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Consumers Power Company

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February 22, 1982

Harold R Denton
Office of Nuclear Reactor Regulation
Division of Licensing
US Nuclear Regulatory Commission
Washington, DC 20555

MIDLAND PROJECT
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION CONCERNING
UNRESOLVED SAFETY ISSUES (QUESTION 400.9)
FILE 0505.8 SERIAL 15993
REFERENCES: (A) NRC (E G ADENSAM) LETTER DATED 11/30/81
(B) CP CO (J W COOK) LETTER DATED 12/11/81, SERIAL 15092
ENCLOSURE: RESPONSES TO UNRESOLVED SAFETY ISSUES

Reference (A) requested that Consumers Power Company (CPCo) provide a summary description of any relevant investigative programs and any interim measures the Company has devised for dealing with certain unresolved safety issues (USIs) applicable to the Midland Plant. Reference (B) provided an interim response and requested a meeting to be held between CPCo and your Generic Issues Branch (GIB). The meeting was held on December 22, 1981 in which the GIB provided additional information as to the detail and nature of the original request (Question 400.9). The response is provided in the enclosure. In addition to the information requested, a response concerning a newly designated USI, Pressurized Thermal Shock (A-49), is included (refer to SECY-81-687).

Where a resolution to the USI has been proposed by the NRC as indicated in NUREG-0606, a discussion of how the issue has been addressed on the Midland Plant, including supporting sections of the FSAR, is given. In those instances where a generic resolution has not yet been proposed by the NRC, a discussion of why it is appropriate for the Midland Plant to commence operation and supporting information on the Midland Plant relative to the particular issue are provided.

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CPCo believes that the information presented in the enclosure provides a basis for concluding that the Midland Plant can be safely operated while longer-term generic review on these issues is still underway.

James W. Cook

JWC/JNL/acr

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UNRESOLVED SAFETY ISSUE A-1

Water Hammer

Incidents that have occurred since 1971 in pressurized water reactors (PWRs) have established water hammer (or steam hammer) as a potential problem in fluid systems. Systematic investigations of the water hammer phenomena in all types of PWRs have established that this is not a significant problem in Babcock and Wilcox designed nuclear steam supply systems. The studies indicate that most of the recorded water hammer events in PWRs occurred when cold auxiliary feedwater was introduced into the main feedwater piping after the feedwater ring spargers (Westinghouse and Combustion Engineering systems) were uncovered, drained and refilled with steam.

There have been only four occurrences of water hammer reported for B&W-designed units. All occurred during start-up and no damage was reported. Changes in design and operating techniques have eliminated any further incidences at these units.

There are several significant design features inherent in the B&W systems that preclude the occurrence of water hammer loads of a destructive nature. These are as follows:

a. The main feedwater piping joins the ring header from below, forming a water seal that minimizes the chance of water draining from the piping into the steam generator. In addition, the nozzles are fed from the top of the ring header, further reducing the chances of drainage.

b. The main feedwater nozzles are designed to operate in the uncovered mode, in direct contrast to the feedwater sparger design.

c. The auxiliary feedwater system feeds directly upon the steam generator tubes via dedicated piping and nozzles rather than through the main feedwater system.

d. The main feedwater system design includes a small bypass line around the main feedwater control valve to allow regulation of low flows during start-up.

In the unlikely event that a large pipe rupture were to occur from a water hammer event, core cooling is ensured by the emergency core cooling system.

The potential for water hammer in other safety systems due to improper filling of the system has been reviewed on the Midland Plant and has been addressed in the design of those systems.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant has adequately addressed this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-2

Asymmetric Blowdown Loads on the Reactor Coolant System

The effects of asymmetric cavity pressure (ACP) on major nuclear steam supply system (NSSS) components are evaluated for the postulated loss-of-coolant accident (LOCA) events listed in FSAR Table 3.6-22.

The reactor vessel cavity is isolated from the steam generator cavities by the primary shield wall. Likewise, the two once-through steam generator (OTSG) cavities are separated by wall isolations. For these reasons, breaks in the reactor vessel cavity cause no ACP in the OTSG cavity and breaks in the OTSG cavity produce no ACP in the reactor vessel cavity. Also, breaks in one OTSG cavity produce no significant ACP in the other OTSG cavity.

The method of evaluating the effects of ACP involve simultaneous application of force and moment time histories from the ACP, along with those from jet impingement, elbow forcing functions, thrust and component internal LOCA forcing functions, to a mathematical model of the NSSS. The result is a set of time phased loadings on the NSSS for each LOCA. The peak LOCA loads at each point in the NSSS are available to the stress analyst for evaluation of the components. Alternatively, time history forces and moments may be used in the stress evaluation.

The structural analyses above address the concerns raised in NUREG-0609. The methods discussed above are given in more detail in B&W report, BAW-1621, "B&W 177-FA Owners Group Effects of Asymmetric LOCA Loadings Phase II Analysis," July 1980.

The results of the structural analyses employing these methods will be documented in the FSAR when they are completed. It is anticipated that the results will show acceptable loadings without significant changes to the system.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-5

Babcock & Wilcox Steam Generator Tube Integrity

This issue deals with the capability of the steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. The following tube damage has been identified in operating once-through steam generators (OTSG's):

1. Underdeposit corrosion combined with high-cycle, low-stress fatigue, and

2. Solid particle impingement.

From the initial OTSG tube leak in 1976 through the end of 1981, about 250 of over 275,000 tubes in service (less than 0.1%) have been plugged due to defects exceeding tube plugging criteria. In addition, the TMI-1 Steam Generators have experienced tube degradation, which is being investigated for relevance to other B&W units. At this time it is not expected that the plant conditions experienced at TMI-1 will be shown to be applicable to those other units.

The B&W OTSG employs a vertical, straight tube and shell design. The specific design features are described in Section 5.4.2 of the FSAR. The OTSG inservice inspection commitments are described in Sections 5.2.4, 5.4.2, and 6.6 of the FSAR and implemented via Section 16.3/4.4.5.1 of the technical specifications. Technical specifications 16.3/4.4.5.1 and 16.3/4.4.6.2 identify corrective actions to be taken should tube degradation or leakage occur.

Additional information on the steam generator design, operation and testing features which will improve tube integrity can be found in FSAR Appendix 3A (Regulatory Guides 1.83 and 1.121) and Responses to NRC Question numbers 110.29, 110.52, 110.55, 121.1, 121.3, 121.23, 122.6, 122.7, 122.9, and Appendix A Open Items MEB-8 and MEB(MI)-3.

Consumers Power Company is presently working with the B&W Owners' Group to identify and develop specific activities toward maintaining or improving the reliability and performance of the OTSG tubes. Several activities which are completed or in progress are highlighted below.

1. Eddy current, profilometry, fiberoptics, and deposit sampling are conducted on OTSG tubes to aid in better understanding of the tube conditions and damage mechanisms present. The results of these examinations enable priorities to be established for dealing with the various types of tube damage present.

2. Consumers Power Company will strictly monitor and control secondary side water chemistry. The water chemistry control and monitoring program is described in FSAR Section 10.3.5. In addition, B&W prepares periodic site chemistry reports for each operating plant from data provided by the

utilities. The information is circulated among member utilities operating OTSGs so that each can gain from the others' operating experiences.

3. A lane flow blocker has been designed and built to divert moisture and reduce the concentration of contaminants in the lane region. Qualification of the hardware will be performed in a 30-tube model boiler presently being built. Consumers Power Company, through EPRI, is funding a program to evaluate the effectiveness of the lane flow blockers. Also, Consumers Power Company has installed handholes which will allow installation of the blockers if the testing and qualification programs demonstrate that the blockers are acceptable and effective.

Consumers Power Company is working with the B&W OTSG Owners' Group to establish a formalized OTSG Integrity Program. The objective is more efficiently addressing tube integrity problems by working together rather than separately pursing common tube problems. The program would be operated on a continuing basis. This program will be directed at preventive and corrective action (i.e., eliminate the source of the problem) prior to the need for remedial action (i.e., plug tubes).

Accordingly, it can be concluded that the Midland Plant can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-9

Anticipated Transients Without Scram (ATWS)

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram") and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If a transient occurred which required a reactor trip and a sufficient number of control rods did not scram as required, then an "anticipated transient without scram," or ATWS, would have occurred.

Babcock & Wilcox in B&W Topical Report BAW-10099, "Babcock & Wilcox Anticipated Transients Without Scram Analysis," has analyzed ATWS events for a generic B&W plant and contends that the shutdown system reliability is high enough that design modifications are not necessary to satisfy ATWS safety goals. However, in order to further reduce the likelihood of an ATWS event, the Midland Plant has incorporated an Anticipatory Reactor Trip System (ARTS). This system is discussed in Section 7.2.2 of the FSAR. ARTS provides a redundant safety grade scram actuation signal on high reactor coolant system pressure or loss of control oil pressure of both main feedwater pump turbines. ARTS is independent and diverse from any other scram actuation signal in the reactor protection system.

Consistent with FSAR Section 15.8 and pending resolution and implementation of the NRC proposed rulemaking on Anticipated Transients Without Scram, Consumers Power Company will:

a. Develop emergency procedures which contain guidance to alert operators to ATWS events, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety valve indicators, and any other alarms annunciated in the control room, with emphasis on alarms not processed through the electrical portion of the reactor scram system.

b. Train operators to take actions in the event of an ATWS, including consideration of manually scramming the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to ensure delivery to the full capacity of this system, and initiation of turbine trip. The operators will also be trained to initiate boration by actuating safety-injection systems to bring the facility to a safe-shutdown condition.

Consumers Power Company will continue to remain informed on issues related to NRC rulemaking on USI A-9.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-11

Reactor Vessel Materials Toughness

10 CFR 50, Appendix G and draft NUREG-0744, September 1981 (Page 6-1, Second Paragraph) states that the reactor pressure vessel is acceptable, without application of special materials evaluation requirements, if the end-of-life (EOL) upper shelf energy (USE) remains greater than or equal to 50 foot-pounds. All of the Midland Unit 2 beltline materials and all but one of the Midland Unit 1 beltline materials meet that requirement. Current analyses show that one circumferential weld on Unit 1 may not meet the 50 foot-pound end-of-life requirement. For this Unit 1 weld, based on prediction methods documented in Report BAW-1511P, the 50 foot-pound criterion will be met for at least 15.1 effective full power years (EFPY) at the 1/4T location. The 15.1 EFPY would be equivalent to 18.9 calendar years if an 80% utilization factor were applied.

Consumers Power Company is participating in the B&W Reactor Vessel Owners Group activities as described in BAW-1543 which will provide data to allow accurate predictions of irradiated material performance for the full 40-year life of the plant. This program will provide experimental material toughness J-R curves to meet the 10 CFR 50, Appendix G, requirement that alternative analyses be performed to demonstrate equivalent margins of safety. This would also satisfy the recommendations of draft NUREG-0744 that J-T curves be established for materials in the reactor vessel. Such data will be available well before the 50 foot-pound level is reached.

Consumers Power Company expects to meet the draft NUREG-0744 recommendation that an Elastic Plastic Fracture Mechanics (EPFM) analysis be performed, if the EOL USE is equal to or less than 50 foot-pounds, well before the 50 foot-pound level is reached. Such an analysis is expected to support the continued safe operation of Unit 1. In the unlikely event that the EPFM analysis does not support the continued operation of Unit 1, Consumers Power Company would consider other actions such as performing an in-place thermal annealing of the weld in question consistent with the description of FSAR Section 5.3.3.8. The thermal annealing would satisfy the requirements of 10 CFR 50, Appendix G.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-12

Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

The NRC has raised generic questions regarding the fracture toughness and potential for lamellar tearing of steam generator or reactor coolant pump support materials. A postulated support failure could conceivably impair the effectiveness of systems designed to mitigate the consequences of an accident.

Support failure from inadequate fracture toughness is not expected to occur except under the unlikely combination of:

a. The occurrence of an initiating event (eg, a large pipe break) which has been determined to be of low probability (normal operating stresses on piping are very low);

b. The existence of nonredundant and critical support structural member(s) with low fracture toughness;

c. The existence of support structural members at operating temperatures low enough that the fracture toughness of the support material is reduced to a level at which brittle failure could occur if a large flaw existed; and

d. The existence of a flaw of such size that the stresses imparted during the initiating event could cause the flaw to rapidly propagate resulting in brittle failure of the member(s).

An extensive literature search (documented in the "For Comment" version of NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports") conducted by Sandia National Laboratory revealed that no documentation exists describing inservice failures resulting from lamellar tearing.

Consumers Power Company is participating in the Atomic Industrial Forum (AIF) Subcommittee on Material Requirements in concert with the Metals Property Council to develop an industry response to Unresolved Safety Issue (USI) A-12 (NUREG-0577) and will continue to remain informed regarding developments related to USI A-12. In response to a request by the AIF Subcommittee on Materials Requirements, Consumers Power Company has initiated a study to identify the material specifications and obtain copies of material test reports for steam generator and reactor coolant pump support materials.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic safety issue and can be operated before the ultimate resolution of this issue without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-17

Systems Interactions in Nuclear Power Plants

In mid-1977, Task A-17, was initiated to confirm that present procedures take into account the potential for undesirable interactions between and among systems. The task was divided into Phase I and Phase II by the NRC.

Phase I was structured to identify areas where interactions are possible between and among systems and have the potential of negating or seriously degrading the performance of safety functions. Phase I was to identify where NRC review procedures may not have properly accounted for these interactions.

Phase II is a follow-up phase structured to take specific corrective measures in areas where the Phase I shows a need. In a status summary of Task A-17 (NUREG-0606, November 16, 1981), it is stated that Phase II will not be pursued as an unresolved safety issue. The work originally planned under Task A-17 will now be performed under TMI Action Plan Item II.C.3, Systems Interaction.

A NRC staff summary letter report from Thomas E Murley, Director, Division of Safety Technology, on the approach to systems interactions in light water reactors was issued on June 25, 1981. The report summarizes the present staff thinking on the approach to be taken by a systems interaction program. This summary divides the types of systems interactions into two groups. One group is for internally caused systems interactions and the second group is for externally caused (spatial) systems interactions. This summary further states that a combination of probabilistic risk assessment (PRA), fault trees, failure modes and effects analysis, site walkdowns, etc, can be applied to identify systems interactions.

As a result of existing programs, the Midland Plant already incorporates many of the ideas brought out in the forementioned staff summary. Some of these programs have always been a part of the design and construction of the Midland Plant and others are the result of specific NRC staff requests and NRC regulations.

Normal design practice for the Midland Plant includes design reviews which assure compatibility of design and equipment prior to design approval. Design controls were established to control the design interfaces between the various engineering disciplines and to control design changes and modifications.

The Midland Plant has conducted several failure modes and effects analyses (FMEAs), which are documented in the FSAR, for various safety systems. In addition, Consumers Power Company is addressing potential control systems interactions on the Midland Plant as presented in response to USI A-47. The risk assessment effort currently in progress for the Midland Plant analyzes selected systems using a fault tree/block diagram methodology. This effort will identify functional interactions that may exist such as intersystem dependencies, human interactions from testing and maintenance and generic common cause agents.

Site walkdowns are also scheduled for the Midland Plant. A fire protection walkdown, with NRC presence, will be utilized to verify that Midland meets the NRC fire protection criteria. Other walkdowns are included for high energy line breaks, flooding, proximity, missiles and seismic Class II over Class I. These walkdowns will complement the design controls and reviews utilized for the Midland Plant to prevent installation of components and commodities in a manner which could result in significant adverse systems interactions.

Based upon the measures described above, including sound design, procurement and construction practices coupled with a comprehensive preoperational and start-up testing program, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-24

Environmental Qualification of Safety-Related Electrical Equipment

In addition to the conservative design, construction and operating practices and quality assurance measures required for nuclear power plants, safety systems are installed at nuclear plants to mitigate the consequences of postulated accidents. Certain of these postulated accidents could create severe environmental conditions inside the containment. The most serious of these accidents would be a high energy pipe break in the reactor coolant system piping or in a main steam line. Postulated accidents would cause certain Class 1E equipment to be exposed to environmental conditions which the equipment would not see during normal operation including high temperature, high humidity (including steam), high pressure, chemical spray, radiation and submergence.

In order to ensure that Class 1E electrical equipment will perform its function under these accident conditions, Consumers Power Company has undertaken an extensive environmental qualification program as discussed in Section 3.11 and a two-volume Environmental Qualification Supplement of the FSAR. Using NUREG-0588, "Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," and IEEE 323, "IEEE Standards for Qualifying Class 1E Equipment for Nuclear Generating Stations," as guidelines for assessing the qualification of electrical equipment, the Midland Plant will be assured a high level of equipment reliability for safe operation and accident mitigation.

Based on the Midland Construction Permit Safety Evaluation Report published in 1972, the Midland Plant is a Category II plant in accordance with NUREG-0588. In general, IEEE 323-1971 and Category II of NUREG-0588 are utilized for qualifying Class 1E electrical equipment for purchase orders executed before November 15, 1974 and wherever feasible, IEEE 323-1974 (as endorsed by Regulatory Guide 1.89) and Category I of NUREG-0588 are utilized for qualifying Class 1E electrical equipment for purchase orders executed after November 15, 1974. In addition, IEEE 334-1971 (as endorsed by Regulatory Guide 1.40) is used for qualifying Class 1E motors inside containment, IEEE 382-1972 (as endorsed by Regulatory Guide 1.73) is used for qualifying Class 1E valve operators inside containment and is also used for qualifying valve operators outside containment. IEEE 317-1972 (as endorsed by Regulatory Guide 1.63 dated 10/73) is used for qualifying electrical penetrations. The extent of compliance to the above Regulatory Guides is discussed in Appendix 3A of the FSAR.

Qualification will be accomplished in several ways including: type testing, operating experience, analysis, ongoing testing or combinations of these methods. Equipment specifications include items such as: the equipment's Class 1E requirements, performance characteristics, electrical characteristics and environmental conditions. As indicated in FSAR Section 3.11, the qualification documentation will verify that all Class 1E equipment is qualified for its application and meet its specified performance requirements.

In the case where qualification to the requirements of IEEE 323-1974 is to be accomplished by type testing, the equipment is aged to a condition equivalent to the end-of-life condition. The aging program addresses temperature, radiation and operating cycles. Electromechanical equipment such as relays and circuit breakers are operated to simulate the expected wear and electrical contact degradation in the absence of previous test results. For qualification of electronic components, reliability data and burn-in techniques are utilized.

Where the qualification yields an equipment qualified life that is less than the anticipated installed life of the equipment, qualification restrictions will be incorporated into the preventative maintenance program.

Environmental qualification for Class 1E electrical equipment which is not exposed to an accident environment will be accomplished by verifying that substantiating documentation exists to ensure that the equipment is capable of performing its safety function, throughout its design life, under all specified environmental conditions.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-26

Reactor Vessel Pressure Transient Protection

The Midland Units 1 and 2 are designed to prevent the reactor vessel pressure from exceeding the pressure restriction imposed by the technical specifications in accordance with 10 CFR 50, Appendix G. This pressure restriction ensures that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel materials.

The reactor coolant systems (RCS) always operate with a steam or gas space in the pressurizer; no operations involve a "solid water" condition, other than system hydrotest. The nonsolid condition provides dampening of any potential pressure increase incidents and slows the rate of pressure increase so that time exists for operator action. In addition to this inherent protection, an overpressure protection system protects the reactor vessel by preventing RCS pressure from exceeding the pressure limit during increasing pressure transients or incidents. The system utilizes various combinations of pressurizer safety valves, the pressurizer power-operated relief valve (PORV), the decay heat removal (DHR) system drop line relief valve and operator action to terminate the transient. The combination used depends on RCS temperature and the combination always provides single failure proof protection.

FSAR Section 5.2.2.11 provides a detailed discussion of the design and means to protect the Midland reactor vessel from overpressurization. For RCS temperatures above 330°F, overpressure protection is provided by the two pressurizer code safety valves. For RCS temperatures between 280° and 330°F, redundant overpressure protection is provided by the pressurizer PORV and operator action to terminate the transient. When the RCS temperature is below 280°F, redundant overpressure protection is provided by the decay heat removal drop line relief valve and the PORV. The design of this system addresses the concerns raised in NUREG-0224.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant has adequately addressed this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-31

Residual Heat Removal Requirements

The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident (LOCA) has been typically interpreted as achieving a "hot-standby" condition (ie, the reactor is shutdown, but system temperature and pressure are still at or near normal operating values). Considerable emphasis has been placed on the hot-standby condition of a power plant in the event of an accident or abnormal occurrence. A similar emphasis has been placed on long-term cooling, which is typically achieved by the decay heat removal (DHR) system. The DHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot-standby condition values.

Even though it is considered safe to maintain a reactor in hot-standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason, the ability to remove heat from the reactor after a shutdown is an important safety function for both PWRs and BWRs. Although it is not a design basis, the present Midland design incorporates the ability to be taken to the cold shutdown condition using only safety-grade equipment assuming only onsite or offsite power is available and considering a single failure. The systems used to achieve cold shutdown under these conditions are safety-grade and redundant. Instrumentation and control systems which may be used to achieve cold shutdown are described in Section 7.4.1 of the FSAR.

The decay heat removal system at the Midland Plant is designed to remove decay heat and sensible heat from the reactor coolant system and core during the latter stages of cooldown as discussed in Section 5.4.7 of the FSAR. The system is also used to maintain the reactor coolant temperature during refueling. In the event of a loss-of-coolant accident, the decay heat removal pumps are used for low pressure injection of borated water into the reactor vessel for emergency core cooling.

The cooldown and decay heat removal are capable of being achieved from the control room with only onsite or offsite power available and limited outside operator action. The actions which must be accomplished outside of the control room consist of opening two valves in the decay heat removal suction path, opening valves to the auxiliary pressurizer spray, and the re-establishment of electric power to the two cooler bypass valves. This amount of outside operator action is considered acceptable because at least six hours are available for its accomplishment as discussed in FSAR Section 5.4.7.

The Midland Plant cold shutdown design capability was the subject of a presentation to the Utility Design and Review Board, "Design to Achieve and Maintain Cold Shutdown." Members of the NRC participated in this meeting which was held on April 27, 1981. A comparison of the Midland design to revised BTP 5-1 as well as draft 2, revision 1 of Regulatory Guide 1.139 was included in this presentation. The transcript of this meeting and the disposition of the questions raised at this meeting were transmitted to the NRC by a letter from J W Cook to H R Denton dated June 4, 1981. This documentation, as well as the response to Question 211.35, contains a more detailed discussion of the Midland Plant cold shutdown capabilities.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant has adequately addressed this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-36

Control of Heavy Loads Near Spent Fuel

Overhead cranes are used to lift heavy objects, sometimes in the vicinity of spent fuel, in both PWRs and BWRs. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool or in the reactor core during refueling and damage the fuel, there could be a release of radioactivity. In response to this safety issue, the Midland Plant has incorporated several significant improvements in equipment and procedures used for handling heavy loads.

As presented in FSAR Section 9.1.4, the Midland 125-ton auxiliary building crane used to handle spent fuel casks is single failure proof, designed to retain structural integrity during and after a seismic event and meets the intent of Regulatory Guide 1.104 and NUREG-0554 as described in Ederer Topical Reports EDR-1(P)-A and EDR-1(NP)-A. Information concerning Regulatory Guide 1.104 conformance relating to plant specific design parameters is provided in FSAR Tables 9.1-9 and 9.1-10. The special safety features incorporated into the design of the main hoisting system of the auxiliary building crane preclude a cask drop accident by preventing a load drop in the event of a single failure in the hoisting or braking systems.

Because of lifting envelope limitations, the auxiliary building crane was not able to accommodate dual load path attaching points. In view of this limitation, a factor of safety of 10:1 has been applied to the trunion and attaching point components that have a single load path. As an added measure of safety, the auxiliary building crane incorporates mechanical stops to preclude heavy loads from passing over the spent fuel pool. These mechanical stops are designed to stop the crane while it is traveling at design speed and fully loaded. During cask handling operations, other than above the railroad bay, the trolley will be prevented from moving on the bridge through electrical interlocks. Also, the spent fuel shipping cask utilizes a double yoke lifting rig which would prevent a cask drop in the event of a single failure in the yoke.

Preoperational tests of the fuel handling equipment are outlined in Subsection 9.1.4.4.2 of the FSAR and are conducted as described in Chapter 14 of the FSAR. The test of the auxiliary building crane is described in Test Abstract 14A.1.72.

The 190-ton containment polar crane, which is used to handle the missle shield, the reactor vessel head and reactor internals during refueling operations, is designed to retain structural integrity during and after a seismic event as discussed in Sections 3.2 and 9.1.4 of the FSAR.

The NRC technical resolution of Generic Task A-36, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was transmitted to Consumers Power Company for action by the generic NRC letters dated December 22, 1980 and February 3, 1981. These documents provide guidelines for necessary actions to assure safe handling of heavy loads once the plant becomes operational.

Enclosure 2 to the December 22, 1980 letter identified a number of measures dealing with safe load paths, procedures, operator training and crane inspections, testing and maintenance.

Consumers Power Company has prepared and submitted the first part of the required two-part response in a letter from J W Cook to H R Denton (Serial 14928) dated December 21, 1981. This response to request for additional information on control of heavy loads addresses the general requirements of NUREG-0612 for the safe handling of heavy loads at Midland. The second part response will provide an extensive, detailed assessment of the Midland Plant's capability to safely handle heavy loads in the vicinity of spent fuel and plant systems required for safe shutdown and decay heat removal. The second part response will also identify design changes and/or procedural modifications, if any, required as a result of first part review and will be submitted to the NRC for review in the first quarter of 1982.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-40

Seismic Design Criteria - Short Term Program

NRC regulations require that safety related nuclear power plant structures, systems and components be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in Regulatory Guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, further reviews of the seismic design of various plants are being undertaken. Task A-40 is, in effect, a compendium of short term efforts to support such reevaluations by the NRC staff, in particular, those related to older operating plants.

The seismic design basis and seismic design of the Midland Plant are being evaluated at the OL licensing stage. The Midland Plant design response spectra are shown in Figures 3.7-1 and 3.7-2 of the FSAR for the operating basis earthquake (OBE) and safe shutdown earthquake (SSE), respectively, with the 50% increase described in Subsection 3.7.1.1. These spectra correspond to a maximum horizontal ground acceleration of 0.06g for the OBE and 0.12g for the SSE. The damping factors utilized in the Midland Plant design are those provided in Appendix 3A of the FSAR as a response to Regulatory Guide 1.61. These positions have been reviewed and found acceptable to the NRC as indicated in the NRC letter dated June 8, 1976 on the subject of Regulatory Guides in the structural engineering category.

In accordance with the criteria in effect prior to issuance of Regulatory Guide 1.92, the Midland Plant piping design combines all individual modes in one spatial coordinate by the square root of the sum of the squares. To demonstrate the the effect of Regulatory Guide 1.92 guidelines for closely spaced modes, representative systems were chosen and reanalyzed using the 10% method. The discussion of the analysis and the results are presented in Appendix 3D to the FSAR.

In addition to meeting the requirements of the design bases as stipulated in respective sections of the FSAR, Consumers Power Company has initiated a Seismic Margin Review of the Plant. The purpose of this review is to evaluate the seismic safety margins in the Seismic Category I structures and components necessary for the safe shutdown of the reactor when subjected to ground motion characterized by the Site Specific Response Spectra defined for the Midland site. The criteria for this review has been submitted to the Staff via letter from J W Cook to H R Denton dated September 25, 1981 and is currently under Staff review.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-43

Containment Emergency Sump Performance

Following a postulated loss-of-coolant accident (LOCA), the water would be collected in the emergency sump at a low point in the containment. This water would be recirculated to the reactor system by the decay heat removal pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. A postulated loss of the ability to draw water from the emergency sump could disable the decay heat removal and containment spray systems.

One postulated means of losing the ability to draw water from the emergency sump could be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release of the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be swept into the sump, potentially causing blockage.

A second postulated means of losing the ability to draw water from the emergency sump could be abnormal conditions in the sump or at the pump inlet, such as air entrainment, vortices, or excessive pressure drops. These conditions could result in pump cavitation, reduced flow, and possible damage to the pumps.

Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Systems," provides guidance on the design of ECCS sumps. To minimize the risk of sump clogging during a LOCA, thermal insulation used inside the containment consists primarily of metallic reflective insulation for the primary piping, a portion of the steam generators, and the reactor vessel, as well as encapsulated nonmetallic insulation for the pressurizer, the reactor coolant pumps, the remainder of the steam generators, and the secondary system piping and components as discussed in Section 6.2.2.1.2.4 of the FSAR.

Additional insulation includes small amounts of microtherm (encapsulated metallic) insulation on the hot and cold leg pipes from the reactor vessel to the outside surface of the biological shield. Encapsulated microtherm insulation is also used in areas where conventional insulation (encapsulated fiberglass or metallic) cannot be installed because of insufficient space.

Each insulation assembly is a rigid body jacketed in heavy gage stainless steel and designed to withstand vibration and seismic shock associated with postulated accident conditions inside containment. With the exception of local failure in the vicinity of postulated reactor coolant system pipe ruptures, insulation assemblies are expected to remain intact during and after a LOCA.

The floor level in the vicinity of the sump slopes generally downward from the sump in accordance with Regulatory Guide 1.82. There are three layers of trash screens to prevent floating debris from interfering with sump performance: an outer trash rack, a center screen with 3/8" square openings, and an inner screen of 16 mesh with 0.023" diameter wire. All screens and trash racks are vertically mounted on a 3" high concrete curb and attached to the intake structure of the sump. Sufficient screen area exists to allow 50 percent clogging of the fine inner screen without degrading spray pump or decay heat pump net positive suction head (NPSH). As detailed in Section 6.2.2.1.3 of the FSAR, analysis shows that both the spray pumps and decay heat removal pumps have an available NPSH which exceeds the required NPSH. The trash rack and screens are designed to be seismic Category I. The highest approach velocity in the vicinity of the containment sump is 0.5 fps. Thus, velocities in the containment are sufficiently low to allow settling of the high density particles. In addition, intake losses are only about 0.25 feet at the runout flow of 6000 gpm and 50 percent blockage of the screens. This head loss would have negligible impact on the available NPSH at the spray and decay heat removal pumps. A detailed comparison of the Midland design with Regulatory Guide 1.82 is given in Table 6.2-23 of the FSAR.

Midland was one of the plants selected for the survey on the in-plant use of insulation in the Burns and Roe report NUREG/CR-2403. The report notes that insulation debris on the basement floor would have to float through the five openings in the shield wall and find its way to the emergency sump around building obstacles in its path. Floor drains at various elevations in the reactor containment are connected by 4-inch piping to the emergency sump. Each drain is equipped with a basket strainer. The basket strainer is lined with 16 mesh screen with 0.023 inch diameter wire. The basket strainer passes debris no larger than that permitted through the fine mesh sump inner screen. Given the design of the sump, with downward sloping floor and the three screens as described above, it would be anticipated that little debris would reach the innermost sump screen. Any debris which does elude the trash racks and settling regions passes into the sump only through the 16 mesh screen and can be drawn into the suction piping for the reactor building spray and decay heat removal systems. Such debris is of small enough dimension to pass through any restriction to flow encountered by either system and eventually is pumped back into the containment.

Separate suction lines are located in the sump, one for each train of the decay heat pump and spray pump. Each intake is separated from the other by a divider plate in the sump. The suction piping has sufficient submergence to ensure continuous intake flow. Each intake is covered on the end by a grating cage which eliminates any potential vortex formed in the sump from entering the suction lines. To maintain pump suction lines free of entrapped air, the portion of the line downstream of the reactor building sump isolation valve will be vented and filled during initial system fill. Since the remaining section of the line, from the sump to the sump isolation valves, is run horizontally, no high points for air entrapment exist as the sump and suction pipe are filled during a LOCA.

Conformance to Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," addresses the testing of the recirculation function of ECCS. A series of tests has been performed by Western Canada Hydraulic Laboratories Ltd to evaluate the performance capability of the Midland sump design. A full scale sump model was built and tested to verify vortex control and to determine the head loss associated with the trash rack, grating cage, and inlet piping. Results of the test program were submitted in a letter dated June 26, 1980 from J W Cook to A Schwencer of the NRC. Additional information was provided in amended repsonse to Q221.189 (Q&R p 6.3-70). In the Alden Research Lab interim report on containment sump reliability studies dated June 1981, it is noted that a solid top cover plate over the sump is very effective in suppressing vortices as long as the cover plate is submerged, as in the Midland design. Further, it also pointed out that even the strongest air drawing vortices were completely suppressed by the vortex suppressor tested. The grating cages in the Midland design serve to accomplish this.

An onsite test to determine borated water storage tank suction piping losses will be performed. Onsite testing of the ECCS and reactor building spray pumps' suction piping was performed for the Davis-Besse Unit 1 plant. The results of a comparison between the predicted head loss values obtained through calculations and the actual head loss values obtained through onsite testing were conclusive and showed the calculated values were more conservative than the actual values obtained by testing. The calculation method used for Midland is identical to that used for Davis-Besse. A first-of-a-kind test for flowrate measurement was performed on Oconee, which provided the initial verification of design. Midland is similar in design to Oconee. Further details on the ECCS design to meet functional requirements following a LOCA, including NPSH calculation methodology, are given in FSAR Section 6.3.2. This methodology has been reviewed by the Staff and found to be acceptable; the Staff evaluation is given in a letter to Consumers Power Company dated September 29, 1976.

Housekeeping requirements inside containment will further reduce potential for sump clogging. Surveillance requirements provide for visual examination of sump components (trash racks, screens, pump suction inlets). An access hatch is provided for inspection and maintenance of inside structures. These inspections are conducted at the frequencies specified in FSAR Section 16.3.5.2.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant has adequately addressed this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-44

Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite ac power connection, a standby emergency diesel generator ac power supply and dc sources.

Task A-44 involves a study by the NRC to determine the extent to which nuclear power plants should be designed to accommodate a complete loss of all ac power (that is, loss of both the offsite and the emergency diesel generator ac power supplies). This issue arose because of industry operating experience regarding the reliability of ac power supplies. A number of operating plants have experienced a total loss of offsite electrical power and relied upon the standby emergency diesel generators to supply ac power. In one instance, these emergency power supplies failed to start. In addition, there have been instances where an emergency diesel generator in an operating plant failed to function during periodic surveillance tests.

The issue of station blackout was also considered by the Atomic Safety and Licensing Appeal Board (ALAB-603) for the St Lucie Unit 2 facility. In addition, in view of the completion schedule for Task A-44 (October 1982), the Appeal Board recommended that the Commission take expeditious action to ensure that other plants and their operators are equipped to accommodate a station blackout event. The Commission has reviewed this recommendation and determined that some interim measures should be taken at all facilities while Task A-44 is being conducted. The staff notified Consumers Power Company of these requirements in a letter from D Eisenhut, NRC, dated February 25, 1981. Procedures and operator training for safe operation of the facility address the loss of ac power at the Midland Plant as described in the Consumers Power Company response dated June 17, 1981.

Midland has two diverse sources of preferred offsite power as discussed in Section 8.2.1.1 of the FSAR. A loss of offsite ac power involves a loss of both these sources of offsite power. Transient stability and grid reliability studies have been conducted and are discussed in Appendix 8A. The results of these analyses show that the offsite power system can withstand major transients without system breakup or uncontrolled cascading.

If offsite ac power is lost, standby power for each safety-related load group is supplied by a diesel generator complete with its accessories and fuel storage and transfer systems. Each diesel generator is capable of supplying loads necessary to shut down and isolate the associated reactor reliably and safely. Each diesel generator is rated at 5250 kW for continuous operation and at 5775 kW for 2-hour short-time operation in any 24-hour period. Functional aspects of the diesel generator including load shedding, starting and sequencing, etc, are discussed in detail in FSAR Section 8.3. As noted in that section, in the 300 start test, there were no failures to start in 315 official starts. Details on this qualification test were submitted to R S Boyd by letter dated February 8, 1979.

Criteria for periodic testing will be given in Technical Specification 16.3/4.8. The position with regard to Regulatory Guide 1.108 is given in Appendix 3A of the FSAR which indicates Midland's compliance with the guidelines and gives additional clarification on some points.

Midland has implemented a program for enhancement of diesel generator reliability in accordance with the recommendations in NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability." All applicable recommendations in that report are addressed in the Midland design. Given the expected reliability of the diesel generator and the ongoing efforts to assure long-term reliability, station blackout has been adequately addressed in the Midland design.

In the unlikely event that the diesel generators should fail to operate, the plant could cope with the situation by other features. For example, the auxiliary feedwater system is designed with two independent full capacity systems, each with diverse motive and control power sources. In the highly unlikely event of a complete loss of ac power, the turbine-driven auxiliary feedwater pump is capable of meeting the feedwater requirements for a minimum of two hours on dc power alone, as discussed in Section 10.4.9.3 of the FSAR.

Decay heat can be removed through the main steam line relief valves with the reactor coolant system in natural circulation, supplying feedwater using the auxiliary feedwater system taking suction from the condensate storage tank. The turbine-driven pump bearings do not require cooling from an ac dependent source. Actuation and control of this train are provided from the vital dc power source.

Restoration of offsite ac power in the event of a loss of the grid would be by the Consumers Power Company area power controller and area operators. Once power is restored to the site, procedures direct the activities required to return to normal operation.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-45

Shutdown Decay Heat Removal Requirements

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity via a turbine generator. Following a reactor shutdown, a reactor produces insufficient power to operate the turbine; however, the radioactive decay of fission products continues to produce heat (so-called "decay heat"). Therefore, when reactor shutdown occurs, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressures do not develop which could jeopardize the reactor and the reactor coolant system. It is evident, therefore, that all light water reactors (LWRs) share two common decay heat removal functional requirements: (1) to provide a means of removing decay heat from the reactor coolant system and (2) to maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The reliability of a particular power plant to perform these functions depends on the frequency of initiating events that require or jeopardize decay removal and the probability that required systems will remove the decay heat.

In response to an ACRS request, the NRC Staff will conduct a study which will evaluate the benefit of providing alternate means of decay heat removal which could substantially increase the plants' capability to handle a broader spectrum of transients and accidents. The study will consist of a generic system evaluation and will result in recommendations regarding the desirability of and possible design requirements for improvements in existing systems or an alternative decay heat removal method if the improvements or alternative can significantly reduce the overall risk to the public.

The primary method for removal of decay heat from the Midland reactors is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feedwater or auxiliary feedwater systems, and it is rejected to either the turbine condenser or the atmosphere via the steamline atmospheric dump valves. Following the TMI-2 accident, the importance of the auxiliary feedwater system was highlighted and a number of steps were taken to improve the reliability of the auxiliary feedwater system. As stated in FSAR Section 10.4.9.3, the Midland auxiliary feedwater system provides redundant and diverse means of supplying feedwater to the steam generators for cooling the reactor coolant system under emergency conditions. Either pump has the capability of supplying 100 percent of the feedwater requirements for safe cooldown of the reactor coolant system. The system can perform its safety-related function assuming any single active component failure coincident with loss of offsite power. Complete physical and electrical separation is maintained throughout the pump controls, control signals, electrical power supplies and instrumentation for each auxiliary feedwater pump. Assuming a temporary loss of all offsite, normal onsite and emergency onsite ac power (station blackout), the Midland auxiliary feedwater system is capable of performing its safety function for at least 2 hours. The steam turbine driven auxiliary feedwater pump provides the required feedwater to both steam generators during station blackout and is dependent only on vital dc power. A detailed comparison of the Midland auxiliary feedwater system with the NRC acceptance criteria in Standard Review Plan 10.4.9 and the recommendations in NUREG-C611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," is provided in Appendix 10A of the FSAR. Additionally, a detailed reliability analysis of the system was performed by Pickard, Lowe, and Garrick, Inc., and forwarded to the NRC on February 23, 1981.

At low primary system pressure (below about 300 psi), the long-term decay heat is removed by the decay heat removal system to achieve cold shutdown conditions. A more detailed description of the Midland decay heat removal system is presented in FSAR Section 5.4.7 and the response to Unresolved Safety Issue A-31. Suitable procedures will be available for bringing the plant from normal oeprating power to cold shutdown for both the forced circulation and the natural circulation conditions.

Additionally, the Midland reactors have alternate means of removing decay heat if an extended loss of feedwater is postulated. This method is known as "feed and bleed" and uses the high pressure injection system to add water coolant (feed) at high pressure to the primary system. The decay heat increases the system pressure and energy is removed through the power-operated relief valves and/or the safety valves (bleed), if necessary.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-46

Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably among plants licensed in different time frames. The NRC Staff has determined that the seismic qualification of the equipment in operating plants should be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The NRC's objective of this "Unresolved Safety Issue" is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria. This guidance will concern equipment required to safely shut down the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

Consumers Power Company has instituted an extensive seismic qualification program for the Midland Plant. For electrical equipment and instrumentation, the program is discussed in Section 3.10 of the FSAR. In general, balance of plant equipment purchased prior to July 1, 1975 has been qualified to IEEE 344-1971 as supplemented by Position B.1 of NRC Branch Technical Position EICSB-10. Equipment purchased after July 1, 1975 is qualified to IEEE 344-1975 as supplemented by Regulatory Guide 1.100 (refer to FSAR Appendix 3A for a discussion of Regulatory Guide 1.100). NSSS equipment has been qualified to IEEE 344-1971 as supplemented by B&W Topical Report BAW-10003 for the reactor protection and ECC systems. Identification of which version of IEEE 344, the methods used for qualification, and test results of each category of safety-related equipment is discussed in FSAR Section 3.10.4.

Seismic qualification of mechanical equipment is discussed in FSAR Section 3.9.2.2. The criteria and methods used for qualifying mechanical equipment meet the intent of Standard Review Plant 3.9.2 and, where applicable, Regulatory Guide 1.48 as discussed in Appendix 3A of the FSAR. A summary of individual component criteria and methods of qualification is provided in Table 3.9-17 of the FSAR.

Consumers Power Company is presently re-evaluating the seismic qualification of all equipment due to revised response spectra and in the process is generating the Seismic Qualification Review Team (SQRT) information required by the NRC. This information will be submitted at a later date in a separate submittal in accordance with the NRC's SQRT procedures. A site visit will be made by the NRC to audit Consumers Power Company's seismic qualification program.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-47

Safety Implications of Control Systems

This issue addresses the NRC concern that the potential may exist for accidents or transients to be mc : severe than previously estimated as a result of control system failures or malfunctions.

The control systems for Midland Plant Units 1 and 2 have been designed and built with a high degree of reliability. These non-safety grade control systems have been the subject of four evaluations to date to identify adverse interactions which might impact the safety analysis for the plant.

1. FMEA of ICS

   In April 1979, the NRC Commissioners requested that failure modes and effects analysis (FEMA) of the Integrated Control System (ICS) be performed by B&W. This analysis, which was completed in August of 1979 provided recommendations for improving the reliability of the ICS. Based on this analysis, several improvements to the Nonnuclear Instrumentation System (NNI) and ICS were incorporated into the Midland design. These improvements are documented in the FSAR Response to TMI Issues, NUREG-0667, Recommendation 5.

2. IE Bulletin 79-27

   Consumers Power Company has reviewed the Class 1E and Non-class 1E buses supplying power to safety- and non-safety-related instrumentation and control systems which could affect the ability to achieve a cold shutdown condition. This review has been completed and is available in the FSAR, Response to TMI Issues, NUREG-0667, Recommendation 5, Item h, where it is concluded that no design modifications or additional administrative controls are necessary.

3. IE Information Notice 79-22

   Consumers Power Company has performed an evaluation of potentially adverse environmental effects on those non-safety grade control systems that could have a possible effect on the Midland Plant safety analysis. This is a similiar evaluation to the evaluation performed by B&W operating plants in 1979 and reflects that many control systems have been upgraded to safety grade in the Midland Plant. This evaluation, forwarded to the NRC on February 15, 1982, stated that no adverse impact on the plant safety analysis report was identified.

4. Control Systems Failure Analysis

   The NRC staff during the November 17-20, 1981 meeting to review Chapter 7 of the FSAR requested a failure evaluation which will address potential non-safety grade control systems interactions at the Midland Plant. This evaluation, which involves the ICS, evaporator steam demand development system and the NNI, is scheduled for completion in June 1982. This evaluation will consider loss of single sensor input, breakage of instrument lines having more than one instrument with at least one input into the above systems, failure of individual fuses or breakers and complete loss of power to these systems.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant is adequately addressing this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-48

Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Following a loss-of-coolant accident (LOCA) in a light water reactor plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Due to the potential for significant hydrogen generation, 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light Water Cooled Power Reactors," and GDC 41, "Containment Atmosphere Cleanup," require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident. The purpose of the requirements is to ensure that containment integrity is maintained and that essential equipment required for safe shutdown of the reactor is able to survive the adverse environment created by a postulated accident.

The design basis for combustible gas control in the containment at Midland is presented in Section 6.2.5 of the FSAR. The Midland design meets all current requirements. The hydrogen control system is designed to ensure that the hydrogen concentration within the Midland containment is maintained below the lower combustible limit of 4.0 volume percent as specified in Regulatory Guide 1.7. The system includes redundant safety grade hydrogen recombiners located inside containment, a safety-related hydrogen monitoring subsystem and a backup hydrogen purge subsystem. Hydrogen mixing is provided by the containment spray system, the recirculating air cooling units and the containment internal design which permits convective mixing and prevents entrapment of hydrogen.

Preliminary calculations have been performed to provide a reasonable estimate of the maximum pressure and temperature resulting from a substantial release of hydrogen into the post-LOCA containment atmosphere. The postulated hydrogen transient involves a complete reaction of 75 percent of the core zirconium which starts 5 minutes post-LOCA. The LOCA assumed is initiated by a 4.27 square foot reactor coolant pump suction break (FSAR Figure 6.2-8). The maximum temperature and pressure are 650°F and 108 psia, respectively. By the end of the transient, containment spray has reduced the temperature and pressure to 450°F and 104 psia, respectively. Although credit was taken for mixing due to containment spray, no credit was taken for mixing due to fan actuation. Even for the worst possible case, when containment spray is terminated at 5 minutes post-LOCA, the containment pressure is less than twice the containment design pressure (the Midland containment has 1.67×10^6 cubic feet of net-free volume and a design pressure of 70 psig).

SECY letter 80-107, "Proposed Interim Hydrogen Control Requirements for Small Containments" dated February 22, 1980, Section 3.5, concludes that the effects of assumed hydrogen burns are not expected to exceed the values used in the existing equipment qualification tests for LOCA conditions. In addition, essentially all required Three Mile Island Unit 2 systems and components have continued to sucessfully function following a containment hydrogen burn. Therefore, it is judged that potential hydrogen burns do not constitute a threat to safety and that essential equipment will survive the adverse environment created by the postulated accident.

Based upon the measures described above, Consumers Power Company is confident that the Midland Plant has adequately addressed this generic issue and can be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-49

Pressurized Thermal Shock

This issue addresses the possibility of brittle fracture of the reactor vessel during either a Small Break LOCA transient or an overcooling transient resulting in a cooldown of the reactor vessel metal, followed by repressurization of the pressure boundary above a critical level during the cooling period. The probability of vessel brittle fracture during an overcooling/repressurization transient depends upon its material properties and the time history of the thermal and pressure transients to which it may be subjected.

To address this concern, Consumers Power Company, in concert with the B&W Owners Group, participated in generic evaluations to address the Small Break LOCA in 1979 and 1980. A highly conservative, bounding analysis was performed to cover all of the 177 FA B&W plants and the results reported in BAW-1628.

Results of thermal mechanical analyses are highly weld-property and irradiation dependent. Even with the highly conservative bounding analysis used in BAW-1628, the analyses show acceptable results for all operating B&W reactor vessel welds through several effective full power years (EFPY) of operation. This analysis conservatively bounds the Midland vessels.

The Staff's draft Task Action Plan states, "The vessels of concern are those which...and which are made of material that has a high sensitivity to neutron irradiation (such as those made with welds of high copper content)." This provides a significant additional safety margin for Midland Unit 2 which can be excluded from the "vessels of concern" because of its weld properties. Additional conservatism exists in BAW-1628 for the Midland vessels, since the limiting case welds were longitudinal welds which are not used in the Midland vessels. Therefore, thermal shock is not a concern with respect to startup and the initial years of operation of the Midland reactors.

It is expected that Midland specific thermal shock analyses to be performed in the future will benefit from additional knowledge gained from programs currently in progress. As examples, significant benefits are expected from EPRI sponsored thermal mixing studies, from the continuing Reactor Vessel Surveillance Program (BAW-1543), and from NRC activities associated with the A-49 issue (especially in the definition of transients to be analyzed). These new data and other modeling improvements will be used to demonstrate an expected extension of calculated vessel integrity by additional EFPYs as they become available. Accordingly, it can be concluded that the Midland Plant can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.