ARKANSAS POWER & LIGHT COMPANY
POST OFFICE BOX 551 LITTLE ROCK, ARKANSAS 72203 (501) 371-4000

January 18, 1980

1-010-22
2-010-18

Director of Nuclear Reactor Regulation
ATTN: Mr. Darrell G. Eisenhut, Acting Director
Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Arkansas Nuclear One-Units 1 & 2
Docket Nos. 50-313 and 50-368
License Nos. DPR 51 & NPF 6
Lessons Learned Implementation
(File: 1510.3, 2-1510.3)

Gentlemen:

Mr. H. Denton's letter of October 30, 1979, requested Arkansas Power and Light Company to document our method of compliance with the "short-term" Lessons Learned requirements which do not require prior NRC approval. As committed in our November 20, 1979 letter, a description of the methods of implementation/compliance for those items is hereby provided.

Very truly yours,

David C. Trimble

David C. Trimble
Manager, Licensing

DCT:nak

Attachment

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Item 2.1.1 Emergency Power Supply

ANO-1

We have determined that 126 Kw of pressurizer heating is required within two hours of a loss of offsite power to establish and maintain natural circulation of the RCS if forced circulation is lost. The ANO-1 pressurizer has available 84 Kw of proportional heaters on each channel of the class 1E power system. During this January-February 1980 outage, forty-two Kw of backup (on-off) heaters will be added to the swing bus (connected to either one of the diesel generators). The combination will produce the necessary 126 Kw.

The diesel generator loading was reviewed with the conclusion that the additional load is permissible. The added 42 Kw will neither effect the diesel loading sequencing nor exceed the seven day rating. Therefore, these non-class 1E loads will be automatically loaded and will not be shed upon SIAS actuation. The class 1E interfaces for main power and control power will be protected by safety grade circuit breakers.

The ANO-1 pressurizer level instrument channels are presently powered from vital instrument buses.

The ANO-1 PORV motive and control power is presently provided from the channel 1 (red) D.C. system. The PORV block valve, following this shutdown, will be powered from the channel 2 (green) A.C. system.

The changeovers of motive and control power from the normal offsite power source to the emergency power source will be automatic.

ANO-2

We have determined that 150 Kw of pressurizer heating is needed within 30 minutes of loss of offsite power to maintain natural circulation. One Hundred Fifty Kw of proportional heaters is currently powered from each safety bus.

The existing control circuit for the proportional heaters will be modified, during the January-February 1980 shutdown, to allow closing of the power circuit breakers upon loss of offsite power following a Safety Injection Actuation Signal. The control circuit also contains an undervoltage relay which will trip the circuit breakers when the motor control center is transferred to the onsite emergency source. Manual operation of a handswitch in the Control Room is necessary to close the circuit breaker. The main and control power interfaces are protected by safety-grade circuit breakers.

The diesel generator loading was reviewed for LOCA, MSLB and Blackout conditions. It has been concluded that under Blackout conditions (loss of offsite power) the diesel generator ratings will not be exceeded. Under loss of coolant or main steam line breaks, the continuous rating of one diesel could be exceeded, however, neither the seven day nor the two hour ratings were approached or exceeded. Therefore, the diesel generators are capable of carrying the additional load without compromising their ability to handle the emergency load.

The ANO-2 design does not incorporate a PORV or block valve.

The ANO-2 pressurizer level indicators are safety-grade, qualified, redundant and powered from safety buses; therefore, they presently meet the necessary requirements.

Item 2.1.2 - Relief and Safety Valve Testing

ANO-1 & 2

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

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

Item 2.1.3.a Direct Indication of PORV and Safety Valve Position

ANO-1 & 2

An acoustical valve monitoring system is being installed on ANO-1 during January 1980. An acoustical valve monitoring system is being installed on ANO-2 during its January-February 1980 outage. These systems will provide the operator with an indication of valve position (i.e., open/closed) on an annunciator panel in the Control Room. This alarm will sound when either valve is opened.

Redundant sensors which will be located on each valve will transmit redundant signals to a signal conditioner located in the Control Room. Only one of the inputs will be connected to the signal conditioner at a time. However, the operator can manually switch to the other input.

The system is single failure proof from the sensors to the signal conditioners. In addition, if one signal conditioner fails, the operator can still get a position indication by connecting to signal conditioners for one of the other valves.

This acoustical valve position indication equipment will be powered from a safety grade power source. All equipment will also be mounted as seismic class I installations. In addition, the equipment is environmentally suited for its application. Sufficient QA documentation, however, does not exist to qualify the equipment seismically or environmentally. Generic qualification should begin in February, 1980, and will require approximately six (6) months to complete.

Temperature elements downstream of the PORV and safety valves on ANO-1 and downstream of the safety valves on ANO-2 provide backup indication of valve position. These are monitored in the control room and alarm on high temperature.

Item 2.1.3.b - Instrumentation for Dection of Inadequate Core Cooling

ANO-1

Our letter of December 13, 1979, to Mr. R. Reid provided the guidelines incorporated into our plant procedures used for detection of inadequate core cooling with currently available instrumentation. A description of the existing instrumentation used is not available at this time, but will be supplied by January 31, 1980.

The design for additional instrumentation for detection of inadequate core cooling is not available at this time, but will be supplied by January 31, 1980.

For indication of primary coolant saturation conditions, we will install two (2) channels of margin to saturation measurement and indication during the January 1980 outage. A functional block diagram of a single channel is shown on the attached figure and is described in the following paragraphs.

For indication of temperature margin to saturation, the calculator selects the highest temperature input. The calculator utilizes the pressure input as a pointer to locate the corresponding saturation temperature in steam tables resident in the calculator memory. The process temperature is subtracted from the saturation temperature and the difference is then available for recording and display. The pressure margin to saturation calculation is done in a similar manner.

Each channel calculator receives a single, safety-grade, wide-range (0-2500 psig) primary coolant pressure input from existing buffered outputs present in the Engineered Safeguards System. One channel is fed its pressure signal from reactor coolant loop A; the other channel drives its pressure input from loop B. Also, each channel calculator receives two (2) wide-range (120°-920°F) RTD temperature inputs, one from each reactor coolant hotleg loop. As referenced in our November 20, 1979, letter to D. G. Eisenhut, the interim installation will use existing non-safety grade temperature signals available in the Non-Nuclear Instrumentation System. This is necessary as qualified RTD bridges will not be available until May 1980. The temperature inputs will be upgraded to safety grade requirements during the first outage of sufficient duration upon receipt of the equipment, but no later than the next refueling outage.

The temperature margin to saturation conditions from each calculator is to be continuously recorded on a two (2) channel strip chart recorder located on the main control board. The temperature margin from each calculator may also be individually read from digital indicators mounted in cabinets located in the control room. By manual selection, pressure margin to saturation may also be read from these digital indicators. Annunciation of low margin to saturation (temperature only) will be provided on the main control room panels. Isolation will be provided between the calculator and recorder as well as the calculator an annunciator.

In addition to the redundant margin to saturation monitoring described above, backup capability already exists on the ANO-1 plant computer. This computer program functions in a similar manner to the margin calculators. In addition, primary coolant temperature and pressure are directly available to the operators

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by means of existing console indicators. Steam tables are provided in the control room for use by the operators to determine saturation margin manually. These measures will be used until the upgraded temperature inputs are installed.

The margin to saturation monitoring channels will be installed prior to startup from the present outage.

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INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	Tsat - T or P-Psat
Display Type (analog, Digital, CRT)	Digital and recorded
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Control Room
Alarms (include setpoints)	To be supplied later
Overall uncertainty (°F, PSI) **	± 8°F; 45 psi
Range of Display	Display: 0-199.9°F; 0-1999 psig Recorder: 0-100°F
Qualifications (seismic, environmental, IEEE323)	Seismic, environmental per IEEE 323 and 344

Calculator

Type (process computer, dedicated digital or analog calc.)	Dedicated digital calcula.
If process computer is used, specify availability. (% of time)	NA
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	See description
Qualifications (seismic, environmental, IEEE323)	Seismic, environmental per IEEE 323 and 344
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam tables

Input

Temperature (RTD's or T/C's)	Rosemount 177GY RTD
Temperature (number of sensors and locations)	Each channel has a hot-leg A and B input for a total of 4. on hotleg piping.
Range of temperature sensors	120° - 920° F

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Uncertainty* of temperature sensors (°F at 1)	**	$\pm$ 5.0° F
Qualifications (seismic, environmental, IEEE323)		Seismic, environmental
Pressure (specify instrument used)		Foxboro E11GH-INM2 P.T.
Pressure (number of sensors and locations)		Channel A-one, hotleg A Channel B-one, hotleg B located in containment
Range of Pressure sensors		0-2500 psig
Uncertainty* of pressure sensors (PSI at 1)	**	$\pm$ 30.0 psi
Qualifications (seismic, environmental, IEEE323)		Seismic, environmental

Backup Capability

Availability of Temp & Press	See description
Availability of Steam Tables, etc.	See description
Training of operators	
Procedures	

* Uncertainties must address conditions of forced flow and natural circulation

** - Includes cumulative uncertainties of all intermediate transmitters, bridges, and signal converters between sensors and calculators.

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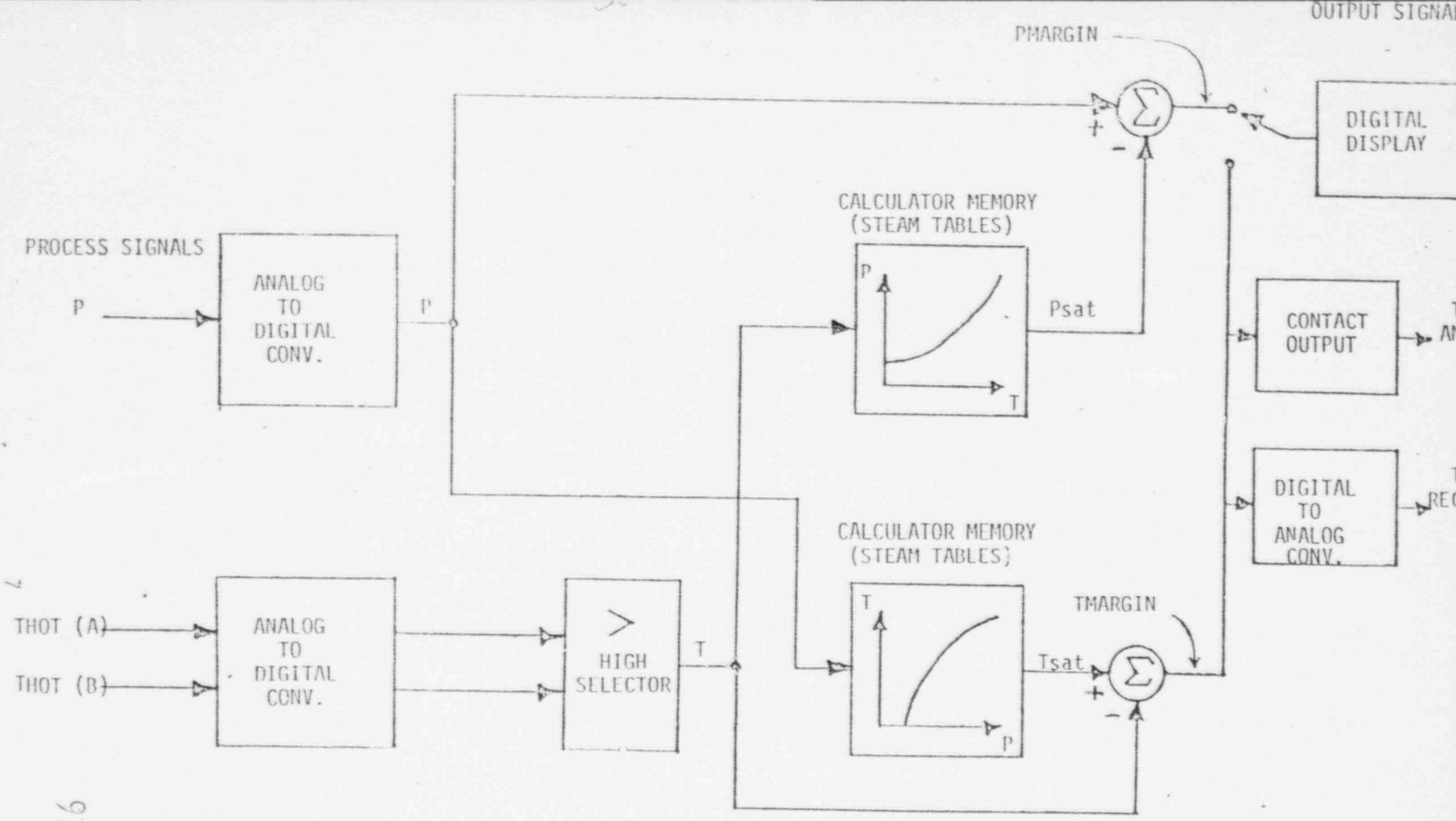

TYPICAL MARGIN TO SATURATION CHANNEL

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Guidelines for detection of inadequate core cooling using currently available instrumentation have been incorporated into operating procedures.

The evaluation for additional instrumentation for detection of inadequate core cooling is complete. The instrument response of a reference C-E plant for events which have the potential for inadequate core cooling is documented in CEN-117, "Inadequate Core Cooling -- A Response to NRC IE Bulletin 79-06C, Item 5 For Combustion Engineering Nuclear Steam Supply Systems". The conclusion reached in CEN-117 is that there currently is sufficient instrumentation in the plant to be used to detect inadequate core cooling. ANO-2 has updated its emergency procedures and associated operator training based on recommended guidelines from Combustion Engineering.

We have received several variations of a conceptual design for Reactor Vessel Water Level Indication from C-E as part of the C-E Owner's Group effort. This design(s) is currently being evaluated as to feasibility and value in assessing the extent of core uncovering during an accident. We will advise you by February 1, 1980, of the results of our evaluation. However, as stated in the above discussion, it is not expected that C-E's contention of no need for this instrumentation will be invalidated.

For indication of primary coolant saturation conditions, we will install two channels of saturation margin measurement and indication. A functional block diagram of a single channel is shown on the attached figure and is described in the following paragraphs.

For indication of temperature margin to saturation, the calculator selects the highest temperature input. The calculator utilizes the pressure input as a pointer to locate the corresponding saturation temperature in steam tables resident in the calculator memory. The process temperature is subtracted from the saturation temperature and the difference is then available for recording and display. The pressure margin to saturation calculation is done in a similar manner.

The pressure input for each subcooled margin calculator is derived from redundant, safety-grade, wide-range (0-3000 psig) pressurizer pressure transmitters. These transmitters also provide the pressure signals to the Plant Protection System. The two temperature inputs for each calculator are from redundant, safety-grade, wide-range (150-750^{o}F) T_{HOT} RTD's in each loop. Redundancy requirements are satisfied by separate and redundant subcooled margin monitoring channels.

The temperature margin to saturation from each calculator is to be continuously recorded on a strip chart recorder located on the main control board. The temperature margin from each calculator may also be displayed on a digital indicator in the back of the control room. Optionally, the pressure margin to saturation may be displayed on the indicator. Separate annunciators are provided on the main control room panels. Isolation is provided between the calculator and recorder as well as the calculator and annunciator.

Although not required upon completion of installation, a backup to the redundant channels of subcooled margin monitoring is provided by means

of an existing program on the ANO-2 plant computer. This program functions in a similar manner to the margin calculators previously described. In addition, primary coolant temperature and pressure are directly available to the operators by means of existing indicators. Steam tables are provided in the control room for use by the operators to determine saturation margins.

The subcooled margin monitors will be installed during the outage beginning on or before January 31, 1980.

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	Tsat - T or P-Psat
Display Type (Analog, Digital, CRT)	Digital and recorded
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Control Room
Alarms (include setpoints)	To be supplied later
Overall uncertainty (^{o}F, PSI)	$\pm$ 5^{o}F, $\pm$ 46 psi
Range of Display	0-199.9^{o}F; 0-1999 psi
Qualifications (seismic, environmental, IEEE323)	Seismic, environmental per IEEE 323 and 344

Calculator

Type (process computer, dedicated digital or analog calc.)	Dedicated digital calcula.
If process computer is used, specify availability. (% of time)	NA
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	See description
Qualifications (seismic, environmental, IEEE323)	Seismic, environmental per IEEE 323 and 344
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam tables

Input

Temperature (RTD's or T/C's)	Rosemount RTD
Temperature (number of sensors and locations)	2 Hotleg/loop (1 from each loop/calc.)
Range of temperature sensors	150-750^{o}F

Uncertainty* of temperature sensors (^{o}F at 1)	0.75^{o}F
Qualifications (seismic, environmental, IEEE323)	Seismic, environmental
Pressure (Specify instrument used)	Rosemount 1153A
Pressure (number of sensors and locations)	1 pressz/calc.
Range of Pressure sensors	0-3000 psig
Uncertainty* of pressure sensors (PSI at 1)	30 psig
Qualifications (seismic, environmental, IEEE323)	Post LOCA environmental, seismic

Backup Capability

Availability of Temp & Press	See Description
Availability of Steam Tables etc.	See Description
Training of Operators	
Procedures	

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

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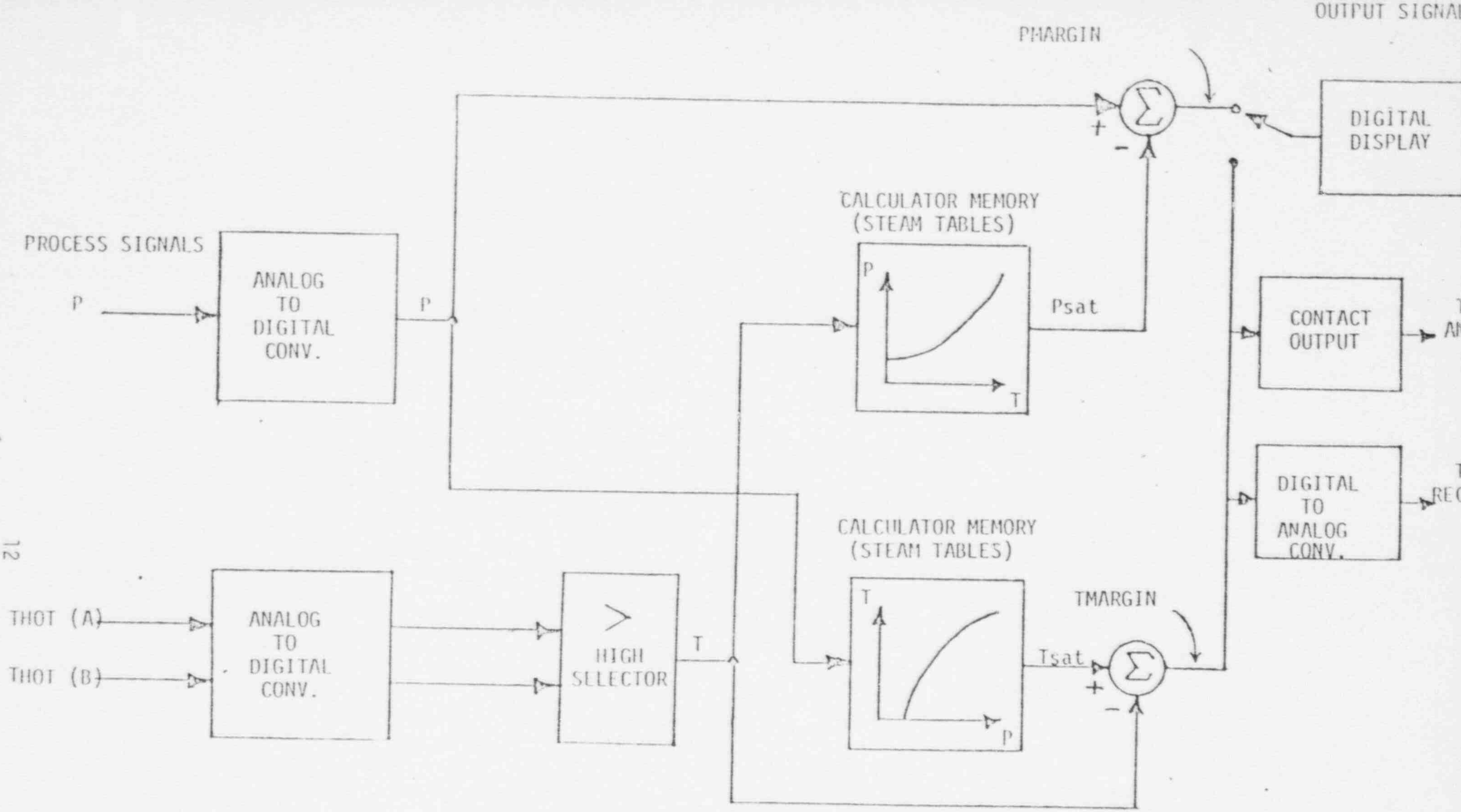

TYPICAL MARGIN TO SATURATION CHANNEL

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Item 2.1.4 Diverse Containment Isolation

ANO-1

We have reviewed the Reactor Building Isolation System design and procedures and find the following automatically actuated valves which provide penetration isolation (valve numbers in parentheses):

1) Reactor Coolant Pump Controlled Bleed-off (CV-1270, CV-1271, CV-1272, CV-1273, and CV-1274).

2) Reactor Coolant Letdown (CV-1214, CV-1216, and CV-1221).

3) Intermediate Cooling Water to Control Rod Drives (CV-2235).

4) Chilled Water to Reactor Building Coolers (CV-6202).

5) Intermediate Cooling Water to Reactor Coolant Pump Motors and Lube Oil Coolers (CV-2234).

6) Intermediate Cooling Water to Letdown Cooling and Seal Water Coolers (CV-2233).

7) Chilled Water from Reactor Building Coolers (CV-6203 and CV-6205).

8) Intermediate Cooling Water from Reactor Coolant Pump Motors, Lube Oil Coolers and Control Rod Drives (CV-2220 and CV-2221).

9) Fire Water (CV-5611 and CV-5612).

10) Nitrogen Supply to Quench Tank (CV-1667).

11) Reactor Building Sump Drain to Reactor Auxiliary Sump (CV-4446 and CV-4400).

12) Quench Tank Drain to Auxiliary Building Equipment Drain Tank (CV-1053 and CV-1052).

13) Reactor Building Vent Header to Auxiliary Building Gas Collection Header (CV-4803 and CV-4804).

14) Reactor Building Air Particulate Monitor to Auxiliary Building Gas Collection Header (CV-7453 and CV-7454).

15) Reactor Building Purge (CV-7402, CV-7404, CV-7401, and CV-7403).

16) Quench Tank Gas Sample (CV-1054 and CV-1845).

Items 1 and 2 above are isolated upon receipt of a Low Reactor Coolant System Pressure Signal ($\leq$ 1500 psig) or High Reactor Building Pressure ($\geq$4 psig) Signal.

Item 3 through 16 isolate upon a High Reactor Building Pressure Signal. The valves noted in Items 3 through 8 are normally open during power operation, as these valves are in systems which provide support to systems within the Reactor Building. The valves noted in Items 9 through 16 are normally closed

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during operation and require specific manual operation for opening, i.e., there is no automatic actuation of any of these valves to permit a connection from the Reactor Building Atmosphere to the Auxiliary Building Atmosphere or environment. In addition, to provide full opening of the penetration, specific manual operation of at least two valves is required, each of which requires a separate and deliberate action.

In the case of Items 3 through 8, it is felt that in order to ensure that a "normal", orderly cooldown ensues following receipt of an ES Signal (Low Reactor Coolant System Pressure); to take maximum advantage of systems available without unnecessarily proceeding to degraded modes; and to ensure no unwarranted equipment damage, no changes to the Reactor Building Isolation System are needed.

In reference to Items 9 through 16, based on the fact that these valves are normally closed and that specific manual action is required to breech Reactor Building isolation, no changes to the Reactor Building Isolation System are needed.

However, to provide for an increased margin of safety, we have prepared a design change to provide the valves in Items 9 through 16 above with an ES Signal to isolate on Low Reactor Coolant System Pressure ($\leq$ 1500 psig).

This design change will be implemented during the January-February 1980 outage. Specific manual operation will be required to reset individual valves even though the ES channel has been reset.

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ANO-2

We have reviewed the Containment Isolation Actuation System (CIAS) design and procedures and have listed below the automatically actuated valves which provide penetration isolation (valve numbers in parenthesis):

Category I

1) Chemical and Volume Control System Letdown (2CV-4821-1 and 2CV-4823-2).

Category II

2) Chilled Water Supply to containment Coolers (2CV-3852-1).

3) Chilled Water Supply from Containment Coolers (2CV-3850-2 and 2CV-3851-1).

4) Component Cooling Water to Reactor Coolant Pump Coolers (2CV-5236-1).

5) Component Cooling Water from Reactor Coolant Pump Coolers (2CV-5254-2 and 2CV-5255-1).

Category III

6) Containment Vent Header (2CV-2400-2 and 2CV-2401-1).

7) Reactor Coolant System and Pressurizer Sample (2SV-5833-1 and 2SV-5843-2).

8) Nitrogen Supply to Safety Injection Tanks (2CV-6207-2).

9) Quench Tank Liquid Sample (2SV-5878-1 and 2SV-5871-2).

10) Safety Injection Tank Sample (2SV-5876-2).

11) Quench Tank Makeup Water Supply (2CV-4690-2).

12) Containment Sump Drain (2CV-2060-1 and 2CV-2061-2).

13) Containment Purge Inlet (2CV-8289-1, 2CV-8284-2 and 2CV-8283-1).

14) Containment Purge Outlet (2CV-8291-1, 2CV-8286-2 and 2CV-8285-1).

15) Low Pressure Nitrogen Supply (2CV-6213-2).

16) Reactor Drain Tank Drain (2CV-2202-1 and 2CV-2201-2).

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Category IV

17) Reactor Coolant Pump Controlled Bleedoff (2CV-4847-2 and 2CV-4846-1).

18) Steam Generator Sample (2CV-5852-2 and 2CV-5859-2).

19) Air Particulate Monitor in Hydrogen Purge System (2SV-8231-2, 2SV-8273-1 and 2SV-8271-2).

20) Air Particulate Monitor in Containment Atmosphere Sample (2SV-8261-2, 2SV-8265-1 and 2SV-8263-2).

21) Fire Water Supply (2CV-3200-2).

Item 1 (Category 1) above is isolated upon receipt of a Safety Injection Actuation Signal (SIAS) or a Containment Isolation Actuation Signal (CIAS). SIAS is generated when Reactor Coolant System pressure is less than or equal to 1740 psia or when Containment Building pressure is greater than or equal to 18.4 psia. CIAS is generated when Containment Building pressure is greater than or equal to 18.4 psia.

Items 2 through 21 isolate upon receipt of a CIAS. The valves noted in Items 2 through 5 (Category II) are normally open during power operation since they are in systems which provide support to needed systems within the Containment Building. The valves noted in Items 6 through 16 (Category III) are normally closed during power operation and are only opened periodically by specific manual operation, i.e., there is no automatic opening of any of these valves. The valves noted in Items 17 through 21 (Category IV) are normally open during power operation, but are not necessary to be open following receipt of a SIAS.

Since Items in Category II are providing support to systems within the Containment Building, the valves should stay open upon receipt of a SIAS to prevent unnecessary equipment damage. The systems represented in Category II contribute to a "normal", orderly cooldown following receipt of a SIAS.

Items in Category III are normally closed during power operation and specific manual operation is required to open them. Furthermore, to cause full opening of the penetration, specific manual operation of at least two valves is required, each of which requires a specific and deliberate action. Based on this fact, no changes to the Containment Building Isolation System are needed.

Items in Catagory IV are normally open during power operation and specific manual operation is required to close these valves following receipt of a SIAS. Each of the Category IV systems were reviewed and it has been verified that a direct connection between the Containment Building atmosphere and the Auxiliary Building atmosphere or the environment does not exist while these penetrations are open. Based on this fact, no changes to the Containment Building Isolation System are needed.

However, to further increase the margin of safety, a design change will be implemented during the January-February 1980 outage for items in Category III and IV to add a SIAS to those valves. This design change will provide an

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additional degree of assurance that no release path to the environs exists upon receipt of a SIAS without a concurrent CIAS.

The valves selected to receive SIAS will continue to receive CIAS. The ESFAS response of the valves will not be affected by the additional actuation signal. Resetting of the actuation signal will not automatically reset the valves.

Item 2.1.5.a Dedicated H2 Control Penetrations

ANO-1

The original ANO-1 license required redundant, safety-grade and dedicated hydrogen purge systems which have been in place since the unit was licensed. Therefore, no changes are necessary to meet the requirements of NUREG-0578.

ANO-2

The ANO-2 license is based on redundant, safety-grade, qualified in-containment hydrogen recombiners and a safety grade, dedicated hydrogen purge system. Therefore, no changes are necessary to meet the requirements of NUREG-0578.

Item 2.1.5.c - Recombiner Procedures

ANO-1

The hydrogen purge system at ANO-1 for hydrogen control inside containment was used as the design basis for licensing. The procedures for use of this system have been thoroughly evaluated in light of the NUREG-0578 requirements. Therefore, our present procedures satisfy the requirements of this item.

ANO-2

The hydrogen recombiners located inside containment for hydrogen control were used as the design basis for licensing. The procedures for use of this system have been thoroughly evaluated in light of the NUREG-0578 requirements. Therefore, our present procedures satisfy the requirements of this item.

Item 2.1.6.a - Systems Integrity for High Radioactivity

ANO-1 & 2

Arkansas Power and Light has undertaken a program to reduce leakage of all the applicable systems given on page A-26 of NUREG-0578.

The affected systems are being tested during the January-February 1980 outages for leakage using either a volumetric makeup or direct leakage collection/measurement method. An initial leak test is being performed to determine the "As Found" condition. An inspection is being made of the mechanical joints and mechanical interfaces (Flanges, Unions, Valve Bonnets, Packing, etc.). Identified leakage of 0.01 cc/min will be repaired/reduced as practical and a second leak test will be performed. The initial leak testing and repairs capable of being made during operations have been completed for ANO-2. All ANO-1 systems will be inspected, tested, and repaired prior to achieving criticality following the current outage.

AP&L will perform a review of the leak test results and any identified source of leakage. This review will be aimed at identifying needed improvements in our present Preventative Maintenance program. In addition, the leak tests will be re-performed at least once every 18 months.

The systems to be tested are:

- Dirty Liquid Radwaste
- Clean Liquid Radwaste
- Gas Radwaste
- Makeup & Purification
- Decay Heat System
- RB Spray System

North Anna Incident Evaluation

ANO-1 & 2

As per procedures for Reactor-Turbine Trip (and HPI), the Volume Control Tanks of both Units 1 and 2 remain aligned to a charging pump, and equilibrium make-up is provided to maintain VCT level.

It is possible to isolate the VCT in Unit 2 from the charging pumps and take suction from the Refueling Water Tank (though it is not in the Reactor-Turbine Trip Procedure to do so). Should the liquid safeties lift at the VCT, the relief path is to one of four Boron Management Holdup Tanks, with a capacity of 51,270 gallons each. These tanks relieve to one another, and gases are expelled through a pressure control valve to the Gas Collection Header. Therefore, no liquid or gaseous releases to the reactor auxiliary building atmosphere would be expected.

All tanks and pumps associated with contaminated materials are hard piped to either the Gas Collection Header if they are sources of aerated gases or the Waste Gas System if they are sources of hydrogen gas.

The Waste Gas Systems in both units are currently being investigated for possible design improvements to further ensure that inadvertent releases can be avoided. The schedule for completion of the investigation is anticipated by March 1, 1980. The results of this investigation and the extent of modification proposed will dictate our completion of hardware changes.

2.1.6.b - Design Review of Plant Shielding of Spaces for Post-Accident Operations

ANO-1 & 2

Our plant shielding review has been completed. Applicable portions of this report entitled "Design Review of Plant Shielding and Sampling Capabilities in Response to NUREG-0578" are attached. For accident conditions which assume a Regulatory Guide 1.4 release of fission products, radiation levels throughout the reactor auxiliary building are in excess of General Design Criteria A by a considerable amount. However, it should be noted that the results of this review are misleading due to the non-existence of a mechanistic sequence of events leading to 100% fuel chadding failure and, therefore, the resulting radiation source term which form the basis of this report.

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We intend to administratively control access to certain areas of the reactor auxiliary building based on assessment of the situation through the plant hardware installed or being installed (i.e., margin to saturation meters, incore thermocouples, high range containment radiation monitors, etc.) as a result of the TMI-2 event and NUREG-0578. The severity of the situation will be detailed by this instrumentation and will dictate those systems which will remain operational following the incident. For the most severe incident, we envision that only those systems designed for post-accident conditions (i.e., ECCS systems such as High Pressure Safety Injection, Low Pressure Safety Injection and Reactor Building Spray) will be operational, thereby, eliminating a large part of the reactor auxiliary building which will be affected by any high radiation dose levels which may exist.

Based on the above, and because of changes initiated in plant design and operator training as a result of experiences gained from TMI-2, we do not reasonably expect the original shielding design of the plant to be inadequate. Therefore, no plant shielding modifications are proposed at this time; however, if our review indicates shielding modifications may be feasible, it shall be considered.

Item 2.1.7.a - Auto Initiation of the Auxiliary Feedwater System

ANO-1 & 2

This item is not required to be addressed by the January 2 confirmatory orders. However, based on your January 18, 1980 letter of our proposed design, we will install the Emergency Feedwater Control System as described in our letters of October 31, 1979, and December 18, 1979, prior to startup from the January 1980 outage.

Item 2.1.7.b - Emergency Feedwater Flow Indication

ANO-1

At the present time, the ANO-1 emergency feedwater system has an orifice plate with one differential pressure transmitter and one control room panel mounted flow indicator for each emergency feedwater line to the two steam generators. This system will be upgraded, during the January 1980 outage, by adding two (2) differential pressure transmitters powered from separate emergency power supplies. Four indicators will be located in the control room supplying two indicators for each steam generator. Sketch 2.1.7-1 shows a simplified layout of the basic equipment. These changes will meet your single failure criteria, testability, accuracy, and safety-grade power supply requirement. These changes also meet the long-term safety-grade requirements except for the four (4) panel mounted indicators which will be installed prior to January 1, 1981.

ANO-2

The ANO-2 emergency feedwater system is designed to meet BTP 10-1 Rev. 1, and as such currently meets both short and long-term requirements of NUREG-0578. Therefore, no modifications are necessary.

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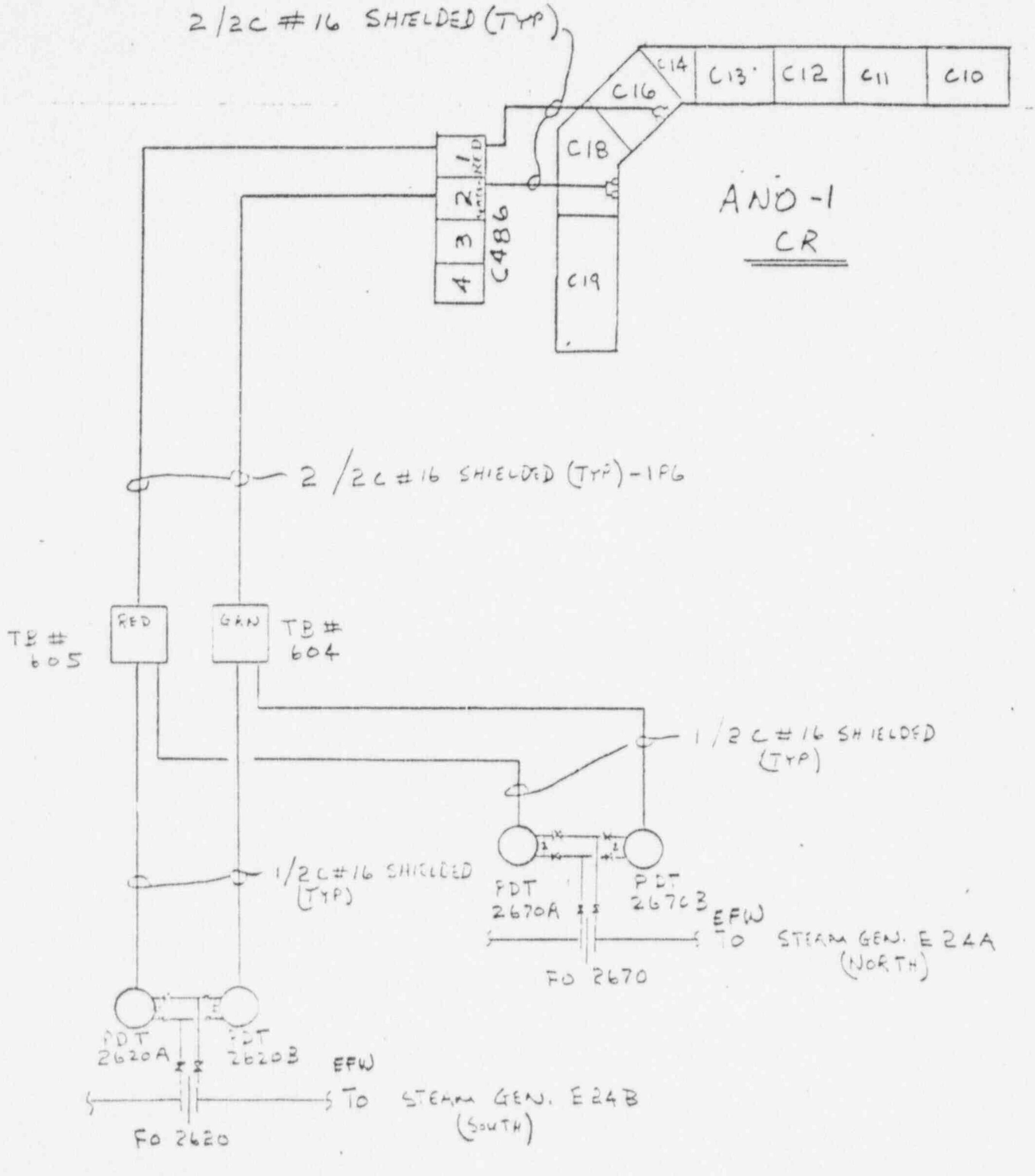
FIGURE 2.1.7.-1



ANO-1 EMERGENCY FEEDWATER FLOW INDICATION

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2.1.8.a - Improved Post-Accident Sampling

ANO-1 & 2

A design and operational review of the reactor coolant and containment atmosphere sampling systems has been performed. Applicable portions of a report entitled "Design Review of Plant Shielding and Sampling Capabilities in Response to NUREG-0578" are attached. As can be seen from this report, when Regulatory Guide 1.4 assumptions are utilized for release of fission products and gases to the coolant, personnel could not obtain samples for radiological and chemical analyses without exceeding General Design Criteria 19 guidelines. However, the results of this review are misleading in that there is a non-existence of a mechanistic sequence of events leading to 100% fuel cladding failure and the resulting radiation source terms which form the basis of this report and prohibit work in the reactor auxiliary building.

Our existing sampling system is undergoing review (to be completed by April 1, 1980) and will be modified to the extent possible to reduce the postulated radiation dose to personnel and still maintain the capability to obtain meaningful samples which will totally reflect the continuities of the sampled environment. Varying levels of sample dilution and remote operation are envisioned as possible modes for achieving this end. This design, however, may not totally reflect dose reduction to GDC 19 levels, but will reflect a reasonable attempt to comply with the NUREG-0578 requirements.

Based on changes made to plant design and operator training which have resulted from the TMI-2 event, we reasonably expect that cladding failure will not exceed the 1% failed fuel criteria to which the plant was initially designed. However, changes to the post-accident sampling system will be initiated and completed by January 1, 1981, to alleviate potential for postulated employee over exposure during the course of these activities.

2.1.8.b - Increased Range of Radiation Monitors

ANO-1 & 2

The following responses cover the sections listed under the NUREG-0578 item for radiation monitoring by January 1, 1980. The numbering system below coincides with the requested response format.

1. Noble Gas Effluents

90030183

   a. System/Method Description

      i) The instrumentation to be used will consist of a detector with a remote readout in one of the Control Rooms. The detector(s) will consists of GM and/or Ionization chambers. The detectors energy response for gaseous radiation will be at least ± 20 from 60 KeV to >2 MeV. One or two detectors will be used as necessary to provide for measuring Xe-133 gas concentration from 10^0 to 10^3 µCi/cc as a minimum. The readout device will be capable of accepting the signal generated by the detector(s). The readout will read in mR or R per unit time. The range of the readout will, as a minimum, correspond to 10^2 to 10^3 µCi/cc for Xe-133. A visual conversion aid to convert dose rate to µCi/cc , such as

a graph, chart, etc., will be located at the readout device.

Calibration of the device prior to its operation will be accomplished using a known Cs-137 standard and standard techniques. The system will be checked against a remotely installed source monthly, and calibrated every three months.

ii) During an emergency, the Reactor Building Purge Systems and Spent Fuel Ventilation Systems will not be operated. Also. other systems that may create a release path in the Unit 2 Auxiliary Building Extension (e.g., ANO Penetration Room Ventilation System) will be isolated. Therefore, the building release paths will be restricted to the Auxiliary Building Exhausts on Units 1 and 2. Because a more accessible location is needed for particulate and iodine collection media, a new isokinetic sampling system will be installed. Isokinetic sample nozzles are being installed in the Unit 1 Auxiliary Building Exhaust duct and in the Unit 2 Auxiliary Building Exhaust duct. These lines are routed down into the Spent Fuel area where the iodine and particulate sample media will be located. The sample lines will be tied together and valved such that a dual set of filters and one pump can be used for either Unit. A fixed sample volume will be defined by use of shielding. The detector will also be shielded from interference from background radiation since this will be a low background area. These design modifications will be completed during the January 1980 outage.

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iii) The readout will be located in either Unit 1 or Unit 2 Control Rooms.

iv) The system will provide continuous readout in the Control Room.

v) The system will be powered by vital AC power in the Control Room.

b. Procedures for conducting all aspects of the measurement/ analysis.

i) Exposure is minimized since the readout is in the Control Room.

ii) Calculations such as those found in the "Reactor Shielding Design Manual" by Rockwell were used to determine the radiation intensities at the detector. Previous Reactor Building air samples during operation plus the gas activities given in the Unit 1 FSAR show that 96% to 98% of the initial noble gas activity will be Xe-133. Calculations are also being made for periods following an accident when the isotopic mix will change. In addition, provisions will be incorporated to allow grab sampling for noble gas so that calculations can be verified or modified if necessary after the isotopic mix starts to change.

iii) Readout will be monitored at least every fifteen minutes by an individual located in the Control Room.

iv) The system will be calibrated to dose rate from Cs-137 using the calibration sources available at ANO. Initial calibration may be performed by the vendor.

<u>2.1.8.c - Improved Iodine Instrumentation</u>

<u>ANO-1 & 2</u>

As noted in the November 11, 1979 response to Lessons Learned Task Force Recommendations, ANO has nine portable air samplers and procedures for obtaining and performing spectral analysis of the samples, and therefore, presently satisfies the requirement.

<u>Item e - Reactor Coolant System Venting</u>

The following are descriptions of the venting systems for ANO-Units 1 & 2 which are proposed for implementation as Category B items.

<u>ANO-1</u>

The hot leg venting system will consist of piping, solenoid operated valves and instrumentation designed to permit the main control room operator to remotely vent the high points of the hotleg piping and the top of the

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pressurizer to containment during post-accident conditions. The system will be designed to vent superheated steam, steam water mixtures, water, fission gases, helium, nitrogen and nydrogen as high as 2500 psia and 670°F.

The system will be safety grade with the same qualifications as were accepted for the RCS at the time of licensing. Redundant vent paths, each path to consist of two valves in series powered from the same class 1E power supply, will be provided at each hot leg vent. The redundant path at each location will be powered from a different class 1E power supply. A single safety grade vent path will be added to the pressurizer; the PORV and block valve will provide a redundant vent path. Each vent will be seismically qualified.

Each vent path will be capable of venting a gas volume of at least 1/2 the RCS volume in one hour. The RCS mass loss from an open vent path will be less than the definition of a LOCA in 10 CFR 50, Appendix A. To minimize inadvertant vent opening, power will be removed from the valves during normal operation. Opened/closed indication will be provided in the control room for all power operated valves.

Analyses demonstrating that venting will not result in violation of combustible gas concentration limits and procedural guidelines for the operators' use of the vents will be provided to you by June 1, 1980. These analyses are intended to support your review and approval of our design.

ANO-2

The RCS high point venting system will consist of piping, solenoid operated valves and instrumentation designed to permit the main control room operator to remotely vent the reactor vessel head and the top of the pressurizer to containment during post-accident conditions. The system will be designed to vent superheated steam, steam water mixtures, water, fission gases, helium, nitrogen and hydrogen as high as 2500 psia and 700°F.

The system will be safety grade with the same qualifications as were accepted for the RCS at time of licensing. Redundant vent paths, each path to consist of two valves in series powered from the same class 1E power supply, will be provided at the reactor vessel head and the top of the pressurizer. The redundant path at each location will be powered from a different class 1E power supply. Each vent will be seismically qualified.

Each vent path will be capable of venting a gas volume of at least one-half the RCS volume in one hour. The RCS mass loss from an open vent path will be less than the definition of a LOCA in 10 CFR 50, Appendix A. To minimize inadvertant opening, power will be removed from the valves during normal operation. Opened/closed indication will be provided in the control room for all power operated valves.

Analyses demonstrating that venting will not result in violation of combustible gas concentration limits and procedural guidelines for the operators' use of the vents will be provided to you by June 1, 1980. These analyses are intended to support your review and approval of our design.

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Item 2.2.1.a - Shift Supervisor's Responsibilities

ANO-1 & 2

Once per year, the Vice-President, Generation and Construction, will issue a management directive to the personnel primarily responsible for plant operations and safety, which will emphasize that the primary management responsibility of the shift supervisor is for the safe operation of the plant. This directive will also clearly establish the shift supervisor's command duties under all plant conditions. The first of such directives has been issued for 1980.

Plant procedures are being reviewed and modified, as appropriate, to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the Control Room relative to other plant management personnel. Particular emphasis is placed on the following:

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

b. The shift supervisor, until properly relieved, will remain in the Control Room at all times during accident situations to direct the activities of Control Room operators. Persons authorized to relieve the shift supervisor are specified.

c. If the shift supervisor is temporarily absent from the Control Room during routine operations, the Plant Operator who is the lead control room operator will be designated to assume the Control Room command function. These temporary duties, responsibilities, and authority are clearly specified.

Training programs for shift supervisors will emphasize and reinforce the responsibility for safe operation and the management function the shift supervisor is to provide for assuring safety.

The administrative duties of the shift supervisor have been reviewed by the Director of Generation Operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant will be delegated to other operations personnel not on duty in the Control Room.

Procedures to implement the above have been completed and training provided.

Item 2.2.1.b - Shift Technical Advisor

ANO-1 & 2

The Shift Technical Advisor (STA) function has been added (as required by NUREG-0578) as a method of improving the plant staff's capabilities to respond to abnormal operating conditions and for improving the evaluation of operating experience.

The Shift Technical Advisor personnel have been selected from our site engineering staff personnel. As of January 1, 1980, thirteen graduate engineers have been selected from the normal plant staff. The present complement of STA's was selected from the plant Engineering and Technical Support Group. They will serve a 24-hour duty day on a rotating basis and will be onsite at all times during their duty. Sleeping quarters have been added on the plant site for their use and they will be available to the control room within 10 minutes of being called by the Shift Supervisor.

There are two major functions to be provided by the STA: Accident Assessment and Operating Experience Assessment. An essential ingredient for the satisfactory performance of these functions is that the efforts of the STAs must be dedicated to concern for the safety of the plant.

In the accident assessment fuction, the STA's duties will be primarily for diagnosis of off-normal events and to advise the Shift Supervisor. The Shift Supervisor has primary responsibility for the safe operation of his unit and must judge for himself as to the validity of advice given by the STA.

Primary responsibility for the Operating Experience Assessment function at Arkansas Nuclear One will lie with the Plant Performance group. This group consists of three graduate engineers with Nuclear/Mechanical/Electrical backgrounds and having power plant experience. In addition, reviews of operations and recommendations to improve safety are the responsibility of all Shift Technical Advisors.

In the operating experience assessment function, the STAs will evaluate plant operations, plant design, and operating events from a safety point of view. Examples of assignments which may be used to accomplish this objective include:

(1) Review of operating and maintenance records to detect unsafe practices, potential equipment failures or reliability problems.

(2) Review of plant transients to detect needs for procedure or equipment changes.

(3) Review of reportable occurrences from Arkansas Nuclear One and from other plants with similar designs to detect developing problems.

(4) Review of operating, maintenance, quality control, and surveillance testing procedures and practices to detect possible safety problems/improvements.

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(5) Discussions with operating and maintenance personnel and evaluation of their comments regarding plant problems potentially affecting safety.

(6) Prepare written safety evaluations and recommendations resulting from the reviews outlined in (1) through (5).

In order to be able to provide meaningful and accurate evaluations, the STAs should have knowledge and training in at least the following general areas: mathematics, reactor physics, chemistry, materials, thermodynamics, fluid mechanics, heat transfer, electrical engineering, instrumentation and controls, as well as experience and training in plant design, control room layout and reactor operations.

Due to the incompleteness of the diverse background desirable to provide the STA functions by most or all of our present STAs, the first year (1980) must be dedicated largely to training. Emphasis in most cases is needed in specifics of plant design features, plant layout, and reactor operations. Therefore, much of the time and duties of the STAs will be spent in activities closely resembling that for Reactor Operator trainees.

The program, as presently formatted, will encompass three categories of training. Category A consists of a lecture series and on-the-job training (OJT) in plant structures, systems, component design and layout. Category B consists of a lecture series and OJT on the functions and capabilities of Instrument and Control Systems. It will include training provided by CE and B&W and simulator training. Category C consists of a lecture series and OJT on Plant Response and analyses of transients and accidents. It also will include training by B&W and CE and simulator training.

Item 2.2.1c - Shift and Relief Turnover Procedures

ANO-1 & 2

AP&L has reviewed and revised, as appropriate, plant procedures for shift and relief turnover. These revised procedures are consistent with the clarification of this recommendation provided at our Regional Meeting.

1. Procedure(s) have been provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to assure a complete and effective turnover. The following items have been included in the procedure(s):

    a. Assurance that critical plant parameters are within allowable limits;

    b. Assurance of the availability and proper alighment of all systems essential to the prevention and mitigation of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode will be compared with the Technical Specifications action statement;

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

2. Procedure(s) have been provided to assure a complete and effective turnover by the offgoing to the oncoming auxiliary operators and technicians. These procedure(s) address any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient; and

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

The reviews, peocedural modifications, and training have been completed.

Item 2.2.2.a Control Room Access

ANO-1 & 2

AP&L has reviewed the plant procedures for Control Room access. We have implemented procedures which will limit Control Room access during an emergency. These procedures include the following:

1. Administrative procedures that establish the authority and responsibility of the person in charge of the Control Room to limit access.

2. Procedures that establish a clear line of authority and responsibility in the Control Room in the event of an emergency. The line of succession for the person in charge of the Control Room has been established and limited to persons possessing a current senior reactor operator's license. The plan clearly defines the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the Control Room.

   These procedures have been implemented and training provided.

Item 2.2.2.b - On-Site Technical Support Center (TSC)

ANO-1 & 2

ANO has completed the following items per NUREG-0578:

A. A TSC has been established on the 4th floor of the Plant Administrative Building which is located south of the Turbine Building. The TSC presently serves as the Planning and Scheduling facility and contains approximately 1000 sq. ft. work space, including a 150 sq. ft. office for the Duty Emergency Coordinator.

B. Procedures have been revised to reflect the engineering/management support and staffing for the TSC. The TSC staff is shown on Figure 2.2.6-1.

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C. Direct communications between the TSC and the Control Room, Near Site Emergency Operations Center and the NRC has been established.

D. Portable monitoring equipment has been provided for near the Technical Support Center direct radiation and airborne radioactive contaminants with warning capability if radiation levels in the TSC are approaching potentially dangerous levels. The TSC will be evacuated based on recommendations of the Health Physics personnel prior to reaching dangerous radiation levels.

E. Technical Data (such as layout drawings, P & ID's, electrical schematics, Isometric drawing and Tech Manuals) is available from the plant's Records Management System. This data is located in the Administrtion Building within one floor of the TSC. Plant parameters will be monitored by CRT display in the TSC from the plant computer.

F. Accident accessment from the Control Room, should the TSC become uninhabitable, will be conducted using present procedures. In the event the TSC becomes uninhabitable, the Duty Emergency Coordinator, Operations Superintendent, Operations & Maintenance Manager, Engineering and Technical Support Manager, Technical Analysis Superintendent, and Plant Analysis Superintendent will evacuate to the Control Room. The remaining members of the On-Site Technical Support Center Engineering/Management staff will evacuate to the Near Site Emergency Operations Center, which is located approximately 0.65 miles north of the plant, with continued direction being provided from the TSC personnel located in the Control Room. If the TSC becomes uninhabitable, technical data such as plant layout drawings, P & ID's and electrical schematics will be relocated to the Near Site Emergency Operations Center.

Item 2.2.2.c - On-Site Operations Support Center

ANO-1 & 2

The On-Site Operations Support Center (OSC) is located in the vicinity of the On-Site Technical Support Center. Operations Support Personnel will be located in the OSC for response to the control room and/or TSC needs. Telephone communications presently exist with the control room and the TSC.

3.0 PLANT SHIELDING FOR POST-ACCIDENT OPERATIONS

This section provides an assessment of potential post-accident shielding problems associated with those systems outside of the containment structure which might contain large radioactive inventories. The purpose of recommendation 2.1.6.b is to facilitate post-accident operations using systems that may contain high levels of radioactivity and to ensure that safety equipment in proximity to the resulting radiation fields is not unduly degraded. Corrective action can consist of design change, additional fixed or portable shielding, post-accident procedure optimization, or equipment upgrading. For these systems Section 3.1 contains an estimate of potential radiation source terms, Section 3.2 identifies the access requirements, and Section 3.3 indicates the post-accident shielding requirements necessary to accommodate these access requirements. Section 3.4 provides a discussion of potential equipment problems which may result from the high radiation levels.

3.1 SOURCE TERMS

In response to the NRR Lessons Learned Task Force Section 2.1.6.b, estimates for source terms and doses have been made for the following systems:

- Makeup and purification/Chemical and volume control
- Decay heat removal/Residual heat removal
- Reactor building spray
- Safety injection
- Liquid and gaseous radwaste
- Sampling

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The initial primary coolant and containment sump fission product inventories are listed on Table 3-1. These figures reflect the assumptions set forth in subsections 3.1.1 and 3.1.2.

3.1.1 PRIMARY COOLANT ACTIVITY

In accordance with Regulatory Guide 1.4, it is assumed that 100% of all noble gases, 50% of all iodine, and 1% of all remaining radioactive particulates are dissolved into the primary coolant. The components that may be affected by these source terms include:

- Letdown heat exchanger, piping and valves,
- Reactor makeup pumps, piping and valves,
- Decay heat pumps, heat exchangers, piping and valves if operating in a shutdown decay heat removal mode (24 hours after accident).

3.1.2 CONTAINMENT SUMP ACTIVITY

It is assumed that primary coolant entering the containment will release 99.9% of the noble gases to the containment atmosphere; the primary coolant that collects in the containment sump will then contain 0.1% of the noble gases, and all of the radioactive iodines and particulates.

The components that may be affected by these source terms include:

- Decay heat pumps, heat exchangers, piping and valves (operating in the containment recirculating mode).
- Safety injection pumps, piping and valves (operating in the containment sump recirculating mode).
- Containment spray pumps, piping and valves.

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Table 3-1. Fission Product Inventory Primary Coolant and Containment Sump, Units 1 and 2

Isotope	Fission Product Inventory, Curies/Gram: Primary Coolant		Containment Sump	
	Unit 1	Unit 2	Unit 1	Unit 2
^{131}I	1.3×10^{-1}	1.7×10^{-1}	1.3×10^{-1}	1.7×10^{-1}
^{132}I	2.0×10^{-1}	2.6×10^{-1}	2.0×10^{-1}	2.6×10^{-1}
^{133}I	3.0×10^{-1}	3.9×10^{-1}	3.0×10^{-1}	3.9×10^{-1}
^{134}I	3.5×10^{-1}	4.5×10^{-1}	3.5×10^{-1}	4.5×10^{-1}
^{135}I	2.7×10^{-1}	3.5×10^{-1}	2.7×10^{-1}	3.5×10^{-1}
^{83m}Kr	2.7×10^{-2}	3.5×10^{-2}	2.7×10^{-5}	3.5×10^{-5}
^{85m}Kr	8.9×10^{-2}	1.1×10^{-1}	8.9×10^{-5}	1.1×10^{-4}
^{85}Kr	2.2×10^{-3}	2.8×10^{-3}	2.2×10^{-6}	2.8×10^{-6}
^{87}Kr	1.6×10^{-1}	2.1×10^{-1}	1.6×10^{-4}	2.1×10^{-4}
^{88}Kr	2.5×10^{-1}	3.2×10^{-1}	2.5×10^{-4}	3.2×10^{-4}
^{131m}Xe	2.3×10^{-3}	3.0×10^{-3}	2.3×10^{-6}	3.0×10^{-6}
^{133m}Xe	1.3×10^{-2}	1.7×10^{-2}	1.3×10^{-5}	1.7×10^{-5}
^{133}Xe	5.4×10^{-1}	6.9×10^{-1}	5.4×10^{-4}	6.9×10^{-4}
^{135m}Xe	1.4×10^{-1}	1.8×10^{-1}	1.4×10^{-4}	1.8×10^{-4}
^{135}Xe	1.1×10^{-1}	1.4×10^{-1}	1.1×10^{-4}	1.4×10^{-4}
^{138}Xe	5.4×10^{-1}	6.9×10^{-1}	5.4×10^{-4}	6.9×10^{-4}
Solids	1.5×10^{-1}	1.9×10^{-1}	1.5×10^{-1}	1.9×10^{-1}

Based on a Unit 1 reactor coolant mass of 5.35×10^5 lb (2.43×10^8 grams) a Unit 2 reactor coolant mass of 4.619×10^5 lb (2.1×10^8 grams), and a Unit 2/Unit 1 power ratio of 1.11. Of the total reactor core fission product inventory, the primary coolant is assumed to contain 50% of the iodines, 100% of the noble gases, and 1% of the particulates. The containment sump liquid is assumed to contain 50% of the iodines, 0.1% of the noble gases, and 1% of the particulates.

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3.1.3 DOSE RATE CALCULATIONAL BASIS

The dose rate as a function of time, shown on Figure 3-1, was calculated using the CYLDOSE code. CYLDOSE calculates the linear attenuation, scatter buildup, and resulting tissue dose rate from a cylindrical gamma radiation source. Multiple source materials and shield materials may be specified. Dose points may be selected anywhere along the side of the source or at its end; a line source approximation is used for dose points along the side, whereas a truncated cone source approximation is used for dose points at the end. For convenience of calculation the gamma energy emitted by the source(s) is divided into groups and each group is designated by a number and a group average energy. Calculations may be done considering one or a combination of these groups. Source strengths associated with these energy groups may be read into the code as data or calculated by the code.

3.2 AREAS AND COMPONENTS REQUIRING ACCESS FOR POST-ACCIDENT OPERATIONS

In developing the system and component access requirements, particular attention was directed at the systems for which the shielding evaluation was performed. Additional components within the Auxiliary Building that are likely to require some access are tabulated in Subsection 3.2.7.

3.2.1 UNIT 1 REACTOR COOLANT SYSTEM SAMPLING EQUIPMENT

3.2.1.1 Reactor Coolant Sampling

The valves used to draw a sample of the reactor coolant are all located near the sample sink; a motor operated valve, CV-1814, is controlled from control room panel C-18. All other valves are manually operated.

FIGURE 3-1(a)
RELATIVE DOSE RATE AS A FUNCTION OF TIME
0 – 30 DAYS
UNIT 1/UNIT 2

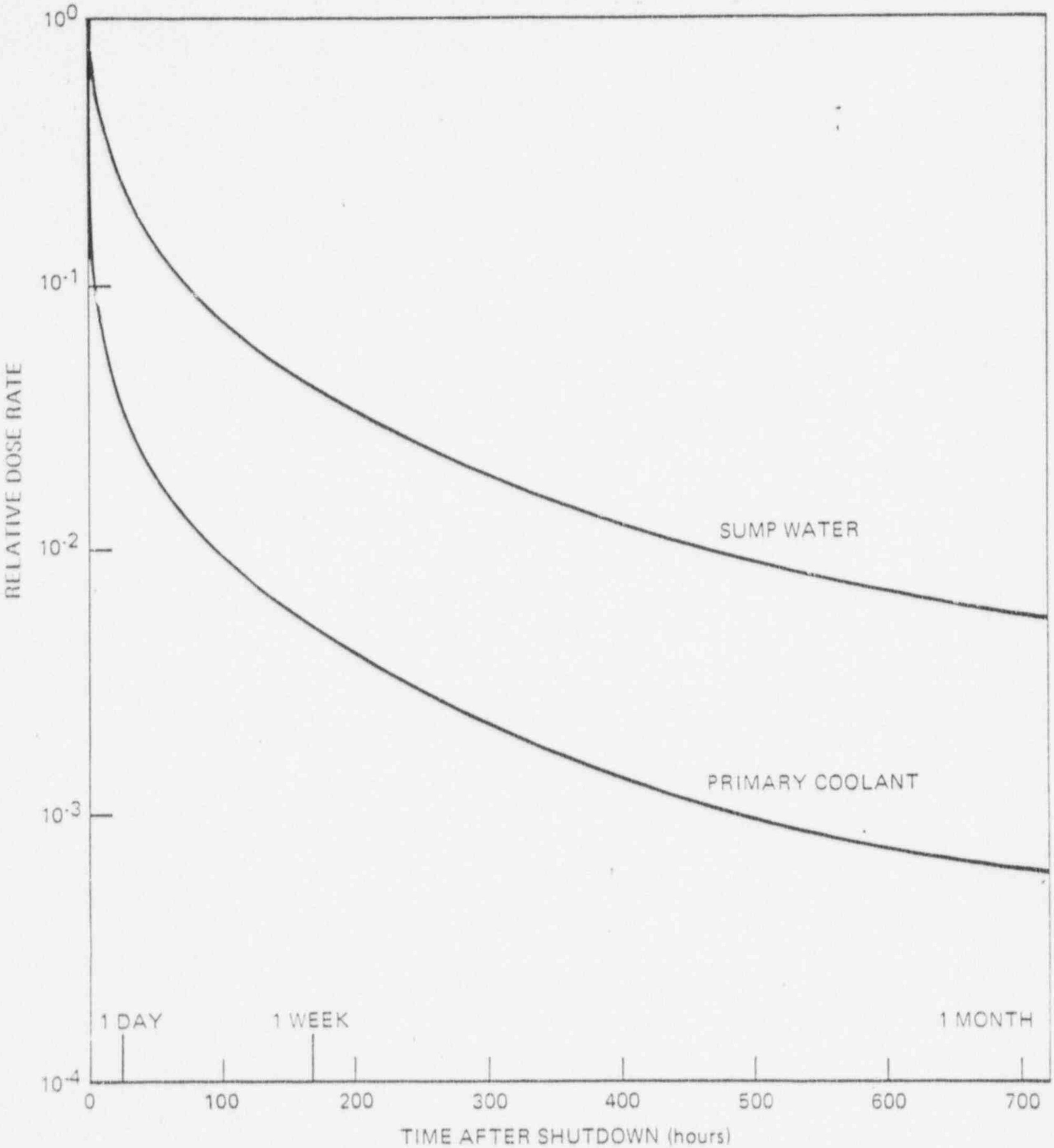

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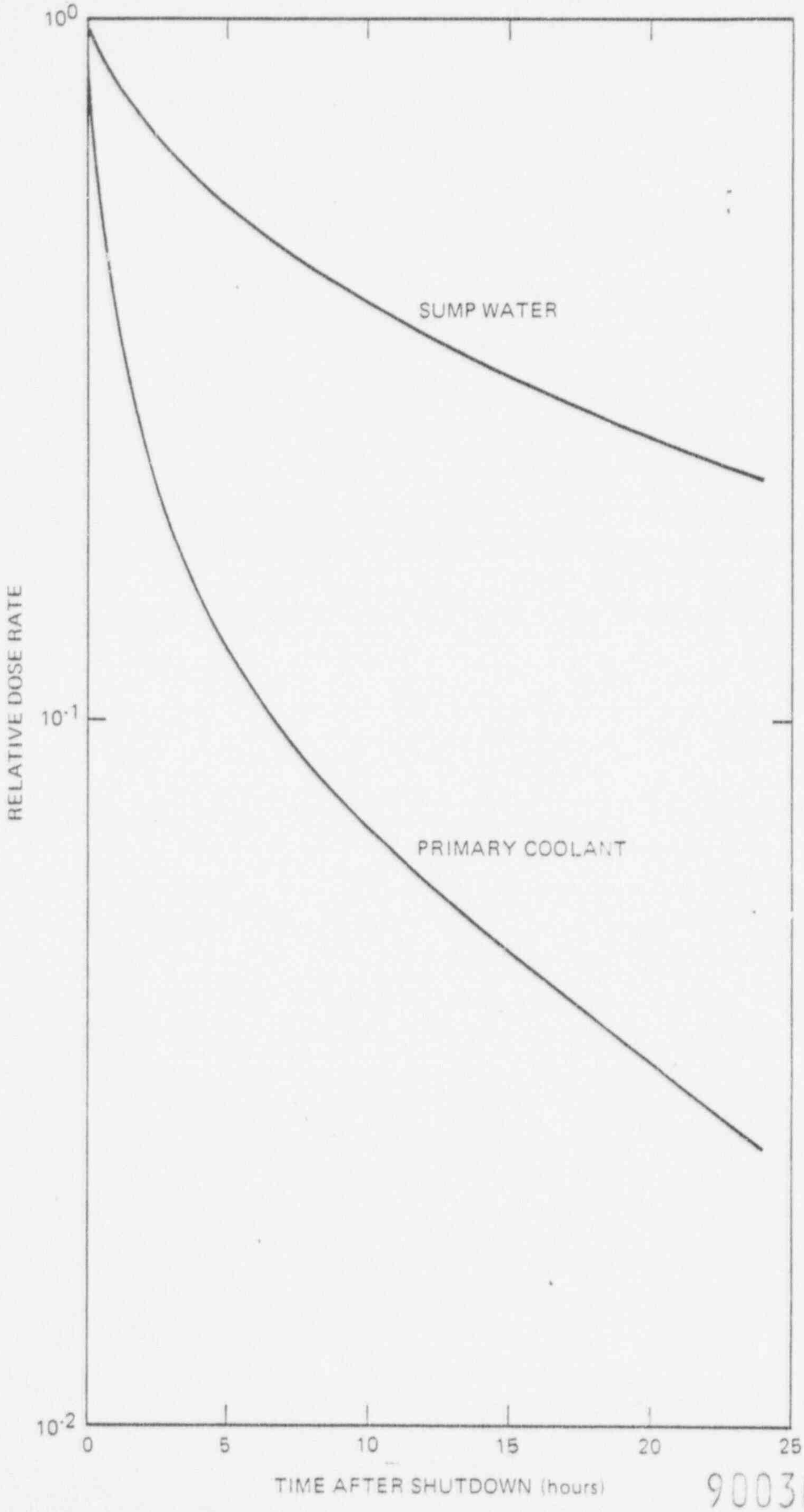
FIGURE 3-1(b)
RELATIVE DOSE RATE AS A FUNCTION OF TIME
0 – 24 HOURS
UNIT 1/UNIT 2



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3.2.1.2 RCS Decay Heat Removal System Sampling

Valves

SS-41A
SS-41B

3.2.2 DECAY HEAT REMOVAL OPERATIONS

3.2.2.1 Decay Heat During Cooldown

Valves	Other Equipment
BW-8A	Pump P34A
DH-1A	Pump P34B
DH-15	
DH-19	
BW-8B	
DH-1B	

3.2.2.2 Decay Heat Coolant Purification

Valves

MU-7
MU-5
MU-6
ABD-15
ABV-3
DH-4B
DH-5B
DH-6 (Throttling)
DH-4A
DH-5A

3.2.3 LOSS OF COOLANT/RC PRESSURE OPERATIONS

3.2.3.1 Rupture Greater Than HPI Capacity

Valves	Other Equipment
MU-13	Purge Dampers (actuate from ventilation control panel)
DH-7A	Isolate DH rooms by closing watertight doors
DH-7B	
MU-14	
MU-15	
MU-16	
MU-17	
ABS-13	
ABS-14	

3.2.3.2 Rupture Within HPI Capacity

Valves	Other Equipment
MU-13	Same as 3.2.3.1
DH-7A	
DH-7B	
MU-14	
MU-15	
MU-16	
MU-17	
MU-23	
MU-24	
MU-25	
MU-26	
ABS-13	
ABS-14	
MU-21A, B&C	

3.2.4 UNIT 2 REACTOR COOLANT SAMPLING

3.2.4.1 Reactor Coolant HPI Leg for Liquid

Valves	Other Equipment
2PS55	Sample sink
2PS56	
2PCV5922	
2PS59	
2PS57	

3.2.4.2 Isolation of 2T120 for Total Gas Analysis

Valves	Other Equipment
Valves listed in 3.2.4.1	Sample sink
2PS93B	
2PS93A	
2PS91	

3.2.4.3 Sampling of the Shutdown Cooling System

Valves

2BS18A
2BS18B
2PS62
2PCV5923
2PS63

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3.2.5 SHUTDOWN COOLING SYSTEM OPERATIONS

3.2.5.1 Initiation of Shutdown Cooling

Valves

2SI-1A
2SI-1B
2SI-2A
2SI-2B
2SI-4A
2SI-4B
2SI-5A
2SI-5B
2SI-20 (Throttle)

3.2.5.2 SDC Purification

Valves

2CVC-147
2CVC-146
2SI-35
2SI-34

3.2.6 UNIT 2 LOSS OF REACTOR COOLANT OPERATIONS

In the event of a rupture greater than the capacity of the high pressure injection pump, post-LOCA containment radiation monitoring will be required in accordance with Appendix A of OP 2202.06. In addition, ECCS pump rooms must be isolated by closing the watertight doors and SDC pump room floor drains 2ABS-5 and 2ABS-6. Case II (rupture within HPI pump capacity) also requires post-LOCA containment radiation monitoring.

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Valve operations listed in OP 2102.10, Plant Shutdown and Cooldown will also have to be performed during the loss of reactor coolant event, in accordance with procedure OP 2202.06.

3.2.7 Based on experience at TMI, the following areas and/or components may also require post-accident access:

- Control Room
- Liquid Radwaste Control Panel/Equipment
- Gaseous Radwaste Control Panel/Equipment
- Solid Radwaste System Control Panel/Equipment
- Auxiliary Feedwater Pumps/Valves
- Auxiliary Shutdown Panel(s)
- Component Cooling Water Pumps
- Auxiliary Building Ventilation System (fans, filters, adsorbers)
- Control Room HVAC Systems
- Cable Spreading Room
- Electrical Equipment/Switchgear Rooms
- Diesel Generators

3.3 ACCIDENT SCENARIOS - UNITS 1 AND 2

The dose contributions from the subject systems were calculated for several different accident scenarios. Depending on the scenario and operating mode, components in these systems will contain varying amounts of activity. By examining each scenario, the associated dose rates and attendant operational problems, the need for modification of equipment, shielding and/or operating procedures can be identified.

3.3.1 FUEL FAILURE EVENT WITHOUT COINCIDENT LOCA

In this condition, it is assumed that the primary coolant will attain the activity level specified on Table 3-1. Since a LOCA condition does not exist, no containment isolation signal is

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generated (assuming the absence of any other isolation signal). Consequently, primary coolant activity would be introduced into the Makeup and Purification System (Unit 1) or Chemical and Volume Control System (Unit 2), including the following components:

- Reactor makeup pumps/charging pumps
- Purification demineralizers
- Reactor coolant filters
- Seal water cooling system
- Letdown coolers
- Associated valves and piping

(Note: It is recognized that Unit 2 has provisions for radiation monitoring equipment to sample and monitor the activity in the reactor coolant upstream of the reactor coolant filters. A high-activity signal from this monitoring equipment will alarm in the control room. This same equipment could, on a high activity alarm, provide a signal to isolate the letdown line. It is not clear whether this automatic isolation feature presently exists. Unit 1, at least, does not appear to have this feature.)

In this mode, the makeup and purification system/CVCS can continue to operate for purposes of makeup, letdown, and their associated functions. By examining Tables 3-2 and 3-3, it is noted that the makeup and letdown loops are responsible for major dose contributions within the auxiliary building. The attendant dose problems impact most of the analyzed access points (refer to dose point numbers on Tables 3-2 and 3-3; these numbers correspond to the locations marked on Figures 3-3 through 3-12). The initial dose rates will decrease as a function of time in accordance with the curves shown on Figures 3-1(a) and (b).

3.3.2 FUEL FAILURE COINCIDENT WITH LOSS OF COOLANT

In the event of a loss of coolant, engineered safety features will initiate safety injection and containment isolation. As a result of automatic containment isolation, the flow of primary coolant to the makeup and purification system will be terminated. The introduction of containment sump water into these systems is expected to cause the following problems:

Unit 1

- Operation in the containment sump recirculation mode will make decay heat valves DH-1A and DH-1B inaccessible for manual realignment if required. As noted on Table 3-2, dose rates range up to 2×10^4 R/hr, depending on location relative to the recirculation equipment.

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- With containment sump recirculation in operation, the liquid and gaseous radwaste system control panels will be inaccessible. High dose rates will be encountered both at the panels and along the access routes to the panels.

Unit 2

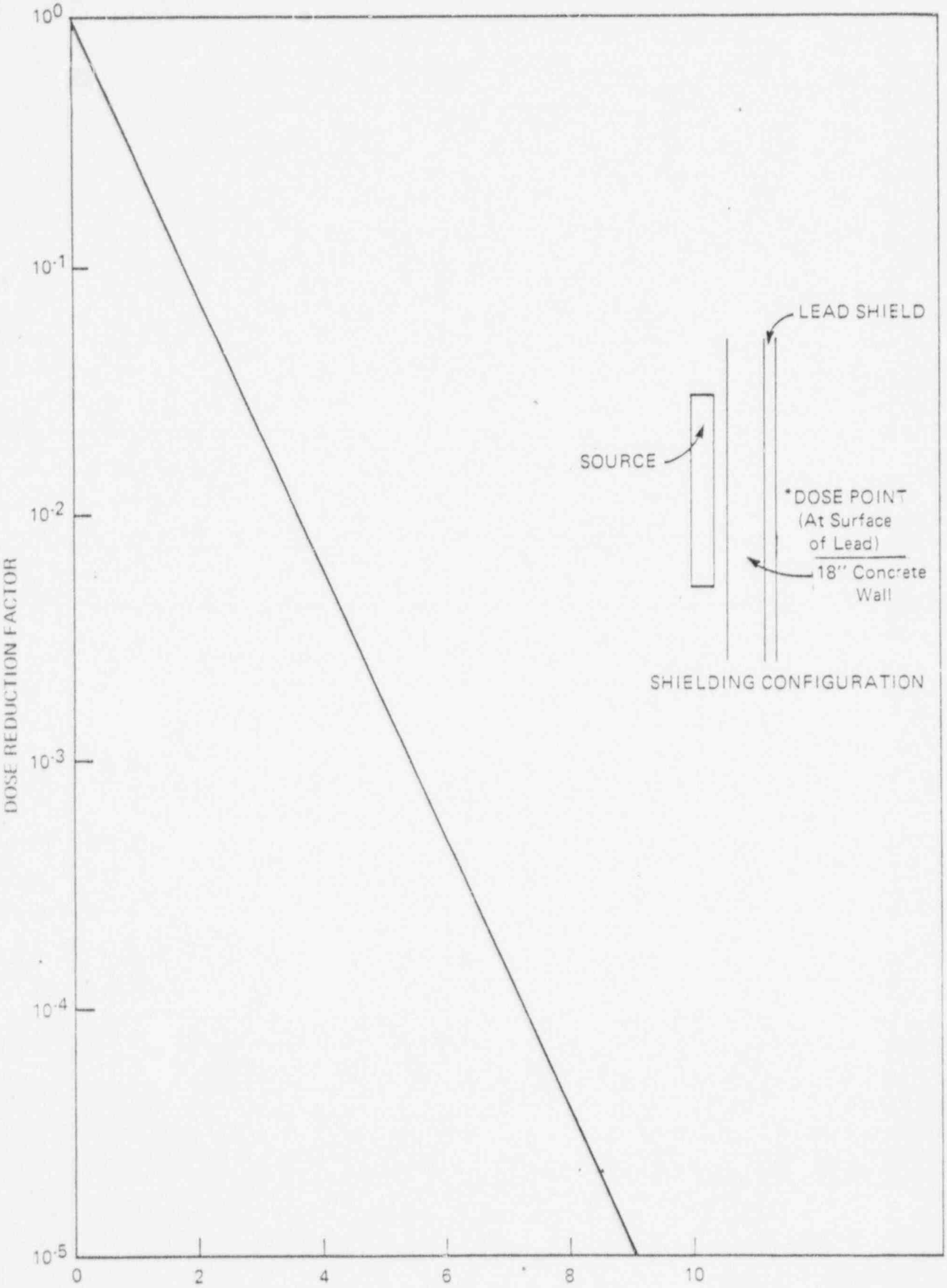
- Primary coolant and containment sump water activity levels will cause high doses in sump recirculation/decay heat valve operating areas; manually-operated valves that will be inaccessible include: 2SI4A, 2SI4B, 2SI1A, 2SI1B, 2SI5A, and 2SI5B. These valves would normally require alignment when initiating decay heat removal system operation.

3.3.3 FUEL FAILURE WITH COINCIDENT LOCA; LETDOWN IN OPERATION

This scenario is identical to that discussed in subsection 3.3.2, with the additional condition that primary coolant is being brought out of containment through the letdown loop. The containment isolation/loss of letdown discussed in 3.3.2 above would present some operational difficulties by impeding the use of some functions (e.g., degasificaton, boron control); thus, it may be desirable to utilize the letdown feature. The radiation hazards and problems that will be encountered are essentially a combination of the disadvantages of the first two scenarios discussed. Table 3-2 indicates that the worst case dose figures could apply in this operational mode. It should be further noted that these figures only reflect the "shine" doses from equipment, and do not consider possible airborne (immersion) doses.

If Figure 3-2 (effect of lead shielding) is examined in light of these calculated doses, it is readily seen that in order to reduce doses to acceptable levels (100mr/hr), substantial thicknesses of

FIGURE 3-2
EFFECT OF LEAD SHIELDING



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lead shielding will be required. In addition to the obvious problems of physical installation, the possible use of such quantities of lead shielding raises additional questions regarding the structural adequacy of the buildings (to support the additional loads).

3.4 POTENTIAL EQUIPMENT PROBLEMS

The integrated dose received by safety-related instrumentation should not be sufficient to result in any degradation in performance, because the equipment that is expected to perform under such extreme accident conditions is generally qualified for service in high radiation fields. Because of their sensitivity to ionization energy, electronic devices are not normally located in areas which may be subjected to high radiation fields; however, this must be verified.

The following items are prone to deterioration when subjected to significant ionization energy:

- Rubber, Teflon, and plastic components
- Glass (transmittance degradation)
- Electrical components (insulation and solid state devices)

Using Regulatory Guide 1.4 assumptions, the integrated 30 day dose increase factor to any component over the actual normal operation integrated dose received by that component, is estimated to be 1.4×10^4.

For components in extremely high radiation areas, it is assumed that under normal operating conditions that doses received by the components would be no greater tha 0.1 rad/hr. Therefore, the conservatively estimated 30 day dose recieved by the same equipment, based on Regulatory Guide 1.4 assumptions, is less that 1.0×10^6 rads.

Electronic equipment employing solid-state devices is subject to failure under these dose conditions; in addition, the high radiation field will cause device performance characteristics to change regardless of the total integrated exposure. It is recommended, therefore, that a survey be made to establish the locations of electronic equipment with respect to (potential) high radiation areas.

Table 3-2. Unit 1 Post-Accident Dose Rates at Selected Locations

Dose Point	Description	Initial Dose Immediately Following Accident
1	Adjacent to primary coolant filter and containment air sampling station El. 335-354	6500 R/hr; attributable to letdown line source term
2a	Adjacent to primary coolant filter and makeup pump P360C (pump running) El. 335-354	400 R/hr; attributable to letdown line source term
2b	Corridor adjacent to makeup pump P36C area (pump running) El. 335-354	100 R/hr
3	At elevator El. 335-354	0.2 R/hr from letdown line/makeup pump, plus 20 R/hr from recirculation system at El. 317. 1 R/hr in decay heat mode, assuming initial 24-hour decay.
4	Demineralizer valve operating area	23 R/hr
5	Makeup pumps valve operating area	640 R/hr, excluding gaseous and liquid waste lines
6	a. Corridor adjacent to makeup pumps	1 to 600 R/hr, depending on location relative to opening to valve operating area
	b. Inside makeup pump cubicle	500,000 R/hr
7	a. Corridor adjacent to reactor coolant makeup tank	7,200 R/hr, assuming 50% liquid, 50% gas; approximately 6 times normal gaseous activity due to noble gas buildup. The activity levels will be produced only if letdown is used (i.e., activity can be minimized if tank is isolated).

Table 3-2. Unit 1 Post-Accident Dose Rates at Selected Locations (Continued)

Dose Point	Description	Initial Dose (Immediately Following Accident)
8a	Outside door of "A" train decay heat/LPSI recirculation system, near unshielded recirculation line No. GBC-3-4"	1 x 10^4 R/hr, from containment sump recirculation operation
8b	Same location, except operating in decay heat mode only	40 R/hr.
8c	Same location, containment spray in location	Less than 1 R/hr.
8d	Same location, with unshielded re circulation line No. GCB-3-4 excluded	70 R/hr.
9a	Outside door of "B" train decay heat LPSI recirculation system; containment sump recirculation and containment spray systems operating	1500 R/hr. Of this figure, 1300 R/hr is due to the exposed "B" train recirculation line No. GCB-3-4" is due to the "A" train recirculation
9b	Same location, except operating in decay heat mode.	1 R/hr
10a	Vicinity of decay heat, LPSI, recirculation equipment	900 R/hr, with containment sump recirculation and containment spray operating; 0.3 R/hr with decay heat operating only
10b	Near stairway No.1, El. 317-335, 30 ft. away from unshielded recirculation line GCB-3-4"	1700 R/hr in containment sump recirculation mode; less than 0.3 R/hr in decay heat mode only.
10c	Inside containment recirculation equipment cubicle, at decay heat pump manual valves.	In excess of 20,000 R/hr., after 24-hr. decay. Recirculation and containment spray in operation.

Table 3-2. Unit 1 Post-Accident Dose Rates at Selected Locations (Continued)

Dose Point	Description	Initial Dose (Immediately Following Accident)
11	Control room, above makeup tank T-4	15.4 R/hr from makeup tank, 7.3 R/hr from vacuum degasifier, and an additional 1.4 R/hr from line HCC-5-3" (assuming that letdown is in operation).
12	Control room, in area of main control boards	5.0×10^{-4} R/hr from makeup tank, 1.8×10^{-4} R/hr from vacuum degasifier (assuming that letdown is in operation).
13	Diesel generator area	17-20 R/hr, from makeup tank. An additional 81 R/hr from vacuum degasifier, if this equipment is in operation.
14	Proposed remote sample station location (refer to Section 4)	7 mR/hr, from makeup tank
15	Vacuum degasification tank valve room	160 R/hr from degasification equipment, if operating.
16	Vicinity of waste gas surge tank T-17	16 R/hr from tank T-17; 14,000 R/hr from line HRC-2-2"; 8,500 R/hr from line HRC-2-2.5"; 520 R/hr from line HSC-1-3"
17	Vicinity of tank T-12D	290 R/hr from tank T-12D; 1,700 R/hr from line HSC-1-3"; 1,700 R/hr from valves in the adjacent valve operator corridor.
18	Vicinity of filter F-16	29 R/hr, from filter.

Table 3-2. Unit 1 Post-Accident Dose Rates at Selected Locations (Continued)

Dose Point	Description	Initial Dose (Immediately Following Accident)
19	Vicinity of gas decay tank T-18C	53 R/hr from Tank T-18C; 2,800 R/hr from line HRC-2-2.5"; 6,400 R/hr from line HSC-1-3".
20	Outside shield wall, vicinity of tank T-18C	20,000 R/hr from line HSC-1-3"; 7,300 R/hr from line HRC-2-2.5"; 1 R/hr from tank T-18C.
	Liquid Waste Control Panel	From 1200 R/hr. at stairway to 3000 R/hr. walking to panel; 100 R/hr. at panel during containment sump recirculation mode.
	Gaseous waste control panel	1 R/hr. at panel when operating in decay heat mode only. 1700 R/hr. at panel when operating containment spray and containment sump recirculation. 1 R/hr. at panel when operating in decay heat mode only.

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Table 3-3. Unit 2 Post-Accident Dose Rates at Selected Locations

Dose Point	Description	Initial Dose Immediately Following Accident
30	Control room, above volume control tank	300 R/hr from volume control tank (VCT).
31	Control room, near main control boards	1.5 x 10^{-4} R/hr, primarily from the VCT.
32	Decontamination room adjacent to control room	10 R/hr, from the vacuum degasifier tank, if degasification equipment is operating.
33	Electrical equipment room adjacent to cable spreading room	1820 R/hr, from VCT.
34	Access corridor, elevation 372', between diesel generators and cable spreading room.	750-800 R/hr, from VCT.
35	Stairway east of diesel generators (elevation 372').	1.8 mR/hr, from VCT.
36	Diesel generator room.	2.4 R/hr, from VCT.
37	Stairway entrance near hot machine shop (elevation 354').	0.3 R/hr from VCT; 50 R/hr from the vacuum degasification equipment, if operating.
38	Sample room, elevation 354'.	0.2 R/hr from VCT; 2 R/hr from the vacuum degasification equipment, if operating.
39	Corridor adjacent to VCT, elevation 354'.	1,820 R/hr, from VCT; 500 R/hr from degasification equipment, as noted above.
40	Stairway leading to south piping penetration area.	900 R/hr from VCT; 1 R/hr from degasification equipment, and 160 R/hr from the seal water heat exchanger, if operating.

Table 3-3. Unit 2 Post-Accident Dose Rates at Selected Locations (Continued)

Dose Point	Description	Initial Dose Immediately Following Accident
41	East of letdown heat exchanger, outside shield wall.	10 R/hr from heat exchanger; 1000 R/hr from the "B" train shutdown cooling equipment on Elevation 317 below.
42	Wall adjacent to charging pump 2P36B on elevation 335.	65 R/hr from charging pump "B", if running. The "B" train shutdown cooling equipment contributes the same dose as at point 41.
43	a. Corridor at entrance to charging pumps 2P36B and 2P36C; both pumps running.	110 R/hr (55 R/hr per pump), plus an additional localized dose of 1000-2,000 R/hr from the unshielded "window" to lines 2HCD-2-4" and 2HCD-63-3".
	b. Passageway to pump 2P36A and tank room 2054 on elevation 335.	1,000 R/hr from line 2HCD-63-3", plus 54 R/hr from pump 2P36A.
	c. At stairway entrance, elevation 335.	4-5 R/hr.
44	Elevation 317, in valve corridor containing two shutdown cooling manual valves.	160,000 R/hr, from containment spray system; 80,000 R/hr after 24-hour decay. (Shutdown cooling will not be initiated until 24 hours has elapsed.)
45	Inside area containing train "B" shutdown cooling, containment spray, HPSI/LPSI equipment.	10^6 R/hr.
46	a. Outside steel door to area described for point 45.	7,000 R/hr.
	b. Outside shield wall enclosing above area.	300 R/hr.

Table 3-3. Unit 2 Post-Accident Dose Rates at Selected Locations (Continued)

Dose Point	Description	Initial Dose Immediately Following Accident
47	Same as 46(a) and (b), except train "A" equipment considered.	See doses under 46(a) and (b).
48	Door outside HPSI pump 2P89C area.	150 R/hr.
49	Entrance to stairway at elevation 317.	20 R/hr.
50	Unshielded letdown line 2HCD-63-3" at elevation 335.	9,000 R/hr.

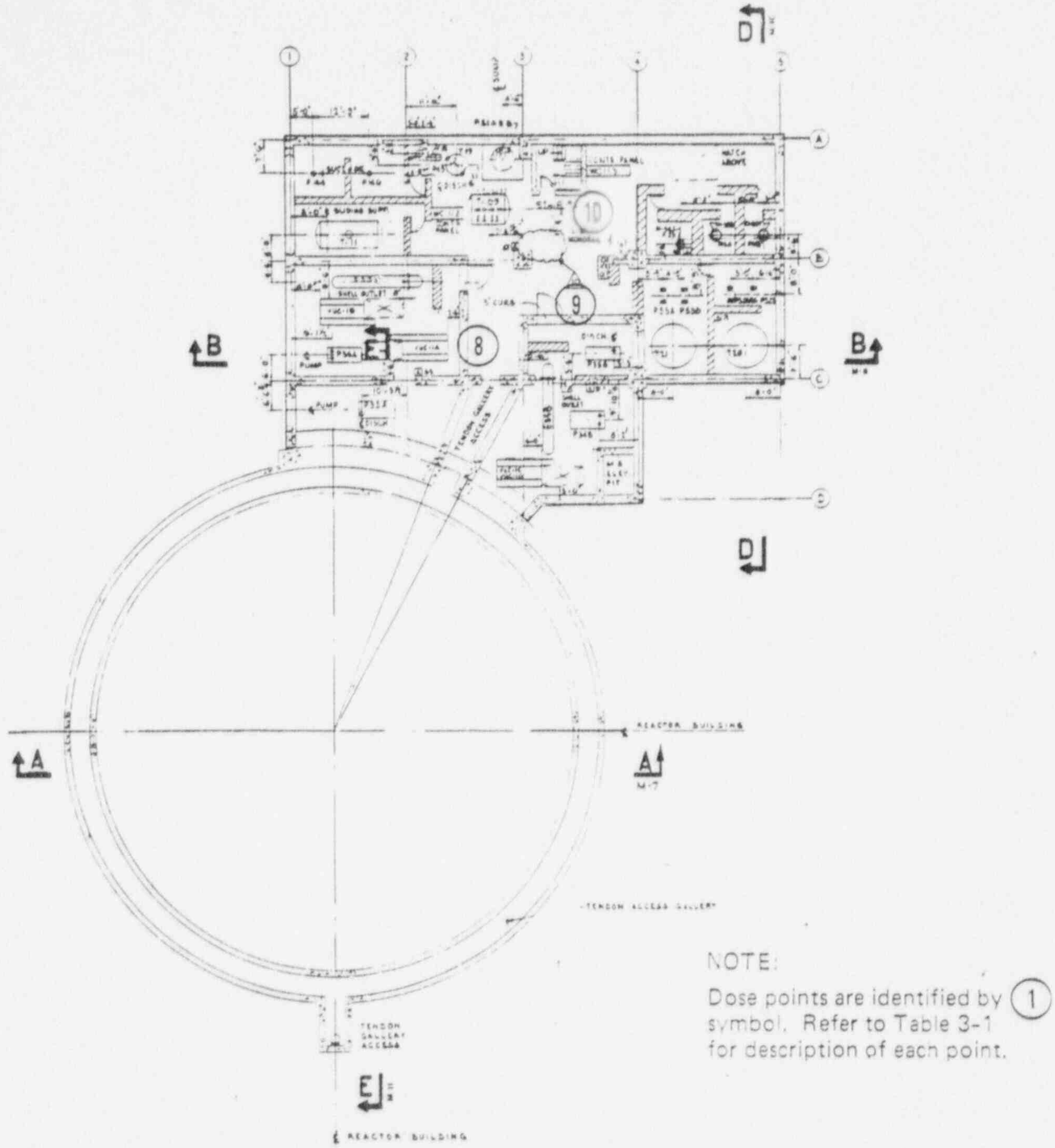

FIGURE 3-3
UNIT 1 PLAN BELOW GRADE
(EL. 317) SHOWING ANALYZED DOSE POINTS
REF.: AP & L DRAWING M-11, REV. 10

POOR ORIGINAL

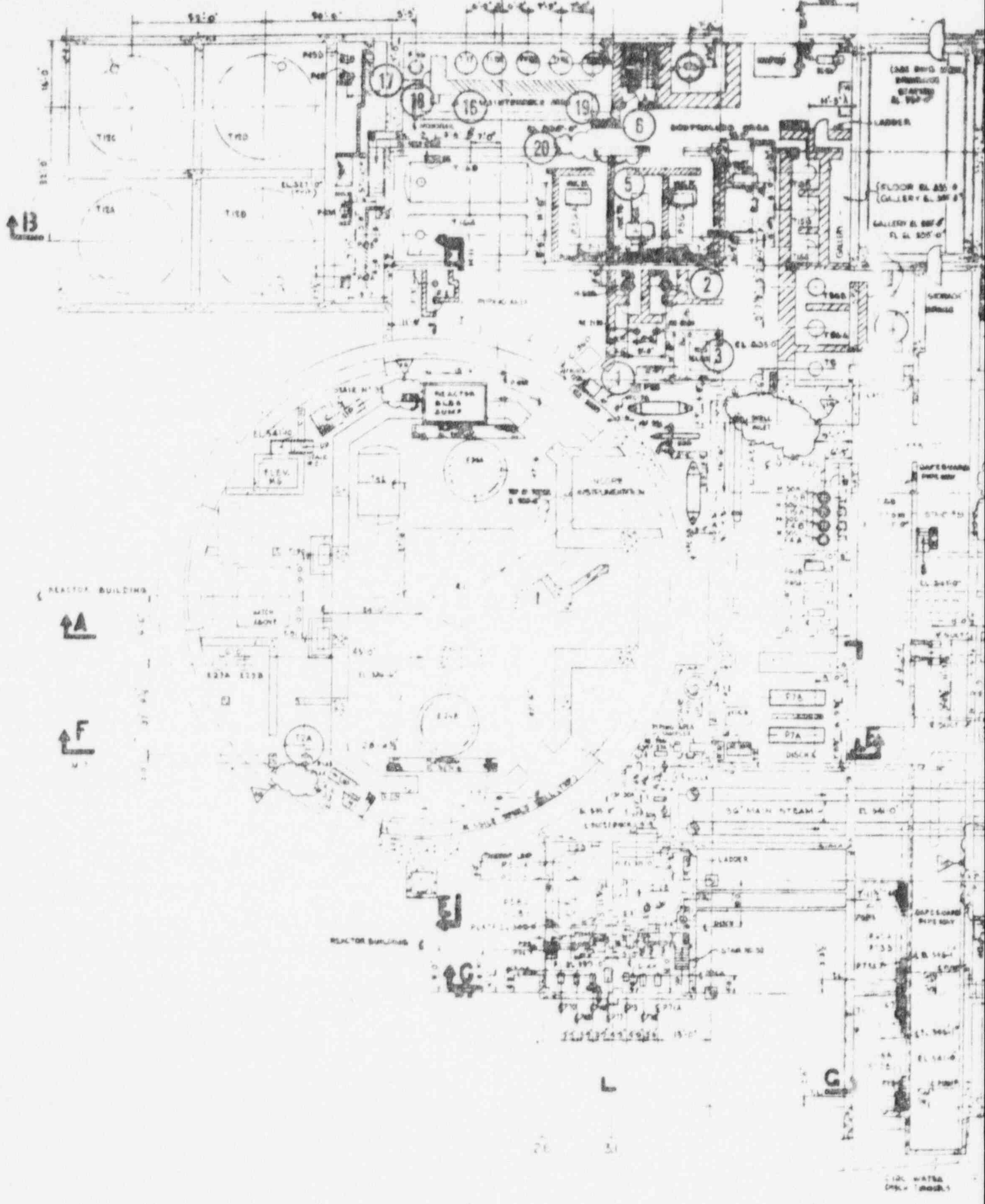

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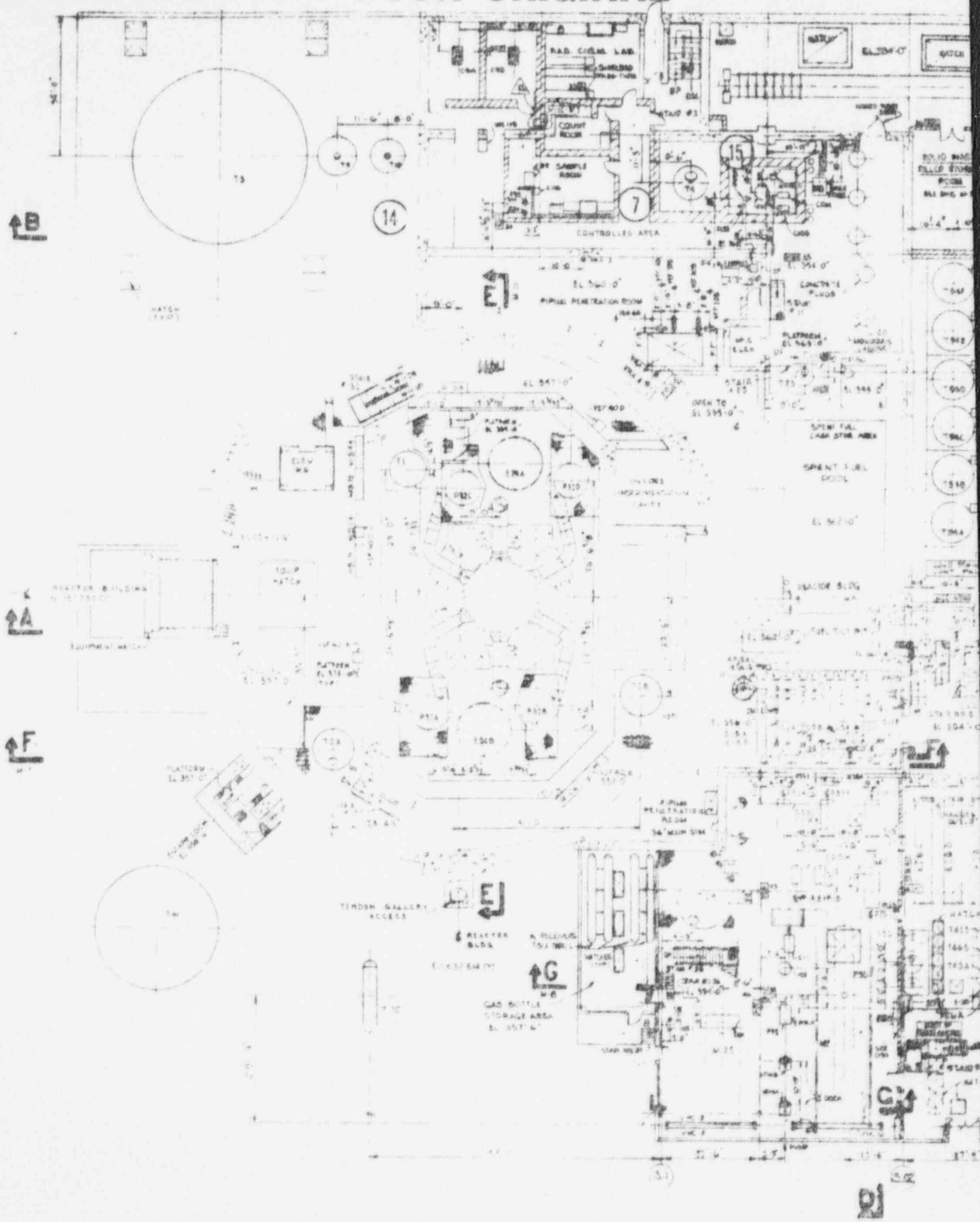
FIGURE 3-4
UNIT I PLAN BELOW GRADE SHOWING
ANALYZED DOSE POINTS
REF: AP & L DRAWING M-6, REV. 10

-27 90030218

POOR ORIGINAL



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POOR ORIGINAL

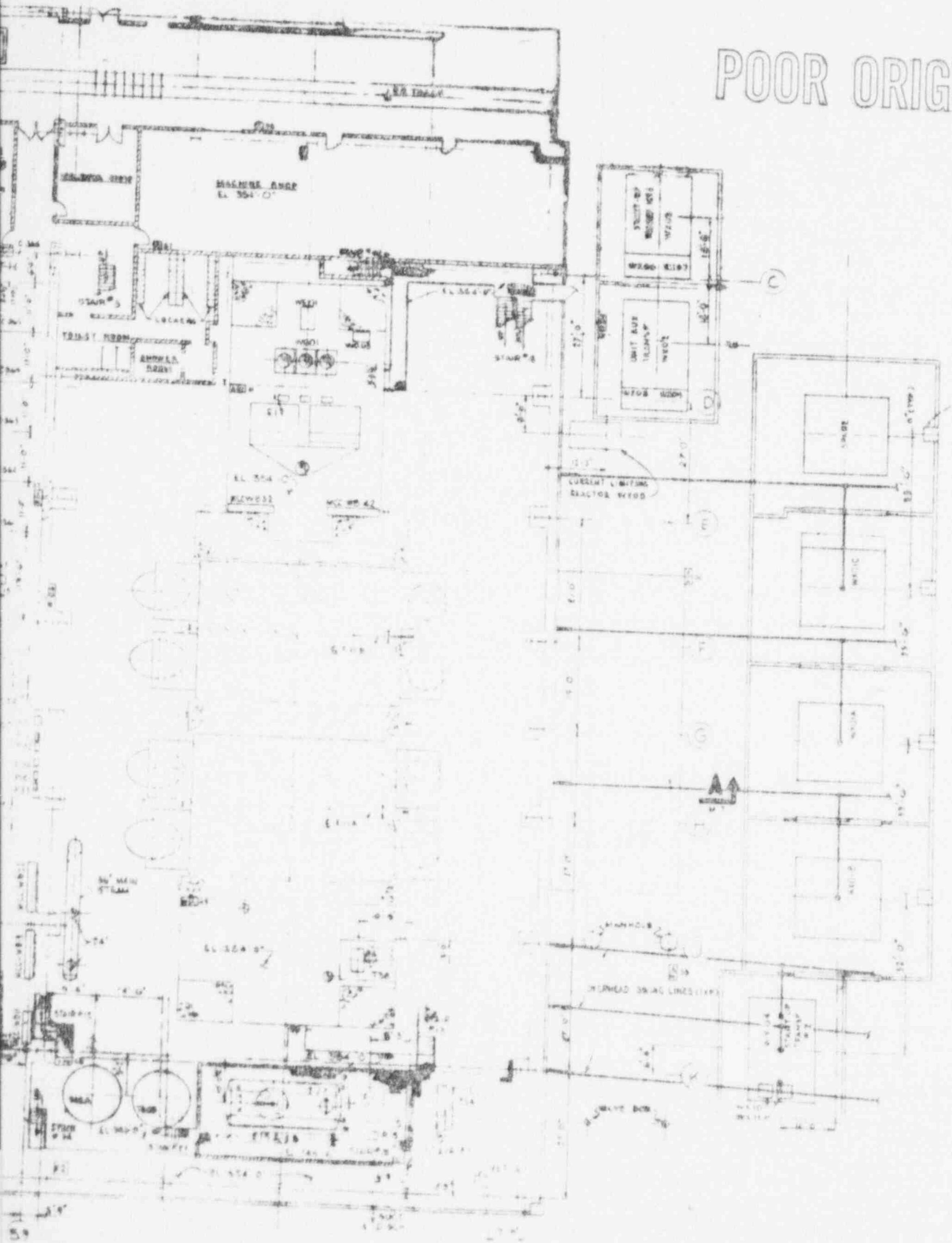

FIGURE 3-5
UNIT I GROUND FLOOR PLAN SHOWING
ANALYZED DOSE POINTS
REF: AP & L DRAWING M-5, REV. 12

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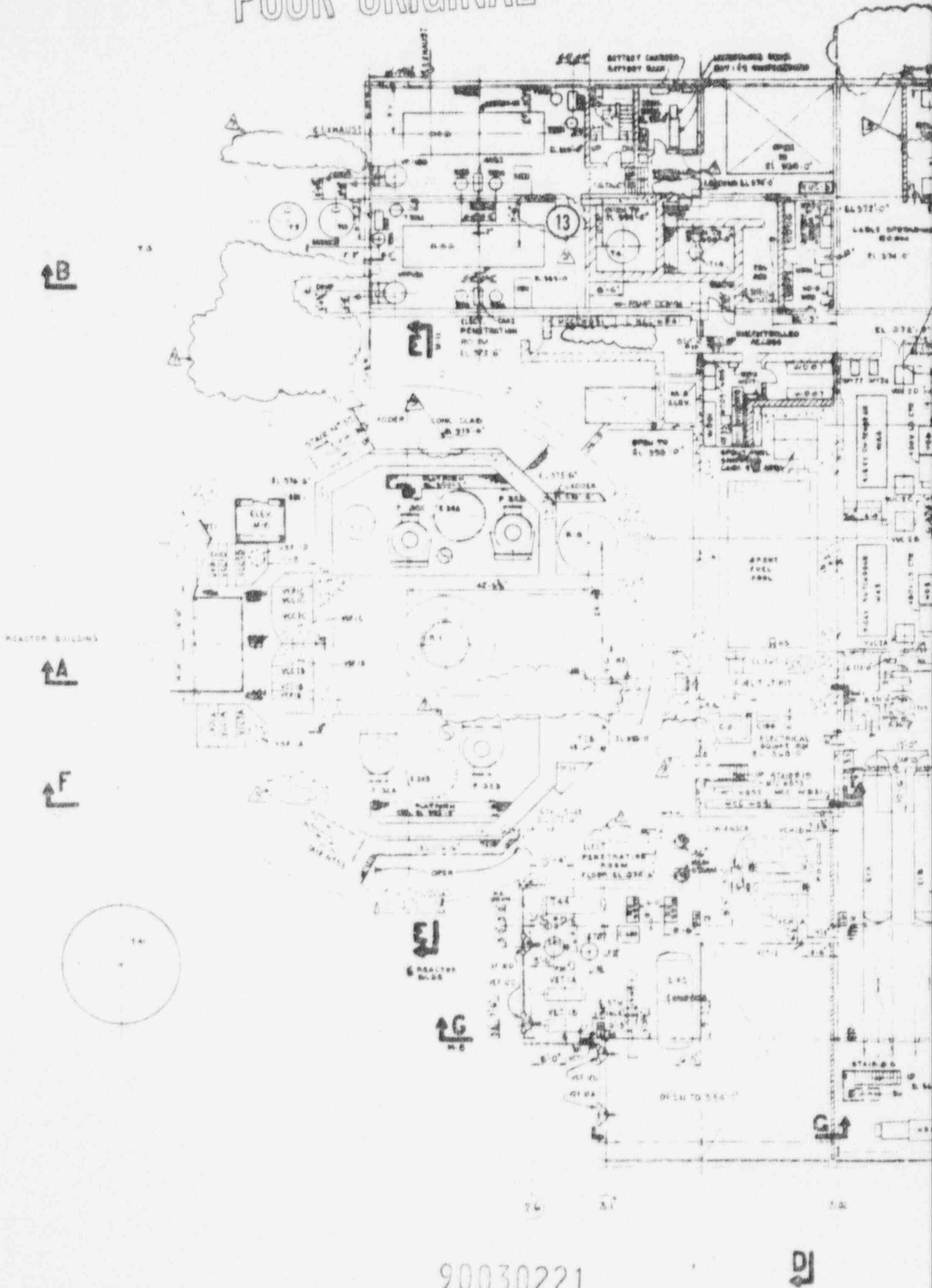
POOR ORIGINAL



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POOR ORIGINAL

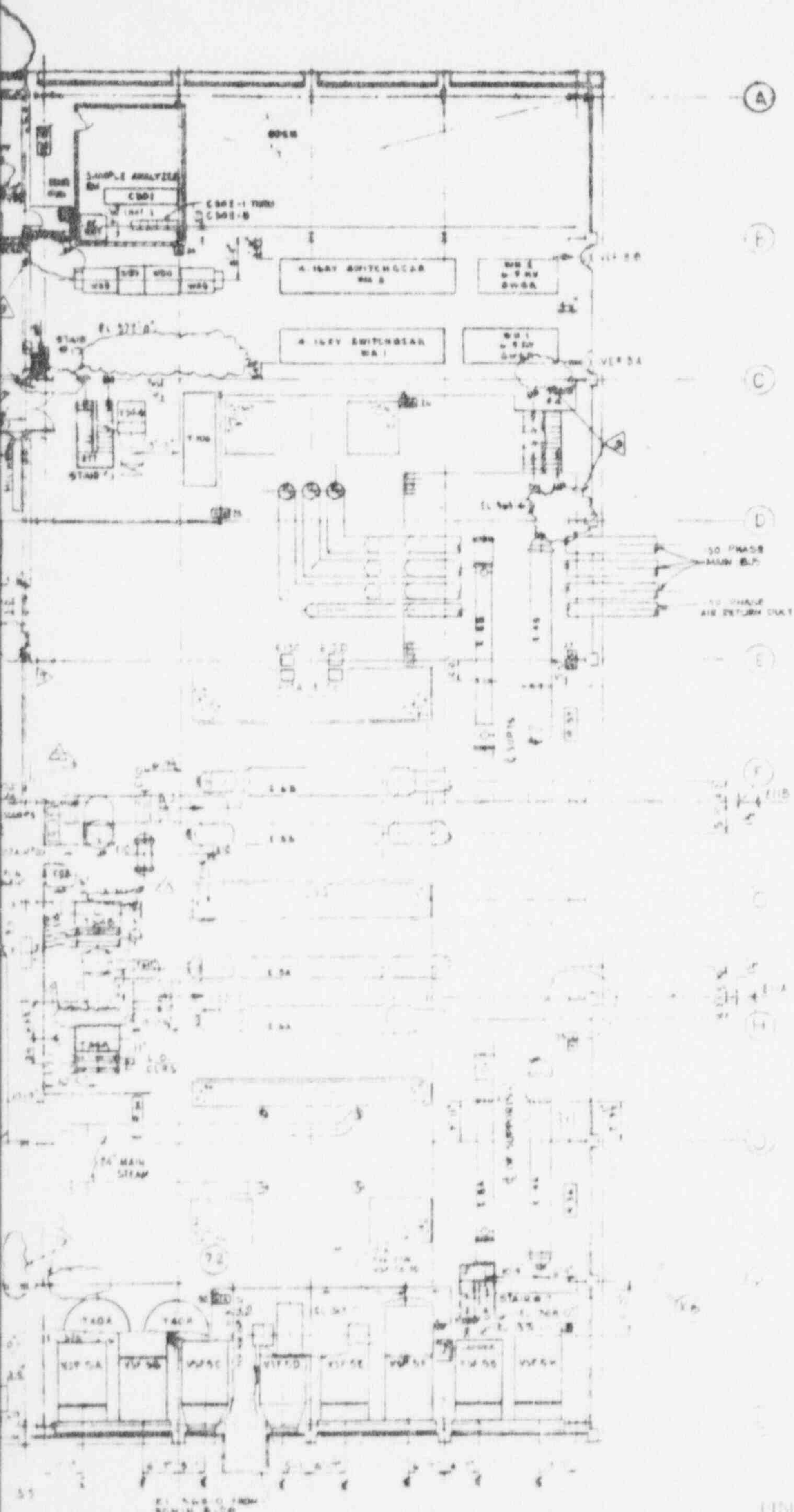
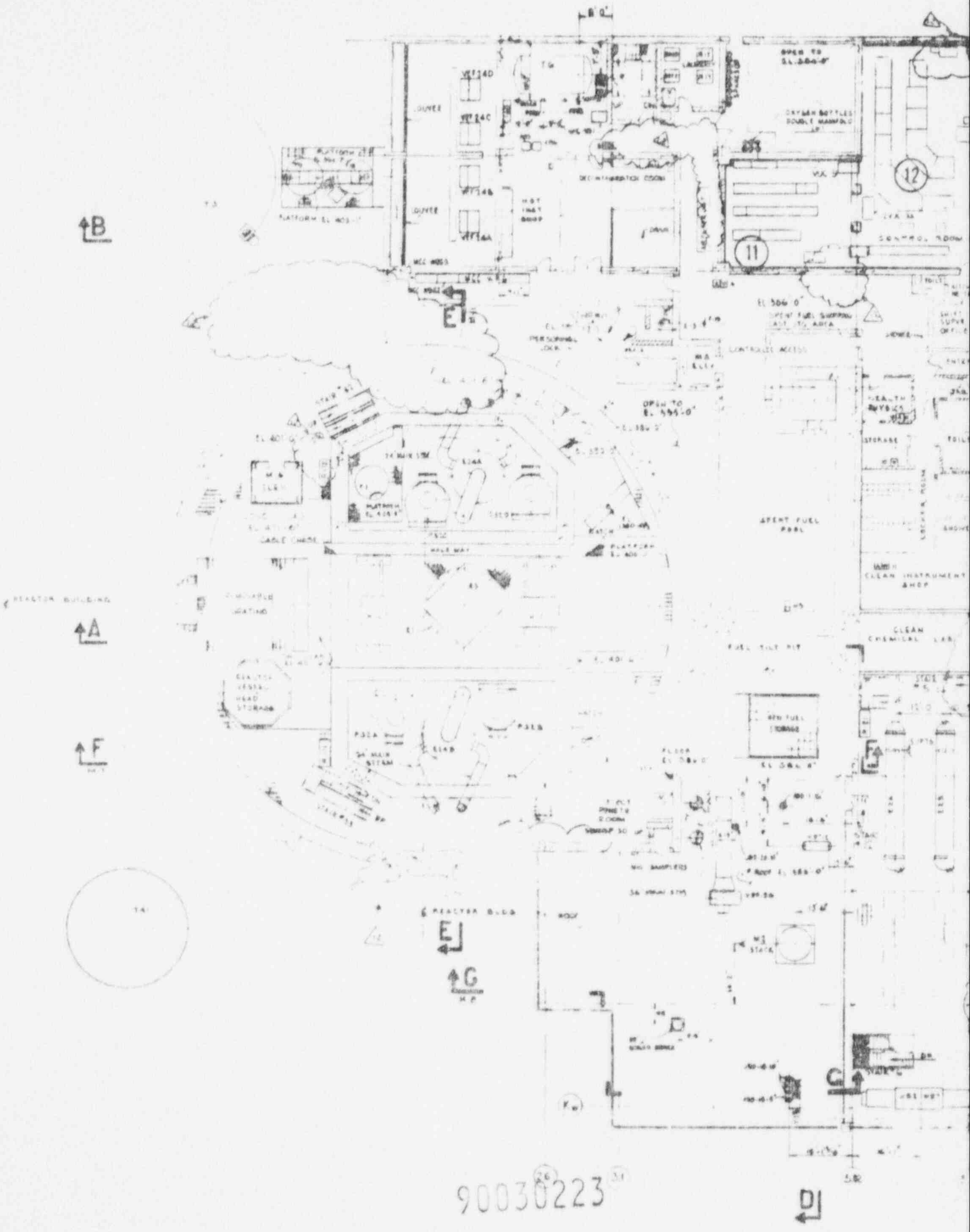


FIGURE 3.6
UNIT I INTERMEDIATE FLOOR PLAN SHOWING
ANALYZED DOSE POINTS
REF: AP & L DRAWING M-4, REV. 9

9003022 2

POOR ORIGINAL



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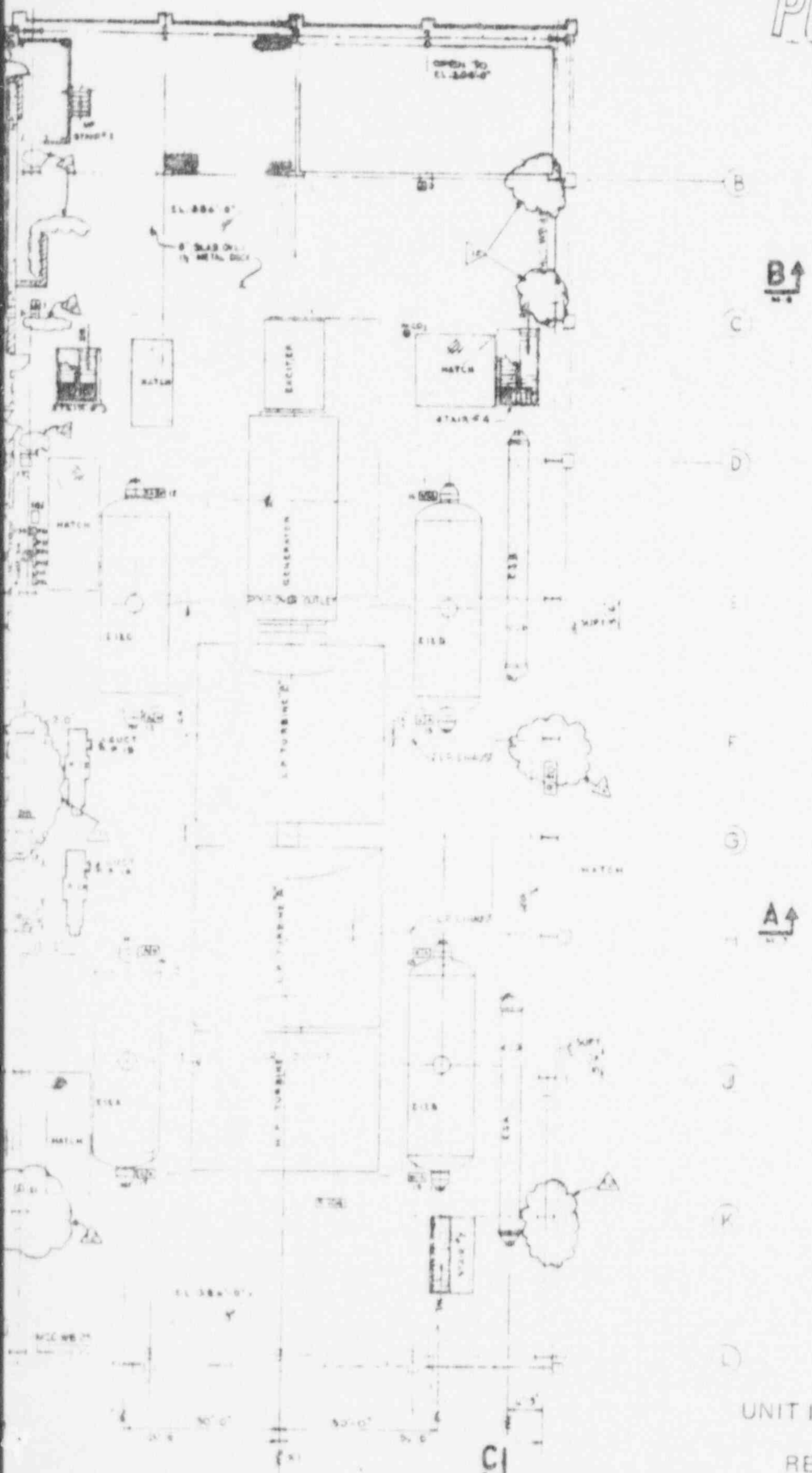

POOR ORIGINAL

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FIGURE 3-7
UNIT I OPERATING FLOOR PLAN SHOWING
ANALYZED DOSE POINTS
REF: AP & L DRAWING M 3, REV. 12

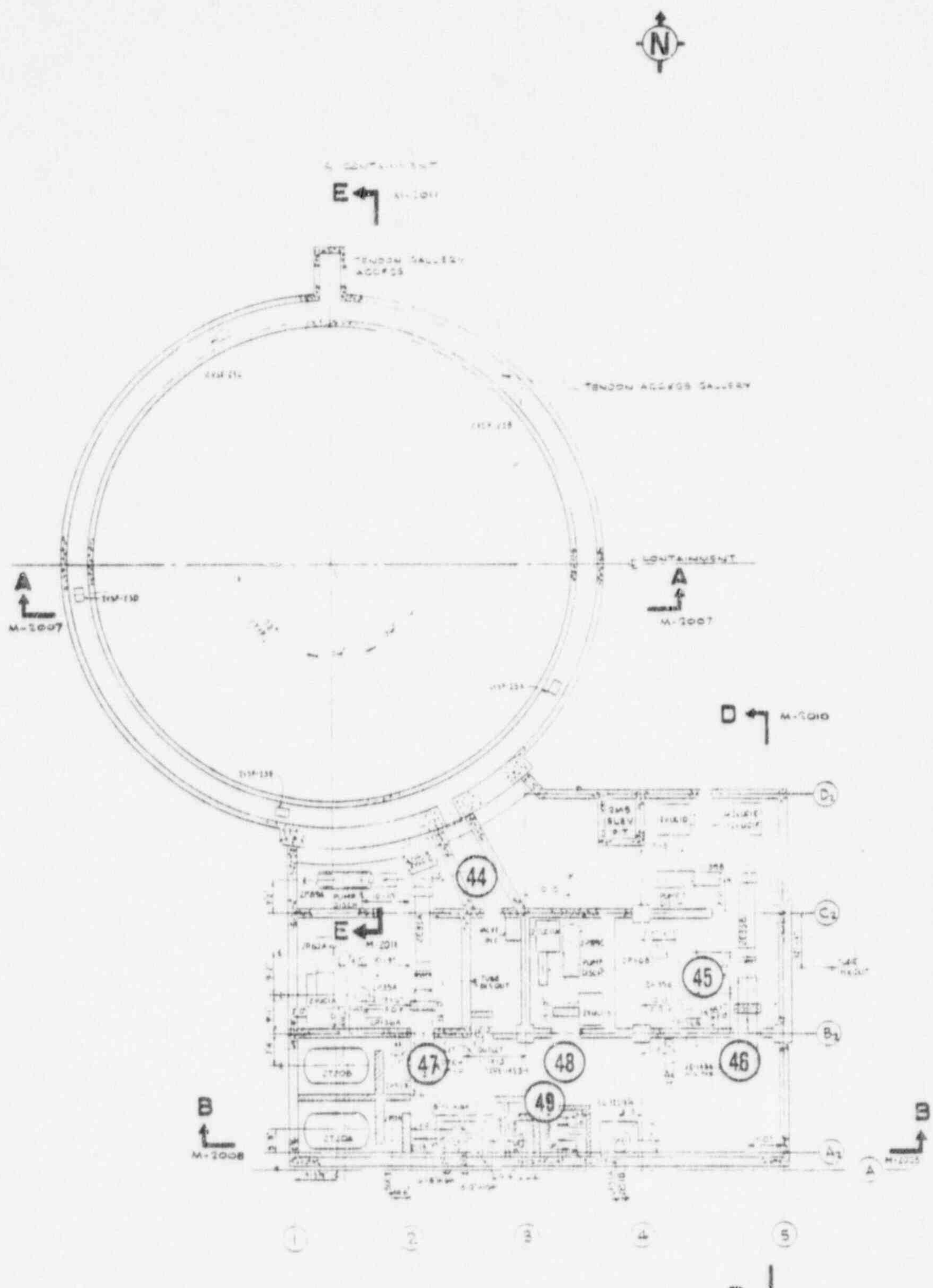

FIGURE 3-8
UNIT 2 PLAN BELOW GRADE
(EL-317) SHOWING ANALYZED DOSE POINTS
REF: AP & L DRAWING M-2011 REV. 15

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POOR ORIGINAL

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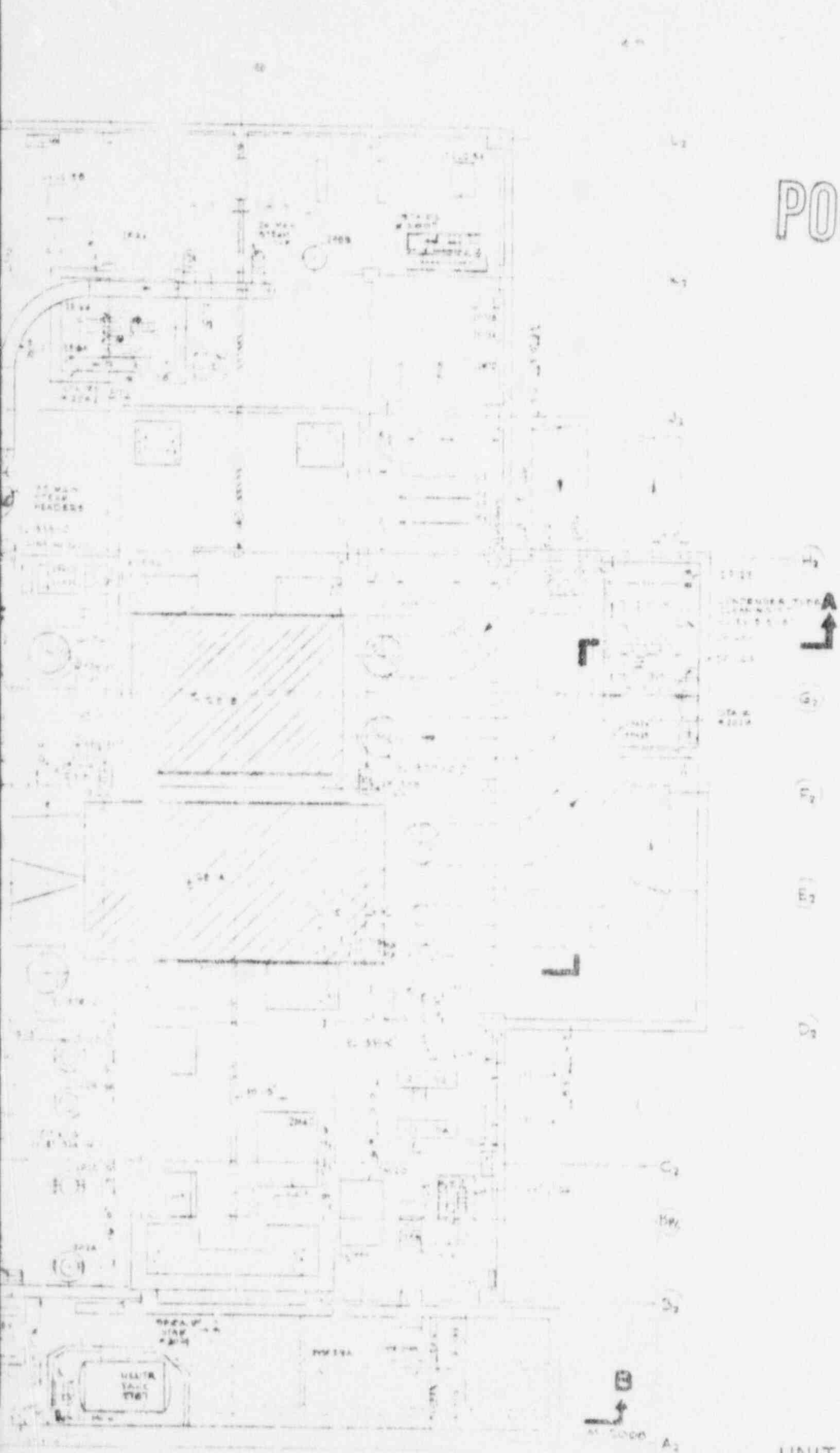

POOR ORIGINAL

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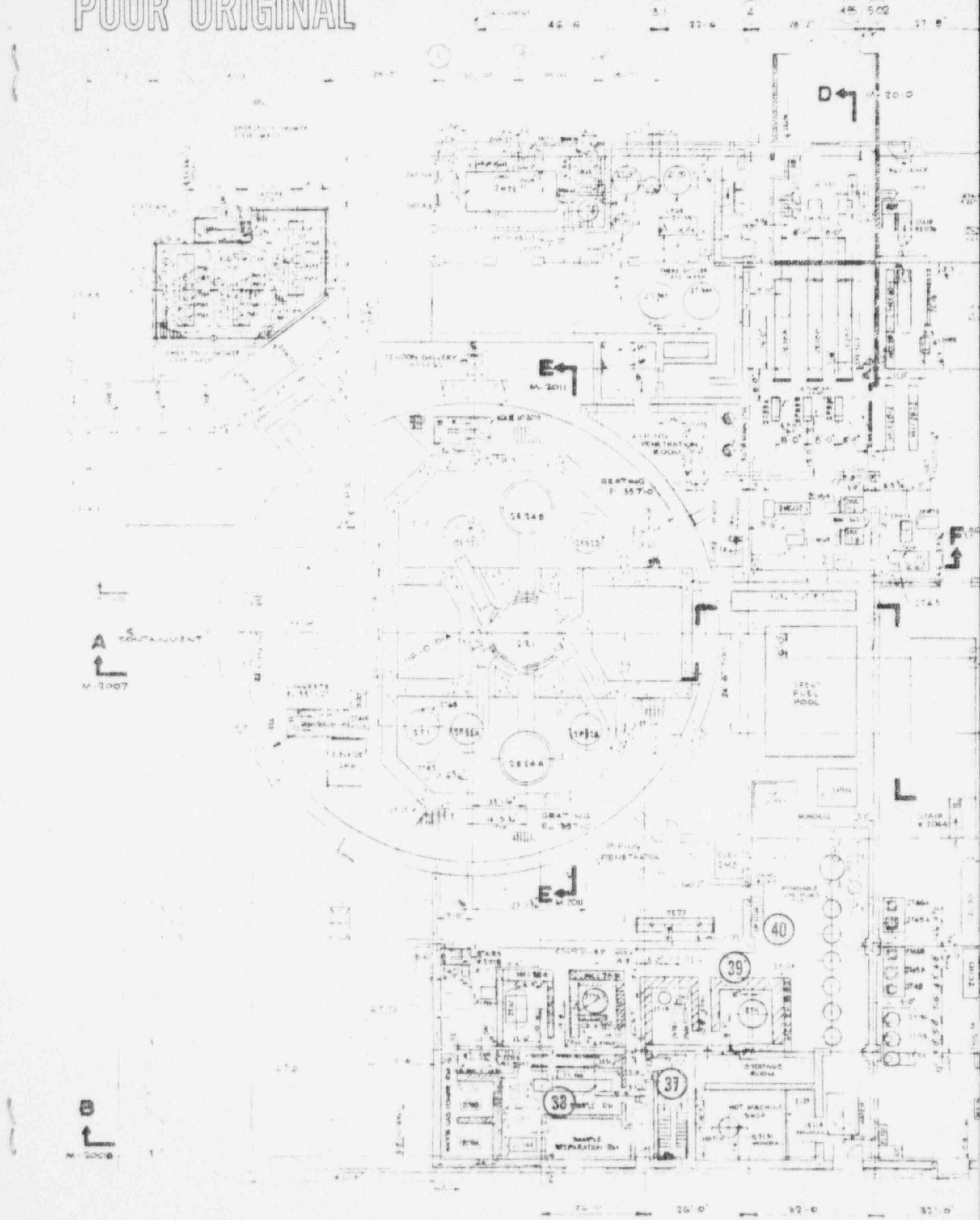
FIGURE 3-9
UNIT 2 PLAN BELOW GRADE SHOWING
ANALYZED DOSE POINTS
REF: AP & L DRAWING M-2006, REV. 25

POOR ORIGINAL



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POOR ORIGINAL

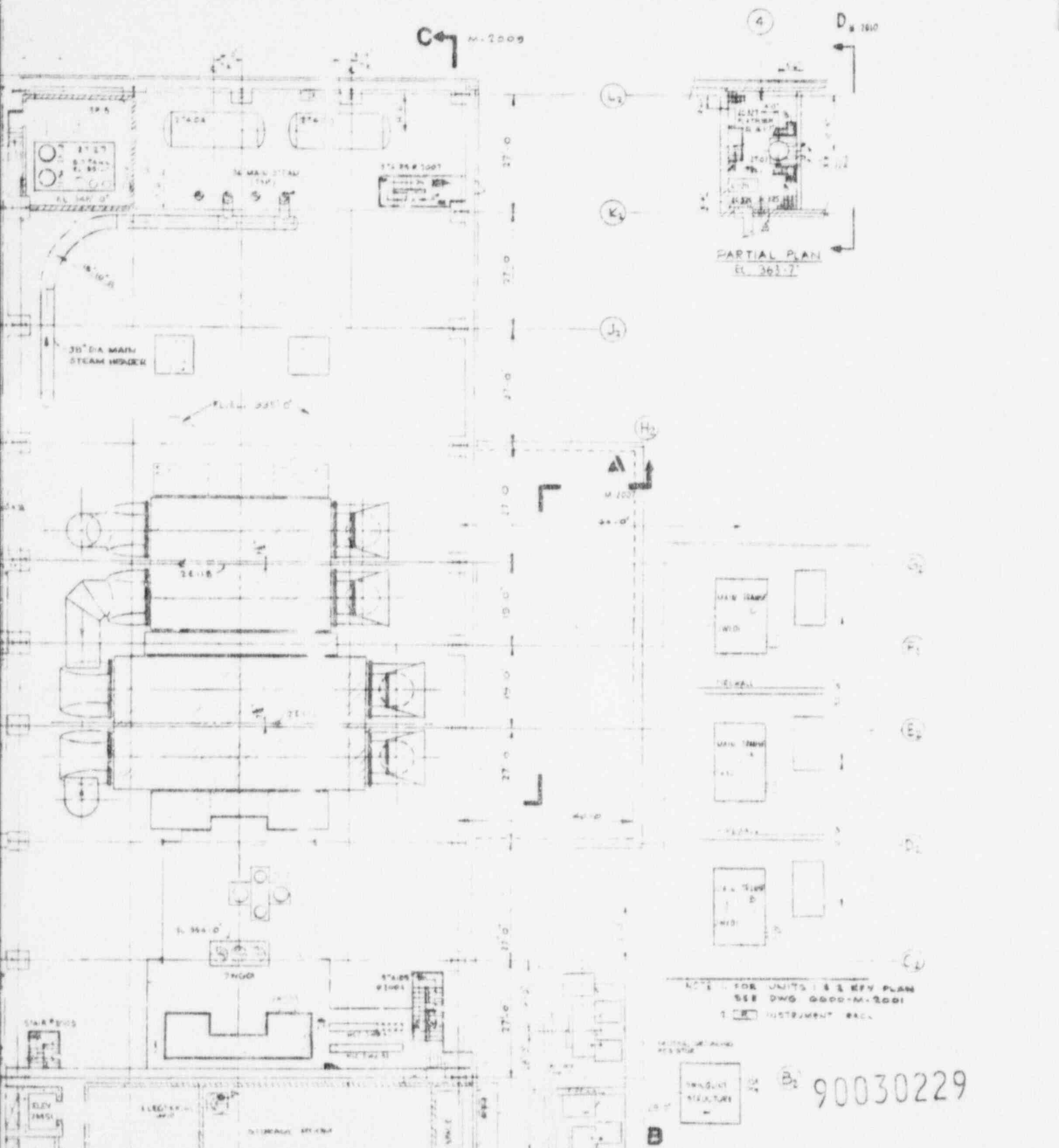

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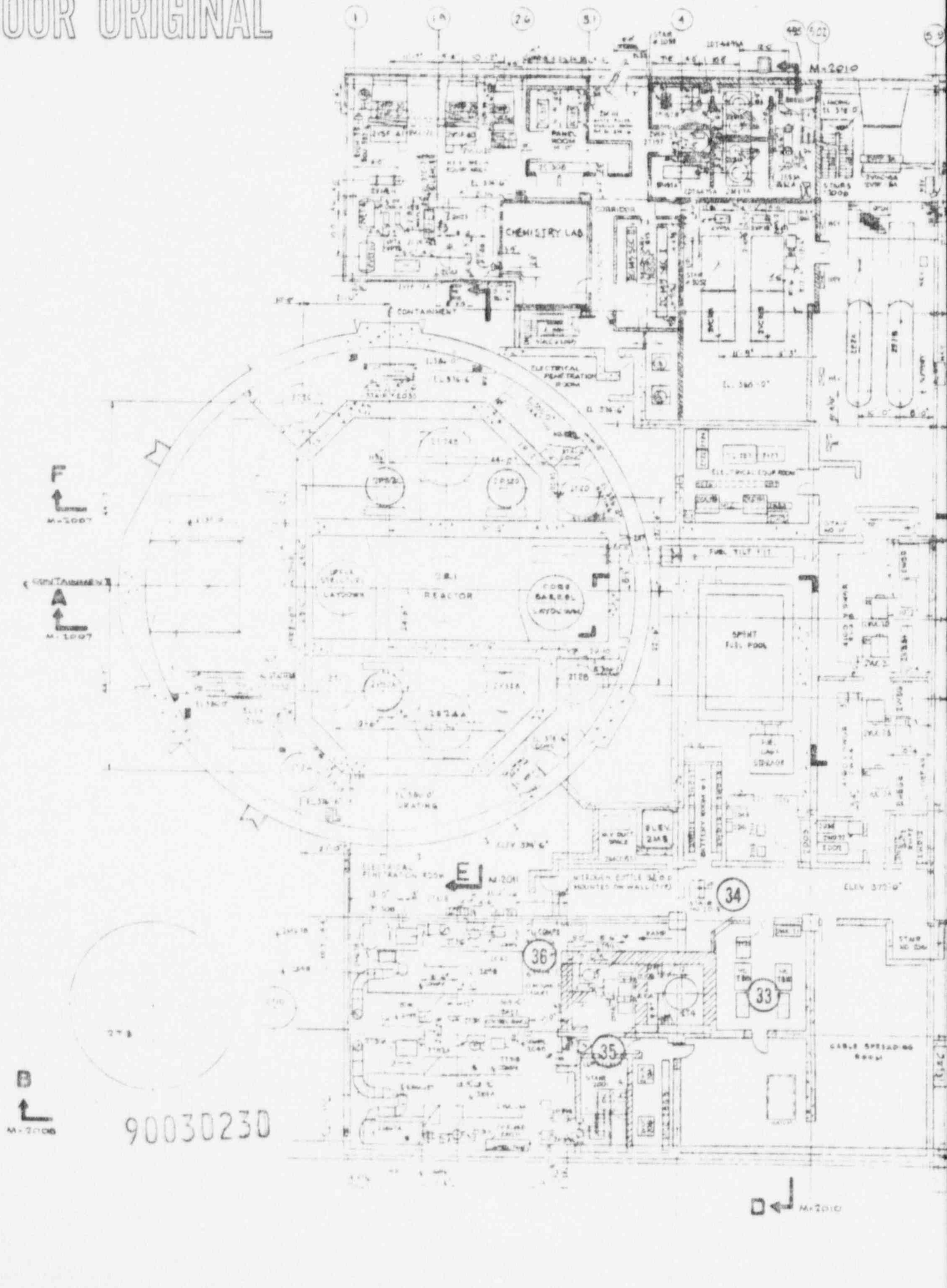
FIGURE 3-10
UNIT 2 GROUND FLOOR PLAN
SHOWING ANALYZED DOSE POINTS
REF: AP & L DRAWING M-2005, REV. 26

POOR ORIGINAL



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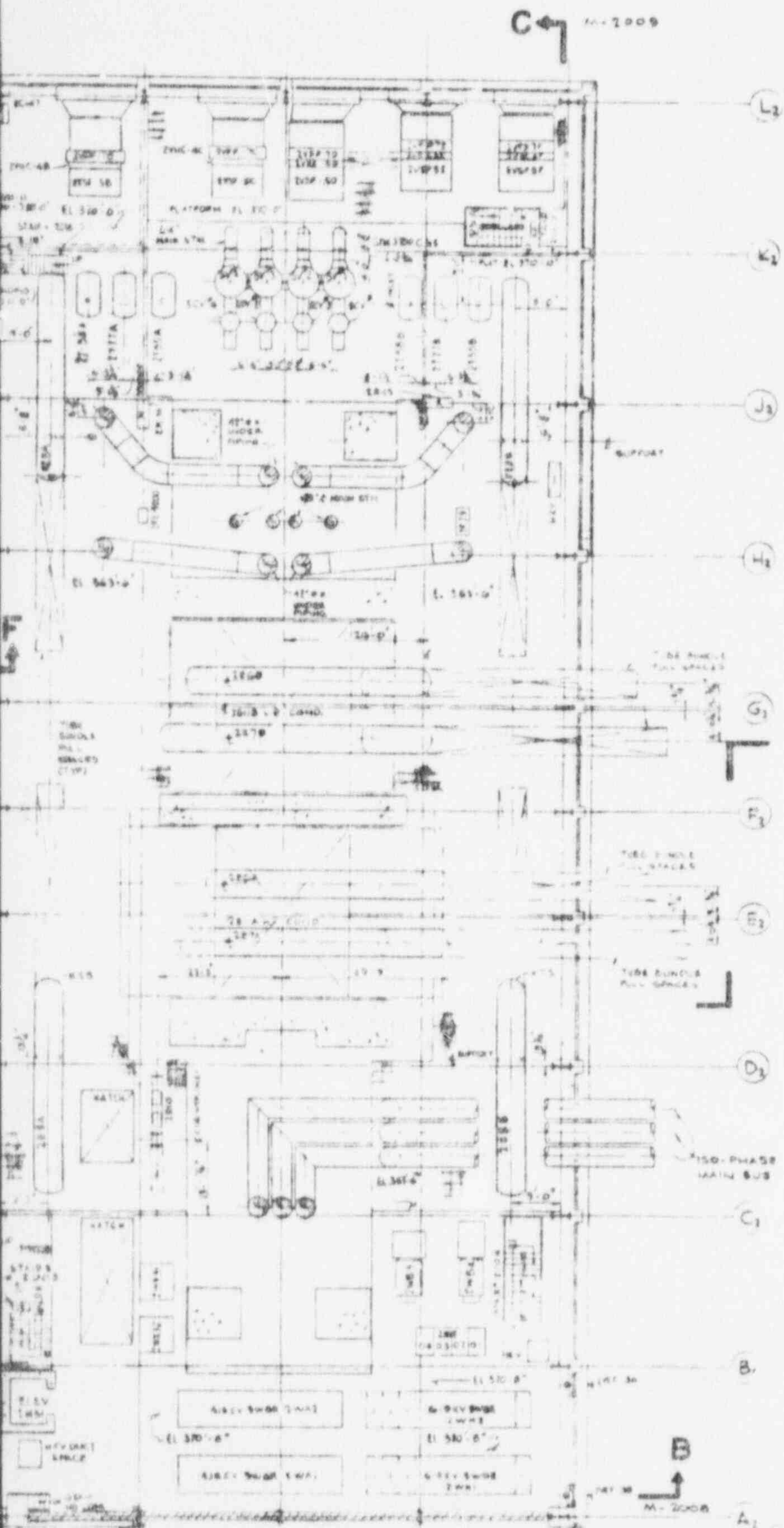

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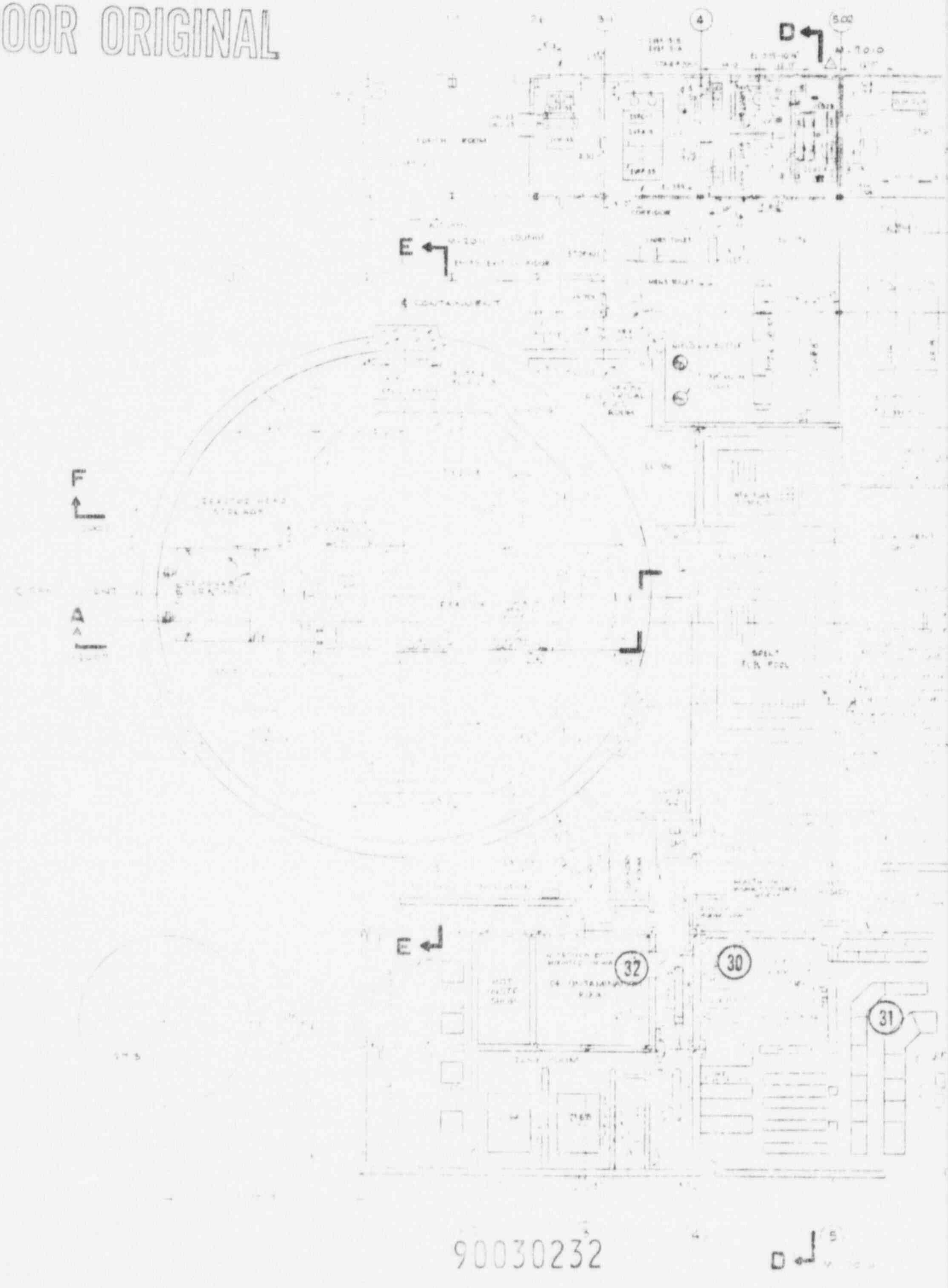
FIGURE 3-11
UNIT 2 INTERMEDIATE FLOOR PLAN
SHOWING ANALYZED DOSE POINTS
REF: AP & L DRAWING M-2004, REV. 20

POOR ORIGINAL



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POOR ORIGINAL

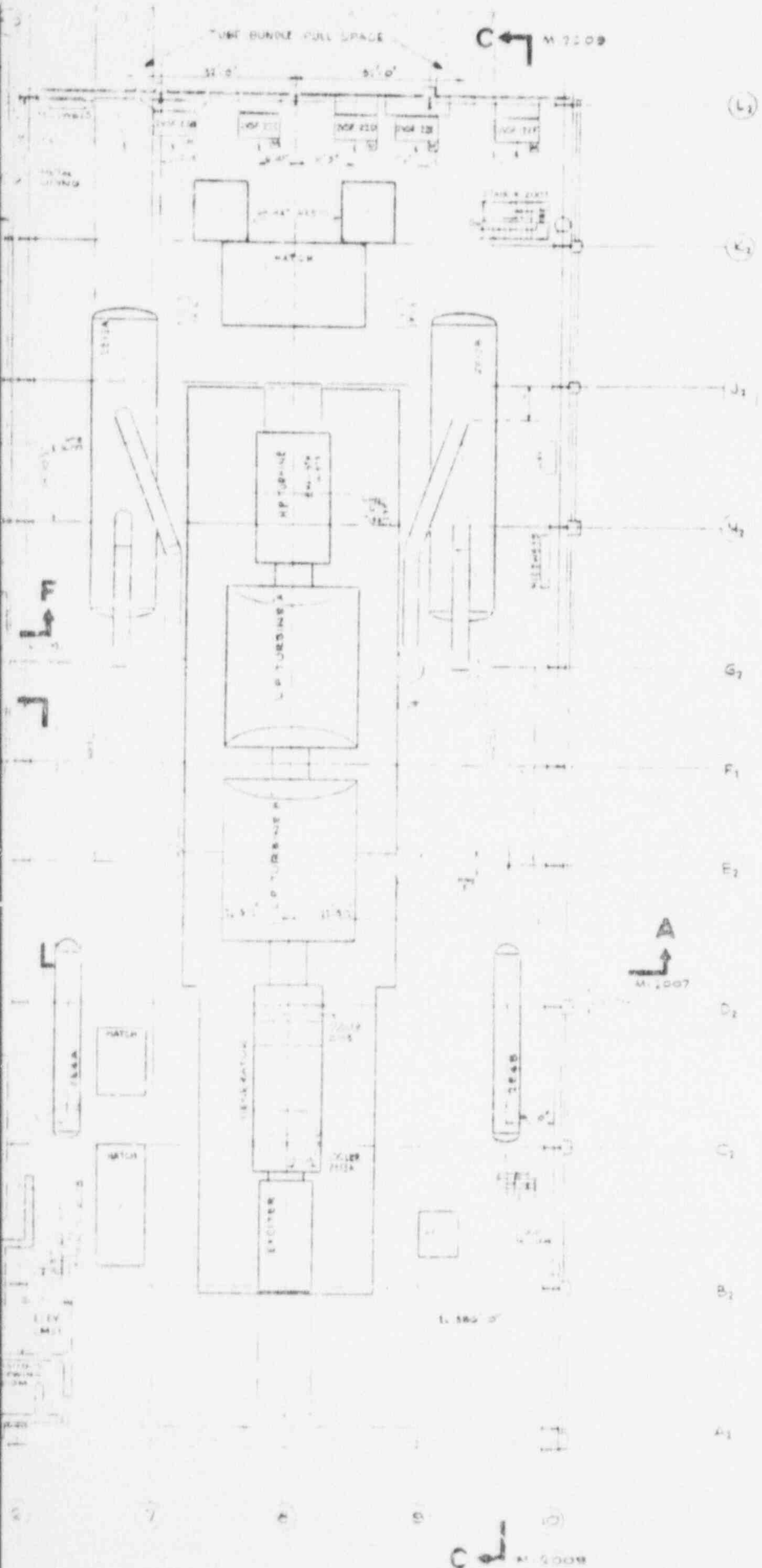

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FIGURE 3-12
UNIT 2 OPERATING FLOOR PLAN
SHOWING ANALYZED DOSE POINTS
REF: AP & L DRAWING M-2003,
REV. 19

4.0 IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

4.1 EXISTING PRIMARY COOLANT SAMPLING SYSTEMS

4.1.1 UNIT 1

The present sampling station for Unit 1 is located on elevation 354 adjacent to the reactor coolant makeup tank and waste gas compressors. This sampling station permits the taking of reactor coolant samples from the letdown line, pressurizer water space, pressurizer steam space, and decay heat cooler outlet.

4.1.2 UNIT 2

The present sampling station for Unit 2 is also located on elevation 354. Unlike Unit 1, this sampling station is farther away from the reactor coolant makeup tank but is also adjacent to the waste gas compressors. This station permits the taking of reactor coolant samples from a reactor coolant hot leg, pressurizer surge line, pressurizer steam space, decay heat removal/safety injection system, and letdown/purification system.

4.1.3 COMMON FEATURES OF UNITS 1 AND 2

Because the sampling station of Unit 2 is adjacent to the sampling station of Unit 1 and connected by a passageway, the count room and chemical lab in Unit 1 also serves Unit 2.

4.2 REQUIREMENTS FOR SAMPLING: PRE- AND POST-ACCIDENT

4.2.1 NORMAL SAMPLING

Primary coolant sampling is performed on a regular basis; sampling provides a feedback control feature essential for long-term reactor operation. Based on analysis of the sampled primary coolant, action can be taken to adjust boron levels and water chemistry.

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In addition, the integrity of the reactor core can be assessed by measuring the presence of fission products in the primary coolant. Those items that must be determined by sampling and the justification for each are listed below:

Boron concentration	- part of the reactivity control and safe shutdown margin
Hydrogen concentration	- hydrogen is required to keep the oxygen concentration to a minimum
Oxygen concentration	- high oxygen concentration can lead to corrosion and crud buildup on the core and core internals
Chlorides	- the presence of chlorides should be kept to a minimum to prevent embrittlement and cracking of metals
Radionuclides (noble gases, iodine, and cesium)	- their presence in the reactor coolant indicates a degree of fuel-rod failure

4.2.2 POST-ACCIDENT PRIMARY COOLANT SAMPLING

Following an accident, primary coolant sampling will be required to provide the following data:

Boron concentration	- to ascertain the safe shutdown reactivity margin; sample should be taken within 1 hour after accident

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Radionuclide analyses (Noble gases, Iodine, and other non-volatile fission products	- to determine degree of core damage; sample should be taken within Cesium 2 hours of accident
Chlorides	- NRC requirement; sample should be taken within one shift

4.3 SAMPLING PROBLEMS ASSOCIATED WITH HIGH-ACTIVITY REACTOR COOLANT

4.3.1 HAZARDOUS DOSES TO PERSONNEL

There are essentially two major dose-related problems to solve in order to sample the reactor coolant with high activity levels resulting from a fuel-failure event. The first problem is that (in Unit 1) dose levels (from the reactor coolant makeup tanks) at the sampling stations and in the access corridors of the auxiliary building can preclude access to the Unit 1 sampling station. In Unit 2, the dose levels are considerably lower; access to and occupancy of the Unit 2 sampling stations is possible for short periods of time. Any reactor coolant leakage into the auxiliary building (e.g., from the reactor coolant makeup pump seals) will result in airborne activity that will cause large whole-body (immersion) doses.

The second major problem is that the present sampling equipment and reactor coolant sample itself will cause hazardous doses on the order of 2,000 to 4,000 rem/hr at 1 meter away from the equipment. The airborne activity that could result from attempting to take a sample would also cause an extremely hazardous condition.

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4.3.2 POST-ACCIDENT DOSES AT PRIMARY COOLANT SAMPLE STATION

In the event of a loss-of-coolant accident, automatic containment isolation will cause isolation of the letdown line. This isolation is desirable in that it will prevent high-activity reactor coolant from entering the makeup and purification system (i.e., the regenerative heat exchanger, reactor coolant filters, demineralizers, makeup tank, and makeup pumps). Unless there is leakage past the isolation valves, the total dose to the existing sampling stations from the adjacent reactor makeup tank and waste gas compressors would be essentially the same as before the accident. Uner these conditions, it would be possible to get to the sampling station, but the doses from the sample and sampling equipment would still be hazardous. If, however, post-accident reactor coolant is processed by the letdown system (which may be required for degassing operations or for boron control), then the doses from the reactor makeup tank in the corridors and sampling room (Unit 1) would be extremely hazardous, as shown on Table 3-2. The doses at the Unit 2 sampling stations are significantly less hazardous, as shown on Table 3-3.

4.3.3 PRIMARY COOLANT SAMPLING LOCATIONS

4.3.3.1 Unit 1

Unit 1 has three reactor coolant sampling locations that draw reactor coolant into the sampling station. Samples are normally taken from the letdown system just upstream of the demineralizer. Because the letdown system is automatically isolated on a safety injection signal, it will not be possible to take a sample from this point unless the letdown flow is resumed.

However, with the letdown system isolated, samples can still be taken from the pressurizer liquid, steam space, and decay heat cooler sample lines if modifications are made. Because molecular diffusion will tend to keep concentrations of dissolved chemicals

uniform throughout the reactor coolant system, the steady-state concentrations of boron, nuclides, and other chemicals in the pressurizer liquid leg should be essentially the same as the steady-state concentration in the main reactor coolant loops. Therefore, by allowing time to pass for concentration gradients to equalize, samples extracted from the pressurizer liquid phase should be representative of the reactor coolant that would normally be sampled from the letdown system.

The pressurizer liquid sample is brought via 3/8-inch tubing into the sampling station where it is first cooled by the shielded pressurizer sample cooler. It is estimated that the personnel dose from this shielded cooler is approximately 600 to 1,000 rem/hr (in addition to any other contributing doses). Therefore, it is concluded that even with the letdown leg isolated (which would allow personnel to enter the sampling station without being subjected to hazardous doses from the reactor makeup tank), the direct shine doses from the pressurizer sample cooler would be very hazardous, in addition to the doses that would be received from the sample specimen (bomb).

It is recognized that the pressurizer liquid sample is not the most desirable sample location for radionuclides because the sample does not come from the main reactor coolant loop. However, short of making any modifications to the Unit 1 reactor coolant piping (e.g., providing for a sample point in the hot leg), the pressurizer liquid sample is the most attractive alternative.

4.3.3.2 Unit 2

Unit 2 has a greater number of locations where reactor coolant can be drawn for samples. These locations are the reactor coolant letdown system, reactor coolant hot leg, pressurizer surge line, pressurizer steam space, and safety injection/shutdown cooling system. All of these samples are taken in the Unit 2 sampling room. As discussed above, it is possible to occupy the Unit 2

sampling room for short periods of time. However, the same hazards encountered with the sample cooler and sample specimen in Unit 1 will also be present in Unit 2. In addition to the pressurizer liquid sample, Unit 2 is provided with the capability for taking samples from one of the reactor coolant system hot legs and from the safety injection/shutdown cooling system.

letdown system will be automatically isolated if an accident occurs. However, even though letdown flow is terminated, the boronmeter should still provide a reading that is indicative of the reactor coolant boron concentration before any borated water is injected into the reactor coolant system. This information can provide important data for prediction of the shutdown margin.

4.5 CONTAINMENT AIR SAMPLING

4.5.1 CONTAINMENT AIR SAMPLING EQUIPMENT

4.5.1.1 Radionuclide Analysis

The present containment air sampling equipment for Unit 1 and 2 draws containment air through a particulate paper and charcoal cartridge for a specific time period. The cartridge then removed and is taken to a count room, where the accumulated activity is counted, both on the filter paper (for particulate) and on the charcoal (for iodine).

4.5.1.2 Hydrogen Monitoring

In the event of a loss-of-coolant accident, the reactor core could reach temperatures leading to a Zircaloy-water reaction, and hydrogen gas could be released to the containment (in addition to the normal hydrogen that would be liberated from the primary coolant). It is desirable to monitor the containment for hydrogen so that the hydrogen recombiners can be adequately employed to maintain the hydrogen concentration at safe levels, and to provide indication of a possible Zircaloy-water reaction.

4.5.2 UNIT 1 CONTAINMENT AIR SAMPLING STATION

The Unit 1 containment air sampling station is on elevation 335 (in a radiation Class IV area) near the containment wall, opposite the shield walls for the reactor coolant filters. Approximately 30 feet from the sampling station in the same room is the shielded seal water injection filter and other decay heat and high pressure injection piping. The estimated maximum dose received in attempting to reach the sampling station is 6500 R/hr in the walkway corridor (from the reactor coolant filter) plus an additional dose of approximately 1000 R/hr from the seal water injection filter and other nearby piping. If the containment air sampling equipment is moved approximately ten feet to the other side of the adjacent 2.5-ft.-thick shield wall (separating this area from the adjacent radiation Class II zone), then the dose at this new sampling location would be reduced to about 6.5 R/hr, which is primarily due to the reactor coolant filter.

The present nuclide sampling apparatus is as described in subsection 4.5.1.1. However, based on information supplied to date, hydrogen monitoring is performed by physically obtaining a containment air sample in an evacuated bottle (approximately 100cc in volume), using sample equipment at the same location; the bottle is then taken to a laboratory for analysis.

It is estimated that the dose from a 100cc glass bottle (containing a containment air sample taken immediately after the accident) at 0.1 meter is approximately 116 R/hr; 1.10 R/hr at 1.0 meter. Two hours after the accident, this same sample would give a dose at 0.1 meter of approximately 20 R/hr.

4.5.3 UNIT 2 CONTAINMENT AIR SAMPLING STATIONS

Unit 2 has two separate containment air sampling stations. One station is a continous air monitor located on elevation 360 in piping penetration room 2084, adjacent to the shielded seal water

heat exchanger and also decay heat and high/low pressure injection piping. The dose from this equipment is estimated to be on the order of 5,000 to 10,000 R/hr.

The other containment air sampler is a bottle sampler located near the spiral staircase in containment penetration area 2081, on elevation 356. This area is shielded from equipment and piping containing reactor coolant so that shine doses are not significant and personnel can enter this area to take a sample. However, the dose received from the sample bottle itself will be of concern; the dose from a 100cc bottle is approximately the same as that identified in subsection 4.5.2.

Unit 2 also has containment air hydrogen analyzers located in the reactor coolant sampling room. Fortunately, the Unit 2 sampling room is sufficiently shielded from the volume control tank to allow access to the sampling station from several different entrances. The dose received at the stairway entrance leading to the sampling room is on the order of 3 R/hr (from the VCT) but approximately 50 R/hr from the vacuum degasification tank, if the degasification system is processing reactor coolant. Within the sample room, the doses from the volume control tank and vacuum degasification tank (if operating) are 0.2 R/hr and 2 R/hr, respectively. These dose levels may be tolerated for a short period of time while obtaining a hydrogen analysis of the containment air. Doses received from the containment hydrogen analyzer itself have not been estimated.