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

OCT 13 1978

Docket Nos. 50-329
50-330

Mr. S. H. Howell, Vice President
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Howell:

SUBJECT: REVISED SCHEDULE AND SUPPLEMENTAL REQUESTS FOR ADDITIONAL INFORMATION: PART 2

In continuing our review of the FSAR for Midland Plant Units 1 & 2, we find that insufficient information has been provided for our review to proceed with development of staff positions which had been scheduled for issuance prior to this time.

The first part of our supplemental requests for information which we require for developing our positions was provided by our letter of August 30, 1978. The second part of our requests is provided by Enclosure 1.

We have assessed the status of our review in conjunction with some existing limitations on staff manpower resources. Our revised schedule for Midland is provided in Enclosure 2. The revised schedule is generally consistent with our preliminary schedule assessment during our meeting of August 31, 1978.

Please contact us if you desire clarification or other discussions of these or previous information requests.

Sincerely,

Steven A. Varga, Chief
Light Water Reactors Branch No. 4
Division of Project Management

Enclosures:
As stated

cc: Listed on page 2

8006250 398

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## ENCLOSURE 1

## SUPPLEMENTAL REQUEST FOR ADDITIONAL INFORMATION (Q1's)

## PART 2

## MIDLAND PLANT UNITS 1 & 2

These requests for additional information are numbered such that the three digits to the left of the decimal identify the technical review branch and the numbers to the right of the decimal are the sequential request numbers. The number in parenthesis indicates the relevant section in the Safety Analysis Report. The initials RSP indicate the request represents a regulatory staff position.

Branch Technical Positions referenced in these requests can be found in "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-75/087.

022.0 CONTAINMENT SYSTEMS BRANCH

022.27
(6.2)
The response to request 022.4 regarding the containment sump does not provide sufficient information relative to alternative approaches considered. Provide the following information to justify deviating from our position in Regulatory Guide 1.82:

1. Our position in paragraph C.1 of Regulatory Guide 1.82 states that two sumps should be provided to serve the ECCS and CSS. Some other plants at your advanced stage of construction have utilized a vertically placed plate or screen in the sump design to physically partition the sump and assure a continuation of recirculation flow in the event a portion of the sump structure screening is damaged. The plate or screen is designed to minimize vortex formation. Discuss the technical consideration given to this alternative to comply with our guide's position. Also submit detail and arrangement drawings which show your sump structure relative to suction piping and potential for partition.

2. Our position in paragraph C.7 of Regulatory Guide 1.82 states that the coolant velocity at the inner screen should be approximately 0.2 ft/sec, assuming 50% blockage of the screen area. FSAR Section 6.2.2.1.2.2 states that during maximum flow conditions, the velocity of recirculated fluids reaching the inner screen is 0.5 ft/sec. Specify and justify the screen blockage assumed in determining your velocity and provide justification for any increase in velocity.

022.28
(6.2)
(7.3)
RSP
Your response to request 022.9, regarding diversity of parameters sensed for the initiation of containment isolation, is unacceptable. It is our position that each automatic containment isolation valve be capable of actuating from a diversity of parameters being sensed; e.g., each isolation valve that actuates on RBIS-I (which currently senses only containment high pressure) should be capable of actuating on either containment high pressure, low pressurizer level, or some other diverse parameter. Modify your design and FSAR discussion to comply with this position.

022.29
(6.2)
We find that your maximum external containment pressure due to the inadvertent actuation of the spray systems has not been determined in a sufficiently conservative manner. A more appropriate calculation would assume that the containment, which is initially at the conditions stated in FSAR Table 6.2-7, becomes saturated at the minimum BWST temperature of 40°F. Therefore, revise your external containment pressure analysis using these assumptions.

022.30 (6.2) The following request is made in addition to (and not in lieu of) request 110.46. Your response to request 022.19 states that the reactor cavity analyses sought by requests 022.2 and 022.18 are to be provided in a future amendment. Your response should also include the following:

1. The reactor cavity analyses of FSAR Section 6.2 assumes a cold leg break located in the shield wall piping penetration. It does not appear that hot and cold leg guillotine breaks at the reactor vessel terminal ends (as postulated in FSAR Section 3.6) were assumed for the reactor cavity analyses. We require that these breaks be analyzed to determine the worst case in calculating peak pressures, load, and moments in the reactor cavity analyses.

2. The reactor cavity nodalization drawings referenced in FSAR Section 6.2.1.2.3.2 are not adequate to verify that all physical restrictions and obstructions have been properly nodalized. Provide drawings as discussed in request 022.2(d).

3. Clarify how insulation is treated for the subcompartment analysis: Your response to request 022.2(g) states that insulation is assumed to stay in place, but FSAR Section 6.2.1.2.2.1 states that insulation in the reactor cavity is assumed to drop to the bottom of the cavity and block flow paths in this region.

4. FSAR Section 6.2.1.2.2.1 discusses a shield plug located on top of the reactor cavity. Discuss the potential for the shield plug becoming a missile during the reactor cavity transient. In addition, provide drawings which show detailed views of the shield plug, including its arrangement relative to surrounding structures.

022.31 (9.4) (RSP) State the operating modes in which you plan to permit the reactor building purge system to be operated, particularly in regard to use of the 48-inch lines of the system. It is our position that if this system is to operate during the startup, normal operation, hot standby or hot shutdown, it should meet the provisions of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

022.32 (6.8) RSP Your response to request 022.2, concerning the liner plate weld channel pressurization system is unacceptable. It is our position that if the liner plate weld channel pressurization system detects leakage which is greater than the containment design leak rate, repairs must be made to reduce the leak rate below design before resuming power operation. Modify your position and discuss your plans to comply with our position.

022.33 It is our position that the penetration pressurization system
(6.8) shall not be used during the containment integrated leak
RSP rate tests (CILRT). The CILRT should be performed with the accident differential pressure existing across the containment isolation barriers. Similarly, it is our position that the liner plate weld channels be vented to the containment atmosphere during the CILRT. Discuss your intentions to comply with these positions.

022.34 FSAR Section 6.2.6.1.2 states that the decay heat removal
(6.2) system, plant heating system, and essential service water
RSP systems will not be vented and drained for the containment integrated leak rate tests (CILRT) because these systems are needed to maintain the plant in a safe condition. However, III.A.1(d) of Appendix J, 10 CFR 50 requires that the isolation valves of these systems be locally Type C tested. It is our position that the leak rates of these valves must be added to the CILRT results prior to determining the acceptability of the CILRT. Discuss your intentions to comply with this position.

022.35 We disagree with the proposed procedures in Section 6.2.6.1.3
(6.2) to be used when repairs must be made to satisfy the acceptance
(RSP) criteria of a containment integrated leak rate test (CILRT). It is our position that differences in the post-repair leakage rates of affected components shall not be subtracted from the final integrated leakage rates. In addition, our position has been developed to preclude the necessity for total depressurization of the containment during the course of CILRT involving repairs.

If, during the performance of a CILRT, leakage occurs through testable penetrations or isolation valves to the extent that it could interfere with satisfactory completion of the test or result in the CILRT not meeting the acceptance criteria, the leak paths may be isolated and the Type A test continued until completion. Only containment penetrations which are designed to permit local leak testing may be isolated during a Type A test.

Local leak rates measured before and after each repair must be reported, and the <u>sum</u> of (1) the total post-repair leak rates plus (2) the upper 95% confidence limit of the overall containment leak rate, must satisfy the acceptance criterion for the CILRT. If this sum fails to satisfy the acceptance criterion for the CILRT, then the CILRT shall be repeated

022.35 (cont'd) to further identify leak paths that are contributing to the inability to satisfy the acceptance criterion for the CILRT. We emphasize that the difference in the local leak rates measured before and after repair may not be deducted from the upper 95% confidence limit of the overall containment leak rate in order to satisfy the acceptance criterion for the CILRT.

Modify your proposed procedures accordingly.

022.36 (6.2) Provide the following information concerning the penetration pressurization system:

1. Paragraph III.C.3(a) of Appendix J to 10 CFR Part 50 states that isolation valves in a seal system must demonstrate lower leakage rates than those specified in the technical specifications or associated bases. Provide the maximum allowable leak rates and discuss how the individual isolation valve leak rates will be quantified.

2. Paragraph III.C.3.(b) of Appendix J to 10 CFR 50 states that the minimum pressure of a seal system shall be 1.1 Pa (i.e., 77 psig for the Midland Plants). Either correct or justify your proposed minimum pressure of 73.7 psig.

022.37 (6.2) Your ECCS minimum containment backpressure calculation references topical report BAW-10103. Table B-1 of this topical report gives a delay time to initiate containment sprays of 35 seconds, but FSAR Table 6.2-15 states that the sprays are initiated at 88 seconds. State the minimum time necessary to initiate containment sprays and state the time which was used for your ECCS backpressure calculation. Clarify and justify your assumptions in terms of availability of offsite power, transient sensing and signal processing pump startup, and line sweepout.

022.38 (6.2) Describe and justify the analytical model which you used to determine the maximum containment temperature and pressure for a spectrum of postulated main steam line breaks for various reactor power levels. Include the following in your discussion:

022.38 (cont'd)

1. Provide a single active failure analysis which specifically identifies those safety grade systems and components relied upon to limit the mass and energy release and containment pressure/temperature response. The single failure analysis should include, but not necessarily be limited to, main steam and connected systems isolation; main feedwater, auxiliary feedwater, and connected systems isolation; main feedwater, condensate, and auxiliary feedwater pump trips; the loss of or availability of offsite power; diesel failure when loss of offsite power is evaluated; and partial loss of containment cooling systems. Justify reliance on any equipment which is nonsafety grade in whole or in part.

2. Discuss and justify your assumptions as to the time at which active containment heat removal systems become effective.

3. Discuss and justify the heat transfer correlation(s) (e.g., Tagami, Uchida) used to calculate the heat transfer from the containment atmosphere to the passive heat sinks. Provide a graph of the heat transfer coefficient versus time for the most severe steam line break accident analyzed.

4. Specify and justify the temperature used in the calculation of condensing heat transfer to the passive heat sinks; (In other words, specify whether you used the saturation temperature corresponding to the partial pressure of the vapor, or the atmospheric temperature which may be superheated, and justify your selection).

5. Discuss and justify your analytical model, including the thermodynamic equations, used to account for the removal of the condensed mass from the containment atmosphere due to condensing heat transfer to the passive heat sinks.

6. Provide a table of the peak values of containment atmospheric temperature and pressure for the spectrum of break areas and power levels analyzed.

7. For the case which results in the maximum containment atmospheric temperature, graphically show as a function of time the containment atmospheric temperature, the containment liner temperature, and the containment concrete temperature.

8. For the case which results in the maximum containment pressure, graphically show the containment pressure as a function of time.

9. Specify and justify the design temperature of the containment structure liner and concrete, the design temperature of the internal concrete structures, and the temperature used to qualify the safety-related instrumentation located within the containment.

022.39 (6.2) Your response to request 022.1, concerning the environmental qualification of safety related equipment is incomplete. The response assumes that all safety related equipment can be modeled as a carbon steel slab with a thickness of 1/4 to 1/8 inch. Justify that this model is conservative for all safety related equipment which would experience the environment resulting from a postulated main steam line break (MSLB).

We require the following information describing the component thermal analyses performed as part of your environmental qualification program. Each component required to function during or following the MSLB should be addressed explicitly.

1. Provide external and sectional diagrams of each component analyzed, showing principal dimensions, materials of construction, and cross-sections modeled for analyses.

2. Provide a detailed description of each thermal model, indicating basic assumptions and showing the model mock-up with principal dimensions, materials, and material thermal properties.

3. Perform the analyses using the correlation provided in the attached CSB Interim Evaluation Model.

4. Identify the specific point on the component which was analyzed and justify that this location is the most critical or conservative with regard to potential component failure.

ATTACHMENT TO REQUEST 022.39

CSB Interim Evaluation Model
Environmental Qualification for Main
Steam Line Break Inside Containment
(Operating License Applicants Only)

Analyses of main steam line break (MSLB) accidents inside PWR dry-type containments have predicted temperature transients which exceed the qualification temperature of some safety related equipment. As a result there is a concern regarding the capability of this equipment to survive such an event to assure safe plant shutdown. This concern is related to Issue 25 of NUREG-0153 dated September, 1976.

The NRC has identified this matter as a Category A Technical Safety Activity and is currently pursuing a program to resolve this concern. In the meantime it is required that you perform an evaluation of the containment environmental conditions associated with a MSLB accident as well as a LOCA and justify that the essential equipment needed to mitigate these accidents have been adequately qualified.

Since the NRC generic effort on this concern is still in progress, we are providing the analytical assumptions which are acceptable for the interim period. These models and assumptions are acceptable for the spectrum of MSLB accidents.

1. Containment Environmental Response
   a. Heat transfer coefficient to heat sinks.
      The Uchida heat transfer correlation (data) should be used while in the condensing mode. A natural convection heat transfer

coefficient should be used at all other times. The application of these correlations should be as follows:

(1) Condensing heat transfer

$q/A = h_u \cdot (T_s - T_w)$

where $q/A$ = the surface heat flux

$h_u$ = the Uchida heat transfer coefficient

$T_s$ = the steam saturation (dew point) temperature

$T_w$ = surface temperature of the heat sink

(2) Convective heat transfer

$q/A = h_c \cdot (T_v - T_w)$

where $h_c$ = convective heat transfer coefficient

$T_v$ = the bulk vapor temperature.

All other parameters are the same as for the condensing mode.

b. Heat sink condensate treatment

When the containment atmosphere is at or below the saturation temperature, all condensate formed on the heat sinks should be transferred directly to the sump. When the atmosphere is superheated a maximum of 8% of the condensate may be transferred to the vapor region. The revaporization should be calculated as follows:

$$M_r = X \cdot q / (h_v - h_L)$$

where $M_r$ = revaporization rate

$X$ = revaporization fraction (0.08)

$q$ = surface heat transfer rate

$h_v$ = enthalpy of the superheated steam

$h_L$ = enthalphy of the liquid condensate entering the sump region (i.e., average enthalpy of the heat sink condensate boundary layer)

c. Heat sink surface area

The surface area of the heat sinks should correspond to that used for the containment design pressure evaluation.

d. Single active failure evaluation

Single active failures should be evaluated for those containment safety systems and components relied upon to limit the containment temperature/pressure response to a MSLB accident. This evaluation should include, but not necessarily be limited to, the loss or availability of offsite power (whichever is worse), diesel generator failure when loss of offsite power is evaluated, and loss of containment heat removal systems (either partial or total).

e. Containment heat removal system actuation
The time determined at which active containment heat removal systems become effective should include consideration of actuation sensors and setpoints, activation delay time, and system delay time (i.e., time required to come into operation).

f. Identification of most severe environment
The worst case for environmental qualification should be selected considering time duration at elevated temperatures as well as the maximum temperature. In particular, consider the spectrum of break sizes analyzed and single failures evaluated.

2. Safety Related Component Thermal Analysis
Component thermal analyses may be performed to justify environmental qualification test conditions less than those calculated during the containment environmental response calculation. The thermal analysis should be performed for the potential points of component failure such as thin cross sections and temperature sensitive parts where thermal stressing, temperature-related degradation, steam or chemical interaction at elevated temperatures, or other thermal effects could

result in failure of the compartment electrically or mechanically. The heat transfer rate to components should be calculated as follows:

a. Condensing heat transfer rate

$q/A = h_{cd} \cdot (T_s - T_w)$

where $q/A$ = component surface heat flux

$h_{cd}$ = condensing heat transfer coefficient

= the larger of 4x Tagami Correlation or 4x Uchida Correlation

$T_s$ = saturation temperature (dew point)

$T_w$ = component surface temperature

b. Convective heat transfer

A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period, a forced convection heat transfer correlation should be used. For example:

$NU = C\ (Re)^n$

where Nu = Nusselt No.

Re = Reynolds No.

C,n = empirical constants dependent on geometry and Reynolds No.

The velocity used in the evaluation of Reynolds number may be determined as follows:

$$V = 25 \frac{M_{BD}}{V_{CONT}}$$

where $V$ = velocity in ft/sec

$M_{BD}$ = the blowdown rate in lbm/hr

$V_{CONT}$ = containment volume in $ft^3$

After the blowdown has ceased or reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable. However, use of a natural convection heat transfer coefficient must be fully justified whenever used.

3. Evaluation of Environmental Qualification

The component peak surface temperature(s) ($T_{cs}$) should be computed using items 1 and 2 above. The component qualification temperature ($T_{cq}$) should be determined from the actual environment test conditions. Where components have been "bathed" in a saturated steam or steam/air environment for extended periods (e.g., 10 minutes), the qualification temperature is the test chamber temperature. For components subjected to test conditions substantially removed from the steam saturation point or for short durations (e.g., less than 10 minutes), the qualification temperature must be justified by experimental thermocouple readings on the component surface or analyses which minimizes the heat flux to the component.

If the component surface temperature, $T_{cs}$, is less than or equal to the component qualification temperature, $T_{cq}$, the component may be considered qualified for an MSLB environment during the interim period.

If the component surface temperature is greater than the qualification temperature, then (a) provide additional justification that the component can operate in environments equal to or greater than that which would result in the calculated peak surface temperature, or (b) provide a requalification package for the component, or (c) provide appropriate protection to assure that the component will not experience a surface temperature in excess of the qualification temperature, $T_{cq}$.

022.40 (6.2) Your response to request 022.16, regarding local Type C testing of containment isolation valves, is incomplete and unacceptable. We disagree with your assumption that penetrations associated with the secondary system (main steam, feedwater, auxiliary feedwater, etc.) do not provide credible leak paths for the leakage of containment atmosphere out of containment. Primary-to-secondary steam generator tube leakage provides a potential leak path for containment atmosphere out of containment following a loss of coolant accident. Therefore, justify not performing local Type C tests on secondary system containment isolation valves by either:

1. Showing a water seal exists which precludes containment atmospheric leakage as discussed in request 022.16; or

2. Providing a calculation which conservatively predicts the offsite dose attributed to containment atmospheric leakage through the steam generator tubes. Identify the dose contribution due to leakage and show that the dose contribution in addition to the offsite accident dose, does not exceed the exposure guidelines of 10 CFR Part 100.

031.0 INSTRUMENTATION AND CONTROL SYSTEMS BRANCH

031.11 (1.2.2) Section 1.3.2 of your FSAR does not satisfy the intent of Revision 2 of Regulatory Guide 1.70. Your FSAR states that "due to extensive reformating and the additional information provided in the FSAR, cross-reference of changes is not considered appropriate and is therefore not included." We do not agree that this information should be omitted.

The purpose of this section is to identify all significant changes from the original design which we approved during the construction permit review. We require that your FSAR describe all significant changes from the construction permit design and identify the FSAR location where the revised design is described. The description should include the basis for the change.

This section should also provide assurance that the Midland units have not been constructed to any safety criteria that are less conservative than those to which you committed and which we approved during the review for the construction permits.

Amend your FSAR to reflect these requirements.

031.12 (3.11) Your FSAR does not provide all of the information specified by Section 3.11 of Regulatory Guide 1.70 and our Standard Review Plan, NUREG-75/087. Notable examples are:

1. All safety related equipment should be qualified to perform its function under all expected environmental conditions. These environmental conditions are not limited to an accident environment such as that inside of containment during a LOCA.

   Some of the tests discussed in Table 3.11-4 of your FSAR indicate that environmental qualification is not required when there is no extreme environment such as that produced by an accident. Qualification is required even though the environmental envelope does not include these extreme conditions. Clarify such areas in your FSAR accordingly.

2. Where Heating, Ventilation, and Air Conditioning (HVAC) are relied on to control the environment of safety related equipment within the envelope to which such equipment is qualified, these HVAC systems must meet at least one of the following requirements:

(a) The HVAC must be designed and qualified to meet all requirements of a safety related system, or,

(b) The control room should receive an alarm when the acceptable temperature range has been exceeded. This alarm should be provided by instrumentation which:

(b.1) is of high quality

(b.2) is checked periodically to verify its functional capability by plant technical specfication requirements, and

(b.3) is powered from a continuous power source or is redundant with separate channels and power sources.

Also, the operator should have a method of obtaining a continuous record of the temperature during the time that the temperature range is exceeded.

Applicants are also required to report the occurrence of the temperature exceeding the equipment qualification range as an abnormal occurrence to the NRC. In addition, the applicants are required to provide the results of an analysis to demonstrate that the excess temperature has not degraded the involved Class 1E equipment below an acceptable level for continued plant operation.

In either a or b above, we require applicants to demonstrate the capability of the environmental control system to prevent degradation of redundant Class IE equipment beyond the point where the safety function cannot be accomplished within the time required.

We require that you address this concern in the qualification program for all equipment which relies on HVAC systems for environmental control.

3. Several places in Section 3.11 of your FSAR indicate that information will be available later. We require this information to complete our review.

031.13 (3.11) With the concerns of request 031.12 in mind and in order to ensure that your environmantal qualification program conforms with General Design Criteria 1, 2, 4, and 23, with Sections III and XI of Appendix B to 10 CFR Part 50, and with the national standards identified in Standard Review Plan Section 3.11, Part II "Acceptance Criteria" (which includes IEEE 323), the following information on your qualification program is required for all Class IE equipment :

I. <u>On the Basis of One Item per Specified [1] Group</u>

1. Identification of Equipment including,

   a. Manufacturer
   b. Manufacturer's type number
   c. Manufacturer's model number

2. Equipment design specification requirements, including,

   a. The system function requirements
   b. An environmental envelope which includes all extreme conditions, both maximum and minimum, expected to occur during plant shutdown, normal operation, and abnormal operation including any design basis event.
   c. Time required to fulfill its function when subjected to any of the extremes of the environmental envelope specified above.

3. Test plan,

4. Test set-up,

5. Test procedures,

6. Acceptability goals and requirements,

7. Test results

8. Identification of the documents which include and describe the above items.

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1/ The above information shall be provided to us for at least one item in each of the following groups of Class IE equipment.

a) Switchgear
b) Motor control centers
c) Valve operators (in containment)
d) Motors
e) Logic equipment
f) Cable
g) Diesel generator Control equipment
h) Sensors
i) Limit switches
j) Heaters
l) Control Boards
m) Instrument racks and panels
o) Penetrations - including design provisions for the overcurrent protection circuits, and
p) Splices

II. Remaining Equipment

In accordance with the requirements of Appendix B to 10 CFR Part 50, we also require a statement verifying:

1. That all remaining Class 1E equipment has been qualified in accordance with the program described above, and

2. That the qualification information for this equipment is available for an NRC audit.

031.14 (7.1.2.2) (8.3.1.4) (3.11) FSAR Section 7.1.2.2 discusses independence of redundant safety related instrumentation and control systems. We request the following additional information:

1. Identify each type of device used to isolate Class 1E circuits from non-Class 1E circuits.

2. Describe the method used to qualify each type of isolator.

3. Provide a summary of the results of the qualification program for each type of isolator.

4. Describe the power supplies for each type of isolator. This description should demonstrate that Class 1E power supplies will not be degraded by the isolators and that non-Class 1E power supplies will not degrade any Class 1E circuit.

031.15 (7.1.1) FSAR Table 7.1-1 lists the pressurizer heat controls and the decay heat removal isolation valve interlock as safety related instrumentation and control systems supplied by the NSSS vendor. Your FSAR does not describe how these systems will be environmentally qualified. Describe your associated qualification program, including criteria used to qualify these systems and all of their components and equipment.

031.16 (7.1.1) FSAR Table 7.1.1 identifies other plants with similar safety related instrumentation and control systems. This table, however, does not identify the differences between the Midland designs and the designs of the other plants. Nor does it discuss differences and their effects on safety related systems. Provide this information in accordance with Section 7.1.1 of Regulatory Guide 1.70.

031.17 (7.1.2) FSAR Section 7.1.2.5 takes exception to our Branch Technical Position ICSB 4. We disagree with your exception and require that items 1 and 4 therein be fully implemented in the Midland designs. Valve lock-out as provided in the response to satisfy Branch Technical Position ICSB 18 is intended to assure availability of the core flooding system during normal operations. Branch Technical Position ICSB 4 is intended to insure availability of the core flooding systems during other times such as startup, when pressurizing the main coolant system, and during power operations when the core flooding system is isolated (as allowed by the technical specifications) for short periods of time.

Modify your design to satisfy all the requirements of Branch Technical Position ICSB 4.

031.18 (7.1) (App 3A) Your conformance to the recommendations of Regulatory Guide 1.52, as discussed in Appendix 3A of the FSAR, is unacceptable. Your FSAR states that compliance to the applicable IEEE Standards is not known. It also states that the qualification program for the electrical components will be provided when available.

Provide a discussion of your conformance to the recommendations of Regulatory Guide 1.52, including position C.2.h. Identify and justify all exceptions.

031.19 (7.1) Your FSAR states that compliance to Branch Technical Position ICSB 24 will be discussed in a later amendment. We request that you expedite this submittal consistent with our established review schedule.

031.20 (7.1) In order to ensure that implementation of your separation criteria is acceptable, we require that the following information :

(Appendix 3A)

1. Verify that the two diesel generator synchronizing circuits are the only Class 1E to non-Class 1E circuits that will be analyzed and not make use of the isolation cabinets.

2. Identify and describe each type of device used to isolate the Class 1E circuits from the non Class 1E circuits.

3. Provide a description of the qualification program used to demonstrate that each type of isolator will prevent degradation of the Class 1E circuits. This description should include a summary of the test results and the acceptance criteria. (See related request 031.14 parts 1 and 2.)

4. Provide drawings to show worse case examples where terminations of Class 1E and non-Class 1E circuits are made on a common device (isolator). Identify these drawings if presently available in the FSAR.

5. Identify and justify all terminations on devices other than isolators where the requirement for a separation distance of six inches between Class 1E and non Class 1E circuits is not met.

6. Provide a description, including a summary of results, of the method used to qualify the isolation relays in the CRDCS trip breaker cabinet. This device is discussed in item 7 of your NSSS separation criteria.

7. Provide a sketch showing the physical separation between redundant channels which are connected to the reactor trip switch.

8. Provide a description, including a summary of results, of the method used to qualify the CRDCS trip breaker. This method should include a demonstration that the breaker contacts will open, when tripping, during and following a seismic event.

031.21 (7.1) (App. 3A) Your response to Regulatory Guide 1.105 in FSAR Appendix 3A states that "compliance of the NSSS safety-related instrumentation will be provided by amendment." Expedite this submittal consistent with our established review schedule.

031.22 (7.2) (7.8) The response to request 031.9 does not satisfy our concerns. The reactor protection system inputs from the power range detectors are described in Section 7.8 of the FSAR. This section does not, however, address any safety requirements that these systems should meet. Since these inputs are a vital part of the reactor protection system, we require that a description of this system be provided in the FSAR and that the criteria be specified. If this information is provided in a revision of Section 7.8, then a reference to this revised section should be provided in Section 7.2.

031.23 (7.3) (15.2) FSAR Section 15.2.8.2.1 discusses a reverse-flow monitor which is used to actuate the main steam line isolation system (MSLIS) and the auxiliary feed water actuation system (AFWAS). Since credit is taken for this monitor, describe it and include drawings, and show how it satisfies all requirements for a safety system, including environmental and seismic qualification requirements. This monitor should be included in FSAR Sections 7.3.3.2.6 and 7.3.3.2.7.

031.24 (7.3) FSAR Section 10.3.2.2 states that closure of the main steam line isolation valves is accomplished by redundant spring assemblies requiring no additional energy assist. It also states that two channels of actuation provide for positive valve closure on a trip signal (MSLIS). Provide a description, including both logic and mechanical diagrams, to show how each redundant signal accomplishes valve closure. Sufficient detail is required for our review to verify that no single failure will preclude valve closure when required.

031.25 (7.3) (3.11) 12.3.3) The fresh air intake system for the control room is required to have monitors to detect and automatically initiate the emergency mode of operation at sufficiently low activity concentrations so as to assure Criterion 19 of the General Design Criteria is not exceeded during the course of certain accidents. These detectors are therefore considered to be safety grade and all requirements for a safety related system apply, including IEEE 279-1971.

031.25 (cont'd) Describe these detectors. Also describe your associated qualification program and provide a summary of the results which verify that this equipment satisfies all safety requirements.

031.26 (7.3) FSAR Table 7.3-2 provides your design data for the engineered safety features actuation system (ESFAS). This table does not identify the ESFAS subsystem that will be affected by high radiation in the fuel pool area. Identify and describe all such subsystems.

031.27 (7.4.1) FSAR Section 7.4.1 identifies the pressurizer heater controls as a system required for safe shutdown. Yet Section 7.4.1.1.6(e) indicates that this system does not meet all requirements of IEEE-Standard 279-1971 and discusses your basis for not meeting our requirement.

Identify the worst case events which would result from failure of this system while attempting to achieve and maintain hot shutdown. Also demonstrate that each of these conditions will not violate the Commission's requirements.

Identify and justify all sections of IEE-279-1971 which are not met in the design of the pressurizer heater control system. Coordinate the response of this request with the response to request 211.35.

031.28 (7.4) Criterion 19 of the General Design Criteria requires in part that equipment at appropriate locations outside the control room be provided with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. The staff interprets this to mean that the equipment is required for safety and should meet all requirements for a safety related system. These systems should be identified in FSAR Section 7.4 as systems required for safe shutdown.

Modify your description in FSAR Section 7.4 to include all systems required to achieve and maintain safe shutdown of the reactor. This description should include all information specified in Section 7.4 of Regulatory Guide 1.70, and should identify and justify all exceptions.

031.29 (7.4.2) The discussion of Regulatory Guide 1.53 in FSAR Section 7.4.2.3 states that "a failure modes and effects analysis of the control rod drive control system (CRDCS) trip position will be performed and submitted at a later date." This section also states that the discussion of conformance to Regulatory Guide 1.75 for the trip portion of the control rod drive control system will be submitted later. Please expedite these submittals consistent with our established review schedule.

031.30 (7.5) Provide the following information regarding FSAR Section 7.5.2.2.1:

1. By omission, this FSAR section takes exception to several sections of IEEE-279-1971. Describe how your design of the safety related display instrumentation satisfies the missing sections of IEEE-279-1971 or provide justification for each omission.

2. Justify not including safety-related recorders in this design as required by Branch Technical Positio- ISCB 23

3. This FSAR section includes design criteria for only the engineered safety features actuation system (ESFAS). Identify all other safety related display instrumentation and describe how it satisfies Branch Technical Position ICSB 23. Also include the information requested by parts 1 and 2 above.

4. Justify not including control rod position indication as safety related display instrumentation.

031.31 (15.4.6) (7.5) FSAR Section 15.4.6.3.3 takes credit for operator action to terminate the dilution flow during a chemical addition system malfunction. To initiate this action, the operator relies upon a high makeup flow alarm. Describe and provide drawings showing how this alarm satisfies all of the requirements for a protection system or justify the design on some other basis. Where drawings are included in the FSAR drawing package submitted for our detailed drawings review, a reference to the proper drawings should be indicated.

031.32 (7.3.4) (7.5) FSAR Section 9.3.4.3.1 states that "the make up tank has a 10 minute supply of water below the low-level alarm point to enable the operator to line up the BWST (open valve) following a small break." This is described as part of the safety related function of the makeup and purification system. Provide a description, with drawings, to show how this low-level alarm satisfies all of the requirements for a protection system or, justify your design on some other basis. Where drawings are included in the FSAR drawing package submitted for our detailed drawings review, a reference to the proper drawings should be indicated.

031.33 (15) (7.5) With regard to our requests 031.31 and 031.32, identify all other alarms in the Midland designs which are relied upon to inform the operator when to take any necessary manual safety actions. Describe how each of these alarms satisfies all of the requirements of a protection system, or justify each design on some other basis.

031.34 (5.4) (7.6) Your decay heat removal system does not satisfy our requirements:

1. As shown on Figure 5.4-10, a single failure of motor-operated valves 045 or 046 or their power supplies (corresponding to valves 046 and 048 on Figure 5.4-11 for Unit 2) can cause the complete loss of the DHR system. This does not satisfy Criterion 34 of the General Design Criteria.

2. FSAR Section 7.6.1.2 discusses the decay heat removal isolation valve interlock. We find this design unacceptable since it does not satisfy our requirements for independence or diversity for these interlocks. We require that interlocks satisfy item B2 of Branch Technical Position ICSB 3. Also, the design must assure that failure of a single power supply will not preclude isolation between the DHR system and the reactor coolant system when required.

Accordingly, provide a modified design which satisfies both Criterion 34 and Branch Technical Position ICSB 3 or justify this design on some other basis.

031.35 (7.7) Following a steam line break upstream of a main steam isolation valve (MSIV), the single failure of the other MSIV to close could cause the second steam generator to blow down. To preclude this incident, credit is generally implied for all downstream valves and associated control systems to limit blow down of the second steam generator in an acceptable manner. This approach has been found to be acceptable to the staff as expressed in Issue No. ` of NUREG-0138.

The design of Midland 1 and 2 presents an additional aspect which must be considered in the steam line break accident: In addition to the turbine generator pathway, steam is also supplied to the Dow Chemical Company process steam evaporators. Valves intended for isolation and routing of steam to this external system are controlled by the process steam transfer system (PSTS). Following a potential steamline break accident, the PSTS is relied upon to control steam flow such that both steam generators do not blow down.

031.35 (cont'd) Since this system is relied upon to mitigate the consequences of a steam line break accident, then it should satisfy requirements for a safety related system or our position in NUREG 0138:

1. Describe how the PSTS satisfies the requirements of IEEE Std 279-1971 or

2. Provide justification on some other bases that failure of the process steam transfer system will not preclude plant cooldown following any postulated steam line break accident.

   a. Identify all steam pathways downstream of the MSIVs and all control systems which would be expected to isolate such pathwyas following a MSLB.

   b. Identify any such pathway not automatically isolated by control systems following a MSLB, specify the diameter and destination of each (i.e., its significance and reasons for remaining unisolated), and verify that this subsequent steam release has been included in your safety analyses. Identify any credit you have assumed for manual operator action in this regard.

031.36 (9.5.2) FSAR Section 9.5.2 states that a two-way radio system is installed to supplement the public address system and the sound-powered phone system. Describe the procedures and results of the tests used to demonstrate that this equipment will not degrade operation of safety related instrumentation, through radio frequency interference (RFI).

031.37 (7.7.2.2) FSAR Section 7.7.2.2 states that "no accident analyzed in Chapter 15 requires proper functioning of the integrated control system (ICS). Chapter 15 also addresses various abnormalities that could result from failures of the ICS. In all cases, the reactor protection system (RPS) provides the necessary plant protection."

This statement does not support the conclusion that all abnormalities, resulting from all possible failure modes of the ICS, will be kept within acceptable limits by the RPS. Provide the summary of an analysis which identifies all possible failure modes of the ICS, which would not cause an abnormal condition outside of acceptable limits.

040.0 Power Systems Branch

040.77 (8.0) (3A) (RSP) Your response to our request 040.19 is unacceptable relative to Regulatory Guide 1.63. It is our position that Regulatory Guide 1.63, Revision 1 is applicable to Midland Plant Units 1 & 2. Identify and justify each exception taken to the recommendations of revision 1 of this guide.

040.78 (8.3) It is not clear from the referenced drawings provided in your response to request 040.25 how control power is locked out to active and passive valves in your design. Provide a modified response that includes a description of:

1. How power lockout is accomplished for active and passive valves,

2. How power can be re-instated from the control room if repositioning of active valves is required later, and

3. How redundant valve position indication meets the single failure criterion.

040.79 (8.2) (RSP) Your nonconformance to our positions regarding offsite power systems in request 040.14 is unacceptable. It is our position that the design changes required by our positions must be implemented in you design. Provide a modified response and supplement the description of your design in the FSAR to show how it meets our positions.

040.80 (5.4) (8.3) (RSP) Your response to request 040.26 is not satisfactory. The Decay Heat Removal (DHR) system is required to comply with the requirements of criterion 34 of the General Design Criteria. It is our position that the DHR system should be capable of performing the normal shutdown cooling function even with the system experiencing a single active failure of a fluid component or any single active or passive failure of an electrical system. To demonstrate that your design satisfies this criterion, provide an analysis assuming failure of 1MO-1110 (2MO-1110) valve to open. In addition provide a sketch that shows how power is supplied to valves 1MO-1110, 1MO-1111 and 1MO-1112 in Figure 5.4.11 of the FSAR.

040.81 (8.3) (RSP) Your response to request 040.28 is incomplete. We require that the circuit breaker protection system trip set points and breaker co-ordination between primary and backup protection shall have the capability for test and calibration. Provisions for test under simulated fault conditions should be provided. For designs where protection is provided by a combination of a breaker and a fuse or two fuses in series, provisions shall be provided for testing fuses. Revise your FSAR to include this information.

040.82 (8.3) (3A) (RSP) Your exceptions taken to positions C.2.a(3), C.2.a(7), C.2.c(2) and C.2.e(3) of Regulatory Guide 1.108 in your response to request 031.3 are unacceptable. We require full conformance to all the provisions of Regulatory Guide 1.108. Revise Appendix 3A of the FSAR to include our requirements.

040.83 (8.1) With regards to IEEE Standard 336-1971, "Installation, Inspection and Testing Requirements for Instrumentation and Electrical Equipment During the construction of Nuclear Power Generating Stations," it is not clear from the discussions presented in Section 8.1 of the FSAR whether the requirements of this standard have been or will be met for the installation, inspection and testing of electrical equipment. Provide a discussion defining the degree of conformance to the requirements of this standard.

040.84 (8.2) Your present design does not include any provisions for the disconnection of the reactor coolant pumps from the electric system in the event of an underfrequency condition. We are concerned with underfrequency transient(s) that would affect reactor coolant (RC) pump speed, i.e., the assumed RC pump coastdown flow rate. Our concern is further described by Issue No. 9 of NUREG-0138.

Since the RC pump motors remain connected to the power system, identify the frequency decay rates that would result in a braking action on the pumps, resulting in flow rates below that required to maintian the DNBR above the 1.3 limit. Translate these frequency decay rates into a plot of RC flow versus time and compare this with the flow provided by normal pump coastdown. Discuss the method by which this was accomplished. Provide this information for the worst-case core life condition. Identify possible initial grid operating conditions that could be expected and that would allow significant frequency decay rate or other undesirable influences that could adversely affect the design basis reactor coolant coastdown flow rate.

040.85 (8.3) Section 8.3.1.1.2 of the FSAR states the, "If preferred power is available to a 4.16 kv Class 1E bus following a LOCA, the loads are sequentially started." Provide your basis and justification for sequencing safety loads when preferred power is available during the accident.

Provide a comparison on a bus by bus basis for all emergency buses of the voltage and motor starting transients associated with sequences versus instantaneous loading for the condition of grid voltage at the low end of its normal range and maximum plant auxiliary load.

Provide a description of what would be required to remove this non-standard design feature from your design and the associated safety implications, if any.

040.86 (8.3) Table 8.3-11 of the FSAR indicates diesel generator ventilation fan control switch in pull-to-lock position and low lube oil temperature or low jacket water temperature conditions render the diesel generators inoperable for emergency start. Provide your basis and justification for these conditions to render a diesel generator inoperable for emergency start.

040.87 (8.3) With regard to the Class 1E d-c power system, address the following:

1. Does the battery charger have sufficient capacity to operate all non-accident shutdown loads assuming the battery is not available?

2. Is the stability of the battery charger output load dependent?

3. Is there any annunciator to alarm whenever the charger gets into a current limiting condition?

040.88 (6.3, 8.3) Section 6.3.2.2.2 of the FSAR describes an additional makeup pump that is to be an installed spare between the two safety trains. We require that the spare make-up pump provided in your design and all associated signals, power cabling and control devices that may interface with portions of either safety division must be treated as a third safety division for separation purposes. Provide the details of your design that safisfy this requirement.

040.89 (10.4, 8.3) It appears that your design of Auxiliary Feed Water (AFW) System is susceptible to single failure if the AFW isolation valve 1FV3875A or B to the unit affected steam generator inadvertently close, resulting in loss of all AFW flow for the affected unit following a main steam or feedwater line break inside containment. Provide a sketch that shows how power to these valves is supplied and demonstrate that no single electrical failure in the valve control circuit will result in inadvertent closure of isolation valve 1FV3875A or B (Unit 1) or 2FV3975A or B (Unit 2).

040.90 (10.4, 8.3) It is our position that the Auxiliary Feed Water System should be capable of operating even if all alternating current power (other than static inverter) is unavailable. Accordingly, provide information which clearly indicates and verifies conformance to this position. Identify the DC source that is associated with the steam turbine portion of the Auxiliary Feed Water System.

040.91 (8.3) (RSP) Your present criteria for color coding cable and raceways (for distinguishing purposes) for each separation group up to (but not including) the terminal equipment, is unacceptable. We require that terminal equipment be included in your color coding scheme, to provide a visual means of separation group identification. Provide your criteria for identifying Class 1E terminating equipment. Include in your response a color coding scheme for panels where cables from redundant divisions terminate.

040.92 (8.3) Provide your criteria for separating Class 1E cables and raceways from non-seismic field routing piping.

040.93 (8.3) Your description of separation group 'E' is inadequate. Identify all Class 1E "swing" loads (Class 1E loads that can be manually connected either to load group I or load group II, but not simultaneously) and demonstrate that (1) the independence of both standby power sources will not be compromised and, (2) at least one interlock is provided in its circuitry to preclude an operator error from paralleling their standby power sources.

040.94 (9.5.4) (9.5.5) (9.5.6) (9.5.7) (9.5.8) State the degree of conformity of the design of the emergency diesel generator (including the following subsystems: fuel oil storage and transfer, engine cooling water, engine starting, engine lubrication, and combustion air intake and exhaust) to the following regulatory guides: 1.26, 1.29, 1.68, 1.102, 1.117, and 1.137.

040.95 (9.5.4) (RSP) In response to request 040.7, you reference Figure 9.5-31, "Emergency Diesel Engine Fuel Oil Piping Schematic." This figures shows that there are two duplex strainers on the engine and one duplex strainer on the auxiliary module. The strainers on the engine are shown with pressure differential switches that provide indication when there is a high differential pressure across the strainers. However, the duplex strainer on the auxiliary module does not show a means for measuring the pressure differential across the strainer. We require that a differential pressure indicator be provided for the duplex strainer in the auxiliary module. Revise your design accordingly.

040.96 (9.5.4) (RSP) In reference to Figure 9.5-25, "Emergency Diesel Generator Fuel Oil Storage and Transfer, Units 1 and 2":

1. Provide explanations for the notes shown on the drawings.

2. A strainer is shown in the 1½" line between the storage tank and the day tank. Indicate if the strainer has a means for measuring the differential pressure across the strainer. We require a means of measuring the differential pressure across the strainer.

040.97 (9.5.4) (RSP) Your response to request 040.35 indicates that the emergency diesel fuel oil storage and transfer system meets the requirements of ANSI N-195-1976 with the following exceptions:

1. The storage tank fill line is not provided with a strainer or a shutoff valve. The fill connection is, however, provided with a weatherproof cover which may be locked closed.

2. The fuel oil transfer pump is a submersible type which necessitates that the fuel oil strainer be alternatively located on the suction

of the fuel oil booster pumps rather than the suction of the transfer pumps.

In regard to Item 1, we require that a shut-off valve be provided in the fill line of each storage tank. Also we recommend that a strainer should be provided for each fill line. Revise your design accordingly.

In regard to Item 2, from Figure 9.5-25, "Emergency Diesel Generator Fuel Oil Storage and Transfer, Units 1 and 2," the location of the fuel oil booster pump is not clear. Provide this location together with the location of the strainer ahead of the booster pump on Figure 9.5-25.

040.98 (9.5.4) Your response to request 040.45 regarding the nearness and names for supplying additional fuel oil in a timely manner when needed, is not sufficiently complete. Provide additional information on the sources, distance and means of transportation of diesel fuel to the nuclear plant site.

040.99 (9.5.4) In response to request 040.6, you indicate that the location of the buried underground emergency diesel fuel oil storage tanks is shown on Figure 1.2-1. For clarification, provide a detailed description and drawing (plan, elevation and sections) of the buried emergency diesel fuel oil storage tanks and the associated piping to and from the day tanks.

040.100 (9.5.6) Your response to request 040.52 provided revisions to Figure 9.5-27 to include the seismic design boundaries of the different portions of the diesel generators starting system. However, it appears that the "S-1" symbol on the upper compressor in Figure 9.5-27 is pointing in the wrong direction since it is in the reverse direction of the "S-1" symbol shown for the lower compressor. Either correct the direction of the "S-1" symbol or explain the reversed directions.

040.101 (9.5.7) In reference to Figure 9.5-28, "Emergency Diesel Generator Lubrication System," provide the following information:

1. Note 8 has the symbol "LS" for a low level alarm switch. However this symbol is not shown on the figure. Provide the location of the low level alarm switch.

2. Note 10 is the a pressure switch for low lube oil pressure. However the symbol is not given and the location is not shown on the figure. Provide this information.

3. Along the lines on Figure 9.5-28 are numbers from 2 to 12. Explain what these numbers represent.

040.102 (9.5.8) Your response to request 040.58 relative to exposure of the diesel generator intake and exhaust system from atmospheric conditions (ice, freezing rain, or snow) referred to revised subsection 9.5.8.2.1 in the FSAR. This section stated that - "any snow or rain entering the exhaust stack would fall vertically down the stack into the silencer --- the snow would melt and drain through the exhaust silencer drain." This answer applies when there is heat from the operating diesel generator. Indicate how the diesel generator exhaust would be prevented from clogging up from freezing rain and snow when the diesel is not operating.

040.103 (10.2) Your response to request 040.61 is not complete. You did not show that a single valve failure cannot preclude the turbine overspeed trip from functioning. Discuss the effect of one of the valves not closing upon a signal from the overspeed protection system.

040.104 (10.4.1) Your response to request 040.71, relative to hydrogen production in the secondary side water, indicates that subsection 10.4.1.2.2 has been revised. However, no changes were made. Provide your revision as stated.

110.0 MECHANICAL ENGINEERING BRANCH

110.46 (3.9.3) (6.2) During our meeting of May 23, 1978, on asymmetric LOCA loads, you requested that the staff identify the information which will be needed for all system components and structures in a composite manner consistent with our requests for operating plants. The following is in response to that request.

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 and structures that require reassessment include:

a. Reactor pressure vessel
b. Fuel assemblies, including grid structures
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel, steam generator, pressurizer, and pump supports
g. Reactor internals
h. Biological shield wall and neutron shield tank (where applicable)
i. Steam generator, pressurizer, and pump compartment walls

110.46.1 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 the steam generator, pressurizer, and pump support systems and the various cavity structures in sufficient detail to show the geometry of all principal elements and materials of construction.

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

   a. Reactor vessel hot and cold leg nozzle to piping terminal ends.

   b. Pump suction and discharge nozzles to piping terminal ends.

   c. Steam generator inlet and outlet nozzles to piping terminal ends. [1]

[1] Postulated steam line breaks may control the design of certain steam generator supports and, therefore, must also be considered in support design.

d. Pressurizer inlet nozzle to piping terminal end.

Provide an assessment of the effects of asymmetric pressure differentials[2] on the structures, 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

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

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

   b. For structures, provide the maximum predicted and the allowable ductility ratios when considering the effects of localized impact and impulsive loads.

4. 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.

5. For the various cavity structures, describe the extent to which the design meets the structural design criteria identified in Section 3.8.3 of your Safety Analysis Report.

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

7. For the combination of dynamic responses within the reactor coolant pressure boundary and its supports, which result from the coincidence of an SSE and LOCA, the square root of the sum of the squares (SRSS) technique is acceptable contingent upon performance of an elastic dynamic analysis to meet the appropriate ASME Code, Section III, service limits. In all other cases, dynamic responses shall be combined by absolute summation unless justification acceptable to the staff is provided for any other method of combination.

[2] 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.

110.46.2 In order that we may evaluate your methods employed to compute the asymmetrical pressure differences across the core support barrel during subcooled portion of the blowdown analysis, the following information is requested:

1. A complete description of the hydraulic code(s) used including the development of the equations being solved, the assumptions and simplifications used to solve the equations, the limitations resulting from these assumptions and simplifications and the numerical methods used to solve the final set of equations. Provide comparisons with experimental data, covering a wide range of scales, to demonstrate the applicability of the code and of the modeling procedures to the subcooled blowdown portion of the transient. In addition, discuss application of the code to the multi-dimensional aspects of the reactor geometry.

   If an approved vendor code is used to obtain the asymmetric pressure difference across the core supprot barrel, state the name and version of the code used and the date of the NRC acceptance of the code.

2. If the assessment of the asymmetric pressure difference across the core support barrel is made without the use of a hydraulic blowdown code, present the methodology used to evaluate the asymmetric loads and provide justification that this assessment provides a conservative estimate of the effects of the postulated LOCA.

110.46.3 A compartment multi-mode, space-time pressure response analysis is necessary to determine the external forces and moments on components. Analyses should be performed to determine the pressure transient resulting from postulated hot leg and cold leg reactor coolant system pipe ruptures within the reactor cavity and any pipe penetrations. If applicable, similar analyses should be performed for steam generator, pressurizer, and reactor coolant pump compartments that may be subject to pressurization and where significant component support loads may result. The proposed method of evaluation and principal assumptions to be used in the analysis should be provided for review in advance of the final load assessment.

The following type of information is to be provided in the FSAR. Although this request was primarily developed for reactor cavity analyses, it should be applied to other component subcompartments by general application.

1. Provide a descripton of the computer program used to calculate the mass and energy release from the postulated pipe breaks. Provide the nodalization scheme for the system model, and specify the assumed initial operating conditions of the system. Discuss the conservatism of the blowdown model with respect to the pressure

response of the subcompartment. If the computer code being used has not been previously reviewed by the staff, provide a comparison of the blowdown to that predicted by a previously accepted code as justification of its acceptability.

2. Provide the assumed initial operating conditions of the plant.

3. Provide and justify the pipe break type, area, and location for each analysis.

4. 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 evalution. This mass and energy release data should be provided in tabular form, with time in seconds, mass release rate in lbm/sec, enthalpy of mass released in Btu/lbm, and energy release rate in Btu/sec. A minimum of 20 data points should be given from time zero to the time of peak pressure. The mass and energy release data should be given for at least the first three seconds.

5. 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.

6. 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.

7. 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. The nodal model development studies should show that a spatially convergent differential pressure distribution has been obtained for the selected evaluation model.

   Describe and justify the nodalization sensitivity study performed for the major component supports evaluated, if different from the structural analysis model, where transient forces and moments acting on the components are of concern. Where component loads are of primary interest, show the effect of noding variations on the transient forces and moments. Use this information to justify the nodal model selected for use in the component supports evaluation.

If your design is such that the pressurization of subvolumes located in regions away from the break location is significant, show that the selection of parameters which affect the calculations have been conservatively evaluated. Particular attention should be given to pressurization of the volume beneath the reactor vessel. In this case, a model which predicts the highest pressurization below the vessel should be selected for the evaluation.

NOTE: It has been our experience that for the reactor cavity three regions should be considered (i.e., nodalized) when developing a total model. These are:

(1) The volume around or in the vicinity of the break location out to a radius approximated by the adjacent nozzles, and including portions of the penetration volume for some plants;

(2) The volume or region covering the upper reactor cavity, primarily the RPV nozzles other than the break nozzle; and

(3) The region encompassing the lower reactor cavity and other portions of the reactor cavity not included in items (1) and (2) above.

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

9. Graphically show the pressure (psia) and differential pressure (psi) response as functions of time for a representative number of nodes to indicate the spatial pressure response. Discuss the basis for establishing the differential pressure on structures and components.

10. 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 uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending upon 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, particularly for the reactor cavity as discussed above.

11. Provide the peak and transient loading on the major components used to establish the adequacy of the support 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. The centerline of the break nozzle is recommended as the X coordinate and the center line of the vessel as the Z axis. Provide the projected area used to calculate these loads and identify the location of the area projections on plan andsection 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.

110.47 (3.9.2.1) Recent reactor operating experience suggests that the suction and discharge piping of positive displacement pumps may experience unaceptable vibration and high cycle fatigue. Question 110.36 described the systems which we require to be tested for abnormal transient or steady-state vibration. Therefore, for the systems listed in 110.36:

(1) Provide a commitment to monitor vibration in the suction and discharge piping of any postive displacement pumps during the preoperational test program.

(2) Describe and provide justification for the acceptance criteria against which the observed or measured values will be compared.

(3) Discuss the methods you will use to eliminate unacceptable vibration in this piping if found during the test program. Pulsation dampeners and stabilizers are possible solutions.

110.48 (3.9.5) (RSP) You have referenced topical report BAW-10008, Rev. 1, Part 1, for the design of the core support structures and other reactor internals. This topical report describes the methods for calculating loads on the reactor internals resulting form both OBE and SSE. The report also describes the stress and deformation analyses performed and provides a comparison of the calculated and allowable values. The staff approved this topical report in August 1972.

BAW-10008 describes an analog technique for calculating the LOCA induced differential pressures acting on the core barrel and other reactor internals. Recently, Babcock and Wilcox submitted topical report BAW-10132 which describes newly developed analytical techniques for calculating LOCA related loads. The loads calculated by the methods of BAW-10132 may be larger than the loads calculated by the analog technique used in the BAW-10008 stress analysis.

BAW-10008 included in the loads the vibratory motion of the reactor vessel due to LOCA thrust forces. However the analysis did not consider the motion of the reactor vessel due to the asymmetric cavity pressurization effects of a pipe break at the reactor vessel nozzle.

Provide a commitment to perform a reanalysis of the Midland reactor internals and core support structures. This analysis shall include all the loading conditions of BAW-10008 with the addition of reactor vessel motion caused by asymmetric cavity pressure differentials. The thermal-hydraulic analyses shall be in accordance with the staff approved version of BAW-10132. The staff evaluation of this topical is expected durign October, 1978. The resultant calculated stresses and deformations shall be compared against the allowable values in BAW-10008 or against the allowables of Article NG-3000 of the ASME Boiler and Pressure Vessel Code. Provide a schedule for completion of this reanalysis. The results of the analysis must be reviewed and accepted by NRC prior to OL issuance.

110.49 (3.10) (3.8.2.2) In question 110.26 we provided a description of the staff's Seismic Qualification Review Team (SQRT). We also requested seismic qualification information for selected Class 1E electrical equipment. In 110.44 we requested a schedule for your submittal of the remaining qualification summaries missing from FSAR Tables 3.9-1 and 3.9-17.

During its review, SQRT will emphasize the mechanical and electrical equipment required for achieving safe cold shutdown assuming the following scenario:

(i) safe shutdown earthquake, with coincident
(ii) loss of offsite power, and
(iii) assumption of any single active failure.

SQRT will begin its review after your submittal of all requested infomation, including the electrical equipment seismic qualification forms (110.26), the mechanical equipment qualification summaries (110.44), and the active pump and valve appurtenance qualification summaries (110.39). After an initial review of this information, SQRT may request additional information on selected components. Finally, a site visit will be necessary to inspect and otherwise evaluate selected components.

1. So that SQRT may optimize its efforts, denote in FSAR Table 3.9-1 those mechanical components required for safe cold shutdown assuming the scenario above.

2. Verify that the electrical seismic qualification forms will include all NSSS electrical equipment required for safe cold shutdown assuming the scenario above.

110.50 (3.9) Bechtel's Interim Report #6 to MCAR # 22, dated August 29, 1978, states that the containment spray piping anchors will be evaluated against Appendix F of Section III of the ASME Code for waterhammer loads. This implies that the piping itself might be stressed above Service Limit B for the waterhammer loads. We have asked you in 110.28 and 110.41 for information related to the functional capability of essential Class 2 and 3 piping. Describe how your design assures that the containment spray piping can deliver rated flow when subjected to waterhammer loads.

110.51 (3.9.3) Appendix XVII-2461.1 of the ASME Code Section III requires that bolt loads in bolted connections under tension for linear component supports include prying effects due to the flexibility of the connection.*

1. Provide confirmation that the loads in bolted connections for linear component supports were determined by considering the deformation of the connection, including component-to-component connections and component-to-plant structure connections such as base plates and anchor bolt connections. This information should include representative diagrams of the connections, the analytical techniques and models used, and the maximum stresses in the bolts and the connections under static, cyclic, and impulsive type loading.

2. If the connection was assumed to be rigid, provide complete analytical or experimental justification for this assumption.

110.52 (5.4.2) Several OTSG tube failures have occurred at Oconee Units 1 and 2 and other B&W operating plants. A suspected contributing factor to these failures is the flow induced vibrations caused by frequent testing of the turbine stop valves. Describe the mechanical modifications in the steam generator, main steam line and associated piping, and/or other measures which are being proposed to preclude the occurrence of similar problems in the Midland OTSG's.

*Similar requirements for structural joints are also stated in the AISC Manual of Steel Construction, 1970 Edition, for plants in which support design predates Subsection NF of Section III of the ASME Code.