UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of:	)	
	)	Docket Nos. 50-329
CONSUMERS POWER COMPANY	)	50-330
	)	
(Midland Plant, Units 1 and 2)	)	Operating License

REVISED CONTENTIONS OF MARY SINCLAIR BASED
ON DISCOVERY PURSUANT TO BOARD ORDER OF MAY 25, 1982

August 12, 1982

Contention 28 deals with the water hammer problem of pressurized water reactors of the Midland type. This problem is identified as one of the unresolved safety issues applicable to Midland 1 & 2 in the SER, C-4. Babcock and Wilcox(B&W) plants with an internal auxiliary feedwater (AFW) feed ring of the same design as Midland in recent events, have shown a marked susceptibility to internal damage of the feed ring as a result of water hammer. From this, reduced cooling in the steam generators could occur as a result of inadequate AFW flow following loss of normal feedwater flow. (NRC Response to Interrogatory 7) Since this effect involves critical safety systems, the Task A-1 report (Jan., 1980) states that systematic review procedures in the OL review process will require the Applicant to : 1) address potential water hammer problems in various systems; 2) demonstrate that there are adequate design features and operating procedures to prevent damaging water hammer events; and 3) expand the preoperational testing program to insure that these design features and operating procedures do prevent damaging water hammer events.

However, the SER does not indicate that these criteria have been met by the Applicant. As a result of this omission, the findings required by 10 CFR §§50.57 (a)(3)(i) and 50.57(a)(6) cannot be made.

Contention 30. The degradation of steam tube integrity due to corrosion induced wastage, cracking reduction in tube diameter, and vibration induced cracks

is a serious unresolved safety problem at the Midland nuclear plant. It is admitted that the chemistry of the cooling water is critical to prevention of steam tube failure, (NUREG-0886). However, the fact that these plants depend on cooling water from the cooling pond increases the likelihood of corrosion and poor water chemistry because the DEIS states that the plant dewatering system will first be discharged to the cooling pond. (DEIS at 5-2). That means that many wastes, including radioactive materials from leaks and spills on the reactor site, can enter the cooling pond and disrupt the chemistry of the pond. Therefore, due to this contribution of an undertermined amount and quality of ground dewatering inflows to the cooling pond, the NRC's bland assurance that corrosion is unlikely due to the lack of sodium thiosulfate, is unsatisfactory. (NRC Response to Interrogatory 9.j.) In fact, due to the contribution of groundwater, the NRC is not fully aware of the likely constituents of the cooling pond, and the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

Contention 31. Numerous non-safety related systems, the feedwater system, main stream system, makeup and pernification system, non-vital electrical power systems, and the integrated control systems, can adversely affect safety related systems, such as Anticipated Transients Without Scram (ATWS). (NRC Response to Interrogatory 10.c) Since there has been no routine inspection and quality control standards applied to these non-safety systems, and the general quality control during construction of even safety related systems has been so poorly done (amply documented in the record of these hearings), there is an even greater probability of ATWS at Midland. However, this scenario has not been analyzed in the SER. Furthermore, B&W reactors, such as the Midland reactors, experience the largest pressure rise and thus are the most difficult to modify to achieve adequate safety margins to prevent ATWS events. (NUREG-0460, April, 1978, p 46) Therefore, the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

Contention 32. There is no assurance that suitable safety margins can be maintained throughout the design life of the Midland plant with the materials used for reactor vessel fabrication. This makes the Midland reactors unusually susceptable to reactor embrittlement and to pressurized thermal shock (PTS). For

example, an investigation following the severe PTS at the Rancho Seco reactor indicated that the limiting material in the Rancho Seco reactor vessel was fabricated using the same weld wire and flux as the limiting material in the Midland reactor vessel beltline and has equivalent chemical composition and fracture toughness properties. This indicates that the staff's conclusions concerning the Rancho Seco reactor vessel beltline materials are applicable to the Midland Unit 1 reactor vessel beltline materials. (NRC Response to Interrogatory 11.e) Furthermore, in a memorandum to the Midland file, dated June 14, 1977, by G.S. Keeley of Consumers Power Co. and sent to S. H. Howell, et al., described a memorandum which A. J. Birkle had written to R. C. Bauman on March 22, 1977, on the status of Midland NSSS-12 reactor vessel girth weld fracture toughness. (Discovery Response, Consumers Power Co.) This memorandum pointed out that there was 'a chance that the NSSS-12 reactor vessel could have a low level of fracture toughness at the operating temperature after 10 years of operation'. The low level was with reference to the 50 ft-lb upper shelf criteria of 10 CFR 50, Appendix G & H. It also indicated that this could possibly be corrected by annealing the vessel which is not now a viable approach although an EPRI R&D effort is underway." Moreover, Demetrias Basedekas, NRC reactor safety engineer, in a memorandum addressed to Chairman Palladino (NRC Response to Interrogatory 11.a) made the following major points which emphasize the importance of this deficiency concerning PTS:

> "Substantial uncertainties and non-conservative assumptions in estimates of consequences and of probabilities cast serious doubts on the validity of conclusions stated by industry and the NRC staff.
>
> The lack of badly needed design information on control and electrical power systems, and related neutronic and thermal-hydraulic parameters for representative plants (at least one for each NSSS vendor) makes an independent and thorough assessment of this issue by NRC virtually impossible.
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> Substantial operation experience with PTS precursor events involving control system and steam generator tube failures, coupled with an understanding of functional and some design aspects of control systems and components in operating plants, suggest an unacceptable level of risk associated with a number of older pressurized water reactors."

These points, as well as the fact that the Midland nuclear plants were designed over a decade ago, and contain the same defective material as the Rancho Seco nuclear plant means that findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57 (a)(6) cannot be made.

Contention 35. Assurance of pressure vessel integrity and the ability to detect and adequately size flaws depends, for one thing, on carefully controlling the fabrication, welding and examination of welds to minimize the probability of significant weld defects. The affidavits secured by the Government Accountability Project and recently turned over to NRC, especially that of Dean Dartey and one of the anonymous workers, describes extensive failures in welding. (Midland Daily News, July 20, '82) Therefore, the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

Contention 36. Systems interactions, identified as an unresolved safety problem applicable to Midland in the SER (C-4), has special significance at Midland because the most serious accident resulting from systems interaction failures have occurred in B&W reactors. The serious events and their special problems with system interaction include the following:

> 1) The persistant operator disbelief of high temperature data from incore thermocouples and system RTD's was one major, out of many, causes for the TMI-2 accident. This disbelief was based on the rationale that the former were not safety-grade equipment while the latter were outside the calibrated range of the detectors. (NUREG-0600, p 10, and "Daniel Ford, Three Mile Island, Thirty Minutes to Meltdown") In the case of the high temperatures, acceptance of the temperature data as valid might have prompted a higher high-pressure-injection flow rate and a reluctance to subsequently depressurize the plant to use the core flood tanks. (NUREG-0600 , p 11) This is one example of non-safety related equipment impacting on safety systems.
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> 2) At Crystal River, an accident on February 26, '80, is of interest because of systems interaction where the integrated control system input, the PORV positioning, the instruments used for manual control of ECCS and the entire non-nuclear instrumentation (NNI) power supply depended on one and 24 VDC line within the NNI power supply system. (NUREG-0667)

3) At Davis-Besse I on April 19, 1980, maintenance activities allowed an elimination of redundant power supplies that were supporting the decay heat removal function. Concurrent construction activities caused the loss of working power supply and subsequently decay heat removal was lost for over two hours. (USNRC IE Information Notice 80-20, May 8, 1980) (NRC Response to Interrogatory 15.e)

In spite of this repeated history of system interaction problems at B&W reactors, the staff SER specifically fails to require a comprehensive program to reportedly evaluate all systems which could interact. (SER at C-12.) Moreover, the apparent use of non-safety grade materials for safety grade functions at Midland significantly increases the risk of adverse system interactions. (Howard affidavit).

Contention 40 deals with lack of adequate qualification methods to satisfy the rquirements for safety related equipment.

Contrary to NRC Response to Interrogatory 19 (a), a Commission decision in the UCS Petition for Emergency and Remedial Action (CLI 80-21, May 27, 1980), 11 NRC 707, requires that all plants under licensing review must meet the equivalent of the IEEE 1974 Standard in order to satisfy GDC 4 (10 CFR 50, Appendix 4). In fact, the SER admits that this standard has not been met. (SER p 3-36) Thus, absent further action, the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57 (a)(6) cannot be made.

Contention 45. There is no assurance that offsite power is sufficiently reliable to ensure the maintenance of safety functions during accident conditions. In one of the anonymous GAP affidavits, an electrician described the poor quality control that has gone into the electrical work at the Midland nuclear plant. He stated that the cables shop substituted control cables when the correct type was unavailable. He explained that a cable design may have called for three shielded pairs of 16-guage wire but the cable shop in which he worked would use six stranded 16-guage wire with the shielding around the entire bundle. (Midland Daily News, June 28, 1982)

These types of electrical cable deficiencies built into many parts of the plant do not comply with the General Design Criteria, therefore the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

Contention 50. The occupational exposure of regular workers or transient workers at the Midland nuclear plant cannot be controlled as the NRC Response to Interrogatory 29(a) states, because of the extensive quality control failures that the disclosures of Zack Co. employees and Dean Dartey indicate have been built into the heating, ventilating and air conditioning system at the Midland nuclear plant. Therefore, the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57(a)(6) cannot be made.

Contention 52. The reliability of the emergency onsite diesel generator at Midland is seriously in question. The NRC staff has stated that: "The excessive settlement and cracking of the diesel generator building due to improperly compacted soil can seriously and adversely affect diesel generator performance since this can cause excessive differential movement between diesel generator and building foundations." (NRC Response to Interrogatory 31.d) Also, there is concern at Midland for damaging fuel oil and service water lines entering and exiting the building. Therefore, the findings required by 10 CFR §§ 50.57(a)(3)(i) and 50.57 (a)(6) cannot be made.

Respectfully submitted,

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