Wisconsin Electric POWER COMPANY
231 W. MICHIGAN, P.O. BOX 2046, MILWAUKEE, WI 53201

March 14, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Denton:

DOCKET NOS. 50-266 AND 50-301
IMPLEMENTATION OF NUREG-0578
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On December 31, 1979, we submitted a complete description which documented our current implementation and future commitments in response to NUREG-0578. As noted in our letter, we had met all the January 1, 1980 schedular requirements to the best of our knowledge and belief.

In a telephone conference call with us on February 25, 1980, and a subsequent meeting on March 4, 1980, at Point Beach Nuclear Plant, the NRC Lessons Learned Westinghouse Plant Review Team requested certain additional information and clarification of our December 31, 1979, response. Attached herewith is a supplemental information package which responds to the Staff requests. For each item, the additional discussion provided addresses those areas of further interest to the Staff.

Also attached is a revision to the description of our implementation of Item 2.1.8.b. This revision replaces in its entirety Section 2.1.8.b of the Addendum enclosed with our letter of December 31, 1979. The new material is a major editorial revision for purposes of clarity and includes some minor technical changes which were found to be required in the course of installing the new monitoring equipment.

Very truly yours,

C. W. Fay, Director
Nuclear Power Department

Enclosures

A039
S
1/1

8003190573

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS FOR THE PRESSURIZER HEATERS, POWER-OPERATED RELIEF VALVES AND BLOCK VALVES, AND PRESSURIZER LEVEL INDICATORS IN PWRs

Pressurizer Heat Power Supply

The revisions to the Emergency Operating Procedures ensure the proper use of the pressurizer heaters under accident conditions. The procedures which have been developed and modified include the use of the pressurizer heaters based on the particular procedure and the appropriate need to use the pressurizer heaters in that plant condition. They also consider the capability for using the heaters during a particular set of primary system conditions. Their use is, therefore, event triggered and not simply based on clock time into an event.

In the case of loss of AC, without an accompanying accident, the pressurizer heaters are turned on very early in the Subsequent Action portion of the procedure. This would probably be within ten minutes of the initiating loss of AC. The loading of the heaters to the bus is acceptable at this time since the diesel loading is relatively low and the pressurizer heaters will be needed for proper pressure control.

In the case of an accident, such as a loss of reactor coolant, the pressurizer heaters are used when the plant conditions allow their use to properly control pressure. The actual timing is, therefore, dependent on the break size. Restoration of the heaters is done during the Subsequent Action, after the termination of safety injection flow step. It is done at this time because this step requires a level of greater than 20% and increasing in the pressurizer which will ensure that the heaters are covered.

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

One-line diagrams of the instrument air systems for Point Beach Nuclear Plant, Units 1 and 2, are given in Figures 2.1.1-3 and 2.1.1-4, respectively. A description of the system operation and redundancy of supply (service air system backup) is described in Wisconsin Electric's December 31, 1979, submittal, page 2.1.1-3 and Table 2.1.1-2.

During normal operation, the unit's backup pressurized nitrogen gas bottle, provided for low temperature-overpressure protection during plant shutdown, is isolated from the air system. This prevents depletion of the nitrogen supply due to low air pressure or regulator leakage during normal operation. The redundancy of supply and the capability to manually unisolate the air system and to restart the compressors provides for sufficient operational capability during an accident condition.

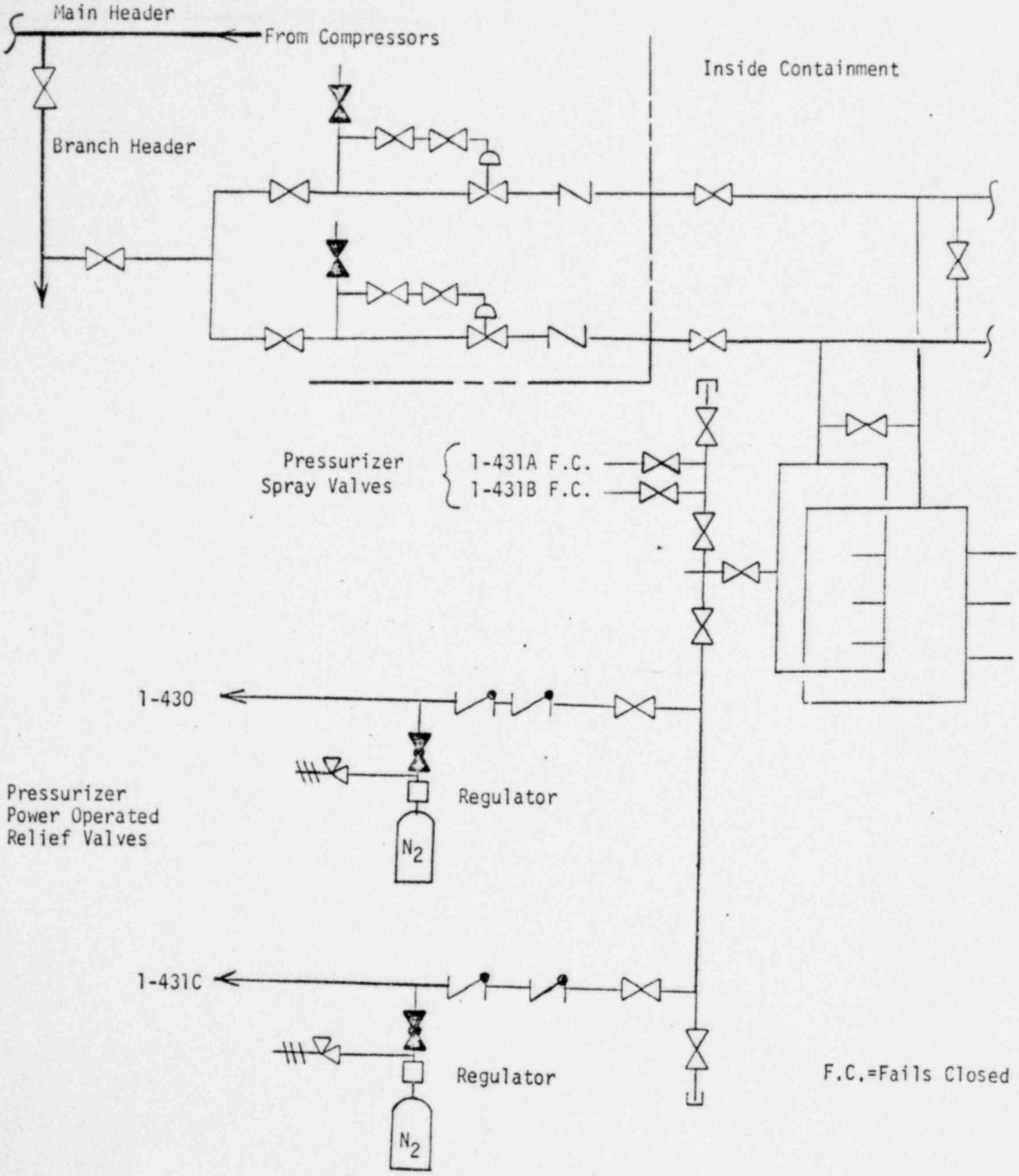


Figure 2.1.1-3 UNIT 1 INSTRUMENT AIR ONE-LINE DIAGRAM

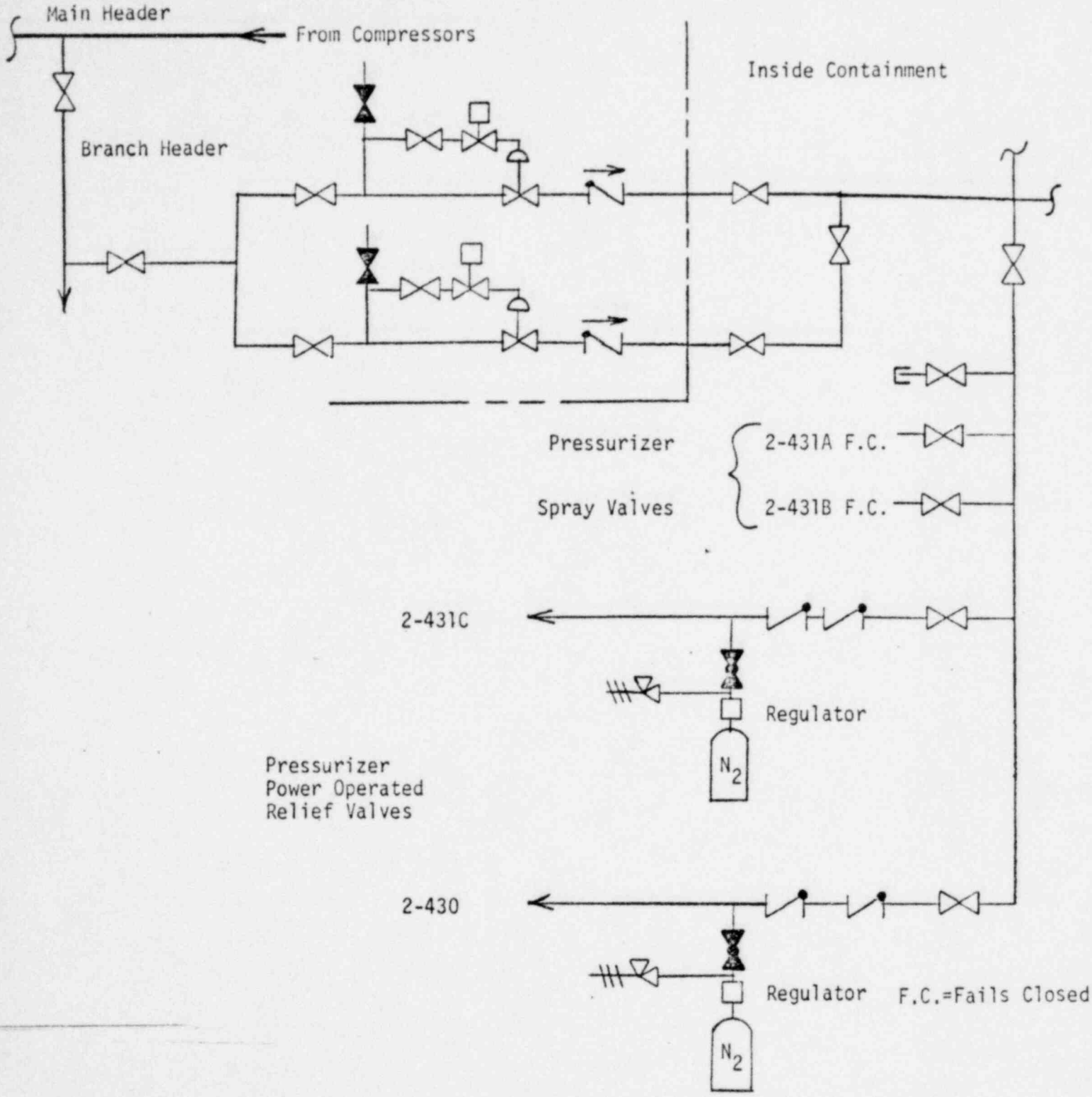


Figure 2.1.1-4 UNIT 2 INSTRUMENT AIR ONE-LINE DIAGRAM

2.1.3 INFORMATION TO AID OPERATORS IN ACCIDENT DIAGNOSIS AND CONTROL

2.1.3.a Direct Indication of Power-Operated Relief Valves and Safety Valve Position for PWRs and BWRs

Power-Operated Relief Valve Indication

As indicated in Figure 2.1.3-2, "Valve Position Alarm Circuitry", of the Wisconsin Electric December 31, 1979, submittal, the position indication circuit power supply for PCV-430 comes from Train "A" 125V DC power ("A" battery) and for PCV-431C comes from Train "B" 125V DC power ("B" battery) on Units 1 and 2. The alarm power for the annunciators also comes from the "A" battery via the red instrument bus.

Code Safety Valve Indication

Omitted from Figure 2.1.3-2 is the fact that the acoustic monitor electronics package receives power from the Unit 2 120V AC blue instrument bus. This bus is inverter-supplied through the "B" battery. A swing inverter is provided as backup.

A more complete description of the Point Beach Nuclear Plant electrical distribution was provided in response to Item 2.1.1 in the Wisconsin Electric December 31, 1979, submittal, and is given in Chapter 8 of the Point Beach Nuclear Plant Final Facility Description and Safety Analysis Report.

Power Operated Relief Valve and Safety Valve Backup Indication

The use of backup indication to provide confirmation of the primary indication is a basic foundation of the operator training program. It is reemphasized in the "Discussion" portion of EOP-1A, Large Loss of Reactor Coolant, and it is backed up by the alarm response procedures which give specific guidelines, if appropriate, for the alarms that would come in as backup to the primary indication, such as pressurizer relief tank indications and/or relief line temperature.

If, such as in EOP-3A, Steam Generator Tube Rupture, the use of the power operated relief valve is part of the procedure, specific guidance is given in the procedure to alert the operator to look at the pressurizer relief tank level, temperature and pressure and to isolate the power operated relief valve by use of the power operated relief valve isolation valve if conditions warrant.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

PBNP Methods Used to Determine Reactor Coolant Systems Subcooling

PBNP control room personnel presently have three methods of determining the subcooled state of the reactor coolant system. The amount of subcooling is defined here as the difference between the measured reactor coolant system temperature and calculated saturation temperature at the measured reactor coolant system pressure. The three methods used to determine the amount of subcooling are:

1. Computer

   PBNP is equipped with an on-line data logging computer (Westinghouse Prodac P-250) for each unit. Each computer contains a program that calculates the amount of subcooling and alarms at a margin of $\leq$50°F of subcooling. Saturation temperature is calculated using the average of four narrow range (1700 psia to 2500 psia) pressurizer pressure signals (U0482) when in range as input to steam table routines. The average thermocouple temperature (U0091), based on all available thermocouples, is used as the reactor coolant system temperature. When out of the narrow range, the one wide range pressure signal (0 psia to 3,000 psia) is used. In addition, both hot leg RTD temperatures will be made available for use when equipment has been received.

   The amount of subcooling is normally continuously trended on a control board recorder when the unit is not in a cold shutdown condition. Alternately, it may be put on continuous digital display.

2. X-Y Recorder

   A dedicated X-Y recorder is set up in the control room as a subcooling monitor in case of computer failure. The X-Y recorder can be made operational for either unit within two minutes of not being able to initialize a failed computer.

   The signals feeding the X-Y recorder are reactor coolant system wide range pressure (0 psia to 3,000 psia) and hot leg temperature from one loop. It was demonstrated during natural circulation tests that the hot leg RTD and thermocouples respond similarly during natural circulation.

   The 36°F subcooled line is pre-drawn on the temperature versus pressure graph placed on the X-Y recorder. When the X-Y recorder is activated, the displacement of the pen from the 36°F subcooled line is the indication of the amount of subcooling.

The control room operators use REI 17.0 (Monitoring Subcooling) procedure for this method. This has previously been verified by a Region III, NRC I&E inspection.

3. Hand Calculations

If both the P-250 computer and the X-Y recorder setup are unavailable for use, the control room operators can graph reactor coolant system temperature versus reactor coolant system pressure by hand using the reactor coolant system temperature and pressure signals available at the time. This would be either the hot leg RTD described above, the manifold RTDs if pumped flow is available, or the manual reading of selected incore thermocouples. The 36°F subcooled line pre-drawn on the graph will allow the operator to determine the amount of subcooling.

The procedure used is REI 17.0, Monitoring Subcooling.

Equipment Used and Power Supply

The qualifications and reliability of the incore thermocouples and computer has been previously documented. The hot leg RTDs have the same qualification as the reactor protection grade manifold RTDs. The single wide range pressure transmitter is a Foxboro Model 611 GM-MC5 while the narrow range pressure transmitters, used for the protection system inputs, are Foxboro Model 611 GM-DS1; the only difference between these models as purchased being the range. A modification request has been issued to put in three upgraded wide range pressure transmitters. At the present time, the delivery time is not known.

The reactor coolant system wide range pressure indication circuit detects coolant loop pressure and records it on the main control board (channel 420). It is treated as a control grade indication circuit and is powered from the yellow instrument bus.

The power for the Unit 1 computer is supplied by 2XY01 from 480V bus 2B31. The power for the Unit 2 computer is supplied from 1XY01 from 480V bus 1B31. The B31 buses are powered from B03 which is supplied by 4160V bus A05. The safeguards bus, A05, is supplied by the emergency diesels during a loss of AC. Some loads are stripped including B31 nonsafeguards loads.

These can be recovered after safety injection is reset and the nonsafeguards lockout is reset. So the recovery of power to the computer (subcooling meter) on the unaffected unit can be accomplished within an hour. The initialization process can be accomplished within another hour.

During the time the computer is not available, the X-Y plotter is used. The instruments driving the plotter are from instrument power supply and the recorder is plugged into instrument power.

No signal to any subcooling indication, the P-250, X-Y plotter, or manual, is a reactor protection signal. Each is approximately isolated by an isolation amplifier or computer block.

Vessel Level System

The reactor vessel level system design and description has changed in some minor concepts since our December 31, 1979, submittal. The system, as currently configured and being evaluated, is schematically shown in the attached replacement Figure 2.1.3-3. The level system is still based on measuring differential pressure across the total height of the reactor vessel. The current design would utilize one lower guide tube penetration for both level channels and a sealed leg to each d/p instrument system. The upper penetration has been disassociated from the reactor coolant system gas vent penetration and would connect to one spare control rod drive mechanism penetration for both level channels. A sealed reference leg would connect to the upper penetration and each d/p instrument system. Parallel instruments in each channel would provide for wide and narrow range indication during conditions where reactor coolant pumps are running or stopped, respectively. If necessary, the temperatures of vertical tubing runs would be monitored to provide a basis for compensation. Design work is continuing and will provide the detailed design configuration, equipment requirements, and plant layout.

The previously stated schedule for implementation is being followed which requires that the NRC provide a proposal review and approval prior to any equipment being ordered.

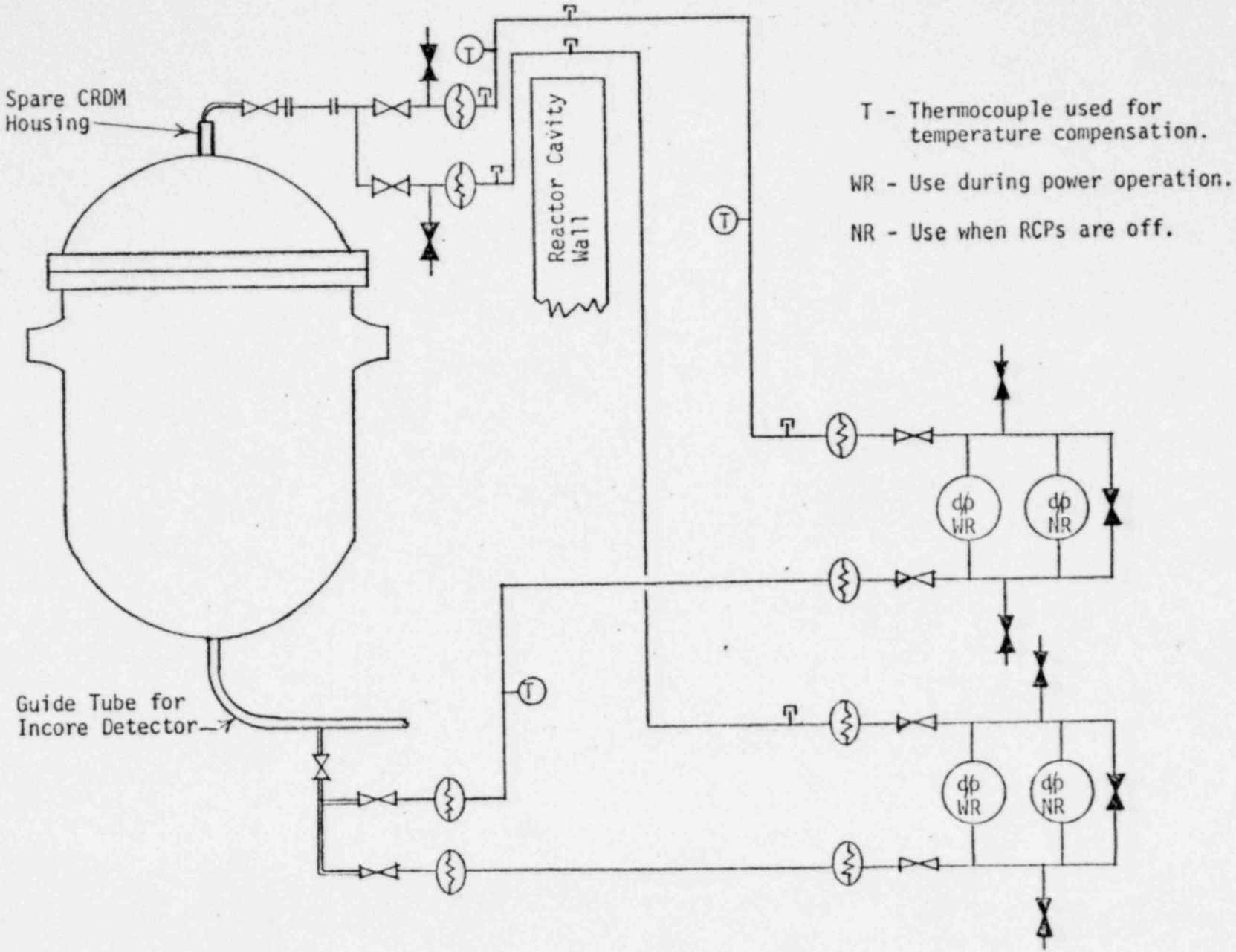


FIGURE 2.1.3-3 REACTOR VESSEL LEVEL INSTRUMENTATION

2.1.4 CONTAINMENT ISOLATION PROVISIONS FOR PWRs AND BWRs

Containment isolation for the Point Beach Nuclear Plant is described in our December 31, 1979, submittal as verbally corrected. The correction on page 2.1.4-1 is to the third paragraph of the "Discussion" where the third sentence should read as follows:

> "This occurs under the following conditions: High Containment Pressure, Pressurizer Low Pressure, and Low Steam Line Pressure in either steam line."

In clarification of the classifications used in Table 2.1.4-1, the penetrations in the Desirable (D) category are a sub-set of Non-Essential (NE) and are all closed in the post-LOCA condition. The resetting of Safety Injection, Containment Ventilation Isolation, or Containment Isolation will not automatically open any of the closed paths to or from containment as previously noted. Modifications are in progress which assure that this now applies to the outboard containment purge exhaust isolation valves (CV 3212 outboard of penetration #V1), which were identified as capable of reopening after isolation when in a remote operation mode not normally used. The remote operation mode capability will be removed from these valves as soon as possible. In the interim, the valves have been tagged to remain closed for operation above 200°F. Precautions in Procedure OP-9C have provided this administrative protection in the past and have proven to be sufficient to prevent inadvertent valve operation.

2.1.5 POST-ACCIDENT HYDROGEN CONTROL SYSTEMS FOR PWR AND BWR CONTAINMENTS

2.1.5.a Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems

The post-accident containment vent system (PACVS) described in our December 31, 1979, submittal is shown in detail in Figure 2.1.5-1. This system is safety grade and meets the single-failure proof criterion (NRC Clarification Item 2) for containment isolation as designed. To meet the single-failure proof criterion for operation which was additionably imposed in Item 2, the system can be considered as providing two independent penetrations, each of which is capable of containment pressurization and venting. If the normal function is impaired, both functions can be achieved sequentially using only one penetration with some post-accident modification. Alternately, the containment atmosphere post-accident sampling system as modified (see Figure 2.1.8-8 of our December 31, 1979, submittal) can also provide limited containment pressurization and venting to backup this capability. This system provides the normal capability for containment venting per Procedure OP-9C.

The PACVS system configuration and location of equipment has been verified in the plant. No problem is expected in accessing the exhaust and sample locations. In the event of an accident, where the recirculation mode of operation with high pressure safety injection is required, access to the pressurization connections may be limited. This problem can be eliminated by making the service air connection prior to the initiation of recirculation.

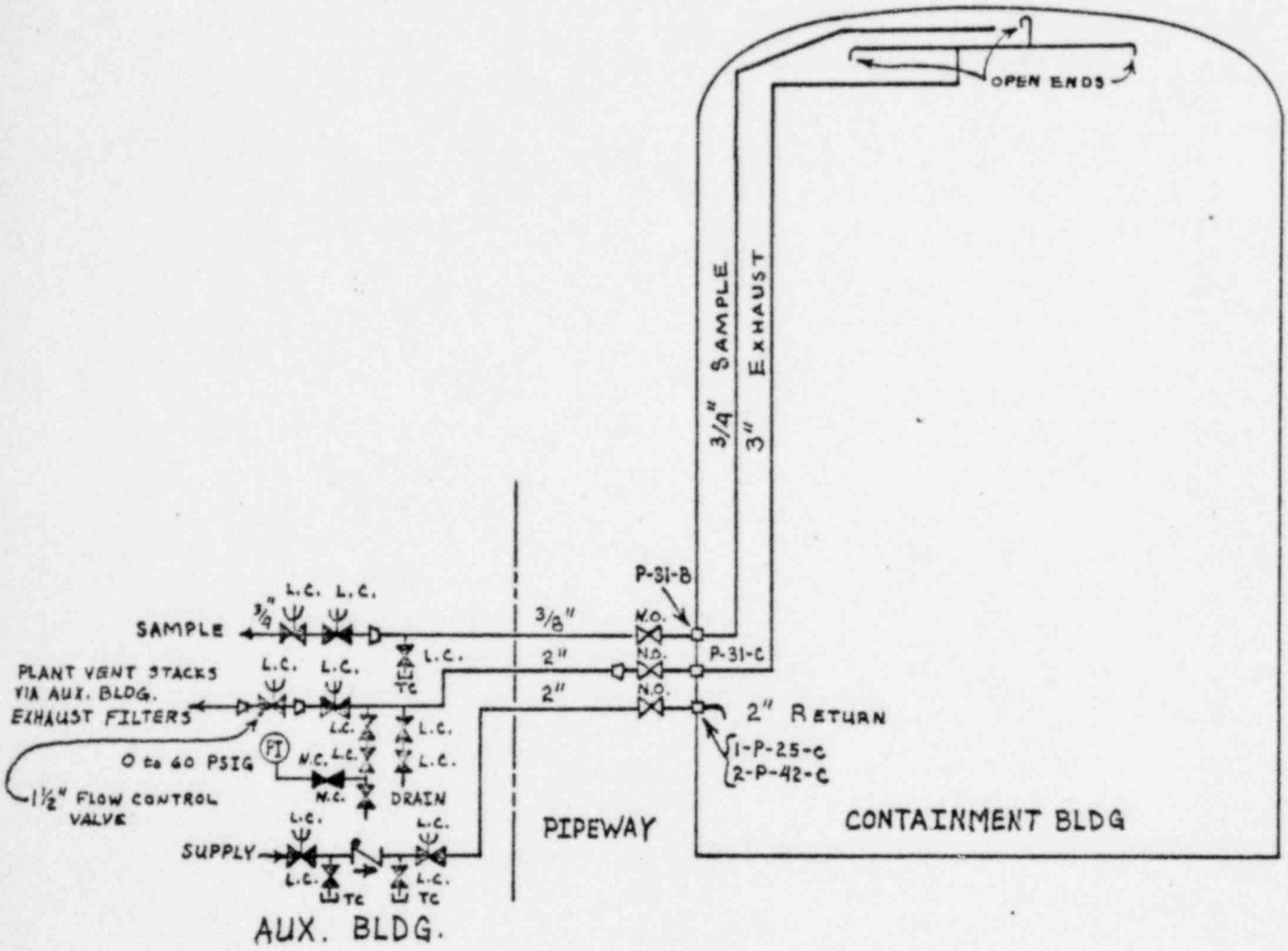


FIGURE 2.1.5-1 POINT BEACH POST-ACCIDENT CONTAINMENT VENT FOR EACH UNIT

NOTE:
NO = NORMALLY OPEN
NC = NORMALLY CLOSED
LC = LOCKED CLOSED
TC = TEST CONNECTION
= GATE VALVE
= DIAPHRAGM VALVE
= CHECK VALVE

2.1.6.a INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS

The basic criterion used in our December 31, 1979, submittal was to initially test all systems that might receive reactor coolant during either a small or large loss of coolant accident.

This approach was chosen because it involved the greatest number of auxiliary systems. The guidance and examples given in NUREG-0578 and in the NRC October 30, 1979, letter were also used to ensure that no systems were overlooked. Excluded systems have been evaluated to ensure that this exclusion will not preclude any option for providing adequate core cooling or for using needed safety systems. The systems chosen for review are basically in two categories:

1. Those which are used to ensure core cooling and containment depressurization:

   a. High head safety injection system;

   b. Low head safety injection system;

   c. Containment spray system;

   d. Cross-connection piping between the low head safety injection system and both the high head safety injection system and the containment spray system;

   e. Liquid sampling system;

   f. Containment sampling system; and

   g. Post-accident containment ventilation system.

2. Those auxiliary systems that might be used at present to aid in removal of gas from the reactor coolant system and in possible longer term areas of operations:

   a. Chemical and volume control system, letdown and charging;

   b. Gaseous systems associated with letdown including the gas strippers, the volume control tank gas space, the waste gas ventilation header, and the waste gas compressors; and

   c. The liquid waste piping to the waste holdup tank.

All of the systems in the above two categories were leak tested. A summary of the as-found leakage is presented in the Tables 2.6.1.a-1 thru 3. Detailed procedures are now being developed to implement a continuing system of leak checks for these systems and an evaluation of these systems in the following manner:

1. Systems in the first category will be leak-checked once per year. The leak-check will include an evaluation of leakage through valves to other areas or systems to ensure that all areas of possible leakage are included. Limiting leakage criteria are being developed using the source terms of 2.1.6.b to ensure proper preventive maintenance is performed as appropriate.

2. The systems in the second category will be leak-checked once per year for leakage to atmosphere as an ALARA type of test. The initial gas system leak testing was performed in 1979 by SAI, Inc. Future testing may be done in-house or may continue to be performed by an outside contractor. Testing of systems in the second category for high source term criteria is not needed for the following reasons:

   a. The letdown and charging system and associated gas systems will not be used for the purpose of removal of gas from the reactor coolant system as soon as the reactor vessel head vent system is installed. Prior to the installation of the head vent system, gases will be removed through the PORV, or, if activity is low enough, via the letdown path of the CVCS or the pressurizer steam space sample line.

   b. The liquid waste system will not receive any significant amount of waste from other systems (since they have been tested also) for the first several days following an accident. With the normal decay of the source terms, a leak-check to atmosphere is all that is necessary. The as-found leakage in the second category which is evaluated as significant will be repaired as a part of a continuing ALARA program at Point Beach.

   The implementation of this program will be a part of the Point Beach Nuclear Plant surveillance program and, as such, all data obtained will be processed for proper review and retrieval.

   The use of radiation monitoring information is an inherent part of the ALARA program at Point Beach Nuclear Plant and will continue to be used to help identify leaks throughout the Plant.

2.1.6.b DESIGN REVIEW OF PLANT SHIELDING OF SPACES FOR POST-ACCIDENT OPERATIONS

Systems Reviewed

As described in the attachment to our December 31, 1979, letter, identification of appropriate systems for consideration was based on a determination of the required use of those systems during and immediately following a postulated accident. Accordingly, a detailed study was performed on the following systems:

1. High head safety injection system;
2. Low head safety injection system (RHR);
3. Containment spray system; and
4. Liquid sampling system.

As explained in our December 31, 1979, submittal, the chemical and volume control system (CVCS) and the gaseous radwaste system were not included since these systems would not be used in a major accident.

A single exception to this categorization exists for the liquid sampling system, since sample flushing liquid discharges to the CVCS volume control tank (VCT). Accordingly, the volume control tank has been reviewed in addition to the four main identified systems. It has been determined that the relatively small amount of sampling liquids would result in a moderate increase in radiation levels in the area of the volume control tank. Existing plant shielding in this area is adequate, and no further modifications are needed.

For all systems analyzed, in-plant identification of pertinent components and lines were performed to verify as-built conditions.

Modifications Planned

It has been determined that shielding in the area of the C-59 control panel in the auxiliary building may be desirable. Accordingly, a design evaluation of the shielding requirements for operation, convenience, or ALARA considerations is in progress. Shielding modifications for the control panel will be implemented by January 1, 1981 to fulfill the requirements of NUREG-0578.

Environmental Qualification

Identification of required systems, a review of system layout and equipment, and a review of system operational requirements have been completed. The calculation of radiation dose rates following a postulated accident in the area of required system components has been also completed. The identification of

equipment consi[illegible] was based on a review of those systems required to oper[illegible]e either in the recirculation mode of the emergency core cooling systems or the residual heat removal mode of the auxiliary cooling system. The systems evaluated and physically traced included the high-head safety injection, containment spray, residual heat removal, and component cooling systems. The equipment and components of those systems were evaluated for degradation due to radiation fields only if they were located in areas where they could be exposed to radiation from the recirculated radioactive fluid and were required to operate in the modes of operation discussed above. The radiation fields in each of these areas were calculated based on the assumptions for post-accident release of core radioactivity required by NUREG-0578, Item 2.1.6.b.

A preliminary review of the components and integrated radiation doses indicates no equipment should fail due to radiation exposure within thirty days following the postulated accident. The final review of equipment and associated radiation exposure will determine specific modifications, if any, to be implemented by January 1, 1981.

TABLE 2.1.6.a-1

TABULATED LEAKAGE FOR SYSTEMS IN CATEGORY 1

SYSTEM (UNIT 1)	MEASURED LEAK RATE (ml/min)
Reactor coolant hot leg sample stop valve (1-971) through leakage	21*
Safety injection pump "A" (P-15A) pump seal	0.5
Safety injection pump "B" discharge strip valve (1-8668) packing	1
Residual heat removal pump "A" (P-10A) pump seal	33.3**
Residual heat removal pump "B" (P-10B) pump seal	1

SYSTEM (UNIT 2 AND COMMON)	MEASURED LEAK RATE (ml/min)
Reactor coolant hot leg sample stop valve (2-971) through leakage	37*

* Scheduled for repair.
** Acceptable because of isolated location.

TABLE 2.1.6.a-2

TABULATED LEAKAGE FOR LIQUID SYSTEMS IN CATEGORY 2

SYSTEM (UNIT 1)	MEASURED LEAK RATE (ml/min)
Charging pump "A" (P-2A) pump seals	40
Charging pump "B" (P-2B) pump seals	90
Charging pump "C" (P-2C) pump seals	10
"A" charging pump discharge valve (CV-286) packing leak	0.2

SYSTEM (UNIT 2 AND COMMON)	MEASURED LEAK RATE (ml/min)
Charging pump "A" (P-2A) pump seals	30
Charging pump "B" (P-2B) pump seals	80
Charging pump "C" (P-2C) pump seals	50

TABLE 2.1.6.a-3

TABULATED LEAKAGE FOR GASEOUS SYSTEMS IN CATEGORY 2

ITEM/SYSTEM	MEASURED LEAK RATE (cfm)
Waste Gas Compressor K1A	
(1) Tap off for temperature indication (TI-1029)	1.0×10^{-5}
(2) Float chamber head flange	2.3×10^{-5}
(3) Float chamber inlet flange	1.3×10^{-5}
(4) Receiver bottom flange	3.3×10^{-5}
(5) Level controller outlet flange	6.7×10^{-6}
(6) Float chamber outlet flange	1.5×10^{-5}
(7) Outlet flange plastic patch	1.7×10^{-4}
(8) Inlet flange	1.6×10^{-4}
(9) Inlet check valve	1.8×10^{-4}
(10) Pressure control valve (PCV-1028) downstream flange	1.1×10^{-4}
(11) Pressure control valve (PCV-1050) body-to-bonnet and flange	1.6×10^{-4}
TOTAL Waste Gas Compressor K1A	8.5×10^{-4}
Waste Gas Compressor K1B	
(12) Pressure control valve (PCV-1035) upstream flange	1.3×10^{-5}
(13) Air receiver drain valve upstream flange	6.7×10^{-6}
(14) Inlet flange	2.5×10^{-5}
(15) Inlet check valve	1.7×10^{-5}
(16) Pressure control valve (PCV-1051) upstream flange	2.0×10^{-5}
(17) Check valve downstream of PCV-1035	1.7×10^{-5}
TOTAL Waste Gas Compressor K1B	9.9×10^{-5}
TOTAL Waste Gas Compressor Area	9.5×10^{-4}

TABLE 2.1.6.a-3 - CONTINUED

ITEM/SYSTEM	MEASURED LEAK RATE (cfm)
Holdup Tanks	
(18) "C" tank reuse header isolation valve (1264) stem leak	2.7×10^{-3}
(19) Pressure transmitter P1-155 cutout valve (1268) union leak	1.3×10^{-4}
(20) Tee fitting downstream of "C" tank gas analyzer isolation valve (1101C)	1.5×10^{-4}*
(21) Fitting downstream of "C" tank level transmitter cutout valve (1271C)	1.3×10^{-4}
(22) Fitting downstream of "C" tank level transmitter cutout valve (1271C)	1.9×10^{-4}*
(23) Second fitting downstream of "C" tank gas analyzer isolation valve (1101C)	2.4×10^{-4}
(24) "C" tank manway	0.3 to 2*
(25) "C" tank top of tank blank flange	1.2×10^{-3}
(26) "A" tank manway	0.3 to 2*
TOTAL Holdup Tank Area	0.6 to 4
TOTAL After Repair	4.4×10^{-3}
Post Accident Containment Vent System	
Unit 1 TOTAL	8.5×10^{-4}
Unit 2 TOTAL	None

*Item Repaired.

2.1.7 IMPROVED AUXILIARY FEEDWATER SYSTEM RELIABILITY FOR PWRs

2.1.7.b Auxiliary Feedwater Flow Indication to Steam Generators for PWRs

Auxiliary Feed Pump Flow

As indicated on Figure 2.1.7-1 in our December 31, 1979, submittal, the power supply for each auxiliary feed pump discharge flow indication circuit is as follows:

1F-4002 Unit 1 steam-driven auxiliary feed pump discharge flow (1P29), red instrument bus, Unit 1

2F-4002 Unit 2 steam-driven auxiliary feed pump discharge flow (2P29) red instrument bus, Unit 2

F-4007 Common motor-driven auxiliary feed pump discharge flow (P38A), blue instrument bus, Unit 1

F-4014 Common motor-driven auxiliary feed pump discharge flow (P38B), blue instrument bus, Unit 2

Note: Each instrument bus is supplied with 120V AC from its associated inverter supplied with 125V DC by a station battery as follows:

Unit 1 Red	"A" Battery
Unit 2 Red	"A" Battery
Unit 1 Blue	"B" Battery
Unit 2 Blue	"B" Battery

One swing inverter serves as a backup for both red inverters and one swing inverter serves as a backup for both blue inverters.

Figure 2.1.7-1 incorrectly labels the channel numbers associated with each auxiliary feed pump. The listing above is correct.

Steam Generator Level Indication

Steam generator level indication provides the ultimate information on secondary system capability for heat removal from the RCS and additionally provides a backup for auxiliary feedwater flow indication. Steam generator level indication circuits on both units are configured as follows:

Transmitter	Steam Generator	Instrument Bus	Grade
LT-461	"A"	Red	Reactor protection
LT-462	"A"	Blue	Reactor protection
LT-463	"A"	Yellow	Reactor protection
LT-460	"A" (wide range)	Blue	Control
LT-471	"B"	Yellow	Reactor protection
LT-472	"B"	Red	Reactor protection
LT-473	"B"	White	Reactor protection
LT-470	"B" (wide range)	Red	Control

Channels 461 (471) through 463 (473) serve as reactor protection system signal sources and are configured appropriately for this function. Indication of these measurements is provided on the main control board through isolation devices. The steam generator level control system also receives signals from these channels to maintain the steam generator level during operation. Channel 460 (470) services as a wide range indication of steam generator level and is recorded on the main control board. It is treated as a control grade indication circuit.

Instrument Bus Power Supply

The white and yellow instrument buses are supplied with 120V AC power through AC/AC MG sets. (MG sets serve as isolation devices.) The yellow bus is supplied with power via B04 and can be directly supplied with power from the emergency diesel. The white bus is supplied with power via B02 and does not normally realign to receive power from the emergency diesel. However, if necessary, power can be supplied from the emergency diesel via B04. This alignment can be shown using Figure 2.1.1-2 of our December 31, 1979, submittal.

2.1.8.a IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

Sampling Times

The presently installed sampling modifications permit chemistry personnel to obtain a sample within one hour after access has been authorized to the Auxiliary Building by health physics personnel. No additional time is required for the installation of any special provisions; however, the requirement for health physics approval prior to building access is consistent with the TMI Lessons Learned item reflecting the need for exercising prudent radiological controls prior to entry. The analyses can be completed within one hour of sampling.

Counting Room and Chemistry Lab Availability

The counting room is marginally available in the event of a Unit 1 accident with the source term assumed in NUREG-0578 due to unique circumstances of acceptable radiation levels in the vicinity of counting equipment and elevated levels in areas of personnel occupancy. It has been determined that gamma spectrometry can be performed with existing facilities for a limited number of higher activity samples, such as coolant samples. Counting facilities are also available at the Kewaunee Nuclear Plant pursuant to the existing reciprocal agreement between Point Beach Nuclear Plant and Kewaunee Nuclear Plant to provide backup analytical laboratory services.

Since the chemistry laboratory may not be accessible in the event of a Unit 1 accident with the source term assumed in NUREG-0578, additional sample handling and preparation facilities will be provided elsewhere on site to meet the January 1, 1981, requirements.

In the event of similar accident assumptions in Unit 2, access to both the chemistry laboratory and the counting room would be unaffected, and analyses can also be performed within one hour of sampling.

The desirability of providing additional gamma spectrometry equipment at another location on site is being investigated.

Materials

Materials used for the fabrication of the new sampling facility are the same safety classification as the original plant sampling facility. Stainless steel tubing and Swagelok fittings are used throughout.

2.1.8.b INCREASED RANGE OF RADIATION EFFLUENT MONITORS

Interim Instrumentation

In our submittal of December 31, 1979, our response to Item 2.1.8.b documented an interim methodology for high range effluent monitoring based on direct radiation measurements taken on vent stacks. However, because of personnel dose and availability considerations, we committed to the provision of additional installed monitoring instrumentation for a more satisfactory interim implementation of high level monitoring capability. The installation of this additional monitoring equipment has been completed.

Enclosed herewith is a revision to the description of our implementation of increased range effluent monitoring. This revision replaces in its entirety Section 2.1.8.b of the Addendum enclosed with our submittal of December 31, 1979. The new material is a major editorial revision for purposes of clarity and includes some minor technical changes which were found to be required in the course of installing the new monitoring equipment.

Monitoring of Steam Dump and Code Safety Valve Header

Interim methods have been developed to permit the estimation of high level release rates based on direct radiation measurements obtained on the header for the atmospheric steam dump and Code safety valves of the steam generators. Additional discussion and the figure showing the relationship between direct radiation measurement and release rates are provided in Section 2.1.8.b of the Addendum enclosed herewith. Appropriate permanent noble gas monitors will be provided to meet the January 1, 1981, requirements.

Analysis of Particulate Filters and Radioiodine Cartridges

Assuming a Unit 1 accident with the source terms specified in NUREG-0578, the marginal availability of the counting room may not be appropriate for performing gamma spectrometry on particulate filters and radioiodine cartridges obtained from effluent sampling and monitoring systems. The reciprocal agreement between Point Beach Nuclear Plant and Kewaunee Nuclear Plant to provide backup analytical laboratory services to both plants is relied upon to provide accurate and sensitive analyses of effluent sampling filters and cartridges in the event of a Unit 1 accident. The desirability of providing backup gamma spectrometry equipment elsewhere on-site is being investigated.

Radiological Safety

Appropriate health physics controls and radiological precautions to be observed during monitoring and sampling procedures are provided in HP17.6.1 and 17.6.4 of the Point Beach Nuclear Plant Health Physics Administrative Control Policies and Procedures Manual.

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION

As indicated in our December 31, 1979, submittal, improved procedures have been implemented at Point Beach Nuclear Plant with respect to the determination of airborne radioiodine. Portable samples are used at the plant in lieu of dedicated iodine monitors because their ease of portability and extreme versatility provides assurance of the ability to monitor areas as required. The procedures call attention to the need for discriminating against noble gas adsorption on charcoal cartridges by either gamma spectrometry or purging of the charcoal cartridges with clean gas prior to counting. In addition, the reciprocal agreement between Point Beach Nuclear Plant and Kewaunee Nuclear Plant to provide backup analytical laboratory services to both plants provides further assurance of accurate radioiodine analyses. In general, radioiodine analyses are performed at Point Beach Nuclear Plant by gamma spectrometry. However, in the event the counting laboratory is unavailable or inaccessible for any reason, gross counting can be used. It is anticipated that selected samples determined by gross counting would be further verified by follow-up gamma spectrometry at Kewaunee Nuclear Plant.

To provide additional assurance of performing accurate radioiodine analyses in as short a time as is reasonably achievable, silver zeolite cartridges have been ordered by Point Beach Nuclear Plant to be used for radioiodine sampling in the event of an accident. These cartridges have the advantage of inherent discrimination against noble gases and require only a brief purging with clean gas prior to gross counting. Further modification of procedures will address the use of these cartridges when the equipment is received.

Gross counting equipment (Eberline RM-14 counter and HP-210 probe) is available on-site in a low background area at the site boundary control center. Further equipment may be provided in the permanent Technical Support Center when its construction is completed.

Training of health physics personnel in the use of accident radioiodine monitoring procedures is included in the overall training of health physics personnel in normal and emergency procedures and is practiced during emergency drills.

ACRS REACTOR COOLANT SYSTEM VENT

The reactor coolant system (RCS) gas vent system which will be installed at Point Beach includes the capability to vent both the reactor vessel head and the pressurizer steam space. Relief paths are provided to the pressurizer relief tank (PRT) and containment atmosphere. In each vent path there are two normally closed, Safety Class 2 isolation valves in parallel from the possible gas source (vessel head or pressurizer) and one normally closed, Safety Class 2 isolation valve in each of the two parallel relief paths (PRT or containment). These are pilot operated, solenoid valves which fail closed. Each valve in a parallel configuration will be powered from a separate safeguards bus. The system is designed such that any single active failure will not prevent venting nor prevent vent isolation. The valves will be remotely operated with key locked controls from the control room. Direct position indication will be provided in the control room for each valve. Leakage in the system can be detected using a pressure indication, with a high pressure alarm, from the piping volume contained between the isolation valves.

The system will be designed to meet ANSI Safety Class 2 specifications up to and including the isolation valves. The remainder of the system will meet ANSI Safety Class 4 specifications. Included in each RCS penetration will be an appropriately sized (7/32") orifice to provide both the required flow capabilities for venting and a limitation on primary coolant loss below that size "corresponding to the definition of a LOCA (10 CFR 50 Appendix A)". The connections to the RCS will be made using existing penetrations and piping, up through the orifices, which have been considered in the plant LOCA analyses and already provide for manual isolation.

The system is shown in Figure ACRS-1, including proposed manual vent, drain, and maintenance isolation valves. Engineering design and analysis for the Point Beach RCS gas vent system is being performed by Combustion Engineering, Inc., consistent with the qualifications applied to similar systems in the plant. The design of the system is being coordinated by Wisconsin Electric with other utilities installing the same system on similar Westinghouse PWRs.

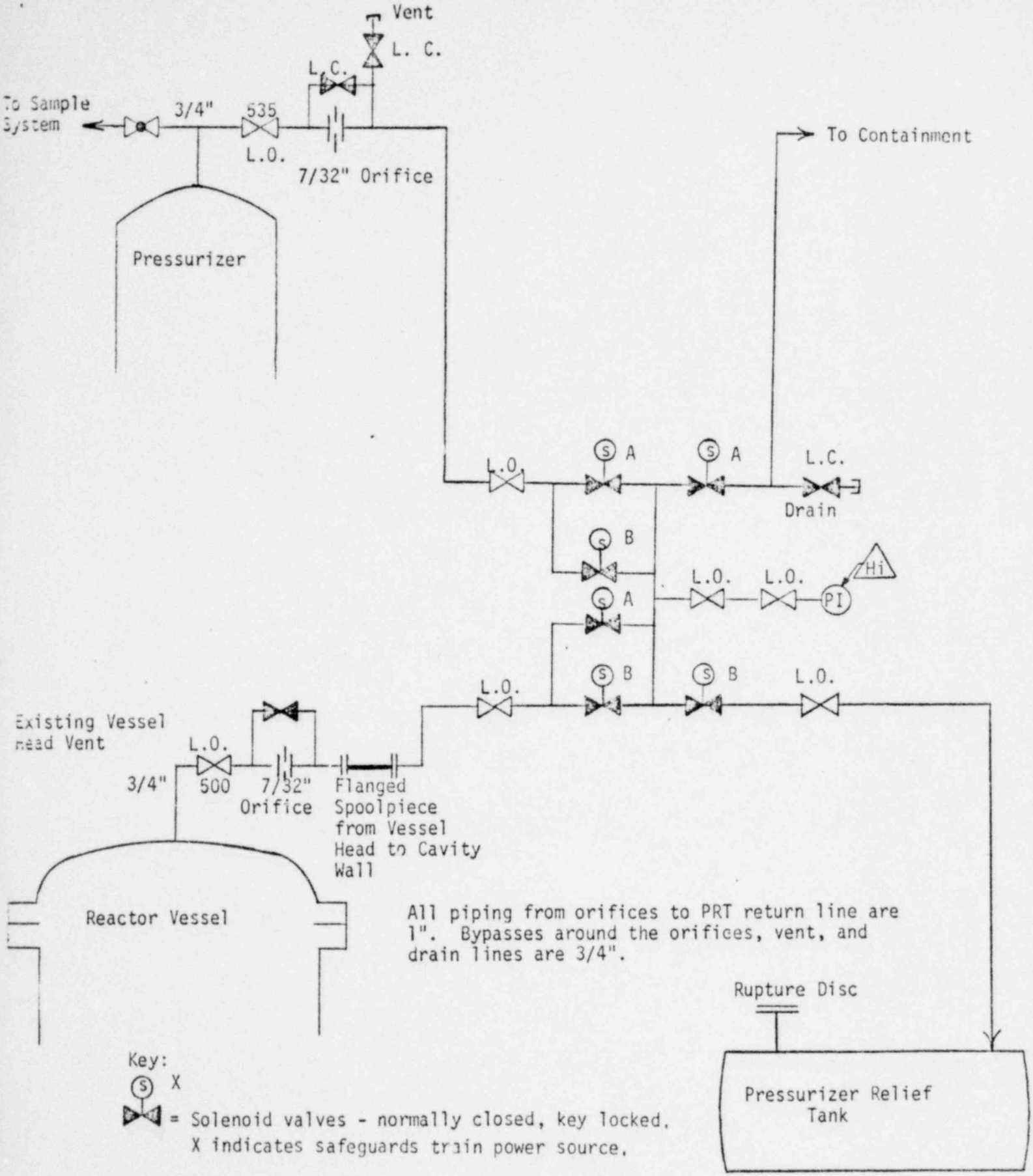


FIGURE ACRS-1 REACTOR COOLANT SYSTEM GAS VENT

2.2.1 IMPROVED REACTOR OPERATIONS COMMAND FUNCTION

2.2.1.a Shift Supervisor's Responsibilities

The Duty Shift Supervisor is responsible for the overall command function of the plant on his assigned shift. As such, he is expected to be anywhere in the plant where the execution of this responsibility is required. In the Point Beach organization, the Operating Supervisor is in charge of the control room operators during both normal and emergency conditions. This is explicit in the plant procedures, as discussed in our December 31, 1979, submittal, in response to the NRC position requiring "a definite line of command". It is, therefore, not realistic or proper to require the Duty Shift Supervisor to "step in" and relieve the Operating Supervisor of his duty of directing the control room operators during emergency conditions. To do so would directly conflict with the intent of NRC Clarification Item 2.a, "to maintain the broadest perspective of operational conditions affecting the safety of the plant". We agree with this philosophy and have provided for its realization through this succession of authority, two levels (Operating Supervisor and control room operators) of which remain in the control room at all times during accident conditions.

2.2.1.b SHIFT TECHNICAL ADVISOR

The specific duties, responsibilities, authority, and station location of Shift Supervisors, Operating Supervisors and Control Operators are clearly presented in the Administrative Control Policies and Procedures Manual as PBNP 4.2, 4.3, and 4.4, respectively. Also, further information and commitments for these licensed people activities have been made in other documents post-dating the TMI-2 accident.

Similarly, the duties, responsibilities, authority and location of Duty & Call Superintendents are spelled out in the Administrative Control Policies and Procedures Manual, PBNP 3.13, and in part, in the Emergency Plan.

For the Duty & Call Technical Advisor, our three previous NUREG-0578 related submittals to the NRC spell out this person's "evaluation and assessment" function and availability.

The purpose of this clarification is to clearly delineate the Duty & Call Superintendent and Technical Advisor coordination and interfacing in a significant event when there is a requirement to report to the control room and/or technical support center. During the year 1980, the Duty & Call Superintendents will, in general, have higher qualifications for "evaluation and assessment" than the Technical Advisors; therefore, coordination shall be as follows:

1. In a "significant event", the Duty Shift Supervisor (or Operating Supervisor) first alerts the Duty & Call Superintendent, as has been the practice over the years. Together they decide on further alerting of personnel and the potential of the significant event in progress.

2. Assuming the significant event decision above falls in the one-hour NRC reporting requirement with serious potential, the Duty Shift Supervisor next alerts the Duty & Call Technical Advisor to report to the control room (within ten minutes) and initiates his "evaluation and assessment" activity. Then the Duty Shift Supervisor focuses on his emergency operational direction activity where it will remain until he is appropriately relieved or the incident is resolved.

3. When the Duty & Call Superintendent arrives (within approximately thirty minutes) in the control room, he will interface with the Technical Advisor and the Duty Shift Supervisor (as operating demands permit.

   The Duty & Call Superintendent will then decide, based upon the type of significant event, whether he or the Technical Advisor is more qualified to continue the "evaluation and assessment" function and whether the Technical Support Center should be activated.

4. In most cases, in the year 1980 while Duty & Call Technical Advisors are developing and training, it is likely the Duty & Call Superintendent will decide he is more qualified to evaluate and assess; and will, therefore, relieve and send the Duty & Call Technical Advisor to the Technical Support Center to initiate and continue NRC communications.

5. The assigned person upon reaching the Technical Support Center will then open up the NRC telephone link and keep it open continuously if required. He will also be logging appropriate information on contacts and response. It is important as far as practicable to log phone calls and data transmission so as much documentation as possible exists for turnover and evaluation. Any information transmitted off-site will be from this log. Moreover, the log will help the "Technical Analysis Team" in the Technical Support Center to watch for trends and relay suggestions to the Duty & Call Technical Advisor or Duty & Call Superintendent in the control room. Finally, such a log would be consulted prior to any request for information from the control room to reduce the needless interference in the control room activity by repeatedly asking for the same information.

   The Coordinator in the Technical Support Center should designate a section of the blackboard as a status board where key parameters are printed for easy display.

6. The assigned person in the Technical Support Center continues according to the Emergency Plan to act as Coordinator and place other telephone calls to other regulatory authorities and to other employes to establish a functioning Site Boundary Control Center and Technical Support Center. When a second Duty & Call Superintendent is available in the Technical Support Center, he may assume the Coordinator functions while the assigned person continues to serve as the primary communications person.

7. The Duty & Call Superintendent (or Duty & Call Technical Advisor) remaining in the control room shall provide information as requested or as felt desirable to the person in the Technical Support Center. It is reemphasized that all communications with the control room shall be handled through the Technical Support Center. Normally, the Duty & Call Technical Advisor or Duty & Call Superintendent in the control room will <u>not</u> be manning the NRC hot line.

All of the above action steps relate to a condition of minimum availability of personnel on backshifts. As more Duty & Call, Technical Advisor, or Group Head people become available on-site, the Coordinator will press them into appropriate service if needed.

A modification request has been issued to transmit key parameters to the Technical Support Center to reduce the voice communications and logging requirements. This will be accomplished prior to 1981, dependent upon equipment availability.

2.2.2.b TECHNICAL SUPPORT CENTER

Health Physics Equipment

Lockers have been provided in the temporary Technical Support Center (TSC) and the temporary Operations Support Center. Typical inventories are shown in Table 2.2.2.b-1. Similar equipment will be provided in the corresponding permanent centers when construction of the latter is completed.

Communications

In addition to the communications described in the Wisconsin Electric December 31, 1979, submittal, a telephone which has an additional direct outside line has been added to the Technical Support Center. Also, access to the microfilmed plant drawings can be achieved using a 3M 201 Dry Silver Reader Printer in the adjacent plant office. This provides for direct viewing and hardcopy prints of all microfilmed materials.

Communications with the control room, in the event of an accident, will be via one of the two plant extensions or the speaker phone in the Technical Support Center. Two dedicated persons (in the control room and TSC) will be assigned to the communications task or the speaker phone will be used. Once the link with the control room is established, it will not be broken and used for other communications unless another link has been established. Walkie-talkie radio or a Bell phone may also be used for this link. Additional clarification for plant personnel regarding the duties and responsibilities of interfacing during an accident are documented in Procedure DCS 1.12.

Instrumentation

In the event of an accident, all plant information will be transferred from the control room to the TSC via the communications link. The instrumentation to be provided by January 1, 1981, will consist of nine (9) two-pen recorders, a datalogger, and several analog instruments. The information which will be displayed and/or recorded on this instrumentation is given in Table 2.2.2.b-2. The primary power supply for the recorders and datalogger will be the white instrument bus, with the yellow instrument bus as backup.

In the event the TSC should become uninhabitable, the Site Boundary Control Center would become the TSC and two selected plant personnel would be assigned to the control room to provide for data transfer and evaluation needs for the relocated TSC.

Monitoring Air Quality in the Temporary Technical Support Center

The present capabilities for monitoring air quality in the temporary Technical Support Center range from basic non-continuous indicating to continuous indicating.

1. Non-Continuous Indicating

   a. Noble Gases

   The basic technique utilizes the simple emptying of a water-filled one liter polypropylene bottle in the atmosphere of interest. Analysis will be accomplished by gamma spectrometry. A sample of this type is easily collected and, assuming access to the multi-channel analyzer, can be analyzed within twenty minutes to one-half hour with available personnel. The existing stack noble gas effluent monitors will be utilized to determine the general magnitude of noble gas concentration in the Technical Support Center (TSC).

   b. Particulates and Iodines

   Low volume and high volume sampling equipment presently exist in the TSC. The particulate and iodine filters from these can be screened with a hand held instrument at the TSC. Quantification of activity will be conducted in the Health Physics Station and Chemistry Counting Room, if accessible; or at the Site Boundary Control Center and the Kewaunee Nuclear Plant.

   For samples obtained with the high volume air samples, gross particulate activity can be determined within twenty minutes to one-half hour after sampling. Gross iodine activity, assuming negligible quantities of noble gases, also can be determined within one-half hour.

   With the pending receipt of silver zeolite cartridges, a "go, no go" screening procedure will be developed for iodine counting which will allow an estimate of gross iodine activity to be determined in the TSC. This estimation can be performed within ten minutes after sampling. Particulate activity also will be estimated at the TSC by this procedure.

2. Continuous Indicating

   a. Gross Particulate Activity Trend

   An Eberline AMS-2 particulate filter monitoring system is available for use in the TSC. It is currently stored in the Health Physics Station source room, pending

receipt of a presently ordered cart which will be dedicated to facilitate its movement.

The AMS-2 incorporates a shielded detector monitoring a fixed filter paper with a continuous indicating meter and adjustable alarm setting. It is useful for indicating the gross particulate activity trend.

b. Gross Noble Gas Activity

A shielded calibrated GM tube monitoring device has been devised to be utilized in conjunction with the AMS-2 low volume pump. Sample air is drawn through a metal chamber by the pump. The GM tube is mounted within the chamber. An Eberline RM-14 type instrument with adjustable alarm setting is utilized to display the activity sensed by the detector. This assembly will be mounted on the moveable cart with the AMS-2. Successful calibration of the detector has been accomplished utilizing known activities of noble gases.

TABLE 2.2.2.b-1

INVENTORY OF HEALTH PHYSICS
EQUIPMENT IN TECHNICAL SUPPORT CENTER
AND OPERATIONS SUPPORT CENTER

ITEM	TSC(1)	OSC(1)
Sampling Equipment		
Low volume Air Sampler	1	1
Particulate Filters, box	1	1
Charcoal Filters, box	4	4
50' Extension Cord	2	2
High Volume Air Sampler	1	1
Particulate Filters, box	1	1
Charcoal Filters, box	1	1
Dosimetry Equipment		
Dosimeters (0-5000 mR)	10	10
Dosimeter Charger	1	1
Batteries, Size AA, pkg.	1	1
Survey and Monitoring Equipment		
Victoreen Vamp Area Monitor	1	1
Rad Owl II	1	1
Thyac III - Side Window Probe	1	1
Batteries, Size D	12	12
Signs		
Three-Pocket Placards	12	12
"Radiation Area" Inserts	12	12
"High Radiation Area" Inserts	12	12
"RWP Required" Inserts	12	12
"Airborne Area" Inserts	12	12
"Contaminated Area" Inserts	12	12
"Radioactive Materials" Inserts	12	12
Respiratory Protection Equipment		
Clear-Vue Respirator	1	1
Ultra-Vue Respirator	6	6
Filter Cartridges, box	1	1
Smoke Test Kit	1	1
Forms		
Air Particulate Sample, CHP-22, pad	1	1
Iodine Survey, CHP-2, pad	1	1
Misc. Survey, pad	1	1
Radiation Work Permit, CHP-31, pad	2	2
Dosimeter Rezero, CHP-34, pad	1	1
Irregular or Offscale Dosimeter Report, CHP-3, pad	1	1
Coordinator Logsheets	5	5

TABLE 2.2.2.b-1 - CONTINUED

ITEM	TSC(1)	OSC(1)
Miscellaneous		
Barricade Tape, Yellow/Magenta, rolls	5	5
Tuck Tape, rolls	2	2
Hot Spot Tags	50	50
Radiation Material Hazard Tags	50	50
Radioactive Material Contamination Tags	50	50
Yellow/Magenta Tape, rolls	6	6

(1) For licensing and inspection purposes, quantities are to be regarded as approximate.

TABLE 2.2.2.b-2

TECHNICAL SUPPORT CENTER INSTRUMENTATION

Parameters to be Displayed on Rack Mounted 2-Pen Recorders

T_H and T_C Loop A or B - Unit 1 (switched)

T_H and T_C Loop A or B - Unit 2 (switched)

Pressurizer Wide Range Pressure - Unit 1 and Unit 2

Pressurizer Level - Unit 1 and Unit 2

Steam Generator A and B Pressure - Unit 1

Steam Generator A and B Level - Unit 1

Steam Generator A and B Pressure - Unit 2

Steam Generator A and B Level - Unit 2

Containment Narrow Range Pressure - Unit 1 and Unit 2

Parameters to be Datalogged

Unit 1

Incore Thermocouples (8-2 per core quadrant)

Auxiliary Feedwater Flow (2 channels - pump discharge flow until steam generator flow is installed)

High Pressure Safety Injection Flow (2 channels - Train A and Train B)

Containment Sump Level (1 channel)

Containment High Range Radiation Monitor (1 channel)

Containment Purge Exhaust Vent Stack High Range Effluent Monitor (1 channel)

Unit 2

Incore Thermocouples (8-2 per core quadrant)

Auxiliary Feedwater Flow (2 channels - pump discharge flow until steam generator flow is installed)

TABLE 2.2.2.b-2 - CONTINUED

High Pressure Safety Injection Flow (2 channels - Train A and Train B)

Containment Sump Level (1 channel)

Containment High Range Radiation Monitor (1 channel)

Containment Purge Exhaust Vent Stack High Range Effluent Monitor (1 monitor)

Common

Auxiliary Building Vent Stack High Range Effluent Monitor

Combined Air Ejector Discharge High Range Effluent Monitor

Drumming Area Vent Stack High Range Effluent Monitor

Gas Stripper Building Vent Stack High Range Effluent Monitor

Parameters to be Displayed Only

Time

Wind Speed

Wind Direction

ERRATA TO DECEMBER 31, 1979, SUBMITTAL

Page 2.1.6-2, first paragraph after item 4, last sentence:

"The liqui lines have been tested at operational pressures sing water, and visual inspection was used for leak detection."

Page 2.1.7-5, item 3, line 3:

". . . the feed flow test jack."

Page 2.2.1-2, item 2, line 3:

". . . Supervisor, Operating Supervisor, and the control operators are . . ."

Addendum, page 2.1.6.a-9, second from the last sentence in Section 3.1.2"

". . ., this method was successful in detecting, locating, and quantifying leakage sources."

ADDENDUM

JANUARY 1, 1980, IMPLEMENTATION STATUS FOR
NUREG-0578 ITEMS

WISCONSIN ELECTRIC POWER COMPANY
DOCKETS 50-266 AND 50-301
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

SECTIONS

2.1.8.b

2.1.8 INSTRUMENTATION TO FOLLOW THE COURSE OF AN ACCIDENT

2.1.8.b INCREASED RANGE OF RADIATION MONITORS

TABLE OF CONTENTS

1.0 Summary of Methods Used to Meet Section 2.1.8.b Requirements for Monitoring High-Level Gaseous Effluents

- 1.1 Specific Requirements for Increasing the Range of Radiation Monitors Which Must be Provided by January 1, 1980
- 1.2 Specific Requirements for Increasing the Range of Radiation Monitors Which Must be Provided by January 1, 1981

2.0 Description and Design Review of Existing Plant Monitoring Equipment For Vents Producing Gaseous Effluents

- 2.1 Introduction
- 2.2 Auxiliary Building Vent Stack
- 2.3 Drumming Area Ventilation Stack
- 2.4 Containment Purge Exhaust Stacks
- 2.5 Combined Air Ejector Decay Duct Exhaust
- 2.6 Gas Stripper Exhaust
- 2.7 Steam Safety Valves and Atmospheric Dump Valves
- 2.8 Evaluation of Existing Systems

3.0 Gaseous Effluent Monitoring: Design Modifications

TABLE OF CONTENTS

(Continued)

3.1 Introduction

3.2 Auxiliary Building Ventilation Stack, Gas Stripper Stack, and Drumming Area Ventilation Stack.

3.3 Containment Purge Stacks

3.4 Combined Air Ejector Decay Duct

3.5 Steam Safety Valves and Atmospheric Dump Valves.

4.0 References

TABLES

1. High-Range Noble Gaseous Effluent Radiation Monitoring Equipment

FIGURES

1. Deleted

2. Deleted

3. Deleted

4. Auxiliary Building Stack Rad/hr to Ci/sec Conversion Chart

5. Deleted

6. Deleted

7. Drumming Area Rad/hr to Ci/sec Conversion Chart

8. Deleted

9. Containment Purge Exhaust (1 Fan) Rad/hr to Ci/sec Conversion Chart

10. Containment Purge Exhaust (2 Fans) Rad/hr to Ci/sec Conversion Chart

11. Combined Air Ejector Decay Duct Rad/hr to Ci/sec Conversion Chart

12. Gas Stripper Stack Rad/hr to Ci/sec Conversion Chart

13. Auxiliary Building Stack R/hr to Ci/sec Conversion Chart (Measurement at 8"x8"x9" Expansion Chamber)

14. Containment Purge Exhaust (1 Fan) R/hr to Ci/sec Conversion Chart (Measurement at 3/4" Containment Purge Line)

FIGURES
(Continued)

15. Containment Purge Exhaust (2 Fans) R/hr to Ci/sec Conversion Chart (Measurement at 3/4" Containment Purge Line)

16. Drumming Area R/hr to Ci/sec Conversion Chart (Measurement at 8"x8"x9" Expansion Chamber)

17. Gas Stripper Stack R/hr to Ci/sec Conversion Chart (Measurement at 8"x8"x9" Expansion Chamber)

18. Combined Air Ejector Decay Duct R/hr to Ci/sec Conversion Chart (Measurement at 4" Sch 40 Exhaust Pipe)

19.-24. Not Used.

25. Steam Line Safety Valve and Atmospheric Dump Valve Exhaust R/hr to uCi/cc Conversion Chart.

1.0 SUMMARY OF METHODS USED TO MEET SECTION 2.1.8.b REQUIREMENTS FOR MONITORING HIGH-LEVEL GASEOUS EFFLUENTS

Each of the gaseous effluent paths to the atmosphere from the Point Beach Nuclear Plant was reviewed to determine the adequacy of existing instrumentation to monitor the release rates postulated in NUREG 0578. This evaluation indicates that the following modifications should be made to the existing gaseous monitoring systems:

a. Installation of high-range detectors on all gaseous effluent vent duct sampling lines to allow monitoring of expected releases;

b. Installation of high-range detectors on the exhaust of the combined air ejector decay duct;

c. Installation of shielding around each detector to mimimize the effects of background levels;

d. Modifications to existing procedures to instruct personnel in case the use of the high-range detectors and sampling lines is required.

e. Modifications to existing procedures to instruct personnel in the use of portable instruments to monitor the steam safety valve and atmospheric dump valve exhausts.

Although the installation of new equipment is not required by the NRC for interim compliance, it has been concluded that new equipment is desirable, based on personnel exposure and accessability considerations.

The above modifications have been completed. The high-range radiation monitoring equipment, listed in Table 1, has been purchased from the Eberline Instrument Corporation. In addition to these steps, plans have been made to establish a design and procure equipment necessary to meet the January 1, 1981, requirements for permanent modifications.

1.1 SPECIFIC REQUIREMENTS FOR INCREASING THE RANGE OF RADIATION MONITORS WHICH MUST BE PROVIDED BY JANUARY 1, 1980

There are two basic requirements in NUREG-0578 Section 2.1.8.b: increasing the range of existing radiation monitoring equipment to monitor noble gas effluents, and adding to the capability to sample radioiodine and particulates from the high-level gases.

The requirements have been satisfied at the Point Beach Nuclear Plant by the modifications to the existing vents described in Section 3.0.

1.1.1 REQUIREMENTS FOR NOBLE GAS EFFLUENT MONITORS

1.1.1.1 GENERAL

All operating reactors must provide an interim method for quantifying high-level releases of noble gases from all potential release points. At Point Beach Nuclear Plant, this requirement is satisfied by the use of portable survey instruments and figures showing the relationships between direct radiation readings obtained at the effluent ducts or the main steam line and release rates or concentrations. Improved interim monitoring capability is available using the high range monitors which have been installed on the gaseous effluent duct sample lines and the combined air ejector decay duct exhaust pipe. Figures showing the relationships between the direct radiation readings obtained at the sampling locations and the release rates have been prepared.

1.1.1.2 SYSTEM DESCRIPTION

An Eberline RMS II radiation monitoring system is installed to monitor each plant vent. High-range detectors are located on sample lines for each vent except the combined air ejector decay duct. For the combined air ejector decay duct, the high-range detector is mounted on the discharge piping. The ranges of each instrument are described in Table 1. Monitoring locations are described in detail in Section 3.0. The monitoring locations are shielded to minimize the effect of background on the radiation readings. By January 1, 1981, the sensors and readouts will be powered from an instrument bus which is supplied by an emergency power source to ensure continuous operation should a loss of offsite power occur. Readouts are displayed in the control room.

1.1.1.3 PROCEDURES

Procedures for noble gas monitoring have been developed, and figures showing the relationships between meter readings and release rates are attached, both for the case in which the measurement is made directly on the duct with a portable survey instrument and for the case in which the reading is made at the sample location by the Eberline RMS II system. A figure has also been developed showing the relationship between meter readings and release concentration when the readings are taken on the steam line to monitor steam safety valve and atmospheric dump valve releases.

1.1.2 REQUIREMENTS FOR THE MONITORING OF HIGH-LEVEL RADIOIODINE AND PARTICULATE GASEOUS EFFLUENTS

1.1.2.1 SYSTEM DESCRIPTION

Existing sample locations and equipment will be used to obtain samples for determining radioidine and particulate activity in the gaseous effluent steams. When sampling is necessary, the operator will use a portable sample pump to draw a sample through a particulate filter and a charcoal canister. Analysis of the filter paper and charcoal canister will be performed in the chemistry laboratory using standard methods. The existing sample locations are in areas which allow sufficient access time for obtaining representative samples.

1.1.2.2 PROCEDURES

Procedures at Point Beach Nuclear Plant have been reviewed and improved to allow the processing of particulate filters and radioiodine charcoal filter canisters to ensure that the results of the analysis are not biased by external contamination or noble gases. Isotopic multichannel analysis will be used to determine the presence of particulates and radioiodine. These procedures contain suitable safety precautions and detailed instructions to minimize radiation exposure during the processing of the filters and charcoal canisters.

1.2 SPECIFIC REQUIREMENTS FOR INCREASING THE RANGE OF RADIATION MONITORS WHICH MUST BE IMPLEMENTED BY JANUARY 1, 1981

Modifications which involve installation and redesign of the equipment are required by January 1, 1981. This section describes specific requirements which must be met by January 1, 1981, along with the description of the plans to meet those requirements.

1.2.1 HIGH-RANGE NOBLE GAS EFFLUENT MONITORING

By January 1, 1981, high-range noble gas effluent monitors are to be provided for each release path which meets the requirements of Table 2.1.8.b.2 of NUREG-0578.

At Point Beach Nuclear Plant, fixed high-range noble gas effluent radiation gas monitors have been installed on each plant vent stack to provide improved interim monitoring capability. The use of these monitors to satisfy the requirements for January 1, 1981, is being evaluated. Additional modifications, or refinements to interim modifications, will be made as necessary so that high-range noble gas effluent monitors meeting the requirements of Table 2.1.8.b.2, NUREG-0578, will be fully installed by January 1, 1981.

1.2.2 RADIOIODINE AND PARTICULATE MONITORING

Means are required to continuously sample and provide analysis of sampling media for radioiodine and particulate levels.

Point Beach Nuclear Plant will implement provisions and modifications as necessary for continuously sampling and providing analysis of the high-level particulate and radioiodines from each release path, by January 1, 1981.

1.2.3 CONTAINMENT RADIATION MONITORS

Two radiation monitoring systems in containment shall be provided to meet the requirements of Table 2.1.8.b.3, NUREG-0578.

Point Beach Nuclear Plant will install high-range radiation monitors at suitable locations to enable monitoring of containment radiation by January 1, 1981.

2.0 DESCRIPTION AND DESIGN EVALUATION OF EXISTING PLANT MONITORING EQUIPMENT FOR VENTS PRODUCING GASEOUS EFFLUENTS

2.1 INTRODUCTION

Gaseous effluents at Point Beach Nuclear Plant flow from stack vents supplied by ventilation fans which take suction from air in the auxiliary building, the waste drumming and the spent fuel pool area, and both containments. The remainder of the gaseous effluents result from the letdown gas stripper vent and the combined air ejector decay duct. All of these vents currently have low-level radiation monitors installed.

The steam safety valves and atmospheric dump valve may also release entrained gaseous effluents. No installed monitoring provisions exist for the steam safety valves and atmospheric dump valves.

2.2 AUXILIARY BUILDING VENTILATION STACK

The auxiliary building ventilation stack consists of a 54"-diameter, circular, sheet steel ventilation duct with a flow rate of approximately 61,400 CFM. The auxiliary building ventilation system consists of two intake fans located at the 8.0' elevation which exhaust through the ventilation stack. The ventilation stack exhaust is located above the containment facade.

2.2.1 EXISTING GASEOUS EFFLUENT MONITORING EQUIPMENT

The monitoring equipment, designated channel R-14, for the auxiliary building ventilation stack is a Tracer Lab, Model MD-12C, beta-gamma Geiger-Mueller tube detector with a check source. The log rate meter display unit is located in the control room for use by the operator in monitoring low level release rates.

The existing auxiliary building vent stack monitor was designed to detect low levels of radioactivity in the gaseous effluent stream. Radioiodine and particulates are monitored at sampling locations in the duct. A sample pump is installed on the duct which draws some of the effluent stream through a filter paper and a charcoal filter canister and returns the sample stream to the duct. The filter and charcoal canister are removed and laboratory analysis determines the radioiodine and particulate content.

2.3 DRUMMING AREA VENTILATION STACK

The drumming area ventilation stack consists of a 46-inch diameter, circular, sheet steel stack which starts at the 65' elevation in the solid waste processing area. The flow rate is approximately 43,100 CFM, and all air from the spent fuel pool area and waste handling equipment is exhausted by the stack.

2.3.1 EXISTING GASEOUS EFFLUENT MONITORING EQUIPMENT

The existing equipment used for monitoring radiation levels of gaseous effluents for the drumming area is similar to that used in the auxiliary building ventilation stack and consists of a Geiger-Mueller tube assembly in the duct with readout in the control room. Particulates and radioiodines are collected on a paper filter and a charcoal filter and are analyzed in the laboratory.

2.4 CONTAINMENT PURGE EXHAUST STACKS

The Unit 1 and 2 containment purge exhaust stacks consist of 36"-diameter, circular, steel ducts with a purging flow rate of approximately 12,500 CFM with one fan and 25,000 CFM with two fans operating. The stacks are based at elevation 66' and

exhaust containment atmosphere which has been first passed through HEPA and charcoal filters. Air flow exists only when containment purge or continuous venting is in process.

2.4.1 EXISTING GASEOUS EFFLUENT MONITORING EQUIPMENT

Radiation monitoring for the purge ducts utilizes vacuum pump and sampling probe arrangments to obtain samples of gaseous effluents passing through the purge ducts. A 10 CFM pump draws the gas sample from the probe to the monitor, where it is filtered to collect particulates and then passed to a constant-volume chamber for gaseous activity monitoring. After monitoring, the sample is returned to the stack. This system is designed to monitor only the low levels of radiation which are expected to be encountered during containment purges or venting. Radioiodine monitoring is accomplished by laboratory analysis of grab samples.

2.5 COMBINED AIR EJECTOR DECAY DUCT EXHAUST

The combined air ejector decay duct exhaust consists of a 36"-diameter circular, sheet steel duct and has a typical flow rate of approximately 25 CFM. The purpose of the decay duct is to allow the gaseous effluents produced by the Unit 1 and 2 condenser air ejectors time to decay (approximately 3-4 hours) prior to venting. Gases exit the decay duct through a 4" schedule 40 carbon steel pipe to the auxiliary building ventilation stack.

2.5.1 EXISTING GASEOUS EFFLUENT MONITORING EQUIPMENT

The monitoring equipment (Channel CR-9) for the combined air ejector decay duct is a beta scintillation detector, with the display unit located in the control room. This equipment is designed to monitor air ejector exhausts at the very low levels of radiation associated with small steam generator tube leaks.

Radioiodines and particulates are monitored at a sampling location in the discharge of the duct. A sample pump is used to draw some of the effluent stream through a filter paper and a charcoal filter canister. The filter and charcoal canister are removed and laboratory analysis determines the radioiodine and particulate content.

2.6 GAS STRIPPER EXHAUST

The gas stripper building exhaust flows into a 36"-diameter, circular, sheet metal duct with a typical flow rate of approximately 13,000 CFM. The stack is located in the gas stripper building and vents that area through the Unit 2 containment purge vent stack.

2.6.1 EXISTING GASEOUS EFFLUENT MONITORING EQUIPMENT

The monitoring equipment is similar to that of the auxiliary building ventilation stack and consists of a Geiger-Mueller tube detector located in the vent duct with the display unit located in the control room. Radioiodine and particulates are sampled, collected on filters, and analyzed in the laboratory.

2.7 STEAM SAFETY VALVES AND ATMOSPHERIC DUMP VALVES

Each main steam line has a safety valve and atmospheric dump valve header upstream of the main steam isolation valve. On each header there are six exhaust paths: four steam safety valves and one atmospheric dump valve which exhaust directly to atmosphere and one supply line to the steam driven auxiliary feed pump turbine, which has an atmospheric exhaust. No installed monitoring is provided for these release paths because release of radioactive material occurs only in conjunction with primary to secondary leakage. The monitoring provided by the combined air ejector monitor provides early indication of primary to secondary leakage, allowing the isolation of the affected steam generator.

2.8 EVALUATION OF EXISTING SYSTEMS

The installed instrumentation on all of the ventilation ducts is designed for detecting the potential low-level releases from normal operation and accidents analyzed in the Final Facility Design and Safety Analysis Report. The potential release rates postulated in Table 2.1.8.b.2 of NUREG-0578 are significantly higher, so that the installation of high-range effluent monitors in each of the stacks is required.

Radioiodine and particulate monitoring capabilities are adequate to meet the interim requirements of NUREG-0578 with additional procedural precautions to ensure that the analytic results are not biased by external contaminants or noble gases and to ensure that exposure to persons handling and analyzing the filter media is minimized.

3.0 GASEOUS EFFLUENT MONITORING: DESIGN MODIFICATIONS

3.1 INTRODUCTION

As a result of the evaluation of the ranges of installed gaseous effluent monitoring equipment compared to the ranges of radiation levels expected from the concentrations postulated by the NRC, two specific types of changes have been made as required to meet the NUREG-0578, January 1, 1980, commitments. These are:

a. Installation of high-range, shielded radiation detectors with a display unit in the control room for monitoring noble gas effluents;

b. Use of portable survey instruments to permit the estimation of high-level release rates.

Alternative methods were investigated for monitoring high-level gaseous effluents, but these methods were selected based on manpower limitations, cost, flexibility, and ease of use. Table 1 lists the equipment which is installed for noble gas effluent monitoring.

As a result of the evaluation of auxiliary building access times and current radioiodine and particulate sampling methods, it has been determined that, with improved procedural controls, the existing methods of obtaining and processing filters and charcoal canisters for collecting and analyzing particulates and radioiodines comply with the interim requirements of NUREG-0578.

3.2 AUXILIARY BUILDING VENTILATION STACK, GAS STRIPPER STACK, AND DRUMMING AREA VENTILATION STACK

These ventilation stacks will be discussed together because of the similarity of the modification made on each.

From each duct a sample is withdrawn at a constant flow rate with a sample pump. The sampling line provides a suitable location for high-range effluent monitoring. On the suction line of the pump at each location, an expansion chamber is installed so that the detectors can see a source of sufficient strength and known volume to provide accurate measurement in the range of interest. An Eberline RMS II Detector Model DA1-4 is mounted in contact with the expansion chamber and is connected to an electronics channel (Model EC1-3 for the drumming area ventilation stack, Model EC1-4 for the gas stripper stack and the auxiliary buiding ventilation stack) which gives readings in the range of interest on a log rate meter displayed in the control room. The sensors and the readouts will eventually be powered from an instrument bus which is supplied by an emergency power source. Each of the detector/expansion chamber assemblies is shielded from background radiation by the equivalent of 4" of lead to minimize the effect of background radiation.

For the auxiliary building ventilation stack, the range of detection for the Eberline RMS II system is 1.06 uCi/cc to 1.06E+04 uCi/cc.

For the drumming area ventilation stack, the range of detection is 1.06E-01 uCi/cc to 1.06E+03 uCi/cc.

For the gas stripper stack, the range of detection is 1.06 uCi/cc to 1.06E+04 uCi/cc.

In the event of releases in the range not covered by installed pre-existing equipment designed for normal operation and not covered by the new high-range Eberline RMS II System, direct measurements taken at the duct provide the required information on release rates.

Figures 4, 7, and 12 provide the relationship between meter readings and release rate when measurements are taken directly on the duct for the auxiliary building ventilation stack, the drumming area ventilation stack, and the gas stripper stack, respectively.

Figures 13, 16, and 17 provide the relationship between meter readings and release rate when measurements are taken at the expansion chamber mounted on the sample pump suction line for the auxiliary building ventilation stack, the drumming area ventilation stack, and the gas stripper stack, respectively.

3.3 CONTAINMENT PURGE STACKS

Each of the containment purge stacks has the capability of being monitored by the process radiation monitoring channel (R11/R12) which normally monitors containment gaseous and particulate activity . The alignment to monitor the containment purge stack

can be accomplished remotely from the main control room. A sample is pumped from the containment purge stack to the R11/R12 monitors and returned to the stack.

For each unit, an Eberline RMS II Detector Model DA1-5 is mounted in contact with the 3/4" sample tubing of the R11/R12 channel and is connected to an electronics channel Model EC1-5 which gives readings in the range of interest on a log rate meter displayed in the control room. The sensors and the readouts will eventually be powered from an instrument bus which is supplied by an emergency power source. At each of the detector locations, the detector and the tubing is shielded by the equivalent of 4" of lead to minimize the effect of background radiation on the detector readings and to ensure that the detector sees a collimated source of known volume and that the geometry is the same for both units.

For Unit 1, the detector is located on a vertical run of tubing in the facade area adjacent to the East wall of the facade at the 66' elevation.

For Unit 2, the detector is located on a horizontal run of tubing in the facade area directly above the Unit 2 R11/R12 cubicle at an elevation of approximately 63'.

For both units, the range of detection is 2.8E+02 uCi/cc to 2.8E+06 uCi/cc.

In the event of releases in the range not covered by installed pre-existing equipment and not covered by the new high-range Eberline RMS II System, direct measurements taken at the duct provide the required information on release rates.

Figures 9 and 10 provide the relationship between meter readings and release rate when measurements are taken directly on the duct with one fan operating and with two fans operating, respectively.

Figures 14 and 15 provide the relationship between meter readings and release rates when the measurements are obtained at the detectors mounted on the R11/R12 sample lines with one fan operating and two fans operating, respectively.

3.4 COMBINED AIR EJECTOR DECAY DUCT

The discharge from the main condenser air ejectors from both units is diverted into a 36" duct for the purpose of delay release in order that short-lived radioactive contaminants might decay. At the discharge of the 36" duct, the air ejector stream is exhausted through a 4" schedule 40 carbon steel pipe.

An Eberline RMS II Model DA1-4 Detector is mounted in contact with the 4" schedule 40 exhaust pipe and is connected to an electronics channel Model EC1-3 which provides readings in the range of interest on a log rate meter displayed in the control room. The sensor and the readouts will eventually be powered from an instrument bus which is supplied by an emergency power source. The detector location is shielded to minimize the effect of background radiation on the detector readings and to ensure that the detector sees a collimated source of known volume.

The detector is located on the west wall of the turbine building where the combined air ejector decay duct exhaust penetrates the wall at an elevation of approximately 56'.

The range of detection is 2.72 uCi/cc to 2.72E+04 uCi/cc.

In the event of releases in the range not covered by installed pre-existing equipment and not covered by the new high-range Eberline RMS II System, direct measurements taken at the duct provide the required information on release rates.

Figure 11 shows the relationship between meter readings and release rate when measurements are taken directly on the duct.

Figure 18 shows the relationship between meter readings and release rate when measurements are taken at the detector in contact with the 4" schedule 40 exhaust pipe.

3.5 STEAM SAFETY VALVES AND ATMOSPHERIC DUMP VALVES

On each main steam line, a monitoring location has been identified upstream of the steam safety valves and atmospheric dump valves. Direct measurements taken with portable survey instruments at these locations will provide the required information on release concentrations. Estimates of the amount of steam released, or the release flow rate will then be used to determine release amounts or release rates.

Figure 25 shows the relationship between meter readings and release concentration when the readings are taken on the steam safety valve and atmospheric dump valve header.

4.0 REFERENCES

1. Drawings

   a. Heating, ventilation and air conditioning drawings for auxiliary building, containment and drumming area. Drawing Nos. M-114, 116, 118, 119, 120 to 133, M-145 to 149.

   b. General plan views: M1-9.

   c. Miscellaneous drawings:

      E83D195-electrical one-line containment isolation signal path

M-2215 -- P&ID: heating and ventilation

M-2097 -- containment post-accident sampling system

G-276-P -- noble gas system

F-2069P -- let-down gas stripper

G-276-P -- cryogenic noble gas removal

C-342-344 -- construction details of various plant vents

2. FFDSAR Sections

   a. Section 9.6.3: auxiliary building ventilation system

   b. Section 11.2: radiation monitoring system

3. Radiation monitoring system: Operating and Maintenance Manual, Tracer Laboratories.

4. ISOSHLD Computer Runs, EDS Version 1, 12/05/79

   a. 12/13/79 Run, 20.45.28, Various Plant Stacks and 3/8" Sample Pipe

   b. 12/20/79 Run, 13.09.31, 3/8" Tube, 1/2" Tube, 5/8" Tube

   c. 1/15/80 Run, 17.58.37 4" SCH 40 Pipe

   d. 1/15/80 Run, 17.58.40, 3/4" Tube

e. 1/19/80 Run, 14.39.18, 8 by 8 Inch Duct 9 In Long

5. NRC Documents

a. Harold Denton (NRC)/All Nuclear Power Plants, letter dated October 30, 1979, "Division of Lessons Learned Short-Term Requirements"

b. Office of Nuclear Reactor Regulation, USNRC, NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations"

6. WEPCO DOCUMENTS

a. Burstein (WEPCO)/Denton (NRC) letter dated October 20, 1979, "Dockets 50-266 and 50-301 Implementation of NUREG-0578, Point Beach Nuclear Plant Units 1 and 2"

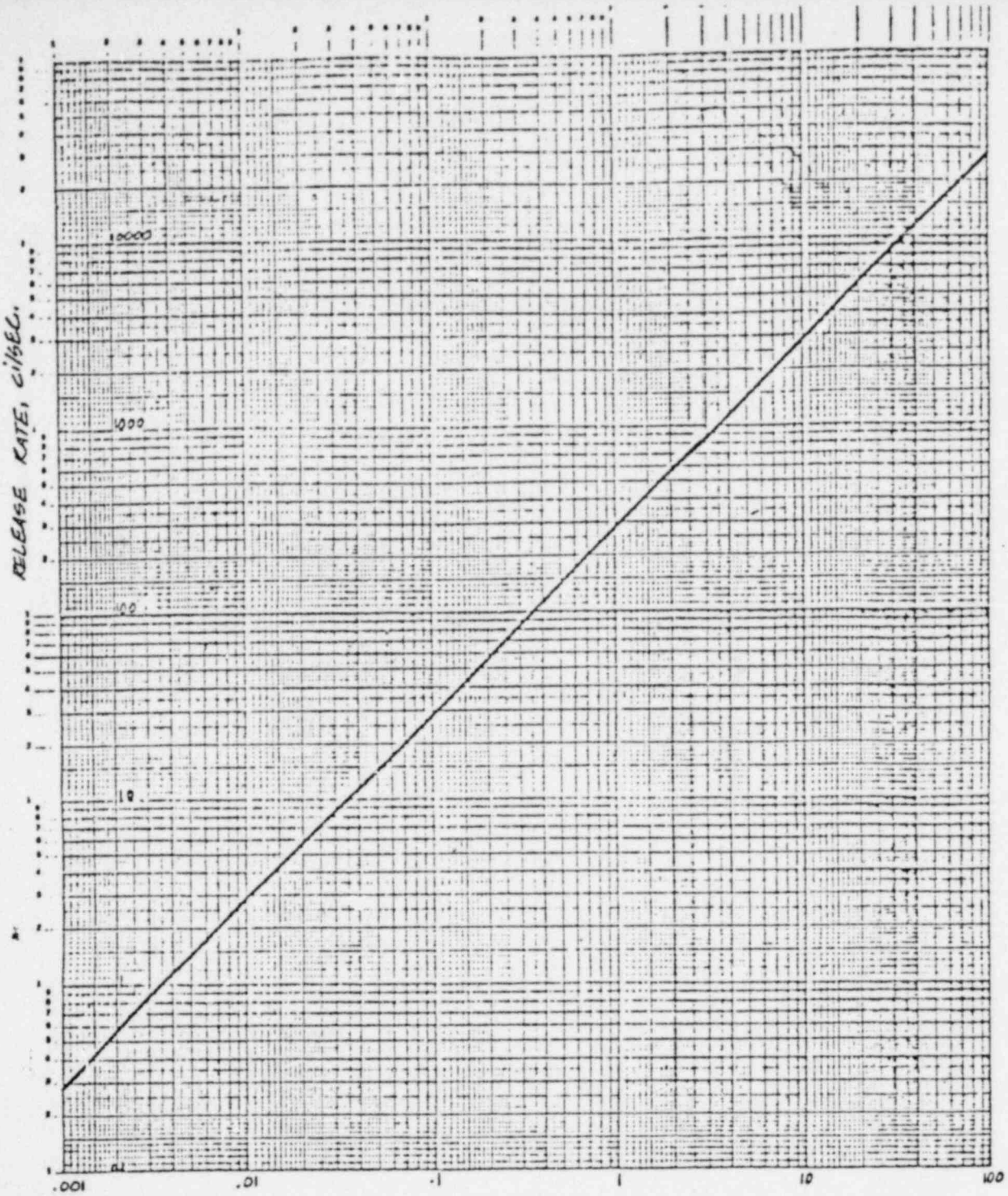
b. Burstein (WEPCO)/Denton (NRC) letter dated November 27, 1979, "Dockets 50-266 and 50-301 Implementation of NUREG-0578, Point Beach Nuclear Plant, Units 1 and 2"

TABLE 1

HIGH-RANGE NOBLE GASEOUS EFFLUENT MONITORING EQUIPMENT

VENDOR: Eberline
P.O. Box 2108
Santa Fe, New Mexico 87501

Effluent Stack	Electronic Channel	Detector Model	Detection Range (R/Hr)	Detection Range (uci/cc)	Detection Required by Table 2.1.8.b.2 (NUREG-0578) (uci/cc)
Auxiliary Building Ventilation Stack	EC1-4	DA1-4	10^{-2} - 10^{2}	1.06 to 1.06×10^{4}	10^{3}
Drumming Area Ventilation Stack	EC1-3	DA1-4	10^{-3} - 10^{1}	1.06×10^{-1} to 1.06×10^{3}	10^{2}
Containment Purge Vent Stacks	EC1-5	DA1-5	10^{-1} - 10^{3}	2.80×10^{2} to 2.80×10^{6}	10^{5}
Combined Air Ejector Decay Duct	EC1-3	DA1-4	10^{-3} - 10^{1}	2.72 to 2.72×10^{4}	10^{4}
Gas Stripper Stack	EC1-4	DA1-4	10^{-2} - 10^{2}	1.06 to 1.06×10^{4}	10^{3}



METER READING R/HR.

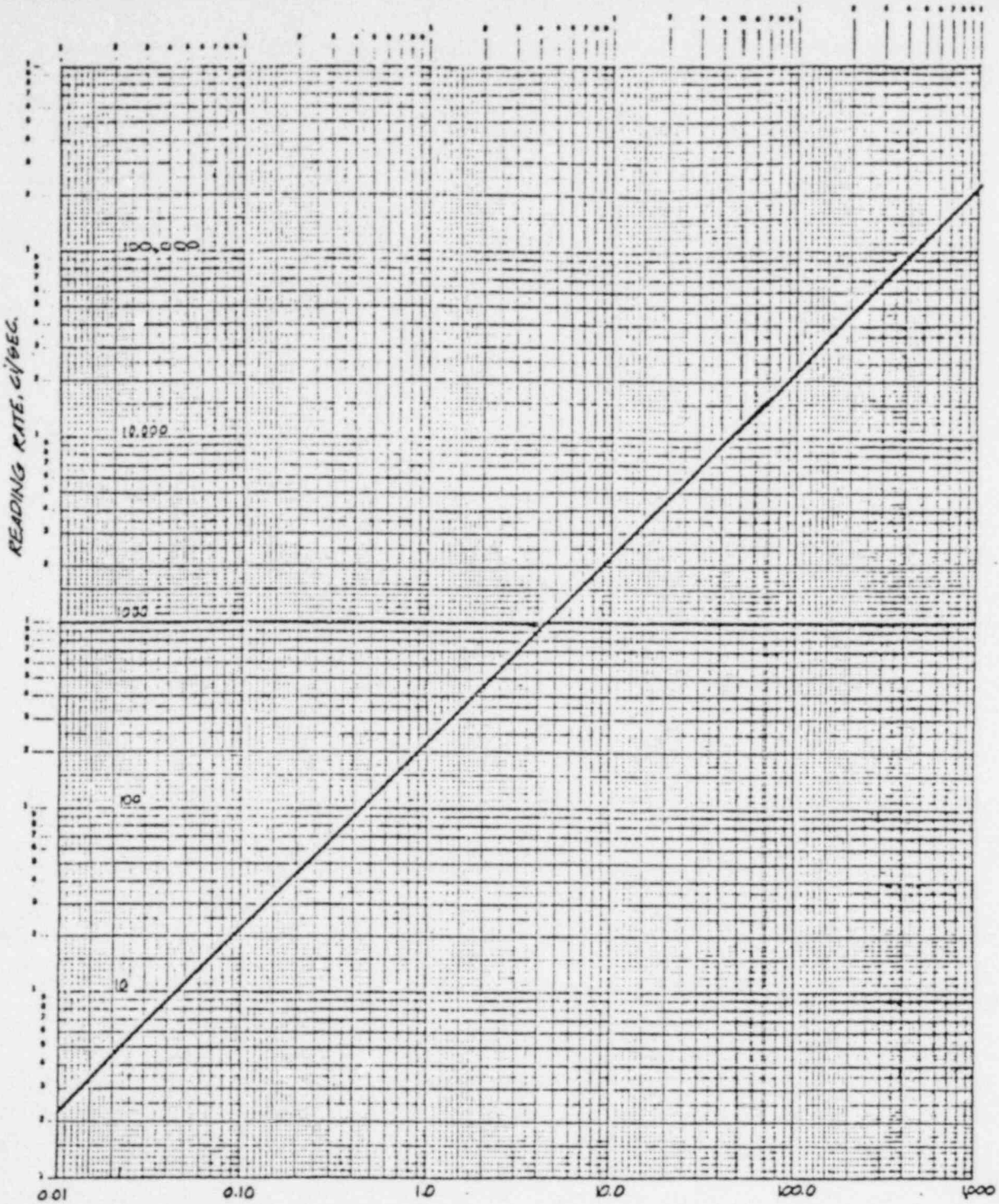
POOR ORIGINAL

REV	DATE	REVISIONS	BY	REVD	APPR'D
A	4.11.77	INITIAL ISSUE FEBRUARY	HLR	MAM	
B	1.26.80	REVISION	HLR	MAM	MRM

eds nuclear

AUXILARY BUILDING STACK R/HR. TO CURIES. CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY
MILWAUKEE, WISCONSIN

SCALE N.T.S.
JOB NO 0870-001
DRAWING NO FIGURE 1 REF. 2.1.5 D
REV B
SHEET 1 OF 1



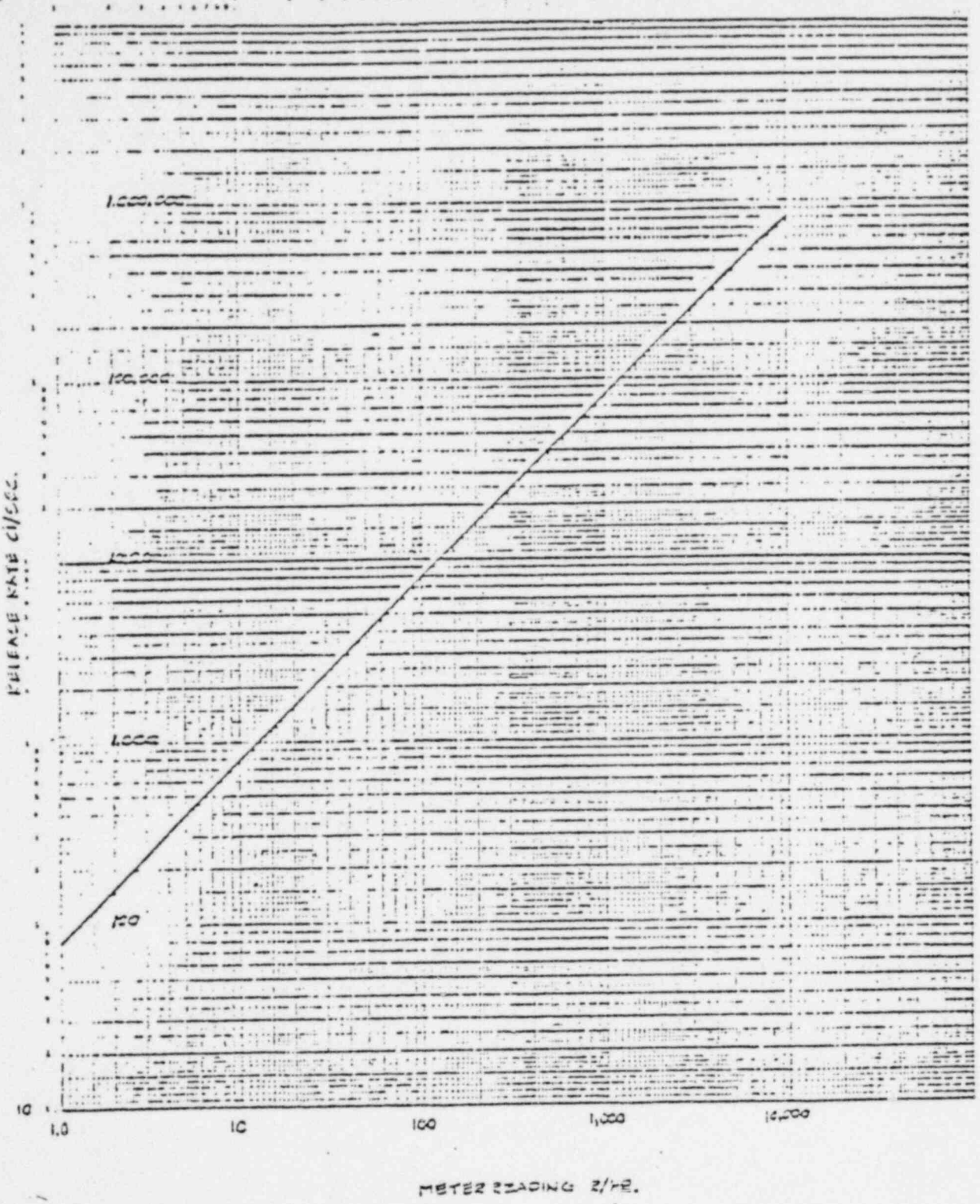
POOR ORIGINAL

REV.	DATE	REVISIONS	BY	REVD	APPR'D
A	4/1/M	INITIAL ISSUE PRELIMINARY	HLR	WAM	
B	11/22/81	REVISION	HLR	WAM	HRM

eds nuclear

DRUMMING AREA R/HR. CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY, MILWAUKEE, WISCONSIN

SCALE N.T.S.
JOB NO. 0870-001
DRAWING NO. FIGURE 7 SEC. 6.1.8.b
REV. B
SHEET 1 OF 1



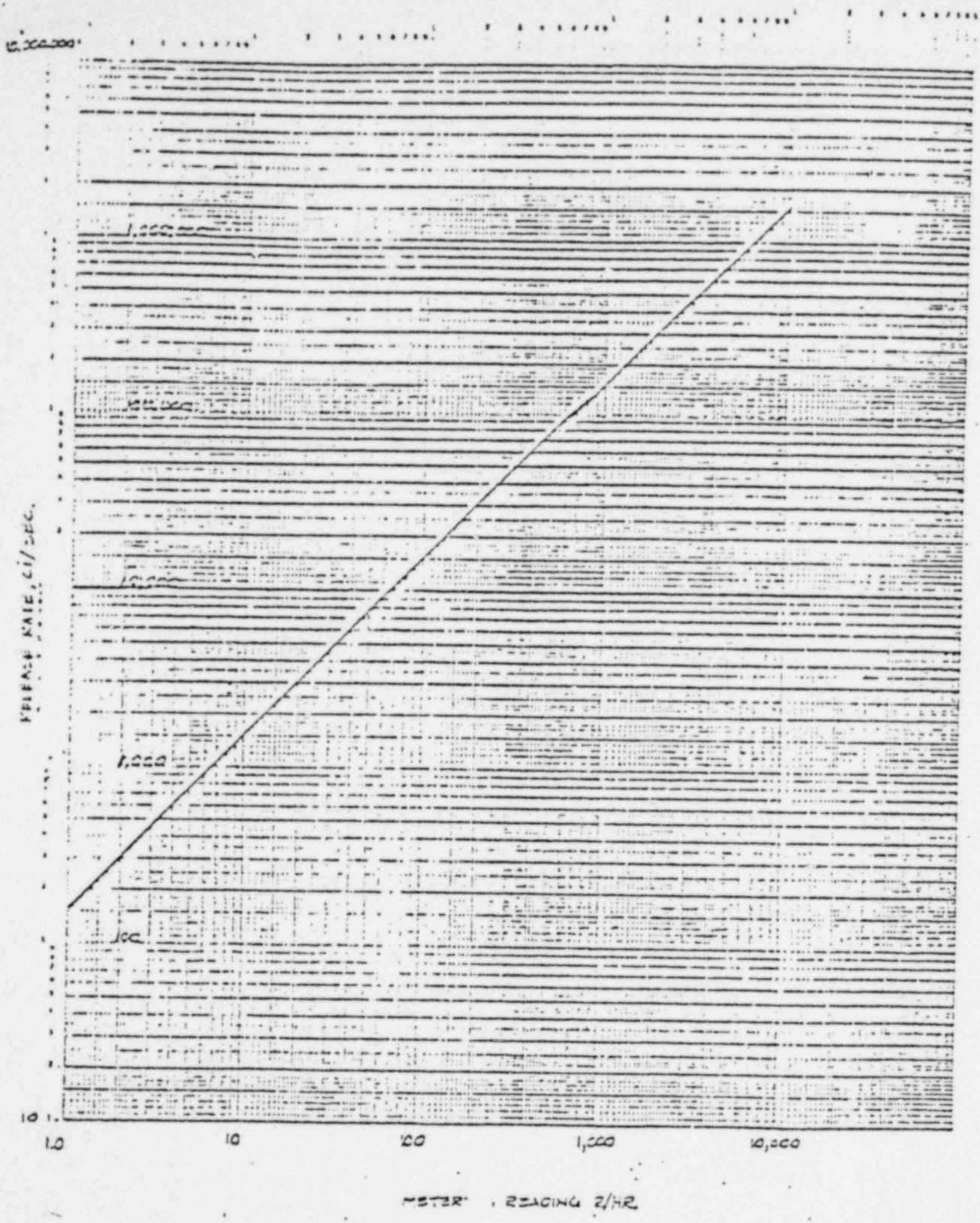
POOR ORIGINAL

REV	DATE	REVISIONS	BY	REVIEWED	
A	11/22/80	INITIAL [illegible]	[illegible]	[illegible]	[illegible]

ECS NUCLEAR

CONTAINMENT PURGE EXHAUST R/HR (1 FT) TO Ci/SEC. CONVERSION CHART
POINT BEACH NUCLEAR PLANT UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY, MILWAUKEE, WISCONSIN

SCALE: [illegible]
DRAWING NO. FIGURE 4 | REV. A
SHEET [illegible] OF [illegible]



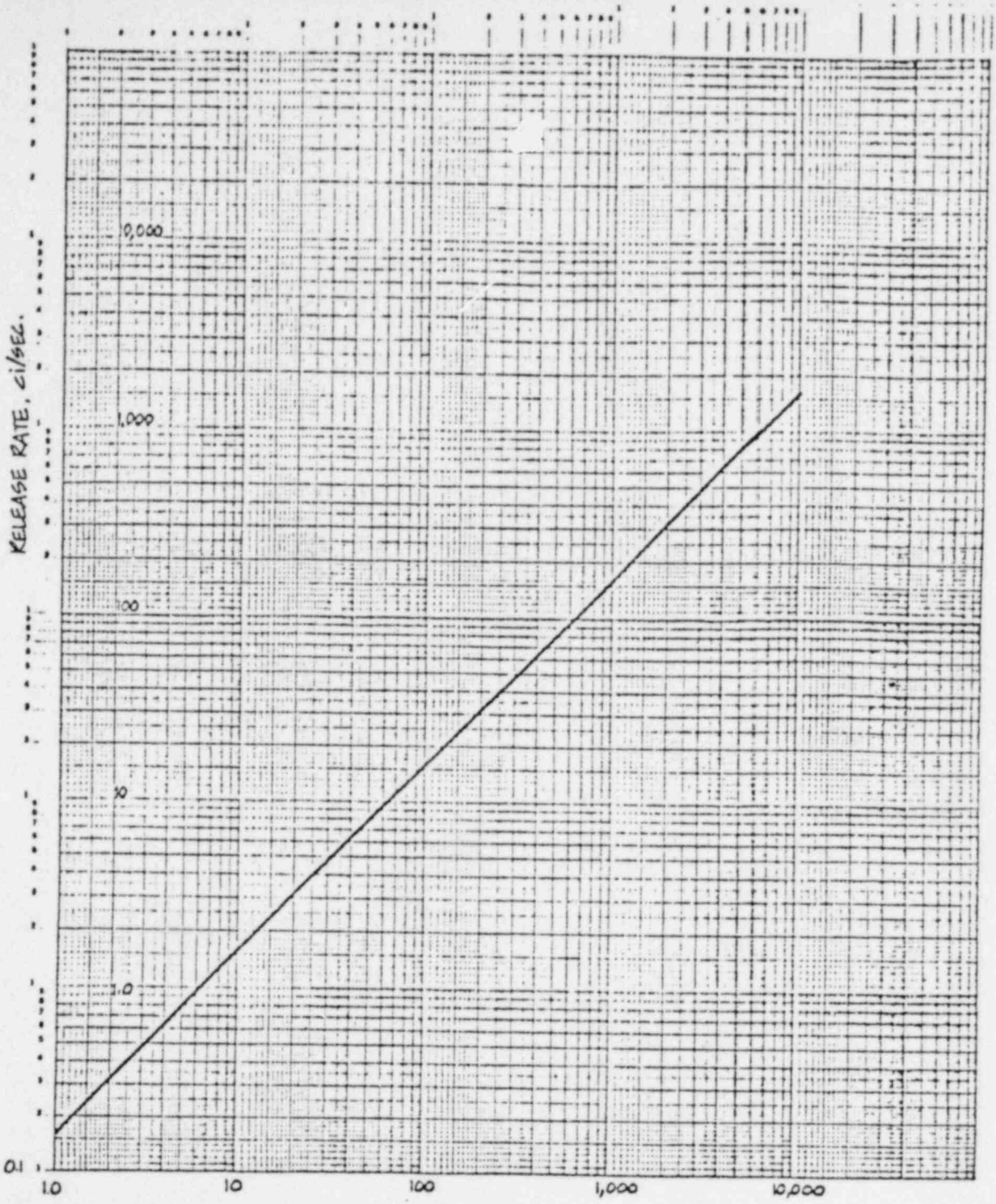
POOR ORIGINAL

REV. DATE	REVISIONS	BY	REVIEWED BY
A	INITIAL ISSUE		

BCS nuclear

CONTAINMENT PURGE EXHAUST
3 (FANS) R/HR TO CI/SEC.
CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER
COMPANY, MILWAUKEE, WISCONSIN

SCALE NONE
JOB NO. 2470-30
DRAWING NO. REV
FIGURE 0
ACC. 2.1.2.D A
SHEET 1 OF 1



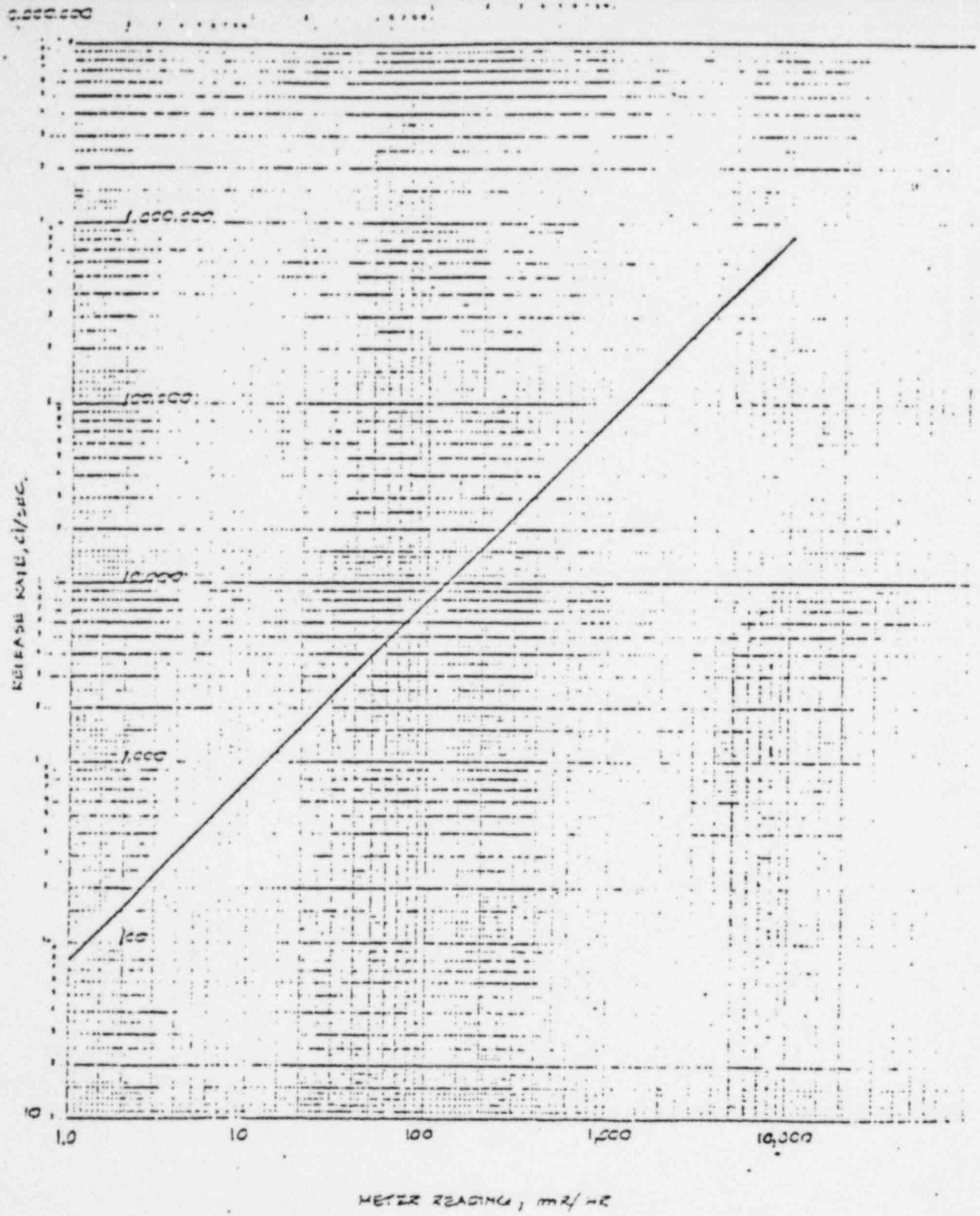
POOR ORIGINAL

REV	DATE	REVISIONS	BY	REVD	APPR'D
A	12/11/81	INITIAL ISSUE - PRELIMINARY	WLR	NPW	
B	2/18/82	REVISION	WLR	NPW	M.A.

eds nuclear

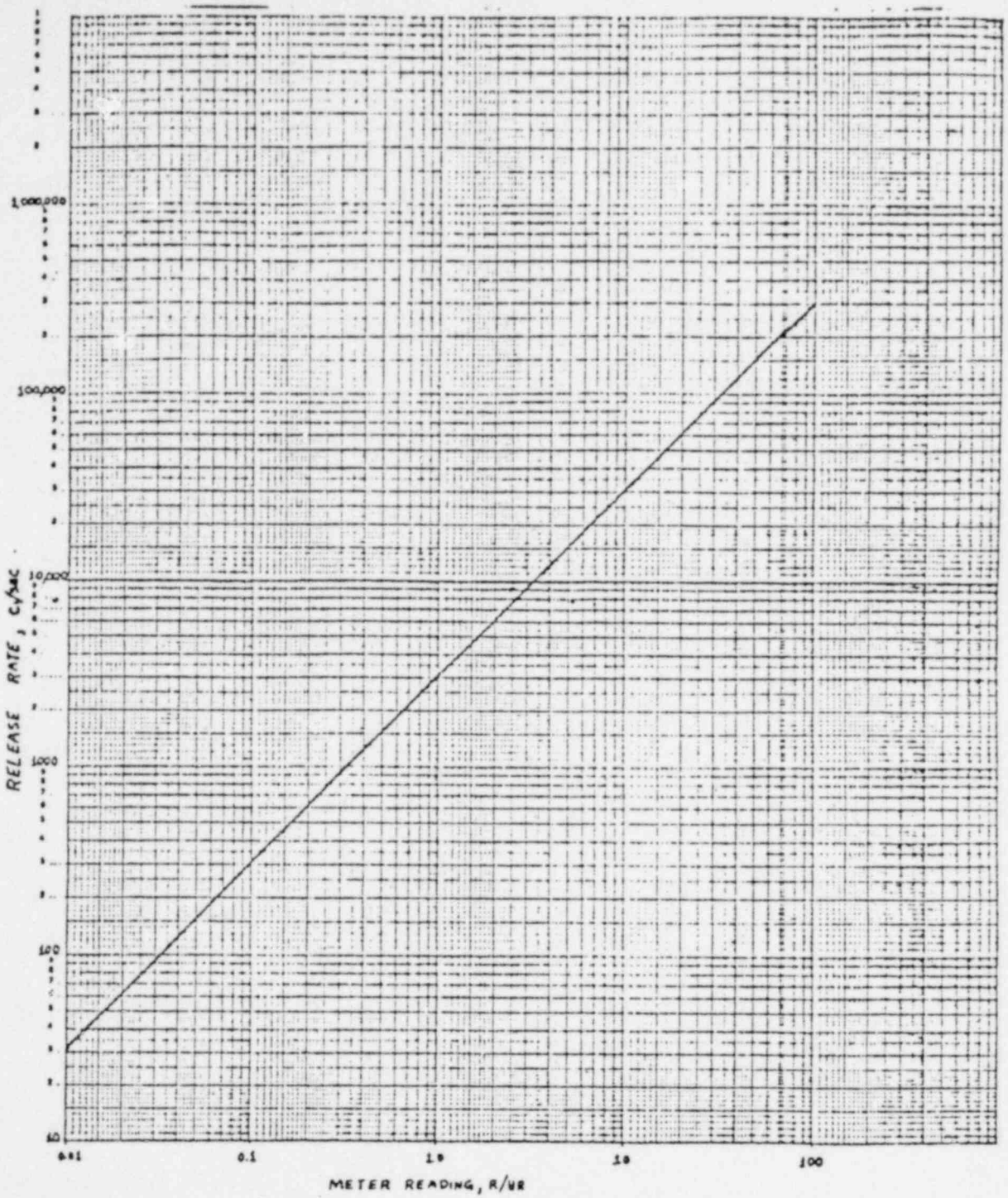
COMBINED AR EJECTOR PLUS PLANT R/HR TO ci/SEC. CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY
MILWAUKEE, WISCONSIN

SCALE N.T.S.
JOB NO 0870-001
DRAWING NO: FIGURE 11 SE.2.1.8b
REV: B
SHEET 1 OF 1



POOR ORIGINAL

DATE	REVISIONS	BY	APPR.		BGS nuclear	GAS STRIPPER SCALE FOR TO DIESEL CONVERSION CURVE POINT BEACH NUCLEAR PLANT UNITS 1 & 2 WISCONSIN ELECTRIC POWER COMPANY MILWAUKEE, WISCONSIN	SCALE NONE	DRAWING NO. REV. FIGURE 12 SEP. 2, 80	A	SHEET OF



(MEASUREMENT AT 8"x8"x9" EXPANSION CHAMBER)

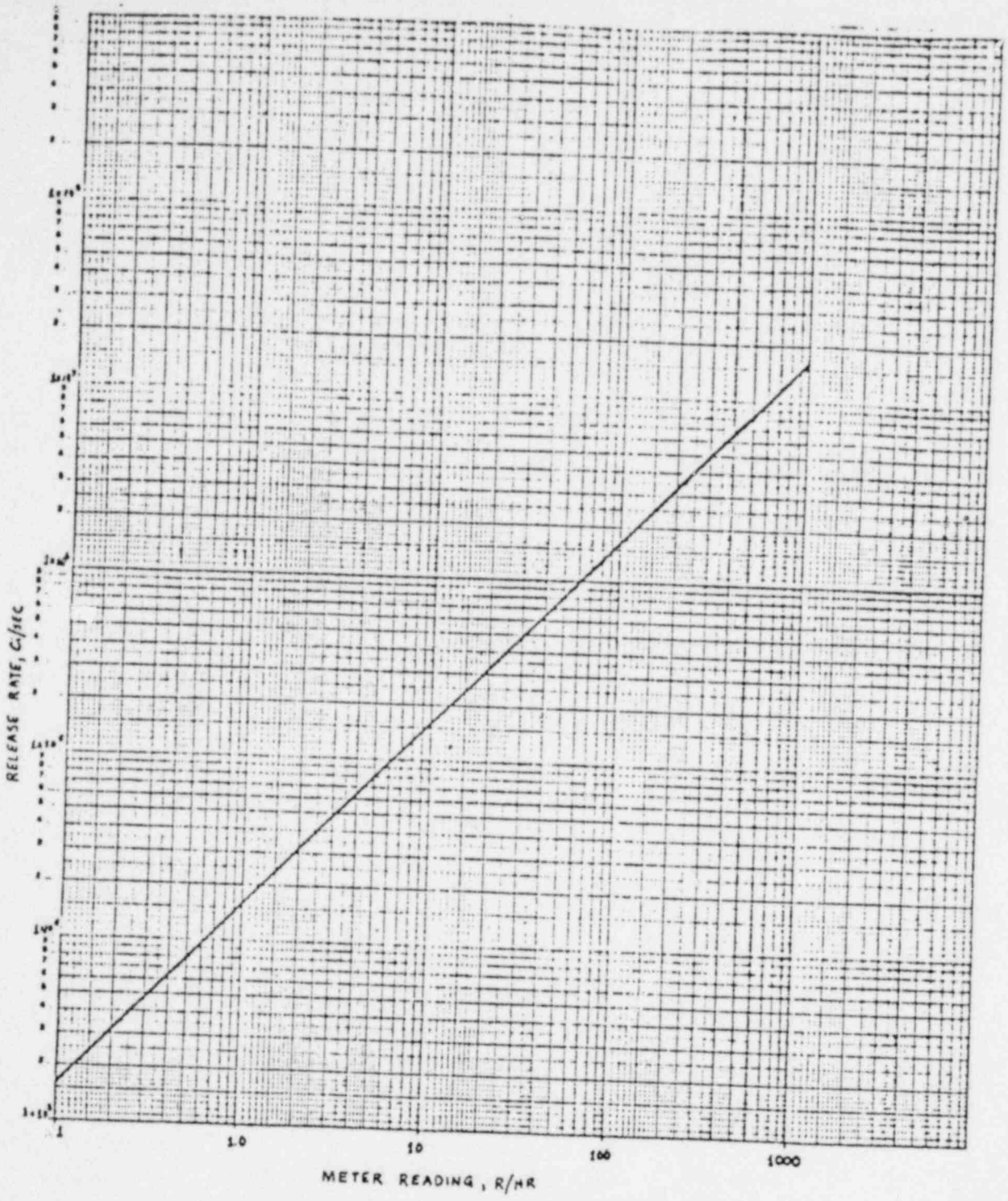
POOR ORIGINAL

REV.	DATE	REVISIONS	BY	REV'D	APPR'D
A	1/22/80	INITIAL ISSUE PRELIMINARY	HR	[illegible]	[illegible]
B	1/5/80	REVISION	HR	WPM	[illegible]

eds nuclear

AUXILIARY BUILDING STACK R/HR TO CI/SEC. CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY
MILWAUKEE, WISCONSIN

SCALE: NT.S.
JOB NO. 0830-001
DRAWING NO. E.J.B.D. FIGURE 13
REV. B
SHEET 1 OF 1



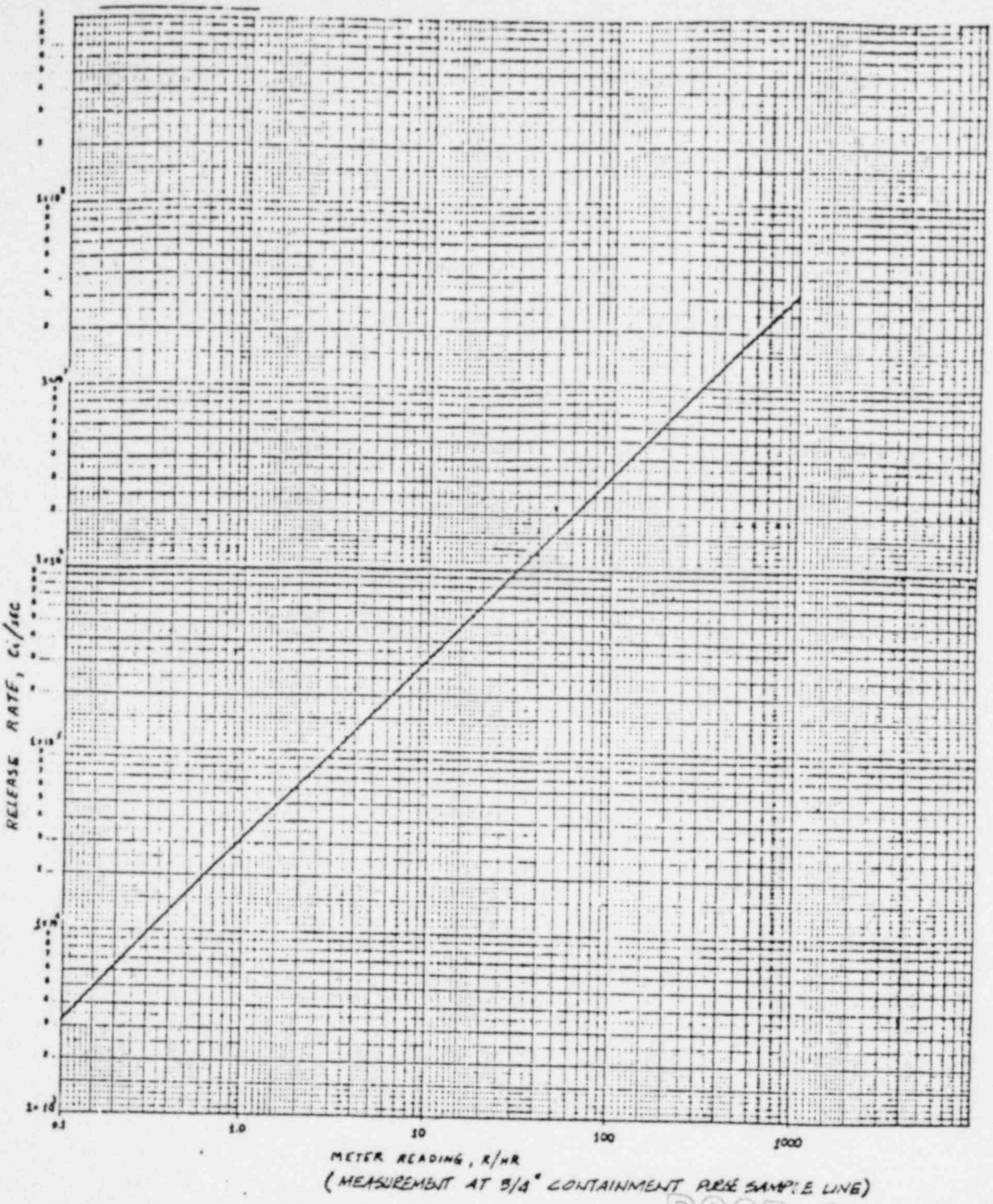
POOR ORIGINAL

REV.	DATE	REVISIONS	BY	REVIEW	APPR'D
A	1/28/80	INITIAL ISSUE PRELIMINARY	HR		

eds nuclear

CONTAINMENT PURGE EXHAUST (1 FAN)
R/HR TO C/SEC CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNIT 1 & 2
WISCONSIN ELECTRIC POWER Co.
MILWAUKEE, WISCONSIN

SCALE N.T.S.	
JOB NO. 0870-001	
DRAWING NO	REV
2.1.8 D FIGURE 14	A
SHEET 1 OF 1	



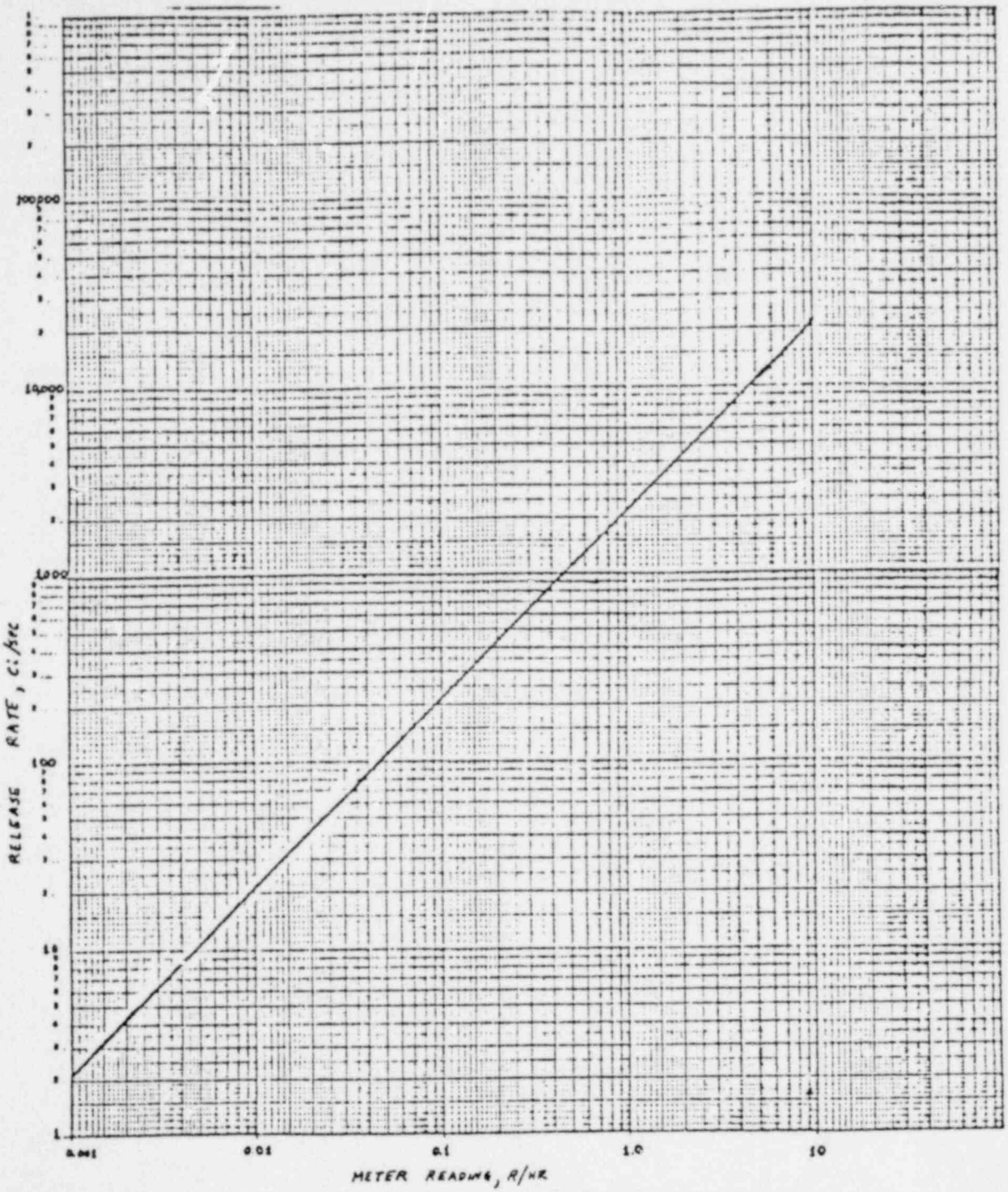
POOR ORIGINAL

REV	DATE	REVISIONS	BY	REVD	APPR'D
A	1/[illegible]	INITIAL ISSUE PRELIMINARY	[illegible]	[illegible]	[illegible]

eds nuclear

CONTAINMENT PURGE EFFLUENT (2 FAN) R/HR TO Ci/SEC CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER Co.
MILWAUKEE, WISCONSIN

SCALE N.T.S.	
JOB NO. 0870-001	
DRAWING NO	REV
1.1.B.D FIGURE 15	A
SHEET 1 OF 1	



(MEASUREMENT AT 8"x8"x9" EXPANSION CHAMBER)

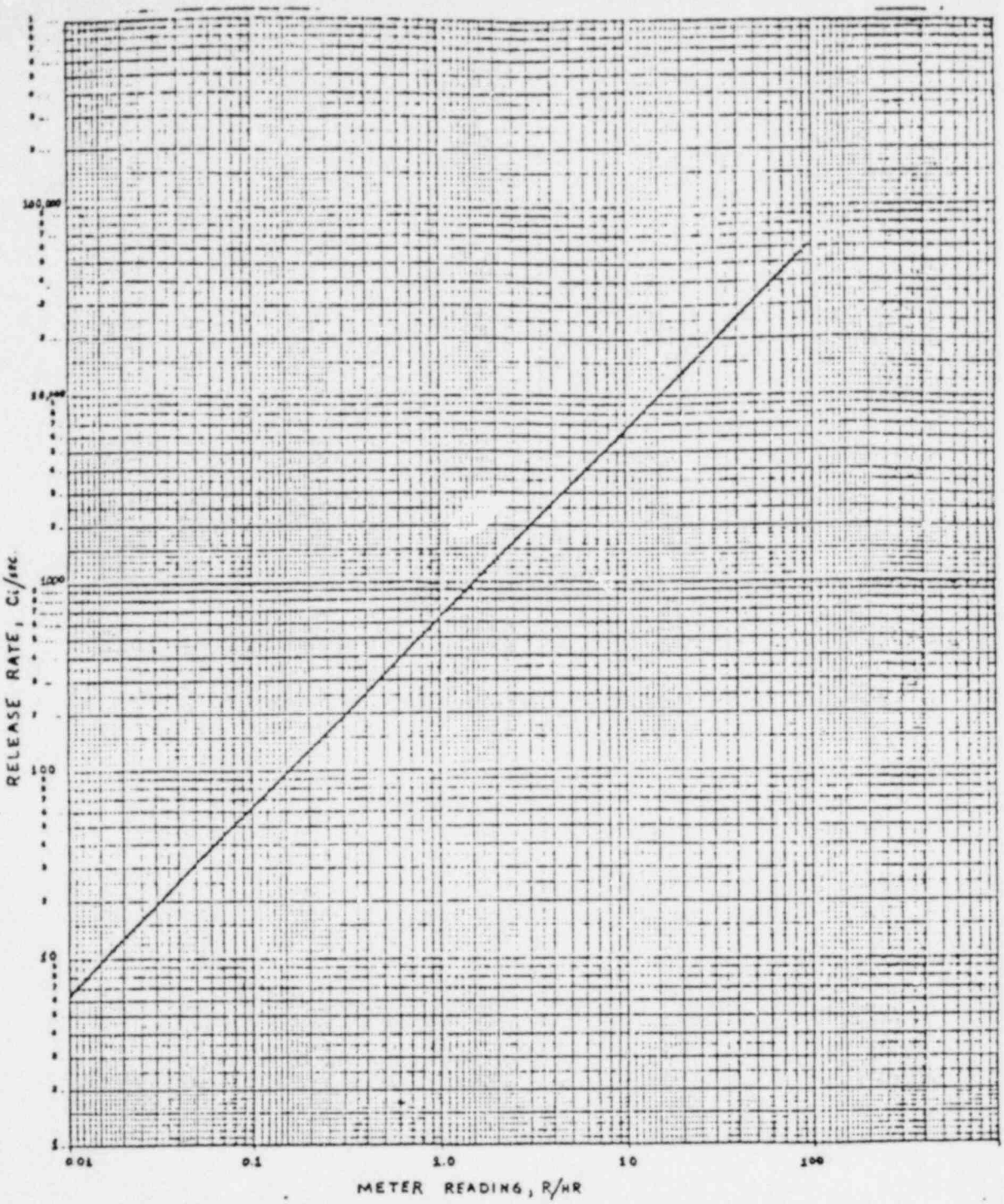
POOR ORIGINAL

REV	DATE	REVISIONS	BY	REVW	APPR'D
A	12/31/80	INITIAL ISSUE PRELIMINARY	[illegible]	[illegible]	[illegible]
B	4/5/80	REVISION	[illegible]	[illegible]	[illegible]

eds nuclear

DRUMMINS AREA R/HR
TO Ci/SEC. CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY
MILWAUKEE, WISCONSIN

SCALE	N.T.S.
JOB NO.	C370-001
DRAWING NO	REV
2.1.8D FIGURE 16	D
SHEET 1 OF 1	



(MEASUREMENT AT 8"x8"x9" EXPANSION CHAMBER)

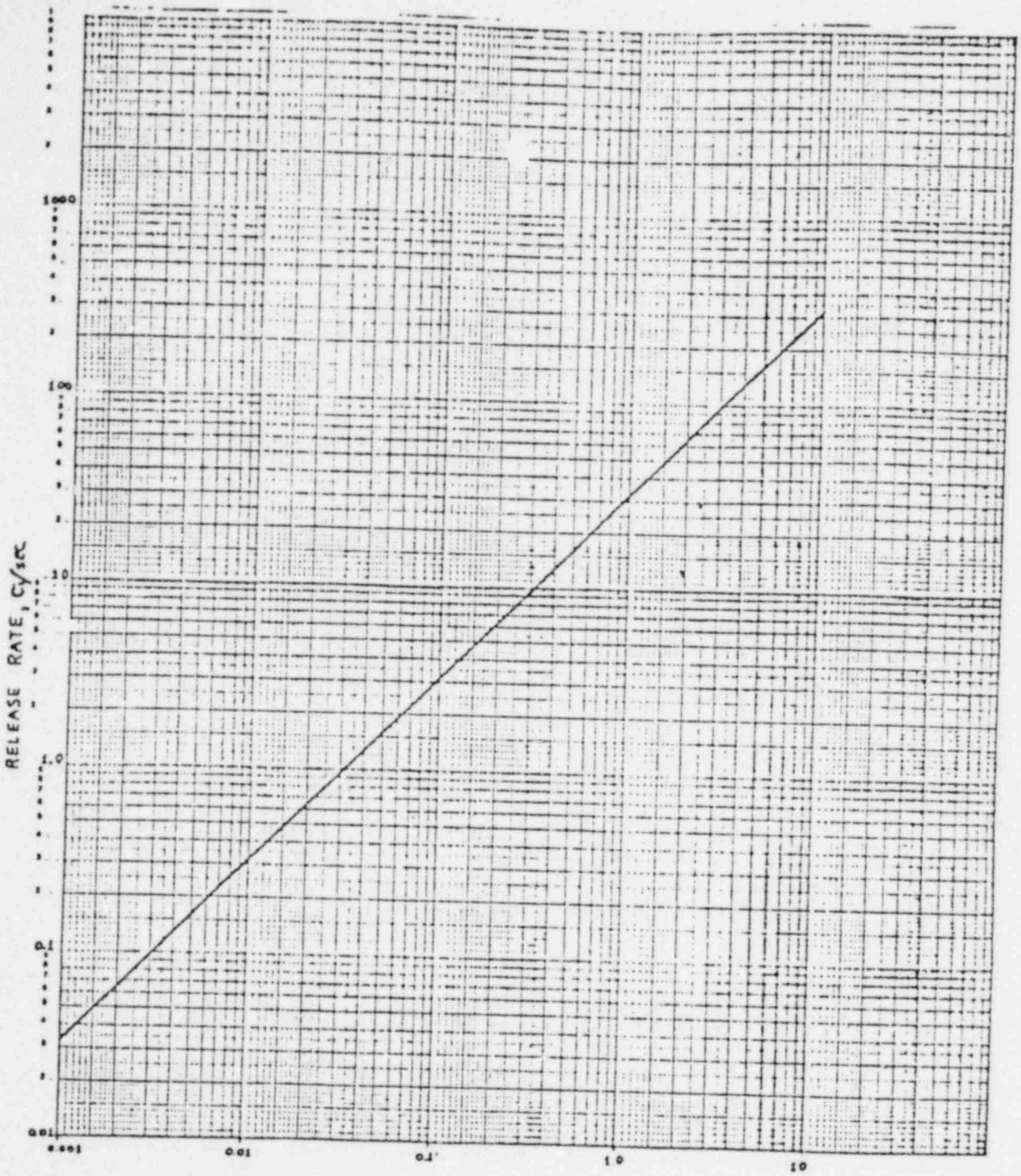
REV.	DATE	REVISIONS	BY	REV'D	APPR'D
A	1/27/80	INITIAL ISSUE PRELIMINARY	[illegible]	[illegible]	MJM
B	2/5/80	REVISION	[illegible]	KM	[illegible]

eds nuclear

GAS STRIPPER STACK R/HR TO Ci/SEC. CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER COMPANY
MILWAUKEE, WISCONSIN

SCALE N.T.S.
JOB NO. 0970-001
DRAWING NO. 2.1.8 D FIGURE 17
REV. B
SHEET 1 OF 1

POOR ORIGINAL



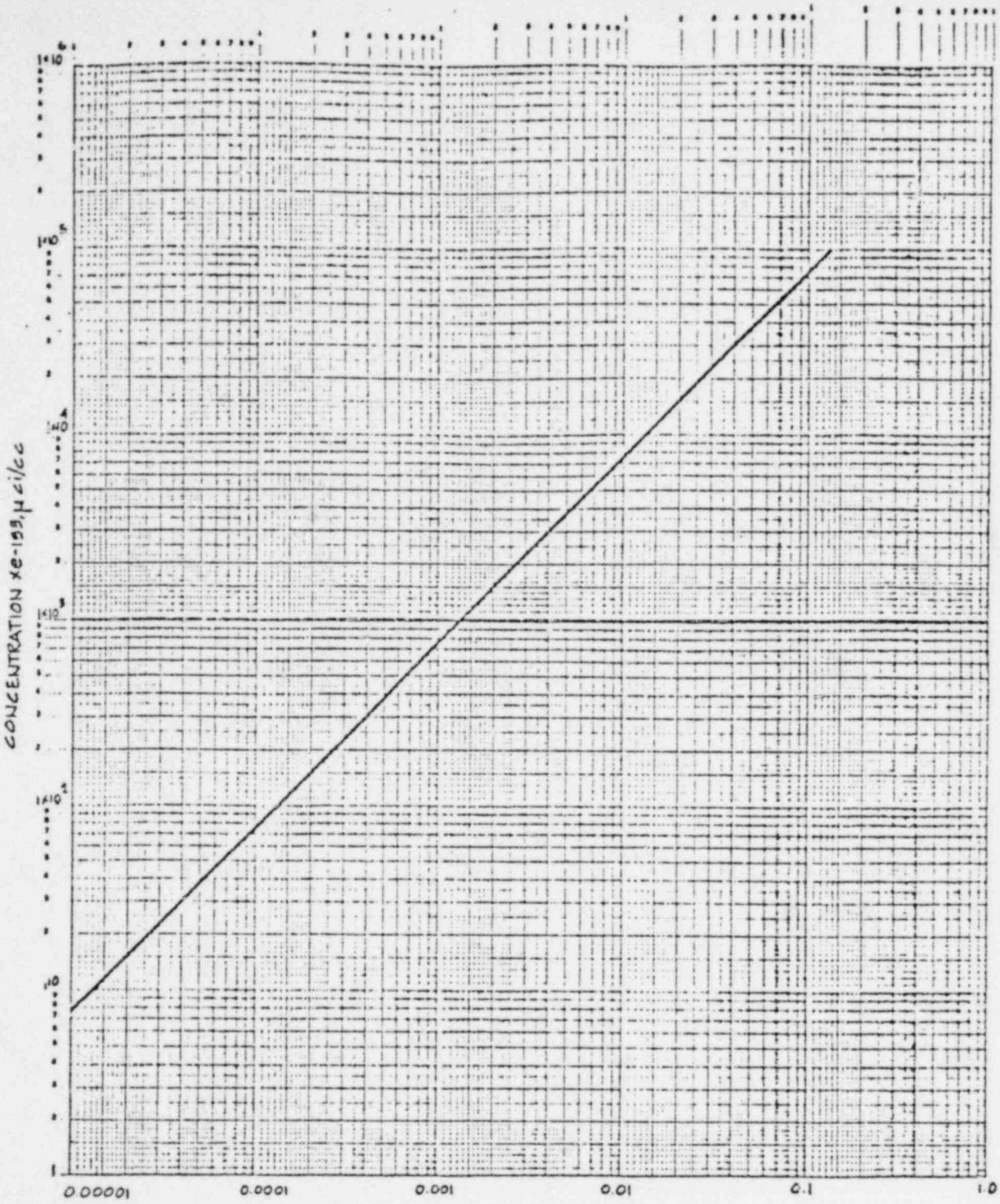
POOR ORIGINAL

REV	DATE	REVISIONS	BY	REVD	APPR'D
A	1-22-80	INITIAL ISSUE - PRELIMINARY	[illegible]	[illegible]	M/M

eds nuclear

COMBINED AIR EJECTOR GRAY DUCT MONITOR TO Ci/SEC CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNITS 1 & 2
WISCONSIN ELECTRIC POWER
MILWAUKEE, WISCONSIN

SCALE N.T.S.
JOB NO. 2570-001
DRAWING NO. 2.1.8b FIGURE 18
REV A
SHEET 1 OF 1



METER READING, R/HR

MEASUREMENT AT STEAMLINE SAFETY VALVE
AND ATMOSPHERIC DUMP VALVE HEADER

REV	DATE	REVISIONS	BY	REVD	APPR'D
A	5/7/80	INITIAL ISSUE - PRELIMINARY	KJR	[illegible]	[illegible]

eds nuclear

STEAMLINE SAFETY VALVE AND ATMOSPHERIC DUMP VALVE EXHAUST R/HR TO μ CI/CC CONVERSION CHART
POINT BEACH NUCLEAR PLANT
UNIT 1 & 2
WISCONSIN ELECTRIC POWER CO.
MILWAUKEE, WISCONSIN

SCALE N.T.S.
JOB NO 0870-001
DRAWING NO 5.1.80 FIGURE 15
REV A
SHEET ___ OF ___

POOR ORIGINAL