Consumers Power Company

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March 4, 1980

Director, Nuclear Reactor Regulation
Att Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - REQUIREMENTS RESULTING FROM
REVIEW OF TMI-2 ACCIDENT - ACTIONS TAKEN
IN RESPONSE: REVISION 1

Consumers Power Company letter dated December 27, 1979 described actions to be taken at the Palisades Plant in response to requirements resulting from NRC review of the TMI-2 accident. Review of this letter has identified a need to make minor changes.

Transmitted herewith is Revision 1 of the December 27, 1979 letter incorporating the necessary changes to reported actions. These changes are identified by a vertical line in the margin.

David P. Hoffman

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Nuclear Licensing Administrator

CC JGKeppler, USNRC

Attachment: 88 Pages

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CONSUMERS POWER COMPANY
Palisades Plant

NUREG 0578 Commitments
Revision 1
March 4, 1980

EMERGENCY POWER SUPPLY (2.1.1)

Pressurizer Heaters

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of off-site power, the following positions shall be implemented:

Pressurizer Heater Power Supply

2.1.1(1) The pressurizer heater power supply design shall provide the capability to supply, from either the offsite power source or the emergency power source (when offsite power is not available), a predetermined number of pressurizer heaters and associated controls necessary to establish and maintain natural circulation at hot standby conditions. The required heaters and their controls shall be connected to the emergency buses in a manner that will provide redundant power supply capability.

Response to 2.1.1(1)

The present design of the pressurizer heater power(1) supply at the Palisades Plant is such that one half of the heater capacity (750 kW) is supplied from an off-site power source; the other half of the pressurizer heaters are supplied from the emergency power source (diesel generators)(2). The capability of being able to supply these heaters by either source ensures the ability of being able to depend on the heaters under any condition of power loss that may be encountered. By using these heaters and associated controls, needed for heater operation, natural circulation will be able to be established and maintained in a hot standby condition.

Modifications to the pressurizer heaters on the 1-E bus will be completed to provide the capability of being powered from a redundant emergency power supply and will allow sufficient heater capacity to maintain natural circulation.

One half of the pressurizer heaters (750 kW) through review and experience(3) is greater than the number required to maintain natural circulation in the hot standby condition(4). During the power physics testing following the present

(1) CEN-125 , "Input for NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," December 1979, Pg 2-7, (Attachment 1)
(2) Palisades Plant FSAR, Section 4, Pg 4-23
(3) Palisades Test Report (PTR-9) 60% Power Test, June 8, 1972
(4) CEN-125 , "Input for NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," December 1979, Pg 2-25 and 2-26, (Attachment 1)

refueling, a test will be conducted using a nominal 225 kW of pressurizer heater capacity to demonstrate the ability to maintain 20°F subcooling.

Attachment 1 has concluded that natural circulation cooling is feasible until the pressure has decayed to the point that only about 20°F subcooling is available.

This test will be somewhat conservative since the test will be conducted under a "clean core" condition. This condition will not take any credit for core heat since 1/3 of the core will be new and the other 2/3 has not been under power operation conditions for over 6 months and doesn't have much decay heat. Under conditions where the utilization of these heaters will be needed, some degree of decay heat will be present. This decay heat will help establish natural circulation and, therefore, will not require the pressurizer heaters to operate as long. Results of the tests and data analysis will be sent to the NRC upon request.

Attachment 1 (Page 2-20) states that the amount total kilowatt need to maintain natural circulation is 150 kW on emergency power. The amount of heaters needed to maintain natural circulation will be incorporated into the appropriate procedures to assist the operators in case they must load the heaters on the diesel.

The heaters powered from the emergency bus are currently automatically shed from load upon actuation of the SIS.

2.1.1(2) Procedures and training shall be established to make the operator aware of when and how the required pressurizer heaters shall be connected to the emergency buses. If required, the procedures shall identify under what conditions selected emergency loads can be shed from the emergency power source to provide sufficient capacity for the connection of the pressurizer heaters.

Response to 2.1.1(2)

Procedures that make the operator aware of when and how the required pressurizer heaters are connected to the emergency buses are presently being reviewed/revised. This will be completed prior to start-up from the present refueling outage.

The training of the operators on procedures is handled through the Training Department at Palisades. Training is continuously being provided so operators can be provided with the most complete and latest operating techniques and procedures. All plant operators at the Palisades Plant will be trained on all revised procedures prior to plant start-up.

2.1.1(3) The time required to accomplish the connection of the preselected pressurizer heater to the emergency buses shall be consistent with the timely initiation and maintenance of natural circulation conditions.

Response to 2.1.1(3)

Based on Figures 3.a and 3.b of reference(5), the time permitted to load additional heater capacity at Palisades and thereby maintain natural circulation is:

14 Hours With a Safety Valve Leakage of 1 GPM
6.8 Hours With a Safety Valve Leakage of 2 GPM

These times assume that (1) 150 kW of heaters are energized at 30 minutes after loss of off-site power, (2) a nominal operating pressure of 2,000 psia, and (3) an ambient heat loss of 500,000 Btu/h.

Pressurizer heaters on Bus 1-D are connected to a hand switch in the control room. This switch enables the operator to manually connect these heaters on an emergency bus.

Modifications to the pressurizer heaters on the 1-E bus will be completed to provide the capability of being powered from a redundant emergency power supply and will allow sufficient heater capacity to maintain natural circulation. The time required to complete this connection to the redundant power supply will be less than 5 hours. This modification will be completed prior to start-up.

(5) CEN-125 , "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," December 1979, Pg 2-25 & 2-26 (Attachment 1)

2.1.1(4) Pressurizer heater motive and control power interfaces with the emergency buses shall be accomplished through devices that have been qualified in accordance with safety-grade requirements.

Response to 2.1.1(4)

All Class 1E interfaces for main and control power are protected by safety grade circuit breakers.

EMERGENCY POWER SUPPLY (2.1.1)

Pressurizer Level and Relief Block Valves

Position

Consistent with satisfying the requirements of General Design Criteria 10, 14, 15, 17 and 20 of Appendix A to 10 CFR Part 50 for the event of loss of off-site power, the following positions shall be implemented:

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

2.1.1(1) Motive and control components of the power-operated relief valves (PORVs) shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

Response to 2.1.1(1)

Palisades operates at pressure with the PORV block valves closed. Credit for PORV relief capacity is not assumed in any Palisades Plant FSAR Chapter 14 analyses. During shutdown conditions, the block valves can be opened and the PORVs used for overpressure protection.

Either PORV has adequate relieving capability to protect the primary coolant system (PCS) from overpressurization when the transient is limited to either (1) the start of an idle primary coolant pump with the secondary water temperature of the steam generator $\leq$ 70°F above the PCS cold leg temperatures or (2) the start of the HPSI pump and its injection into a water solid PCS(6)(7)(8).

The present status of the power-operated relief valves shows that:

PRV 1042B is powered from safety-related Class 1E motor control center (MCC) 1, and PRV 1043B is powered from nonsafety-related MCC 9.

(6) Technical Specifications for Palisades Plant
Section 3.1.8, Page 3-25a
(7) Palisades Overpressurization Analysis, June 1977.
Submitted to NRC by letter dated June 24, 1977.
(8) Palisades Plant Primary Coolant System Overpressurization
Subsystem Description. Submitted to NRC by letter dated
November 28, 1977.

Prior to start-up of Palisades, from the present refueling outage, both of the power-operated relief valves will be powered from an emergency bus. This modification will allow the motive and control components of the power-operated relief valves the capability of being supplied by either off-site or emergency power.

In case the emergency power is interrupted, these valves will fail closed.

Changeover of the PORV and block valve motive and control power from the normal off-site power to the emergency on-site power can be accomplished manually in the control room.

It should also be pointed out that the design of the Palisades Plant is such that instrument air is not needed for operation of the PORV/block valves.

2.1.1(2) Motive and control components associated with the PORV block valves shall be capable of being supplied from either the offsite power source or the emergency power source when the offsite power is not available.

Response to 2.1.1(2)

This motive and control power for the block valve will not be supplied from an emergency power bus different from that which supplies the PORV. Based on Failure Modes Effects Analysis (FMEA)(9) performance by Consumers Power Company, it has been concluded that "crisscrossing" the power supplies to the PORVs and block valves is less conservative than having each PORV powered from the same source as its block valve. Palisades operates at pressure with the PORV block valves closed.

The present status of the block valves shows block valves MO-1042A and MOV-1043A(10) are supplied from safety-related Class 1E MCCs 1 and 2, respectively.

(9) PORV and Block Valve FMEA - Palisades Plant, December 1979 (Attachment 2)
(10) PMIDs E-1 & E-5

2.1.1(3) Motive and control power connections to the emergency buses for the PORVs and their associated block valves shall be through devices that have been qualified in accordance with safety-grade requirements.

Response to 2.1.1(3)

Because the block valves are connected in series with the PRVs, a stuck-open PRV can be isolated by closing the respective block valve. However, PRV-1043B will be disconnected from MCC 9 and reconnected to safety-related Class 1E MCC 2 to ensure continued operation on a loss of off-site power.

A particular PORV and its associated block valve are supplied from the same power supply. To switch power supplies would be less conservative than the present arrangement.

2.1.1(4) The pressurizer level indication instrument channels shall be powered from the vital instrument buses. The buses shall have the capability of being supplied from either the offsite power source or the emergency power source when offsite power is not available.

Response to 2.1.1(4)

Pressurizer level indication instrumentation channels are powered from the vital instrument buses which are powered from the plant batteries. The batteries have the capability of being supplied from either the off-site power source or the emergency power source when off-site power is not available.

PERFORMANCE TESTING FOR BWR AND PWR RELIEF AND SAFETY VALVES (2.1.2)

Position

Pressurized Water Reactor and Boiling Water Reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

Response to 2.1.2

By letter dated December 17, 1979, Mr William J Cahill, Jr, Chairman of the EPRI Safety and Analysis Task Force, submitted "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," December 13, 1979.

Consumers Power Company considers the program to be responsive to the requirements presented in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" dated July 1979, Item 2.1.2 which recommended in part, "commit to provide performance verification by full scale prototypical testing for all relief and safety valves. Test conditions shall include two-phase slug flow and subcooled liquid flow calculated to occur for design basis transients and accidents."

The EPRI Program Plan provides for a completion of the essential portions of the test program by July 1981. Consumers Power Company will be participating in the EPRI Program to provide program review and to supply plant specific data as required.

DIRECT INDICATION OF POWER-OPERATED RELIEF VALVE AND SAFETY VALVE POSITION FOR PWRs AND BWRs (2.1.3.a)

Position

Reactor System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

Response to 2.1.3.a

The plant is installing acoustical transducers(11) on the two PORVs and the three safety valves on the pressurizer. The construction of valve indication system will be completed in two phases. Phase One is construction for inside containment(13) and Phase Two is construction outside containment(12). The installation will meet the same requirements as other engineered safety features (except redundancy). Seismic and temperature testing(14)(15) is being conducted on the relief valve indication monitoring system. This testing is being performed by Environmental Testing Corporation. The installation and checkout procedure is also enclosed(16). These will provide a direct valve indication in the control room prior to plant start-up from the present refueling outage. There will be annunciation when a valve has opened and there will be position indication for each valve in the control room. The alternative method of determining valve position is through the use of the RTDs located on the exhaust lines of the valve and the quench tank level indication (hi, low). The position indication will be powered from the safety grade bus prior to start-up. The system will be operable, following final adjustment of the bandwidth which will be performed during zero power physics testing.

The signal conditioning equipment is being tested for the environment it will be operating in. The equipment is also being tested so it would be qualified in a LOCA environment. Consumers Power Company is attempting to complete qualification requirements by January 1, 1981.

All associated equipment will be seismically qualified as per B&W schedule which has not been established to date.

(11) "Project Requirements for Palisades Nuclear Plant PCS Relief Valve Position Indication," November 8, 1979 (Attachment 3)
(12) "Construction Specifications for Outside Containment Portion of Pressurizer Relief Position Indication System at Palisades Nuclear Plant," 12-6-79 (Attachment 4)
(13) "Construction Specifications for Inside Containment Portion of Pressurizer Relief Position Indication System at Palisades Nuclear Plant," 11-3-79 (Attachment 5)
(14) "Seismic Vibration Testing of Various Electric Items for B&W Company," Test Report No 14689, February 28, 1979 (Attachment 6)
(15) "Results From Charge Preamplifier Temperature Test," BAW 1590, November 1979 (Attachment 7)
(16) "Installation and Checkout Procedure C-010-00 for Valve Monitoring System," (Attachment 8)

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

Subcooling Meter

Position

Licensees shall develop procedures to be used by the operator to recognize inadequate core cooling with currently available instrumentation. The licensee shall provide a description of the existing instrumentation for the operators to use to recognize these conditions. A detailed description of the analyses needed to form the basis for operator training and procedure development shall be provided pursuant to another short-term requirement, "Analysis of Off-Normal Conditions, Including Natural Circulation" (see Section 2.1.9 of NUREG-0578).

In addition, each PWR shall install a primary coolant saturation meter to provide on-line indication of coolant saturation condition. Operator instruction as to use of this meter shall include consideration that is not to be used exclusive of other related plant parameters.

Response to 2.1.3.b

Palisades will install two Safety Class 1, Seismic Class 1, Quality Class 1 "Subcooled Margin Monitors"; they will meet IEEE 344-1975 and IEEE 323-1974 standards. These will continuously calculate and display the degrees F of primary coolant subcooling. One will monitor coolant Loop 1 and the other coolant Loop 2. Redundant safety grade temperature inputs from each hot leg will be used. The redundant pressure indication used will come from Q-Listed pressure transmitters supplied with safety grade power. If certification of conformance to IEEE 323 and 344 cannot be obtained for all elements in the existing pressure current loops, new Class 1E components will be obtained and installed for Phase II. A continuous display of the primary coolant saturation conditions will be provided. Two wide range Class 1E pressure and temperature inputs will be installed during the next refueling outage.

Safety grade calculational devices and display (minimum of two meters) will be used. The plant computer is not needed for this process.

To avoid the problem of the subcooling meter having an adverse impact on the reactor protection or engineered safety features system, one meter will be installed on primary coolant Loop 1 while the other will be installed on Loop 2. Each monitor will receive Channel "C" and "D" inputs. Isolation will be used so channels cannot be crossed.

In the long term, Consumers Power Company will address the requirements of Regulatory Guide 1.97, which is under development, as it applies to the subcooling meter.

Previously existing instrumentation to detect inadequate core cooling:

Primary system pressure. In addition to monitoring pressure instruments, the operator is directed to monitor the following instruments/indicators which could be used to indicate the cause of abnormal pressure:

- Pressurizer Heaters (On or Off)

- Pressurizer Spray Valve Position Indication

- Charging Flow

- Letdown Flow

- PORV and Pressurizer Relief Valve Position

  + PORV Position Indication
  + PORV Block Valve Position Indication
  + Relief Valve Discharge Piping Temperatures
  + Quench Tank Level, Pressure and Temperature Instruments

HPSI pumps running indication and HPSI flow indication.

Atmospheric dump valves and steam generator relief valves positions.

T_{avg}

T_h

T_c

Incore thermocouples.

Start-up neutron detectors.

In addition to the above existing instruments, a subcooling meter and more reliable PORV/pressurizer relief valve position indication will be installed. Procedures will be revised accordingly upon completion of this installation.

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, Etc)	Tsat-T or Psat-P
Display Type (Analog, Digital, CRT)	Digital
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Main Control Room Panel
Alarms (include setpoints)	≤ 50°F of Subcooling
Overall uncertainty (°F, PSI)	± 3°F
Range of Display	0°F-999°F of Subcooling
Qualifications (seismic, environmental, IEEE323)	Seismic 1, Class 1E, IEEE 323 IEEE 344

Calculator

Type (process computer, dedicated digital or analog calc.)	Dedicated Digital
If process computer is used specify availability. (% of time)	N/A
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	Highest Temp Lowest Pressure
Qualifications (seismic, environmental, IEEE323)	Seismic 1 IEEE 323 Class 1E IEEE 344
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

Input

Temperature (RTD's or T/C's)	RTDs
Temperature (number of sensors and locations)	8; Both Hot Legs & All 4 Cold Legs
Range of temperature sensors	515°F-615°F

Uncertainty* of temperature sensors (°F at 1)	0.2%
Qualifications (seismic, environmental, IEEE323)	Environmental
Pressure (specify instrument used)	Rosemount 1151GP PT
Pressure (Number of Sensors and Locations)	2 Pressurizers
Range of Pressure Inputs	0-2500 PSJA
Uncertainty* of pressure sensors (PSI at 1)	0.25%
Qualifications (seismic, environmental, IEEE323)	None
<u>Backup Capability</u>	
Availability of Temp & Press	Temp & Pressure Indicators
Availability of Steam Tables etc.	Steam Tables
Training of Operators	Yes
Procedures	Yes

*Uncertainties must address conditions of forced flow and natural circulation.

*Temperature sensors are located on the top of the coolant pipes so that under stratified conditions, they will read the highest temperature in the pipe. This will result in conservative indication.

Pressure sensors are on the pressurizer and will indicate pressure that is somewhat lower than that in the reactor core. Consequently, this will result in a more conservative indication.

INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (2.1.3.b)

ADDITIONAL INSTRUMENTATION

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement those devices cited in the preceding section giving an unambiguous, easy-to-interpret indication of inadequate core cooling. A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Response to 2.1.3.b Additional Requirements

The instrument response of a reference C-E plant for events which have the potential for inadequate core cooling is documented in CEN-117(17). The conclusion reached in CEN-117 is that there currently is sufficient instrumentation in the plant to be used to detect inadequate core cooling. The Palisades Plant is currently updating its emergency procedures and associated operator training based on the information in CEN-117. This effort will be completed prior to start-up. The functional requirements for and a conceptual design description of a reactor vessel water level measurement system is provided as an attachment(18). This design was prepared by the C-E Owners Group for discussion with NRC in a generic resolution review. If required by the generic resolution review with the NRC staff, the functional requirements and conceptual design will be submitted for Proposal Review by the NRC staff prior to implementation. The implementation schedule for such a device, should it be deemed necessary, will be established during the Proposal Review.

(17) Combustion Engineering Report CEN-117 in Letter Dated October 31, 1979 From Consumers Power Company to the NRC

(18) CE Post-TMI Evaluation Task 2 Conceptual Design for a Reactor Vessel Level Monitoring System, December 21, 1979 (Attachment 9)

CONTAINMENT ISOLATION (2.1.4)

Position

2.1.4(1) All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e., that there be diversity in the parameters sensed for the initiation of containment isolation.

Response to 2.1.4(1)

The existing control circuit design of containment isolation valves will result in automatic opening of the valves upon resetting of the containment isolation signal (CIS) if their hand switch is positioned to "open" because the control switch for each isolation valve has maintained contacts. This will be modified such that all cycling valves will be locked closed following CIS, such that deliberate operator action is required to reopen the penetration.

It will be verified that all nonessential systems receive an isolation signal on containment isolation. This will be done by start-up.

A survey of the Palisades containment will be carried out, and the precise location of the existing radiation detectors will be noted. Possible locations for additional radiation detectors will also be determined.

Containment high radiation and containment high pressure are the parameters that provide the diversity for containment isolation.

Palisades has diverse containment isolation signals that satisfy safety-grade requirements.

2.1.4(2) All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to NRC.

Response to 2.1.4(2)

Essential systems are those critical to the immediate mitigation of any event that results in automatic containment isolation. Essential systems provide RCS inventory and pressure control, reactivity control, core cooling, secondary heat sink, containment cooling (depressurization) and safe shutdown.(19) Essential systems must be available in response to accident parameters without operator action and, therefore, should not be automatically isolated as part of the Containment Isolation System.

Essential Systems

High-Pressure Safety Injection (HPSI)

An ESF system that automatically provides cooling water makeup and borated water for negative reactivity in the event of an RCS depressurization or containment pressurization.

Low-Pressure Safety Injection (LPSI) and Shutdown Cooling

An ESF system that automatically provides large quantities of borated water for LOCAs and a flow path for shutdown cooling through the shutdown cooling heat exchanger. It may be in operation at the time of containment isolation.

Containment Spray System (CSS) (With/Cont Sump Recirculation)

An ESF system that automatically removes heat and iodine from the containment atmosphere in the event of a LOCA or MSLB. The sump recirculation automatically provides a long-term water source for both containment spray and safety injection.

Containment Emergency Cooler Cooling Water

A system required to remove heat from the containment atmosphere in conjunction with the CSS. Some plants may use the normal containment cooler for post-accident conditions with the same or alternate cooling water source.

(19) CEN-125 , "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," December 1979, Pages 5-6 (Attachment 1)

Auxiliary Feedwater

A system required to automatically supply feedwater to the steam generators for residual heat removal.

Main Steam and Feedwater System (MSS and MFS)

These systems, which are categorized as essential, differ somewhat from those above. The MSS and MFS, although not the only means, are the preferred means for heat removal during a small LOCA. These systems should not, therefore, be arbitrarily isolated on a Containment Isolation Actuation Signal (CIAS) necessitating a steam release to the atmosphere. However, based on containment pressurization considerations for a steam line break, isolation may be required on a Main Steam Isolation Signal (MSIS) or other actuation signal which occurs on containment high pressure or steam generator low pressure.

Charging

A system required for reactivity and inventory control at normal system operating pressures.

Nonessential Systems

Those systems not included above.

2.1.4(3) All non-essential systems shall be automatically isolated by the containment isolation signal.

Response to 2.1.4(3)

Palisades system penetrations will be reviewed prior to plant start-up to assure that all nonessential systems do automatically isolate on CIS. The review is expected to verify that there are two exceptions to the automatically isolated CIS criteria which are:

(a) Component Cooling Water
(b) Instrument Air

In both of these cases, isolation is affected in a way which is felt to be functionally equivalent to automatic isolation on CIS. The reasons for this conclusion are discussed below:

Instrument Air

The isolation valves for this system consist of a remote manual valve and a check valve - both outside containment. This line is designed for higher pressure than containment and, therefore, it would not be expected to fail outside containment as a result of a LOCA inside containment. Hence even without automatic isolation, no leak path should be created for radioactive fluids to leave containment. A check valve is considered to be an automatic isolation valve. Thus, although the instrument air line is not automatically isolated on CIS, a break upstream of the check valve would be automatically isolated if containment high pressure existed.

CCW

CCW to the containment is isolated on SIS rather than CIS. An SIS is generated on high containment pressure as well as on low primary coolant system pressure. Thus, CCW is automatically isolated by two diverse signals. The CCW system is FSAR Class 1 outside of containment and designed for a higher pressure than containment; therefore, it would not be expected to fail as a result of a LOCA. Hence, even if the integrity of the CCW system was breached inside containment, opening of the isolation valves would not create a leak path for radioactive fluids to leave containment. This system will not be latched due to the effect that it would have on the RCP seals.

2.1.4(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

Response to 2.1.4(4)

All automatic containment isolation valves will be electrically locked-closed to preclude automatic opening upon resetting of the containment isolation signal. Subsequent to resetting of CIS, the control switch for each valve will need to be moved to the "close" position and then to the "open" position to reopen the valve. This will be accomplished by the use of a "seal in" relay, thereby requiring deliberate operator action to reopen a containment isolation valve. Modifications will be completed prior to start-up of the unit after present refueling outage.

DEDICATED H_2 CONTROL PENETRATIONS (2.1.5.a)

Position

Plants using external recombiners or purge systems for post-accident combustible gas control of the containment atmosphere should provide containment isolation systems for external recombiner or purge systems that are dedicated to that service only, that the redundancy and single failure requirements of General Design Criterion 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

Response to 2.1.5.a

The Palisades Plant does not use or take credit for their purge system for post-accident combustible gas control and has internal hydrogen recombiners. This requirement is not applicable to the Palisades Plant.

CAPABILITY TO INSTALL HYDROGEN RECOMBINER AT EACH LIGHT WATER NUCLEAR POWER PLANT (2.1.5.c)

Position

The procedures and bases upon which the recombiners would be used on all plants should be the subject of a review by the licensees in considering shielding requirements and personnel exposure limitations as demonstrated to be necessary in the case of TMI-2.

Response to 2.1.5.c

The plant will review and revise procedures as to the use of the hydrogen recombiner. In the event of a LOCA, the procedures will require the operator to energize the hydrogen recombiner immediately. This action will be completed prior to plant start-up.

The shielding and associated personnel exposure limitations associated with the recombiner use is not applicable to the Palisades Plant due to the location of these recombiners in containment.

INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS FOR PWRs and BWRs (2.1.6.a)

Position

Applicants and licensees shall immediately implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate Leak Reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

   b. Measure actual leakage rates with system in operation and report them to the NRC.

Response to 2.1.6.a(1)

I. System Definition

To assist in defining the systems which would contain highly radioactive fluid following an accident, and provide a third party review function, MPR Associates was contracted. An extensive review of plant system drawings, operating and emergency procedures, and other documents was performed jointly by Consumers Power Company and MPR to define those systems of concern. The results of this review and the basis for including or excluding systems or portions of systems were summarized in MPR Associates letter P-490 dated December 4, 1979, included as Attachment(10). A brief synopsis of the systems included is as follows:

High-pressure Safety Injection (HPSI) - Required to operate in the recirculation mode to draw water from the containment sump and return it to the PCS following a nonisolable breach of the PCS.

Low-Pressure Safety Injection (LPSI)/Shutdown Cooling - Can operate in the recirculation mode similar to HPSI or in the shutdown cooling mode with the LPSI pump suction being taken from the PCS. Note that Palisades does not have a separate system for the shutdown cooling, but uses the LPSI system in a slightly modified lineup for this purpose.

Containment Spray - Required to operate in the recirculation mode to draw water from the containment sump and return it to the containment building spray nozzles following a LOCA or MSLB.

Sampling System - Post-accident use would be required at some point to determine the extent of fuel damage and PCS boron concentration and chemistry. This system, however, is being upgraded or replaced in accordance with Recommendation 2.1.8.a, "Improved Post-Accident Sampling

Capability." In the interim, the existing system will be tested as part of the overall program to minimize leakage.

Systems which will not be included in this program are as follows:

Chemical and Volume Control - In the event of an accident involving fuel damage, letdown is isolated at the containment penetration by a containment high radiation (CHR) or containment high-pressure (CHP) signal. The letdown lines, ion exchangers, volume control tank, etc, would therefore not become contaminated with highly radioactive PCS water. Letdown flow will not have to be reestablished following an accident.

Liquid Radwaste System - As above, liquid radwaste lines are isolated at the penetrations by CHR/CHP, and would not be required to be used. This is dependent on the completion of four modifications which are discussed in the next section.

Vacuum Degasifier/Gaseous Radwaste System - As above, containment lines to these systems are automatically isolated at the penetrations by CHR/CHP and would not be required to be used. This is also dependent on the completion of four modifications, discussed later.

Leak reduction measures for all systems that could carry radioactive fluid outside of containment will be completed prior to plant start-up of present refueling outage.

The measurement of actual leakage rates with the system will be completed prior to criticality. The majority of the systems was tested with the greatest leakage of 1/2 gpm observed on Valve CV-3006. This valve was repacked and leakage was reduced to zero. The highest leakage identified was approximately 4 ml per minute. A report with results will be sent to the NRC upon completion of this program.

2.1.6.a(2) Continuing Leak Reduction

Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response to 2.1.6.a(2)

II. The following is a summary description of the leak reduction program for systems which would or could contain highly radioactive fluids during a serious transient or accident.

The Palisades leak reduction program has three basic phases. First, as a result of the system review, several modifications will be performed to prevent the unnecessary contamination of certain systems. These modifications, when completed, will greatly limit the number of systems which would become contaminated, as well as localize these highly radioactive areas to make shielding more effective and minimize personnel exposure. These modifications are as follows:

1. The Engineered Safeguards Room sump pump discharge lines will be modified to return any highly radioactive leakage back into containment rather than into the liquid radwaste system. This modification will be complete prior to start-up.

2. The various relief valve discharge lines in the Engineered Safeguards systems route any highly radioactive water to the Engineered Safeguards Room sumps.

3. The air line which connects the Engineered Safeguards (ESS) Room air compressors with the turbine building high-pressure air compressor will be modified to prevent any highly radioactive gases released in the ESS rooms from being transferred into the turbine building. This modification will be complete prior to start-up.

4. As discussed above, the post-accident sampling system is included in accordance with Recommendation 2.1.8.a. This system will include provisions to return sample effluents back into containment to prevent contamination of other systems, the capability to draw samples from Engineered Safeguards systems when the PCS is depressurized, and an appropriate location with adequate shielding to minimize personnel exposure. Per Recommendation 2.1.8.a, this modification is required prior to January 1, 1981. In the interim, provisions are being made in the existing system to return sample system effluent back into containment.

The second phase of the leak reduction program includes the inspection of the systems of concern to identify and measure existing leakage and to repair significant leaks as necessary. For the inspections, checklists were generated to list all major valves and other components within the system boundaries of concern; and each component is then individually inspected for leakage with the system pressurized. To date, the LPSI/Shutdown Cooling system has been inspected, and the total leak rate determined to be 1/2 gallon per minute. This water will go through an existing floor drain to the Engineered Safeguards Room sump and empty into the equipment drain as radwaste (normal operation) or back into containment (accident condition). The remaining inspections, with one exception, will be performed when system conditions permit prior to start-up.

All significant leakage from the systems of concern will be corrected prior to start-up. On a long-term basis, the systems of concern will continue to be inspected and/or tested at a frequency that will not exceed each refueling cycle.

The third aspect of the leak reduction program involves consideration of the October 17, 1979 letter concerning radioactive material spills and releases at North Anna and other sites. In addition to the above responses, these considerations are addressed as follows:

The gaseous radwaste is maintained as an ASME Class 3 system (Quality Group C), and as such is periodically pressure tested per ASME B&PV Code, Section XI. This testing adequately monitors the integrity of the system.

Because the Palisades Auxiliary Building is designed so that flooding will not occur from a Lake Michigan seiche or storm surge, the building also would very effectively keep major spills within. Even if the installed watertight doors were open, it would take a spill of more than 25,000 gallons to raise the water level in the building enough for the water to begin spilling over into the turbine building or outside. In addition, even a major liquid spill would not contribute significantly to the off-site radiation dose rates. Therefore, since the consequences of a liquid spill are small, and since plant design is such that spills would be contained, we consider that further action is not warranted.

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS (2.1.6.b)

Position

2.1.6.b With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids, are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Response to 2.1.6.b

Personnel Protection

Consumers Power Company is committed to initiate measures to adequately assure the protection of the health and safety of the plant work force during and after the postulated accident. Systems and areas outside the containment with large radiation sources after the postulated accident have been identified. These include the engineering safeguards room equipment, those penetrations assumed to fill with containment source terms, the emergency sampling system and the area in the shine path through the personnel air lock. Radiation dose calculations were performed using Regulatory Guide 1.4 source terms with appropriate delay times based on plant design. Considerations used in defining the systems which contain radioactivity are found in discussion of Section 2.1.6.a.

Operator actions which are assumed to be performed if warranted by accident conditions have been determined from a review of the Palisades Plant Emergency Plan, Emergency Implementation Procedures and the appropriate Emergency Operating Procedures. The identified areas that require access as a result of one or more types of accident are: The control room, shift supervisor's office, viewing gallery (outside control room), training center (east of main gate), parking lot outside main gate, service building conference rooms, security building, 1C switchgear room (next to diesel generators), both diesel generator rooms, 1D switchgear room (below the control room), main steam isolation valve room, C-33, C-40, motor control centers No 7 and No 8 room, primary system sample panel area (new), west engineered safeguards room, auxiliary bay roof east of the turbine building (625' elevation), auxiliary feed pump room, screenhouse room, north end of the east mezzanine in the turbine building (607' elevation), northeast corner of turbine building (590' elevation), condensate pump room (in turbine building), heater drain pump area (in turbine building), cable spreading room, valve pit below T-90, primary makeup transfer pumps and heat exchanger room, charging pump room, in the 602' pipeway above the boronometer, motor control center No 10 on the main turbine floor, south end of the turbine building at the 590' elevation at the MSIVs and the component cooling water room.

Of these 30 areas, only 6 are determined to have radiation levels immediately after the postulated accident which would unduly limit or preclude access. The areas include the emergency safeguards room, the 602 pipeways adjacent to the containment, the personnel air lock area, the component cooling room, the C-33/C-40/MCC 7 and 8 area, the purge line penetration area, and the main steam isolation valve room.

Modifications are in the process of being implemented to assure adequate operations capability. Operator actions are being reviewed to determine the time frame for required actions to allow more accurate dose estimates. Several options are available to keep personnel exposures as low as reasonably achievable. Some areas will require additional shielding or remote operating techniques. Procedure controls are being added to assure that areas are entered under controlled and monitored conditions for measured time periods. Access routes are being reviewed and, where necessary, auxiliary routes will be specified in the appropriate procedures. Refer to Section 2.1.8.a, sampling capabilities, for interim solution to access the personnel air lock area. Refer to the response in Section 2.1.8.b(2) for interim methods of obtaining high range effluent monitoring data from the area adjacent to the purge line penetration room.

Liquid and gas containing systems were taken under consideration in the response to 2.1.6.b. These systems were also discussed in Section 2.1.6.a.

Accumulated doses for personnel in a vital area will be within guidelines set by Appendix A of 10CFR50 General Design Criteria 19. The dose rate of areas requiring continuous occupancy during the course of an accident, the control room and On-Site Technical Support Center, were based on control room occupancy factors contained in the Standard Review Plan 6.4. Those areas requiring infrequent access, sufficient shielding will be provided to allow access at a frequency and duration estimate by Consumers Power Company.

Item 2.1.6.b Equipment Qualification at Palisades

The task involved determining which equipment would be required during a LOCA, its location, the dose it receives and the ability of the equipment to withstand the effects of the dose. The equipment needed was determined based on other safety-related equipment lists made up to respond to various SEP topics (safe shutdown, seismic and electrical equipment qualification). The location was determined through equipment location drawings, cable tray drawings and consultations with the plant. The dose was calculated using TMI source terms and assuming present plant design. Outside containment doses were primarily due to piping carrying potentially radioactive material. This piping was determined and also located on the above drawings. Doses to equipment were calculated by summing the contributions from various pipe lengths.

The safety-related equipment list totaled 500 pieces of equipment at the start. Scoping dose calculations were performed to determine the high dose areas. These areas are:

a. Throughout the Engineered Safeguards Room.

b. Within a 30' radius (except where intersected by an equivalent of 2' of concrete wall) of points along each of the source pipes penetrating the containment at El 602' and at 0° (north). The source pipes are air supply to the air room, containment spray, low-pressure safety injection and high-pressure safety injection.

c. Within a 15' radius (except where intersected by an equivalent of 2' of concrete wall) of points along each of the source pipes penetrating the containment at El 602' and 45°. The calculation of doses includes the termination of the source at the isolation valves. The source pipes include charging, letdown, primary coolant pump bleed off and containment vent header.

d. Within about 15' to 20' of the containment supply, air lines at El 625' and the containment exhaust air lines which leave at El 625' and have isolation valves at El 619' and 615'.

e. Between the penetrations and the equivalent of 2' of concrete wall.

f. In the vicinity of the boronometer.

Equipment in these areas is presented in the table. Throughout, an integrated dose of 10^4 (rads) was used as being the dose below which radiation effects are insignificant.

All of the sources available to Consumers Power Company (see references at end of memo entitled SEP Topic III-12, Environmental Qualification of Electrical Equipment, November 30, 1978) indicated that there were no materials that could not withstand 10^4 rads. Oil and grease, however, were not well covered. Special efforts will be made to determine ability to withstand the effects of irradiation for and assess the failure modes of the oils and greases used in safety-related equipment at Palisades.

Materials and components not capable of withstanding these doses have been identified during previous environmental qualification efforts (IE Information Notice 79-22 concerning the effects of nonsafety control systems on the results of high energy line breaks and SEP Topic III-12, "Environmental Qualification of Electrical Equipment") performed within the past 13 months. These components include seats, gaskets and discs on solenoid valves, diaphragms, seals on control valves, bearings, bearing lubrication and windings on motors, etc. The materials of construction of the equipment listed in Table 1 will be reviewed to see if any of these radiosensitive materials are present. The dose levels are such, however, that it is believed that not much will have to be replaced. Replacement materials, additional shielding or other actions to reduce doses to equipment will be installed by January 1, 1981, unless prohibited by hardware availability.

Equipment

1. Safeguards Rooms	Function	Integrated Dose Over 30 Days (Rads)
SV-3030A	Containment Sump Lines	4×10^4 to 1.2×10^6
SV-3030B	to SI Pumps	4×10^4 to 1.2×10^6
CV-3030		4×10^4 to 1.2×10^6
SV-3029A		4×10^4 to 1.2×10^6
SV-3029B		4×10^4 to 1.2×10^6
CV-3029		4×10^4 to 1.2×10^6
SV-3031A	SIRW Tank	4×10^4 to 1.2×10^6
SV-3031B	Outlet Valves	4×10^4 to 1.2×10^6
CV-3031		4×10^4 to 1.2×10^6
SV-3057A		4×10^4 to 1.2×10^6
SV-3057B		4×10^4 to 1.2×10^6
CV-3057	(In Emergency Procedures)	4×10^4 to 1.2×10^6
HPSI Pump P66A,B	Engineered Safeguards	4×10^4 to 1.2×10^6
HPSI Pump P66C	Pumps	4×10^4 to 1.2×10^6
LPSI Pump P67A		4×10^4 to 1.2×10^6
LPSI Pump P67B		4×10^4 to 1.2×10^6
Containment Spray Pump P54A,B,C		4×10^4 to 1.2×10^6
SV-3027A,B	Valves on SI Miniflow	4×10^4 to 1.2×10^6
CV-3027		4×10^4 to 1.2×10^6
SV-3056A,B		4×10^4 to 1.2×10^6
CV-3056		4×10^4 to 1.2×10^6
SV-3037	P66A Discharge (In Emer-	4×10^4 to 1.2×10^6
CV-3037	gency Procedures)	4×10^4 to 1.2×10^6
SV-3059	P66B, C Discharge	4×10^4 to 1.2×10^6
CV-3059	(In Emergency Procedures)	4×10^4 to 1.2×10^6
E/P-0306	Shutdown Cooling Flow	4×10^4 to 1.2×10^6
CV-3006	Regulation (In Emergency Procedures)	4×10^4 to 1.2×10^6
SV-3055A,B	SI Flow to Shutdown	4×10^4 to 1.2×10^6
CV-3055	Cooling Hx (In Emergency Procedures)	4×10^4 to 1.2×10^6
CV-3212	Inlet to Shutdown Cooling	4×10^4 to 1.2×10^6
CV-3223	Hx (In Emergency Procedures)	4×10^4 to 1.2×10^6

Safeguards Rooms	Function	Integrated Dose Over 30 Days (Rads)
CV-3213 CV-3224	Outlet of Shutdown Cooling Heat Exchanger to Primary Cooling System (In Procedures)	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6
SV-3070 CV-3070	Miniflow From SDC Outlet	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6
SV-3071 CV-3071	Miniflow Line From Outlet of SDC Hx to HPSI Pump	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6
CV-3036	Line From HPSI to SI Tank Test Line (In Emergency Procedures)	4×10^4 to 1.2×10^6
CV-3018	Line From P66B, C to SI Tank Test Line (In Emergency Procedures)	4×10^4 to 1.2×10^6
CV-3025	Isolation From Shutdown Cooling Heat Exchangers to LPSI Line (In Emergency Procedures)	4×10^4 to 1.2×10^6
E/P-0737,0736 CV-0736A CV-0737A	Auxiliary Feedwater Valves To Tie in With Main FW	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6
SV-0947,0948 CV-0947,48	Inlet to Engineered Safeguards Pump Cooling	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6
CV-0949,0913 SV-0949,0913	Service Water Backup to Engineered Safeguards Pump Cooling and SDC Hx	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6
SV-0951,50 CV-0951,50	Return to Service Water	4×10^4 to 1.2×10^6 4×10^4 to 1.2×10^6

Safeguards Rooms	Function	Integrated Dose Over 30 Days (Rads)
P72A,B;P73A,B	Engineered Safeguards East and West Room Sump Pump	4×10^4 to 1.2×10^6
V-27 A Through D	Safeguards Room	4×10^4 to 1.2×10^6
VHX-27 A Through D	Fans and Coolers	4×10^4 to 1.2×10^6
CV-3029	Containment Sump	4×10^4 to 1.2×10^6
CV-3030	Isolation Valves	4×10^4 to 1.2×10^6
SV-3030		4×10^4 to 1.2×10^6
SV-3029		4×10^4 to 1.2×10^6

2. Isolation valves and components affected by doses from lines carrying sources at El 590' and El 607'.

E/P-0212	Letdown Flow Downstream	10^4 to 10^6
CV-0212	of Containment	10^4 to 10^6
SV-3002	Containment Spray Lines	10^4 to 10^6
SV-3001	(In Emergency Procedures)	10^4 to 10^6
CV-3001		10^4 to 10^6
CV-3002		10^4 to 10^6
FE-0301	Containment Spray Flow	10^4 to 10^6
FE-0302	(In Procedures) Indicator	10^4 to 10^6
PT-1805&14	Containment Pressure	10^4 to 10^6
PS-1801&1A	Containment Pressure	10^4 to 10^6
PS-1803&3A	Containment Pressure	10^4 to 10^6
SV-1813,14	Containment Ventilation Isolation	10^4 to 10^6
CV-1813,14	Containment Ventilation Isolation	10^4 to 10^6
SV&CV-2009	Letdown Isolation Valve	10^4 to 10^6
SV&CV-0155	Makeup Water To Quench Tank Containment Isolation Valve	10^4 to 10^6
SV&CV-1045,44	Clean Radwaste Isolation	10^4 to 10^6
SV&CV-1038,36	Valve	10^4 to 10^6
SV&CV-1037		10^4 to 10^6
SV&CV-1065,64		10^4 to 10^6

Safeguards Rooms	Function	Integrated Dose Over 30 Days (Rads)
SV&CV-1001	Primary System Drain Tank Containment Isolation Valve	10^4 to 10^6
SV&CV-1002	Primary System Drain Tank Containment Isolation Valve	10^4 to 10^6
SV&CV-1004	Discharge To Clean Waste Receiver Tanks Containment Isolation Valve	10^4 to 10^6
SV&CV-1102	Containment Systems Vent Containment Isolation Valve	10^4 to 10^6
SV&CV-1210,11	Instrument Air Containment Isolation Valve	10^4 to 10^6
SV&CV-1503 SV&CV-1501,02	Heating System Containment Isolation Valve	10^4 to 10^6 10^4 to 10^6
SV&CV-1813,14	Containment Air Room Supply Containment Isolation Valve	10^4 to 10^6
CV-2014 SV-2014 AE-0203 RE-0202	Boronometer and Process Radiation Monitor in CVCS System	10^4 to 10^6 10^4 to 10^6 10^4 to 10^6 10^4 to 10^6

3. Isolation valves and components affected by doses from sources in pipes up to isolation valves El 607' to El 625'.

Steam Generator Code Relief Valves

CV-0779 Through 0782 POS-0779 Through 0782 E/P-0781,80 E/P-0779,82	Steam Dump Valves and Positioners	Between 10^4 and 10^6 Between 10^4 and 10^6 Between 10^4 and 10^6 Between 10^4 and 10^6
SV&CV-0824 SV&CV-0827	Service Water Isolation Valves	Between 10^4 and 10^6 Between 10^4 and 10^6

Safeguards Rooms	Function	Integrated Dose Over 30 Days (Rads)
P&ID M-218		
SV&CV-1803,05,06	Containment Air Exhaust Containment Isolation Valves	Between 10^4 and 10^6
SV&CV-1808,07	Containment Air Supply Containment Isolation Valves	Between 10^4 and 10^6

4.	Electrical penetrations at 0°, El 625' to El 631' and 235°, El 609' to El 621' and cables in cable tray in near vicinity of these penetrations.	10^4 to 10^7
5.	Containment air locks and equipment hatch (seals and air lock equalizing valves).	2×10^7
6.	Safety-related components located in the direct shine path from containment penetration blockouts between containment penetration and the first 12' equivalent thickness of concrete wall if not previously listed or if direct shine dose exceeds previously listed dose ranges.	Between 10^4 to 10^7

7. Cables Located:

a.	Throughout the Engineered Safeguards Room.	10^4 to 10^7
b.	Within a 30' radius (except where intersected by an equivalent of 2' of concrete wall) of points along each of the source pipes penetrating the containment at El 602' and at 0° (north). The source pipes are an air supply to the air room, containment spray, low-pressure safety injection and high-pressure safety injection.	10^4 to 10^7
c.	Within a 15' radius (except where intersected by an equivalent of 2' of concrete wall) of points along each of the source pipes penetrating the containment at El 602' and 45°. The calculation of doses includes the termination of the source at the isolation valves.	10^4 to 10^7

Safeguards Rooms	Function	Integrated Dose Over 30 Days (Rads)
	The source pipes include charging, letdown, primary coolant pump bleed off and containment vent header.	
d.	Within about 15' to 20' of the containment supply, air lines at El 625' and the containment exhaust air lines which leave at El 625' and have isolation valves at El 619' and 615'.	10^4 to 10^7
e.	Between the penetrations and the equivalent of 2' of concrete wall.	10^4 to 10^7
f.	In the vicinity of the boronometer.	

AUTO INITIATION OF THE AUXILIARY FEEDWATER SYSTEM (AFWS) (2.1.7.a)

Position

Consistent with satisfying the requirements of General Design Criterion 20 of Appendix A to 10 CFR Part 50 with respect to the timely initiation of the auxiliary feedwater system, the following requirements shall be implemented in the short term:

2.1.7.a(1) The design shall provide for the automatic initiation of the auxiliary feedwater system.

Response to 2.1.7.a(1)

Consumers Power considers the existing design adequate "with respect to timely initiation of the auxiliary feedwater system"; however, we will implement the NRC's imposed position.

The existing design for the Palisades Plant doesn't have the provision for automatic starting of turbine-driven auxiliary feedwater Pump P-8B or motor-driven auxiliary feedwater Pump P-8A. The modification of the AFW will provide this automatic actuation(20).

The turbine-driven and motor-driven auxiliary feedwater pump control circuits will be modified to provide automatic start on the tripping of both main feed pump drive Turbines K-7A and K-7B or low flow at the suction of the main feed pumps. Suitable output signals will be obtained from the main feed pump drive turbine trip logic and the existing flow transmitters to implement this modification.

The automatic initiation of the auxiliary feedwater system will be slightly time delayed to eliminate possible transients in the feedwater system. The motor-driven auxiliary feedwater pump will be started first, and if flow is not established within a preset time, the turbine-driven auxiliary feedwater pump will be started automatically.

Two automatic flow indicating controllers, one for each steam generator, will be installed to maintain constant flow rates for auxiliary feedwater to each steam generator. In addition to flow control, these indicators will also provide flow indication.

Additional flow transmitters will be installed to provide redundant auxiliary feedwater flow indication.

In case qualified devices cannot be procured for Phase I, the installation will be completed with control grade equipment, but Class 1E safety grade

(20) Automatic Auxiliary Feedwater System Drawing for Palisades, Bechtel Company, GWO 8428 Job 12477-039 (Attachment 11)

power supplies will be used. All control grade equipment will be replaced with qualified, safety grade equipment under Phase II.

2.1.7.a(2) The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

Response to 2.1.7.a(2)

The auto starting signal for the two auxiliary feedwater pumps can originate from either of two independent initiating signals. The signals are (1) all stop valves closed on both feed pump turbines and (2) low suction flow on both feedwater pumps. The failure of the Auxiliary Feedwater Auto-Start (AFAS) channel power supply does not result in the failure of the system; at least one pump will start. The initiating signal logic is failsafe; ie, a failure in the signal would start the auxiliary feed pumps.

2.1.7.a(3) Testability of the initiating signals and circuits shall be a feature of the design.

Response to 2.1.7.a(3)

Direct manual starting and stopping of the pumps is a design feature as is testing of the logic, beginning where the initiating signals enter the logic through and including starting of the pumps(21).

The initiating signals are powered from safety grade power supplies though the logic for starting comes for nonqualified equipment located in the secondary plant. This approach is consistent with control grade requirements of NUREG-0578.

Periodically during normal plant operation, the automatic start circuit will be tested to verify operability. The test will start the pumps and deliver water to the steam generators.

(21) Bechtel Drawing SK-JL-95, Rev C, dated 11-14-79, (Attachment 12).

2.1.7.a(4) The initiating signals and circuits shall be powered from the emergency buses.

Response to 2.1.7.a(4)

This requirement will be met prior to start-up. At the present, one system feed pump turbine stop valve indication is powered from d-c buses and the low flow power supply is off the instrument bus. To make the appropriate modification, the low flow power supply will be changed from the instrument bus to the safety bus.

2.1.7.a(5) Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

Response to 2.1.7.a(5)

This feature is incorporated into the new design of the AFW system. The operator will have the capability of overriding the start or stop action of either pump from the control room such that a single failure in the manual circuit will not result in a loss of system function.

Auxiliary feedwater system pumps and valves may be manually operated from the control room. The separation of manual and automatic initiation circuits insures that a single failure in the manual circuits does not impair the ability of the automatic circuits to actuate the AFW system.

2.1.7.a(6) The a-c motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads onto the emergency buses.

Response to 2.1.7.a(6)

The plant cannot meet this requirement under certain conditions, without jeopardizing the diesel generator. To comply to the extent possible while not compromising the diesels, the following logic has been used for the motor-driven pump. If either of the off-site power supply breakers are closed, the motor-driven pump will start automatically, regardless of whether the diesel is or is not running. If no off-site power is available, however, the auto start is blocked for the motor-driven pump. It can still be started by the operator when the diesel load permits.

The turbine-driven auxiliary feed pump is not inhibited by loss of off-site power. All support equipment is powered from the safety buses.

2.1.7.a(7) The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFWS from the control room.

Response to 2.1.7.a(7)

The separation of manual and automatic initiating circuits insures that the failure of the automatic function will not result in the loss of manual capability to operate AFW system pumps and valves from the control room.

In the long term, the automatic initiation signals and circuits shall be upgraded in accordance with safety-grade requirements.

This is incorporated into the total AFW system modification to achieve control grade prior to plant start-up and safety grade by January 1, 1981.

AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS (2.1.7.b)

Position

Consistent with satisfying the requirements set forth in GDC 13 to provide the capability in the control room to ascertain the actual performance of the AFWS when it is called to perform its intended function, the following requirements shall be implemented:

2.1.7.b(1) Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

Response to 2.1.7.b(1)

Prior to plant start-up a new safety grade flow transmitter supplied with safety grade power will be installed. The readout device, however, will not be safety grade. The existing control grade indication will be supplied with power from the vital instrument buses.

2.1.7.b(2) The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9.

Response to 2.1.7.b(2)

Existing auxiliary feedwater flow indication loops will be reconnected so as to be powered from safety grade Class 1E power sources.

The control grade indication meets the single failure criterion because with the installation of the proposed safety grade system two complete control grade indications powered from vital buses will exist for each generator.

The testability of the flow indication will be provided by actual system functional testing(22).

The safety grade indication will be installed by 1/1/81 and satisfy safety grade requirements.

Each auxiliary feedwater channel will provide an indication of feed flow with an accuracy of ±10%.

(22) Automatic Auxiliary Feedwater System Drawing for Palisades, Bechtel Company GWO 8418, Job 12477-039, (Attachment 11)

IMPROVED POST-ACCIDENT SAMPLING CAPABILITY (2.1.8.a)

Position

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and non-volatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly; i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift.

Response to 2.1.8.a

Sample piping does exist outside containment. However, plant systems and procedures do not currently provide for the sampling, handling and analysis of highly radioactive reactor coolant under Regulatory Guide 1.4 accident conditions.

Short-Term Solution

The Palisades Plant will modify the existing plant sampling so that a sample of pressurized and unpressurized reactor coolant and containment atmosphere can be obtained in less than one hour. On-site radiological and chemical analysis can be provided within one hour of obtaining the sample. Analytical techniques will be modified to include radiological precautions that will maintain the planned exposures to sampling and analysis personnel ALARA and less than 3 Rem whole body and 18.75 Rem extremity dose equivalent.

Draft procedures have been written and training shall be implemented prior to plant start-up to allow sampling and analysis of reactor coolant samples under severe radiation conditions.

The procedure is accomplished in three distinct stages to permit the spreading of accumulated dose equivalent among lab personnel. Entry is made to the sample room in stage one to draw a small coolant sample and transfer to the auxiliary building cold lab. Initial dose rate studies have indicated that the route normally used for entry to the sample room would be prohibitive during the first day of the accident. The use of additional shielding or an alternate route is being considered and will be resolved before plant start-up. In stage two, the sample is diluted in the cold lab and aliquots for boron and radioactivity analyses are transferred to the service building or other designated backup chemistry lab location. Chloride analysis will not be necessary since the letdown system will not be used due to an extremely high collection of activity (> 1×10^5 Ci/minute) on the demineralizers. There is no reason to expect unanticipated chloride concentrations and no method of controlling chlorides if the letdown system is not used. The boron analysis is performed in the backup chemistry lab and the radioactivity content is determined on the MCA/Ge(Li) in the feedwater purity building as stage three.

The ALARA provisions of the procedure include the following:

a. A presample survey of the cold lab to ensure that existing radiation levels are low enough to permit the performance of the sample dilution in this location.

b. Continuous radiation survey by the individual making entry to the sample room and drawing the sample into a shielded container with abort values specified for general area and contact dose rates based upon exposure to that individual and others who will subsequently have to handle the sample.

c. Local shielding and filtered exhaust hood at the sample dilution location.

d. Sample and analysis personnel equipped with high range direct readout whole body dosimetry and extremity TLDs. Respiratory protection equipment, thyroid blocking agents and anti-C clothing will be used as necessary for conditions.

Necessary equipment and materials for handling radioactive liquids and wastes will be available in the backup chemistry lab location.

The minimum training for lab personnel shall consist of a required review of the procedure prior to plant start-up. Each individual who may be called upon to draw the sample will perform a dry run of Stage One of the procedure. In addition, sample personnel will routinely use a radiac during sampling in order to develop a proficiency in survey techniques.

The modification of existing procedures shall be performed to allow the estimation of containment gas activities from the dose rate readings of installed area radiation monitors prior to implementation of the long-term solution.

Long-Term Solution

In order to meet the sampling and analysis criteria for a Regulatory Guide 1.4 accident, Consumers Power is proposing a combination of in-line and laboratory analyses for a long-term solution (1-1-81).

To avoid handling the high activity samples, a Ge(Li) detector (or intrinsic germanium) will be installed in a shielded glove box type arrangement to quantify and identify isotopes present in a small diameter tube. The detector and/or equipment will be set up to preferentially analyze a sample containing reactor coolant.

Blockage of the small diameter tubing in the reactor coolant line can be prevented by a series of cascading filters installed prior to entry to the tube. After analysis by the Ge(Li) system, the tube will enter a small shielded borated demineralizer to remove radioactivity. A sample point will be located after the demineralizer for obtaining dissolved gas and boron samples. To avoid a large buildup of activity in the demineralizer, reactor coolant will be recirculated prior to entry to the cascading filters. This will require a minimum amount of purging of the tube to obtain a representative sample and, consequently, less buildup in the demineralizer. Both the reactor coolant line and the containment air line will be routed back to containment to minimize the waste.

Figure 1, Page 5, Section 2.1.8.a, is a schematic of the proposed sampling and analysis system. Remote operated isolation valves which could be closed in case of a sample line break or leakage. The method of selecting either a pressurized coolant sample or a recirculation sample from the low-pressure safety injection discharge header are shown.

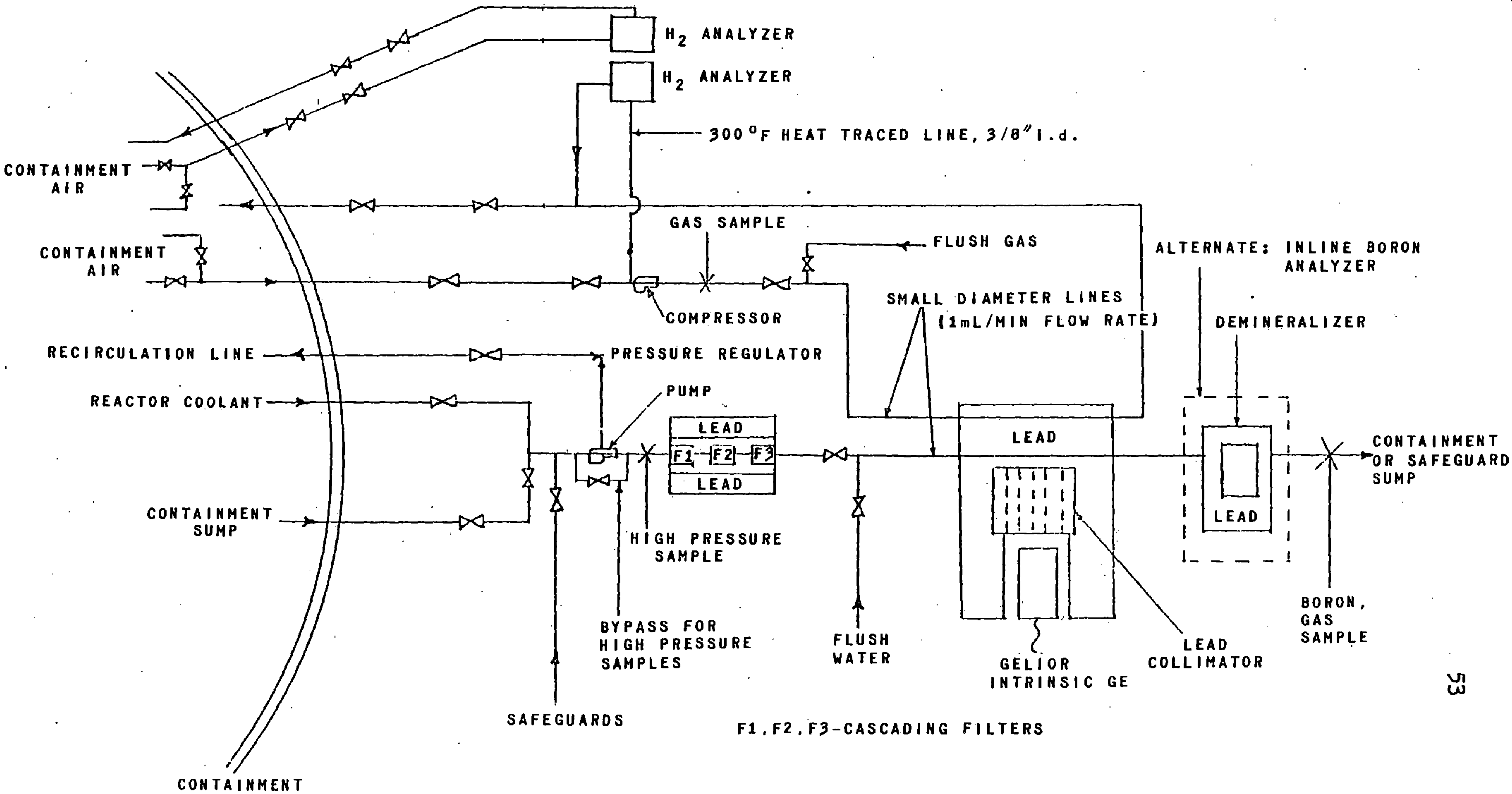
The proposed location of the Ge(Li) detector and demineralizer is the room used for existing sample panel. Access to this room under accident conditions to obtain the dissolved gas and boron sample will be performed according to an established procedure to assure that personnel exposures are ALARA.

Changes in design, equipment or location may be required as a result of other plant modifications. The intended purpose and results of the sampling and analysis system will not be compromised, however, as the result of any change. The sampling lines and components will conform to the classification of the system to which each sampling line is connected.

The analysis of hydrogen concentration in the containment atmosphere will be performed by an inline analyzer placed in the circulating loop of the containment air sampling line. More detailed design of the monitoring system will be provided by March 16, 1980 along with detailed designs for each of the other high range monitoring systems. System installations will be complete by 1-1-81.

FIGURE 1

POST - ACCIDENT SAMPLING AND ANALYSIS



INCREASED RANGE OF RADIATION MONITORS (2.1.8.b)

Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near-term.

2.1.8.b(1) Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions; multiple monitors are considered to be necessary to cover the ranges of interest.

a. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

b. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (ALARA) concentration to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

Response to 2.1.8.b(1)

1. Noble Gas Effluents - Interim Methods

Current stack monitor capabilities are limited to a release rate of approximately 1800 Ci per minute (30 Ci/sec) which represents three times the Regulatory Guide 1.4 MHA source term with leakage at a rate of 0.1% of gaseous inventory per day. The current monitor samples a stream which would contain leakage from containment penetrations into the auxiliary building, internal auxiliary building leakages and condenser offgas. The monitor does not sample atmospheric steam reliefs.

Interim methods to be used for quantifying noble gas releases at rates of up to 10,000 curies per second from stack and steam dump sources are being implemented as follows:

a. System and Method

(1) Portable instrumentation with readouts which are remote from the detection chamber are being dedicated to emergency use in quantifying high level releases. Dynamic range requirements of approximately 20 mR/h to 20 R/hr (net above background) have been determined for chosen sensor locations in order to quantify release rates from the level of current instruments and

procedures to the level of 10,000 Ci/s. Sensitivity of each sensor to the expected mix of gases, including Xenon-133 (81 KeV), will be accounted for in graphs for dose rate to release rate conversion. The graphs will provide conversion values as a function of time after shutdown such that the larger percentage contribution from Xe-133 with time will be acknowledged. Graphical displays will be available for use with plant procedures prior to plant start-up.

Calibration capability for portable instrumentation exists on site to a level of approximately 50 R/h. Calibrations above this level are not proposed for these instruments. However, certain other instruments which have ranges far in excess of this value have been calibrated using an NBS certified source at the University of Michigan. Calibrations of this sort are available to a level of several thousand R/h, but are not available on a routine basis.

(2) Monitor location for interim high level stack and steam dump release monitoring is on the auxiliary building roof and consists of two shielded probes with four inches of lead shielding on all sides for background reduction; one will be directed at the stacks and the second in the same area but shielded from the stacks. Main ventilation stack flow rate is instrumented with readout in the control room. Steam dump rates are fixed. Ventilation exhaust measurements will be adjusted as necesary for indicated flow rate changes, in addition to the time dependent conversion factors discussed in (1) above, in order to ensure measurements which are representative of stack emission rates.

Since atmospheric steam releases will not occur at significantly long times after shutdown, temporal variation in Xenon-133 contribution is not factored into instrument sensitivity considerations. With the exception of this difference, all the foregoing discussion on stack monitoring applies equally to our interim steam release monitoring.

(3) Radiation readings are accessed by a relatively short distance excursion (approximately 50 feet) from the control room, most of which distance is adequately shielded in the event of the extreme accident. A decision on preferred method of personnel protection for access to these readings through the limited areas of excessive dose rates has not been made (added shielding, construction of a new access route or extension of the remote readout capability). However, one of these alternatives will be implemented such that exposure rates will not exceed 100 mR/h dose rate average over duration of the accident at any area required for routine passage to, or readout of, these monitors. Actions necessary to assure these improvements will be made prior to plant start-up.

(4) The health physicist or his designated alternate will appoint a radiation protection technician to take readings at 15-minute intervals. The data obtained will be relayed from the radiation protection technician to the health physicist who will then inform the site emergency director or his designated alternate.

(5) Each of the interim monitors is battery powered. These monitors will be supplied with battey changes at regular intervals with batteries dedicated to emergency use and contained in the emergency kits. It is recognized that battery changes are not an ideal substitute for continued operation without battery change. To this end, Consumers Power Company is in the process of obtaining a number of instruments rated for 20-day operation between battery changes.

b. Description of Procedures

Procedures are now in draft format and awaiting decisions on specific alternatives to be used for access to monitoring locations and entry of the tabulated response conversion factors. All necessary decisions and procedure aids as described below will be completed prior to plant start-up. The procedures as now drafted or currently available plant approved procedures include specific instructions in each of the following areas:

(1) Minimization of personnel exposure.

(2) Calculational methods for determining release rates.

(3) Reporting of results.

(4) Instrument calibration.

Additional details as to procedures are discussed in the description of System and Method (Item 1.a. above).

2.1.8.b(2) Since iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by on-site laboratory analysis.

Response to 2.1.8.b(2)

Radioiodine and Particulate Effluents - Interim Methods

Shielding analysis has shown that the current stack monitor location would be inaccessible for recovery of particulate and iodine samples during the first five days after a Regulatory Guide 1.4 Maximum Hypothetical Accident. Furthermore, a release rate of 10,000 Ci/s noble gas, with Regulatory Guide 1.4 noble gas to iodine partitioning of 4 to 1 gives 2,500 Ci/s iodine, or 80 Ci of iodine plus 130 Ci of noble gas in the filter media after 15 minutes of sampling. Precautions are being added to emergency procedures which address the potentially high radiation levels associated with filter removal. A filter transfer shield and other specialized handling equipment will be utilized, as necessary, to minimize dose due to filter handling. In extreme cases, however, the current sampler may not allow safe filter removal. In such instances, a method described in our current emergency procedures for estimating thyroid dose would be substituted for stack filter data.

Current emergency plan procedures relate Xenon-133 and gross noble gas levels to expected conservative concentrations of I-131 for five types of accident which could cause significant off-site doses. The procedures relate stack monitor noble gas response to off-site thyroid dose described by thyroid dose isopleths for Pasquill meteorological categories A through F, respectively. The procedures also call for immediate verification of the estimated levels by air sampling and iodine cartridge analysis at the locations indicated by the isopleths as having maximum off-site ground level concentration. These current procedures will be modified to take into account the response of interim high level instruments in estimating iodine releases. Particulates are not a limiting dose consideration for this plant since HEPA, but not charcoal filtration, is present in the stack. Furthermore, the ratio of particulate activity to iodine activity indicates iodine remains limiting for accidents which do not involve release through the stack. Nevertheless, particulate filters will be sent immediately to our Big Rock Point nuclear facility for analysis within an expected time frame of four to five hours. Silver zeolite filters will be returned to the Palisades counting facility for Ge(Li) spectral analysis in order to refine the field instrument (portable multichannel NaI) measurements.

The criteria for required range to 1 x 10^5 µCi/cc of noble gas, coupled with the 4 to 1 ratio of noble gas to iodines, results in a requirement to handle charcoal sample cartridges at minimum sampler flow with as much as approximately 15,000 curies of adsorbed iodine plus 24,000 curies of entrained noble gas in a 15-minute sampling interval. With NRC credit for exhaust dilution (1 x 10^4 µCi/cc) these numbers decrease to 1500 curies iodine plus 2,400 curies noble gas - still much too hot for safe handling.

Only one commercially proposed sampling system (Science Applications/RadeCo) meets the requirement of automated (hands off) sampling and analysis which is required for personnel protection at these levels. This system handles noble gas to a maximum level of 1×10^4 μCi/cc. Automatic fresh air purge of the charcoal releases the noble gas. Automatic analysis of iodines is performed by an intrinsic germanium spectrometer.

We believe that the capability to analyze effluent release rates to 1×10^4 μCi/cc of noble gas will be sufficient for any type of accident hypothesized to date. Consequently, we intend to continue discussions toward the goal of having an installed system for stack monitoring by 1-1-81. The monitor will be installed in a low dose rate area to enable normal servicing during accident conditions. Preferred locations are the auxiliary bay roof, hallway outside the control room or turbine building on the 625' elevation level. Control room readout and off-site dose computation capability will be available with the instrument.

A separate instrument is proposed for determination of atmospheric steam release activity. The instrument is envisioned as a noble gas monitor similar to the interim system, but handwired to the control room. A small sample of condensed steam collected automatically during the release would be laboratory analyzed for radioiodine and particulates.

2.1.8.b(3) In-containment radiation level monitors with a maximum range of 10^8 rad/h shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be designed and qualified to function in an accident environment.

Response to 2.1.8.b(3)

High Range Containment Monitors

A monitor that is installed during a refueling outage located within containment provides a permanent location for the installation of a new environmentally qualified high range containment monitor by 1-1-81. We are proposing that this single environmentally qualified monitor be provided a backup by a second monitor located outside the containment adjacent to the equipment lock. The location external to containment offers the advantage of a much less severe environment with subsequently higher reliability under accident conditions. Range of both monitors will allow determination of containment gamma dose rates of 1×10^7 R/h.

IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS (2.1.8.c)

Position

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Response to 2.1.8.c

Airborne radioiodine monitoring with conventional charcoal canisters may produce overly conservative results under accident conditions due to interference from radioactive noble gases. Occupancy of various locations may be forbidden if airborne radioactive iodine levels are high.

a. Short-Term Solution

RadeCo Model GY-130 silver zeolite and/or equivalent (ie, silver gel) radioiodine sampling cartridges demonstrate an acceptable collection efficiency for inorganic as well as organic iodines (94-96% in the normal sampling flow rate range) but do not retain noble gases to any appreciable degree. Although these cartridges are too expensive for normal plant sampling, they are justifiable for emergency monitoring. Two hundred of these filters are being procured.

Respiratory protection is currently available and potassium iodide as a thyroid blocking agent will be available prior to start-up to reduce thyroid burdens of radioactive iodine.

b. Procedural Method

A 10-minute (10-20 cubic foot) air sample is passed through a combination particulate filter and silver zeolite cartridge holder at a flow rate of 1-2 cfm using the standard Palisades air sampler, the RadeCo Model H809V. The silver zeolite cartridge is then counted using a standard frisker (Eberline RM-14 or Ludlum 177) equipped with a pancake GM probe. Collection efficiency of 94% and detector efficiency of 25% yield a minimum detectable activity of 6.0E-10 μCi/ml for the smallest sample size (10 cubic feet), which is a factor of 10 below the I-131 MPC. Projected iodine levels in excess of 520 MPC hours (40 x 13) will require respiratory protection to be worn or potassium iodide to be prescribed.

Evaluation of the type of protection will be made on a case-by-case basis depending upon the individual action required in the airborne area. The Company's general medical consultant has developed a procedure for the use of potassium iodide in emergency situations. An initial draft of procedure has been submitted to the Company and will be implemented prior to start-up.

c. Equipment Impact - Short Term

1. Without reuse, a quantity of 200 silver zeolite and/or equivalent (ie, silver gel) cartridges currently on order is adequate to permit sampling of the Control Room/TSC and the Operations Support Center atmospheres every 15 minutes for the first 10 hours during an accident and then every 4 hours for the next 10 days. With reuse of cartridges which have insignificant accumulations of radioactivity, sampling can be expected well in excess of this minimum frequency.

2. Backup power supplies for air samples are necessary in the event of nonavailability of the primary source (115 VAC). Batteries are not practical for a continuous 7 days of operation. A purchase of 4 gasoline powered a-c generators has been initiated. These generators will be limited to outdoor areas near the required use centers with power transfer by temporary cables.

3. An adequate number (10) of RadeCo H809V air samplers are available at Palisades; a minimum of 2 samplers will be dedicated for emergency use.

4. Eight standard size (2" x 4" x 8" or 2" x 4" x 6") lead bricks will be stored in both the TSC and in the Operations Support Center such that they are available for the construction of counting shields for pancake GM probes in the event that background radiation levels preclude air sample cartridge counting without shielding. These lead bricks are presently available on site.

d. Training

1. The sample procedure is similar to the existing surveillance air sample procedure (HP 2.2) and utilizes existing air sampling equipment and counting instruments. Therefore, the minimum training for health physics personnel shall consist of a required review of the procedure prior to plant start-up. A group discussion on each shift led by the responsible assistant supervisor will be performed prior to plant start-up.

2. Hands-on training will be conducted for the portable gasoline powered generators upon receipt of this equipment. Each health physics technician will be given an opportunity to start a generator. The normal locations of gasoline supplies on site and safety precautions for operating combustion engines in proximity to inhabited spaces will be covered in this training.

e. Longer Term Actions

Orders are being placed for 10 additional a-c/battery operated friskers (Ludlum Model 177) to supplement the existing supply and ensure that adequate numbers of friskers will be available for contamination monitoring and air sample counting during an emergency.

The purchase of 5 battery operated ratemeters has been initiated to permit long-term (500 hours per battery change) monitoring and air sample counting in the field during emergencies.

Orders are being placed for several dual channel analyzers (Eberline SAM-2) to permit isolation of I-131 gamma energy when counting air sample cartridges.

TRANSIENT AND ACCIDENT ANALYSIS (2.1.9)

Position

NUREG 0578 Requirement 2.1.9: "Analysis of Design and Off-Normal Transients and Accidents."

a. Provide the analysis, emergency procedures, and training to substantially improve operator performance during a small break loss-of-coolant accident.

b. Provide the analysis, emergency procedures, and training needed to assure that the reactor operator can recognize and respond to conditions of inadequate core cooling.

c. Provide the analysis, emergency procedures, and training to substantially improve operator performance during transients and accidents, including events that are caused or worsened by inappropriate operator actions.

Response to 2.1.9

These analyses are being performed and emergency procedure guidelines are being developed as part of the Combustion Engineering Owners Group efforts. These efforts are being conducted in accordance with schedules established in conjunction with NRC Bulletins and Orders Task Force. Consumers Power Company is participating in this effort.

CONTAINMENT PRESSURE INDICATION

Position

A continuous indication of containment pressure should be provided in the control room. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and minus five psig for all containments.

Response

The present status of the containment pressure indications shows that two pressure indicators are powered by safety bus which leaves 2 indicators to be upgraded prior to 1-1-81.

CONTAINMENT WATER LEVEL INDICATION

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

Response

Palisades presently has continuous indication of containment water level. The two wide range instruments (0-120") cover the range from the bottom to approximately 5 feet above the top of the containment sump. These two indications (No 0358 and No 0359) are annunciated on hi-hi level. The present power supplies for these annunciators show that one of these indications is powered from a safety grade bus while the other is powered from a control grade bus. Therefore, the condition of utilizing emergency power is available. An operator will have containment water level indication while the plant is on either off-site or emergency power.

A wide range safety grade containment water level instrument will be installed by 1-1-81.

The narrow range containment water level instruments will meet the requirements of Regulatory Guide 1.89 (Qualification of Class 1E Equipment of Nuclear Power Plants).

Since the Palisades maximum water volume that can end up in containment is smaller than 600,000 gallons, Consumers Power Company will deviate from this requirement. This is taken under consideration by Bechtel in their design of the containment water level indication.

In the long term Consumers Power Company will address the requirements of Regulatory Guide 1.97, which is under development, as it applies to the containment water level indication.

CONTAINMENT HYDROGEN INDICATION

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Response

In containment and out of containment instrumentation setups will be reviewed for installation.

This indication will be installed by 1-1-81. In the long term Consumers Power Company will address the requirements of Regulatory Guide 1.97, which is under development, as it applies to the containment hydrogen indication.

REACTOR COOLANT SYSTEM VENTING

Position

Each applicant and licensee shall install reactor coolant system and reactor vessel head high point vents remotely operated from the control room. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50 General Design Criteria. In particular, these vents shall be safety grade, and shall satisfy the single failure criterion and the requirements of IEEE-279 in order to ensure a low probability of inadvertent actuation.

Each applicant and licensee shall provide the following information concerning the design and operation of these high point vents:

RCS Venting

(1) A description of the construction, location, size, and power supply for the vents along with results of analyses of loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should be demonstrated to be acceptable in accordance with the acceptance criteria of 10 CFR 50.46.

Response to RCS Venting (1)

The Reactor Coolant Gas Vent System is designed to be used to remotely vent gases from the reactor vessel head and pressurizer steam space during post-accident situations when large quantities of noncondensible gases may collect in these high points.

Small amounts of gas can be vented to the quench tank and thus not enter the containment atmosphere. Larger volumes will require venting to the containment - either through the blown quench tank rupture disc or directly - where the hydrogen concentration will be controlled by the containment hydrogen recombiners.

As shown in Figure 1(24) the head vent ties into an existing 3/4" reactor vessel vent and will be flanged to permit head removal for refueling. The pressurizer vent ties into an existing 3/4" line also shown in Figure 1(23).

(23) CEN-125 , "Input for Response to Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply System," December 1979, Pg 8-11 (Attachment 1)

(24) CEN-125 , "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply System," December 1979, Pg 8-3 (Attachment 1)

The design criteria for the vent system will also provide:

1. The system shall permit remote (control room) venting from the reactor vessel head vent or the pressurizer.

2. The vent flow rate capability shall be based upon the following considerations:

   a. The vent rate shall be sufficient to vent one half of the RCS volume in standard cubic feet in one hour.

   b. Coolant liquid loss through the vent should not exceed makeup capacity.

   c. The vent mass flow rate should not result in heat loss from the RCS in excess of the normal pressurizer heater capacity. (When only 1E powered heaters are available, the venting should not result in uncontrollable pressurizer pressure or level changes.)

3. A safety grade vent path meeting the same qualifications as were accepted for the RCS at time of licensing. The power operated valves shall be 1E powered with power removed during normal operation to minimize the possibility of inadvertent operation. Although not required by the NRC, redundant paths may be provided.

4. The design shall minimize modification to currently installed safety class equipment and piping which may be radioactive or require major reactor coolant system hydrostatic testing after installation.

5. Consistent with NRC requirements, the system will be designed to limit mass loss to less than a LOCA as defined in 10 CFR 50, Appendix A, and thus a separate analysis of inadvertent system operation or pipe breakage is not required.

6. A vent system operable following all design basis events except those requiring evacuation of the control room, and loss of all a-c power (plant blackout).

7. A vent system capable of venting directly to containment or to the quench tank.

8. A vent system designed to vent superheated steam, steam water mixtures, water, fission gases, helium, nitrogen, and hydrogen as high as 2500 psia and 700°F.

9. Control room position indication shall be provided for all power operated valves.

10. A system designed not to interfere with refueling maintenance actions.

In CE's analysis of venting the RCS, 100 scfm design capacity was used. This exceeds the NRC requirements which desire the capability of venting a gas volume of 1/2 the RCS in one hour.

The RCS vents are smaller than the size corresponding to definition of a LOCA (10 CFR 50, Appendix A); thus, not challenging to the ECCS.

The system is designed(25) to be controlled remotely from the main control room. All valves and instrumentation are powered from emergency power sources, and alternate sources are used as necessary to meet single failure criteria. Position indication (open/shut) is provided for all remotely operated valves and displayed in the control room. Pressure instrumentation is also provided to monitor system performance and displayed in the control room.

The vents will be seismically qualified and will be powered from the emergency bus.

(25) CEN-125, "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply System," December 1979, Pg 8-3 (Attachment 1)

(2) Analyses demonstrating that the direct venting of noncondensable gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment as described in 10 CFR Part 50.44, Regulatory Guide 1.7 (Rev. 1) (Control and Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident), and Standard Review Plant Section 6.2.5.

Response to RCS Venting (2)

These vents will be located such that they will be near the air coolers to provide a maximum circulation for the vented noncondensable gases. This circulation will disperse the gases around containment thus not violating combustible gas concentration limits in containment as described in 10 CFR, Part 50.44, Regulatory Guide 1.7 (Rev 1) (Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident) and Standard Review Plan Section 6.2.5. This location also provides extra cooling of the vented gas.

The reason for venting this noncondensable gas into containment is due to the fact if an accident occurred and substantial core damage occurred, venting the gases to the atmosphere would be totally unacceptable.

The reactor coolant gas vent system is not required to operate under power operation nonaccident conditions. To preclude inadvertent operation of this system, valves are electrically key locked shut with power disconnected during plant operation, with administrative controls applied to place the system in operation. Should inadvertent operation occur, a flow limiting orifice is provided to limit mass loss from the RCS to less than a single charging pump.

(3) Procedural guidelines for the operators' use of the vents. The information available to the operator for initiating or terminating vent usage shall be discussed.

<u>Response to RCS Venting (3)</u>

Procedures for the RCS venting will be issued after completion of the modification. The content of the procedures will define the conditions under which the vents will be used.

A procedure will be initiated by 1/1/81 to assure that sufficient liquid or steam can enter the steam generator U-tube region so that decay heat can be effectively removed from the reactor coolant.

SHIFT SUPERVISOR RESPONSIBILITIES (2.2.1.a)

Position

2.2.1.a(1) The highest level of corporate management of each licensee shall issue, and periodically reissue, a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

Response to 2.2.1.a(1)

The following is an example of a periodic reissue of a managment directive that emphasizes the primary management responsibility of the shift supervisor as issued by the Vice President of Nuclear Operations:

The on-duty shift supervisor has primary management responsibility for the safe operation of the plant under all conditions on his shift. He is the only person authorized to direct the activities of licensed operators. He is also the only person authorized to direct the performance of the licensed activities. He shall consider advice from the plant staff (including those senior to him), and the shift technical advisor, couple this with his own knowledge and experience, and then direct that the required actions to maintain safe operation of the plant be taken.

The shift supervisor shall provide direct command oversight of operations and perform Management review of ongoing operations in the plant that are important to safety. In performing these functions, the shift supervisor shall not become totally involved in any single operation to the extent that it detracts from his ability to overview plant status, nor should he normally involve himself in the manipulation of controls, but rather direct that others do it.

The shift supervisor's normal duty station is the control room/shift supervisor's office. During routine operations, the shift supervisor will make periodic rounds and inspections of the plant systems and equipment. Prior to his leaving the control room/shift supervisor's office, he shall inform the #1 control room operator to assume the command of the control room.

The shift supervisor may be relieved during emergency situations by another qualified shift supervisor as directed by the operations and maintenance superintendent or by the operations supervisor. A proper watch relief must be effected prior to his leaving the control room.

In certain rare instances the shift supervisor may leave the control room during emergency situations (such as on the back shifts when the auxiliary operators require his expertise on the scene) provided the shift technical advisor remains in the control room during his absence. Prior to his leaving the control room, he shall inform the #1 control room operator to assume the command of the control room.

2.2.1.a(2) Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilites, and authority shall be clearly specified.

Response to 2.2.1.a(2)

Palisades plant procedures will be reviewed prior to plant start-up for completeness and to assure that the responsibilities, duties and authority of the shift supervisor and control operators are properly defined.

The main responsibility and authority of the shift supervisor are to maintain the broadest perspective of operational conditions affecting the safety of the plant. (See Response 2.2.1.a(1).) The shift supervisor will not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

The basic scope of the shift supervisor is as follows:

The shift supervisor's normal duty station is the shift supervisor's office, although he may be anywhere in the plant where his attention is required. In abnormal or emergency situations, the shift supervisor must direct his attention to the actions necessary to insure reactor safety.

The duty shift supervisor will remain in charge of the control room and the licensed operators. He has authority to restrict control room access. No one shall relieve the shift supervisor unless he does possess a current Senior Reactor Operator's License for the Palisades Plant.

The Administrative Procedures, Section 4.0, titled "Operations Administration," for the Palisades Plant assure that the duties, responsibilities and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant Management personnel.

2.2.1.a(3) Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

Response to 2.2.1.a(3)

Training programs for the shift supervisors presently emphasize and will continue to emphasize and reinforce the responsibility for safe operation as well as the management function of the shift supervisor. This training program is done at the plant through the instruction of the Operations Superintendent. The continuous training consists of verbal one-on-one contact/communication between the shift supervisor and the Operations Superintendent.

2.2.1.a(4) The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

Response to 2.2.1.a(4)

The administrative duties of the shift supervisor are reviewed by the Vice President of Nuclear Operations on an annual basis. This review analyzes the administrative functions of the management responsibilities for assuring the safe operation of the plant. The functions that detract from or are subordinate to these responsibilities will be delegated to other operations personnel not on duty in the control room, to the extent feasible.

SHIFT TECHNICAL ADVISOR (SECTION 2.2.1.b)

Position

2.2.1.b Each licensee shall provide on-shift technical advisor to the shift supervisor. The shift technical advisor may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The Shift Technical Advisor shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the Shift Technical Advisors that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

Response to 2.2.1.b

Shift Technical Advisors (STAs) have been hired and will be on shift watch prior to plant start-up. The STAs are presently on day shift and will remain there to make better use of their capabilities until Palisades starts up from the present refueling outage. They have been provided seven weeks of training in plant design and layout, plant administrative and emergency procedures, response and analysis of the plant to transients and accidents and various other topics necessary for successful commencement of watchstanding duties prior to start-up. The Palisades STAs will receive one week of CE simulator training prior to start-up.

A continuing training program is being developed which will include response and analysis of the plant to transients and accidents, and the capabilities of instrumentation and controls in the control room.

The major functions of an STA are to perform an <u>engineering evaluation</u> of plant operations from a safety point of view as outlined below:

1. Operating history of the plant (equipment failures, design problems, operation errors, etc) and Licensing Event Reports from other plants of similar design. The STA will provide any engineering assistance and recommendations to correct any of these problems on an as needed basis or to avoid any reoccurence.

2. Plant conditions required for maintenance and testing. Engineering judgement provided by the STA will be supplied, if needed, to assure that any testing or maintenance activity will not jeopardize the safety of the plant or specific piece of equipment being tested or worked on.

3. The STA will review administrative procedures on an engineering evaluation basis as they apply to maintenance and testing of engineering safeguards equipment.

4. Adequacy of plant emergency procedures. These procedures will be reviewed by the STA to identify potential problem areas from an engineering point of view. Upon implementation of emergency procedures, the STA will be ready to give engineering assistance that may be needed by the shift supervisor.

   Consumers Power Company's (CPC) Topical Report No CPC-1A. "Consumers Power Company Quality Assurance Program Manual for Nuclear Power Plants" establishes the requirements for QA and technical evaluations for equipment procurement and plant operation quality assurance. Therefore, this will not be covered as an STA responsibility.

   STAs will also review valve lineups as directed by the shift supervisor.

SHIFT AND RELIEF TURNOVER PROCEDURES (2.2.1.c)

Position

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

2.2.1.c(1) A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console.

(what to check and criteria for acceptable status shall be included on the checklist);

c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

Response to 2.2.1.c(1)

The Shift Turnover Procedure will be revised to provide a checklist for the shift supervisor to assure that critical plant parameters and proper alignment of all systems essential to the prevention and migration of operational transients and accidents by a check of the control console, critical plant parameters are within allowable limits, and that identification of systems and components that are in a degraded mode of operation.

In addition, prior to assuming his shift:

Shift Supervisors

a. Will review entries in the Shift Supervisor's Logbook and Daily Orders Book since he was last on duty.

b. Will review the Control Operator's Shift Turnover Checklist.

The oncoming and offgoing supervisors will discuss verbally the condition of the plant and, in particular, any special work or tests in progress or to be carried into the next shift. The oncoming supervisor will assure himself that critical plant parameters are within allowable limits.

Control Operators

It is an inherent responsibility of each operator to stay on the job until he is relieved.

The oncoming Control Room Operator prior to assuming his shift duties shall:

a. Review his logbook for the period since he was last on shift.

b. Review and sign the Control Operator's shift turnover checklist.

c. Identify which systems and components are in a degraded mode of operation permitted by the Technical Specifications.

The oncoming and offgoing Control Room Operators will discuss verbally the condition of the plant.

2.2.1.c(2) Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

Response to 2.2.1.c(2)

Auxiliary Operators

It is an inherent responsibility of each operator to stay on the job until he is relieved.

Prior to assuming the shift, the Auxiliary Operator will review his logbook.

The Auxiliary Operator's Logbook (Feedwater Purity Logbook, Secondary Logbook, and Primary Logbook) contains a record of action associated with the respective side including equipment under test or maintenance. Such items as switching orders and process evolutions are recorded in the logbook.

Checklists are available for start-ups, shutdowns and for some routine operations when they are long or complex. In addition whenever equipment is returned to service following maintenance or testing, the equipment is checked for operability before relying on it for its intended function.

2.2.1.c(3) A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

Response 2.2.1.c(3)

Procedures were specifically designed to verify the operability of Technical Specifications limiting condition of operation. Associated equipment will be checked on a monthly interval following the present refueling outage and their check will provide the evaluation of the effectiveness of the shift and relief turnover procedure. See response to Item 2.2.1.b for additional methods of verifying procedure effectiveness.

CONTROL ROOM ACCESS (2.2.2.a)

Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

2.2.2.a(1) Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

Response to 2.2.2.a(1)

The Duty Shift Supervisor or, in his absence, the No 1 Control Operator is responsible for maintaining control of personnel entering the Control Room who are not assigned to the operating shift. This responsibility applies whether entry is for observing operations, conducting tests, performing maintenance or other reasons. The Duty Shift Supervisor or, in his absence, the No 1 Control Operator is authorized to refuse entry or to direct personnel to leave the Control Room if their presence interferes with operations or may compromise plant safety.

When the reactor is critical, no one has the right to violate the area in front of the control boards inside the brown strips on the Control Room floor without the permission of the Shift Supervisor or Control Operator, except operators on duty and the plant technical staff who give deference to the operator in maintaining safe plant operations. Controls are manipulated by or under the direction of licensed personnel (10 CFR 40.54(i)).

Nonplant personnel must have the permission of the Shift Supervisor or Control Operator to enter the Control Room. Visitors may be asked to leave if their presence is a distraction to safe operations or if they knowingly or unknowingly violate any procedure, regulation or instruction.

2.2.2.a(2) Develop and implement procedures that establish a clear line of authority and responsibility in the Control Room in the event of an emergency. The line of succession for the person in charge of the Control Room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for Plant Management personnel not in direct command of operations, including those who report to stations outside of the Control Room.

Response to 2.2.2.a(2)

The emergency procedures for Palisades establish a clear line of authority and responsibility in the Control Room in the event of an emergency.

Basically the duties are:

The Shift Supervisor on duty will assume all responsibilities of Site Emergency Director until relieved of these responsibilities by the Site Emergency Director. He will continue in charge of plant operations during the emergency and will insure the reactor is maintained in a safe condition as prescribed in appropriate sections of the Plant Procedures Manual.

ONSITE TECHNICAL SUPPORT CENTER (TSC) (2.2.2.b)

Position

Each operating nuclear power plant shall maintain an onsite technical support center separate from and in close proximity to the control room that has the capability to display and transmit plant status to those individuals who are knowledgeable of and responsible for engineering and management support of reactor operations in the event of an accident. The center shall be habitable to the same degree as the control room for postulated accident conditions. The licensee shall revise his emergency plans as necessary to incorporate the role and location of the technical support center. Records that pertain to the as-built conditions and layout of structures, systems and components shall be readily available to personnel in the TSC.

Response to 2.2.2.b

The on-site Technical Support Center (TSC) will be located in the present shift supervisors and control room Viewing Areas just outside of the Control Room. This total area is 1,000 ft^2 including the shift supervisor's office. One half of the center is habitable to the same degree (shielding and air supply) as the Control Room for postulated accidents. The other half has equivalent shielding but a different air supply. Key TSC personnel will be provided with a means for gaining entry to the Control Room.

Dedicated communications between the TSC, the Control Room, Emergency Operations Center, and the NRC are provided.

All necessary equipment to fulfill this requirement is ordered.

Radiation monitoring will be provided for both direct radiation and airborne radiation contaminants in this TSC and will provide warning if the radiation levels in the support center are reaching potentially dangerous levels. Action levels will be designated to define when protective measures should be taken. This will be completed by January 1, 1981.

To ensure access to Technical Data, Palisades TSC will be provided with a card reader and aperture cards, system descriptions, containment photographs and necessary plant drawings which would be necessary for assessment of an accident in the TSC. These drawings will be controlled and kept up to date by the Plant Document Control Section. The plans and procedures for Engineering/Management support and staffing of the TSC will be part of the Palisades Emergency Plan. The present schedule is to submit the emergency plans for NRC review the first week in January per requirements of Darrell Eisenhut's September 13, 1979 letter, Items 7 & 8, "Follow-Up Action and Results From the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident."

ONSITE OPERATIONAL SUPPORT CENTER (2.2.2.c)

Position

An area to be designated as the onsite operational support center shall be established. It shall be separate from the control room and shall be the place to which the operations support personnel will report in an emergency situation. Communications with the control room shall be provided. The emergency plan shall be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

Response to 2.2.2.c

The on-site operational support center will be completed by plant start-up. The center will be located in conference rooms #3 and #4 of the new administrative building where plant support personnel will report to await further work instructions, etc. Auxiliary operators on duty will assemble in a small separate room within the Control Room boundry. Communications with the Operational Support Center and Control Room have been upgraded.

The emergency plan will be revised to reflect the existence of the center and to establish the methods and lines of communication and Management.

Near Term Requirements for Improving Emergency Preparedness

(1) Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.

(2) Assure the implementation of the related recommendations of the Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned Task Force's recommendations on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.

(3) Determine that an emergency operations center for Federal, State and local personnel has been established with suitable communications to the plant and that upgrading of the facility in accordance with the Lessons Learned Task Force's recommendation for an in-plant Technical Support Center is under way.

(4) Assure that improved licensee off-site monitoring capabilities (including additional thermoluminescent dosimeters or the equivalent) have been provided for all sites.

(5) Assess the relationship of State/local plans to the licensees' and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of ten miles. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.

(6) Require test exercises of approved emergency plans (Federal, State, local and licensees), review plans for such exercises and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor start-up for new licensees. Exercises of State plans will be performed in conjunction with the concurrence reviews of the Office of State Programs. As a preliminary planning basis, assume that joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about ten per year, which would result in all sites being exercised once each five years. Revised planning guidance may result from the ongoing rulemaking.

Action To Be Taken at Palisades

1. The plant emergency plan is being upgraded to meet the requirements of Regulatory Guide 1.101 and will be submitted in draft form for NRC review by January 18, 1980.

2. Actions in response to NUREG 0578 requirements are discussed separately in the contents of this letter.

3. The Emergency Operations Center has been established at Consumers Power Company's South Haven Conference Center. Communications' links to the Technical Support Center were discussed in response to Requirement 2.2.2.b above. Alternate locations for the Emergency Operations Center are the State Police Post, South Haven, MI, and the Van Buren County Sheriff's Office, Paw Paw, MI.

4. Provisions for plume location and monitoring, including the use of isopleths, are described in response to NUREG 0578, Item 2.1.8.b, Response Section 2, "Interim Methods for Quantifying Radioiodine and Particulate Effluents." In addition, we have reviewed the advisability of additional environmental thermoluminescent dosimeter (TLD) stations with analytical backup from our central TLD Laboratory in Jackson, Michigan. We are proceeding with plans for installation of additional TLD monitoring stations in each overland sector. Two rings of dosimeters will be provided. The rings will be approximately one to two miles and four to five miles from the plant, respectively.

   Each station will be provided with a package of three emergency environmental TLDs. During emergency conditions, the top TLD will be removed from each location at daily intervals. A three-day supply of dosimeters will be maintained in a shielded off-site location to allow a new dosimeter to be placed at the bottom of the package.

5. State emergency plans have been completed and submitted to NRC Region III. Local emergency plans will be submitted by January 18, 1980. They will be reviewed against current criteria by July 30, 1980 and against the upgraded criteria of Regulatory Guide 1.101 by January 1, 1981.

6. Individual tests of plant and State emergency plans will be accomplished by July 30, 1980. A joint test exercise will be held within five years.