6.2.4) It is not clear on many of the lines listed in Table 6.2-44 where only one isolation valve is listed if a second isolation barrier exists for these lines. Examples of this are penetrations 48, 67, 71A & B, 73, 76 and 77. Provide information on any further isolation barriers in those lines.

021.29 (6.2.4) Provide a complete drawing for each closed system outside containment for which credit is claimed as an isolation barrier. Show all piping connecting to the closed system up to a second isolation barrier. Identify all lines connected to the closed systems that leave the secondary containment.

021.30 (6.2.4) After reviewing Table 6.2-44 and Figures 6.2-76 through 6.2-80, the following was noted:

1) The 1-1/2" lines for penetrations 5, 6, 7, and 8 show only one isolation valve.

2) The following penetrations contained valves whose post-accident position is closed, but did not have any isolation signals to close the valves should they be open: penetration numbers 23, 27, 32, 33, 44, 45, 48, 67 and 77. These isolation valves should be provided with isolation signals to assure that these valves are closed if an accident occurs or these valves should be locked closed with the appropriate administrative controls applied.

3) The remote-manual valves for penetrations 44 and 45, which are open during normal operation, are said to fail close in the post-accident period. No information on how this can occur is presented.

4) Certain lines (penetrations 48 and 77) have the same source of power to both isolation valves in the line.

5) Penetration number 76A that appears in Figure 6.2-79 does not appear in Table 6.2-44.

6) Penetration number 77 has a 1-1/2" line with two check valves in series. Provide justification for this arrangement.

7) It is not clear what kind of isolation barriers are present in penetrations 110, A, C, and F.

Please provide clarification statements for these items.

021.31 (6.2.4) Provide a detailed drawing of the recirculation pump seal water supply line that will show if it connects to the CRD supply line between the isolation valve inside containment and the containment itself. If it does, provide information on any isolation valves in the recirculation pump seal water supply line.

021.32 (6.2.5) Table 6.2-51 indicates that the two drywell hydrogen sample points and the two containment hydrogen sample points will be located at the same elevation. Justify placing the hydrogen samples in this manner as being more conservative than placing each sample point at a different location (e.g., place sample points towards the top of the drywell and containment and sample points towards the bottom of the drywell and containment) or revise your design accordingly.

021.33 The statement is made in Section 6.2.5.3.3.1 of the FSAR that there will be no hydrogen generation resulting from spray water contact with any aluminum or zinc components because the containment spray water contains no boron. The staff has determined that hydrogen release from zinc corrosion following a postulated loss-of-coolant accident should be considered in the analysis of hydrogen production and accumulation within the containment. Therefore, justify your position that corrosion of zinc will not occur, or provide the following:

(a) the corrosion rate as a function of temperature for all materials in the containment that could become a source of hydrogen due to corrosion.

(b) describe how the corrosion rates for the various materials were established. In so doing, identify the experimental data used as a basis (and provide references) and discuss the conservatism in the applicability of the data in view of the environmental conditions that are expected following LOCA.

(c) Graphically show the hydrogen concentration inside the containment as a function of time, with no recombiners operating, with one recombiner operating (minimum engineered safety features), and with both recombiners operating.

(d) Graphically show the contribution of each source of hydrogen as a function of time.

In either case, provide the mass and surface area of zinc paint and

galvanized steel and other corrodible material in both the drywell and the wetwell for our confirmatory analysis.

021.34 (3.2.6) With regard to system venting and draining for the Type A containment leak rate test provide the following information:

1) Itemize each system penetrating containment and discuss the venting and draining provisions for each system. Systems that are not designed to remain intact following a LOCA should have the isolation valves exposed to the containment atmosphere to permit the test differential pressure to be applied across them; i.e., the system should be vented and drained both upstream and downstream of the isolation valves. For each system penetrating containment that is not vented and drained, provide justification.

2) Identify any gas filled lines that will not be vented for the Type A test and provide justification for not doing so.

021.35 Closed systems outside containment having a post accident function, become extensions of the containment boundary following a LOCA. Certain of these systems may also be identified as one of the redundant containment isolation barriers. Since these systems may circulate contaminated water or the containment atmosphere, system components which may leak are relied on to provide containment integrity. Therefore, discuss your plans for specifying a leakage limit for each system that

becomes an extension of the containment boundary following a LOCA, and leak testing the systems either hydrostatically or pneumatically. Also discuss how the leakage will be included in the radiological assessment of the site.

021.36 (6.2.6) Containment isolation valves may be exempted from the Type C test requirements if it can be shown that the valve does not constitute a potential containment atmosphere leak path following a loss of coolant accident.

Table 6.2-44 identifies those containment isolation valves that will not be Type C tested. Therefore, justify that they do not constitute potential containment atmosphere leak paths following a LOCA. In this regard, a water seal may be shown to exist that will preclude containment atmosphere leakage. If this approach is taken, discuss how a water seal can be established and maintained using safety grade pipes and components, and considering single failure of active components. System drawings showing the routing and elevation of piping should be used to show the existence of a water seal.

When operation of a system is needed to maintain a water seal in the system, the ECCS for example, show that the system will keep its water seal for a sufficient period of time if the system is removed from operation.

021.37 Where more than one bellows is utilized on a penetration, provide
(6.2.6) assurance that each bellows will be subjected to Type B testing.

Enclosure

Branch Technical Position CSB 6-3

DETERMINATION OF BYPASS LEAKAGE
PATHS IN DUAL CONTAINMENT PLANTS

A. BACKGROUND

The purpose of this branch position is to provide guidance in the determination of that portion of the primary containment leakage that will not be collected and processed by the secondary containment. Bypass leakage is defined as that leakage from the primary containment which can circumvent the secondary containment boundary and escape directly to the environment i.e., bypasses the leakage collection and filtration systems of the secondary containment. This leakage component must be considered in the radiological analysis of a loss-of-coolant accident.

The secondary containment consists of a structure which completely encloses the primary containment and can be maintained at a pressure lower than atmospheric so that primary containment leakage can be collected or processed before release to the environment. The secondary containment may include an enclosure building which forms an annular volume around the primary containment, or the auxiliary building where it completely encloses the primary containment and other regions of the plant that are provided with leakage collection and filtration systems. Depressurization systems are provided as part of the secondary containment to decrease the pressure to below atmospheric and/or maintain the secondary containment volume at a negative pressure.

All primary containment leakage may not be collected because (1) direct primary containment leakage can occur while the secondary containment is being depressurized and (2) primary containment leakage can bypass the secondary containment through containment penetrations and seals which do not terminate in the secondary containment.

Direct leakage from the secondary containment to the environment can occur whenever an outward positive differential pressure exists across the secondary containment boundary. The secondary containment can

experience a positive pressure transient following a postulated loss-of-coolant accident in the primary containment as a result of thermal loading and infiltration from the environment and the primary containment that will occur until the depressurization systems become effective. An outward positive differential on the secondary containment wall can also be created by wind loads. In this regard, a "positive" pressure is defined as any pressure greater than -0.25 in. w.g. (water gauge), to account for wind loads and the uncertainty in the pressure measurements. Whenever the pressure in the secondary containment volume exceeds -0.25 in. w.g., the leakage-prevention function of the secondary containment is assumed to be negated. Since leakage from the secondary containment during positive pressure periods cannot be determined, the conservative assumption is made that all primary containment leakage is released directly to the environment during these time periods. Therefore, it becomes necessary to determine the time periods during which these threshold conditions exist.

The existence and duration of periods of positive pressure within the secondary containment should be based on analyses of the secondary containment pressure response to postulated loss-of-coolant accidents within the primary containment and the effectiveness of the depressurization systems.

The evaluation of bypass leakage involves both the identification of bypass leakage paths and the determination of leakage rates. Potential bypass leakage paths are formed by penetrations which pass through both the primary and secondary containment boundaries. Penetrations that pass through both the primary and secondary containment may include a number of barriers to leakage (e.g., isolation valves, seals, gaskets, and welded joints). While each of these barriers aid in the reduction of leakage, they do not necessarily eliminate leakage. Therefore, in identifying potential leakage paths, each of these penetrations should be considered, together with the capability to test them for leakage in a manner similar to the containment leakage tests required by Appendix J to 10 CFR Part 50.

B. BRANCH TECHNICAL POSITION

1. A full secondary containment structure should completely enclose the primary containment structure, with the exception of those parts of the primary containment that are imbedded in the soil, such as the base mat of the containment structure. For partial dual containment concepts, leak rates less than the design leak rate of the primary containment cannot be used in the calculation of the radiological consequences of a loss-of-coolant accident, because it has not been possible for applicants to quantitatively assess and confirm the extent of undetected and unprocessed leakage.

2. Direct leakage from the primary containment to the environment, equivalent to the design leak rate of the primary containment, should be assumed to occur following a postulated loss-of-coolant accident whenever the secondary containment volume is at a "positive" pressure; i.e., a pressure greater than -0.25 in. w.g.. Positive pressure periods should be determined by a pressure response analysis of the secondary containment volume that includes thermal loads from the primary containment and infiltration leakage.

3. The secondary containment depressurization and filtration systems should be designed in accordance with Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants". Preoperational and periodic inservice inspection and test programs should be proposed for these systems and should include means for determining the secondary containment infiltration rate, and the capability of the systems to draw down the secondary containment to the prescribed negative pressure in a prescribed time.

4. For secondary containments with design leakage rates greater than 100 volume percent per day, an exfiltration analysis should be provided.

5. The following leakage barriers in paths which do not terminate within the secondary containment should be considered potential bypass

leakage paths around the leakage collection and filtration systems of the secondary containment:

a. Isolation valves in piping which penetrate both the primary and secondary containment barriers;

b. Seals and gaskets on penetrations which pass through both the primary and secondary containment barriers; and,

c. Welded joints on penetrations (e.g., guard pipes) which pass through both the primary and secondary containment barriers.

6. The total leakage rate for all potential bypass leakage paths, as identified in item 5 above, should be determined in a realistic manner, considering equipment design limitations and test sensitivities. This value should be used in calculating the offsite radiological consequences of postulated loss-of-coolant accidents and in setting technical specification limits with a margin for bypass leakage.

7. Provisions should be made to permit preoperational and periodic leakage rate testing in a manner similar to the Type B or C tests of Appendix J to 10 CFR Part 50 for each bypass leakage path listed in item 5 above. An acceptable alternate for local leakage rate testing for welded joints would be to conduct a soap bubble test of the welds concurrently with the integrated (Type A) leakage test of the primary containment required by Appendix J. Any detectable leakage determined in this manner would require repair of the joint.

8. If air or water sealing systems or leakage control systems are proposed to process or eliminate leakage through valves, these systems should be designed, to the extent practical, using the guidelines for leakage control systems given in Branch Technical Position APCSB 6-1 (Ref. 3).

9. If a closed system is proposed as a leakage boundary to preclude bypass leakage, then the system should:

a. Either (1) not directly communicate with the containment atmosphere, or (2) not directly communicate with the environment,

following a loss-of-coolant accident.

b. Be designed in accordance with Quality Group B standards, as defined by Regulatory Guide 1.26. (Systems designed to Quality Group C or D standards that qualify as closed systems to preclude bypass leakage will be considered on a case-by-case basis.)

c. Meet seismic Category I design requirements.

d. Be designed to at least the primary containment pressure and temperature design conditions.

e. Be designed for protection against pipe whip, missiles, and jet forces in a manner similar to that for engineered safety features.

f. Be tested for leakage, unless it can be shown that during normal plant operations the system integrity is maintained.

C. REFERENCES

1. 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors".

2. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants", Revision 1.

3. Branch Technical Position APCSB 6-1, "Main Steam Isolation Valve Leakage Control Systems", attached to Standard Review Plan 6.7.

110.0 MECHANICAL ENGINEERING

110.6 (3.6A.1.1) Expand the first paragraph of Section 3.6A.1.1.d (page 3.6A-5) to clearly indicate that the systems, components and equipment necessary to mitigate the consequences of a postulated piping failure are also considered to be essential and, therefore, protected in accordance with the remainder of Section 3.6A.

110.7 (3.6A.1.3) (3.6B.2.1) Sections 3.6A.1.3.2.d (page 3.6A-8) and 3.6B.2.1.1.4 (pages 3.6B-2 and 3) identify the criteria used to exempt certain postulated pipe breaks from consideration of pipe whip.

Verify that the other affects (such as jet impingement, pressure, temperature, humidity, wetting of all exposed equipment, flooding) of such breaks are considered.

110.8 (3.6A.2.1) (3.6B.2.1) (RSP) It is the Staff's position that a branch connection to a main run need not be considered as a terminal end when all of the following are met:

(1) The branch and main runs are of comparable size and fixity (i.e., the nominal size of the branch is at least one half of that of the main);

(2) The intersection is not rigidly constrained to the building structure; and

(3) The branch and main runs are modeled as a common piping system during the piping stress analysis.

Expand the third item of Sections 3.6A.2.1.c.1.(a)(1) (page 3.6A-11) and 3.6A.2.1.c.1.(b)(1) (pages 3.6A-12 and 13) and the note to Section 3.6B.2.1.1.5a (page 3.6A-3) to correspond with this definition of terminal ends.

110.9 (3.6A.2.1) Expand Section 3.6A.2.1.c.1 (pages 3.6A-11 through 13) to indicate the criteria used for postulating break locations in high energy piping not designed to seismic Category I standards.

110.10 (3.6A.2.1) (3.6B.2.1) Table 3.6B-3) Modify the reference citations to indicate valid sections of the FSAR in the following sections:

1) Section 3.6A.2.1.c.2.(a) (page 3.6A-14)
2) Section 3.6A.2.1.c.2.(b) (page 3.6A-14)
3) Section 3.6A.2.1.c.3. (page 3.6A-15)
4) Section 3.6A.2.1.c.3.(a)(3) (page 3.6A-16)
5) Section 3.6A.2.1.c.3.(a)(4) (page 3.6A-16)
6) Section 3.6B.2.1.3.4 (page 3.6B-7)
7) Table 3.6B-3, both side and bottom
8) Appendix 3c.2.3 (page Appendix 3c-7)

110.11 (3.6A.2.1) (3.6B.2.1) (RSP) It is the staff's position that piping between the containment isolation valves for which no breaks are postulated shall receive a 100 percent volumetric examination of all circumferential, longitudinal, and branch to main run welds during each inspection interval (IWA-2400 of the ASME Code).

Expand Sections 3.6A.2.1.c.2.(g) (page 3.6A-15) and 3.6B.2.1.3 (page 3.6B-6 and 7) to provide a commitment to such an augmented inservice inspection program.

110.12 (3.6A.2.1) (RSP) It is the Staff's position that circumferential pipe breaks are to be postulated in all high energy piping systems exceeding a nominal size of 1 inch.

Modify Section 3.6A.2.1.c.3.(a)(1) (page 3.6A-16) to provide a commitment to postulate pipe breaks in the CRD piping.

110.13 (3.6A.2.1) Modify the last sentence of Section 3.6A.2.1.c.3.(a)(4) (page 3.6A-16) to indicate that longitudinal pipe breaks will be the only type postulated when the circumferential stress is at least 1.5 times the longitudinal stress.

110.14 (3.6A.2.1) (3.6B.2.1) Expand Sections 3.6A.2.1.c.3.(a)(4) (page 3.6A-16) and 3.6B.2.1.1.6.d.3 (page 3.6B-5) to indicate how consideration of the maximum stress range is used to exempt circumferential pipe breaks when the postulated break location is due to a usage factor in excess of 0.1.

110.15 (3.6A.2.1) Expand Section 3.6A.2.1.c.4.(a) (pages 3.6A-16 and 17) to provide justification for the assumption that restraining one end of the postulated circumferential pipe break reduces the jet force and reaction by reducing the flow area.

110.16 (3.6A.2.1) Expand the first item of Section 3.6A.2.1.c.4.(d)(1) (page 3.6A-17) to indicate the criteria for postulating cracks in moderate energy ASME Class 1 piping.

110.17 (3.6A.2.1) Expand the first two items of Section 3.6A.2.1.c.4.(d)(1) (page 3.6A-17) to indicate the plant operating conditions to be considered in the evaluation.

110.18 (3.6A.2.1) Expand the first two items of Section 3.6A.2.1.c.4.(d)(1) (page 3.6A-17) to include a definition of maximum stress range.

110.19 (3.6A.2.1) Expand the second item of Section 3.6A.2.1.c.4.(d)(1) (page 3.6A-17) to provide the justification for the use of Equation 9 in lieu of Equation 10 to determine postulated pipe crack locations in ASME Class 1 piping.

110.20 (3.6A.2.1)	Expand Section 3.6A.2.1.c.4.(d)(1) (pages 3.6A-17 and 18) to indicate the criteria used for postulating cracks in moderate energy piping not designed to seismic Category I standards.
110.21 (3.6A.2.3)	Clarify the second sentence of the first paragraph of Section 3.6A.2.3.1 (page 3.6A-19).
110.22 (3.6A.2.3)	Expand Sections 3.6A.2.3.3.2.d.1 (pages 3.6A-23 and 24), 3.6B.2.3.3.2.b.3 (page 3.6B-16) and 3.6B.2.3.3.2.c.3 (page 3.6B-17) to provide assurance that the design procedures used will insure that the design strain does not exceed 0.5 of the ultimate uniform strain of the materials used in the GE U-bolt pipe whip restraints.
110.23 (3.6A.2.3) (RSP)	It is the Staff's position that crushable honeycomb material may be used for pipe whip restraints provided that, under design loading, the honeycomb material will not experience a deflection in excess of that which is defined by the horizontal portion of its load deflection curve. Expand Section 3.6A.2.3.3.2.d.2 (pages 3.6A-24 and 25) to provide a commitment to such a design limit.
110.24 (3.6A.2.5)	The first paragraph of Section 3.6A.25 (pages 3.6A-29 and 30) indicates that Tables 3.6A-1 through 3.6A-13B and Figures 3.6A-14 through 3.6A-30 will be updated only for changes which significantly affect the pipe break evaluation. While we agree that minor changes need not be immediately reported, we do believe that the latest data should be in the FSAR at two key points in the licensing process - namely, at the time of issuance of the SER and the OL. Therefore, we request that you provide a commitment to update these Tables and Figures prior to issuance of the SER and, again, prior to issuance of the OL
110.25 (3.6A.2)	Expand Section 3.6A.2 (pages 3.6A-10 through 30) to describe the protection criteria for the effects due to jet impingement.
110.26 (Tables 3.6A-2, -4, -6, -8, -9, -11 thru -13, and -27)	Tables 3.6A-2, 4, 6, 8, 9, 11 through 13, and 27 are indicated as "Later." Provide a schedule for their inclusion in the FSAR.

110.27 (Table 3.6A-14) Expand Table 3.6A-14 to include the following systems:

(1) Head vent

(2) Condensate

110.28 (3.6B.2.1) Expand Section 3.6B.2.1.1.6.e (page 3.6B-5) to describe the "mechanistic approach" used to justify longitudinal breaks with a break area less than the flow area of the pipe.

110.29 (3.9.2.5) Previous analyses for other nuclear plants 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. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel

b. Core supports and other reactor internals

c. Control rod drives

d. ECCS piping that is attached to the primary coolant piping

e. Primary coolant piping

f. Reactor vessel supports

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 various cavity structures.

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

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural, mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. 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 the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other faulted condition loads for the postulated breaks described above. 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

4. If the results of the assessment in item 3. above indicate loads leading to inelastic action in these systems or displacment exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect on the load transmitted to the backup structures to which these systems are attached.

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

5. For all analyses 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.

6. Demonstrate that active components will perform their safety functin when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

110.30 (3.9.3) Your response to 110.4 is not entirely acceptable. Indicate if the design of both NSSS and non-NSSS Class 1, 2, and 3 component supports complies with paragraph 1370(c) of Appendix F of the ASME Code Section III. The staff has noted in Regulatory Guides 1.124 and 1.130 that the increases in allowable stress described in NF-3231.1(b) and (c) do not apply to buckling analyses and that the F-1370(c) buckling limit is applicable.

110.31 (3.9.1.1) (3.7.3) There are conflicting statements in the FSAR regarding the consideration of the OBE in NSSS and BOP Class 1 component fatigue calculations. FSAR table 3.9-1 and various parts of FSAR Section 3.9.1.1 state that the OBE has not been considered in your fatigue analyses. However, FSAR section 3.7.3.8.1.2.6 and FSAR table 3.7-14 describe the number of maximum stress cycles due to OBE that were considered in the BOP and NSSS Class 1 fatigue analyses respectively.

Provide clarification of the consideration of OBE loads for NSSS scope ASME Class 1 components to resolve the apparent conflicts between the FSAR sections. As noted in Enclosure 110-2, OBE loads are to be evaluated against Service Level B requirements which include fatigue analyses.

110.32 (3.9.1.1) (3.6.2) Provide confirmation that Mark III containment SRV discharge and suppression pool vibratory loads have been taken into account, i.e., load cases 1 and 2 of Enclosure 110-1, for determination of postulated pipe break locations in ASME Class 1, 2, and 3 piping using the stress and usage factor criteria specified in 3.6 of the FSAR.

110.33 (3.9.2.1) We have identified several portions of your vibration, thermal, and dynamic effects testing program for NSSS and BOP piping which deviate from the criteria of SRP Section 3.9.2. We require certain additional information to more fully define your program. Modify FSAR sections 3.9.2.1.1 and 3.9.2.1.2 to provide this information for both NSSS and BOP systems.

(1) Expand your program to include the following piping systems, including their supports and restraints:

(a) all ASME Class 1, 2, and 3 systems,

(b) other high energy piping systems inside seismic Category I structures,

(c) high energy portions of systems whose failure could reduce the functioning of any seismic Category I plant feature to an unacceptable safety level, and

(d) seismic Category I portions of moderate energy piping systems located outside containment.

A visual check of many of these systems is acceptable.

(2) Describe how your program will verify that no restraint of normal thermal movement occurs in the systems listed in (1) for the NSSS scope of supply. FSAR section 3.9.2.1.2.2 already describes your procedures for the BOP scope of supply.

(3) Describe in more detail how your program will verify the adequate performance of snubbers for the systems listed in (1).

(4) For NSSS piping you provide various references to "Code limits" as the Level 1 criteria against which measurements will be compared. Indicate in more detail how the measurements of your test program will be related to such limits.

(5) Provide a brief list of the major transient events to be included in the BOP program such as relief valve actuation, turbine stop valve closure, major pump starts and stops, etc.

(6) Provide a list of any BOP systems which will be instrumented. Instrumentation is not required by the staff if visual observation can provide a meaningful inspection.

(7) For the BOP program, describe the bases for the acceptance criteria against which any measurements will be compared.

(8) Provide a cross reference between FSAR section 3.9.2.1 and the appropriate test descriptions in FSAR chapter 14.

110.34 (3.9.3) (3.9.4) (3.9.5) (RSP) FSAR section 3.9.3, 3.9.4, and 3.9.5 reference several tables (3.9-2, 3.9-6, 3.9-17, etc.) that describe the various loading combinations considered in the design of ASME Class 1, 2, and 3 components, component supports, core support structures, control rod drive components, and other reactor internals.

We have had discussions with the Mark II Owner's Group concerning the load combinations appropriate for the design of BWR Mark II plants. Our position with respect to load combinations has been documented as Attachment A to Enclosure 5 of the NRC Mark II Generic Acceptance Criteria for Lead Plants. This staff position is repeated here as Enclosure 110-1. These loading combinations are applicable to a Mark III plant such as Grand Gulf.

Therefore, provide a commitment that all ASME Class 1, 2, and 3 components, component supports, core support structures, control rod drive components, and other reactor internals have been or will be analyzed or otherwise qualified in accordance with Enclosure 110-1, as modified by the following two clarifications:

(a) For load cases 1 and 2, all ASME Code Service Level B requirements are to be met, including fatigue usage factor requirements, and should take into account all SRV discharge load effects (initial actuation and continuous suppression pool vibratory) taken for the number of cycles consistent with the 40 yr. design life of the plant.

(b) For load case 10, SRV should be assumed to be one SRV.
x

110.35 (3.9.3) For reactor coolant pressure boundary components and supports, we have accepted the use of the square root of sum of squares methodology for combining dynamic responses resulting from LOCA and SSE. This acceptance is documented in NUREG-0484 "Methodology for Combining Dynamic Responses." At this time, we have not accepte the use of SRSS for combining responses from other combinations of dynamic loads and for other components and supports. Our review of the SRSS methodology is continuing and we are concentrating on the proposed Kennedy-Newmark criteria, which is being proposed by the Mark II Owner's Group. The eventual outcome is expected to establish our position and criteria for general acceptance of response combination using SRSS methods.

110.35 We request that you provide in the FSAR a specific listing of all combinations of dynamic loads and all components for which combination of dynamic responses by the SRSS method is proposed. The listing should specifically include such loads as OBE inertia loads, OBE anchor point movement loads, SRV loads, turbine stop valve closure loads, Mark III containment hydrodynamic vibratory loads, SSE loads, and LOCA loads (including annulus pressurization.)

110.36 (3.9.3) Provide information regarding effects of seismic sloshing loads on the piping and components. An acceptable response can be found in the LaSalle (docket 05000373) response to Question 111.78.

110.37 (3.9.3) (3.6.2) For ASME Class 1, 2, and 3 components that could be exposed to jet impingement or pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping, describe the procedure used to determine the stress levels in the components and all other components in the target system resulting from exposure to such loads in combination with those resulting from other applicable loads. Provide specific assurance that the calculated stress levels are kept below ASME Service Level D limits or, if applicable, more conservative limits for active components or where piping functional capability must be assured.

110.38 (3.9.3.2) Your FSAR indicates that active valves will be qualified for operability under seismic loading on a prototype basis. We agree that a prototypical test can qualify a limited range of similar valves. Your FSAR does not sufficiently describe the characteristics you consider in determining that a valve is similar to the tested prototype valve, and therefore can be qualified by analysis only.

Provide a discussion of how you establish the "similarity" of valves to a tested prototype. This discusson should include, but not be limited to, characteristics such as valve type, size, geometry, pressure rating, stress level, manufacturer, actuator type, and actuator load rating.

110.39 (3.9.3.4) Describe in more detail the dynamic testing performed to demonstrate the operability of safety related NSSS and BOP snubbers under upset, emergency and faulted load combinations. Describe the magnitudes of the applied loads, the frequency content, and the number of load cycles at each applied load level in these tests.

110.40 (3.9.6) As required by 10 CFR 50.55a(g) we request that you submit your preservice and initial 20 month inservice testing program for pumps and valves. Enclosure 110-3 provides a suggested format for this submittal and a discussion of information we require to justify any relief requests.

110.41 (3.10) (3.9.2) (RSP) It is not clear from the FSAR how the seismic analyses of seismic Category I electrical and mechanical equipment have taken into consideration all three seismic accelerations (i.e., x, y, and z directions) acting on the equipment.

Regulatory Guide 1.92 provides methods acceptable to the staff for combining the response to the three spatial components of seismic excitation.

Describe how your analyses have considered the three spatial components of seismic excitation.

110.42 (3.10) (3.9.2) (RSP) Hydrodynamic vibratory loadings result from the flow of a steam-water-air mixture into the suppression pool. This flow may result from SRV actuation or from a postulated pipe break. In either case the resultant vibration of the suppression pool may affect components in other portions of the reactor building. Therefore, hydrodynamic vibratory loadings of various magnitude and frequency content can be associated with the following cases: SRV_1, SRV_x, SRV_{ADS}, SRV_{ALL}, IBA, and DBA.

The staff will require that electrical and mechanical equipment required for cold shutdown be demonstrated capable of performing their safety function under the most severe of the following combinations of seismic and hydrodynamic vibratory loadings: (as appropriate depending upon the location of the equipment)

(1) SRV_x or SRV_{ALL} (whichever is controlling) + OBE

(2) SRV_x or SRV_{ALL} (whichever is controlling) + SSE

(3) SRV_{ADS} + OBE + IBA

(4) SRV_{ADS} + SSE + IBA

(5) SSE + DBA

(6) SRV_1 + SSE + DBA

110.42 Provide a commitment that all NSSS and BOP seismic Category I mechanical and electrical equipment will be qualified for the most severe combined seismic and hydrodynamic vibratory loadings. Recent BWR Operating License applicants, such as the LaSalle (docket 05000373) and Zimmer (docket 05000358) plants, have stated that, in general, the SRV_{ALL} case imposes the most severe hydrodynamic vibratory loadings on safety-related equipment. However, this does not preclude the possibility that other hydrodynamic loads might be limiting for particular components at your plant. As noted above, you should consider the most limiting case.

110.43 (3.10) (3.9.2) A review of the design adequacy of your safety-related electrical and mechanical equipment under seismic and hydrodynamic loadings will be performed by our Seismic Qualification Review Team (SQRT). A site visit at some future date will be necessary to inspect and otherwise evaluate selected equipment after our review of the following requested information. The SQRT effort will be primarily focused on two subjects. The first is the adequacy of the original single-axis, single-frequency tests or analyses.

The second subject is the qualification of equipment for the combined seismic and hydrodynamic vibratory loadings. The frequency of this vibration may exceed 33 hertz and negate the original assumption of a components rigidity in some cases.

Attached Enclosure 110-4 describes the SQRT and its procedures. Section V.2.A requires information which you should submit so that SQRT can perform its review.

Several of the BWR Mark II OL applicants have stated in their Closur Reports that equipment will be qualified for the SRSS combination of the hydrodynamic and seismic required response spectra (RRS). Similarily, when qualified by analysis, the peak dynamic responses of the equipment to the hydrodynamic and seismic loads will be combined by SRSS. The combining by SRSS of either the RRS or peak dynamic responses for hydrodynamic and seismic loadings is not acceptable at this time. (See 110.35)

To aid the staff in its review, provide a compilation of the required response spectra listed below for each floor of the seismic Category I buildings at your plant.

(1) the RRS for the OBE or SSE, whichever is controlling. If the OBE is controlling, explain why.

(2) the controlling hydrodynamic RRS

(3) items (1) and (2) combined by SRSS

(4) items (1) and (2) combined by absolute sum.

Enclosure 110-1

ACCEPTANCE CRITERIA FOR MARK II PIPING SYSTEMS

MECHANICAL ENGINEERING BRANCH

DIVISION OF SYSTEMS SAFETY

LOAD CASE	N(3)	SRV_X	SRV_{ADS}	OBE	SSE	IBA(1&5)	DBA(5)	ACCEPTANCE CRITERIA
1	X	X	-					B
2	X	X		X				B
3	X	X			X			C(4)
4	X		X			X		C(4)
5	X		X	X		X		C(4)
6	X		X		X	X(2)		C(4)
7	X				X		X(2)	C(4)
8	X							A
9	X			X				B
10	X	X			X		X(2)	C(4)

(1) Use SBA or IBA whichever is governing.

(2) Loading due to DBA/SBA/IBA is determined from rated steady state conditions.

(3) N - Normal load consists of pressure, dead weight, thermal & fluid reaction loads.

(4) Piping functional capability should be assured per Enclosure 110-2 or alternate means. Service level limits higher than the level specified in this table may be used, provided piping functional capability is demonstrated.

(5) SBA, IBA and DBA shall include all event induced loads whichever are applicable, such as possible annulus pressurization load, pool swell load, condensation oscillation load, chugging load, etc.

Enclosure 110-2

INTERIM TECHNICAL POSITION
FUNCTIONAL CAPABILITY OF PASSIVE PIPING COMPONENTS

MECHANICAL ENGINEERING BRANCH

DIVISION OF SYSTEMS SAFETY

I. Introduction

The functional capability of all piping components in essential ASME Class 1, 2 and 3 piping systems designed to Levels C or D service limits is required to be demonstrated. Applicants may choose to use the criteria in Section II which require no further proof of functional capability. Piping components within Section III require additional analytical or experimental proof that functional capability has been maintained.

The technical content of this position is based upon integrated experimental and analytical studies of piping system components performed at the Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission. The program of studies, the analytical and experimental results, discussions and recommendations have been documented in a report, "Evaluation of the Plastic Characteristics of Piping Products in Relation to ASME Code Criteria, ORNL/Sub-2913/8."

II. Situations in which Functional Capability is Assured without Further Proof

A. Class 1 Piping Components:

1. Functional capability may be considered assured without further proof for any Class 1 piping component when the Level "A" or "B" or "C" limit is used with Equation (9) of NB-3650 provided $D_o/t < 50$, where D_o is the outside diameter and t is the wall thickness of the piping component. The Level "C" limit to be satisfied for the above verification procedure is 1.5 S_y.

   A value of B_1 not less than 0.5 may be used in Equation (9) for the functional capability evaluation.

2. For tees and branch connections, the Level "D" limit may be used with Equation (9) of NB-3650 without additional requirements for functional verification, provided $D_o/t < 50$.

The Level "D" limit to be satisfied for the above verification procedure is 2.0 S_y.

A value of B_1 not less than 0.5 may be used in Equation (9) for the functional capability evaluation.

3. P_d concurrent with M_i may be used in Equation (9).

B. Class 2/3 Piping Components:

1. Functional capability may be considered assured for Class 2/3 piping components for Levels "A" and "B" limits in Equation (9) of NC-3652.1 or ND-3652.1 provided $D_o/t \leq 50$.

2. For tees and branch connections, Level "C" limits may be used without additional requirements for functional verification. However, for elbows or bends, the following additional requirements shall be met whenever Level "C" limits are specified:

(a) Use (0.8 B_2) instead of (0.75 i) but not less than 1.0.

(b) Use 1.5 S_y for the right-hand side of Equation (9).

In each of the above cases, D_o/t shall be equal to or less than 50.

3. P_d concurrent with M_i may be used in Equation (9).

4. Class 2/3 piping components may be evaluated as Class 1 piping components for verifying functional capability, provided the rules and limits as specified in item II.A., above, are met.

III. <u>Situations in which Functional Capability Requires Additional Demonstration</u>

A. Class 1 Piping Components:

1. Piping components other than tees and branch connections, such as elbows, pipe bends and straight pipe, using Level "D" limits.

2. Any piping components with $D_o/T > 50$.

B. Class 2/3 Piping Components:

1. Straight pipe when Level "C" limits are used.

2. Elbows or pipe bends which cannot meet the requirements specified in item II.B.2, above, when Level "C" limits are specified.

3. All piping components when Level "D" limits are used. (NOTE: The ORNL report recommends against the use of Level "D" limits when functional capability must be maintained.)

4. Any piping components with $D_o/t > 50$.

IV. Definitions

Functional Capability - Capability of piping components to deliver rated flow and retain dimensional stability when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

Piping Components - These items of a piping system, such as tees, elbows, bends, pipe and tubing and branch connections, constructed in accordance with the rules of Section III of the ASME Code.

Piping System - A group of connected piping components and other associated Code components (i.e., pumps, valves, vessels) performing jointly a specified plant function or, in the case of multifunctional systems, more than one function.

Essential Piping Systems - Piping systems which are necessary (a) for safe shutdown of the plant and to maintain the plant in a safe shutdown condition, or (b) to prevent or mitigate the consequences of an accident which could result in potential offsite exposures exceeding the guidelines of 10 CFR Part 100.

Enclosure 110-3

NRC STAFF COMMENTS ON INSERVICE PUMP AND VALVE TESTING PROGRAMS AND RELIEF REQUESTS

The NRC staff, after reviewing a number of pump and valve testing programs, has determined that further guidance might be helpful to illustrate the type and extent of information we feel is necessary to expedite the review of these programs. We feel that the Licensee can, by incorporating these guidelines into each program submittal, reduce considerably the staff's review time and time spent by the Licensee in responding to NRC staff requests for additional information.

The pump testing program should include all safety related* Class 1, 2 and 3 pumps which are installed in water cooled nuclear power plants and which are provided with an emergency power source.

The valve testing program should include all the safety related valves in the following systems excluding valves used for operating convenience only, such as manual vent, drain, instrument and test valves, and valves used for maintenance only.

PWR

a. High Pressure Injection System
b. Low Pressure Injection System
c. Accumulator Systems
d. Containment Spray System
e. Primary and Secondary System Safety and Relief Valves
f. Auxiliary Feedwater Systems
g. Reactor Building Cooling System
h. Active Components in Service Water and Instrument Air Systems which are required to support safety system functions.
i. Containment Isolation Valves required to change position to isolate containment.
j. Chemical & Volume Control System
k. Other key components in Auxiliary Systems which are required to directly support plant shutdown or safety system function.

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*Safety related - necessary to safely shut down the plant and mitigate the consequences of an accident.

l. Residual Heat Removal System

m. Reactor Coolant System

DWR

a. High Pressure Core Injection System

b. Low Pressure Core Injection System

c. Residual Heat Removal System (Shutdown Cooling System)

d. Emergency Condenser System (Isolation Condenser System)

e. Low Pressure Core Spray System

f. Containment Spray System

g. Safety, Relief, and Safety/Relief Valves

h. RCIC (Reactor Core Isolation Cooling) System

i. Containment Cooling System

j. Containment isolation valves required to change position to isolate containment

k. Standby liquid control system (Boron System)

l. Automatic Depressurization System (any pilot or control valves, associate hydraulic or pneumatic systems, etc.)

m. Control Rod Drive Hydraulic System ("Scram" function)

n. Other key components in Auxiliary Systems which are required to directly support plant shutdown or safety system function.

o. Reactor Coolant System

Inservice Pump and Valve Testing Program

I. Information required for NRC Staff Review of the Pump and Valve Testing Program

A. Three sets of P&ID's, which include all of the systems listed above, with the code class and system boundaries clearly marked. The drawings should include all of the components present at the time of submittal and a legend of the P&ID symbols.

B. Identification of the applicable ASME Code Edition and Addenda

C. The period for which the program is applicable.

D. Identify the component code class.

E. For Pump testing: Identify

1. Each pump required to be tested (name and number)
2. The test parameters to be measured
3. The test frequency

F. For valve testing: Identify

1. Each valve in ASME Section XI Categories A & B that will be exercised every three months during normal plant operation (indicate whether partial or full stroke exercise, and for power operated valves list the limiting value for stroke time.)
2. Each valve in ASME Section XI Category A that will be leak tested during refueling outages (Indicate the leak test procedure you intend to use)
3. Each valve in ASME Section XI Categories C, D and E that will be tested, the type of test and the test frequency. For check valves, identify those that will be exercised every 3 months and those that will only be exercised during cold shutdown or refueling outages.

II. Additional Information That Will Be Helpful in Speeding Up the Review Process

A. Include the valve location coordinates or other appropriate location information which will expedite our locating the valves on the P&IDs.

B. Provide P&ID drawings that are large and clear enough to be read easily.

C. Identify valves that are provided with an interlock to other components and a brief description of that function.

Relief Requests from Section XI Requirements

The largest area of concern for the NRC staff, in the review of an inservice valve and pump testing program, is in evaluating the basis for justifying relief from Section XI Requirements. It has been our experience

that many requests for relief, submitted in these programs, do not provide adequate descriptive and detailed technical information. This explicit information is necessary to provide reasonable assurance that the burden imposed on the licensee in complying with the code requirements is not justified by the increased level of safety obtained.

Relief requests which are submitted with a justification such as "Impractical", "Inaccessible", or any other categorical basis, will require additional information, as illustrated in the enclosed examples, to allow our staff to make an evaluation of that relief request. The intention of this guidance is to illustrate the content and extent of information required by the NRC staff, in the request for relief, to make a proper evaluation and adequately document the basis for that relief in our safety evaluation report. The NRC staff feels that by receiving this information in the program submittal, subsequent requests for additional information and delays in completing our review can be considerably reduced or eliminated.

I. <u>Information Required for NRC Review of Relief Requests</u>

A. Identify component for which relief is requested:

1. Name and number as given in FSAR
2. Function
3. ASME Section III Code Class
4. For valve testing, also specify the ASME Section XI valve category as defined in IWV-2000

B. Specifically identify the ASME Code requirement that has been determined to be impractical for each component.

C. Provide information to support the determination that the requirement in (B) is impractical; i.e., state and explain the basis for requesting relief.

D. Specify the inservice testing that will be performed in lieu of the ASME Code Section XI requirements.

E. Provide the schedule for implementation of the procedure(s) in (D).

II. Examples to Illustrate Several Possible Areas Where Relief May Be Granted and the Extent and Content of Information Necessary to Make An Evaluation

A. Accessibility: The regulation specifically grants relief from the code requirement because of insufficient access provisions. However, a detailed discussion of actual physical arrangement of the component in question to illustrate the insufficiency of space for conducting the required test is necessary.

Discuss in detail the physical arrangement of the component in question to demonstrate that there is not sufficient space to perform the code required inservice testing.

What alternative surveillance means which will provide an acceptable level of safety have you considered and why are these means not feasible?

B. Environmental Conditions (e.g., High radiation level, High temperature, High humidity, etc.)

Although it is prudent to maintain occupation radiation exposure for inspection personnel as low as practicable, the request for relief from the code requirements cannot be granted solely on the basis of high radiation levels alone. A balanced judgment between the hardships and compensating increase in the level of safety should be carefully established. If the health and safety of the public dictates the necessity of inservice testing, alternative means or even decontamination of the plant if necessary should be provided or developed.

Provide additional information regarding the radiation levels at the required test location. What alternative testing techniques which will provide an acceptable level of assurance of the integrity of the component in question have you considered and why are these techniques determined to be impractical?

C. Instrumentation is not originally provided
Provide information to justify that compliance with the code requirements would result in undue burden or hardships without a compensating increase in the level of plant safety. What alternative testing methods which will provide an acceptable level of safety have you considered and why are these methods determined to be impractical?

D. Valve Cycling During Plant Operation Could Put the Plant in an Unsafe Condition
The licensee should explain in detail why exercising tests during plant operation could jeopardize the plant safety.

E. Valve Testing at Cold Shutdown or Refueling Intervals in Lieu of the 3 Month Required Interval
The licensee should explain in detail why each valve cannot be exercised during normal operation. Also, for the valves where a refueling interval is indicated, explain in detail why each valve cannot be exercised during cold shutdown intervals.

III. Acceptance Criteria for Relief Request

The Licensee must successfully demonstrate that:

1. Compliance with the code requirements would result in hardships or unusual difficulties without a compensating increase in the level of safety and noncompliance will provide an acceptable level of quality and safety, or
2. Proposed alternatives to the code requirements or portions thereof will provide an acceptable level of quality and safety.

Standard Format

A standard format, for the valve portion of the pump and valve testing program and relief requests, is included as an attachment to this Guidance. The NRC staff believes that this standard format will reduce the time spent by both the staff in our review and by the licensee in their preparation.

of the pump and valve testing program and submittals. The standard format includes examples of relief requests which are intended to illustrate the application of the standard format and are not necessarily a specific plan relief request.

ATTACHMENT

STANDARD FORMAT

VALVE INSERVICE TESTING PROGRAM SUBMITTAL

Valve Number	Class	Coordinates	Valve Category A	B	C	D	E	Size (inches)	Valve Type	Actuator Type	Normal Position	Test Requirements	Relief Requests*	Testing Alternative	REMARKS (Not to be used for relief basis)
710	3	D-14					X	4	GA	M	LO	ET			
700	3	D-15				X		6	DE	NA	C	DT			
717	3	C-15			X			16	CK	SA	-	CV	X	CS	
702C	3	C-15			X			16	CK	SA	-	CV			
707	3	E-14			X			3	REL	SA	-	CV			
834	3	D-11		X			X	4	GL	M	C	Q	X	ET	
												MT			60 sec.
722B	3	B-11			X			3/4	REL	SA	-	SRV			
722C	3	B-11			X			3/4	REL	SA	-	SRV			
715	2	A-10			X			3	REL	SA	-	SRV			
729	2	B-10			X			3	REL	SA	-	SRV			
744B	2	D-14	X					10	GA	MO	C	Q			
												LT	X		
												MT			30 sec.

Legend for Valve Testing Example Format

Q - Exercise valve (full stroke) for operability every (3) months

LT - Valves are leak tested per Section XI Article IWV-3420

MT - Stroke time measurements are taken and compared to the stroke time limiting value per Section XI Article IWV 3410

CV - Exercise check valves to the position required to fulfill their function every (3) months

SRV- Safety and relief valves are tested per Section XI Article IWV-3510

DT - Test category D valves per Section XI Article IWV-3600

ET - Verify and record valve position before operations are performed and after operations are completed, and verify that valve is locked or sealed.

CS - Exercise valve for operability every cold shutdown

RR - Exercise valve for operability every reactor refueling

Relief Request Basis

System: Auxiliary Coolant System, Component Cooling

1. Valve: 717
   Category: C
   Class: 3
   Function: Prevent backflow from the reactor coolant pump cooling coils
   Impractical test requirement: Exercise valve for operability every three months
   Basis for relief: To test this valve would require interruption of cooling water to the reactor coolant pumps motor cooling coils. This action could result in damage to the reactor coolant pumps and thus place the plant in an unsafe mode of operation.
   Alternative Testing: This valve will be exercised for operability during cold shutdowns.

2. Valve: 834
   Category: B-E
   Class: 3
   Function: Isolate the primary water from the component cooling surge tank during plant operation. It is normally in the closed position, but routine operation of this valve will occur during refueling and cold shutdowns.
   Impractical Test Requirement: Exercise valve (full stroke) for operability every three (3) months.
   Basis for Relief: This valve is not required to change position during plant operation to accomplish its safety function. Exercising this valve will increase the possibility of surge tank link contamination.
   Alternate Testing: Verify and record valve position before and and after each valve operation.

3. Valve: 744B

Category: A

Class: 2

Function: Isolate the residual heat exchangers from the cold leg R.C.S. backflow and accumulator backflow.

Test Requirements: Seat leakage test

Basis for Relief: This valve is located in a high radiation field (2000 mr/hr) which would make the required seat leakage test hazardous to test personnel. We intend to seat leak test two other valves (875B and 876B) which are in series with this valve and will also prevent backflow. We feel that by complying with the seat leakage requirements we will not achieve a compensatory increase in the level of safety.

Alternative Testing: No alternative seat leak testing is proposed.

Enclosure 110-4

SEISMIC QUALIFICATION REVIEW TEAM (SQRT)

Interim Procedures

I. SCOPE

SQRT tasks include both generic and site specific reviews. Generic reviews cover equipment supplied by the NSSS and A/E common to more than one plant. Specific plant reviews as delineated in the Standard Review Plan Sections 3.9.2 and 3.10 will be supplemented by SQRT site visits and evaluation.

II. OBJECTIVES

SQRT is a group of NRC staff members established to conduct reviews of the design adequacy of safety related mechanical components, instrumentation and control equipment, and their supporting structures for various vibratory loads. SQRT is charged with accomplishing the following three tasks.

1. Determine the design adequacy of mechanical and electrical components and their supports for the required vibratory loading conditions which include:

   (a) Seismic

   (b) hydrodynamic (as applicable)

   (c) explosive (as applicable)

   (d) other vibratory inputs from the operating environment (as applicable)

   (e) appropriate combinations of the above events.

2. Changes in seismic qualification criteria, such as the revision of IEEE Std. 344 and other IEEE Standards, and the issuance of Regulatory Guides 1.100 and 1.89 require that the staff verify:

   (a) For older plants having components qualified by previous criteria; that components have adequate margin to perform their intended design functions during and after a seismic event.

   (b) For new plant applications; that there has been uniformity and consistency in implementing the current criteria.

3. In the case of plants which have design basis seismic ground motion levels and/or other required vibratory loads increased, review to assure adequate design margin exists at the revised levels.

III. GENERAL CRITERIA

The bases used by the staff to determine the acceptability of equipment qualification will be IEEE Std. 344-1975 as supplemented by Regulatory Guides 1.100 and 1.92, and Standard Review Plan Sections 3.9.2 and 3.10

IV. GENERAL PROCEDURES

SQRT will conduct generic and plant specific reviews:

1. Generic reviews will be conducted of all NSSS vendors and most architect engineers (major equipment vendors and testing laboratories may be included if necessary) to assure proper interpretation and implementation of the current equipment qualification criteria applied

to plants applying for construction permits and operating licenses.

2. A plant specific equipment qualification review will be conducted of each plant now undergoing licensing review having components qualified to criteria different from current requirements.

   A. For components having multi-plant application (such as those within the scope of an NSSS vendor), an equipment qualification review at specific sites will provide generic qualifications.

   B. For components which have only specific plant application (mostly those within the scope of the BOP supply), an equipment qualification review at specific sites will provide site-specific qualifications.

3. Equipment qualification review for plants with revised increased design basis seismic ground motion levels and/or other required vibratory loads will be conducted on a plant by plant basis.

V. SPECIFIC PROCEDURES

SQRT procedures provide for both generic discussion meetings and plant site visits.

1. Generic Discussion Meeting:

   To implement the generic review specified in IV.1 and IV.2.A, a generic discussion meeting will be held to discuss the following:

   A. Meeting Agenda

      Meeting Objectives by SQRT

B. NSSS or A/E personnel should be prepared to present the following information:

(1) A detailed description of current practice followed in equipment qualification, including acceptance criteria, methods, and procedures used in conducting testing and analysis. Present and discuss the equipment qualification program on certain specified items (i.e., pumps, valves, diesel generators, motors, bistable units, relays, electrical cabinets, etc.)

(2) Information regarding administrative control of equipment qualification, especially the handling of interface problems, documentation, and internal review procedures.

(3) Identifying the scope of their suppliers. A list of equipment should be made available if possible prior to the meeting.

C. For the cases specified in IV.2.A, methods and procedures for conducting equipment qualification review are discussed, including selection of plants for site visits and setting up a tentative schedule for such visits.

D. Discuss necessary documentation.

E. Inspect testing facilities, if any. Testing capability, format of testing reports, wave forms of shaker table motions, and monitoring and control devices are the major items for inspection.

F. SQRT concludes the meeting and specifies the follow-up items.

2. Plant Site Reviews:

To implement plant specific equipment qualification reviews specified in IV.2 above, on-site inspection of equipment and supporting structures in question is required. Site visits generally follow the following procedures:

A. Pre-visit information submission:

The applicant (plant owner) receives initial information concerning the intended visit, and should subsequently submit two summary equipment lists (one for NSSS supplied equipment and one for BOP supplied equipment). These lists should include all safety related mechanical components, instrumentation, and control equipment, including valve actuators and other appurtenances of active pumps and valves. In the lists, the following information should be specified for each item of equipment:

(1) Method of qualification used:

(a) Analysis or test

(b) If by test, describe whether it was a single or multi-frequency test and whether input was single axis or bi-axial

(c) If by analysis, describe whether static or dynamic,

single or multiple-axis analysis was used. Present natural frequency of equipment.

(2) Indicate whether the equipment is required for:

(a) hot stand-by

(b) cold shutdown

(c) both

(d) neither

The scenario to be considered for this determination is:

(i) SSE or OBE, with coincident

(ii) loss of offsite power, and

(iii) assumption of any single active failure.

(3) Location of equipment, i.e., building, elevation.

(4) Availability for inspection (Is the equipment already installed at the plant site?)

(5) Provide a description of how cold shutdown is reached using the equipment in item (2) above.

B. SQRT screens the above information and decides which items will be evaluated during our forthcoming site visit. The applicant

will be informed of these items and will be expected to submit two weeks prior to the visit an equipment qualification summary as shown on pages 10-12 for each of the selected items.

C. A brief meeting is held at the beginning of a site visit with the following agenda:

(1) SQRT explains the objectives of the site visit and procedures to conduct equipment inspection.

(2) Utility personnel or their designees present an overview of the seismic qualification program conducted.

(3) The seismic qualification of certain specified items may be discussed as necessary.

(4) SQRT specifies items that need to be inspected.

D. SQRT conducts inspection of specified items.

E. SQRT describes findings of the inspection.

F. General discussion.

G. SQRT concludes the visit and specifies needed information and the follow-up actions.

3. After each visit SQRT will issue a trip report, which identifies findings, conclusions and follow-up items. Status reports may be issued as necessary. The site review will include the issuance of

an Evaluation Report for the specific plant. Generic evaluations will be referenced to the NSSS vendor or A/E.

VI. RESPONSIBILITIES OF NRC PARTICIPANTS:

A. The Seismic Qualification Review Team consists of members of the Mechanical Engineering Branch (MEB), the Instrumentation and Control Systems Branch (ICSB), and the Power Systems Branch (PSB). One additional member from MEB will join the team when a review of a specific plant is going to be conducted. This member will be the reviewer of the plant.

The Team Leader is responsible for scheduling actions, coordinating staff positions, and contacting appropriate authorities for work assignments to each member. He reports to the MEB Branch Chief regarding the progress of SQRT performance. He will set up necessary contacts for generic reviews and will contact project management for specific plant site visits. He will specify the meeting objectives and concludes meetings.

The MEB members and Team Leader are responsible for reviewing assigned equipment qualifications in the area of responsibility of the Mechanical Engineering Branch, including the methods and procedures used in test and analysis.

Members representing the Power Systems Branch (PSB) and the Instrumentation & Control Systems Branch (ICSB) are responsible for reviewing assigned equipment qualification in the area of responsibility of

their branch, including equipment signal interpretations for functional verification. They serve as a liaison between SQRT and ICSB and PSB.

All members shall present their opinion and professional judgement to the Team Leader in order to arrive at consistent and uniform SQRT positions.

B. The MEB, PSB, and ICSB project reviewers will be advised of SQRT activities which relate to specific plants. The MEB project reviewer is responsible for evaluating the impact of SQRT activity on the specific plant review and for taking appropriate action to include pertinent information in the plant safety evaluation. The MEB project reviewer is expected to participate in the site visit and attend pertinent generic meetings as necessary.

The DPM project manager, after being informed of the intended plant visit, is expected to contact the applicant and arrange for the visit. The project manager serves as a liaison between the SQRT and the applicant.

C. Generic meetings will be arranged by the SQRT or via the DPM generic project manager if one is assigned.

D. Representatives from I&E Regional Offices and other interested organizational groups within NRC are welcome to attend either generic meetings or plant site visits as observers. The SQRT should be informed of expected attendance at such meetings or site visits.

Qualification Summary of Equipment

I. Plant Name: [redacted] Type:

1. Utility: _____ PWR _____

2. NSSS: _____ 3. A/E: _____ BWR _____

II. Component Name [redacted]

1. Scope: [ ] NSSS [ ] BOP

2. Model Number: _____ Quantity: _____

3. Vendor: _____

4. If the component is a cabinet or panel, name and model No. of the devices included: _____

_____

5. Physical Description a. Appearance _____

b. Dimensions _____

c. Weight _____

6. Location: Building: _____

Elevation: _____

7. Field Mounting Conditions [ ] Bolt (No. _____, Size _____)
[ ] Weld (Length _____)
[ ] _____

8. Natural Frequencies in Each Direction (Side/Side, Front/Back, Vertical):

S/S: _____ F/B: _____ V: _____

9. a. Functional Description: _____

_____

b. Is the equipment required for [ ] Hot Standby [ ] Cold Shutdown

[ ] Both _____

10. Pertinent Reference Design Specifications: _____

_____

III. Is Equipment Available for Inspection in the Plant: [ ] Yes [ ] No

IV. Equipment Qualification Method: Test: _____

Analysis: _____

Combination of Test and Analysis: _____

Test and/or Analysis by _____
(name of Company or Laboratory & Report No.)

V. Vibration Input:

1. Loads considered: 1.[ ]Seismic only 2.[ ]Hydrodynamic only 3.[ ]Explosive only

   4.[ ] Other (Specify)_____ 5.[ ] Combination of_____

   6. Method of combining RRS: [ ] Absolute Sum [ ] SRSS [ ]_____
   (other, specify)

2. Required Response Spectra (attach the graphs):_____ _____

3. Required Acceleration in Each Direction:

   S/S =_____ F/B =_____ V =_____

VI. If Qualification by Test, then Complete:

1. [ ] Single Frequency [ ] Multi-Frequency: [ ] random [ ] sine beat [ ]_____

2. [ ] Single Axis [ ] Multi-Axis

3. No. of Qualification Tests: OBE_____ SSE_____ Other_____
   (specify)

4. Frequency Range:_____

5. TRS enveloping RRS using Multi-Frequency Test [ ] Yes (Plot TRS on RRS graphs) [ ] No

6. Input g-level Test at S/S = _____ F/B = _____ V = ____

7. Laboratory Mounting:

   1. [ ] Bolt (No._____, Size_____) [ ] Weld (Length_____) [ ]_____

8. Functional operability verified: [ ] Yes [ ] No [ ] Not Applicable

9. Test Results including modifications made:_____

   _____

10. Other tests performed (such as fragility test, including results):_____

    _____

VII. If Qualification by Analysis or by the Combination of Test and Analysis, then Complete:

1. Description of Test including Results: _____

_____

_____

2. Method of Analysis:

[ ] Static Analysis [ ] Equivalent Static Analysis

[ ] Dynamic Analysis: [ ] Time-History
[ ] Response Spectrum

3. Model Type: [ ] 3D [ ] 2D [ ] 1D

[ ] Finite Element [ ] Beam [ ] Closed Form Solution

4. [ ] Computer Codes: _____

Frequency Range and No. of modes considered: _____

[ ] Hand Calculations

5. Method of Combining Dynamic Responses: [ ] Absolute Sum [ ] SRSS
[ ]Other: _____
(specify)

6. Damping: _____ _____ Basis for the damping used: _____

7. Support Considerations in the model: _____

8. Critical Structural Elements:

A. Identification	Location	Governing Load or Response Combination	Seismic Stress	Total Stress	Stress Allowable

B. Max. Deflection	Location	Effect Upon Functional Operability

231.0 REACTOR FUELS

231.1 (4.2.1.1.1.4) Throughout Section 4.2, Reference 4 is applied for various analysis. The section notations for reference 4, as given in the FSAR, appear to have added a prefix of 4, thus are in error.

231.2 (4.2.1.2.1.8) With regard to fuel rod internal pressure, discuss the conservatisms that are applied for determining the available gas retention volume and temperature. Provide calculated values with the new 8x8 design (0.483 O.D.) for both average and maximum internal pressure and fission gas release fractions as a function of irradiated exposure. Provide the exposure history used in each calculation and discuss why the history is appropriate for average/maximum results.

231.3 (4.2.1.2.1.8) NRC has questioned the validity of fission gas release calculations in most fuel performance codes including GEGAP-III for a burnup greater than 20,000 MWd/tU. General Electric was informed of this concern on November 23, 1976 and was provided with a method of correcting gas release calculations for burnups greater than 20,000 MWd/tU. Since there was no question of the adequacy of GEGAP-III for burnups below 20,000 MWd/tU, the Grand Gulf 1 and 2 calculations are acceptable for operation early in life until the peak local burnup reaches 20,000 MWd/tU. For burnups in excess of that value, GEGAP-III calculations (and other affected analyses) must be redone using the correction method mentioned above or such modified methods that might be submitted by Mississippi Power and Light or General Electric and approved by the NRC.

231.4 (4.2.1.2.1.9) Provide a reference for thermal-hydraulic tests with rod-to-rod deflection clearance of <0.060 inches.

231.5 (4.2.1.2.1.15) Routine fuel surveillance is discussed in paragraphs I.D and II.D of Section 4.2 (Revision 1) of the Standard Review Plan. Post irradiation examination of fuel is discussed in the FSAR. Please refer to the document and submit a description of the on-line fuel rod failure detection method.

231.6 (4.2.1.2.2.5) For control rods with B_4C granules in stainless steel, provide the following information:

(1) Intended lifetime of a control rod in terms of neutron captures per cubic centimeter and years of in-core residence.

(2) The internal gas pressure of the control rods at EOL.

231.7 (4 2.3.1.1) In Section 4.2.3.1.1, Section 11 of Reference 4 is noted. The data in Reference 4 are derived for the old 8x8 fuel rod design. Provide the data as given in Tables 11-8, 11-9, 11-10, 11-11, 11-12 and 11-13 for the new 8x8 design.

231.8 (4.2.3.1.2) Provide the LHGR values for UO_2 - Gd_2O_3 fuel as a function of irradiation exposure that give a 1% plastic diametral strain. Provide the exposure history used and show why it is appropriate for the calculation requested.

231.9 (4.2.3.2.10) Provide the incipient center melting parameters, i.e., integral to melt and LHGR, for UO_2 - Gd_2O_3 fuel. Provide the exposure history used and show why it is appropriate for the calculation requested.

231.10 (4.2.3.2.12) Reference 19 is used to describe the behavior of fuel rods in the event of coolant flow blockage. This report was not accepted by NRC in September 1971 on the basis that the information was inadequate. Provide an updated reference for this event.

231.11 (4.2.3.2.13) Reference 20 for Section 4.2.3.2.13 appears to be in error. Provide a brief summary of the fixed-beam tests for channel evaluation and the correct reference for this section.

231.12 (4.2.3.2.15) Loads from seismic and LOCA events including asymmetric blowdown loads are being reviewed by NRC for both PWRs and BWRs. The results of an analysis that show that the fuel assemblies and channel boxes can withstand these phenomena and that coolable geometry is maintained should be provided. This analysis should be performed using state-of-the-art methodology and criteria for the Grand Gulf 1 and 2 design. Grand Gulf should further agree to perform a reanalysis should the criteria developed from generic task A-2 (see NUREG-0371, November 1978) warrant such evaluation.

232.0 REACTOR PHYSICS

232.1 (4 .2.1) The test indicates three different enrichment distributions in fuel bundles of which one is entirely natural uranium. Figure 4.3.1 shows four different enrichments (two different "high" enrichments). Please clarify.

232.2 (15.4.9.2) Is the reference to the RSCS and RWM here correct? It is our impression that these have been replaced by the RPCS. See also Figure 15 A.6-17. Please clarify.

232.3 (15.4.2.2) Provide a summary description of the operations of the Rod Withdrawal Limiter or reference the section of the FSAR where one exists. After selecting and withdrawing a rod to its limit, what restrictions are enforced relative to the next selection and withdrawal of the same rod?

232.4 (15.4.2.2) Discuss the effect of rod drift on the rod withdrawal transient. Does the alarm occur in time to permit the operator to take action before limits are reached?

232.5 (15.4.2.2) Comment on the consequences of a rod withdrawal transient starting at powers below 70 percent where there are no restrictions on rod motion. What is the action of the Rod Withdrawal Limiter when the 70 percent power level is reached during the transient?

232.6 (15.4.2.2) Discuss the analysis which establishes the value of MCPR to be assumed as the starting point for the rod withdrawal transient at 70 percent power (or reference the appropriate section in the FSAR).