Consumers Power Company

David J VandeWalle
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July 9, 1982

Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20
PALISADES PLANT - CORRECTION TO JUNE 29, 1982 LETTER CONCERNING
UNRESOLVED SAFETY ISSUES

The intent of this letter is to rectify Consumers Power Company June 29, 1982 correspondence, entitled "Response to Request for Information Concerning Unresolved Safety Issues", by providing the NRC with a complete response to the staff's May 3, 1982 letter. The May 3, 1982 NRC letter requested Consumers Power Company to provide up-to-date information concerning the status of nineteen (19) USI's applicable to the Palisades Plant. This information will be incorporated by the staff into a SER which is intended to evaluate the Palisades application for conversion of the Provisional Operating License to a Full-Term Operating License.

It was discovered in the Consumers Power Company's June 29, 1982 letter that nine (9) of the nineteen (19) USI's were inadvertently omitted from that submittal. We sincerely apologize for any inconvenience that this error might have caused the staff.

The enclosure to this letter furnishes Consumers Power Company response to all of the applicable USI's. Also provided for the purpose of completeness is a chart illustrating the USI's applicable to Palisades along with an indication of those missing from the June 29, 1982 letter.

Your understanding in this matter is very much appreciated.

David J VandeWalle

David J VandeWalle
Nuclear Licensing Administrator

CC Director, Region III, USNRC
NRC Resident Inspector - Palisades

A001

Attachments

CONSUMERS POWER COMPANY

Palisades Plant

Docket 50-255
License DPR-20

CORRECTION TO JUNE 29, 1982 LETTER CONCERNING

UNRESOLVED SAFETY ISSUES

At the request of the Commission and pursuant to the Atomic Energy Act of 1954, and the Energy Reorganization Act of 1974, as amended, and the Commission's Rules and Regulations thereunder, Consumers Power Company submits our response to NRC letter dated May 3, 1982, entitled "Unresolved Safety Issues Status for Palisades Plant". Consumers Power Company response is dated July 9, 1982.

CONSUMERS POWER COMPANY

BY R B DeWitt 7/9

R B DeWitt, Vice President
Nuclear Operations

Sworn and subscribed to me this 9th day of July 1982.

Dorothy H. Bartkus

Dorothy H Bartkus, Notary Public
Jackson County, Michigan

My commission expires March 26, 1983.

UNRESOLVED SAFETY ISSUES APPLICABLE TO PALISADES

		TASK #	TITLE
	(1)	A-1	Water Hammer
*	(2)	A-2	Asymmetric Blowdown Loads on the Reactor Coolant Syst
	(3)	A-4	Pressurized Water Reactor Steam Generator Tube Integrity
	(4)	A-9	Anticipated Transients Without Scram (ATWS)
*	(5)	A-11	Reactor Vessel Materials Toughness
*	(6)	A-12	Fracture Toughness of Steam Generators and Reactor Coolant Pump Supports
*	(7)	A-17	Systems Interactions in Nuclear Power Plants
*	(8)	A-24	Environmental Qualification of Safety-Related Electrical Equipment (EEQ)
	(9)	A-26	Reactor Vessel Pressure Transient Protection
	(10)	A-31	Residual Heat Removal Requirements
*	(11)	A-36	Control of Heavy Loads Near Spent Fuel
*	(12)	A-40	Seismic Design Criteria Short-Term Program
	(13)	A-43	Containment Emergency Sump Reliability
	(14)	A-44	Station Blackout
	(15)	A-45	Shutdown Decay Heat Removal Requirements
*	(16)	A-46	Seismic Qualification of Equipment in Operating Plants
	(17)	A-47	Safety Implications of Control Systems
	(18)	A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment
*	(19)	A-49	Pressurized Thermal Shock

* Denotes response to applicable USI's missing from Consumers Power Company June 29, 1982 submittal.

UNRESOLVED SAFETY ISSUE A-1
Water Hammer

Since 1971, there have been over 200 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feed rings and piping, the RHR system, ECCS and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump start-up with partially empty lines and rapid valve motion. It should be made clear that none of these events has resulted in release of radioactivity to the environment. Thus, the primary safety concern relates to the very low probability that a water-hammer event could result in a failure of the reactor coolant system boundary or a disabled safety system. An NRC assessment of water-hammer related events through mid-1981 disclosed that about one-half of these causes were operational in nature, or plant procedure oriented, and the remainder were induced by plant system design.

The Palisades Plant reported a water-hammer event in 1974 which resulted from inadvertently pumping water into a partially voided safety injection line while testing and filling the sodium hydroxide system. The damage reported was failure of the anchor bolt mounting at the pump suction line restraint. Operating procedures were reviewed and revised to eliminate introduction of air into the safety injection lines. No further water-hammer events have been reported for the Palisades Plant.

With regard to the water hammer potential associated with the steam generator auxiliary feedwater, Consumers Power Company proposed modifications to the auxiliary feedwater system by submittal dated December 1, 1980. The proposed modification consisted of disconnecting the auxiliary feedwater lines from the main feedwater lines, routing seismic Category I lines through existing spare containment penetration to existing auxiliary feedwater nozzles on the steam generators and providing a separate auxiliary feedwater sparger inside each steam generator. These modifications were completed in December 1981 during the last refueling outage. The NRC found the implementation of these modifications acceptable.

Subsequent to the completion of these modifications, auxiliary system flow tests were performed at 200 psia with 200, 300 and 400 gpm flow. Despite the fact that these tests were performed with the water level below the sparger and during water level recovery of the sparger, no water hammer occurred. Operating history at Palisades coupled with the installation of the above described modifications give adequate assurance that the probability of water hammer is very low. However, in the remote event that a large pipe break did result from a severe water hammer, adequate core cooling is ensured by redundant steam generator and auxiliary feedwater system.

On the basis of the foregoing, Consumers Power Company is confident that the Palisades Plant has adequately addressed this generic issue and can continue operation without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-2
Asymmetric Blowdown Loads on the Reactor Coolant System

This generic safety issue pertains to the capability of reactor vessel supports and other affected structures (eg, fuel assemblies) to withstand asymmetric loss-of-coolant (LOCA) loads or safe shutdown earthquake loads. The effects of these asymmetric loads are evaluated against applicable industry standards and NRC criteria to determine the adequacy of the reactor vessel structural integrity.

Consumers Power Company was an active participant in the Combustion Engineering Owners' Group (CEOG) for the resolution of the effects of asymmetric LOCA loads for the Palisades Plant. We believe that all but one issue has been satisfactorily answered by the CEOG Final Report submitted July 1, 1980 and updated by the CEOG submittal made on July 31, 1981. The July 31, 1981 submittal was made in response to NRC questions (Mr Dennis M Crutchfield's letter dated February 23, 1981 to Mr Arthur E Lundwall, Jr, Baltimore Gas & Electric). The structural integrity of the Palisades reactor vessel was verified via the Bechtel analysis as summarized in Volume 2, Section P of the July 31, 1981 submittal. The one issue that has not been fully resolved for Palisades is qualification of the current fuel assembly design.

Consumers Power Company is currently preparing revised mechanical design specifications for Palisades fuel. These specifications will contain adequate requirements to ensure that the new fuel design will be capable of withstanding the loads calculated using the results of the hydraulic analyses performed by Combustion Engineering for the Palisades Plant. It is expected that vendor selection, fuel design, fabrication and delivery of the first reload of the new fuel design will be completed by mid-1986. Due to design lead time and possible hydraulic testing associated with spacer redesign, this is as reasonably as changes can prudently and cost-effectively be implemented.

In view of the acceptable analysis results previously submitted to the NRC by the CE Owners' Group, the extremely low probability of a large break LOCA at a reactor vessel nozzle and the fact that fuel procured in future contracts will have demonstrated adequate strength to withstand the loads generated during postulated LOCA or seismic events, we have concluded that Consumers Power Company is adequately addressing this generic issue and that the Palisades Plant can continue to be safely operated with the current fuel design without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-4
Steam Generator Tube Integrity

This issue deals with the capability of the steam generator (S/G) tubes to maintain their integrity during normal operation and postulated accident conditions. The following tube damage has been identified in operating recirculating steam generators (RSG):

1. Underdeposit corrosion resulting in tube wall thinning (wastage)
2. Denting at support plate intersections
3. Intergranular attack
4. Stress-corrosion cracking

Corrosion resulting in steam generator tube wall thinning (wastage) identified in several Combustion Engineering and Westinghouse units for a number of years has been essentially eliminated by major changes in secondary side water chemistry treatment (changing from phosphate to all-volatile treatment). The corrosion-related phenomenon known as denting, results from a buildup of support plate corrosion products in the annulus between the tubes and support plates. This eventually causes diametral reduction in tubes and deformation of the tube support plates. This has led to other problems, such as stress corrosion cracking, leaks at the tube support plate intersections and U-bend section cracking of highly stressed tubes.

The number of tubes plugged per unit varies from less than 1% to more than 25%. Replacement S/G have even been installed at several plants and are being prepared for others including Palisades.

The CE RSG utilizes an inverted "U" tube and shell design and, hence, recirculates secondary water through the shell prior to exit. Specific design features are described in the Palisades Plant FSAR. Various measures to maintain steam generator integrity are and have been in place at Palisades. These include Technical Specifications for limits and surveillance of secondary water chemistry, primary to secondary leakage limits and inservice inspection surveillance and repair requirements, and close monitoring for condenser tube leaks. Tube plugging, tube sleeving, increased and more frequent inspections have been and are routinely implemented at Palisades to assure steam generator tube integrity. The result of these activities has been a remarkable reduction in the rate of tube degradation. From 1972 to 1975, nearly 18% of the tubes in each steam generator were plugged or sleeved because of degradation. Since the 1974-75 conversion to all volatile chemistry, only an additional 4% (approximately) of the tubes have required repairs.

Consumers Power Company is working with the EPRI S/G Owners' Group to continue a formalized S/G Integrity Program. The objective is to more efficiently address tube integrity problems by working together rather than separately pursuing common tube problems. The program is operated on a continuing basis and is directed at preventive and corrective action (ie, eliminate the source of the problem) prior to the need for remedial action (ie, plug tubes). Several activities which are completed or in progress are highlighted below.

1. Eddy current, profilometry, fiberoptics and deposit sampling have been conducted on RSG 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 degradation encountered.

2. Consumers Power Company strictly monitors and controls secondary side water chemistry. The EPRI S/G Owners' Group acts as a medium of information exchange regarding operating experience and related chemistry control.

Accordingly, it can be concluded that Consumers Power Company is adequately addressing this issue and that the Palisades Plant can continue to be operated before the ultimate resolution of this generic issue without undue risk to 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 temporarily abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, less frequently occurring, 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 ATWS would occur. A malfunction of the RPS which precludes adequate insertion of negative reactivity when a reactor trip is needed, following an abnormal event, is an "anticipated transients without scram" (ATWS).

The Reactor Protective System at Palisades has multiple channels, diverse inputs and has proved to be highly accurate and reliable. The probability of an ATWS event, therefore, is highly unlikely. In addition, generic analyses of ATWS events performed by Combustion Engineering for its NSSSs (CE Reports CENPD-263-P and CEN-134-P) concluded that:

a. The most severe radiological release from any CE NSSS during any ATWS event is within 10 CFR 100 limits;

b. For all plant classes, service level C stress limits are met for most components. Where service level C limits were exceeded, the stresses were well below level D limits. Analysis determined that necessary piping and valves would remain functional so that the plants could be brought to safe shutdown conditions;

c. Long-term coolable geometry of fuel rods in any CE NSSS would be maintained following any ATWS; and

d. Containment integrity would be maintained since maximum pressure from any ATWS was calculated to be only about one-third containment design pressure.

Consumers Power Company has monitored the progress of this USI for some time and has participated in the efforts of AIF and the ATWS Utility Group (coordinated by KMC, Inc) to resolve this generic safety issue. Three separate proposed rules on ATWS prevention and/or mitigation, including one developed by the ATWS Utility Group, are currently being evaluated by the NRC. Although we do not believe that plant modifications will be necessary to mitigate the highly improbable ATWS events postulated, they may be considered in response to the final NRC rule or implementation position.

In addition to the above activities, Consumers Power Company has also participated in the Combustion Engineering Owners Group efforts to develop and implement Emergency Procedure Guidelines (EPGs) for numerous events, including ATWS. The ATWS EPG specifically addresses control of reactivity, PCS pressure, PCS inventory, PCS heat removal and containment considerations during

and following an ATWS event. Following approval of the final EPGs by the NRC, Consumers Power Company will incorporate the EPGs into plant procedures and provide the necessary operator training to implement those procedures.

Based on the measures described above, Consumers Power Company is confident that this USI is being adequately addressed and that the Palisades Plant can continue to operate without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-11
Materials Toughness

Materials fracture toughness and initial safety margins may be reduced due to neutron irradiation and associated thermal effects. 10 CFR 50, Appendices G and H require, respectively, compliance with minimum fracture toughness requirements to be demonstrated and the maintenance of a materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region. The NRC, in a March 5, 1980 letter, submitted a report to Consumers Power Company entitled "Evaluation of the Integrity of SEP Reactor Vessels," (NUREG-0569). NUREG-0569 documents NRC's review of the design specifications and quality assurance programs, status of material surveillance programs, pressure-temperature operating limits, inservice inspection programs and other related issues.

Appendix J of NUREG-0569 discusses the results of NRC's review of the integrity of Palisades RPV. This report finds the RPV to be operating with pressure-temperature limits that are in conformance with Appendix G. Palisades Material Surveillance Program was developed in accordance with ASTM 185-66. However, according to staff reviews, the program meets, and in many areas exceeds, the requirements of ASTM 185-73. The staff concludes that this program is in compliance with Appendix H and will provide an excellent basis for predicting radiation damage on the vessel materials throughout their service life.

It should be pointed out that the Palisades RPV was designed by Combustion Engineering to the 1965 Edition of ASME Code Section III. Additional CE quality control measures supplemented the required ASME Code. Reverification of the initial integrity of the RPV considered performance of a preservice nondestructive examination. Based on the review of the design criteria and the results of the preservice examination and the fracture toughness tests on the vessel materials, the NRC staff concluded that the initial integrity of the vessel is acceptable.

Based on the measures described, Consumers Power Company is confident that this issue is resolved for the Palisades Plant and that the Palisades Plant can continue to be safely operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-12
Fracture Toughness of Steam Generator and RCS Pump Supports

Generic concerns regarding potential for low fracture toughness and Lamellar Tearing of steam generator or reactor coolant pump support materials have been raised by the NRC. Toughness tests of materials of certain supports of the North Anna Power Station No 1 and 2 showed that fracture toughness of one type of material (Steel Spec ASTM A-572) was relatively poor at operating temperatures of 80°F. The toughness of one of the supports made from A36 steel, however, was found to be adequate. The NRC is reassessing the fracture toughness of the SG and RCS pump support materials of these PWR plants with similar designs and materials. It is postulated that support failure could conceivably impair the effectiveness of systems designed to mitigate the consequences of an incident.

Support failures from inadequate fracture toughness are 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 stress on piping is very low);

b. The existence of nonredundant and critical support structural member(s);

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;

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

Regarding potential for Lamellar Tearing, the NRC reported a technical study (Appendix C 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 which revealed that no documentation exists describing inservice failures resulting from Lamellar Tearing. The NRC is recommending future research regarding the relationship of Lamellar Tearing and structural integrity of nuclear power plant support systems.

The generic implications of the North Anna low fracture toughness do not apply to Palisades since a different type of steel is used for the Palisades S/G and RCS pump supports than that used for North Anna.

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 USI A-12 (NUREG-0577) and will continue to remain informed regarding developments related to USI A-12.

Based upon adequate fracture toughness and measures described above, Consumers Power Company is confident that Palisades can continue to be operated before 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

This issue concerns the task of identifying where the present design, analysis and review procedures may not completely account for potentially adverse systems interactions. The design and analyses by the plant designers and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. However, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems. Pipe failure induced systems interactions, fire induced systems interactions, severe environment induced systems interactions, etc, are just a few examples of this concern. This USI was divided into two phases 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. Also, Phase I was to identify areas where NRC review procedures may not have properly accounted for these interactions.

The anticipated Phase II program will not be pursued as a USI. Phase II, which was originated to take specific corrective measures in areas where the Phase I shows a need, will be performed under TMI Action Plan Item II.C.3, Systems Interaction (reference: NUREG-0606, November 16, 1981).

The Palisades Plant design is founded on basic principles of defense in depth. Redundancy of safety systems and protection of equipment from various internal and external phenomena form a part of the design bases for the plant as discussed in various FSAR sections including Section 1.1 and Appendix I. Over the years, in addition to original construction and licensing reviews by Consumers Power Company, the plant architect-engineer and the NRC, numerous other reviews have been conducted which address various aspects of systems interactions. Under the Systematic Evaluation Program these reviews included seismic design, wind and tornado loads, various internal and external missiles, high and moderate energy piping system leaks and breaks, flooding, ventilation and various aspects of the plant's ability to reach and maintain safe shutdown conditions (Topics III-2, III-4.A, III-5.A, III-5.B, III-6, VII-3, IX-5). In response to the TMI Action Plan several other related activities were performed including an auxiliary feedwater reliability study and development of the emergency procedure guidelines. Still other directly related studies include the very extensive programs performed to assure safe shutdown capability following a major destructive fire anywhere in the plant and environmental qualification of safety-related electrical equipment.

The combination of all these studies is likely to satisfy a large portion of any criteria the NRC may develop to address systems interactions. Based on the diverse activities described above, Consumers Power Company is confident that the Palisades Plant is adequately addressing this issue and can continue to operate without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-24
Environmental Qualification of Safety-Related Electrical Equipment

The issuance of NUREG-0588 by the NRC in July 1981 completed this unresolved safety issue. Consumers Power Company submitted to the NRC a report addressing the environmental qualification of safety-related equipment on October 7, 1980. This report superseded a November 30, 1978 report on the same subject which had been developed for the Systematic Evaluation Program. The October 7, 1980 report was revised by Consumers Power Company on October 20, 1980 and June 11, 1981 as more information became available. The NRC's evaluation of the Consumers Power Company submittals was issued in the form of an SER on June 1, 1981.

Consumers Power Company's reply to the SER was provided to the NRC in a submittal dated September 3, 1981. This reply was further amended by a December 4, 1981 report which committed Consumers Power Company to providing one additional follow-up submittal in March 1982 in order to provide the status for implementation of improved surveillance and maintenance program. Consumers Power Company letter dated March 3, 1982 provided this information. The letter also provided additional information regarding justification of interim operation with equipment scheduled for replacement. Consumers Power Company expects to submit one more status report by June 30, 1982 identifying the progress made up to that time.

Consumers Power Company is currently engaged in an extensive program to ensure adequate qualification of all critical components which could be exposed to harsh environments. This program includes the replacement of some components as well as making modifications and upgrading of qualification documentation.

The NRC consultant's preliminary evaluations of our submittals to date demonstrate that adequate justification exists for continued operation until all equipment qualification is completed. Therefore, it was concluded by NRC staff that the Palisades Plant can operate without undue risk to the health and safety of the public.

Based on the extensive program currently in progress, Consumers Power Company is confident that this issue is being adequately addressed and that the Palisades Plant can continue to be safely operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-26
Reactor Vessel Pressure Transient Protection

The Palisades Plant is 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.

This generic safety issue was the subject of extensive overpressurization analyses, plant modifications and procedural changes implemented by Consumers Power Company for the Palisades Plant prior to and during the January 1978 refueling outage. This work was conducted in response to a January 10, 1977 NRC letter which requested plant unique corrective/preventive measures to ensure adequate reactor vessel overpressure protection. Consumers Power Company letters dated June 24, 1977 and November 28, 1977 provided the requested reactor vessel overpressurization implementation program.

Consumers Power Company letter dated June 24, 1977, which contained the Palisades Plant overpressurization analyses, identified plant modifications and procedural changes that were achieved in the January 1978 refueling outage. The results presented in the plant unique analyses demonstrated that the PORV (power-operated relief valve) has sufficient relief capacity to mitigate all potential overpressurization incidents at Palisades. Measures were also implemented to strengthen related procedural and administrative controls. Two significant changes were made to the plant operating procedures. One change to the procedures incorporated a warning that the overpressurization protective system, during cooldown, must be activated when a primary cooling system (PCS) pressure is greater than 400 psi (when T-avg is about 300°F). Moreover, the plant operating procedures for start-up have been modified so that they include a caution during initial conditions that the PCS overpressurization system must be placed in the normal operation mode when the PCS temperature reaches 300°F (approximately 400 psi). Additional administrative modifications were also made.

Consumers Power Company November 28, 1977 submittal documented our commitment to perform hardware modifications to the PCS. This submittal also provided details of the design of the overpressure protection system. Essentially, these hardware modifications, which were implemented during the January 1978 refueling outage, included provisions for automatic operation of both PORVs when pressure exceeds a predetermined set point if PCS temperature is less than a specified value. The modification consisted of installing two individual and completely redundant PCS pressure sensing instrument loops. Each loop derives its operating power from a source independent of the other. An increasing pressure alarm displays PCS OVERPRESSURE on the annunciator window.

Due to these design and procedural modifications, Technical Specifications (TS) changes had to be made. Consumers Power Company letter dated January 3, 1978 requested NRC approval of the TS changes. The NRC issued Amendment No 51 in a letter dated September 10, 1978 authorizing the implementation of the TS modifications. In that letter, the NRC indicated that the TS changes will

enhance low-temperature overpressure protection and increase assurance that the reactor vessel will not be subjected to pressure transients which could exceed the limits established in accordance with Appendix G of 10 CFR 50.

Based upon the measures described above, Consumers Power Company is confident that the Palisades Plant has adequately addressed this generic issue and can continue to 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 shut down, 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 other abnormal occurrences. A similar emphasis has been placed on long-term cooling, which is achieved by the Residual Heat Removal (RHR) system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot standby condition values.

Consumers Power Company has adequately addressed RHR requirements as part of SEP Topics V-10.B/V-11.A. The NRC's documented review of our submittals regarding these SEP topics was published in NUREG-0820, "Integrated Plant Safety Assessment - Systematic Evaluation Program," April 1982, as well as individual topic SERs. RHR reliability is the focus of discussion of this NUREG-0820 report. The report encompasses two areas of concern: overpressurization protection of the shutdown cooling system and use of safety-grade systems for safe shutdown.

Overpressure relief capacity is specified by 10 CFR 50, as implemented by SRP Section 5.4.7, Branch Technical Position ASB 5-1, and Regulatory Guide 1.139, for the shutdown cooling system (SCS) when in operation; that is, when it is not isolated from the reactor coolant system. Since the overpressurization protection system (OPS) fulfills this function as required by Technical Specifications (TS), procedural requirements would be necessary to ensure that OPS is in service whenever the SCS is in service. The basis for this requirement is to prevent the overpressurization of the SCS that could lead to a Loss of Coolant Accident outside containment.

Consumers Power Company, in a letter dated March 31, 1982, documented the measures to be taken to resolve the issue and the date of implementation. We agreed to amend the TS to require the low-temperature OPS to be in service whenever the SCS is in service. The scheduled date for submission of revised TS to the NRC is July 31, 1982. The NRC staff finds this acceptable as reported in NUREG-0820.

With respect to the issue of using only safety-grade systems for safe shutdown as specified in 10 CFR 50, implemented by SRP Section 5.4.7, RSB 5-1 and Regulatory Guide 1.139, the Palisades Plant can be taken from normal operating conditions to cold shutdown using only safety-grade systems, assuming a single failure, and utilizing either onsite or offsite power only through the use of suitable procedures. The Palisades Plant has safety-grade, as well as nonsafety-grade, plant systems capable of safe shutdown under these conditions.

In the March 31, 1982 letter, Consumers Power Company committed to review existing procedures and, if appropriate, modify them to verify that operators

are provided with sufficient guidance to direct use of safety-grade systems in the event of failures in nonsafety grade systems. We proposed to implement this review by the end of first refueling outage after July 31, 1982. NUREG-0820 finds this acceptable.

Based upon the measures described above, Consumers Power Company is confident that the Palisades 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. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall or tip over onto spent fuel in the storage pool or into the reactor core during refueling and damage the fuel, there could be a release of radioactivity. NRC implementation requirements concerning this safety issue were issued to licensees by letters dated December 22, 1980 (Generic Letter 81-07) and February 3, 1981. These documents requested a determination of the extent to which the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," are presently met at Palisades Plant and the identification of what changes and modifications would be required in order to fully satisfy these guidelines. In addition, they requested the implementation of five interim actions for control of heavy loads.

Consumers Power Company letter dated September 23, 1981 documented our efforts to resolve this safety issue. This letter summarized our previous submittals to the NRC which addressed the aforementioned NRC requests. Also, the letter provided, inter alia, identification of heavy load carrying devices, analyses and evaluations conducted to demonstrate compliance with NUREG-0612, and descriptions of Administrative procedures and controls utilized in the handling of heavy loads. A May 15, 1981 letter provided our confirmation that the intent of the five interim action items was presently being met at Palisades. No changes were identified as being necessary to satisfy NUREG-0612 as indicated in our July 6, 1981 submittal.

The NRC has commissioned Franklin Research Center (FRC) to review the responses from all operating plants. FRC is currently reviewing the Consumers Power Company submittals for Palisades. FRC will conclude its review by issuing a Technical Evaluation Report (TER). A preliminary TER has been recently transmitted to Consumers Power Company and is currently being reviewed. Consumers Power Company intends to resolve any questions that the FRC might have.

Based on our belief that the Palisades Plant meets the objectives of NUREG-0612 coupled with the implementation of appropriate interim measures, Consumers Power Company is confident that this generic safety issue is being adequately addressed for Palisades and the Plant can continue to be operated without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-40
Seismic Design Criteria Short-Term Program

Present NRC criteria require that safety-related nuclear power plant structures, systems and components be designed to withstand the effects of natural phenomena such as earthquakes. While these criteria and the associated seismic design sequence include many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. The objective of this program from NRC standpoint is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified.

In addition, a number of plants were constructed before the NRC's current regulations and regulatory guidance were issued. USI A-40 is essentially a compendium of short-term efforts to reevaluate various seismic design criteria and, in particular, assess the seismic design adequacy of older operating plants.

Even though the plant site is located in a seismically inactive area, the Palisades Plant was originally designed to a 0.2g Hausner spectrum. Under the Systematic Evaluation Program (SEP), Topic III-6, the NRC undertook two significant programs to develop a current assessment of the seismic design adequacy of the plant. First, a major effort was expended with consultants to develop a site specific seismic spectrum appropriate for the Palisades Plant site. The product of this effort is documented in NUREG CR-1582 and in various other documents including NRC letter of June 8, 1981. Secondly, the NRC formed a Senior Seismic Review Team (SSRT) to thoroughly inspect the Palisades Plant to identify systems and components which appeared to be either typical samples or the most seismically susceptible. These selected systems or components were then analyzed to a 0.2g R.G.1.60 spectrum. The 0.2g R.G.1.60 spectrum is slightly higher than the original plant design spectrum. The product of the SSRT effort can be found in NUREG CR-1833.

In conjunction with NRC efforts, Consumers Power Company performed extensive reanalyses of piping systems and various components under IE Bulletins 79-02 and 79-14, as well as in response to the SEP. In essence, the NRC efforts confirmed the adequacy of the basic plant design criteria and the Consumers Power Company analysis and modification efforts provided seismic upgrades (improved anchorage, support, etc) for selected plant components where it was deemed necessary.

The net result of these efforts has been a confirmation of the seismic design adequacy of the Palisades Plant.

Based on the extensive efforts discussed above, Consumers Power Company is confident that this generic issue is being adequately addressed and that the Palisades Plant can continue to operate without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-43
Containment Emergency Sump Reliability

Following a postulated Loss of Coolant Accident (LOCA), a break in the reactor coolant system piping, the water flowing from the break would be collected in the containment emergency sump for long-term recirculation through the reactor system by the emergency core cooling (ECC) pumps to maintain core cooling. In addition, this water would be recirculated through the containment spray system to remove heat and fission products from within containment. Loss of the ability to draw water from the containment emergency sump could significantly degrade long-term cooling and potentially disable the ECC system.

Two principal concerns are drawn from these safety considerations: (1) potentially adverse sump hydraulic conditions, and (2) LOCA-generated debris effects. Adverse sump hydraulic conditions are related to specific sump design and plant features that could lead to vortex formation, air ingestion and other poor hydraulic conditions that could lead to a loss of net positive suction head (NPSH) requirements. Debris concerns relate to insulation debris being generated because of the postulated high-energy pipe break, and this debris then migrating to the sump and blocking the sump screens. Again, loss of NPSH is postulated should significant screen blockage occur.

The evaluation of the safety concerns summarized above is being carried out as Unresolved Safety Issue (USI) A-43 on a generic basis. Under sponsorship of the NRC and others, experiments and studies have been under way for several years. Sump hydraulics have been evaluated in full-scale experiments at the Alden Research Laboratory (ARL) for a wide range of sump designs and simulated adverse plant effects (for example, blockage and break flow). Generally speaking, the results show low levels of air ingested (that is < 1%-2%) for a wide range of suctions conditions. Debris effects have been studied for a selective spectrum of plants and reveal that conclusions arrived at are plant dependent. The plant surveys carried out also show that the recent plants employ primarily metallic reflective and encapsulated insulation; these types of insulation pose a lesser blockage potential than previously hypothesized.

The Palisades containment emergency sump is a cylindrical pit 22' in diameter and 3.25' deep, with multiple inlets. There are 5 - 16" diameter, 1 - 24" diameter, 6 - 4" diameter and 2 - 1" diameter inlet pipes to the containment sump. There are also 2 - 24" diameter outlet pipes from the sump. The outlet pipes serve as suction lines for the safety injection systems during recirculation. The bottom of the sump is located at elevation 585'-0". The estimated water level in containment after a LOCA (including the contents of the SIRW Tank) is at the 596' elevation. The center line of the HPSI, LPSI and containment spray pumps is at or below the 572'-7" line. This provides a suction head of greater than 23'. Utilization of experimental results from the ARL test program leads to an estimate of less than 2% air ingestion. With this low level of air, RHR pumping is not expected to degrade. The estimated velocities at the suction line inlet are relatively low (1 to 3.5 fps). The estimated velocities at the inlet to the sump are also relatively low (50% lower than suction inlet piping). Based on the above, satisfactory hydraulic conditions should prevail in the Palisades sump.

There are four types of insulation used inside the containment building at Palisades. The types are: (1) prefabricated metallic reflective utilizing 304 SS inner and outer shells with reflective aluminum alloy sheets supported within, (2) prefabricated, chemically stable, low-density, molded mineral wool encapsulated with an external aluminum jacket, (3) prefabricated, chemically stable, molded calcium silicate encapsulated with an external aluminum jacket (with the exception that main steam is not encapsulated), and (4) chemically stable fiber glass encapsulated with an external aluminum jacket. The prefabricated reflective type and the mineral wool type insulation are used on the NSSS. The fiber glass insulation is antisweat insulation used on the service water lines. The calcium silicate insulation is used on the balance of the hot piping in containment (ie, main steam, feedwater, charging, letdown, safety injection, pressurizer spray).

The suction lines of the RHR are protected from debris ingestion by intake screens. The 16" and 24" sump inlet lines were set into the sump wall with the bottom of the inlet line 2" above the 590'-0" elevation. This minimizes the potential for heavy debris ingress to the sump inlet (ie, heavy debris carried by fluid flow will not enter the sump due to a 2" high curb). The flow path of the containment spray downward from the 649'-0" elevation is through the 6" to 4" drainpipes. The drainpipes run through the reactor shield and deposit the spray water on the 590'-0" deck (after flowing through a 1-1/4" restricting orifice and normal drain gratings). The deposited water then flows directly into the 590'-0" floor drains which empty directly into the sump. Blockage of the floor drains causes water to fill the plenum around the reactor vessel (which drains directly back to the sump). This water also serves to cool the reactor. Alternate flow paths for the containment spray water from the 649'-0" elevation is out and around the biological shield and straight down to the 590'-0" elevation. Operation of the containment spray is expected to be of short duration (~ 30 minutes). This, coupled with the fact that relatively little insulation is present on the 649'-0" elevation, leads to a low probability of debris formation as a result of containment spray operation.

Water paths from the 607'-0" elevation (main coolant loop deck) are limited and tortuous and debris generated from a LOCA in this area is not expected to travel far due to the tortuous path. Heavy debris is expected to sink and not be carried by the flow in that the flow velocity is expected to be low (except in the immediate vicinity of the break). Floating debris is not expected to be a problem at this elevation or at the 590'-0" elevation due to the large flow areas and small flow rates; eg, 300-3,000 gpm. The majority of the debris expected to be generated at the 607'-0" elevation will be NSSS insulation (reflective and mineral wool) which would tend to settle. No debris is expected to be generated at the 590'-0" elevation.

Palisades standard practice is to inspect the sumps and/or sump area commensurate with activities which take place during refueling outages. Standard practice is to conduct a complete visual inspection of the sump, the intake screens and the suction piping at the conclusion of each refueling outage.

Based on the fact that the effect on NPSH margin is expected to be negligible, combined with the housekeeping administrative procedures being implemented

during refueling outages at the plant, Consumers Power Company is confident that the Palisades 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

Station blackout is the complete loss of ac power to the essential and nonessential buses in a nuclear power plant. Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. Each electrical division for safety systems includes an offsite ac power connection, a standby emergency diesel generator ac power supply and dc sources. The systems used for decay heat removal to cool the reactor core following a reactor shutdown are safety systems that are required to meet these criteria.

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 (ie, the 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 (ASLAB-603) for the St Lucie Unit 2 Facility. In addition, in view of the completion schedule for Task A-44 (October 1982), the ASLAB 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 investigated.

By letter dated February 25, 1981 (Generic Letter 81-04), the NRC requested a review of the Palisades Plant capability to mitigate a loss of offsite power coupled with a loss of all onsite ac power. It also requested a review of procedures and training programs pertinent to this event and identification of additional needs in this area.

Consumers Power Company, in a letter dated June 25, 1981, responded to the NRC request by pointing out that the existing design of the Palisades Plant provides inherent capability to withstand a station blackout for significant periods of time without fuel damage. Without any operator action, decay heat can be removed for up to approximately 30 minutes through the main steam line relief valves using the existing steam generator water inventory with the PCS (Primary Cooling System) in natural circulation. Furthermore, to reach a true total loss of ac power at Palisades, multiple independent dc system failures would also have to occur, since the instrument and preferred ac buses are supplied from batteries through inverters. Because dc system failures have been excluded from station blackout considerations, we do not believe that loss of the instrument and preferred ac buses simultaneous with loss of all other onsite and offsite ac power sources was intended by the NRC staff.

Under the Systematic Evaluation Program, dc system reliability and battery capacity was thoroughly reviewed. It was concluded that sufficient capacity exists to supply required control and instrumentation loads for extended periods, while action is taken to restore at least one ac power source.

It was further noted in our responses to Generic Letter 81-04 that operating procedures exist and operator training has been conducted which address mitigation of station blackout events. Annual requalification including simulator training is currently provided for actions required to accommodate a station blackout event. Also, Consumers Power Company has participated in CE Owners' Group activities to develop improved emergency procedure guidelines as required by NUREG-0737 Action Item I.C.1. The Palisades Plant procedures will be modified to implement the CE Owners' Group report on Emergency Procedure Guidelines by the end of first refueling outage after July 31, 1982.

Based upon the measures described above, Consumers Power Company is confident that the Palisades 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 turbines; 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 (RCS). By definition, the decay heat removal (DHR) system is composed of those components and systems required to maintain primary and/or secondary coolant inventory control and to transfer heat from the RCS and containment building to an ultimate heat sink following shutdown of the reactor for normal and transient events (eg, loss of offsite power, loss of main feedwater) and small LOCAs (ie, 1/2" to 2"). The DHR system does not encompass those emergency core cooling components and systems required only to maintain coolant inventory and dissipate heat during the first ten minutes following medium or large LOCAs.

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shutdown decay heat. 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 plant's 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 improving decay heat removal reliability.

The primary method for removal of decay heat from the Palisades reactor is via the steam generators to the secondary system. This energy is transferred on the secondary side to cold feedwater which is supplied by either the main feedwater or auxiliary feedwater system, and it is rejected to either the turbine condenser or the atmosphere via the steam atmospheric dump valves. Improvements of the reliability of the auxiliary feedwater system to assure adequate cooling to the RCS during accidents and other abnormal events were highlighted in NUREG-0737, "Clarification of TMI Action Plan Requirements." NUREG-0737 included Items II.E.1.1 and II.E.1.2 which documented NRC recommendations regarding Auxiliary Feedwater System Evaluation and Auxiliary Feedwater System Automatic Initiation and Flow Indication, respectively.

Consumers Power Company letter dated September 28, 1981 provided an update of our response to the aforementioned NUREG-0737 Action Items. Concerning Action Item II.E.1.1, it was reported in this letter that Phase I of the AFW system modifications consisted of conducting and incorporating the results of an Auxiliary Feedwater Reliability Study into the AFW system redesign. A deterministic review of our AFW system against Standard Review Plan 10.4.9 and Branch Technical Position ASB 10-1 was also conducted to determine the extent NRC requirements could be met through practical modifications to the AFW system. In addition, it was identified that parallel flow paths around all

auxiliary feedwater pump suction valves were installed during the 1981 refueling outage. Installation of the redundant condensate storage tank level indication and alarms and the installation of wide-range steam generator level instrumentation were also completed during the 1981 refueling outage.

The AFWS Phase II modifications to enhance the reliability of the system were described in a Consumers Power submittal dated November 2, 1981. The modifications consist of using portions of the existing AFWS and supplement the existing AFWS with a third dedicated AFW pump. The third pump is the existing, in-place, spare high-pressure safety injection pump, P-66C, adapted for AFWS service. The Phase II modifications are designed to: (1) provide the capability of manual or automatic initiation of AFW flow on low steam generator water level, (2) provide safety grade AFW flow indication in the control room, (3) provide testability of the control circuits, (4) meet applicable seismic and environmental requirements (SRP 10.4.9, ASB 10-1 and NUREG-0737), and (5) deliver AFW to the steam generators within two (2) minutes of low steam generator water level. Consumers Power Company plans to implement the Phase II modifications during the 1983 plant refueling outage.

In response to NUREG-0737, Item II.E.1.2, and pursuant to General Design Criterion 20 of Appendix A, 10 CFR 50, requirements for timely initiation of the AFW system, Consumers Power Company, by letter dated June 1, 1982, furnished the NRC with an implementation schedule for the AFW System Automatic Initiation and Flow Indication. The NRC was notified in this submittal that all of the automatic initiation signals and circuits for the AFW system flow control have been upgraded to safety grade except for E/P Devices 0736A and 0737A and their associated valve positioners. Consumers Power Company expects to upgrade the aforementioned equipment to safety grade during the 1983 refueling outage. The necessary modifications to satisfy the requirements of Item II.E.1.2, Part 2, AFW system flow rate indication, have been completed. Moreover, a human factor analysis will be done during the Control Room Design Review, Item V.D.1.

The Palisades reactor possesses alternate means of removing decay heat if an extended loss of feedwater is postulated. This method is known as "feed and bleed." It uses the high-pressure injection system and/or the charging pumps to add water coolant (feed) at high pressure to the primary system. Decay heat increases the system pressure and energy is removed through the power-operated relief valves and/or the safety valves (bleed).

Based upon the measures described above, Consumers Power Company is confident that the Palisades 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 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 A-46 is to establish explicit guidelines that can 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 a safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

Under the Systematic Evaluation Program (SEP), selected plant components were analyzed on a sampling basis for functionality following a seismic event. These components included the control rod drive mechanisms and the deep draft service water pumps.

In addition, the anchorage of major equipment was addressed. Experience has shown that many, if not most, seismically induced equipment failures in quality industrial facilities have occurred because the components were not adequately anchored to their foundations, and that few equipment failures have occurred in equipment that was anchored. The result of this review has been modifications to substantially upgrade the anchorages of a number of safety-related electrical components at Palisades. These modifications were completed during the 1981 refueling outage.

Consumers Power Company is also participating in a Seismic Qualification Utility Group which is conducting a pilot program to develop an alternative method for seismically qualifying selected nuclear plant components based on actual experience with the equipment during earthquakes. This program is expected to assist the NRC and its consultants in developing qualification methodology for installed equipment at operating plants, in screening and assigning qualification priorities for more efficient utilization of NRC and industry resources, and possibly in qualifying certain classes of equipment on a generic basis without component specific testing or analyses.

Based on the above discussion, Consumers Power Company is confident that this generic issue is being adequately addressed and that the Palisades Plant can continue to operate without undue risk to the health and safety of the public.

UNRESOLVED SAFETY ISSUE A-47
Safety Implications of Control Systems

This safety issue addresses the NRC concern that the potential may exist for accidents or transients to be more severe than previously estimated as a result of control system failures or malfunctions. According to the NRC (NUREG-0606, February 1982), it is not likely that it will be possible to develop generic answers to these concerns since the potential for an accident that would effect a particular control system and the effects of the control system failures will vary from plant to plant. Therefore, the NRC's responsibility in resolving this Unresolved Safety Issue (USI), A-47, is to define generic criteria that may be used for plant specific reviews.

The control systems for Palisades Plant have been designed and built with a high degree of reliability. Operating experience at Palisades has demonstrated this fact. There are safety features in the Palisades nonsafety grade control systems which are designed to prevent transients imposed when single component failures occur.

Some aspects of USI A-47 were the subject of an extensive review and evaluation process conducted by Consumers Power Company in response to IE Notice 79-22, Qualification of Control Systems, dated September 14, 1979 and IE Bulletin 79-27, Loss of Non-Class 1-E Instrumentation and Control Power System Bus During Operation, dated November 30, 1979.

1. IE Notice 79-22:

   Consumers Power Company letters dated October 9, 1979 and October 16, 1980 responded to IE Notice 79-22. This notice discussed a concern involving potential effects of nonsafety grade equipment on safety analyses and performance of safety grade equipment. This concern related to the effect on nonsafety grade equipment of an adverse environment which might be produced by failure of a high-energy line. The October 9, 1979 Consumers Power Company submittal described the evaluation we conducted of the effect of high-energy line break induced adverse environment on nonsafety grade control equipment at Palisades. The results of the evaluation indicated that no potential failures were identified which could constitute a substantial safety hazard. Accordingly, no modifications were needed as a result of this IE Notice concern.

   Consumers Power Company letter dated October 9, 1979 also committed us to perform an additional analysis to evaluate what long-term actions are required. Consumers Power Company letter dated October 16, 1980 furnished our response to this commitment. Briefly, since the Palisades Plant is an SEP plant, this analysis/review was tied into the Environmental Qualification of Electrical Equipment (EEQ) Project. This EEQ Project was completed and submitted to the NRC on October 7, 1980. In the course of addressing the EEQ subject, we fulfilled our commitment of performing the additional analysis to evaluate what long-term actions are required as stated in our October 9, 1979 letter. The necessary long-term actions involve the environmental qualification of certain instruments and controls

by either modifying or replacing the equipment, and are being performed on a schedule consistent with that required for resolution of USI A-24.

2. IE Bulletin 79-27:

Consumers Power Company response to IE Bulletin 79-27 was transmitted to the NRC in our letters dated March 3, 1980 and May 20, 1980. Three items were identified in IE Bulletin 79-27 requiring our review and evaluation. Briefly, the bulletin requested: (1) review and evaluation of the effects of loss of power to I&C buses, (2) review and preparation of emergency procedures required to achieve cold shutdown condition in the event of loss of power, and (3) review of both Class 1E and Non-Class 1E safety-related power supply inverters, as related to the IE Bulletin 79-27, noted failure of 120 V vital ac power supplies. The NRC also requested descriptions of measures (eg, Procedural Modifications) to be implemented by Consumers Power Company to address the above three items.

It should be pointed out that there is no alternate or preferred ac supply with static switches like those described in IE Bulletin 79-27. The Palisades Plant has five independent 120 V ac instrumentation and control (I&C) power divisions. Consumers Power Company March 3, 1980 correspondence addressed Items 1 and 3, and Item 2 was addressed in our May 20, 1980 letter. Based on our evaluation of Item 1, the loss of any single instrument and control bus will not prevent the operators from safely shutting the plant down from the main control room. Regarding Item 3, it was concluded that this item was found not to be applicable to the Palisades I&C buses since the inverters were designed differently from those which malfunctioned at Arkansas Nuclear One-Unit 2. Thus, in both cases, no design modifications or administrative controls were deemed necessary.

With respect to Item 2, Consumers Power Company May 20, 1980 letter documented our response. As a result of our review and evaluation, five new emergency procedures were created to specifically address the control of the plant following loss of power to Class 1E and Non-Class 1E I&C buses. These new emergency procedures were implemented prior to the start-up following the 1980 refueling outage.

Based upon the high degree of reliability of the Palisades Plant control systems, augmented with the measures described above, Consumers Power Company is confident that the Palisades Plant is adequately addressing this generic issue and can continue to be operated without undue risk to the health and safety of the public.

PALISADES UNRESOLVED SAFETY ISSUE A-48
Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Following a Loss of Coolant Accident 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.

Because of the potential for significant hydrogen generation as the result of an accident, GDC 41, "Containment Atmosphere Cleanup," in Appendix A to 10 CFR Part 50 specifies that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

Containment hydrogen recombiners were installed at Palisades in 1973 for post-accident hydrogen control. The recombiners are described in Section 6.6 of the FSAR.

According to NUREG-0820, "Integrated Plant Safety Assessment Systematic Evaluation Program," the NRC staff concluded that the design of the Palisades Plant recombiners is consistent with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident." This Regulatory Guide assumed hydrogen generation corresponds to a 5% metal-water reaction during the LOCA blowdown.

The Hydrogen Control System is designed to insure that the hydrogen concentration inside the containment is maintained below the lower flammability limit of 4.0 volume percent. The system consists of two safety-related hydrogen recombiners inside containment. Hydrogen mixing is provided by the containment air cooler fans. The current requirement for hydrogen control is to limit the concentration to less than 4.0 volume percent produced by a 5% metal-water reaction during a LOCA plus the other sources of hydrogen mentioned above. A 5% metal-water reaction produces a 1.4 volume percent concentration of hydrogen without taking into account the sources which are long-term sources relative to the metal-water reaction. The accident at TMI-2 on March 28, 1979 resulted in hydrogen generation well in excess of 4 volume percent with a resulting hydrogen-air reaction. Because of this, the NRC has been evaluating the extent and consequences of hydrogen burns in excess of what was previously required.

An evaluation by the NRC (SECY 80-107, Denton to Dircks) came to the conclusion that reactor containment buildings could withstand twice the design pressure for short periods without catastrophic failure. The evaluation also concludes that a dry containment (such as Palisades) is least impacted by hydrogen burn considerations because of the large volume and high design pressures and a PWR is generally less impacted than a BWR because the amount of zirconium cladding in a BWR is typically twice that of a PWR.

Calculations have been performed for Midland which has a similar containment to Palisades and about the same amount of zirconium in the core. The calculations show that a 75% metal-water reaction with resulting hydrogen burn could take place without exceeding twice the containment design pressure. Using the Midland results and taking into account the smaller containment volume and design pressure, and the larger amount of zirconium cladding in Palisades, a 45% metal-water reaction could occur without a containment failure at Palisades.

Regarding the effect of a hydrogen burn on safety equipment inside of containment, (SECY letter 80-107, "Proposed Interim Hydrogen Control Requirements for Small Containments" dated February 22, 1980, Section 3.5) it was concluded 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 successfully function following a containment hydrogen burn. Therefore, it is judged that potential hydrogen burns do not constitute a threat to safety at Palisades, 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 Palisades 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 generic safety issue addresses the possibility of brittle fracture of the reactor vessel during either a small break loss-of-coolant (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 extent to which reactor vessel brittle fracture, Pressurized Thermal Shock (PTS), may result during an overcooling/repressurization transient is contingent upon vessel material properties and the time history of the thermal and pressure transients to which it may be subjected.

Consumers Power Company is actively supporting the efforts of the Combustion Engineering Owners' Group (CEOG) in resolving the PTS issues. Analyses contained in CEN-159, "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCAs With Loss of Feedwater for the Combustion Engineering NSSS," shows that the Palisades Plant can be operated to End of Life (EOL) or approximately 32 effective full-power years for the transients analyzed with no predicted loss of reactor vessel integrity. Additional scoping analyses were presented to the NRC during the CEOG status meeting held on March 4, 1982 which also showed EOL capability for the Palisades reactor vessel.

Considering the above favorable results, we have concluded that the integrity of the Palisades reactor vessel would not be compromised and, therefore, the Palisades Plant can be safely operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.