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

November 11, 1977

Docket Nos. 50-329
50-330

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

Gentlemen:

SUBJECT: ACCEPTANCE REVIEW OF FSAR

On August 29, 1977, you tendered an application for operating licenses for Midland Plant Units 1 and 2, filed as Amendment 33 to your application for construction permits and operating licenses. Based upon our acceptance review of the tendered Final Safety Analysis Report (FSAR) and our previous approval of your planned deferral of the Environmental Report, we have concluded that your application is sufficiently complete for us to initiate our detailed review of your FSAR and to permit us to establish a review schedule.

Accordingly, your filing of the application should include three (3) originals signed under oath or affirmation by a duly authorized officer of your organization. In addition, your filing should include fifteen (15) copies of that portion of the application containing the general information and forty (40) copies of the safety analysis report. No additional copies of the physical security plan are required at this time. As required by Section 50.30 of 10 CFR Part 50, you should retain an additional ten (10) copies of the general information and thirty (30) copies of the safety analysis report for direct distribution in accordance with Enclosure 1 to this letter and further instructions which might be provided later. Within 10 days after docketing, you must provide an affidavit that distribution has been made in accordance with this enclosure. All subsequent amendments to the safety analysis report will require sixty (60) copies for distribution.

The conclusion of our acceptance review performed pursuant to Section 2.101 of 10 CFR 2 that the tendered FSAR is sufficiently complete is based upon all of the information filed, taken as a whole. However, during the course of our review of the FSAR, Enclosure 2 was generated to request additional information. These requests are of the type which require an early response for our mutual benefit during the ensuing detailed technical review period. Of particular schedular importance are the requests for additional information in the electrical area. You are therefore requested to transmit complete responses within 5 weeks after your application has been docketed. If you are unable to meet this schedule, please inform us within 7 days after receipt of this letter so that we may incorporate this into our review schedule which is being developed. You will be advised of key milestones of the review as soon as our schedule development is complete.

We note that specific dates are established within your FSAR and within your physical security plan for supplying certain additional information. These dates are being considered further during development of our detailed review schedule. The need for improvements in some of your dates are indicated in Enclosure 2. You will be advised of the need for any further improvements in this regard upon issuance of our schedule. In the interim, our schedule development is proceeding based upon your tendering of the Environmental Report by March 1, 1978, and adhering to the dates presently stated in your FSAR and security plan.

If, during the course of our review, you believe there is a need to appeal a staff position because of disagreement, this need should be brought to the staff's attention as early as possible so that 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 opportunity for applicants to discuss, with management, areas of disagreement in the case review.

Please contact us if you desire discussion or clarification of Enclosure 2 or have questions regarding your application.

Sincerely,

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

Enclosure:
Request for Additional
Information

cc:
See next page

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ENCLOSURE 1

FEDERAL, STATE, AND LOCAL OFFICIALS TO WHOM APPLICATION, SAR, AND AMENDMENTS SHOULD BE SERVED

Federal

U.S. Environmental Protection Agency
Federal Activities Branch
Region V Office
Attention: EIS Coordinator
230 South Dearborn Street
Chicago, Illinois 60606

State Official

Executive Office of the Governor
Division of Intergovernmental Relations
Lewis Cass Building
Lansing, Michigan 48913

Local Officials

Mr. Robert B. Chatterton
Supervisor of Midland Township
928 Clarence Court, Route 7
Midland, Michigan 48640

Mr. Earl D. Morris, Chairman
Midland County Board of Commissioners
Midland, Michigan 48460

National Laboratory

Mr. Phillip F. Gustafson, Manager
Environmental Statement Project
Argonne National Laboratory
9700 South Cass Avenue
Argonne, Illinois 60439

ENCLOSURE 2

REQUEST FOR ADDITIONAL INFORMATION

RESULTING FROM

ACCEPTANCE REVIEW OF

MIDLAND PLANT UNITS 1 & 2 FSAR

This request for additional information was developed during the acceptance review of the Midland Plant Units 1 & 2 FSAR. The requests 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 request 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

010.1 (3.4.1) Section 3.4 of your FSAR states that the number of openings in walls and slabs below flood level have been kept to a minimum. Describe the means of sealing each type of opening below the flood level to prevent flood damage to systems, components and structures important to safety.

0.10.2 (3.4.2) Your failure analysis in Section 3.4.2.5 for the circulating water system states that the flooding analysis was performed using the moderate energy failure criteria to determine that no safety related equipment will be affected. However, the leak rate from a moderate energy crack is very conservative for the circulating water system, since a rupture of the expansion joints to the condenser would result in a full circulating water system flowrate into the turbine building. Reanalyze your flooding protection in the turbine building based on an expansion joint rupture and demonstrate that no safety related equipment will be affected from the resulting flood. Your analysis should include the following:

1. The maximum flow rate through a completely failed expansion joint.

2. The potential for and the means provided to detect a failure in the circulating water transport system barrier such as the rubber expansion joints. Include the design and operating pressures of the various portions of the transport system barrier and their relation to the pressures which could exist during malfunctions and failures in the system (rapid valve closure).

3. The time required to stop the circulating water flow (time zero being the instant failure) including all inherent delays such as operator reaction time, drop out times of the control circuitry and coastdown time.

4. For each postulated failure in the circulating water transport sytem barrier, give the rate of rise of water in the associated spaces and total height of the water when the circulating water flow has not been stopped or overflows to site grade.

5. For each flooded space provide a discussion, with the aid of drawings, of the protective barrier provided for all essential systems that could become affected as a result of flooding. Include a discussion of the consideration given to passageways, pipe chases, and/or the cableways joining the flooded space to the spaces containing safety related system components. Discuss the effect of the flood water on all submerged essential electrical systems and components.

010.3
(3.5)
Provide a tabulation of all safety-related components which are located outdoors and describe the protection to be afforded to these components to prevent their being damaged by tornado generated missiles or a seismic event. Include in this tabulation all HVAC system air intakes and exhausts and the diesel generator combustion air intake and exhaust. Identify the locations of the air intakes and exhausts for these components on the plant arrangement drawings.

010.4
(3.6)
RSP
Section 3.6 of your FSAR does not indicate which design bases and criteria were used in your analyses to provide protection against high and moderate energy piping failures in fluid systems outside containment. Show how you are providing such protection in accordance with the criteria of our Branch Technical Position APCSB 3-1 "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

010.5
(3.6)
RSP
Our review of your main steam and feedwater systems indicates the need for the following changes or additional information:

(1) Your design of the main steam lines has approximately 90-100 ft of piping between the containment and the main steam isolation valves. Additionally, this 90-100 feet of pipe, which contain many bends, is proposed to be within a break exclusion area. It is our position that outdoor piping shall be protected from tornado missiles even if it meets the "superpipe" criteria of Branch Technical Position MEB 3-1. A tornado missile could result in a full pipe break with resulting jet impingement and pipe whip on the auxiliary building roof. Provide a design that can withstand the effects of tornado missiles and demonstrate that a safe cooldown will not be precluded following a tornado missile.

(2) For the feedwater system break exclusion area, it is our position that the surrounding structure housing this piping and all safety related equipment within the structure shall be designed for the environmental effects (pressure, temperature and flooding) of a break equivalent to the full flow area of a single ended rupture. Revise your design to conform with this position. Demonstrate that adequate protection of the control room is provided from the full effects of pipe break of the non-seismic piping in the turbine building, adjacent to the control room.

010.6 (9.0) In regards to potential failures or malfunctions caused by freezing, icing, and other adverse environmental conditions, discuss the protective measures that are provided to assure the proper function of those components not housed within temperature controlled areas and that are essential in attaining and maintaining a safe reactor cold shutdown.

010.7 (9.0) Provide a tabulation of all valves in the reactor coolant pressure boundary and in other seismic Category I systems, per Regulatory Guide 1.29, (e.g., safety valves, relief valves, stop valves, stop-check valves, and control valves) whose operation is relied upon either to assure safe plant cold shutdown or to mitigate the consequences of an accident. The tabulation should identify the sytem in which it is installed, the type and size of valves, the actuation type(s), and the enviornment of conditions to which the valves are qualified.

010.8 (9.1.1, 9.1.2) Provide drawings, which include the dimensionsal details, of the new and spent fuel storage racks. These drawings should show that the spent fuel racks will protect the fuel assemblies from dropped objects.

010.9 (9.1.3) RSP Provide the decay heat release rate due to the spent fuel assemblies for the normal storage conditions (2/3 of a core) and for abnormal storage conditions (1-2/3 cores). In calculating this decay energy, we require use of the methods set forth in our Branch Technical Position APCSB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling."

010.10 (9.1.4) In your analysis of the refueling cask drop accident, describe the safety related equipment that is being protected by your administrative controls of the cask handling crane and by the crushable pad in the storage pit. Also discuss and provide drawings to show that the cask could not drop and tip into the spent fuel pool. Provide the results of your analysis to show the crushable pad can withstand the impact from the dropped cask.

010.11 (9.2.) Figure 9.2-8, component cooling water, shows seismic Category I classification transition points in the piping between the reactor coolant pump supply header and the seal return coolers with no isolation valve at the seismic transition point. Explain the rational for this design and demonstrate that a failure in the non-seismic piping will not affect safe plant cold shutdown.

010.12 (9.2.2) RSP The design of your component cooling water system provides a single supply and return line, supplying cooling water to four reactor coolant pumps. These lines are not designed to seismic Cagegory I requirements and contain motor-operated valves for containment isolation. The seals and bearings of the reactor coolant pumps require continuous cooling by the component cooling water system during all modes of operation. Inadvertent closure of any one of the above motor-operated valves would terminate the coolant flow to all of the pumps which potentially may lead to fuel damage, due to a locked rotor. Therefore, it is our position that you revise your design of this portion of the component cooling water system so that the following criteria are met:

1. A single failure in the component cooling water system shall not result in fuel damage or damage to the reactor coolant pressure boundary caused by an extended loss of cooling to the reactor coolant pumps. Single failure includes operator error, spurious actuation of motor-operated valves, and loss of component cooling water pumps.

2. A moderate energy leakage crack or an accident that is initiated from a failure in the component cooling water system piping shall not result in excessive fuel damage or a breech of the reactor coolant pressure boundary when an extended loss of cooling to the reactor coolant pumps occurs. A single active failure shall be considered when evaluating the consequences of the accident. Moderate leakage cracks should be determined in accordance with the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment."

You may elect one of two approaches to meet the two above criteria:

Approach 1:

That portion of the component cooling water system which supplies cooling water to the reactor coolnat pumps can be designed to non-seismic Category I requirements and Quality Group D if you demonstrate that the reactor coolant pumps are capable of operating with loss of cooling for longer than 30 minutes without loss of function and the need for operator protective action. Also, for this approach, safety grade instrumentation to detect the loss of component cooling water to the reactor coolant pumps and to alarm to the operator in the control room is to be provided. The entire instrumentation system, including audible and visual status indicators for loss of component cooling water shall meet the requirements of IEEE Std 279-1971.

Approach 2:

If you cannot demonstrate that the reactor coolant pumps will operate longer than 30 minutes without loss of function or operator corrective action, then your design must meet one of the following two requirements for the entire component cooling water system:

1. Safety grade instrumentation consistent with the criteria for the protection sytem shall be provided to initiate automatic protection of the plant. For this case, the component cooling water supply to the seal and bearing of the pumps may be designed to non-seismic category requirements and Quality Group D; or

2. The component cooling water supply to the pumps shall be capable of withstanding a single active failure or a moderate energy line crack as defined in our Branch Technical Position APCSB 3-1 and be designed to seismic Category I, Quality Group C and ASME Section III, Class 3 requirements.

010.13 (9.2.5) Revise your FSAR to include the information on Regulatory Guide 1.27 provided by your letter of February 3, 1976 in response to our request 020.1 for additional information regarding heat transient analyses for your ultimate heat sink. Our prior request 020.1 asked for the following information:

"In order to permit an evaluation of the ultimate heat sink and other heat removal systems, provide an analysis of the thirty-day period following a design basis accident listing the total heat rejected, the sensible heat rejected, the station auxiliary system heat rejected, and the decay heat release from the reactor.

In submitting the results of the analysis requested, include the following information in both tabular and graphical presentations:

1. The total integrated decay heat.

2. The heat rejection rate and integrated heat rejected by the station auxiliary systems, including all operating pumps, ventilation equipment, diesels and other heat sources.

3. The heat rejection rate and integrated heat rejected due to sensible heat removed from containment and the primary system.

4. The total integrated heat rejected due to the above.

5. The maximum allowable inlet water temperature taking into account the rate at which the heat energy must be removed, cooling water flow rate, and the capabilities of the respective heat exchangers.

6. The available NPSH to the service water pumps at the minimum Ultimate Heat Sink water level bs. the required NPSH.

The above analysis, including pertinent backup information, should demonstrate the capability of the ultimate heat sink to provide adequate water inventory and provide sufficent heat dissipation for the safe shutdown and cooldown of both units following a LOCA in one unit.

Use the methods set forth in our Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," to establish the input due to fission product decay and heavy element decay. Assume an initial service water temperature based on the most adverse conditions for normal operation."

010.14 (9.2.6) Your condensate storage tanks are not seismic Category I and are not protected from tornado missiles. (see related request 010.5) In the event the condensate storage tank were lost due to a seismic event or a tornado the auxiliary feedwater pumps could automatically start without any water supply and damage all the auxiliary feedwater pumps and prevent safe plant shutdown and cooldown.

a. Following a seismic event, coincident with a loss of offsite power, demonstrate how the auxiliary feedwater pumps will be protected from damage and show how a safe cold reactor shutdown will be attained.

b. A tornado could result in a main steam or feedwater line break since these lines are unprotected, in addition to the loss of offsite power and the loss of the condensate storage tank. Demonstrate how a safe cold shutdown of the reactor will be attained following a tornado which results in a main steam or feedwater line break.

Discuss in detail the design changes necessary to mitigate the consequences of (a) or (b) above and to premit a safe final stabilized condition of the plant following these events.

010.15 (9.3.1) Provide a list of all safety related air operated equipment and valves and describe their failure mode upon loss of air. If any components require a safety related accumulator to perform their safety functions, provide a P&ID typical of each type of accumulator system used.

010.16 (9.3.9) Demonstrate that the failure of any high pressure miscellaneous gas storage system cannot result in damage to safety related equipment. Show that the locations of these storage systems are such that any resulting missiles will not affect any safety related equipment.

010.17 (9.3.1, 10.3) RSP Section 9.3.1 indicates that the compressed air system is not designed to seismic Category I requirements and Section 10.3 indicates that the atmospheric dump valve will fail shut in the event of loss of air supply. It is our position that the atmospheric steam dump valves shall be able to be operated from the control room for cold shutdown of the plant following a steam line break coincident with loss of offsite power. Also, a seismic Category I air supply (or actuator) to the steam dump valves shall be provided.

010.18 (10.4.9) RSP It is our position that the power sources for all controls, valve operators and other supporting systems (e.g., pump lube oil cooling sytem) associated with the turbine driven auxiliary feedwater pump shall be independent of all A/C power sources. This is to comply with the diversity requirements of our Branch Technical Position APCSB 10-1. Modify the system design to comply with this position and confirm that the turbine driven pump lube oil cooler will receive cooling water from the pump recirculation line.

022.0 CONTAINMENT SYSTEMS BRANCH

022.1 (6.2) Provide the following information regarding the environmental qualification of safety related equipment:

a. Provide a comprehensive list of equipment required to be operational in the event of a main steam line break (MSLB) accident to mitigate the accident consequences and assure a safe shutdown of the plant. The list should include, but not necessarily be limited to, the following safety related equipment:

1. Electrical containment penetrations
2. Pressure transmitters
3. Containment isolation valves
4. Electrical power cables
5. Electrical instrumentation cables
6. Level transmitters

Describe the qualification testing that was done, including the test environment, namely, the temperature, pressure, moisture content, and chemical spray as a function of time.

b. It is our position that the thermal analysis of safety related equipment which may be exposed to the containment atmosphere following a main steam line break accident should be based on the following:

(1) A condensing heat transfer coefficient based on the recommendations in Branch Technical Position CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," should be used.

(2) A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period it is appropriate to use a conservatively evaluated forced convection heat transfer correlation. For example:

$$Nu = C(Re)^n$$

where

Nu = Nusselt No.

Re = Reynolds No.

C,h = emperical contants dependent on geometry and Reynolds No.

Since Reynolds number is dependent on velocity, it is necessary to evaluate the forced flow currents which will be generated by the steam generator blowdown. The CVTR experiments provide limited data in this regard. Convective currents of from 10 ft/sec to 30 ft/sec were measured locally. We recommend that the CVTR test results be extrapolated conservatively to obtain forced flow currents to determine the convective heat transfer coefficient during the blowdown period. After the blowdown has ceased or has been reduced to a negligbly low value, a natural convection heat transfer correlation is acceptable.

c. For each component where thermal analysis is done in conjunction with an environmental test at a temperature lower than the peak calculated temperature following a main steam line break accident, compare the test thermal response of the component with the accident thermal analysis of the component. Provide the basis by which the component thermal response was developed from the environmental qualification test program. For instance, graphically show the thermocouple data and discuss the thermocouple locations, method of attachment, and performance characteristics, or provide a detailed discussion of the analytical model used to evaluate the component thermal response during the test. This evaluation should be performed for the potential points of failure such as thin cross-sections and temperature sensitive parts where thermal stressing, temperature-related degradation, steam or chemical interaction at elevated temperatures, or other thermal effects could result in the failure of the component mechanically or electrically. If the component thermal response comparison results in the prediction of a more severe thermal transient for the accident conditions than for the qualification test, provide justification that the affected component will perform its intended function during a MSLB accident, or provide protection for the component which would appropriately limit the thermal effects.

022.2 (6.2) In the unlikely event of a pipe rupture inside a major component subcompartment, the initial blowdown transient would lead to non-uniform pressure loadings on both the structure and the enclosed component(s). To assure the integrity of these design features, we require that you perform a subcompartment, multi-node pressure response analysis, and provide the following information:

a. Provide the results of analyses of the pressure transient resulting from postulated hot-leg and cold-leg (pump suction and discharge) reactor coolant system pipe ruptures within the reactor cavity, pipe penetrations, and steam generator compartments. Provide the results of similar analyses for the pressuirzer surge and spray lines, and other high energy lines located in containment compartments that may be subject to pressurization.

b. Provide and justify the pipe break type, area, and location for each analysis. Specify whether the pipe break was postulated for the evaluation of the compartment structural design, component supports design, or both.

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

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

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

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

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

h. Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressure on structures and components.

i. For the compartment structural design pressure evaluation, provide the peak calculated differential pressure and time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied. If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to assure that regions removed from the break location are conservatively designed.

j. Provide the peak and transient loading on the major components used to establish the adequacy of the supports design. This should include the load forcing functions (e.g., $f_x(t)$, $f_y(t)$, $f_z(t)$) and transient moments (e.g., $M_x(t)$, $M_y(t)$, $M_z(t)$) as revolved about a specific, identified coordinate system. Provide the projected area used to calculate these loads and identify the location of the area projections on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made.

022.3 (6.2) Section 6.2.1.1.3.a of Regulatory Guide 1.70, Revision 2, requires that a description of the method of analysis used to determine the pressure and temperature in the containment be provided in the Safety Analysis Report. However, your SAR references the Bechtel Topical Report BN-TOP-3. This topical report has not been approved by the staff at this time and in its present form is unacceptable as a reference. Therefore, provide a discussion of the methods used to determine your containment pressure and temperature response analyses.

022.4 (6.2) The containment sump design does not comply with the recommendations of Regulatory Guide 1.82 (see Table 6.2-23). Provide justification for deviating from the recommendations of Regulatory Guide 1.82.

022.5 (6.2) Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant OPerations," provides guidance with regards to the design of the containment purge system. Discuss how your containment purge system design complies with the recommendations of BTP CSB 6-4.

031.0 INSTRUMENTATION AND CONTROL SYSTEMS BRANCH

031.1 (3.10) Section 3.10 of Regulatory Guide 1.70, Revision 2, requires that, "All Seismic Category I instrumentation, electrical equipment, and their supports should be identified." The information provided in your FSAR is insufficient both in scope and detail:

1. Your listing of approximately 40 categories and items is inadequate for meeting the information requirements identified in RG 1.70. Expand this list to include all the seismic category instrumentation, electrical equipment, and their supports.

2. Of the 40 categories and items listed in Section 3.10, 28 cases provide no supporting discussion regarding method of qualification or method of analysis. You state that this information will not be available in some cases until mid-1978. We require this information promptly if our anticipated review schedule is to be maintained.

031.2 (3.11) Your information concerning the identification of instrumentation, control, and electrical equipment to be environmentally qualified is incomplete. Your statement in the first paragraph of Section 3.11 that "By August, 1978, 95% of the information will be added while the remaining 5% will be provided in the period of August 1978 to April 1979," is unacceptable. We require that this information be provided prior to these dates if our anticipated review schedules are to be maintained.

031.3 (App 3.A) Your conformance discussion in Appendix 3A to certain recent Regulatory Guides merely states that these regulatory guides are "issued for comment," whereas they have now been reissued as Revision 1. Electrical Guides in this category are: R.G. 1.100, R.G. 1.106, and R.G. 1.108. Update this list of regulatory guides and supplement your FSAR to demonstrate how these regulatory guides have been implemented. Describe and justify any alternate approaches or design features you propose.

031.4 (7.1) Clarify in Section 7.1 how the 13 systems and categories of equipment are related to the list of safety systems in Table 7.1-1.

031.5 (7.1.1) Your Section 7.1.1 references Table 7.1-1 for identification of safety-related systems. Table 7.1-1 is insufficient for providing this information. Modify Table 7.1-1 to provide the information required by Section 7.1.1 of R.G. 1.70.

031.6 (7.1.2) Systems required for safe shutdown are not sufficiently defined in Section 7.1.2 "Identification of Safety Criteria." Your referencing of whole chapters and your lack of system definition is unacceptable. Referencing suitable portions of the FSAR is acceptable; however, the system identity and the appropriate section and subsection must be indicated. Revise your FSAR to provide the information required by Section 7.1.2 of RG 1.70.

031.7 (7.2) (1.7) Only five figures in Section 7.2.1.2.2 are identified as being "Final System Drawings." Since these are the only figures identified, your intent in regards to the drawings listed in Section 1.7 and the attendant tables is not clear. It would appear that many of the drawings in Section 1.7 are "Final System Drawings" and should be referenced as required by Section 7.2.1.3 of R.G. 1.70.

In regards to Section 1.7 "Electrical, Instrumentation and Control Drawings, further clarification is required as to which sections many of the drawings are associated. Your statement on page 1.7-1 of the FSAR, "When appropriate, reference is made to the specific paragraphs in the text which discuss the drawings," has not been adequately implemented. There are no references in the text of Chapter 7 to many of the drawings in Section 1.7. If it is your intent to utilize these drawings to satisfy drawing requirements discussed in Sections 7.2.1.3, 7.3.1.3, and elsewhere in R.G. 1.70, they should be referenced in the appropriate portion of the text.

031.8 (7.4.1) Item C of Section 7.4.1.1.6 states, "Control of the pressurizer heaters is required to ensure the capability of maintaining reactor coolant pressure during safe shutdown." However, your pressurizer heater controls have been classified as non-safety and are discussed in Section 7.7.1.8. Since the function of the pressurizer heaters is safety-related, it is inconsistent that their controls could be non-safety related. Expand the appropriate subsections of Section 7.4 to address this subject as required by Section 7.4.1 of R.G. 1.70 and justify the conflict of safety category between the heaters and their controls.

031.9 (7.5.1) (7.2) You list Reactor Trip in Section 7.5, "Safety Related Display Instrumentation," but Subsection 7.5.1.1.1 states that the RPS display, including reactor trip breaker status, is non safety-related. Your latter statement appears to be in error and is in conflict with the requirements of Paragraph 4.20 of IEEE Standard 279-1971. Expand Section 7.2 to address these indications. Justify any exceptions you have to the requirements of Paragraph 4.20 of IEEE Std 279-1971.

031.10 (7.2) (7.8) Subsection 7.8.1, "Nuclear Instrumentation," states that the power range detectors are required by the RPS to perform safety functions and are part of the RPS. Further discussion in 7.8.1 indicates that the power instrumentation is but one portion of the nuclear instrumentation system. Supporting RPS Figure 7.2-1 refers the user to the "NI System," however, there is no reference to the NI System drawings under Subsection 7.2.1.2.2 "Final System Drawing." Correct and clarify these conflicts in your FSAR.

As part of your response, address the following:

1. Since these power range detectors are primary RPS trip inputs, provide the justification for documenting the system information concerning the dectectors in Section 7.8 "Other Instrumentation Systems" versus Section 7.2 as required by R.G. 1.70.

2. Section 7.8-1, provided to support the text, is inadequate for use as a Final System Drawing. Systems drawings must be provided in the FSAR as required by R.G. 1.70.

3. Figure 7.8-1 labelled "Typical Arrangement" is insufficient to determine the extent or lack of commonality between the safety and the non-safety portions of the Nuclear Instrumentation. Provide the system level drawings and sketches as required by R.G. 1.70 in sufficient detail for the staff to make this determination.

4. Describe how the requirements of IEEE Standard 279-1971 have been implemented in the design of the safety-related portion of the Nuclear Instrumentation. The information required by Sections 7.2.1.2 and 7.2.2 of R.G. 1.70 provides an acceptable basis for this description.

040.0 POWER SYSTEMS BRANCH

040.1 (8.2) Sufficient system description and analyses have not been provided to demonstrate compliance with 10 CFR Part 50 and the General Design Criteria in Appendix A to 10 CFR Part 50 as required by Section 8.2.1 of Regulatory Guide 1.70. Expand the system descripton and analysis for the following areas (see SRP 8.2, Part II, Item 1):

1. The single battery of the 138 Kv Tittabawassee switching station that provides control circuit power for both preferred power sources;

2. The circuitry that transfers power to the Class IE distribution system from the main generator supply to the preferred power system;

3. The physical relationship between the preferred offsite power circuits from the transmission network and the onsite Class IE distribution system.

040.2 (8.2) Provide additional analyses to demonstrate that the single failure of the 138 KV Tittabawasse switching station battery will not jeopardize the independence of the preferred offsite power circuits.

040.3 (8.2) Provide description and analyses of the switching circuitry and sequencing circuitry used to transfer power from the main generator supply to the preferred power system. The analyses shall demonstrate that no single failure will jeopardize the independence of the preferred power sources of the redundant Class IE distribution system.

040.4 (8.2) Provide scaled drawings which clearly show the physical relationship between preferred offsite power transmission lines and that circuitry which is:

1. Within the Tittabawassee 138 KV switchyard and 345 KV substation, and

2. From and including the startup transformers to the 4.16 KV buses numbered 1D2, 1C2, 2D2 and 2C2.

040.5 (8.3.1) Table 1.3-1 indicates that the Midland emergency diesel generators are rated at 5,250 KW each, but that similar plants have emergency diesel generators are rated at 5,250 KW each, but that similar plants have emergency diesel generators rated at approximately 2,500 KW. Explain the reasons for the higher KW rating for the Midland Plant.

040.6 (9.5.4) Provide a diagram showing where the underground emergency diesel fuel oil tanks are located on the plant site relative to the other buildings.

040.7 (9.5.4) Figure 9.5-25, Emergency Diesel Generator Fuel Oil System P&ID, does not indicate filters or instrumentation for measuring the pressure drop across any filters in the discharge line from each seven day storage tank or engine day tank. Describe the means that are provided (if any) for filtering and monitoring filters for cleanliness. Also, indicate this information on the P&ID.

040.8 (10.2) Discuss the following in your turbine generator section:

1. The valve closure times and the arrangement for the main steam stop and control valves and the reheat stop and intercept valves, in relation to the effect of a failure of a single valve on the overspeed control functions;

2. The valve closure times and extraction steam valve arrangements in relation to stable turbine operation after a turbine generator system trip; and

3. Effects of missiles from a possible turbine-generator failure on safety related systems or components. (see SRP 10.2, Part III, Items 3, 4).

040.9 (10.2) (10.4.1) Discuss the effects of a high and moderate energy piping failure, or failure of the connection from the low pressure turbine to the condenser, on nearby safety related equipment or systems. Discuss the protection that is provided the turbine overspeed control system equipment, electrial wiring, and hydraulic lines from the effects of a high or moderate energy pipe failure. This protection should assure that the turbine overspeed protection system will not be damaged so as to prevent performance of its safety function. (see SRP 10.2, Part III, Item 8; SRP 10.4.1, Part III, Item 3a).

040.10 (10.2) (10.4.10) Discuss the effects on nearby safety-related equipment or systems resulting from a high and moderate energy failure of the piping from the secondary steam cycle to and from the process steam evaporator system. Identify and discuss the protection that is provided the turbine overspeed control system equipment, electrical wiring and hydraulic lines from the effects of a high or moderate energy pipe failure. This protection should assure that the turbine overspeed protection system will not be damaged so as to prevent performance of its safety function. Provide a figure showing the location of the piping relative to the turbine generator. (See SRP 10.2, Part III, Item 8).

040.11 (10.4.10) Section 10.4.10 states "The Dow Chemical Company recovers part of the tertiary steam condensate and returns it to provide about 60% of the total feedwater flow rate". Additional information on the remaining 40% required for this tertiary feedwater is required. Indicate:

1. Where it is obtained,

2. How it is treated, and

3. How the secondary system steam cycle is affected by this operation.

040.12 (10.4.13) Show where the auxiliary boiler and its fuel tanks are located in relation to the rest of the plant.

110.0 MECHANICAL ENGINEERING BRANCH

110.1
(3.6.2.1)

The information presented in FSAR Section 3.6.2.1.4 is not complete. In addition to the commitments already made with regards to containment penetration piping, the staff requires a commitment to an augmented inservice inspection program as follows:

(a) The protective measures and structures in the containment penetration area should not prevent the access required to conduct the inservice examination specified in the ASME Code, Section XI, Division 1.

(b) For those portions of fluid system piping in the containment penetration area, the extent of inservice examinations completed during each inspection interval (IWA-2400, ASME Code, Section XI) should provide 100 percent volumetric examination of circumferential and longitudinal pipe welds within the boundary of these portions of piping.

(c) The areas subject to examination should be defined in accordance with Examination Categories B-F and B-J for Class 1 piping welds in Table IWB-2500 and categories C-F and C-G for Class 2 piping welds in Table IWC-2520.

110.2
(3.6.3.1)

When breaks in high energy Class 1 piping are postulated due to the cumulative usage factor (CUF) being greater than 0.1, the staff position is that both a circumferential and longitudinal break shall be postulated at that location. Your position on this subject is unacceptable without further clarification and justification. The staff will have to consider your position on a location by location basis based on the state of stress at the given location.

110.3
(3.6.3.2)

For calculating the piping reaction forces due to postulated pipe breaks, including LOCA, the staff has documented the following positions in Standard Review Plan 3.6.2 (1975).

(a) A rise time not exceeding one millisecond should be used for the initial pulse, unless longer crack propagation times or rupture opening times can be substantiated by experimental data or analytical theory.

(b) When the steady-state jet thrust force is of the form
$T = KpA$, where
K = thrust coefficient
p = system pressure prior to pipe break
A = pipe break area,

110.3 (3.6.3.2) K shall not be less than 1.26 for steam-water mixtures or 2.0 for sub-cooled, non-flashing water.

FSAR Section 3.6.3.2 states that for NSSS piping analysis a longitudinal break opening time of 10 msec and thrust coefficients of 1.0 and 0.61 were used. These values and any other parameters in conflict with the staff position above will require technical justification before they will be acceptable. You should note that this staff position also holds for calculating the force of the fluid jet that emerges from a postulated break. This may have some bearing on the Class 1 jet impingement study scheduled for May 1978.

110.4 (3.9.1.2) The information provided in FSAR Section 3.9.1.2 is not complete. Further verification of the accuracy of the computer programs used in the analysis of seismic Category I items is required. For those programs listed below provide information as follows:

(1) If this particular program version has been previously accepted by the NRC staff, provide reference to the plant application, topical report, etc. for which it was accepted.

(2) Otherwise, provide a verification report for each program version. This verification report should provide a comparison of results to a series of test problems with previously accepted programs, experimental data, hand calculations, or analytical results published in the technical literature. Preferably, these comparisons will be summarized in graphical form.

The computer programs for which the above information is required are:

Part I, Table 3.9.18

ABSA 3

ANSYS (only if used for any non-linear analysis)

Frame Analysis

ME-101
ME-632
ME-660
ME-661
ME-662
ME-913

STARS
SSHOCK

110.4 Part II, Table 3.9.18

91035 - Closure Analysis

91060 - General Interaction Analysis for Shell of Revolution with Axisymmetric Loading

91217 - Analysis of Nozzles, Manways, and Cover Plates

91232 - Classical Solution for Thermal Stress + Temperature in Long Hollow Cylinder with Option for Fracture Mechanics

91249 - OTSG Tubesheet Program

110.5 (3.9.4) Topical report BAW-10029, Rev. 3 has been referenced for purposes of design criteria and performance assurance of the Midland 1 & 2 control rod drive machanisms. As outlined in the staff evaluation of this topical report, the following additional information must be provided:

(1) Data should be provided to show that loads on the CRDM's due to seismic or postulated pipe break events at the Midland plant are enveloped by the loads used in the topical report.

(2) The performance assurance program of BAW-10029, Rev. 3 demonstrates an expected CRDM service life of at least 20 years. Indicate a recognition of this fact since the plant design life is 40 years.

110.6 (3.9.4) Provide the basis for selecting the location, required load capacity, structural and mechanical performance parameters of safety related hydraulic snubbers and achieving a high level of operability assurance including:

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

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

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

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

121.0 MATERIALS ENGINEERING BRANCH - MATERIALS INTEGRITY SECTION

121.1 (5.2.4) (5.4.1) (6.6) (16.0)

We will require that the inservice inspection (ISI) program for ASME Code Class 1, Class 2 and Class 3 components be in accordance with 10 CFR Part 50, paragraph 50.55a(g).

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

A preservice inspection plan.

An ISI plan based on currently applicable requirements.

An updated ISI plan submitted within six months of anticipated commercial operation.

(1) The preservice inspection plan will be reviewed to support the safety evaluation report finding on ISI. The basis for the determination will be compliance with the Edition of Section XI of the ASME Code stated in the FSAR, or later Editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply, and all augmented examinations established by the Commission when added assurance of structural reliability was deemed necessary. Examples of augmented examination requirements can be found in NRC positions on high energy fluid systems in SRP Section 3.2, turbine disk integrity in SRP Section 10.2.3, and steam generator tubing in Regulatory Guide 1.83. Your response should define the applicable Section XI Edition(s) and subsections.

Considering your CP date of December 1972, a preservice inspection plan is acceptable based on the ASME Code, Section XI, 1971 Edition including Addenda through Winter 1971, and all augmented examinations established by the Commission. If any of these preservice examination requirements can not be met, a complete technical justification to support your conclusion must be provided.

(2) The ISI plan based on current requirements will also be reviewed to support the safety evaluation report on ISI. We will require a comparison of your ISI plan with Section XI, 1974 Edition including all Addenda up to the latest referenced in the FEDERAL REGISTER, i.e., Summer 1975 Addenda, and all augmented examinations established by the Commission. Your response should define all examination requirements that you determine are not practical within the limitations of design, geometry, and materials of construction of the components. Particular

attention should be directed to impractical examinations resulting from revised Section XI requirements in 10 CFR Part 50, paragraph 50.55a(b), such as examinations that will result in high radiation exposure to personnel without a commensurate increase in safety, known inaccessible regions due to component arrangement, restricted access to welds in accepted ASME Code weld geometry designs, and limitations in examination methods or procedures due to metallurgical properties in approved materials of construction.

Discuss the ISI (or testing) that will be performed in lieu of the ASME Code Section XI requirements that you determine to be impractical. The technical justification to support your conclusions should contain as a minimum the identification of the applicable ASME Code Edition(s) and subsection(s), the number of components, the safety significance of postulated failure at the inspection location, the Section XI examination category, the examination method, the degree of conformance, and the system modifications, equipment or conditions that would be necessary for total compliance.

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

121.2 (6.6) Confirm that the design and layout of the high energy fluid system piping between (a) the first rigid pipe connection to the containment penetration, or (b) the first pipe whip restraint inside containment, and the first isolation valve outside containment will allow sufficient access to perform adequate augmented inservice inspection. Acceptance augmented inservice inspection requirements for these portions of piping are defined in SRP Section 6.6, paragraph II.8, entitled "Augmented ISI to Protect Against Postulated Piping Failures."

121.3 (5.4.2.2) (16.0) Confirm that the preservice and inservice inspection of steam generator tubing will be conducted in accordance with Regulatory Guide 1.83, Revision 1. If any of these examination requirements can not be met, a complete technical justification to support your conclusion must be provided.

121.4 (3A/1.14) (5.4.1.7) In your discussion of compliance with Regulatory Guide 1.14 and acceptance criteria, the inservice inspection is not mentioned. Confirm that the inservice inspection of the reactor coolant pump flywheels will conform to Regulatory Guide 1.14, Revision 1.

121.5 (5.3.1) Identify each material (plate, forging and/or weld metal) in the reactor vessel beltline region, as defined in 10 CFR Part 50, Appendix G, paragraph II H. Provide the following information for each of those materials, for each reactor vessel:

(1) Chemical analyses, particularly those elements known to affect material sensitivity to irradiation damage and degradation of the upper shelf energy (Cu, P, and S).

(2) Unirradiated mechanical properties, including the fracture toughness properties as required by 10 CFR Part 50, Appendix G, identifying the "limiting" materials in the beltline region and the reactor coolant pressure boundary.

(3) Using the recommendations of Regulatory Guide 1.99, "Effect of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," estimate the shift in RT_{NDT} and USE of the limiting material as a function of the EOL fluence at the inner wall of each reactor vessel.

121.6 (5.3.2) (16.0) Provide pressure-temperature limits, as required by General Design Criterion 31 to assure adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic materials of the reactor coolant pressure boundary, for each of the following operating or test conditions:

(1) Preservice hydrostatic tests,
(2) Inservice leak and hydrostatic tests,
(3) Heatup and cooldown operations, and
(4) Core operation.

121.7 (5.3.1) (16.0) Provide details on the proposed materials surveillance program for Unit Nos. 1 and 2. List the materials (plate, forging and/or weld metal) to be used as surveillance specimens and justify their selection, describe the specimen capsule and tube design and location in the reactor vessel. State any deviations from Appendices G and H, 10 CFR Part 50, and provide technical justification to support your conclusion.

121.8 (5.3.3) Verify that the design of Unit Nos. 1 and 2 will permit in-place annealing of the reactor vessels to restore ductility and toughness, in accordance with Appendix G, 10 CFR Part 50.

121.9 (10.2.3) To provide assurance that high energy turbine missiles will not be produced at operating speed or design overspeed, provide documentation to show the degree of conformance of the proposed turbine-generator with SRP 10.2.3, "Turbine Disk Integrity," paragraph II, "Acceptance Criteria."

130.0 STRUCTURAL ENGINEERING BRANCH

130.1 (3.3.2.3) Section 3.3.2.3 indicates that structures not designed for the design basis tornado (DBT) are checked to determine that they will not generate missiles more severe than those otherwise postulated. We require, in addition, that assurances be given that if these same structures collapse as a result of the DBT loading, they will not jeopardize safety functions of Category I structures, systems, and components.

130.2 (3.4.2) Your FSAR does not follow Regulatory Guide 1.70, Revision 2, for protection against external flooding. Section 3.4.1.2 of the FSAR discusses the so-called "hardened" flood protection approach. Describe this approach and provide additional information regarding static and dynamic loads, coincident wind loading, dynamic effects on foundation properties, and hydrological considerations.

130.3 (3.7.2.3) Insufficient information is presented to evaluate the adequacy of your procedures regarding subsystem decoupling. Specifically, we require more information with respect to mass ratios between subsystems and their supporting masses and with respect to frequency ratios between the fundamental frequency of the subsystem and the frequency of the dominant support motion.

130.4 (3.7.2.7) During our Regulatory Guide review and evaluation for the Midland Plant in 1976, we understood that the sum of the absolute values from the modes which correspond to natural frequencies below 33 Hertz would be used. However, your FSAR states that closely spaced modes are combined by the square root of the sum of the squares method. Clarify these conflicting views.

130.5 (3.8.1.6) Section 3.8.1.6 does not indicate a value for f'c. Document in Section 3.8.1.6 the design values of the ultimate compressive strength for all seismic Category I concrete structures. Assurances should also be given that sample testing confirms that these values have been attained.

211.0 REACTOR SYSTEMS BRANCH

211.1 (15.0) Provide plots of DNBR vs time for those events required by R. G. 1.70 Rev. 2.

211.2 (15.1.3) Your description of the steam pressure regulator malfunction and the inadvertent opening of a steam generator relief or safety valve indicate that the consequences of these events are bounded by the main steam line break. Provide the specific analyses for these cooldown transients to show that DNBR remains greater than 1.30 for each event or show that DNBR remains greater than 1.30 for the worst case main steam line break.

211.3 (15.1.5) The information provided in section 15.1.5 for main steamline break is not adequate. Provide analyses to locate the worst case break, considering the most limiting single active component failure, (FWIV, MSIV etc.) the assumption of offsite power available or **not** available, and the break location. Provide as a minimum the following plots for the worst break:

1. Peak Clad Temperature
2. DNBR
3. Reactivity Margin
4. Break Flow Rate
5. Pressurizer Level
6. Steam Generator Levels
7. Steam Generator Pressures

211.4 (15.0) Operational analyses or failure mode and effects analysis of the various plant responses to the Chapter 15 events are required. To complement the FSAR discussions in this regard, provide a summary of a systematic functional analysis of components required for each event analyzed in Chapter 15.0. The summary should be shown in the form of simple block diagrams beginning with the event, branching out to the various possible protection sequences for each safety action required to mitigate the consequences of the event (e.g., core cooling, containment isolation, pressure relief, scram, operator action, etc.), and ending with an identification of the specific safety actions being provided.

When complete, each protection sequence diagram should clearly identify (for each event) the safety systems required to function to provide the safety actions necessary to mitigate the consequences of the transient or accident (during any plant operating state). An example of such a systematic functional analysis is contained in "Transactions of the American Nuclear Society 1973 Winter Meeting", November 11-15, pages 339-340.

211.5 (6.3) Provide complete NPSH calculations for the ECCS pumps in both the injection and recirculation modes. Provide all assumptions and appropriate justifications.

211.6 (6.3) The reference to BAW-10104 and BAW-10103 for the required ECCS analyses in accordance with 10 CFR 50.46 is insufficient. Provide appropriate calculations and sensitivity studies (or references) which consider the impact of more recent model or equipment changes (such as vessel U-baffle modifications). Also, provide a discussion with references, of all applicable calculations using the small break model.

211.7 (15.2.3) The turbine trip analysis assumes credit for the Integrated Control System and turbine bypass. Provide or reference an analysis for turbine trip taking no credit for any non-safety grade equipment.

211.8 (5.2.2) Submit an analysis of the worst case overpressure transient during startup and shutdown. Provide all assumptions. Plots should include pressure vs. time, reactor coolant temperature vs. time and safety valve flows versus time. Show that the pressure-temperature limits in Technical Specifications are not exceeded. The following position is currently being considered for implementation by the NRC staff. Provide a discussion for the Midland design with respect to each of these points:

1. A system shall be designed and installed which will prevent exceeding the applicable Technical Specifications and App. G limits for the reactor pressure vessel while operating at low temperatures. The system shall be capable of relieving pressure during all potential overpressurization events at a rate sufficient to satisfy the Technical Specification limits, particularly while the Reactor Coolant System is in a water-solid condition.

2. The system must be able to perform its function assuming any single active component failure. Analyses using appropriate calculational techniques must be provided which demonstrate that the system will provide the required pressure relief capacity assuming the most limiting single active failure. The cause for initiation of the event, i.e., operator error, component malfunction, etc., will not be considered as the single active failure. The analysis should assume the most limiting allowable operating conditions (e.g., one RHR train operating or available for letdown, other components such as pressurizer heaters and charging pumps in normal operation when the system is water solid.) All potential overpressurization events must be considered when establishing the worst case.

3. The system must operate automatically, providing a completely independent backup protective feature for the operator. The design must not require manual actions to enable or "turn on" the system or to mitigate the consequences of a potential overpressure event.

4. To assure operational readiness, the overpressure protection system must be tested in the following manner:

   a. A test must be performed to assure operability of the system electronics prior to each shutdown.

   b. A test for valve operability must, as a minimum, be conducted as specified in the ASME Code Section XI.

   c. Subsequent to system, valve, or electronics maintenance, a test on that portion(s) of the system must be performed prior to declaring the system operational

5. The system must meet the design requirements of IEEE-279, Regulatory Guide 1.26, and Section III of the ASME Code.

6. The protection system does not have to meet seismic Category I requirements if it can be shown that an earthquake would not initiate an overpressure transient. The postulated earthquake should be of a magnitude equivalent to the SSE. If the earthquake can initiate an overpressure transient, then it should be assumed that loss of offsite power is an expected consequence of the event and the protection system should be designed to seismic Category I requirements and not require the availability of offsite power to perform its function. Should the applicant show than a postulated earthquake could not cause an overpressure event, the overpressure protection system design must not compromise the design criteria of any other safety-grade system with which it would interface. The requirements of Regulatory Guide 1.29 must be satisfied.

7. The loss of offsite power shall be considered as an anticipated transient which could occur while in a shutdown condition. If this event can initiate an overpressure transient, the overpressure protection system must be independent of offsite power, in addition to performing its function assuming any single active failure.

8. Plant designs which take credit for an active component(s) to mitigate the consequences of an overpressurization event must include additional analyses considering inadvertent initiation/actuation or provide justification to show that existing analyses bound such an event.

211.9 (15.0) Show how the Midland Plants can be maintained at hot shutdown with only safety grade systems assuming the loss of offsite power. How long can the plant be kept in this condition prior to requiring cooldown?

211.10 (6.3) Provide the following information considering a pipe break in a high pressure injection (HPI) line between the reactor coolant system piping and the last HPI check valve:

1. Operator action(s) required,
2. Indications provided for the operator,
3. Time operator action required,
4. HPI pump performance and availability during this event,
5. Flow splits in HPI piping, and
6. Summary table of scenario listing each event and associated times.

211.11 (15.0) Provide a discussion for each Chapter 15 event describing the operator actions required in both the short and long term. Our interest is in evaluating the operator's role in achieving and maintaining stable conditions. (Stable conditions can be assumed to be achieved when the decay heat removal system is placed in operation). An example of such a situation would be the necessity of the operator to secure the HPI pumps after a steam line break to prevent repressurization of the reactor coolant system at low temperature.

211.12 (15.0) Provide an analysis of a break in the normally pressurized makeup line considering all potential single active component failures. As a minimum, submit the following:

1. Table depicting the sequence of events
2. Indications and alarms available
3. Operator action(s) required
4. Plots of RCS pressure, RCS water level

211.13 (15.4.6) Provide additional analyses of the boron dilution event considering the plant conditions other than just power operation or refueling (as specified in Standard Review Plan 15.4.6). Discuss all potential dilution sources for the Midland Plant and address the design aspects which preclude or minimize the potential for a dilution event.

211.14 (5.4.7) The Decay Heat (DH) Removal System incorporates a low-flow DH pump trip interlock. Discuss this feature's potential contribution to the probability of a complete loss of low pressure injection during a LOCA. Balance this risk with the gain in availability of the DH function.

211.15 (15.0) Discuss the loss of instrument air for the Midland Plants showing that it meets the appropriate acceptance criteria for a moderate frequency event. Provide a detailed failure modes and effects discussion consistent with request 211.4. Causes and potential system interactions should be particularly addressed and the loss of instrument air should be considered during all phases of reactor operation. Also, present your plans and capability for preoperational or startup tests to substantiate the analyses.

221.0 REACTOR ANALYSIS SECTION, ANALYSIS BRANCH

221.1 (4.4.6.8) The applicant must commit to the installation of an adequate loose parts monitoring system (LPMS). Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at a number of plants. As a result of a study conducted on the installation of, and experience with, loose parts monitoring systems in operating plants, we have identified the following aspects for a LPMS which we will use to assess the acceptability of the specific system to be provided for Midland Units 1 and 2 when we review the detailed information submitted in the FSAR:

(1) The description of the loose parts monitoring system shall include the location of all sensors and the method for monitoring them. A minimum of two sensors will be required at each natural collection region. For example, in a pressurized water reactor, two sensors should be included at the top and at the bottom of the reactor vessel and at each steam generator primary coolant inlet.

(2) The description of the monitoring equipment shall include the levels and the basis for the alarm settings. In addition, the manufacturer's sensitivity specifications for the equipment shall be provided. Anticipated major sources of internal and external noise shall be identified along with the plans for minimizing the effects of these sources on the ability of the monitoring equipment to perform its intended function.

(3) The loose parts monitoring system will be required to function after any seismic event for which plant shutdown is not required. The procedures of Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants", are acceptable for demonstrating the seismic qualification of this system. An exception to this seismic qualification is that recorders are not required to function within their specified accuracy during or after seismic events without maintenance. However, monitoring (alarm and/or indication) capability must remain available for that channel at all times during and after the seismic event. A description of the precautions to be taken to assure the operability of the system after an operating basis earthquake shall be provided.

(4) The loose parts monitoring system should also be qualified in accordance with the recommendations of Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants", but the qualification program need not include a post-accident environment.

(5) The loose parts monitoring system must be operational and capable of recording vibration signals for signature analysis at the time of initial startup testing. A detailed discussion shall be provided of the operator training program, planned operating procedures, and record keeping procedures for the operation of the system.

312.0 ACCIDENT ANALYSIS BRANCH, SECTION B

312.1 (2.1) Indicate the occupancy factors used for the transient population shown in Table 2.1-7 and discuss the bases for these.

312.2 (2.1) Provide a revised Table 2.1-8 which indicates the maximum number of persons at each of the public facilities listed.

312.3 (2.1) Figure 2.1-1 shows the Midland Plant Unit Nos. 1 and 2 exclusion area crossing the Tittabawassee River and enclosing a portion of land labeled "Dow Chemical Property." Provide a detailed map of the exclusion area which shows the boundary and describe your authority to determine all activities within it, as required by 10 CFR Part 100. Discuss all plant unrelated activities contemplated within the exclusion area and indicate how personnel will be notified in the event of an emergency.

312.4 (2.2) Figure 2.2-1 showing the industrial facilities within five miles of the Midland Nuclear Plant is not legible. Provide a revised map which clearly shows these facilities and which also includes a distance scale.

312.5 (2.2) Discuss any planned or projected uses of underground brine cavities for storage of natural or liquified propane gas.

312.6 (2.2.2) Quantify to the extent possible the terms "almost always" and "isolated occasions" in reference to shipments of hazardous materials, such as explosives, along the C&O Railroad passing through Midland. Use records of past shipments as a basis for estimating shipment frequency, type of material, and amount per shipment.

312.7 (2.2.2) In reference to the major hazardous chemical storage facilities near the Midland Unit Nos. 1 and 2 (e.g., Dow Corning, Dow Chemical) provide estimates of the projected storage quantities of hazardous chemicals beyond those presently being stored (as presented in Tables 2.2-4 and 2.2-5).

312.8 (3.5.1) In reference to the probability for destructive overspeed indicated in Section 3.5.1.3.6.2 of the FSAR, describe the turbine valve testing program (if any) that was used as a basis for the estimation of the probability. Indicate how this valve testing program (if assumed in the analysis for estimating the probability for destructive overspeed) is related to the testing program that is recommended by the turbine vendor and/or the testing program that is to be adopted.

312.9 (3.5.1.3) In reference to the two airports that are indicated as being within 10 miles of the plant site in Table 2.2-8, provide an estimate of the present and projected air traffic (yearly operations) in terms of aircraft type.

312.10 (3.5.1.4) Discuss the proposed plant's capability to achieve a safe shutdown in the event of tornado missile damage of the borated water and condensate storage tanks shown in Figure 1.2-1 of the FSAR. (See related request 010.14).

312.11 (6.1.3) Provide a figure similar to Figure 6A-2, showing the maximum pH in the spray and sump water after an accident.

312.12 (6.2.2.1) In order to assure that the recirculation water will be within the pH limits given in SRP 6.5.3, a) provide information on any dead volumes, i.e., regions that will hold up and retain any injection spray water, and b) describe the pH limits of the spray solution for different combinations of maximum and minimum spray and ECCS flow.

312.13 (6.2.2.1) Provide justification, in the form of theory or experimental results, that hydrostatic equilibrium between water, the sodium hydroxide and sodium thiosulfate can be maintained during the drawdown period so as to afford a solution of constant composition.

312.14 (6.2.2.1) Provide the set point (i.e., water level in the Borated Water Storage Tank, BWST) which generates the Recirculation Actuation Signal.

312.15 (6.2.2.1) Address the criterion given in SRP 6.5.2 that the spray system should be capable of operating for a period of at least 30 days after a postulated accident.

312.16 (6.2.2.1) Regarding testing of the spray pumps, identify the flow capacity of the test line which directs flow back to the BWST.

312.17 (6.2.2.1) State the criteria the operator will use to determine when to valve off the chemical additive tanks.

312.18 (6.4) Describe the capacity of the control room charcoal filter system to prevent overloading of the filter with chlorine from a long-term release in the event of a rupture of the large cryogenic storage tank near the site.

312.19 (15.0) Expand Table 15.0-2 to include information on the steam generator secondary side volume, and the quantity of water and steam in the steam generator during normal operation.

312.20 (9.4) Provide a schematic describing the source of air flow labeled "M-443" in Figure 9.4-2 (M-465 sheet 2, Rev. 0) and indicate the air flow status in this duct during emergency operation of the control room ventilation system.

312.21 (6.4) In reference to Figure 6.4-3, indicate the three doors which will be card-key operated. Discuss briefly the resistance offered by the entry paths leading from the turbine room into the control room (west, between columns 5.0 and 5.3, and east, between columns 8.0 and 8.6) in the event of a massive steam release (e.g., main steam line break) or a CO_2 tank rupture in the turbine room.

312.22 (6.4) Section 6.4.4.2.1.2 indicates that emergency procedures regarding hazardous chemical release are given in Section 13.3 of the FSAR. Section 13.3 does not address specific emergency procedures with respect to hazardous chemical releases. Provide the missing information.

312.23 (6.4) Indicate what provisions are to be made for assuring the availability and transport of bottled air from offsite locations to the control room in the event of a long-term toxic gas release (e.g., see Regulatory Guide 1.95).

312.24 (15.4.8) Regarding the rod ejection accident with loss of offsite power, provide :

a) Curves showing the primary and secondary system temperature and pressure versus time, for a period of two hours.

b) A curve showing percentage of submergence of the once-through steam generators (OTSG) tubes versus time.

c) A table showing the sequence of events and their respective times of occurrence. For events that may have variable times of occurrence (e.g., initiation of emergency feed water), provide the time which is most pessimistic in terms of radiological consequences.

d) A description of all operator actions leading to the final (long term) stabilized plant conditions and available information in the control room to diagnose the accident.

312.25 (15.6.3) For the steam generator tube rupture accident with loss of offsite power, provide:

a) Curves showing the primary and secondary system temperature and pressure versus time, for a period of 2 hours.

b) Curves showing degree of submergence of tubes in the steam generators.

c) A table showing the sequence of events and their respective times of occurrence. For events that may have variable times of occurrence (e.g., initiation of emergency feed water), provide the time which is most pessimistic in terms of radiological consequences.

d) A description of all operator actions leading to the final (long-term) stabilized plant conditions, and available information in the control room to diagnose the accident.

312.26 (15.6.2) State the basis used to calculate the leak rate due to a letdown line break.

312.27 (15.6.5 Identify and describe those safety features provided to mitigate the consequences of leakage from Engineered Safety Features equipment after an accident.

312.28 (15.7.4) In addition to the information requested in Mr. S. Varga's letter of April 6, 1977 with respect to a fuel handling accident inside containment, provide the following:

a) Describe the radiation monitoring instrumentation which will detect a fuel handling accident inside of the containment structure and in the spent fuel storage building.

b) Describe the response time of the containment isolation valves. Indicate closure times which will be included in your technical specifications.

c) Indicate the transient time from the radiation monitor detection to the isolation valve based on the maximum velocity of the air in the exhaust system.

d) Provide drawings of the containment pool area exhaust systems which show the location of the radiation detectors relative to the exhaust inlets and isolation valves.

321.0 EFFLUENT TREATMENT SYSTEMS BRANCH

321.1 (6.5.1) Indicate how the Engineered Safety Features Ventilation System designed to maintain a controlled environment in areas containing safety-related equipment satisfies each regulatory position in Regulatory Guide 1.52, by updating Table 3A.1.52. Justify each item of nonconformance. In addition, for each ESF atmosphere cleanup system, indicate the automatic activation provisions. Activation of the applicable ESF filtration system after a DBA should be automatic by redundant Seismic Category I radiation monitors unless (1) the atmosphere cleanup system is operating during the time the DBA occurs, or (2) activation is the result of another engineered-safety-feature signal (i.e., temperature, pressure).

321.2 (11.3) Provide an analysis with respect to each position in the Branch Technical Position, ETSB No. 11-2, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Reactor Plants," for each atmosphere cleanup system designed to collect airborne radioactive materials during normal plant operation including anticipated operational occurrences. Only the items of noncompliance need be listed with the justification for noncompliance.

321.3 (11.3) In Section 11.3.1.2.2 you indicate how the Radwaste Gas System is designed to prevent hydrogen explosions. Indicate the location of your redundant monitors, where alarms are located and the explosion-proof design provisions (i.e., both nonsparking and capable of withstanding explosion) of the monitors. You should provide dual (i.e., two independent gas analyzers continuously operating and providing two independent measurements verifying hydrogen and/or oxygen concentrations) gas analyzers with automatic control functions to preclude the formation or buildup of explosive hydrogen/oxygen mixtures.

The location of the gas analyzers should be as follows:

1. For PWR systems using recombiners, analysis for hydrogen and/or oxygen should be downstream of the recombiners. In addition, unless the system design features preclude explosive mixtures of hydrogen and oxygen mixtures upstream of the recombiners, analysis for hydrogen and/or oxygen (as appropriate) should be upstream of the recombiners as well. The

371.6 (2.4) Provide additional information regarding the larger reservoirs in the Tittabawassee basin. For Sanford, Edenville, Smallwood, and Secord Dams, this information should include:

1. Detailed topograhic maps of the reservoirs, showing their size, location, and drainage areas.

2. Area-capacity curves for each reservoir, from normal water surface elevation to top of dam elevation.

3. Dam cross-sections showing important elevations, composition, type of dam, and slopes.

4. Tailwater rating curves for each dam.

5. Spillway and outlet racing curves.

6. River cross-sections downstream of each dam (from dam to dam and from Sanford Dam to the site) at approximately 1/2-mile intervals.

371.7 (2.4) Document that flooding of safety-related buildings or equipment does not occur due to runup on the service water pump structure to elevation 640.9 (your estimate of maximum runup due to 112-mph wind.

371.8 (2.4) Provide details of the intake and discharge structures for the service water system, including plans and cross-sections of both structures.

372.0 METEOROLOGY

372.01 (2.3.1) Your discussion of snowpack in Section 2.3.1 references a 1971 study by M. A. Bilello entitled, "Frozen Precipitation: Its frequency and Associated Temperatures," published at the Eastern Snow Conference, New Brunswick. In this study, the mean monthly density of snowpack at Oscoda, Michigan was estimated to be 0.3 g/cm^3. For calculations of the weight of snowpack at the Midland site, a snowpack density of 0.25 g/cm^3 was assumed.

1. Provide further justification for using the assumed value of 0.25 g/cm^3, including elaboration of the statement that estimates of the density of snowpack in the site area were less than 0.1 g/cm^3.

2. Provide a copy of the Bilello reference.

372.02 (2.3.1) Your basis in the discussion of the frequency of lightning strikes to structures is a 1971 publication by D. Bodle, "Electrical Protection Guide for Land-Based Radio Facilities" (JES-159-3-3M 1/74, Joslyn Electronic Systems). Provide a copy of this publication.

372.03 (2.3.1) Provide information on the occurrence of tornados in the vicinity of the site for 1976 through the present, including estimates of the intensity (maximum wind speed) and path area of each.

372.04 (2.3.1) Provide the basis for the temperature values in Section 2.3.1 used for the design of the Midland plant heating, ventilating, and air conditioning systems, operation, such as extreme temperatures over extended time periods and large temperature changes over short time periods.

372.05 (2.3.2) The onsite stability distribution for the 60-10 meter vertical temperature difference appears to be biased toward the neutral (D) stability class and does not correlate well with other sites in similar meteorological regimes and using the same vertical temperature stability classification. For example, data from the Greenwood site (located 12 miles NW of Port Huron) showed 25% D stability while the Midland site indicated 57% D stability. In addition, the Midland stability distribution which is based on a vertical temperature gradient, shows good correlation with the Flint stability distribution (derived from the STAR program) which is based on cloud cover, time of day, and wind speed. However, these two stability classification schemes have historically shown poor agreement. Discuss further the validity of the stability distribution based on the onsite data at the Midland plant.

372.06 (2.3.2) Since the main tower is located in a parking lot, discuss any effects this may have on the meteorological parameters being measured Include discussion on the material the parking lot is made of, how close cars and/or trucks may be parked to the tower, any obstructions or events that may influence the meteorological measurements and any effects the cooling pond may have on the tower measurements. Also compare meteorological variables recorded at the north and south towers with those similar variables at the main tower.

372.07 (2.3.2) Your discussion on the frequency of fog occurrence was based on the following publications: "The Environmental Effects of the Midland Plant Cooling Pond," Report for Consumers Power Company (1972), Bechtel Corporation; "Fog and Plumes from Power Plant Cooling Systems in the Tri-Cities-Saginaw Bay Area," D. J. Portman Report for Consumers Power Company (1975); "An Analytical and Experiemental Study of Transient Cooling Pond Behavior," P. J. Ryan and D. R. F. Harleman, Report No. 181 (1973), Ralph M. Parsons Laboratory for Water Resources and Hydrodynamics, MIT, Cambridge, Massachusetts. Provide a copy of these publications.

372.08 (2.3.3) Provide the calibration results for all calibrations of onsite meteorological instrumentation and data acquisition systems.

372.09 (2.3.3) Describe the process in which the plant control room will receive meteorological data from the main tower, including such things as how the data will be received (i.e., teletype, visual, etc.), in what form the data will be received (i.e., instantaneous values, hourly averaged values, etc.) and what the procedure will be if something happens to render data from the main tower unreceivable at the control room.

372.10 (2.3.5) Compare and explain any differences in the power law exponential values given in Table 2.3-29 with those from the "Recommended Guide for the Prediction of the Dispersion of Airborne Effluents" (Smith, M.E. (ed), 1968, The American Society of Mechanical Engineers, New York, N. Y.). Identify the period of onsite data used for the calculations of the site-specific exponential values.

372.11 (3.3.2) The criteria for a design basis tornado (DBT) in Regulatory Guide 1.76 is based upon a pressure drop followed instantaneously by a pressure rise. However, for the Midland DBT, a 2-second lag time is indicated between the pressure drop and pressure rise. Discuss the basis for this deviation from Regulatory Guide 1.76 and the effects it will have on any safety-related structures.

422.0 QUALITY ASSURANCE BRANCH: CONDUCT OF OPERATIONS

422.1 (13.1.1.3) Identify qualification requirements for headquarters staff personnel, in terms of educational background and experience requirements, for each class of positions identified in Figure 13.1-2 as providing technical support for operation of the Midland facility. In addition, describe the number of technical personnel assigned to each of the groups shown in Figure 13.1-2 that will provide technical support for the operation of the Midland facility.

422.2 (13.1.2.1) Expand your description of the plant organization shown in Figure 13.1-3 to provide the following information; the number of persons assigned or to be assigned to common or duplication position for two-unit operation and, for the operation of the first unit prior to operation of the second unit.

432.0 EMERGENCY PLANNING BRANCH

432.1 (13.3.3) The emergency plan provides protective action guide levels (PAGs) and describes radiation monitoring equipment that will be used to detect activity releases (p. 13.3-4 and 13.3-29). However, the methods by which the doses (PAGs) are converted into instrument readings (or vice versa) are not discussed. This information is required to satisfy Section IV (C) of Appendix E (10 CFR Part 50). Summarize the methods used to determine action levels for each of the accidents analyzed in Section 13.3.3.3.

432.2 (13.3.3) The bases for all quantitative analyses reported in the emergency plan should be made a part of the plan or adequately referenced. For instance, the methods and assumptions used, 1) to construct the dose plots (figures 13B-1, 2 & 3) and 2) to calculate the doses appearing in Section 13.3.3.3 should be provided.

432.3 (13.3.4) Each of the participating agencies are introduced and discussed separately in Section 13.3.4.4. Include in this section a discussion of how each of these agencies fit into the overall emergency organization. The present section does not clearly differentiate the authorities and responsibilities of each agency. For instance, some of the duties assigned to the Department of State Police Emergency Service are quite similar to those of the Midland City County Department of Emergency Services.

Also include the location (relative to the plant) of the active components of each of the agencies.

432.4 (13.3.5.4) The basic protective actions of taking cover and evacuation are not sufficiently described. The pre-planned evacuation sectors should be defined. The procedures for informing the public and carrying out the protective action should be described.

432.5 (13.3.7) With respect to updating of emergency plans, provide a listing of each emergency plan document that is pertinent to the emergency preparedness of the Midland plant. Indicate who is responsible for keeping each document updated and the parties that must be informed of the changes.

432.6 (Appendix 13B) A supplement to Figures 13B-4, 5 is required. This supplement should include road network information keyed to the maps that give the characteristics of each major road, all intersections, the number of lanes, whether improved or unimproved, and other factors that may affect vehicular traffic capabilities during an emergency evacuation.

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