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

February 24, 1978

Docket Nos.: 50-329 & 50-330

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

THIS DOCUMENT CONTAINS
POOR QUALITY PAGES

Gentlemen:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - PART ONE

In continuing our review of the FSAR for Midland Plant Units 1 & 2, we find we need additional information to complete our evaluation. This information request is contained in Enclosure 1.

The information requests provided in Enclosure 1 use a sequential numbering system continuing from those following our acceptance review and provided by our letter of November 11, 1977. As indicated in our letter of December 27, 1977, we have scheduled our round-one requests in three separate parts for which this is the first part. Enclosure 1 is based upon our review of FSAR revision numbers three or four.

We will need complete and adequate responses to Enclosure 1 by April 14, 1978. If you cannot meet this date, inform us within seven days after receipt of this letter so that we may revise our schedule accordingly.

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

Please contact us if you desire clarification or other discussions of the information requested.

Sincerely,

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

Enclosure: As Stated

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Consumers Power Company

ccs:

Michael I. Miller, Esq.
Isham, Lincoln & Beale
Suite 4200
One First National Plaza
Chicago, Illinois 60670

Judd L. Bacon, Esq.
Managing Attorney
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Mr. Paul A. Perry
Secretary
Consumers Power Company
212 W. Michigan Avenue
Jackson, Michigan 49201

Howard J. Vogel, Esq.
Knittle & Vogel
814 Flour Exchange Building
Minneapolis, Minnesota 55415

Myron M. Cherry, Esq.
One IBM Plaza
Chicago, Illinois 60611

Honorable Curt Schneider
Attorney General
State of Kansas
Topeka, Kansas 66612

Irving Like, Esq.
Reilly, Like and Schneider
200 West Main Street
Babylon, New York 11702

James A. Kendell, Esq.
Currie and Kendall
135 North Saginaw Road
Midland, Michigan 48640

Louis W. Pribila, Esq.
Michigan Division Legal Department
47 Building
Dow Chemical USA
Midland, Michigan 48640

Lee Nute, Esq.
Michigan Division
The Dow Chemical Company
47 Building
Midland, Michigan 48640

Enclosure 1

REQUEST FOR ADDITIONAL INFORMATION (Q1s)

PART 1 of 3

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 dated September 1975.

010.0 AUXILIARY SYSTEMS BRANCH

010.19 (3.2.4) Your response to our request 010.2 is not complete. Provide the following information:

a. Provide the bases for assuming that a flood from circulating water system failure will be limited to the plant grade level at elevation 20'-6" inside and outside the turbine building areas. Describe the turbine building wall construction and door arrangement to demonstrate that flood water inside the turbine building will not incapacitate safety related equipment.

b. List all doors and other openings on the walls between turbine building and auxiliary building below the grade level. Describe the water-tight design of these doors and openings and their related administrative controls to ensure that they are closed when isolation is needed.

010.20 (3.6) Provide layout drawings of the safety-related areas outside containment showing all high and moderate energy piping systems and their relation to the safety related equipment.

010.21 (9.0) Your response to our request 010.6 is not complete. Describe the automatic heating system for the borated water storage tank in sufficient detail, including equipment and instruments design classification and power sources. Also, confirm that the borated water heating system design meets the single failure criteria and is Class 1E.

010.22 (9.1.4) Your response to our request 010.10 is not acceptable. Provide detailed arrangement drawings of the fuel pool and fuel cask handling areas including areas housing safety-related equipment below the cask handling path. Modify your design to meet the following positions:

010.22 a. Provide both mechanical and electrical stops for the cask handling crane to preclude possible passing of heavy loads over the fuel pool.

b. Provide the results of an analysis to demonstrate that the floor above safety related equipment can withstand a cask drop from the maximum possible cask lift height assuming a cask drop configuration that will result in the worst effect to the floor and not perforate the floor or generate secondary missiles on the other side of the floor that would damage safety related equipment.

c. Provide physical restrictions, including mechanical or electrical interlocks, in addition to the administrative controls to ensure that the cask lifts will not exceed the maximum elevation assumed in your cask drop analysis.

010.23 (9.2.1) Section 9.2.1 and Table 3.2-1 seem to indicate that the service water traveling screens and screen wash pumps are not designed to seismic Category I requirements and not connected with essential power supplies. Provide the results of an analysis to demonstrate that during post-accident or following a tornado or seismic event with loss of offsite power, the essential service water intake will not be blocked due to accumulation of debris or modify your design accordingly.

010.24 (9.2.1) Section 9.2.1 indciates that the service water pump pit normally receives water supply from the cooling towers. An engineered safety features actuation signal (ESFAS) will automatically shift the water supply to the cooling pond by opening the power operated valves at the discharge lines to the pond, and opening the power operated sluice gates between the pump pit and the pump house forebay. Storage capacity of the pump pit is adequate to feed all four service water pumps while the sluice gate is being opened.

010.24 Provide the results of an analysis to demostrate that the service water pumps are protected from damage due to low suction pressure when the non-safety grade water supply from cooling towers is lost (e.g., SSE) without presence of the ESFAS. If operator action is required in this case, 30 minutes time should be assumed. Confirm Class 1E service water pump pit level indication and alarms are available inside the control room.

010.25 (9.2.1) Figure 9.2-1 and Figure 3.5-1 indicate that the essential service water supply and return lines are tornado missile protected by being buried or located in seismic Category I buildings. Provide results of an analysis to demonstrate that the depth of the buried piping is sufficient to protect the safety related service water piping from tornado generated missiles in the area.

010.26 (9.2.2) (RSP) Your response to our request 010.12 is not acceptable. It is our position that you must modify the design of the component cooling water system supplying cooling water to the reactor coolant pumps to meet the position stated in our request 010.12. If you plan to demonstrate that the RCP's of Midland Plant, Unit Nos. 1 & 2 will not experience shaft seizure following loss of cooling water for longer than 30 minutes without the need for operators corrective action, then an actual pump test should be conducted for verification. Also, to satisfy the criteria in Approach 1 of the above position, safety grade instrumentation to detect the loss of CCW to the RCP's and to alarm to the operator in the control room should be provided.

010.27 (9.2.6) Modify your Figure 9.2-19 to show the auxiliary feedwater supply lines from the condensate storage tanks.

010.28 (9.2.6) Expand Section 9.2.6 to discuss the basis for sizing the condensate storage tanks including the minimum condensate storage available for auxiliary feedwater supply. State assumed time period for plant hot standby and cooldown using condensate supply. Describe

010.28 how the minimum condensate storage capacity is maintained in each condensate storage tank for the auxiliary feedwater system.

010.29 (9.3.4) Design deficiences have been identified in other B&W plants such that following a loss of offsite power, the reactor coolant pump seals cannot withstand the resulting interruption of seal water flow without damage. Expand Section 9.3.4 to address the Midland Plant design relative to the above stated deficiencies. Describe the modifications made to correct the deficiency if there is any, and confirm that the Midland plant can withstand a loss of offsite power without seal damage to the Reactor Coolant Pumps.

010.30 (9.4.1) (RSP) Section 9.4.1 states that the battery room HVAC system is designed to maintain the hydrogen concentration below 4.1 volume percent. This is not acceptable. It is our position that you redesign the battery room ventilation system to limit the hydrogen concentration to well below two volume percent and alarm in the control room when battery room ventilation is lost.

010.31 (9.4.2) Figure 9.4-3 indicates that there are a 22-inch and a 25-inch non-seismic Category I air supply ducts passing through the spent fuel pool area without seismic Category I isolation dampers in the lines. Provide the results of an analysis to demonstrate that the operation of the standby exhaust and filtration system after a postulated fuel accident will maintain its design safety function assuming the spent fuel pool structure boundary is broken due to non-seismic Category I duct failure.

010.32 (9.4.3& 9.4.5) Section 9.4.3 and Section 9.4.5 indicate that the HVAC system for the Auxiliary area is not designed to seismic Category I requirements and yet safety grade unit coolers are provided in each ESF equipment room for control of environment. Provide a discussion to demonstrate how, without a safety grade exhaust and filtration system in the ESF equipment area, a negative pressure can be maintained inside these equipment rooms to preclude possible radioactive release into the environment during post-LOCA operation, or provide a seismic Class I exhaust and filtering system.

010.33 (10.4.9) Your response to our request 010.18 is not complete. Provide the following additional information:

a. Section 10.4.9.2 indicates that the AC operated backup cooling water supply to the turbine driven auxiliary feedwater pump is required when the temperature exceeds 100°F. Verify and confirm that the temperature will not exceed 110°F, during emergency operation using condensate or service water supply at the maximum possible water temperature during summer season.

b. Figure 10.3-1 and Figure 10.4-10 indicate that the AC power operated valves in the turbine driven auxiliary feedwater subsystem may fail in the open position. Discuss the failure mode of these AC valves and confirm that the turbine driven auxiliary feedwater subsystem can operate without AC power supply to meet the diversity requirements of our Branch Technical Position APCSB 10-1.

c. Section 10.4.9.3 states that the reactor coolant temperature can be reduced to about 310°F when using the turbine driven auxiliary feedwater pump (TDAFP). At this point, the TDAFP is stopped and the steam generators boiled down to reduce the reactor coolant temperature to 280°F, at which point the decay heat removal system can operate. This reduction from 310 to 280°F can be accomplished by dumping heat to the main condenser and circulating water system. Assuming a failure of the motor-driven auxiliary feedwater pump and that the main condenser is lost (e.g., loss of offsite power), confirm that safe cooldown of the plant can still be achieved.

010.34 (10.4.9) (RSP) Your response to our request 010.1? is unacceptable. You depend on manual remote control of valves from the service water system to the auxiliary feedwater pump suction following a loss of condensate storage tank supply. There are two events which would lead to unacceptable consequences as a result of your design, since the condensate storage tank is not seismic Category I or missile protected. These are:

a. A seismic event could result in the failure of the condensate storage tank and the loss of offsite power. The auxiliary feedwater pumps would start with no NPSH resulting in the loss of auxiliary feedwater capability; and

b. A tornado could result in a steam line break in the unprotected non-safety portion of the steam line, loss of offsite power, and the loss of the condensate storage tank. Again, the auxiliary feedwater pumps would start with no NPSH resulting in the loss of the auxiliary feedwater system.

It is our position that you protect the plant against these events. Provide a seismic Categoty I, tornado protected condensate storage tank, or provide an automatic switchover to the service water system using safety grade instrumentation and demonstrate that sufficient auxiliary feedwater flow will be available in the time required to prevent unacceptable consequences following these events.

010.35 (10.4.9) The design of your auxiliary feedwater system consists of one motor-driven pump (100%) and one turbine driven pump (100%). A high energy line break at the discharge of one pump and a single active failure of the other pump will result in the inability to perform a safe plant shutdown and cooldown. Revise your design to withstand a high energy line break, coincident with a single active failure in the auxiliary feedwater system.

022.0	CONTAINMENT SYSTEMS BRANCH
022.6 (7.5, 6.2.1.1, 6.2.2.1.5, 6.3.5) RSP	You state in Section 6.2.1.1.3.7 that the instrumentation provided to monitor and record containment parameters following an accident does not include the containment emergency sump water temperature. We require that instrumentation be provided to monitor and record this parameter. Revise your design and discuss your intended compliance with this staff position.
022.7 (6.2.1)	Section 6.2.1.1.1.6.5, Passive Heat Sinks, states that the heat sinks used in the minimum containment pressure analysis for the ECCS evaluation are listed in Table 6.2-10. However, the heat sinks in Table 6.2-10 are also used for calculating the maximum containment pressure. The heat sinks used for both types of analyses should reflect a degree of conservativism to account for the uncertainty in developing the data; i.e., for the minimum containment pressure analysis, conservatively high values should be used, and for the maximum containment pressure analysis, conservativiely low valves should be used. Therefore, discuss how the heat sink data was developed, and its applicability to the maximum and minimum containment pressure analyses. If necessary, revise the analyses using appropriately conservative heat sink data.
022.8 (6.2.1)	Your analysis in Section 6.2.1.1.3.6 of inadvertent actuation of the containment spray system to determine the containment external design pressure assumes a heat transfer coeffecient from the containment shell to the containment atmosphere of 2 BTU/hr-ft^2°F. Justify this value.
022.9 (6.2.4, 7.3)	The Engineered Safety Features Actuation System (ESFAS) provides the signals for containment isolation, which is only high containment pressure. We require diversity in the parameters sensed for the initiation of containment isolation. Therefore, discuss your plans for including other ESFAS; e.g., signals to provide the required diversity for containment isolation.
022.10 (6.2.6)	In Table 6.2-3 and Section 6.2.6, specify the maximum allowable containment leakage rate (La) in weight percent per day.
022.11 (6.2.4)	Identify the containment isolation arrangements which do not comply with the explicit requirements of General Design Criteria 55, 56 and 57, and discuss the rationale for concluding that the isolation arrangements are acceptable on some other defined basis.

022.12 (6.2.5.3) Provide the following information regarding the hydrogen production and accumulation analysis:

a. Discuss the applicability of the experimental data used to support the corrosion rates selected for the aluminum, galvanized materials, and zinc base paints listed in Table 6.2-3a. Identify the key parameters which influence the corrosion rates. Compare the parameters that would be expected in the Midland containment following a LOCA to the parameters for the experimental data.

b. Discuss the converatism in the quantities and surface areas of the galvanized steel and zinc base paint assumed in the analysis. Also, discuss the rationale for not considering aluminum base paint in the analysis.

022.13 (6.2.6.1) Identify those fluid lines penetrating the containment which will be vented and drained to ensure exposure of the system containment isolation valves to the containment atmosphere and the full differential pressure during the containment integrated leakage rate (Type A) test. Those systems that will remain fluid filled for the Type A test should be identified and justified.

022.14 (6.2.6.3) For each fluid line that penetrates the containment, schematically show the isolation valve arrangement and the design provisions (e.g., test, vent and drain connections, black valves) that will permit the isolation valves to be leak tested. Indicate the direction in which the valves will be leak tested. Identify, in Table 6.2-28, all valves for which the applied test pressure will not be in the same direction as the pressure existing when the valve is required to perform its safety function, and provide evidence to show acceptability of testing the valve with pressure applied in the reverse direction.

022.15 (6.2.6.3) 10 CFR 50 Appendix J requires that containment penetrations fitted with expansion bellows be locally tested at the calculated peak containment pressure, Pa. Identify the penetrations fitted with expansion bellows and verify that this requirement can be met.

022.16 (6.2.6.3) Table 6.2-28 identifies the containment isolation valves that will not be subject to Type C leak testing; for example, locked-closed containment isolation valves. Also most of the containment isolation valves under General Design Criterion 57 (closed systems inside containment) will not be subject to Type C leak testing. Discuss your plans to Type C test these valves or justify exempting them from Type C testing.

022.17 (6.2) Your response to request 022.2F did not dicusss the nodalization sensitivity study performed for each subcompartment to determine the minimum number of volume nodes required to conservatively predict the loads acting on compartment walls and component supports. Provide this information. Identify the nodalization scheme used to calculate the loads acting on the compartment walls and that used to calculate loads on the components.

022.18 (6.2) Provide the information requested in 022.2 for the reactor cavity and pressurizer compartment for postulated ruptures in the following piping:

a. Core flood tank lines.

b. Cold leg piping in the reactor cavity.

c. Pressurizer surge line,

d. Pressurizer spray line.

.022.19 (6.2.1.2) RSP Our request 022.2 asked for analyses of the subcompartment pressure transien used in the design of the component supports. Your response references your letter of October 6, 1977, which discusses your participation in a B&W Users Group to evaluate the probability of a reactor coolant system pipe rupture in the reactor cavity relative to our concerns regarding reactor vessel supports. Your letter also states that the results of the probability study were submitted on September 27, 1977 by Science Applications, Inc. as Report No. SAI-050-77-PA. We find the approach described in the topical report unacceptable and require the detailed analyses requested in 022.2. Provide the information requested in 022.2 for the reactor cavity, and other compartments subject to pressurization.

022.20 (6.2) Your response to request 022.5 is unacceptable. Discuss in detail how the containment purge system design complies with the recommendations of our Branch Technical Position CSB 6-4. Also provide the analyses identified in Branch Technical Position CSB 6-4.

040.0 POWER SYSTEMS BRANCH

40.13 (8.3) Identify all safety related cables used in your plants that have polyethelene in its construction . Provide the following information for each type of cable using polyethelene:

Type of cable by name and catalogue number.

Manufacturer.

Type of polyethelene used and how, i.e., insulation and/or jacket.

Results of environmental qualification performed.

Identify the environmental qualification test report for each type of cable.

40.14 (8.2) RSP Recent operating experience has shown that adverse effects on the safety-related power system and safety related equipment and loads can be caused by sustained low or high grid voltage conditions. We therefore require that your design of the safety related electrical system meet the four staff positions in attached Appendix 40-1. Supplement the description of your design in the FSAR to show how it meets these positions or provide appropriate results of analyses to justify non-conformance with these positions.

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

Review and evaluate the alarm and control circuitry for the diesel generators for Midland Plant, Units 1 and 2 to determine how each condition that renders a diesel generator unable to respond to an automatic emergency start signal will be alarmed in the control room. These conditions include not only the trips that lock out the diesel generator start and require manual reset, but also control switch or mode switch positions that block automatic start, loss of control voltage, insufficient starting air pressure or battery voltage, etc. This review should consider all aspects of possible diesel generator operational conditions, for example test conditions and operation from local control stations. One area of particular concern is the unreset condition following a manual stop at the local station which terminates a diesel generator test and prior to reseting the diesel generator controls for enabling subsequent automatic operation.

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

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

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

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

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

(e) Any proposed modifications planned as a result of your evaluation.

40.16 (8.3) Describe how your electrical penetrations and associated connections to the field cables are qualified to withstand LOCA and Steam Line Break environment. Your response should address:

(1) Test Plan

(2) Test set up

(3) Test procedures

(4) Acceptability goals and requirements.

Also, provide an evaluation of the results that demonstrate electrical penetrations are qualified to maintain containment integrity during normal, abnormal, and accident conditions.

APPENDIX 40-1

STAFF POSITIONS ON VOLTAGE VARIATIONS

Position 1: Additional Level of Under-or-Over Voltage Protection with a Time Delay

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

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

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

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

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

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

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

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

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

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

Position 2: Interaction of Onsite Power Sources with Load Shed Feature

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

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

Position 3: Onsite Power Source Testing

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

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

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

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

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

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

Position 4 - Optimization of Transformers Tap Settings

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

110.0 MECHANICAL ENGINEERING BRANCH

110.7 (3.6.2) (RSP) Your response to request 110.1 is unacceptable. The staff will require an augmented inspection program as outlined in request 110.1. FSAR Section 3.6.2.1.4 should be modified accordingly.

110.8 (3.6.3) Regarding your discussion of logitudinal break locations in FSAR Section 3.6 3:

(1) Your response to request 110.2 is not entirely acceptable. We are aware of the discussions on pipe break criteria as referenced in your response. FSAR Section 3.6.3.1.b 2 discusses locations at which longitudinal breaks need not be postulated. We agree with this section with the exception of part of (b). Specifically, we require that both circumferential and longitudinal breaks be postulated at Class 1 intermediate break locations chosen because the cumulative usage factor exceeds 0.1.

(2) Similarly, we disagree with FSAR Section 3.6.3.1.b.1, part b, and require that both circumferential and longitudinal breaks be postulated at Class 1 intermediate break locations chosen on the basis of cumulative usage factor exceeding 0.1.

Justify your positions with respect to our SRP Section 3.6.2.

110.9 (3.6.3.2) Your response to request 110.3 is not entirely acceptable. Your response indicates that topical report BAW-10132P (March 1977) provides justification for thrust coefficients and break opening times other than those allowed in SRP Section 3.6.2. Our review of this report is incomplete. This subject will be considered an open item pending the results of the topical report review.

110.10 (3.6.2) (3.6.3) Additional information regarding strain rate effects is required for Sections 3.6.2.2 and 3.6.3.3:

(1) FSAR Section 3.6.2.2.f states that the minimum specified yield strength may be increased by up to 20% to account for strain rate effects. Our position in SRP Section 3.6.2, Subsection III.2.a states that the yield strength may only be increased by 10%. Justify your position in view of your deviation from our position.

(2) FSAR Section 3.6.3.3 does not discuss this subject. Either state that NSSS analyses do not increase the yield strength by more than 10% to account for strain rate effects or provide justification for doing so.

110.11 (3.6.3) Identify the computer program to be used for the jet impingement analysis described in FSAR Section 3.6.3.3 (bottom paragraph of FSAR page 3.6-56, Revision 3). Provide verification for the program in accordance with SRP Section 3.9.1 if it has not already been addressed in FSAR Section 3.9.1.

110.12 (3.6.3) SRP Section 3.6.2, Subsections III.2.b(2) and (3) describe acceptable methods of analysis for determining the effects of pipe whip. FSAR Section 3.6.3.3 describes the criteria used in your pipe whip analyses.

(1) Describe the factor used to account for rebound effects whenever an energy balance analysis is used. Provide justification for any factor less than 1.2 as required by III.2.b(2) of SRP Section 3.6.2.

(2) Describe the dynamic load factor used when the restraint is analyzed statically as required by III.2.b(3) of SRP Section 3.6.2.

110.13 (3.6.2.2) (3.6.3.3) Provide the following data for both NSSS and BOP pipe whip restraints:

(1) The deformation limits for any energy absorbing materials used in pipe whip restraints.

(2) Justification if these limits exceed 50% of the ultimate uniform strain.

(3) Force-deformation diagrams for the energy absorbing materials. Describe how your analyses consider the non-linearity of these diagrams.

(4) Drawings of any pipe whip restraints which utlize energy absorbing materials.

110.14 (3.9.1) FSAR Section 3.9.1.2.1 states that ANSYS was not used for any nonlinear analysis. FSAR Section 3.9.1.4.1 states that ANSYS is used for elastic-inelastic analyses. We require the information requested in our request 110.4 for ANSYS for any elastic-inelastic analyses.

110.15 (3.9.2) Regarding your discussion in FSAR Section 3.9.2.1 on preoperational vibration and thermal effects test program for piping:

(1) Expand the scope of these tests to include all high energy lines and all seismic Category I moderate energy lines.

(2) Provide a more detailed list of transients that are included in the test program including pump starts and trips, valve closures, etc.

(3) Provide the acceptance criteria against which the measured vibration amplitudes are compared and the bases for these acceptance criteria.

110.16 (3.9.2.2) Provide a list of mechanical components required for achieving hot standby and cold shutdown of the plants after an SSE for both NSSS and BOP scope. Provide a cross reference to each component's seismic qualification summary in FSAR Table 3.9.3-17.

110.17 (3.9.3) (App. 3A) We have reviewed the following FSAR tables for load combinations and allowable stresses:

A) BOP Equipment Purchased After July 1, 1975

3A.1.48-1	Class 1 Piping and Vessels
3A.1.48-2	Class 1 Valves (both active and inactive)
3A.1.48-3	Class 2, 3 Vessels
3A.1.48-4	Class 2, 3 Piping
3A.1.48-5	Class 2, 3 pumps (both active and inactive)
3A.1.48-6	Class 2, 3 Valves (both active and inactive)
3A.1.48-7	Load Combinations Applied to Tables 3A.1.48-1 to 6
3.9-3	Load Combinations Applied to Tables 3.9-6 and 7
3.9-6	Class 1 Component Supports
3.9-7	Class 2, 3 Component Supports

B) NSSS Equipment

3.9-4	Load Combinations and Stress Limits for Class 1 Vessels
3.9-5	Load Combinations and Stress Limits for Class 1 Piping

Additionally, Part I of 3A.1.48 discusses in a general sense the loads considered and the stress limits applied for BOP equipment purchased before July 1, 1975. Part III of 3A.1.48 is referenced by Table 3.9-1 as providing loading combinations and stress limits for NSSS equipment. However, Part III provides no such information. Also, FSAR Section 3.9.3.4.2 briefly addresses NSSS component supports.

Provide the following information so that we may complete our review:

(1) Load combination tables and corresponding allowable stresses for Class 1, 2, and 3 pumps, valves, piping, vessels, and supports in the BOP scope which were purchased before July 1, 1975. These tables should follow the format used in Tables 3A.1.48-1 to 7.

(2) Load combination tables and corresponding allowable stresses for Class 1, 2, and 3 pumps, valves, piping, vessels, and supports in the NSSS scope other than what is addressed in Tables 3.9-4 and 5. The format used in Tables 3.9-4 and 5 is acceptable.

(3) Modification of Table 3A.1.48-7, Tables 3.9-3 to 5, and any forthcoming Class 1 load combination tables to address the design transients in Table 3.9-2. FSAR Section 3.9.1.1 has stated that these transients were considered in the design of all Class 1 items.

110.18 (3.9.3) Provide the allowable buckling loads for Class 1 component supports subjected to faulted load combinations. Provide justification if your criteria exceed the limits of Paragraph F-1370(c) of the ASME Code Section III, Appendix F.

110.19 (3.9.3) (RSP) Your letter of October 6, 1977 regarding asymmetric cavity pressurization loads endorsed topical report SAI-050-77-PA. This topical uses a probability arguement to show that the asymmetric loadings due to a LOCA occurring within the vessel cavity need not be considered in the design of the reactor coolant pressure boundary or its supports.

We have concluded that such probability arguments do not provide an acceptable basis for long term operation without an assessment of the risk resulting from these postulated transient loads. You have already committed to combine LOCA + SSE for the design of ASME Class 1, 2, and 3 components in FSAR Section 3.9.3. We will require that, in addition to these existing commitments, you design the reactor coolant system and its supports for the following load combinations while limiting the resulting stresses to the faulted allowables as listed in FSAR Section 3.9.3:

Weight + Normal Operating Loads + SSE + LOCA
where,
LOCA = all effects of an ASME Class 1 pipe rupture including the asymmetric cavity pressurization caused by a break at a reactor vessel, pressurizer, steam generator, or reactor coolant pump nozzle.

It is our position that the peak loads resulting from SSE and LOCA be combined by absolute sum unless acceptable justification is first provided for any alternative method of combination.

110.20 (3.9.3) Regarding your response to Request 110.14, we note that footnotes in Tables 3.9-9 to 15 state that an elastic-plastic analysis was performed to justify exceeding the elastic limit for primary plus secondary stress intensity. Describe this analysis in more detail and specifically address N-417.6(a) (1-3) of the ASME Code Section III (1968).

110.21 (3.9.3) We require the following additional information regarding your Class 1, 2, and 3 system stress summaries:

(1) As required by Subsection 3.9.3.1 of Regulatory Guide 1.70, provide a summary of the maximum total stress and deformation values compared to the allowable values for each of the normal, upset, emergency, and faulted conditions for all ASME Class 2 and 3 piping, pumps, valves, supports, and vessels required to achieve cold shutdown or mitigate the consequences of a postulated pipe break without offsite power.

(2) Provide the same information with the addition of a cumulative usage factor summary for all ASME Class 1 valves and supports.

110.22 (3.9.3.4) Tables 3.9-17, sheet 2/61, Part E, states that an elastic seismic analysis was conducted on all components whose stresses due to faulted loads were within 10% of the allowable stress limit. This implies that some components were not evaluated for SSE loads. Provide and discuss the following:

(1) Provide the faulted load combinations and allowable stresses for which the Hydraulic Shock Suppressors (C-70) were designed.

(2) Identify and justify any of these components which were not evaluated for SSE loads.

110.23 (3.9.3) (App. 3A) (RSP) We require that all valve operators, and other electrical, mechanical, pneumatic, or hydraulic appurtenances attached to active pumps or valves be qualified by test. This requirement was stated by letter to you, dated September 24, 1976. Modify FSAR Section 3A.1.48 to describe this test program for NSSS and BOP equipment purchased prior to July 1, 1975. A program such as described for BOP equipment purchased after July 1, 1975, will be acceptable.

110.24 (3.9.3) (App. 3A) (RSP) Regarding your discussion in FSAR Appendix 3A.1.48, we require that you modify paragraph (d) at the bottom of FSAR page 3A-72 to specify IEEE 344-1975.

110.25 (3.9.6) The attached Appendix 110-1 provides guidance for submitting your initial 20-month inservice testing program of pumps and valves and for requesting relief from the ASME Section XI, IWP, and IWV requirements. Provide assurance that you will submit your initial 20-month program and any relief requests in a timely manner in accordance with this attachment.

110.26 (3.10) (3.9.2.2) A review of your seismic design adequacy of safety related electrical equipment 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. Attached Appendix 110-2 describes SQRT and its procedures. Notice that Section IV.2.A describes information you should submit so that SQRT can perform its review. Attachment 1 thereto provides a standard format for this information. We require this information for the equipment described in FSAR Subsections:

3.10.4.1.1	4.16 kV Metal - Clad Switchgear
3.10.4.1.6	125 Vdc Distribution Centers
3.10.4.1.9	Containment Electric Penetration Assemblies
3.10.4.1.18	Major Instrument Package
3.10.4.2	NSSS Equipment (only those required to achieve hot standby and cold shutdown)

110.27 (App 3A) Regulatory Guide 1.121 describes a method acceptable to us for establishing the limiting safe conditions of tube degradation of steam generator tubing, beyond which defective tubes as established by inservice inspection should be removed from service by welding plugs at each end of the tube. Discuss your capability for and intended performance in complying with this guide. Justify any alternative criteria you may propose.

110.28 (3.9) It is our position that whenever Service Limit B is exceeded, areas of structural discontinuity in ASME Class 2 and safety related Class 3 piping and thin walled tanks and vessels must be demonstrated to retain sufficient dimensional stability at service conditions so as not to impair the component safety function. While inclusion of secondary stresses produced by constraint of free end displacement is not required to satisfy the stress limits during the emergency or faulted conditions, the reaction loads resulting from the constraint of free end displacement must be included in the functional capability evaluation.

Demonstrate that areas of structural discontinuity will retain sufficient dimensional stability to deliver rated flow whenever Service Limits C or D are used.

You are also referred to Paragraph NA 2142.2 of the ASME Code that discusses large deformations which are possible in areas of structural discontinuity stressed to Service Limit C and gross general deformations which are possible at Service Limit D. Although this does not imply that large deformations will occur in every case where Service Limit B is exceeded, it is our position that an approach such as the following is to be used:

The analyst should examine areas of structural discontinuity, in the context of the geometry and stresses in the system in which they exist, to insure that collapse cannot occur at either the equipment nozzles or in the piping. Examples of possible collapse modes are situations, such as:

(1) A piping system with a cantilvered length of straight pipe where the formation of one hinge would lead to gross plastic deformation, and

(2) A piping system with two anchors, where three points stressed to Service Limits C or D could form hinges and lead to gross pastic deformation.

If a possible collapse mode is identified, a sufficiently detailed analysis should be performed to insure that functional capability is not impaired.

For further explanation of the staff position on Service Limits, operability assurance, and functional capability, see attached Appendix 110-3.

110.29 (5.4.2) The steam generator tube wall defects detected during the baseline inspection on Three Mile Island Unit 2 are suspected to have been caused during the fabrication of the tubes, and during installation of the tubes in the steam generators. Describe the precautionary measures considered both during the manufacturing process of the tubes as well as during the fabrication and installation of the steam generators that would prevent the recurrence of such defects in the Midland steam generators.

APPENDIX 110-1

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.

*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

BWR

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, associated 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. Information Required for NRC Review of Relief Requests

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

TO

APPENDIX 110-1

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	RA	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 accumulatior 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 (876B 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..

WORKING PAPER

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Rev 1

Appendix 110-2

SEISMIC QUALIFICATION REVIEW TEAM (SQRT)

I. SCOPE

SQRT tasks include both generic and site specific reviews. Generic reviews cover equipment supplied by NSSS and A/E common to more than one plant. Specific plant reviews as delineated in the Standard Review Plans, Section 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 seismic design adequacy of safety related mechanical components, electrical instrumentation and their supporting structures to accomplish the following:

1. Changes in seismic qualification criteria, such as the revision of IEEE-344 Standard and the issuance of Regulatory Guide 1.100, require that the staff verify:

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

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

2. Determine the design adequacy of selected components for seismic loading conditions.

3. In the case of plants which have design basis seismic ground motion levels increased, review to assure adequate design margin exists at the revised levels.

III. 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 current seismic qualification criteria applied to plants applying for construction permits and operating licenses.

2. A plant specific seismic review will be conducted of each plant now undergoing licensing review having components qualified to the IEEE-344, 1971 criteria.

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

   B. For components which have only specific plant application (mostly those within the scope of AE supply)seismic qualification review at specific sites will provide information for the site.

3. Seismic qualification review for plants with revised increased design basis seismic ground motion levels will be conducted on a plant by plant basis.

IV. 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 III.1 and III.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 seismic qualification, including acceptance criteria, methods and procedures used in conducting testing and analysis.

      Present and discuss the seismic qualification program on certain specified items (i.e. pumps, valves, diesel generators, motors, bistable units, relays, etc.)

      (2) Information regarding administrative control of component seismic 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 III.2.A., methods and procedures for conducting seismic qualification review are discussed, including selection of plants for site visit 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 shake table motions, 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 seismic qualification reviews specified in III.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 A/E supplied equipment). 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 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 and cold shutdown

(B) hot stand-by

(C) cold shutdown

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

SORT 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 a seismic qualification summary as shown in the Attachment 1 for each of the selected items.

B. 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 over view 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.

C. SQRT conduct inspection on specified items.

D. SQRT describes findings of the inspection.

E. General discussion.

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

3. Plant site reviews for cases involving increased design basis seismic ground motion.
[under development]
In general utility will provide data on systems and components used to bring the plant to shutdown and maintain it in a cold shutdown condition. Safety margin for seismic qualification of equipment should be assessed.

4. 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.

V. 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 Plant 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 with appropriate authorities for work assignments to each member. He reports to the MEB 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 seismic qualifications in the area of responsibility of 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 seismic qualification in the area of their branch, including equipment signal interpretations fo. 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 branch 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 branch project reviewer is expected to participate in the site visit and attend pertinent generic meetings as necessary. The MEB reviewer will have further responsibilities in those cases where revised seismic loads have been established.

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.

ATTACHMENT 1 to APPENDIX 110-2

Seismic Qualification Summary of Equipment

I. Plant Name: ____________________ Type:

1. Utility: ____________________ PWR ______

2. NSSS: ____________________ BWR ______

3. A-E: ____________________

II. Component Name ____________________

1. Model Number ____________________ Quantity: ______

2. Vendor ____________________

3. Physical Description ____________________

____

____

4. Location: (In Plant) Building: ____________________

Elevation: ____________________

5. Natural Frequencies in Each Direction: ____________________

____

6. Functional Description: ____________________

____

____

7. Pertinent Reference Design Specifications: ____________________

____

____

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

Comments: ____________________

____

____

IV. Seismic Qualification Method: Test: __________

Analysis: __________

Combination of Test and Analysis: __________

V. Seismic Input:

1. Required Response Spectra (attach the graphs): __________

____

2. Required Acceleration in each Direction: __________

____

VI. If Qualification by Test, then Complete:

1.[ ] Single Frequency [ ] Multi-Frequency

2.[ ] Single Axis [ ] Multi-Axis

3. Frequency Range: __________

4. TRS enveloping RRS using Multi-Frequency Test [ ] Yes (attach TRS graphs)
[ ] No

5. g-level Test at h_1 = ______ h_2 = ______ V = ______

6. g-level Required h_1 = ______ h_2 = ______ V = ______

7. Mounting:

1. Seismic Report: __________

2. Field Check: __________

8. Functional Verification Performed [ ] Yes [ ] No [ ] Not Applicable

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

   [ ] Response Spectrum [ ] Time-History

3. Model Type (each direction): ____________________

   ____

4. Computer Codes: ____________________

5. Damping: ____________________

6. Support Considerations: ____________________

7. Critical Structural Elements:

A.

Identification	Location	Governing Response Combination	Seismic Stress	Total Stress	Stress Allowable

B.

Max. Deflection	Location	Effect Upon Functional Operability

APPENDIX 110-3

Staff Guidance for Essential Piping, Pumps and Valves

I. In applying the criteria stated below the following prerequisites are to be applied:

(1) All Code requirements for ASME Code Class components must be satisfied. In particular the Code pressure limits specified in conjunction with Service Stress Limits cannot be exceeded (Refer to NC-3000). Code design limits must be satisfied for those loads specified as design.

(2) When a Service Stress Limit above B is specified both operability and functional capability must be demonstrated for that limit. In addition active pumps and valve operability must be demonstrated irrespective of stress limit.

II. Definitions:

Active Component - A pump or valve relied upon to shut down the plant or to prevent or mitigate the consequences of an accident.

Component and Support Functional Capability - Ability of a component, including its supports, in a Safety Related System to deliver rated flow and retain dimensional stability when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

Component and Support Operability - Ability of an active component, including its supports, in a Safety Related System to perform the mechanical motion required to fulfill its designated safety function when the design and service loads, and their resulting stresses and strains, are at prescribed levels.

Essential Systems - Any of the following:

(1) Any ASME Code Class 2 system, or

(2) Those ASME Code Class 3 systems which perform a safety related function.

These are the functional systems necessary to assure:

(1) The capability to shut down the reactor and maintain it in a safe shutdown condition, or

(2) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

III. Service Limits above "B" for Essential Systems

Service Limit C

The use of service limit C for components and component supports of essential systems in lieu of Level B, may be permitted for designated components and component supports provided that (1) the operability of active pumps and valves is demonstrated at service limit C; (2) the functional capability requirements at service limit C are satisfied; and (3) constraint of free end displacement has been considered for the specified loadings.

Service Limit D

The use of service limit D for components and component supports of essential systems in lieu of service limit B, may be permitted for designated components and component supports provided that (1) an analysis comparable to that described in NB-3200 of the Code is conducted which will demonstrate that the designated service loading combination will not produce stresses greater than Level D, and supplemented by additional analyses that show that resulting strains and deflections will not impair the ability of the system to operate normally for extended periods of time; (2) the operability of active pumps and valves is demonstrated at service limit D; (3) the functional capability requirements at service limit D are satisfied; and (4) constraint of free end displacement has been considered for the specified loadings.

IV. Operability and Functional Capability

Active Pumps and Valves

The design of active pumps and valves including their supports, which must perform a mechanical motion to fulfill its safety related function may utilize any of the four service limits including service limits C and D, provided an operability assurance program (meeting SRP 3.9.3 Section II.2) demonstrates that the pump or valve as supported can adequately sustain the designated combined service loadings at a stress level at least equal to the specified service limit, and can perform its safety function without impairment.

The operability requirements specified for mechanical and hydraulic snubbers installed on safety related systems is subject to review by the staff. When snubbers are used, their need shall be clearly established and their design criteria presented.

Functional Capability

Areas of structural discontinuity in piping, tanks and vessels (such as piping elbows, teen connections, tank and vessel nozzle areas, and thin-walled tanks) shall meet service limit B, but alternatively, may be permitted to designate service limits C or D, provided it is demonstrated that the discontinuity areas retain sufficient dimensional stability at service conditions so as not to impair the component functional capability. Justification shall be provided by tests, or analysis or combinations thereof.

Constraint of Free End Displacement

Component and support loads produced by the constraint of free end displacement resulting from thermal or other movement (such as anchor point movement) shall be evaluated for compliance with Code specified stress limits and shall also be included in the operability assurance program for active pumps and valves, and in the functional capability evaluation for areas of structural discontinuity. While inclusion of these loads to satisfy service stress limits when level C or level D stress limits are designated is not required, the reaction loads resulting from the constraint of free end displacement shall be included in the operability assurance program and the functional capability evaluation for any level of stress limit which may be designated, including levels C and D. In each case where the categorization of stresses produced by the constraint of free end displacement is made, the consideration of these stresses and the service limits as secondary rather than primary shall be justified.

121.0 MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.10 (5.3.1) In reference to request 121.5, we require the following additional information on Midland Plant, Unit Nos. 1 and 2 reactor vessels:

(1) A schematic sketch of each reactor vessel showing all welds (longitudinal and circumferential), plates and/or forgings in the beltline region. Welds should be identified by a shop control number (such as a procedure qualification number), the heat of filler metal, type and batch of flux, etc. Each plate and forging should be identified by a heat number and material type.

(2) For each of the above welds, and for welds in the vessel material surveillance program, an identification of the welding process should be provided.

(3) A listing of the following information on all beltline materials (weld, plate and/or forging): chemical composition (particularly Cu, P, and S content), drop weight T_{NDT}, RT_{NDT}, USE and tensile properties. (If any of these fracture toughness requirements have not been determined, use Branch Technical Position - MTEB 5-2 to estimate their value.)

(4) The maximum end of life fluence at the vessel I.D. and 1/4t locations for each weld in the beltline region.

121.11 (5.2.3.1) (5.3.1) Clarify the discrepancies between Table 5.2-3 and Table 5.3-2, with specific reference to materials of construction. Also clarify the discrepancies between Table 5.3-1 and Table 5.3-2, with specific reference to USE and RT_{NDT} of weld deposits.

121.12 (5.2.3.3) (5.3.1) (3A.1.99) (5.3.2) References to Topical Reports BAW-10046P and BAW-10046 are not appropriate. BAW-10046A, for calculation of pre- and post-irradiation properties of reactor vessel materials (using the NRC staff recomendations attached therein), is an acceptable reference. Amend Section 3A.1.99 to reflect the staff recommendations.

121.13 (3A.1.14) (5.4.1.7) Clarify the discrepancies that exist between Sections 3A1.14 and 5.4.1.7 with reference to pump flywheel integrity.

121.14 (5.3.5) (5.4.1) (6.6) (16.0)	Response to request 121.1 is not adequate. Confirm that 100% of all Class 1 components will be given a preservice inspection as defined in article IWB-2100 of ASME Code Section XI, and Class 2 and 3 components will be examined to the extent practical. Parts 1 and 2 of request 121.1 must be fully answered prior to issuance of the OL. Confirm that components subject to inservice inspection, under the applicable rules of the ASME Code that conform to the requirements of 10 CFR Part 50, paragraph 50.55a(g), shall be fully accessible and inspectable. Any exceptions or deviations should be identified along with complete technical justifications for such exceptions and any alternate provisions that may be proposed in lieu of the applicable requirements.
121.15 (3.9.3.4)	Provide information on the fracture toughness characteristics of primary components supports structures and the minimum operating temperature of these supports.
121.16 (6.6) (3.6.2.1) (RSP)	Response to request 121.2 is not adequate. It is our position that augmented inservice inspection is required. Refer to request 110.7.

122.0 MATERIALS ENGINEERING BRANCH - METALLURGY SECTION

122.1 (App. 3A Clarify your discussion of compliance with Paragraph C4 of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." Paragraph C.4 states that unless all of the regulatory positions as stated in Paragraphs C.1, C.2, and C.3 of the guide are met, the weld is subject to rejection unless the soundness is verified by an acceptable examination procedure. Is it your intent to comply with paragraph C.4 of RG 1.50 when the Regulatory Positions C1, C2, and C3 of RG 1.50 are not met?

122.2 Information on the reactor internals has been supplied for only the reactor core support assembly components. Section 4.5.2 of the Standard Review Plan states that the internals for a pressurized water reactor typically consist of the following structures and components: (1) the lower core support structures, including the core barrel, neutron shield pad assembly, core baffle, lower core plate, and core supports; (2) the upper core support structures, including the top support plate, beam sections, upper core plate, support columns, and guide tube assemblies; and (3) the in-core instrumentation support structure. For Sections 4.5.2.1 to 4.5.2.5 of the FSAR, provide the information requested in Revision 2 of Regulatory Guide 1.70.

122.3 (4.5.2.1) Provide the weld metal material specifications for the reactor core support assembly components.

122.4 (4.5.2.2) For the reactor core support assembly components, discuss the degree of compliance with the acceptance criteria of Article NG-5000, "Examination," of the ASME Code, Section III.

122.5 (5.2.3.3.1) Provide a summary of the acceptance criteria and the fracture toughness data for the Class 1 ferritic materials of the steam generators, pressurizer, piping, pumps, and valves of the reactor coolant pressure boundary.

122.6 (5.4.2.1) Recent operating experience with some Babcock and Wilcox once-through steam generators has revealed areas of steam generator tube degradation in the form of circumferential cracks. Expand your discussion in Section 5.4.2.1 of the FSAR to address the actions taken by the Midland Plant, Unit Nos. 1 and 2, to preclude degradation in the steam generators. Discuss the improvements made to prevent inleakage to steam generators from sources such as the condensers. Discuss provisions for access openings to inspect for tube degradation and discuss other steps taken to facilitate steam generator inspection.

122.7 (5.4.2.1.3) (10.3.5) Indicate the degree of conformance with Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators," which is appended to Section 5.4.2.1 of the Standard Review Plan.

122.8 (6.3.2.4) Provide the materials specifications for the principal materials of construction for ECCS components within B&W's scope of supply.

122.9 (5.4.2.2, 16, 3A) Recent operating experience has indicated deficiencies in the techniques and procedures for plugging steam generator tubes. In some cases, plugs which were to have been installed were apparently omitted. Poorly installed, leaky plugs have also been experienced at some plants. In one instance, a steam generator tube plug was found in the reactor vessel.

Describe in detail the plugging technique which would be used for the Midland Plant, Units 1 and 2 should this be necessary because of leaking tubes during plant life. Include a description of any tests of the tube after plugging and your procedure to assure accountability of plugs and plugged tubes.

130.0 STRUCTURAL ENGINEERING BRANCH

130.6 (3.5.3) Provide a table summarizing the wall and roof thicknesses and the strengths, including the age specified for each tornado missile barrier.

130.7 (3.7.2.7) Demonstrate with an example that the use of the square-root-sum of-the squares method for closely spaced modes as opposed to the use of the procedures described in Regulatory Guide 1.92 does not have a significant impact on the Midland piping design.

130.8 (3.7.2.8) Section 3.7.2.8 indicates that non-seismic Category I structures are analyzed and designed to prevent failure under SSE conditions. Describe the method of analysis, the load combinations, and the allowable stresses considered in designing these non-seismic Category I structures.

130.9 (3.7.3.2) Section 3.7.3.2.1 states that the design of structures and the majority of the equipment is not fatigue controlled. Justify this statement in light of our SRP 3.7.3 position of postulating one SSE and five OBEs during the plant life. The number of cycles per earthquake should be obtained from the synthetic time history (with a minimum duration of 10 seconds) used for the system analysis, or a minimum of 10 maximum stress cycles per earthquake may be assumed.

130.10 (3.7.4.1) As a result of our Regulatory Guide Review Evaluation for Midland in 1976, it was our understanding that the only exception made to Regulatory Guide 1.12 was the use of response spectrum analyzers in lieu of discrete response spectrum recorders. In the FSAR, many exceptions to Regulatory Guide 1.12 are indicated. Some of these exceptions include using peak strain gauges in lieu of peak recording accelerographs as well as many exceptions to ANSI N 18.5. We require justification and clarification of these conflicting statements.

130.11 (3.8.1) Describe the specific methods used in the design to account for the radial tensile forces in the dome and walls of the containment.

130.12 (3.8.1.6) Indicate if the value of f'_c stated in Table 3.5-4 is used in all analysis required by 3.8.1 and if so, indicate the age at which the compressive strength of concrete is specified.

130.13 (3.8.1) Your load combination equations and method of analysis in your FSAR Section 3.8.1 deviate from our position in ACI 359 and Section 3.8.1 of the SRP. Demonstrate that the degree of conservatism used in the Midland design is equivalent to that which would have resulted if ACI 359 and the SRP had been used. If your approach is less conservative, provide your basis for concluding that adequate margins of safety exist.

130.14 (3.8.1) You reference the ACI-318-71 Code and ACI 318-63 specifications as the major specifications for concrete work. Specifications for the containment concrete and other materials of construction currently acceptable to the staff are listed in the ACI-359 Code with the exceptions specified in Standard Review Plan Section 3.8.1.II.6(a). Comparisons of your referenced documents reveal several deviations from the staff position. To enable us to evaluate compatibility between the two sets, provide a list of the specifications used for Midland parallel to those listed in the ACI-359 Code. This information should provide for an evaluation and justification of these deviations from the SRP. In addition, provide a list of those specifications required by ACI-359 that were not used for the Midland Plant.

130.15 (3.8.6) Your terminology used in Section 3.8.6 to describe loads and load combinations for steel and concrete structures is not consistent with that used in SRP 3.8.4. Provide a cross reference for the two sets of terminology. Our evaluation indicates that some of your load combinations may be less conservative than those delineated in SRP 3.8.4. Demonstrate that your criteria provide adequate margins of safety for plant design.

221.0 REACTOR ANALYSIS SECTION, ANALYSIS BRANCH

221.2 (4.4.2.5) Provide the radial pressure gradient in the upper plenum and at the core outlet for each allowable loop configuration. Provide an explanation of how these effects are included in the thermal-hydraulic design calculation. Discuss, and support by calculations, the differences in hot channel pressure drop, flow, enthalpy rise and minimum DNBR relative to the assumption of a uniform pressure at the core boundaries.

221.3 (4.4.5 and Chapter 14) The available experimental data for verification of the codes used for predicting the plant response to transient events are limited. Provide the details of your proposed startup test program to obtain the data to verify the analytical methods and to demonstrate the transient characteristics of the plant. The program may reflect the tests performed at similar facilities which are applicable for verification of the Midland analysis.

221.4 (4.4.4.5.6) Provide a description of the instrumentation available and the surveillance requirements and procedures which would alert the reactor operator to an abnormal core flow or core pressure drop (e.g., due to crud buildup) during steady-state operation.

221.5 (4.4.4.4.1) Discuss the basis and analytical procedure for establishing the control band limits of -65 psi on primary coolant system pressure and +2°F on average temperature as those values maintained by the integrated control system (ICS). Justify the use of these limits for less than 4 - pump operation. Justify the assumption that the ICS maintains the system pressure and temperature within control band limits.

Describe the load rationing features of the ICS which control the outlet temperature from each steam generator to prevent gross temperature maldistribution at the core inlet.

232.0 CORE PERFORMANCE BRANCH: PHYSICS SECTION

232.1 (4.3) Several topical reports referenced in Section 4.3 have not yet been submitted. Provide these reports for review or provide detailed summaries of their content on the Midland docket. These reports include:
BAW-10119
BAW-10123
BAW-10121
BAW-10120
BAW-10116
BAW-10118
BAW-10122

232.2 (4.3.2) The incore instrumentation is capable of detecting gross distortions in radial power distributions. However, it may not be capable of detecting localized perturbations (e.g., interchange of Batch 1 and 2 assemblies near core center). Show in Section 4.3.2.2.7c that any fuel loading error that is not detectable with incore instrumentation produces perturbations which do not violate thermal limits when operating at 102% of full power.

232.3 (15.0.1) Have complete analyses been performed to identify all maloperations or failures in the ICS or ICS control functions which might produce more serious consequents in transients?

232.4 (15.0.2) Have safety-related systems also been analyzed with regard to failure of passive components?

232.5 (15.4.1) Discuss the consequences of the Startup Accident as a function of initial core reactivity and indicate the reason for the choice of 1%$\Delta k/k$ subcritical.

232.6 (15.4.3) Since scram occurs at powers significantly less than full power when a pressure trip occurs and the delay time of 0.7 seconds for the pressure trip is at the extreme of the sensitivity analysis, discuss the suitability of the full power scram insertion curve for the startup accident. What effects compensate for the fact that, at low power, the fractional initial reactivity insertion is lower than that shown in Figure 15.0-3?

232.7 (15.4.3.1) Explain the statement (page 15.4-8) that the positive reactivity increase due to single rod withdrawal will cause the inlet temperature to increase in view of the fact that the ICS acts to reduce inlet temperature with increasing power above 15% of full power.

232.8 (15.4.3) Explain the significance of the location of the point labelled "ICS compensation" on Figure 15.4-20. In particular, indicate why it is plotted at an initial power of 105%.

232.9 (15.4.8.4) Describe the techniques used, assumptions made, and results obtained which support your conclusion that no serious core damage or additional loss of coolant system integrity results from the rod ejection accident.

232.10 (15.4.8.2) Describe the features of the CRDM and reactor coolant boundary design that prohibit or render very unlikely the ejection of a second control rod as a result of the first ejection.

232.11 (15.4.8.2) The full power rod ejection accident is performed for an assumed rod worth of 0.65%. What is the expected worth of a rod which might be ejected at full power? What scenario is employed to produce a rod of 0.65% worth? What would be the worth of a second rod which might be ejected?

312.0 SECTION 3, ACCIDENT ANALYSIS BRANCH

312.29 (3.5.1) The response to Request 312.8 is inadequate. The referenced General Electric memo report of March 14, 1973 limits the discussion of the turbine overspeed protection system testing to its capability of being tested during normal operation. A specific periodic testing program is claimed by the report as existing in General Electric "instruction books."

Discuss briefly any plans for implementing the valve test program that is indicated in the General Electric turbine generator operating instructions and describe the program's salient features (e.g., valve testing and inspection procedures and frequencies).

312.30 (3.5.1) The results of your turbine missile analysis, as presented in Table 3.5-3, are in terms of the total probability P_4 for damaging each of the Category I equipment listed in the table. Please revise the table so that the individual probabilities P_2 and P_3 are indicated separately for each target.

312.31 (3.5.1) It appears that the two biggest contributors to the overall risk from low trajectory turbine missiles are the primary system boundary within the containment and the safety related systems within the auxiliary building. In reference to these areas, indicate if your strike and damage analysis includes the effects of concrete scabbing. Describe the modeling that is used in calculating the damage probability of safety related equipment due to concrete scabbing.

312.32 (3.5.1) In reference to the response given to request 312.8, the data presented in Table 2.2-8 are not presented in terms of aircraft type. As indicated in Section 3.5.1.6 of the Standard Review Plan, this information, along with effective plant area and crash probability per square mile are needed in order to estimate the aircraft crash probability for the site.

In order that we may complete our evaluation, please provide appropriate data and an analysis as outlined in Section 3.5.1.6 of the Standard Review Plan regarding the aircraft crash hazard for the Midland plant due to aircraft operations at the Barstow and Tri-City airports. With respect to each aircraft type, use aircraft yearly operations projected for the life of the plant in your analysis.

312.33 (6.1.2) Provide the following information:

A. The type of paints and coatings used in containment.

B. The surface area and thickness or total mass of each type of paint.

312.34
(6.1.2)

Provide the type and amount of plastic and other organic materials in containment such as electrical insulation, machinery lubricants, caulking and seals, etc.

312.35
(6.4)
(3A.1.95)

We are reviewing the control room habitability systems with respect to the chlorine hazards presented by the liquid chlorine storage facility at the Midland Dow Chemical site and also by the nearby railway shipments of chlorine. In accordance with Regulatory Guide 1.95, adequate protection against chlorine for the Midland Nuclear Plant would be provided by Type IV or Type VI control rooms. The proposed control room ventilation system does not comply with either of the above control room categories for the following reasons:

A. The air exchange rate during normal plant operation is greater than 1.0 air change per hour.

B. The air exchange rate for an isolated control room is estimated to exceed 0.015 air changes per hour under 1/8-inch water gauge pressure differential across all penetrations.

C. If the control room were to satisfy the Type VI criteria, control room isolation actuation by remote chlorine detectors (i.e., detectors located at the potential point of chlorine release) would be required.

Review your control room habitability systems design in reference to Regulatory Guide 1.95 and provide appropriate design modifications in order to achieve the level of protection against chlorine indicated by Table 1 of the guide. Indicate those areas (if any) where your design will deviate significantly from the recommendations outlined in items C.1 through C.6 of the guide. In particular, since the present and projected quantities of chlorine stored at the Dow Chemical site are large, long-term releases of chlorine may be possible. Thus, items C.4, C.5, and C.6 should be addressed in detail in the FSAR.

312.36
(6.4)
(7.3)

Discuss briefly the detection sensitivity of the toxic gas detectors listed in Table 7.3-2, specifically in reference to the chemicals identified in Table 6.4-3. Include in the discussion a brief description of the toxic gas detectors in terms of basic principles of operation.

312.37
(15.4.0)

Provide data showing the primary-to-secondary leak rate versus time from accident initiation until the secondary and primary system pressures equilibrate.

312.38 (15.4.8) The response to request 312.24(a) is inadequate, Please provide for the rod ejection accident with loss of offsite power curves showing the primary and secondary systems temperature and pressures versus time, for a period of two hours.

312.39 (15.6.3) The response to request 312.25(a) is inadequate. The revised Figure 15.6-3 does not provide the requested secondary system temperature or pressure, or primary system pressure for the two-hour period following the accident. Please provide curves showing these parameters.

312.40 (15.6.5) Since fuel handling operations for the Midland plant are proposed to take place while the containment is open to the environment, we require that adequate measures exist to mitigate the consequences of a postulated fuel handling accident inside containment. This can be accomplished either by prompt detection of any radioactive release by use of redundant radiation detectors followed by automatic containment isolation, or by purging the containment via ESF grade filters. The staff considers acceptable mitigation for this event to be the criterion given in Standard Review Plan Section 15.7.4 that the dose should be well within the guideline values of 10 CFR Part 100 (taken to be 25% or less of the Part 100 values). Since our preliminary evaluation indicates the dose exceeds our acceptance criterion without prompt containment isolation or other suitable mitigation, provide a full response to the items listed in our earlier request 312.28 and indicate how you plan to comply with this position.

312.41 (15.7.4) The staff requires, based upon Standard Review Plan Section 15.6.5, Appendix B that the dose consequences from leakage of ESF equipment such as pumps, seals, etc., following a design basis LOCA be acceptably mitigated. Acceptable mitigation may consist of release to the atmosphere after filtration through an ESF filter system, or by other suitable means such as an appropriately designed leak-off collection system. Your FSAR does not indicate that such suitable mitigation of ESF leakage has been provided in the Midland design. Revise your design accordingly and discuss your revised conformance with our position. (Also, see related request 321.5).

321.0 EFFLUENT TREATMENT SYSTEMS BRANCH

321.5
(6.5.1)
(3A)
RSP

Your response to Acceptance Review Request 321.1 indicates that the Engineered Safety Features (ESF) Ventilation system designed to maintain a suitable environment for ESF equipment, and described in FSAR Section 9.4.5 is not an ESF filter system and need not be designed in accordance with the recommendation of Regulatory Guide 1.52 (Revision 1). However, it is our position that an ESF filter system is needed to control offsite doses resulting from pump leakage in post-LOCA operation. We require that you provide an ESF filter system as part of your ESF Ventilation System that satisfies all of the positions of Regulatory Guide 1.52 (Revision 1) for this purpose. Revise your discussion in Appendix 3A accordingly.

331.0 RADIOLOGICAL ASSESSMENT BRANCH

331.2 (12.3) Additional information is required regarding the measures you have taken to control buildup, transport, and deposition of the activated corrosion products in the reactor coolant and auxiliary systems. In addition to your discussion on methods used to minimize piping low points, dead legs and crud traps, describe any steps you have taken to minimize the buildup, transport and deposition of Co-58 and Co-60 in the reactor coolant and auxiliary systems. Examples of some methods used to reduce the formation and transport of crud products would include:

1. The use of reduced nickel in primary coolant system alloys

2. Low cobalt impurity specifications in primary coolant system alloys

3. The minimization of high cobalt, hard facing wear materials in the primary coolant system

4. The use of high flow rate/high temperature filtration.

331.3 (12.3.4) Specify the frequency of calibration for your area and airborne radioactivity monitors.

331.4 (12.3) Provide a detailed layout of the solid radwaste area (similar to the one in Figure 11.4-5) indicating radiation zoning for the 634-foot and 652-foot levels.

331.5 (12.5.2) Indicate whether local exhaust systems will be installed in the hot machine shop for work on contaminated items. If such systems are not planned, identify the alternate measures planned to limit airborne contamination from the machining of contaminated items.

362.0 GEOTECHNICAL ENGINEERING

362.1 (2.5.4.5.3) Provide a summary of the results of field density tests for compaction and moisture control of structural fill beneath and adjacent to Category I structures.

362.2 (2.5.4.5.1) Question 1 and the resulting discussion on page 8.00-1 included in Amendment Number 9 to your PSAR stated that all natural sands with relative densities less than 75% would be removed beneath all Class I structures and beneath non-Class I structures so sited that their failure could endanger the adjacent Class I structures. Discuss the methods employed in mapping and removing the sands having less than 75% relative density. Provide plan and sectional figures showing the areas where these materials were removed. Figure A9-2 of the PSAR which displays sub-surface profiles of Class I piping should be updated to show removal of sands of less than 75% relative density and be presented in the FSAR. Figure 2.5-21 of the FSAR shows loose sands beneath the Class I tanks although they were to have been removed. Explain this inconsistency, and provide proper documentation of as-built conditions.

362.3 (2.5.4.10.2.3) Reference is made in section 2.5.4.10.2.3 to Table 2.5-14 for design values of passive pressure. The table number is incorrect and should read Table 2.5-15.

362.4 (2.5.4.13) Provide the results of all benchmark survey measurements taken during construction. Graphically, compare the measured results to predicted settlements. Provide a commitment and schedule to submit the results of future survey settlement measurements.

362.5 (2.5.6.4.2) Provide gradation curves for the 12 inch thick crushed rock bedding layer beneath the riprap. Discuss the adequacy of the bedding material with respect to the requirements of a filter.

362.6 (2.5.6.5.3) Provide figures showing the failure surfaces that resulted in the minimum computed factors of safety for all slope stability conditions studied.

362.7 (2.5.6.5.4) Paragraph four of section 2.5.6.5.4 states that the outer slope of cross-section I was used to simulate the plant area fill and a seismic coefficient of .12g was used. However, Table 2.5-20 indicates that cross-section G was used for this condition. Explain and correct this inconsistency.

362.8 (2.5.6.8) Provide a detail of a typical piezometer as installed in the cooling pond dike. Also provide cross sections showing the development of the phreatic surface from initial piezometric head to full pond steady-state condition and a comparison to the phreatic surface assumed for the stability analysis of the steady-state condition.

371.0 HYDROLOGIC ENGINEERING

371.9
(2.4)
(RSP)

You state that areas adjacent to power plant structures and site drainage facilities are designed for a rainfall intensity of 6.1 inches per hour which corresponds to a 100-year precipitation event. In previous paragraphs, you state that the plant drainage is designed for the 24-hour Probable Maximum Precipitation (PMP) of 13.0 inches. You apparently conclude from this that the 100-year precipitation is more intense than the PMP, since 6.1 in./hr is more intense than 13.0 inches in 24 hours. This approach is incorrect. The 24-hour PMP must be broken down to appropriate time increments, suitable for the drainage areas and times of concentration which exist at the site.

It is our position that site drainage facilities (with times of concentration of about 10 minutes) be designed for the local PMP rainfall intensity of 20 in./hr. This intensity corresponds to a longer-period PMP, broken down into appropriate time increments for small drainage areas. Document the adequacy of your design by providing an adequate response to Request 371.1 using the rainfall intensity described above.

372.0 METEOROLOGY

372.12 (2.3.1) The design basis temperatures used for the design of the Midland plant heating, ventilating, and air conditioning systems are given as 96F dry bulb and 79F wet bulb for summer and -10F dry bulb for winter. For what duration of time would these temperature values have to be equalled or exceeded before operation of the heating, ventilating, and air conditioning systems would be affected?

372.13 (2.3.2) The onsite stability distribution for Midland (3/75-2/77) was compared to the stability distribution from several other sites, [Greenwood (9/72-9/73), Erie (11/73-10/75), Cook (5/75-4/76), Quanicassee (6/72-5/73)]. While Midland showed 57% D stability and 21% E stability, the other sites ranged from 25% to 38% for D stability and 33% to 47% for E stability. Although year to year variability may lead to a biased distribution, the 57% D stability found at Midland is based on two years of data. This lessens the possibility that the large amount of D stability is based on an anomalous year. Although the other four sites cover a variety of meteorological regimes, they all show similar stability distributions (including Quanicassee which is based on σ_θ). However, Midland does not show good agreement with these other sites. Discuss further the validity of the onsite stability distribution based on the onsite data at the Midland Plant. In particular, look at the method that is used to determine vertical temperature differences.

372.14 (2.3.2) Provide monthly joint frequency distributions based on the onsite data from the Midland plant for the 60-10 meter vertical temperature difference and 10 meter winds for the period March 1975 through February 1977.

372.15 (2.3.2) Provide yearly joint frequency distributions based on the Flint data and STAR stability classification for the years 1972 through 1977.

372.16 (2.3.3) Are temperature difference (ΔTs) measured directly or are they determined by subtraction of the temperature at two different levels?

372.17 (2.3.3) In the event an instrument outage renders meteorological data unreceivable by the teletype in the plant control room, "such data will be available via telephone". Where will the telephone supplied meteorological information come from and of what will it consist?

400.0 PROJECT MANAGEMENT

400.1 (1.6) (RSP) FSAR Section 1.6 indicates that you place reliance for the safe design of the plant upon several topical reports which have not yet been submitted for our review. We require, prior to issuing the SER for Midland Plant Units 1&2, that all such reports be submitted, our review completed, and changes for the Midland docket made as may be appropriate based upon our final approved version of the report. Accordingly, it is in your interest to expedite submittal of these reports consistent with your review schedule so that they can be scheduled consistently. Your intended submittal date for each such report should be indicated after your statement "to be submitted." Also, you should check Section 1.6 for completeness since several topicals referenced in subsequent text sections have been omitted (e.g., see request 232.1).

421.0 QUALITY ASSURANCE

421.1 Identify or reference those safety-related structures, systems, and components under the control of the Midland QA program.

421.2 In Topical Report CPC-1-A, Rev. 5 you commit to comply with WASH documents with certain acceptable alternatives. Since the WASH documents contain a number of draft standards which have been superseded and since the Midland FSAR was docketed on 11/18/77, there are Regulatory Guides and Regulatory Guide revisions issued prior to this docket date which apply and should be addressed in the Midland FSAR. These include:

RG 1.33 Rev. 1	RG 1.94, Rev. 1	RG 1.116 Rev. 0-R
RG 1.38 Rev. 2	1.123 Rev. 1	

Accordingly, revise your Report CPC-1-A, Rev. 5 to delete the commitment to the WASH documents and to provide a specific commitment to comply with the Regulatory Position of the following Regulatory Guides and the requirements of the following ANSI Standard:

RG 1.8, Rev. 1-R	RG 1.28, 6-7-72
RG 1.30, 8-11-72	RG 1.33, Rev. 1
RG 1.37, 3-16-73	RG 1.38 Rev. 2
RG 1.39, Rev. 2	RG 1.58, 8-73
RG 1.64, Rev. 2	RG 1.74, 2-74
RG 1.88, Rev. 2	RG 1.94, Rev. 1
RG 1.116, Rev. 0-R	RG 1.123, Rev. 1
ANSI N45.2.12, Draft 3, Rev. 4, 2-74	

Any exceptions, alternatives, or clarifications you believe warranted should be identified with sufficient supporting detail.

Your FSAR should then reference this revised topical by specific revision number and should include any plant specific exceptions or alternatives you consider appropriate.

422.0 CONDUCT OF OPERATIONS

422.3 (13.1.2.2) Describe the responsibilities and authority of your Staff Health Physicist, and Staff Chemist shown in Figure 13.1-3.

422.4 (13.1.3.1) Describe your qualification requirements for the positions of Staff Health Physicist, Staff Chemist, Quality Control Supervisor, and Electrical Supervisor.

422.5 (13.1.3.1) RSP We do not agree with, or need clarification regarding the qualification requirements described in Section 13.1.3.1 for several positions. Below is our request for clarification or a statement of our staff position, relative to these positions:

1. Operations Superintendent - It is our position that the Operations Superintendent should hold a senior operator's license, whether or not the Plant Superintendent has a senior operator's license.

2. Health Physicist - It is our position that three of the five years experience be as stated in Revision 1 to Regulatory Guide 1.8; i.e., "be applied radiation protection work in a nuclear facility...."

3. Chemical Engineer - This should be clarified to assure that the 1 year experience is in radiochemistry.

4. Maintenance Repairman "A"/Maintenance Electrician "A"/Senior Technician - Please clarify such that these positions are comparable to those described in Sections 4.5.2 and 4.5.3 of ANSI N18.1-1971; i.e., technicians and repairmen in <u>responsible</u> positions.

422.6 (13.4.3.3) Describe the specific position title of each member of your Safety and Audit Review Board (SARB) or describe the qualification requirement for members of the SARB .

441.0 OPERATOR LICENSING BRANCH: TRAINING SECTION

441.1 (13.2.1) Amend section 13.2.1.2, to include a commitment that refresher training for non-licensed personnel shall be periodic and conducted at least every two years. This training shall include, at a minimum, refresher training on administrative, radiation protection, emergency and security procedures.

441.2 (13.2.1) Amend section 13.2.1.5.1, to include a commitment that periodic written quizzes will be administered at the conclusion of each lecture or series of lectures.

441.3 (13.2.1) Change Item (e) in Section 13.2.1.5.3.1 from the present wording to, "Any significant (10%) power change in manual rod control." Remove the statement that refers to these reactivity changes as examples and provide a commitment that these listed reactivity changes are those acceptable to meet the operator requalification program.

441.4 (13.2.1, Amend Section 13.2.5.6, to provide a commitment that not more than two licensed individuals who administer the examination shall be exempt from the annual written examination.

441.5 (13.2.1) In section 13.2.1.5.7, it is stated that the operational evaluation can be completed at a simulator. This evaluation at the simulator is only acceptable if the requirements of 10 CFR 55 Appendix A Item (3) (d) are met. If these requirements cannot be met, provide a commitment that the operational evaluation will also include evaluation of performance at the plants.

441.6 (13.2.1) Amend Section 13.2.1.5.8, to include a commitment that records shall be maintained to include copies of written examinations, the answers given by the licensee, results of evaluations, and documentation of any additional training administered.

441.7 (13.2.1) Amend 13.2.1.5.2, to include a commitment that all licensed individuals shall be cognizant of facility design changes, procedure changes and facility license changes and that auditable records are maintained.

442.0 OPERATOR LICENSING BRANCH: PROCEDURES SECTION

442.1 (13.5) In section 13.5.3, include a commitment that the administrative procedures shall include the requirements to meet 10 CFR Part 50.54 (i), (j), (k), (e), and (m).

442.2 (13.5) Provide a commitment that the administrative and operating procedures shall be completed six months prior to fuel loading and revise Figure 13.5-1 to reflect this commitment.

REQUEST FOR ADDITIONAL INFORMATION ______MIDLAND______

Distribution:
NRC PDR
Local PDR
Docket File
LWR #4 File
R. S. Boyd
R. C. DeYoung
D. B. Vassallo
F. J. Williams
S. A. Varga
Project Manager ______DARL HOOD______
M. Service
R. J. Mattson
D. Ross
J. Knight
R. Tedesco
H. Denton
V. A. Moore
R. H. Vollmer
M. L. Ernst
W. P. Gammill
W. McDonald
ELD
IE (3)
W. Haass

bcc: ACRS (16)
T. Abernathy
J. R. Buchanan