MAR 1 5 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 TWO

My letter of February 24, 1978, forwarded the first of three scheduled parts for our requests for additional information for our FSAR review of Midland Plant Units 1 & 2. The second part of that request is enclosed. Also enclosed is an errata sheet correcting two of our previous requests.

We will need complete and adequate responses to Enclosure 1 by May 12, 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.

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

Sincerely,

Original signed by:
S. A. Varga

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

Enclosure: As stated

cc: See Next Page

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Consumers Power Company

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ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION (Q1's)

PART 2 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

010.36 (3.2) Provide the following additional information or changes to FSAR Table 3.2-1:

1. Identify the Quality Group classification and seismic classification of those portions of the Reactor Plant Sampling System that are within the reactor coolant pressure boundary and other fluid systems important to safety. Also identify this information on Figures 9.3-1 through 9.3-6.

2. Revise the component code and code class for the containment penetration of the Equipment and Floor Drainage System to be identified as ASME Section III, Class 2, (III-2).

3. Correct the component code for the filters of the Boron Recovery System to be ASME Section VIII. The code is presently incorrectly identified as ASME Section III, Class 1, (III-1).

4. Correct the Quality Group classification of the Standby Keepwarm Pump to be Quality Group C. The present Quality Group classification for this component is incorrectly indicated to be Quality Group D.

5. Correct the Steam Generator Recirculation Pump, Piping and Valves which are presently identified as Quality Group C/D components and constructed to the B31.1 Power Piping Code. To be acceptable, these components should be classified Quality Group B and constructed to ASME Section III, Class 2 as shown in Figures 10.4-10 and 10.4-13. Resolve this inconsistency in the classification of these components.

010.37 (3.2) Verify that all components within the reactor coolant pressure boundary as defined in 10 CFR Part 50.2(v) are classified Quality Group A in compliance with the Codes and Standards Rule, Section 50.55a of 10 CFR Part 50, or as a minimum, are classified Quality Group B if the components meet the exclusion requirements of the rule.

022.0 CONTAINMENT SYSTEMS BRANCH

022.21 (6.2.6) Closed systems outside containment (e.g., the emergency core cooling system and the contain sp ·· system) will constitute one of the redundant containment isolation barriers, and because of their post-accident function, . become extensions of the containment boundary following a LOCA. Since these systems, which will contain contaminated water, possess system valves and pumps, they may become potential leakage paths for contaminated water outside of containment. Therefore, specify the leakage limit for each of these systems and discuss how the leakage will be included in the radiological assessment for the site. Discuss the test method(s) that will be used to quantify the leak rates and propose a test frequency commensurate with the method(s) of testing employed.

022.22 (6.8) It is our position that the liner plate weld channel pressurization system may not be used in the manner proposed, which will permit reactor operation with the containment leak rate above the maximum allowable leak rate until a more opportune time occurs to effect repairs. We will require that repairs be made prior to resuming reactor operations. Discuss your plans to comply with this position.

022.2. (6.8) Discuss how the air pressurization system will pressurize the personnel airlocks during normal plant operations; e.g., will pressurization of the entire airlock occur, or will only the door seals be pressurized?

022.24 (6.8) The failure modes and effects analysis for the reactor building penetration pressurization system is not complete. Discuss the capability of the system to accomplish its intended function in the event a diesel generator fails to start following a loss of offsite power. Under these circumstances isolation valves would fail to close and may render the system ineffective. Discuss the effect on the pressure response of the containment in the long term following a LOCA if the nitrogen (or air) supply is not terminated.

022.25 (6.8) Assuming normal reactor operations, discuss the consequences of a breach in the boundary of a volume being pressurized by the air (or nitrogen) seal pressurization system, either because of a valve inadvertently opening or a seal failing. Discuss how the operator will be alerted of a malfunction in the system and the actions that will be taken.

022.26 (6.8, 6.2.6) Discuss the proposed use of the penetration pressurization system during the containment integrated leak rate (Type A) test.

110.0 MECHANICAL ENGINEERING BRANCH

110.30 (6.2.4) Table 6.2-28 of the FSAR indicates that the containment purge system isolation valves are containment isolation valves and receive automatic isolation signals. Therefore, should a loss of coolant accident occur while these valves are open they will be called on to close while experiencing the LOCA pressure and temperature within the containment.

It is the staff's position that the containment purge isolation valves are active valves. Therefore, describe the valve operability program applicable to these valves. Demonstrate that this program is capable of verifying the ability of these valves to close when subjected to the pressure and temperature profiles shown in Figure 6.2-4 of the FSAR. Your attention is directed to Section II.2 of Standard Review Plan 3.9.3 for a description of an acceptable operability program.

211.0 REACTOR SYSTEMS BRANCH

211.16 (3.5.1.1) What is the basis for limiting the missile selection criteria to high energy systems? Provide justification to show that missiles with lower energy levels would not fail any safety-related equipment.

211.17 (4.6.3.1) Section 4.6.3.1 of the Midland FSAR states that the CRDM's to be used in the design are "essentially" identical to that supplied on previously reviewed plants. Discuss any modifications and the justification for each. Include relevance to previous prototype testing.

211.18 (5.2.2.4) Provide a discussion and bases for relief valve setpoints and capacities.

211.19 (5.2.2) Since it is assumed that pressurizer and steam safety valve accumulation is 3% or less, how is this verified?

211.20 (5.2.2) Review of Section 5.2.2.2a and 5.2.2.2g shows that increments for flux measurement uncertainties and safety valve setpoint tolerances are not included. Add these items or provide justification for their omission.

211.21 (5.2.2.4) Does the pressurizer safety valve capacity reflect the capacity as installed (i.e., has the effect of associated piping been included)?

211.22 (5.2.2) Check valves in the discharge side of the high pressure injection, low pressure injection, and DHR systems perform an isolation function in that they protect low pressure systems from full reactor pressure. The staff will require that these check valves be classified ASME IWV-2000 Category AC, with the leak testing for this class of valve being performed to code specifications. It should be noted that a testing program which simply draws a suction on the low pressure side of the outermost check valves will not be acceptable. This only verifies that one of the series check valves is fulfilling an isolation function. The necessary testing frequency will be that specified in the ASME Code, except in cases where only one or two check valves separate high to low pressure systems. In these cases, leak testing will be performed at each refueling after the valves have been exercised.

Identify all ECCS check valves which should be classified Category AC as per the position discussed above. Verify that you will meet the required leak testing schedule, and that you have the necessary test lines to leak test each valve. Provide the leak detection criteria that will be proposed for the Technical Specifications.

211.23 (5.2.5.2) Regulatory Guide 1.45 states that identified and unidentified leakage should be collected spearately. The discussion in

(5.2.5.2) Section 5.2.5.2 indicates that all leakage, identified and unidentified, will be collected in the reactor building sump. Provide a discussion of the method used to distinguish unidentified from identified leakage.

211.24 (5.2.5) Provide a discussion of the method of detecting intersystem leakage; specifically leakage to the core flood, decay heat removal, HPI, nitrogen, and vent and drain system as required by Regulatory Guide 1.45.

211.25 (5.2.5.3) Regulatory Guide 1.45 requires charts and graphs to convert containment air monitor signals to equivalent gpm leak rates to assist the operator in interpreting signals. Address this capability for Midland Units 1 and 2. Alarm set points and their correlation to leak rate should also be provided.

211.26 (5.2.5.7) Discuss the capability to take a grab sample of the containment atmosphere on a periodic basis and to manually analyze these samples for particulate activity and to correlate the data to primary system leakage.

211.27 (5.2.5) Regulatory Guide 1.45 requires that the three methods used in unidentified leak detection be able to detect a one gpm leak in one hour. Discuss how you intend to meet this requirement, particularly with regard to the gaseous radioactivity monitor.

Also, the FSAR does not provide a clear explanation of how the sump level and flow monitoring system can detect a one gpm leak in one hour. Discuss in some detail the operation of this system with regard to leak detection sensitivity.

211.28 (5.2.5) State the expected range of the variables monitored for unidentified leak detection and the limits to which the instrumentation can cover this range and still detect and alarm a one gpm leak in one hour.

211.29 (5.2.5) During plant startup or after an extended outage, coolant activity may be low enough such that containment activity due to the presence of small leaks may be below the threshold sensitivity of the radiation monitors used for leak detection. Describe how you intend to monitor RCPB leakage without the use of this equipment until containment activity has increased to a detectable level.

211.30 (5.2.5) Once an unidentified leak has been detected, what procedures will be used to locate the source of leakage?

211.31 (5.2.5) The FSAR does not state that the leak detection systems can perform their functions following a seismic event that does not require plant shutdown as required in Regulatory Guide 1.45. Show that the Midland plant meets this requirement.

211.32 (5.2.5) Describe the provisions for detecting leakage from the primary coolant system to the RHR and ECCS through injection and return lines during normal power operation. Describe the indications, alarms, and procedures for isolation to limit releases and show conformance with Regulatory Guide 1.45.

Discuss the procedures used by the operator to convert all leak detection indications in the control room to a common leakage equivalent, e.g., cpm to gpm.

211.33 (5.2.5) Discuss the basis for determining the alarm setpoints for all three unidentified leakage detection systems.

211.34 (5.2.5) Describe the procedures used to calibrate the radiation monitors and sump level and flow monitors to RCPB leakage.

211.35 (5.4.7) Should the Midland plants experience an event that will require eventual cooldown to permit either long-term cooling with the DHR system or going to cold shutdown for inspection and repairs (extended loss of offsite power, steam generator tube rupture, failure of steam generator relief valves to reclose, etc.), it is desirable that qualified systems be available to perform the operation safely and in an orderly manner. Discuss the capability of the Midland plants to be taken to a cold shutdown condition using only safety-grade equipment, assuming only onsite or offsite power is available, and considering a single failure. Address each of the following areas of concern in your response:

(1) Discuss the capability of the single DHR drop line to provide for the cooldown of the plant assuming a single active failure, including manual actions inside or outside of containment or the return to hot standby until manual actions or maintenance can be performed to correct the failure.

With regard to the Midland shutdown capability, we note that manual operation outside the control room is required for normal shutdown,and containment entry is required for a failure of a motor-operated DHR suction valve. With regard to reducing the need for such manual actions, address the following areas:

(a) Discuss the modifications required to provide the capability to conduct a normal shutdown from the control room.

(5.4.7) (b) Justify the viability of the manual actions required after a suction valve failure (i.e.. opening cross-connects 093, 094). Address times required, doses expected, and potential for inadvertent opening of cross-connects during high primary side pressure conditions. Compare the Midland cross-connect design to Davis-Besse Unit 1. Provide a reliability analysis for the manual action outside the control room and discuss the incremental increase in reliability expected for various selected design modifications.

(2) Provide safety-grade steam generator dump valves, operators, air and power supplies which meet the single failure criterion.

(3) Provide the capability to cool down to cold shutdown assuming the most limiting single failure in less than 36 hours or show that manual actions inside or outside containment or return to hot standby until the manual actions or maintenance can be performed provides an acceptable alternative.

(4) Provide the capability to depressurize the reactor coolant system with only safety-grade systems assuming a single failure, or show that manual actions inside or outside containment or remaining at hot standby until manual actions or repairs are complete provides an acceptable alternative.

(5) Discuss the capability for boration with only safety-grade systems assuming a single failure or show that manual actions inside or outside containment or remaining at hot standby until manual action or repairs are completed provides an acceptable alternative.

(6) Discuss the capability for the collection and containment of DHR system pressure relief valve discharge.

(7) Conduct tests to study the mixing of the added borated water and the cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve.

(8) Commit to providing specific procedures for cooling down using natural circulation and submit a summary of these procedures.

(9) Provide a Seismic Category I AFW supply for at least four hours at hot shutdown plus cooldown to the DHR system cut-in based on the longest time (for only onsite or offsite power and assuming the worst single failure), or show that an adequate alternate Seismic Category I source is available.

211.36 (Table 5.4.10) (6.3.2.2) Table 5.4.10 should be expanded to include sizing criteria and backpressure considerations. This table also indicates 100% accumulation. This error should be corrected and the value selected should be justified. How will this value be confirmed throughout plant life?

211.37 (5.4.7) On Figures 5.4-10 and 5.4-11, the high pressure line designations are "CCA." Per Figure 1.1-2, this corresponds to a 1500 pound pressure rating. It would appear that the line designations or the codes on Figure 1.1-2 are in error. Verify that the letdown lines and injection lines inside the containment isolation valves are rated at primary plant pressure.

211.38 (5.4.7) Section 5.4.7.1.1.3 states that the DHR suction relief valve is sized for the "most rapid rate of pressure increase." Provide the complete quantitative basis for the sizing of this valve. Provide analyses of the component failures or operator errors which could initiate an overpressure transient during plant cooldown. Discuss all assumptions and your analysis techniques.

211.39 (6.3) Referenced figures (5.4-10, 5.4-11, 9.3-32, and 9.3-34) must be expanded to show ECC injection and recirculation valve positions in addition to normal positions. Provide or reference piping identification diagrams which show all of the ECCS including the BWST.

211.40 (6.3.2.2) Figure 6.3-3 shows that since HPI flow from injection pressure and below is expected to be greater than 300 gpm, the high flow alarm (263 gpm) will always alarm. Clarify this discrepancy. What is the basis for the high flow alarm setpoint?

211.41 (6.3.2.5) Discuss the provisions and precautions for assuring proper system filling and venting of ECCS to minimize the potential for water hammer and air binding. Address piping and pump casing venting provisions, accessibility, and surveillance frequencies.

211.42 (6.3.3.8) Provide the basis for ECCS lag times. Are these times calculated or verified by test? If calculated, they must be verified in preoperational test, then periodically reverified.

211.43 (6.3.3.9) Provide justification for not including the nitrogen system used to pressurize the core flood tanks in this section (system dependency).

211.44 (Table 6.3.1) Explain the relevance of Table 6.3.1 to the ECCS analyses. Also, address the following:

(1) Justify the use of normal values rather than the worst-case (minimum or maximum) values.

(2) Why is a maximum boron value specified for the core flood tanks and not for the borated water storage tank?

(3) For core flood tanks, units should be added to level alarm setpoints; also, equivalent ft^3 should be listed.

(4) Same comment as above for BWST.

(5) No cleanliness level is given for the BWST. Why are different components in the same system assigned different cleanliness levels (i.e., LPI pumps are level B, while decay heat removal coolers are C)?

211.45 (Table 6.3-6) With respect to the core flood tank line break, Table 6.3-6 is not complete in that the effect on the reactor was not clearly addressed. For example, the table shows that for the CFT line break, the loss of both LPI trains has no effect on plant operation, however, in the post-LOCA mode, the effect may be severe in regard to cooling the core. The FMEA must clearly show the systems available to cool the core.

211.46 (Table 6.3-6) With respect to Table 6.3-6 address the following:

(1) Operator mitigation states RCS pressure maintained with LPI pumps. Per pump data in Section 6.3, shutoff head of LPI pumps is approximately 200 psi. Should this be HPI pumps?

(2) It is noted that CFT pressure is 600 + 15 psig while CFT isolated alarm is 750 psig. Address the consequences of a LOCA with the CFT's isolated (~700 psig).

(3) For the LOCA in the CF line break, explain how flow is divided between two LPI flow paths. What indications are used and what action is required?

211.47 (Table 6.3-7) The Table 6.3-7 is not complete. Add a column for the method of detection. Also, the pump seal failure should be added to this table.

(6.3) (5.2.7) Provide more detailed information on the proposed leakage collection and the detection system. Discuss provisions for identifying the location of the leak and the time required to identify various size leaks under post-LOCA conditions.

(Table 6.3-7) (6.3) (5.2.7) Our requirements for leakage detection of ECCS equipment passive failures (such as valve stem packing and pump seals) are stated below.

Detection and alarms must be provided to alert the operator to passive ECCS failures during long-term cooling which allow sufficient time to identify and isolate the faulted ECCS line. The leak detection system should meet the following requirements:

(1) Identification and justification of maximum leak rate should be provided.

(2) Maximum allowable time for operator action should be provided and justified.

(3) Demonstration should be provided that the leak detection system will be sensitive enough to initiate (by alarm) operator action, permit identification of the faulted line, and isolation of the line prior to the leak creating undesirable consequences such as flooding of redundant equipment. The minimum time to be considered is 30 minutes.

(4) It should be shown that the leak detection system can identify the faulted ECCS train and that the leak is isolable.

(5) The leak detection system should meet the following requirements:

(a) Control room alarm

(b) IEEE-279, except single failure requirements.

211.48 (6.3.4.1) Regulatory Guide 1.79 specifically recommends the following tests be performed:

(1) the capability of the HPSI pumps to take a suction from the LPSI pumps should be demonstrated;

(2) pump flow test should be initiated by the safety injection signal.

Provide a listing of the tests required by Regulatory Guide 1.79 and provide confirmation of your intent to implement each test procedure or address nonconformance. Identify specific deviations and provide justification.

(6.3.4.1) Assurance must be provided that the low pressure injection system can take suction from the recirculation sump, verifying vortex control and acceptable pressure drops across screening and suction lines and valves. Submit a test plan which satisfies this portion of the Regulatory Guide 1.79 requirement.

211.49 (6.3) In Table 6.3-6, isolation for an HPI line break is stated as being ensured by closure of either isolation valve 446 or 499 for Unit 2 (346 or 399 for Unit 1). Valve 499 (or 399) is not presently shown on the respective makeup and purification diagrams in Section 9.

211.50 (6.3) In Table 5.4-12, spurious closure of the DHR reactor building isolation valve 1120A, B is considered for normal DHR operation. In Table 6.3-6, spurious closure of this valve is considered as not being credible for ECCS operation since the valve is locked open. The valve appears to be locked open for either mode of operation. Resolve this discrepancy between the two failure analyses.

211.51 (6.3) The ECCS for the Midland plants contains manual as well as motor-operated valves. Consideration must be given to the possibility that manual valves might be left in the wrong position and remain undetected when an accident occurs. Provide a list of essential manually operated valves in the ECCS and a discussion of the methods used to minimize this occurrence for each valve. The list should also include all manual valves in the ECCS for which valve position is not indicated in the control room. Address the requirements of Regulatory Guide 1.47 in your response. Verify that there is no one manual valve which could interrupt the flow to both ECCS trains.

211.52 (6.3) Submit ECCS P&ID's which are the final plant drawings used for Midland construction. The concern is that inadequate information exists in the presented simplified schematics to allow an adequate evaluation (see question 211.39). The size of the drawings should be large enough to ensure legibility of valve designations, etc.

211.53 (6.3) With regard to the conservatism of NPSH calculations, the "required" NPSH has often been defined as a fixed number as provided by the architect engineer or the pump manufacturer. Since several methods exist to calculate the required NPSH and the method used can affect the suitability of a particular pump, it is requested that Midland provide and justify the basis on which the required NPSH was determined (i.e., testing, Hydraulic Institute Standards) for all ECCS pumps and the estimated NPSH variability between similar pumps. Include a discussion of all inaccuracies.

211.54 (6.3) Recent plant experience has identified a potential problem regarding the operability of the pumps used for long-term cooling (normal shutdown as well as post-LOCA) for the time period required to fulfill that function. Provide the pump design lifetime

(6.3) (including operational testing) and compare to the continued pump operational time required during the short and long term of a LOCA. Submit information in the form of tests or operating experience. Verify that these pumps will satisfy long-term requirements.

211.55 (6.3) So that we may evaluate the dependence of the ECCS equipment on the plant auxiliaries, provide, or reference in the FSAR, the following:

(1) A list of all of the primary auxiliary systems required to directly support each ECCS component.

(2) A brief description of the supporting function performed by the primary auxiliary.

(3) The method of initiating the primary auxiliary to provide support to the ECCS.

(4) The additional secondary auxiliaries required to directly support the primary auxiliary specified in (1).

(5) A brief description of this supporting function performed by the secondary auxiliary.

(6) The method of initiating this secondary auxiliary.

211.56 (6.3) The Midland FSAR references BAW-10103, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS" for preventing excessive boron precipitation during long-term cooling. Approval of this portion of BAW-10103 was deferred by the staff to a plant-specific basis. Therefore, provide or reference information for the Midland plants addressing the following design guidelines:

(1) The boron dilution function shall not be vulnerable to a single failure. A single active failure postulated to occur during the long term cooling period can be assumed. However this failure would then be in lieu of a single active failure during the short term cooling period.

(2) The inadvertent operation of any motor operated valve (open or closed) shall not compromise the boron dilution function nor shall it jeopardize the ability to remove decay heat from the primary system.

(3) All components of the system which are within containment shall be designed to seismic Category I requirements and classfied Quality Group B.

(6.3) (4) The primary mode for maintaining acceptable levels of boron in the vessel should be established. Should a single failure disable the primary mode, certain manual actions outside the control room would be allowed, depending on the nature of the action and the time available to establish backup mode.

(5) The average boric acid concentration in any region of the reactor vessel should not exceed the level of four weight percent below the solubility limits at the temperature of the solution.

(6) During the post-LOCA long term cooling, the ECC system normally operates in two modes: the initial cold leg injection mode, followed by the dilution mode. The actual operating time in the cold leg injection mode will depend on plant design and steam binding considerations, but, in general, the switchover to the dilution mode should be made between 12 and 24 hours after LOCA.

(7) The dilution mode can be accomplished by any of the following means:

(a) Simultaneous cold leg injection and hot leg suction
(b) Simultaneous hot and cold leg injections
(c) Alternate hot and cold leg injections.

(8) In the alternate hot and cold leg injection mode, the operating time at hot and cold leg injection should be sufficiently short to prevent excessive boric acid buildup.

(9) The minimum ECCS flow rate delivered to the vessel during the dilution mode shall be sufficient to accommodate the boil-off due to fission product decay heat and possible liquid entrainment in the steam discharged to the containment and still provide sufficient liquid flow through the core to prevent further increases in boric acid concentration.

(10) All dilution modes shall maintain testability comparable to other ECCS modes of operation (HPI-short term, LPI-short term, etc). The current criteria for levels of ECCS testability shall be used as guidelines (i.e., Regulatory Guides 1.68, 1.79, GDC 37).

(11) The operator should be capable of confirming minimum required flows subsequent to a LOCA.

211.57 (6.3) Discuss the potential for, and the precautions taken, to prevent crystallization of boric acid in the safety injection system. For example, operating experience has shown instances where the high head safety injection pump was able to achieve only about one half of the pump discharge pressure because the suction elbow and the eye of the pump were found to be plugged with solidified boric acid crystals.

211.58 (6.3) Because of freezing weather conditions, blocking of the vent line on the BWST has occurred on at least one operating plant. Describe design basis and features that preclude this condition from occurring in the Midland plant.

211.59 (15.0) Table 15.0.2 gives a pressure/temperature trip delay of 0.7 second. Since one of the inputs to this trip is the hot leg temperature, which has a delay of 6 seconds, the 0.7 second delay would appear to be in error. Provide the correct trip delay and verify that this value has been used in the analyses.

211.60 (15.0) Provide a confirmation, with bases, that all transient events would not exceed the acceptance criteria for abnormal operational occurrences when credit is not taken for nonsafety-grade systems (turbine trip, turbine bypass, pilot-operated relief valves, etc.). The discussion for the turbine trip analysis in Section 15.2 gives the impression that the analysis was conducted assuming the failure of one nonsafety-grade system at a time. Clarify this discussion to show that no credit for nonsafety-grade components is taken in the analysis.

211.61 (15.2.3) With regard to the turbine trip analysis, provide the stroke time for the turbine stop valve closure and verify that the analysis of Section 15.2.2 is conservative if the initiating event for the transient is stop valve closure.

211.62 (15.2.2) Provide a plot of DNBR versus time for the loss of electrical load and/or turbine trip transient.

211.63 (15.0) Many of the sections in Chapter 15 refer to other transients as providing a limit for the transient under consideration (e.g., 15.2.1 refers to 15.2.7). Provide the appropriate justification and bases to support this position.

211.64 (15.2.5) The loss of condenser vacuum transient is stated by Section 15.2.5 to be bounded by the turbine trip analysis of Section 15.2.2. As currently presented, the turbine trip analysis assumes that the condenser dump system remains operational. Additional analysis reflecting loss of the condenser must be presented either here or in Section 15.2.2.

211.65 (15.2.6) Provide your justification for classification of the loss of AC power as an infrequent event. This justification should include an actual operational data base.

211.66 (15.2.6) Section 15.2.6.2 states that for a loss of nonemergency AC power, "It is assumed that the operator further opens the atmospheric dump valves 10 minutes after the loss of power." To take credit for operator action after 10 minutes, a complete description of each action and appropriate justification must be provided, or the assumption of no operator action for at least 20 minutes must be used.

211.67 (15.0) It is noted that many incidents of moderate frequency in Chapter 15 reference 10 CFR 100 for the dose limit. This is not in itself an acceptable reference. All analyses of events of moderate frequency must show that no fuel damage results (MDNBR<1.30) and that the peak pressures of the reactor coolant and main steam systems do not exceed 110% of design pressure. Revise or resubmit your analyses to show how Midland meets these criteria.

211.68 (15.2.7) Provide a plot showing DNBR as a function of time for the loss of normal feedwater transient.

211.69 (15.2.8) The feedwater piping break analysis infers that no fuel failure occurs. Confirm that this is correct and provide a plot of DNBR versus time to justify this conclusion.

211.70 (15.4.1) (15.4.2) Provide a sequence of events table for the uncontrolled rod group withdrawal transients.

211.71 (15.4.1) Provide the minimum DNBR and the maximum linear heat generation rate for the startup rod withdrawal accident.

211.72 (15.4.2) Provide the maximum linear heat generation rate for the uncontrolled rod group withdrawal at power transient.

211.73 (15.4.4) Provide a plot of DNBR versus time for the startup of inactive reactor coolant pumps.

211.74 (15.5.1) With regard to an inadvertent operation of ECCS during power operation, Section 15.5.1.2 states that after reactor trip, (15.5.1) the operator terminates HPI flow. Is this action necessary for plant safety? Provide a sequence of events table which includes the time frame for operator action. Also, provide figures showing appropriate plant parameters as a function of time (pressure, DNBR, pressurizer level, etc.).

211.75 (15.6.1) Section 15.6.1 states that the inadvertent opening of a pressurizer safety valve is limited by the small LOCA analysis and thus is not specifically analyzed. This transient is defined as an incident of moderate frequency and thus the acceptance criteria for a LOCA do not apply. Provide an analysis for this transient evaluating the consequences with respect to the minimum DNBR limit of 1.30 or demonstrate that the small LOCA analysis results meet the acceptance requirements for an incident of moderate frequency.

211.76 (15.6.2) With regard to a break in an instrument line or line from a primary system that penetrates containment, discuss the effects of an additional single active failure resulting in the failure of the letdown isolation valve to close.

211.77 (15.4.6) With regard to the dilution event, verify that the maximum dilution rates given in Table 15.4-11 are conservative, especially for lower reactor vessel pressures. (See Figure 6.3-3 which shows a flow rate from one pump of approximately 350 gpm at 2000 psi and increasing to 600 gpm at runout.) Provide plots of appropriate plant parameters versus time for the dilution accident at power (e.g., power level, RC pressure, DNBR, etc.).

211.78 (15.4.6) Recently, an operating PWR experienced a boron dilution incident due to inadvertent injection of NaOH into the reactor coolant system while the reactor was in a cold shutdown condition. Discuss the potential for a boron dilution incident caused by dilution sources other than the CVCS.

211.79 (15.1.5) Table 15.1-5 gives a time for auxiliary feedwater flow initiation of 16.3 seconds, or 15 seconds after the initiating setpoint. This is inconsistent with the value of 40 seconds given in Section 10.4.9.2.3. Correct this discrepancy and verify that the proper delay was assumed in the steam line break analysis.

211.80 (15.2.6) With regard to a loss of AC power, Section 15.2.6-2.a infers that the operator initiates the CVCS for addition of boric acid to maintain shutdown margin. Discuss the time frame associated with this operator action and show that it is acceptable.

211.81 (15.1.4) The response to question 211.2 does not provide sufficient information to justify using the steam pressure regulator malfunction as the bounding analysis for the inadvertent opening of steam generator atmospheric dump or safety valve. Provide the appropriate steam flows resulting from the turbine throttle valve valve wide open condition, the safety valve and dump valve flows and show that your analysis assumptions representing these steam flows are justified.

211.82 (15.0) The response to question 211.15 does not provide sufficient information for the staff to make an adequate evaluation. Provide a discussion of the loss of instrument air event for the Midland plant. Recently, a loss of instrument air at an operating plant caused a loss of reactor coolant pump (RCP) seal water injection flow and component cooling water to the RCP thermal barrier and a resultant need for cooldown with natural circulation. Please provide a complete discussion addressing all systems important to plant operation (CVCS, component cooling water, auxiliary systems, etc.) which are affected by a loss of instrument air. Show that the loss of air would not introduce a failure mode which would prevent safe shutdown of the plant and address all potential system interactions. Discuss any actions which would have to be taken by the operator.

211.83 (15.0) With respect to a break in a high or moderate energy piping system outside containment (DHR, CVCS, letdown, etc.) provide the following:

(1) Determine the maximum discharge rate from the systems based on its classification as a high or moderate energy line.

(2) Determine the time frame available for recovery based on the discharge rates calculated above and their effect on core cooling.

(3) Discuss the alarms that are available to alert the operator to the event, the recovery procedures to be used, and the time available for the required operator actions.

(4) In evaluating the recovery procedures, the single failure criterion should be applied consistent with Standard Review Plan 3.6.1 and Branch Technical Position APCSB 3-1.

211.84 (15.1.2) Section 15.1.2.3.3 infers that several nonsafety-grade systems (i.e., steam generator level control, condenser dump, and atmospheric vent) are assumed to operate during the feedwater system malfunction transient. Use of these systems would appear to reduce the effects of the transient. Justify the use of these systems as being conservative or provide an analysis which does not consider their operation (see question 211.62).

211.85 (3.5.1.1) It is noted that the pressurizer heaters and pressurizer safety valves are not included in Tables 3.5-1 or 3.5-2 nor are they discussed in Section 3.5.1.1.1. If there is justification for not considering them as potential missiles, this should be addressed in Section 3.5.1.1.1. If no justification exists, they should be included in Table 3.5-1 or 3.5-2.

211.86 (3.5.1.1) Verify that the following testing as required by Regulatory Guide 1.68 is being accomplished:

(1) Rod control system alarms are verified to operate as required.

(2) The automatic reactor power control system is verified to operate as required.

(3) At least two scram timings of each rod will be performed at extremes of temperature and flow.

211.87 (5.2.2.2) Provide the ratio of pressurizer and steam generator safety valve volumetric flow rate to the peak surge rate as obtained from the safety valve sizing analysis. Since the pressurizer safety valves may be required to discharge liquid as well as steam, provide this ratio for both liquid and steam flows. Show that this latter capacity is adequate for those transients requiring liquid relief.

211.88 (5.2.2.4) Provide or reference a discussion of the analytical basis for the steam generator safety valve setpoints, tolerance, and capacity.

211.89 (6.3) Verify that the safety-related pumps, valves, and associated controls are located abc the hypothetical water level resulting from a LOCA. Provide the predicted flood level calculation with all assumptions identified and justified for the worst break location (justify the break selected). Submit layout diagrams showing all systems below this elevation. Add a factor to the flood level calculation to account for uncertainties and justify this factor.

211.90 (6.3) Discuss how the suction line from the LPI pumps to the containment sump is maintained air free such that an air volume would not become entrapped when the sump is filling during a LOCA and cause damage to the LPI pumps when shifted to the recirculation mode.

211.91 (6.3.2.2) Discuss the recirculation flow requirem nts and provisions to protect operating LPI and HPI pumps from overheating while pumping against a shutoff head.

211.92 (6.3.2.2) It appears that in the ECC mode (suction from BWST) the HPI suction is not provided with overpressure protection. Discuss the need for such protection in the suction piping. Also, Table 9.3-14 should be expanded to include data on sizing criteria. Include possible backpressure effects. Table 9.3-14 indicates 100 percent accumulation. This error should be corrected and the selected value should be justified.

211.93 (6.3) Per Table 3.2-1, it is noted that the "BWST recirculation pump" is not Seismic Category I. Describe the function of this ECCS support component and justify its design.

211.94 (15.0) Provide the basis for the use of less conservative moderator coefficients in some of the analyses (e.g., steam line break, loss of normal feedwater flow, and feedwater line break). Figure 15.0-4, which is referenced for these events, does not show what specific value is used in the analysis and also appears to be a factor of 10 too low (10^{-5} vice 10^{-4}).

222.0 SYSTEMS ANALYSIS SECTION, ANALYSIS BRANCH

221.1 (15.1.5) Additional information is required for the steam line break accidents in Section 15.1.5 of your FSAR:

1 Provide a discussion of the calculational methods used, including all the codes used.

2 Provide a detailed flow diagram for the primary and secondary systems identifying all the components considered.

3 Describe how the initial and transient power distributions were calculated. Provide the initial and transient power distributions used in these analyses.

4 Describe in detail how the thermal-hydraulic effects were evaluated, including calculations of DNBR.

5 Provide transient axial and radial power distributions for each case analyzed. Describe how these peaking factors were considered in the thermal-hydraulic calculations. How were these peaking factors calculated?

6 Provide all the time dependent reactivity feedbacks during the accident (provide for all the cases analyzed).

7 Provide nuclear and thermal-hydraulic analyses for the first 15 seconds for both BOL and EOL conditions from full power.

8 For the high pressure safety injection system and the flow of borated water from core reflood tanks, describe the flow path into the core, the method for evaluating the time for these fluids to reach the center of the core and the method for determining the resultant reactivity feedback.

9 Describe in detail how the coolant flow reduction in the hot channel is evaluated. Discuss the potential for coolant flow blockage due to fuel swelling.

10 Describe in detail how the time dependent pressure drop in the fuel channel was calculated.

11 Provide a plot of the core coolant density and average fuel temperature for the time period from zero to 15 seconds.

12 Describe how the peaking factors in the hot channel were determined.

222.2 (15.2.8) Additional information is required for the feedwater piping break in Section 15.2.8 of your FSAR:

1 Provide a discussion of the calculational methods used, including all the codes used.

2 Provide a detailed flow diagram for primary and secondary systems identifying all the components considered by this analysis.

3 Provide and justify the worst single failure considered.

4 Provide a spectrum of feedline break accidents and identify the worst case.

5 Provide the discharge rates out of the feedwater line including the mass inventory in the steam generators during the accident.

6 Provide the time dependent primary and secondary pressures.

231.0 CORE PERFORMANCE BRANCH: FUEL DESIGN SECTION

231.1 (4.2.1) The design bases listed in the Midland FSAR are correlated with plant conditions such as normal operation (Condition I), upsets (Condition II), emergencies (Condition III) and faulted conditions (Condition IV). This categorization differs from that used in B-SAR-205. In the latter case the design bases are separated into only three categories in terms of fuel assembly loading conditions, one of which, viz. shipping and handling conditions, is not addressed in the Midland grouping. Please explain the rationale for the change in design bases categorization and the deletion of shipping and handling considerations.

231.2 (4.1.1.1) The FSAR listing of stress and strain limits concludes with the statement that "those limits are consistent with current practice." This section should also contain a brief statement regarding the origin, evolutionary history, and rationale for each limit. The limits should also be correlated with the design bases (Condition I thru IV events where applicable. For example, the relationship (if any) of the 1% inelastic + 0.4% elastic strain limit to Condition I, II and III events should be indicated; e.g., how do these limits preclude fuel failure during Condition II transients or limit failure to a small calculable fraction of rods during Conditon III accidents?

231.3 (4.2.1.1) The "cumulative fatigue damage factor" mentioned in the discussion of "Vibration and Fatigue," FSAR section 4.2.1.1.3 should be defined. Please explain also the rationale for limiting the cumulative fatigue damage factor to 90% of the allowable material fatigue life, i.e., why not some other fraction of fatigue life--for example, 80%? Why are all Condition I and II events to be encompassed by this rule, but only one Condition III event? Please show, by use of examples, when the O'Donnell-and-Langer curve is modified by a factor of two on stress amplitude and under what conditions a factor of 20 on the number of cycles is, instead, used to get a properly conservative design basis. Discuss the design features used to ensure that flow-induced vibrations do not lead to excessive fuel rod and guide tube fretting, and reference the test data and operational experience that supports this conclusion.

231.4 (4.2.1.1) Please discuss the specifications for dryness of the pellets and cladding as a Zircaloy hydriding preventive measure. Discuss the statistical sampling technique and the method of moisture detection used to ensure that the moisture has been removed.

231.5 (4.2.1.2) Tabulate the UO_2 and Zircaloy thermal-physical properties used in the fuel design analysis and reference the data supporting these values.

231.6 (4.2.1.2) The discussion of UO_2 chemical properties should list the major impurities known to impair the performance of the fuel or cladding, the level of impurity known to cause a problem, and the impurity limits used in fabrication.

231.7 (4.2.1.3) The CROV code used in the creep collapse analysis was received and accepted for use in safety analysis related to licensing subject to the following conditions:

1) The creep-related material properties used in the analysis should be similar to those characteristic of current B&W cladding.

2) The initial ovality input to CROV should both bound the as-fabricatedd cladding and be not less than 0.0005 inches. (Ovality = OD max. - OD min)

3) The results of long-term, inreactor confirmatory tests will continue to be favorable.

Please indicate how these conditions were met in the use of the CROV code to analyze creep collapse in Midland fuel.

231.8 (4.2.1.3) In our review of the "TACO" thermal analysis and fuel performance code, we determined that several modifications to the code were needed, including a revision of the fission gas release model to account for enhanced release at high exposure. The Midland FSAR does not reference the approved version of TACO, viz. BAW-10087A, Rev. 1, August 1977. Please reference and use the approved version of TACO for the Midland Fuel Thermal Analysis and modify the FSAR as required.

231.9 (4.2.1.5) The FSAR discussion of fuel assembly structural design contains the following statement: "The Fuel Assembly is designed to ensure safe operation for Condition I and II events." One can interpret this statement to have, by implication, one or more of the following meanings:

a) Condition III and IV events need not be handled safely.

b) Condition III and IV events need not be considered in the design analysis.

c) Condition I and II events constitute a definition of the term "operation," but III and IV do not.

Please indicate which, if any, of these interpretations is correct and explain the rationale for the interpretation. Show how the stress intensity limits used for stainless steel and zirconium - based alloys bound the calculated maximum expected loadings for Condition I thru IV. List, in tabular form, numerical values for these limits and loadings and show how the loadings were calculated.

231.10 (4.2.1.6) Please list the numerical values and equations, along with their reference sources, for the pertinent thermal-physical properties used in the design of the Ag-In-Cd and Al_2O_3-B_4C absorber materials. A minimum list of pertinent properties should include melting point, swelling, thermal conductivity, thermal expansion and gas release. Also list the calculated expected values for end-of-life swelling and gas release and compare these to the maximum allowable design values under normal and off-normal conditions; i.e., Conditons I thru IV.

231.11 (4.2.1.6) Discuss the fabrication specifications of the Al_2O_3-B_4C lumped burnable poison pellets; in particular discuss the specifications to ensure that residual moisture levels are below those which could result in hydriding and perforation of the Zircaloy-4 cladding. If the poison rod cladding were perforated, the B_4C would react with primary coolant water to form H_3BO_3, which would then be leached into the coolant. Please discuss the potential safety implications of the reactivity insertion resulting from the loss of B-10 from the burnable poison rods via this mechanism. Would resulting power changes be detected?

231.12 (4.2.1.6) List and discuss the testing performed to show that the absorber materials are compatible with their respective cladding materials i.e., Ag-In-Cd alloy with stainless steel and Al_2O_3-B_4C with Zircaloy-4. Show the respective rate equations and the amount of attack over the design life.

231.13 (4.2.1.6) Please demonstrate that the use of minimum unirradiated strength values for the control rod and burnable poison rod cladding alloys is conservative under all postulated reactor conditions; e.g. demonstrate that the increased strength due to irradiation is not affected by a decrease in ductility. Please discuss the bases for the 1% and 3% strain limits for 304SS and Zircaloy-4 cladding respectively. Show how these limits are consistent with analytical and test results, as stated in FSAR section 4.2.1.6.3. Please list and briefly describe the control component examinations mentioned in FSAR section 4.2.1.6.4.

231.14 (4.2.1.7) The discussion of fuel surveillance, which includes a post-irradiation examination (PIE) program, requires considerable amplification. Please provide a listing of the PIE tests performed and in progress, discuss the results to date and anticipated, and show how the results provide verification of the adequacy of the fuel design.

231.15 (4.2.3.1) The discussion of flow-induced vibration and fretting requires extensive amplification in regard to treatment of the out-of-core testing said to have demonstrated that flow-induced vibration amplitudes of the fuel rods do not cause fretting for PWR operating conditions. The discussion in FSAR section 4.2.3.1.4 does not provide enough quantitative information on the fretting and wear test program on either the fuel rods or control rod guide tubes. Please cite published references for the tests mentioned in this FSAR section. The statement, in FSAR section 4.2.3.1.1, regarding the confirmation of these results by PIE of production B&W fuel also requires support with specific documented examples.

231.16 (4.2.3.1) Please provide numerical values for fuel rod stresses caused by (a) pressure differential (b) ovality bending, (c) thermal, and (d) grid loads for the worst case Condition I thru IV events. Provide numerical evidence to support the assertion that differential fuel rod growth and flow-induced vibration stresses do not affect these worst case stresses.

231.17 (4.2.3.1) Please provide the coolant chemistry specifications said to control the oxidation of the fuel rod cladding and provide data to support this assertion. What provisions are made to control cladding oxidation and crud deposition? Please provide further support, in the form of referable data for the statement that "the majority of the SCC (stress corrossion cracking) experience has been reported for conditions not representative of B&W operating conditions of current design."

231.18 (4.2.3.1) The rod bowing correlation presented in FSAR section 4.2.3.1.8 has not been approved by NRC. The currently acceptable correlation for thermal hydriding calculations for B&W Mark B 15x15 fuel rods is as follows:

$$(\Delta C/C)_{95} = 0.065 + 0.00145\sqrt{BU}$$

where BU is MWd/tU and $(\Delta C/C)_{95}$ is a hot, fractional, 95/95 closure. These coefficients meet the statistical requirements necessary for use in directly assessing DNBR penalties. Please correct the cited FSAR section and revise the rod bowing analysis using the corrected correlation.

231.19 (4.2.3.1) Show the relationship between the fuel rod cladding swelling design curve and rod growth data refered to in FSAR section 4.2.3.1.10. List the data sources.

231.20 (4.2.3.3) Please reference or provide the "recent irradiated cladding ductility data" asserted to indicate that "current production cladding under typical operating conditions retains a ductility with irradiation in excess of that which would lead to a PCI concern." What "atypical" or off-normal operating condition would be expected to cause a PCI problem, based on these ductility data and analyses (also see Q231.16).

231.21 (4.2.3.4) (4.2.3.4) Please discuss the test data and analyses which support the assertion that frictional contact between the spacer grids and fuel rods is adequate to maintain rod position while not leading to undesirable forces due to differential fuel rod growth.

231.22 (4.2) Please provide the dimensions and spring constants for the upper and lower plenum springs and show quantitatively that the resistance to creep and relaxation of the spring alloy is sufficient to withstand the worst postulated flux, temperature, and stress conditions.

231.23 (4.2) Please describe the extent to which the fuel handling and shipping design loads have been confirmed experimentally.

400.0 PROJECT MANAGEMENT

400.2 (3A) Some of the regulatory guides determined by our Regulatory Requirements Review Committee (RRRC) prior to January 1, 1978 to be applicable to all nuclear power plants, are not addressed in Appendix 3A of your FSAR, or your discussion is not based upon the latest applicable revision. These regulatory guides reflect current NRC staff practice. Therefore, except in those cases in which you propose an acceptable alternative method for complying with regulations on which these guides are based, the methods described in these guides are being and will continue to be used to evaluate the subject matter of the guides.

Revise FSAR Appendix 3A to discuss how Midland Plant Units 1 & 2 conform to these regulatory guides, whether by adherence to the positions recommended therein or to an acceptable alternative.

Applicable Regulatory Guide Revision	Revision Currently In FSAR	Subject
RG 1.99, Revision 1 (4/77) (See related requests 121.5, 121.10 & 121.12)	Revision 0 (7/75)	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials
RG 1.101, Revision 1 (3/77)	Revision 0 (11/75)	Emergency Planning for Nuclear Power Plants
RG 1.114, Revision 1 (11/76)	Revision 0 (2/76)	Guidance on Being Operator at the Controls of a Nuclear Power Plant
RG 1.127, Revision 1 (4/78)	None	Inspection of Water-Control Structures Associated with Nuclear Power Plants

421.0 QUALITY ASSURANCE BRANCH

421.3 Our review of the quality assurance aspects in Section 4.3.c of your Midland Plant Fire Protection Evaluation Report indicates the need for the following information:

1. Indicate whether the QA program for fire protection is under the management control of the QA organization. This control consists of (1) formulating and/or verifying that the fire protection QA program incorporates suitable requirements and is acceptable to the management responsible for fire protection and (2) verifying the effectiveness of the QA program for fire protection through review, surveillance, and audits. Performance of other QA program functions for meeting the fire protection program requirements may be performed by personnel outside of the QA organization. The QA program for fire protection should be part of the overall plant QA program. These QA criteria apply to those items within the scope of the fire protection program, such as fire protection systems, emergency lighting, communication and emergency breathing apparatus as well as the fire protection requirements of applicable safety-related equipment.

2. Address in additional detail the ten specific quality assurance criteria in Branch Technical Position APCSB 9.5-1. Examples of the detail we need to evaluate these criteria are provided in Attachment 6 of Mr. D. B. Vassallo's letter of August 29, 1977.

   Alternatively, you may apply the same controls to each criterion that are commensurate with the controls described in your QA topical report CPC-1-A. These controls would apply to the remaining construction activities and for the operations phase of Units Nos. 1 and 2. If you select this method, a statement to this effect should be indicated in your report.

422.0 CONDUCT OF OPERATIONS

422.7 Our review of the organizational aspects in Section 4.1.A of your Midland Plant Fire Protection Evaluation Report indicates the need for the following information:

1. Describe the offsite management position that has responsibility for periodically assessing the effectiveness of the fire protection program, including program activities such as fire drills and fire brigade trainings.

2. You state that the fire protection staff consists of the General Supervisor of Property Protection Engineering and designated Midland power plant personnel. Describe the specific delegation of authority to designated Midland plant personnel for specific program activities such as fire prevention activities, fire safety inspections, fire fighting training, procedures and drills, and maintenance of fire fighting equipment and systems.

3. Describe who will be assigned as head or leader of your fire brigade. If they are not your Plant Supervisors, describe their authority relative to your Plant Supervisors for actions that might affect safety-related systems or equipment.

432.0 EMERGENCY PLANNING BRANCH

432.7 (SEP 5.1) In Section 5.1 of the Site Emergency Plan (SEP), differentiate between plant personnel and unit personnel. The section does not clearly indicate that the plant will have two operating teams, one for each unit.

432.8 (SEP 5.0) Figure 5-1 of the SEP appears to show that the Midland City Fire Department may use its sirens for public notification in case of an evacuation or take cover operation. Indicate whether this is infact the case and if so, provide a discussion of the use of the sirens. This discussion should include the means by which the public will recognize that the sirens are sounding a nuclear emergency.

432.9 (SEP 6.1) In Section 6.1 of the SEP, provide the radiation instrument response levels that will be used to categorize an accident as a Site Emergency or a General Emergency. Show the relation of these levels to the State and Federal government protective action guides. Include the containment radiation monitor in the list of applicable instrumentation. Discuss the use of process instrumentation in classifying an incident as a Site Emergency or a General Emergency.

432.10 (SEP 5.4.8) Section 5.4.8 of the SEP clearly states that "the Site Emergency Director will contact the Midland City Police Office directly and immediately" in the case of serious radiological releases. However, this is contradicted in Section 6.1.4. Amend Section 6.1.4, General Emergency, to specifically address the question of "early warning of the public and prompt initiation of protective action," as requested in Section 4.1.5 of R.G. 1.101 (Revision 1).

432.11 (SEP 6.4.1) Amend Section 6.4.1 of the SEP in response to the following items:

a. State whether or not the Midland City Police has jurisdiction over the offsite areas included in the evacuation and take cover plans. If more than one agency has jurisdiction, indicate how they will coordinate their efforts.

b. State whether those agencies involved in Item (a) have been given authority to begin protective measures, e.g., evacuation, without the need for obtaining approval from any other agency.

c. Your plan states that the emergency response and evacuation plans of offsite State and local agencies will be added to the appendix of the site emergency plan. Provide these plans or indicate their status and provide a schedule for when they will be made apart of the docket record.

d. Discuss the offsite agencies' plans involving areas beyond the LPZ.

432.12 (SEP 6.4.2) (3A)

With respect to respiratory protection, Section C.4.c. of Regulatory Guide 1.95 discusses provisions for an offsite supply of bottled air. Amend Section 6.4.2 of the SEP to address this offsite supply.

432.13 (SEP 8.1)

Amend Section 8.1 of the SEP to include provisions for annual training of the offsite firemen that will ensure their familiarity with the plant, access procedures, and radiation protection precautions; and for their participation in an annual drill or test exercise.

432.14 (SEP 8.2) (3A)

Section 8.2 of the R.G. 1.101 requests that all written agreements be updated at least every two years. All letters of agreement appearing in Appendix A of the Emergency Plan should be kept updated; replace all expired agreements. In addition add a provision to Section 8.2 of the plan indicating your policy involving the length of time a letter agreement will remain inforce, the procedure used to update the agreement, and the procedures for terminating or modifying an agreement.

432.15 (SEP 3.0)

Provide a listing, by title, of written procedures that implement the plan as requested in Section 10 of R.G. 1.101.

ERRATA SHEET

REPLACE THE SAME NUMBERED REQUESTS IN OUR LETTER OF

FEBRUARY 24, 1978 WITH THE FOLLOWING:

022.13 (6.2.6.1) For each gas or liquid filled system penetrating containment. discuss your reasons for not venting and draining portions of the system piping to expose the containment isolation valves to the containment atmosphere and full differential pressure for the containment integrated leak rate test.

022.16 (6.2.6.3) It is our position that all isolation valves provided to satisfy General Design Criteria 54 through 57 (containment isolation valves) be pneumatically (Type C) leak tested. Alternatively, a containment isolation valve may be exempted from the Type C test requirements if it can be shown that the valve does not constitute a potential containment atmosphere leak path following a loss of coolant accident.

Table 6.2-28 indicates the containment isolation valves that will not be Type C tested. Therefore, justify that they do not constitute potential containment atmosphere leak paths following a LOCA. In this regard, a water seal may be shown to exist that will preclude containment atmosphere leakage. If this approach is taken, discuss how a water seal can be established and maintained using safety grade pipes, components and considering single failures of active components. In addition, provide system drawings showing the routing and elevations of the piping.

REQUEST FOR ADDITIONAL INFORMATION RESAR-414

Distribution:
NRC PDR
Local PDR
Docket File
LWR #4 File
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