NRC PDR

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

DEC 15 1978

Docket Nos.: 50-329
50-330

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

Dear Mr. Howell:

SUBJECT: STAFF POSITIONS AND REQUESTS FOR ADDITIONAL INFORMATION (PART 2)

My letter of December 11, 1978 forwarded part 1 of our requests for additional information and our positions that differ from those in your FSAR. Part 2 of our requests and positions is contained in Enclosure 1 hereto.

We will need response and resolution to Enclosure 1 by January 19, 1979. If you cannot meet this date, inform us within seven days after receipt of this letter so that we may revise our schedule accordingly.

Should you desire clarification of Enclosure 1, please contact us.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Light Water Reactors Branch 4
Division of Project Management

Enclosure:
As stated

cc: Listed on following page

7901090041

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ENCLOSURE 1

STAFF POSITIONS (Q-2s) AND REQUEST FOR ADDITIONAL INFORMATION

PART 2

MIDLAND PLANT UNITS 1 & 2

These positions and 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.

211.0 REACTOR SYSTEMS BRANCH

211.176 (15.2) (5.2.2) (RSP) Your response to request 211.147 provides the initial conditions for BAW-10043 to show that it "brackets" the Midland units. It is not clear that BAW-10043 bounds the Midland units. Comparison of parameters from the Midland FSAR and your response is as follows:

	BAW-11043	Midland FSAR
Core Power (MWt)	3105 (112%)	2452 (100%)
Pump Heat (MWt)	16	16
RCS Flow Rate (lb/hr)	137.9×10^6	126.3×10^6
Pressurizer Code Safety Valve Capacity (lb/hr)	690,000	595,690
Secondary Safety Valve Capacity (lb/hr)	13,680,000	12,484,520

The effects of less flow and relief valve capacity are not obvious relative to the lower power level. Submit a plant-specific overpressure valve sizing calculation for Midland.

Also, it is our position that the analysis assume that the reactor scram is initiated by the second safety grade signal from the reactor protection system. Your above analysis for Midland should be performed accordingly.

211.177 (6.3) (5.4.7) (RSP) Your response to request 211.131 does not satisfy our concern with respect to the detection and isolation of passive ECCS failures during the long-term cooling phase after a LOCA. Although the test report addressing injection pump seals indicates that seal integrity was maintained for the conditions under which they were tested, we do not concur with your proposal for LPI seal leakage of 500 ml/min to serve as the bounding leak rate for a passive failure following a LOCA (valve stem packing or pump seal failure). Operating data indicates that leak rates in excess of your proposal have occurred. "Bounding" leak rate assumptions on the order of 30-50 gpm have been accepted by the staff in the past; therefore, we require that you show that the ECCS equipment layout, room water level detectors and airborne radiation monitors in the Midland plant meet the criteria listed in request 211.47 assuming leakage rates of this magnitude, or revise your design accordingly.

211.178 (6.3) (RSP) Your response to our position in request 211.129 does not provide assurance that the single vent on the BWST is adequate since:

1. It is on top of the tank and would be susceptible to blockage due to snow buildup.

2. No heat tracing is provided on the vent.

3. Your response does not describe the "screen inside the BWST" which is heated.

We require that a BWST vent configuration be provided which will preclude vent blockage due to icing or snow accumulation. Revise your design accordingly.

211.179 (6.3) (RSP) Your response to request 211.126 states that flow indication in the "dump-to-sump" lines is not necessary. Our position is that the operator must be provided with flow indication to confirm that at least the minimum required dilution flow exists subsequent to a LOCA. Revise your design and response accordingly.

211.180 (5.2.5) (RSP) Your response to request 211.106 states that the alarm provided in the control room to detect a reactor building sump level increase corresponding to 1 gpm leak within 1 hour will be generated by the plant computer. Since the plant computer may not be available during plant operation, we require that an alarm be provided in the control room which will be available at all times. Revise your design accordingly.

211.181 (6.3) Your response to request 211.113 states that extended operation of the Decay Heat Removal pumps at flows less than 800 gpm would result in damage to the pumps. (This was your basis for not using continued recirculation through the DHR heat exchanger and recirculation line to protect the DHR pumps from closure of a suction valve). Confirm, with basis, that the low pressure injection system will perform its function in the piggyback mode, since the LPI (DHR) pump flow will be less than 800 gpm.

211.182 (6.3) (RSP) Your response to request 211.103 does not meet our requirements with respect to check valve leak testing. The proposal to test two valves in each of the Core Flood and Low Pressure Injection Lines is acceptable for these systems, however, we require that at least two check valves in each of the high pressure injection lines be tested also. This should be done by classifying these valves as AC in accordance with Section XI of the ASME Code. Modify your response accordingly.

211.183 (15.4.5) (RSP) The response to request 211.152 does not satisfy our concern that dilution events could occur at rates less than the makeup flow rate setpoint, and would not be detected. Although these events would take longer than 30 minutes to reach criticality, no indication would be provided from the high makeup flow alarm to alert the operator to terminate the event. We require that the operator have adequate time after indication of the event in accordance with the following criteria:

Plant Condition	Time Prior to Criticality After Indication
Refueling	30 minutes
Startup, cold shutdown, hot standby, and power operation	15 minutes

Provide assurance that these criteria are met or revise your design accordingly.

211.184 (15.2) During the recent review of the loss-of-offsite-power preoperative test procedure for another plant, a concern arose regarding the control of OTSG level by the auxiliary feedwater system during the event. Specifically, overcooling of the primary system could result from feeding the OTSG with the cold auxiliary feedwater. The cooldown could be large enough to empty the pressurizer and cause a steam bubble to form in the hot leg high points, which could impede natural circulation and core cooling. Address this concern for the Midland units. Provide the results of an analysis of a loss-of-offsite power assuming the worst-case initial conditions (low power appears to be worst since programmed steam generator level is lowest). Include plots of steam generator level, reactor coolant system temperature, and pressurizer level. Discuss your assumptions regarding auxiliary feedwater control. Show that MDNBR will remain above 1.30 and core cooling will not be impaired.

211.185 (15D) (15.1.5) (RSP) Your response to request 211.157 regarding worst case single failure for a main steam line break is insufficient. The analysis should consider the following:

1. Inadvertent atmospheric dump valve opening

2. Steam flow through all unisolated lines down stream of the MSIV's (Unit 2). Table 10.3-5 indicates that all lines are not isolated after a steam line break assuming the single failure of one MSIV.

3. Process steam cross-connect valves opening, (see request 211.160 unless power will be removed.

Provide your basis for stating that one HPI pump is the worst single failure with respect to overcooling.

Provide the worst single failure with basis for the worst DNBR main steam line break.

211.186 (15D) (15.1.5) Confirm that the bounding Midland Chapter 15 accidents and transient analyses have considered all events which could occur in Modes 1, 2, 3, and 4 as defined in FSAR Section 7.7.1.6.2.2.

We require that all allowable modes of operation be considered in your safety analysis and be specifically defined in the Midland Technical Specifications. We also require that all modes which are physically possible but which have not been considered in your safety analyses (e.g., Unit 1 NSSS supplying the Unit 2 turbine) be identified and be specifically prohibited by the Midland Technical Specifications.

321.0 EFFLUENT TREATMENT SYSTEMS BRANCH

321.6 (11.4) Justify your position that the proposed extruder/evaporator has the capacity for the combined input from Midland Plant, Unit Nos. 1 and 2. Your estimates of annual quantities of solid waste and your comparison to other plants in Tables 11.4-1 and 11.4.5 of the FSAR is at variance with the capacity of the proposed solidification system. You should consider the data from operating PWR's such as is given in NUREG/CR-0144, "A Review of Solid Radioactive Waste Practices in Light-Water-Cooled Nuclear Power Plants," ONRL/NRC, October 1978, and the expected feedrate for the VRS-T120 extruder/evaporator as recommended in Amendment 1, Table III, of the Topical Report WPC-VRS-001 (Revision 1), May 1978.

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