Report No. WNP #2-01

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

INTERIM SHIELDING EVALUATION RADIATION REPORT

REVISION 2

Report Date: December 1981

REPORT APPROVAL COVER SHEET

Project: Washington Public Power Supply System WNP-2

Title: Shielding Evaluation

Report# WNP#2-01

Revision# 2

Prepared By: Loren Sharp

Reviewed By: Jeffrey Ozawa

Approval By: W. L. Conn

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CONTENTS OF THIS REPORT

Burns and Roe Incorporated (BRI) performed the analysis of radiation levels occurring inside Primary Containment; assembled, edited, reviewed, and approved this technical report for the Washington Public Power Supply System.

EDS Nuclear Incorporated performed the analysis of radiation levels occurring in the Reactor Building Secondary Containment under subcontract to BRI.

The Washington Public Power Supply System performed the analysis of radiation levels occurring in areas outside the Reactor Building Secondary Containment.

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SUMMARY

The Three Mile Island (TMI-2) accident has generated a concern that during an accident in which significant core damage occurs, the post-accident operations requiring the use of systems containing contaminated fluid may induce abnormally high radiation doses to safety-related equipment and components which make it difficult to operate the systems. The NRC initially addressed this concern with NUREG-0578 and NUREG-0737 and recommended a design review to evaluate the functional capability of safety-related equipment and radiation exposure to personnel during the postulated post-LOCA operations.

Radiation levels have been determined for all areas containing safety-related equipment, vital areas, and access routes which are required for the postulated post-LOCA operation.

Radiation levels are currently being determined for safety-related equipment inside primary containment. The safety-related list of equipment located inside primary containment is identified in Table 6.1. Analysis effort in the shadow shielding effect of primary containment hardware and the effect of first order iodine plateout is currently in progress to calculate the radiation levels inside containment.

Radiation levels determined for safety-related equipment in Secondary Containment are reported in Figures 6.1 through 6.8. An analysis effort is continuing in Secondary Containment due to anticipated impact of considering iodine

plateout and time/pressure history of containment following a LOCA. The radiation source term leaking into Secondary Containment will be reduced by the loss of halogens to plateout. Also a reduced pressure inside Containment will impact the pressure related time dependent leakage into Secondary Containment. These impacts are being evaluated to more accurately predict radiation levels inside Secondary Containment.

Beta radiation inside Secondary Containment has also been considered and can be determined from Figure 5.1 and Figures E-1 through E-9.

Radiation levels calculated for safety-related equipment outside Secondary Containment are reported in Table 6.3. Figures 6.19 through 6.23 identify the vital areas outside Secondary Containment which contain safety-related equipment.

Safety-related equipment will either be qualified for the radiation level it functions in, or it will be relocated to a radiation zone it is qualified for, or it will be replaced with comparable equipment which is qualified for the particular radiation level that has been determined.

Vital Areas and Access Routes were evaluated for post-LOCA operations and are reported in Table 6.4 and Figures 6.19 through 6.24. All areas and access routes are in compliance with NUREG-0737 except the security guardhouse and the auxiliary

security center. Security personnel will be relocated in the Technical Support Center per the WNP-2 Emergency Preparedness Plan if guardhouse radiation levels approach those radiation guidelines presented in NUREG-0737 during the post-LOCA situations.

ABSTRACT

This report presents a radiation shielding design review of the equipment and systems of the Washington Public Power Supply System Nuclear Project Unit 2 (WNP-2) that may, as a result of an accident, in addition to normal plant radiation levels during its 40 year life contain highly radioactive fluids. This design review recommended by the NRC (NUREG 0578 and NUREG 0737) evaluates the Functional Capability of safety-related equipment and personnel radiation exposure during the post-accident operations.

This design review evaluates the post-accident radiation conditions for personnel located in vital areas (areas which require access or occupancy during the post-LOCA scenario) on either a continuous or infrequent basis.

The postulated Loss of Coolant Accident (LOCA) scenarios and the operations of the safety-related systems were reviewed. Radioactive sources contained within each system were developed. Radiation levels were calculated at safety-related equipment locations, as well as at selected locations outside the Reactor Building to which access may be required for post-accident operations.

1.0 INTRODUCTION

This report presents a detailed description of and the results from the review of plant shielding and radiation environmental conditions for equipment and systems which may be used in post-accident operations for WNP-2. The review was initiated in response to Section 2.1.6.b of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendation" and to Part II.B.2 of NUREG-0660 "NRC Action Plan Developed as a Result of the TMI-2 Accident."

The design review determined the post-accident radiation environmental conditions for equipment required for post-accident operations inside the Primary Containment, inside the Secondary Containment, and outside the Secondary Containment.

The six month total post-accident radiation dose rate as a function of time and the integrated dose are currently being calculated for safety-related equipment locations inside the WNP-2 Primary Containment. The six month total post-accident radiation dose rate as a function of time and the integrated dose were calculated at safety-related equipment locations inside the WNP-2 Reactor Building and at selected locations (vital areas) outside the Reactor Building.

Section 2.0 discusses the regulatory requirements upon which this report is based and provides a description of the tasks performed for this Shielding Evaluation.

Section 3.0 provides the systems review and source term assumptions used as input for the definition of the post-accident radiological environment.

Section 4.0 discusses the work performed during this project relating to safety-related-equipment located outside of the Reactor Building and the Access and Occupancy of Vital Areas. This consists of the calculation of dose rates outside the Reactor Building.

Section 5.0 discusses the Methods of Calculation including the use of computer codes, identifying the parameters that have a significant effect on the radiation dose rates, and the dose rate and cumulative dose calculation procedure.

Section 6.0 presents a summary of the results.

2.0 REQUIREMENTS

General Design Criterion 4 requires that systems and components important to safety be designed to accommodate the environmental conditions associated with accidents. The Three Mile Island (TMI-2) accident has generated a concern that during an accident in which significant core damage occurs, the post-accident operations requiring the use of systems containing contaminated fluid may induce abnormally high radiation doses to safety-related equipment and components which may make it difficult to operate the systems. The Nuclear Regulatory Commission (NRC) Lessons Learned Task Force initially addressed this concern in Section 2.1.6.b of NUREG-0578 (Ref. 2.1) and recommended a design review be performed on such systems such that the functional capability of safety-related equipment located in close proximity to the resulting high radiation field will not be unduly degraded.

Described in this section is a discussion of the current regulatory requirements and guidelines used.

2.1 Shielding Evaluation Regulatory Requirements

NUREG-0578 Section 2.1.6.b requires that each licensee perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The scope of the review includes the following:

1. Identification of the locations of vital areas and safety-related equipment.

2. Evaluation of the radiation level at each location.

3. Provision for adequate access to vital areas and assurances of post-accident equipment operation through design changes, increased permanent or temporary shielding, or post-accident procedural controls.

In order to perform this review, the NRC has provided guidance in the following documents ("documents of record").

- o NUREG-0578 Section 2.1.6.b Reference 2.1
- o NUREG-0588 Section 1.4 Reference 2.2
- o NUREG-0660 Section II.B.2 Reference 2.3
- o Clarification Letter to NUREG-0578, dated Sept. 5, 1980, Section II.B.2 Reference 2.4
- o NUREG-0737 Section II.B.2 Reference 2.5
- o IE Bulletin No. 79-01B Reference 2.6
- o IE Bulletin No. 79-01B Supplement 2, dated Sept. 30, 1980 Reference 2.7

The regulatory requirements in the above mentioned documents are summarized in the following sections.

2.1.1 Accident Analysis Requirements

The post-accident radiation environment should be based on the most severe design basis accident (DBA) during or following which equipment must remain functional. This includes the consideration of the entire spectrum of Loss of Coolant Accident (LOCA) events which can lead to a degraded core condition. These accident conditions include:

a. LOCA events which completely depressurize the primary system.

b. LOCA events in which the primary system may not depressurize.

2.1.2 Source Term Assumptions

The radioactive source terms for the postulated accident conditions as described in Section 2.1.1 should be equivalent to the source terms recommended in Regulatory Guide 1.3 and 1.7 and Standard Review Plan 15.6.5. The source term assumptions consistent with current licensing requirements used for equipment qualification and access evaluations are summarized as follows:

1. The fission product fractions assumed to be released from the fuel rods during a LOCA are:

noble gases	100%
halogens	50%
remaining fission products	1%

These release fractions are not the sum of (double counting) the assumed liquid and gaseous releases. In effect, 25% of the equilibrium halogen activity is assumed to be mixed in the containment atmosphere and 50% is assumed to be mixed in the Reactor Coolant and recirculated liquids. The post-LOCA source contribution from liquid and gaseous sources are analized separately and the worst dose is tabulated for that evaluation rather than the sum of both doses. Thus, double counting of the fission product fractions is eliminated.

2. The above release is assumed to occur and be distributed instantaneously at the start of the accident.

3. Until depressurized, liquid in the Reactor Coolant System (RCS) and other systems which are not isolated from the core and which contain the reactor coolant at the start of the LOCA contain 100% noble gases, 50% halogens and 1% of the remaining fission products. These radioactive materials are mixed homogeneously in a volume no greater than the RCS liquid space.

4. Liquid in the suppression pool and any system not isolated from the core at the start of the LOCA, and containing only liquid from a depressurized source, is assumed to contain 50% halogens and 1% of the remaining fission products. These radioactive materials are diluted homogeneously in a volume no greater than the combined volumes of the suppression pool and the RCS liquid space.

5. The Primary Containment atmosphere and systems which are not isolated from the Primary Containment atmosphere at the start of the LOCA are assumed to contain at least 100% noble gases and 50% halogens. These radioactive materials are diluted homogeneously in a volume no greater than the combined volumes of the drywell and suppression pool air spaces.

6. Until the Reactor Vessel is depressurized, gases in the steam lines and any other

vapor-containing lines not isolated from the core at the start of LOCA are assumed to contain at least 100% noble gases and 25% halogens. These are diluted uniformly in a volume no greater than the RCS steam space and adjoining, unisolated steam lines.

2.1.3. Vital Area Access Requirements

As defined in NUREG-0737 (Ref. 2.5), a vital area is an area which will or may require occupancy to permit an operator to help in the mitigation of an accident or perform post-accident operations. The accident scenarios discussed in Section 2.1.1 and the source term assumptions in Section 2.1.2 are used for the evaluation of vital area access and occupancy. The total radiation exposure to personnel in vital areas should not be in excess of 5 rem whole body, or its equivalent, to any part of the body for the duration of the accident. For areas requiring continuous occupancy (e.g., the Control Room, Onsite Technical Support Center, etc.), the dose rate criteria limits the total radiation exposure to personnel to less than 15 mrem/hr (averaged over 30 days).

2.1.4 Systems Containing The Sources

Systems considered in the shielding review are those systems that could have the potential of containing a high level of radioactivity post-accident. For those systems connected directly to the Reactor Coolant System or to the Primary Containment atmosphere and not isolated at the start of the accident, the radioactivity is assumed to be instantaneously mixed within the unisolated parts of the system.

2.1.5. Safety-Related Equipment

The safety-related (1E/1M) equipment list contains all equipment necessary to mitigate the consequences of an accident, bring the plant to a safe shutdown condition, and provide long-term cooling capability. This list includes equipment located inside as well as outside the Primary Containment.

2.2 Shielding Evaluation Task Description

The Shielding Evaluation tasks which have been completed to date are as follows:

1. Review all accident scenarios and accident conditions that could result in a limiting radiation environment for all the pieces of safety-related equipment on the 1E/1M (Safety-Related) list that are located in the Reactor Building.

2. Identify systems and components that could potentially contain radioactive materials post-accident.

3. Generate source term assumptions based on regulatory requirements discussed in Section 2.1.

4. Calculate accident radiation service conditions for the safety-related equipment located inside the Reactor Building.

5. Calculate gamma dose rates at selected locations outside the Reactor Building due to radioactive sources inside the Reactor Building.

6. Identify vital areas and equipment outside the Reactor Building to evaluate the access to and occupancy of the vital areas

in accordance with the requirements listed in Section 2.1.

7. Conduct a Primary Containment analysis of LOCA events in which the RCS may not depressurize (or may repressurize) with a degraded core condition. The Primary Containment radiation environment was determined with the use of 100% noble gases, 50% halogens and 1% of the remaining fission products for the period of time during which the activity is isolated to the RCS.

2.3 Shielding Evaluation Item Deleted From Shielding Analysis Consideration

WNP-2 has addressed all the issues needed to comply with the NUREG-660 II.B.2 position except as follows: WNP-2 takes exception to the portion of the task that specifies that a review of "safety-related equipment which may be degraded by radiation during post-accident operation be provided for a Non-LOCA, High-Energy Line Break Source Term". The pipe break/missile analysis performed in Sections 3.5 and 3.6 of the FSAR addresses non-mechanistic pipe breaks inside and outside containment. These pipe breaks do not lead mechanistically to a radiation release due to fuel failures beyond those allowed in normal operation. Hence, the source term identified and applied outside containment is entirely hypothetical and would be a new design basis beyond the scope of current regulations.

2.4 Additional Review Items Required For Final Plant Shielding Report

The items which need to be performed to achieve full compliance with the regulatory requirements are summarized as follows:

1. The equipment considered for this study was limited to the initial 1E/1M (safety-related) list. Additional 1E/1M equipment subsequently identified will be evaluated with integrated radiation results presented in the Final Shielding Evaluation Report.

2. For this report no attempt was made to verify the completeness of the safety-related safety-related equipment list. The safety-related equipment list will contain all equipment required to "mitigate" the consequences of an accident, bring the plant to a safe shutdown condition, and provide long-term cooling capability". This effort is currently underway and will be addressed in the Final Shielding Evaluation Report.

3. Complete the calculation of radiation levels inside primary containment due to the analysis effort in the shadow shielding effect of Primary Containment hardware and the effect of first order iodine plateout.

3.0 ANALYTICAL METHODOLOGY

In order to develop the method used in the calculation of radiation doses, a review of all the postulated accident scenarios and system operations were performed. Source term assumptions were developed based on the results of accident analysis and system review, as well as the regulatory guidelines described in Section 2.1. The systems and components inside the Reactor Building that have the potential of becoming contaminated during or following the accident were identified.

The following subsections describe these activities in greater detail. Section 3.1 describes the accident scenario chosen for this analysis. Section 3.2 identifies all the contaminated systems. Section 3.3 describes the source term assumptions generated for each contaminated system. Section 3.4 identifies the time period considered for this study.

3.1 Accident Scenario

The accident analyses consistent with FSAR Chapter 15 for small and large break Loss of Coolant Accidents (LOCA's) were considered. The entire spectrum of LOCA conditions that could result in a degraded core configuration was reviewed and it was concluded that there is no single accident scenario that could result in a limiting radiation environment for all the safety-related equipment located in the Reactor Building. Therefore, the accident scenario chosen here is based on a non-mechanistic LOCA in which core damage is experienced at the beginning of the accident. Primary Containment isolation is assumed to be achieved prior to radioactivity transport.

A review of the post-accident operation of the 1E/1M (safety-related) systems was conducted. The result of this review indicated that the worst case accident for the steam supply system (highest source term) was the pressurized Reactor Coolant System (RCS). For the liquid systems (the ECCS, the RHR, and the RCIC system), as well as the Primary Containment Atmosphere and Primary Containment Atmosphere Control System (CACS), the worst case accident is the depressurized Reactor Coolant System with the post-LOCA core release fractions dispersed within the Primary Containment.

3.2 Contaminated Systems

In order to perform the radiation dose calculations, it was necessary to identify the systems which would or could contain highly radioactive materials during the post-accident period. Systems required to operate during the post-accident period are those:

- o Systems necessary to mitigate the consequences of a large or small break LOCA.
- o Portions of systems that are in communication with systems containing radioactive liquids or gases.
- o Defined by the NRC as being required; such as Gaseous Radwaste System. (See Section 3.2.3).

3.2.1 Systems Included for Primary Containment Analysis

The following systems are being considered:

- o High Pressure Core Spray (HPCS)
- o Low Pressure Core Spray (LPCS)
- o Residual Heat Removal (RHR)
- o Reactor Core Isolation Cooling (RCIC)
- o Floor Drains and Equipment Drains (FDR-EDR)
- o Reactor Water Cleanup (RWCU)
- o Main Steam (MS)
- o Reactor Recirculation (RRC)
- o Sample Lines (PSR)
- o Automatic Depressurization System (ADS)
- o Low Pressure Coolant Injection (LPCI) Function of the RHR system after depressurization

3.2.2 Systems Included for Secondary Containment Analysis

The following systems were considered:

- o Reactor Core Isolation Cooling (RCIC)
- o Residual Heat Removal (RHR)
- o Low Pressure Coolant Injection (LPCI)
- o Low Pressure Core Spray (LPCS)
- o High Pressure Core Spray (HPCS)

- o Containment Atmosphere Control (CAC, the Hydrogen Recombiners)
- o Main Steam (MS, Up to Second Isolation Valve)
- o Main Steam Line Isolation Valve - Leakage Control (MSIVLCS)
- o Primary Containment
- o Secondary Containment Atmosphere
- o Standby Gas Treatment (SGT)

3.2.3 Systems Excluded

All systems required to mitigate the consequences of an accident have been included. Of those systems recommended for consideration in regulatory documents, one system (Gaseous Radwaste) has been excluded.

The Gaseous Radwaste is isolated by the Primary Containment and Reactor Vessel Isolation Control System and will not receive contaminated gas unless operation is manually initiated. The WNP-2 operating and accident procedures do not take credit for nor anticipate using this system. Since WNP-2 philosophy is based on containment of the core releases within the Primary Containment, this system will not be required and was, therefore, excluded from consideration.

3.3 Source Term Assumptions

Regulatory requirements specify that source terms equivalent to those recommended in Regulatory Guides 1.3 and 1.7 and Standard Review Plan 15.6.5 be used in the LOCA accident analysis. Additional guidance is given

in NUREG-0588 (Ref. 2.2) and NUREG-0737 (Ref. 2.5) and is documented in Section 2.1. Source term assumptions were generated based on the review of the operation of the safety systems. Because a non-mechanistic LOCA scenario was chosen for this analysis, the worst contaminated situation for the fluid contained within each system was conservatively assumed. Tables 3.1, 3.2 and 3.3 list the assumptions involved in the distribution of fission products used in this analysis. These assumptions are consistent with the regulatory requirements discussed in Section 2.1.

A review of the operation of each of the systems discussed in Section 3.2 was also conducted. This review identified the source of contaminated fluid contained within each system post-accident. Using the source term assumptions discussed in Tables 3.1, 3.2 and 3.3, together with the results of this system review, the limiting source term (activity divided by dilution factor) was determined for each system. Table 3.4 is a summary of the system operations and source term assumptions developed for each contaminated system identified in Section 3.2.

3.4 Time Period Considered For Study

All systems were conservatively assumed to become contaminated at the start of the accident and remain contaminated until the integrated radiation dose reached it asymtotic value. It was noted that the integrated dose becomes nearly asymptotic to a constant value beyond about 6 months. Therefore, 6 months is the time period chosen for accident dose qualification in this report.

Table 3.1 Distribution of Fission Products in the Worst Post-LOCA Situation for Areas Inside Containment. Depressurized Reactor Coolant System.

	Primary Containment Air and Steam Space			Suppression Pool (1) and Reactor Coolant System Water Volume	
Fission Products	Fraction (2)	Dilution Volume (3)	Plateout	Fraction (2)	Dilution Volume (3)
Noble Gases	100%	Drywell Air Plus Suppression Pool Air	0%	0%	Suppression Pool Water and RCS Water Volume
Halogens	50% (4)		0-47.5%(5)	50%	
Particulates	0%		0%	1%	

(1) If equipment must be qualified in the suppression pool atmosphere, then a uniform distribution between drywell and suppression pool atmosphere will be assumed.

(2) Expressed in percentage (%) of total core inventory at End-of-Life conditions (1,000 days at 3481 mwt).

(3) Represents the total volume in which the fraction of core fission products is assumed to be homogeneously mixed.

(4) In calculating the radiation dose at a particular location, it is not necessary to assume that all source distribution assumptions are conservative simultaneously. Instead, a set of mutually compatible assumptions will be used which gives the maximum dose for the location being considered. The post-LOCA source contributors are used to calculate independant doses for each contributor. The worst dose is tabulated for that system rather than the sum of all contributors (i.e., 50% halogens airborne and 50% halogens in the water). Thus double counting of the fission product fractions is eliminated.

(5) First order iodine plateout occurs during the first 5-6 hours of the post-LOCA time frame when the elemental halogen concentration is reduced by a factor of 200. This methodology is in accordance with NUREG/CR-0009.

Table 3.2 Distribution of Fission Products in the Worst Post-LOCA Situation for Areas Inside Containment. Pressurized Reactor Coolant System. (1)

	Drywell Air Space (1)	Suppression-Pool Water Volume and Air Space (1)	Reactor Coolant System Water Volume (1)		Reactor Coolant System Steam Space (1)	
Fission Products	Fraction (2)	Fraction (2)	Fraction (2)	Dilution Volume (3)	Fraction (2)	Dilution Volume (3)
Noble Gases	0%	0%	100% *	(5) RCS Water Volume	100% *	Normal RCS Steam Space (6)
Halogens	0%	0%	50% (4)		25%	
Particulates	0%	0%	1%		0%	

(1) The reactor coolant system will remain pressurized for a short period of time (17 hours) and then will be depressurized.

(2) Expressed in percentage (%) of total core inventory at End-of-Life conditions (1,000 days at 3481 mwt).

(3) Represents the total volume in which the fraction of core fission prodcuts is assumed to be homogenously mixed.

(4) In calculating the radiation dose at a particular location, it is not necessary to assume that all source distribution assumptions are conservative simultaneously. Instead, a set of mutually compatible assumptions will be used which gives the maximum dose for the location being considered. The post-LOCA source contributors are used to calculate independant doses for each contributor. The worst dose is tabulated for that system rather than the sum of all contributors (i.e., 50% halogens airborne and 50% halogens in the water). Thus double counting of the fission product fractions is eliminated.

(5) The dilution volume is the RCS water volume plus the RWCU lines up to the isolation valves, RHR lines up to the isolation valves, and the RRC lines during the 17 hours of the pressurized RCS scenario.

(6) The dilution volume is the normal RCS steam space plus the MS lines up to the isolation valves during the 17 hours of the pressurized RCS scenario.

* The 100% of noble gases, present during the 17 hours of the pressurized RCS during a LOCA, are homogeneously mixed in the water and steam dilution volumes identified.

Table 3.3 Distribution of Fission Products

In The Worst Post-LOCA Situation

For Areas Outside Containment

	Primary Containment Air Space		Suppression Pool Water Volume		Reactor Coolant System Steam Space		Reactor Coolant System Water Volume	
Fission Products	Fraction (1)	Dilution Volume (2)	Fraction (1)	Dilution Volume(2)	Fraction (1)	Dilution Volume (2)	Fraction	Dilution
Noble Gases	100%	Drywell Air Plus Suppression Pool Air	0%	Suppression Pool Water Plus RCS Water	100%	Normal RCS Steam Space	100%	RCS Water Volume
Halogens	25% (4)		50% (5)		25%		50%	
Particulates	0%		1%		0%		1%	

(1) Expressed in percentage (%) of total core inventory at End-of-Life Conditions (1,000 days at 3481 MWt).

(2) Represents the total volume in which the fraction of core Fission Products is assumed to be homogeneously mixed.

(3) Based on pressurized Reactor Coolant System

(4) Half of the 50% of the halogens released from the core are assumed to plate-out instantaneously as allowed by NUREG-0588 Rev. 1. The plate-out dose was considered in the total calculation of radiation dose to equipment inside Primary Containment.

(5) In calculating the radiation dose at a particular location, it is not necessary to assume that all source distribution assumptions are conservative simultaneously. Instead, a set of mutually compatible assumptions will be used which gives the maximum dose for the location being considered. The post-LOCA source contributors are used to calculate independant doses for each contributor. The worst dose is tabulated for that system rather than the sum of all contributors (i.e., 25% halogens and 50% halogens). Thus double counting of the fission product fractions is eliminated.

Table 3.4 System Operations and Source Term Assumptions

System	Operation Post-Accident	Contaminated Source	Source Term Assumptions
HPCS	Suction from Condensate Storage Tank and/or Suppression Pool and discharge to the Reactor Vessel.	Suppression Pool	(1)
LPCS	Suction from Suppression Pool and discharge to the Reactor Vessel.	Suppression Pool	(1)
LPCI	Suction from Suppression Pool and discharge to the core.	Suppression Pool.	(1)
(6) RCIC Steam System	Steam bled-off from Reactor Steam Space is used to drive the RCIC turbine, and exhausts into the Suppression Pool.	RCS Steam Space	(2)
RCIC Liquid System	Suction from Condensate Storage Tank or Suppression Pool and discharge to the Reactor Vessel.	Suppression Pool	(1)
RHR System	(1) Shutdown Cooling Mode - suction from reactor recirculation system suction line and discharge into the reactor recirculation discharge line.	RCS Liquid Space	Note A
	(2) Alternate Shutdown Cooling Mode - suction from Suppression Pool and discharge to core recirculates and cools the water in the Suppression Pool.	Suppression Pool	(1)
	(3) Containment Spray Cooling Mode-suction from Suppression Pool and discharge into the Drywell and Suppression Pool.	Suppression Pool	(1)

Table 3.4 cont'd

System	Operation Post-Accident	Contaminated Source	Source Term Assumptions
(6) RHR System (cont'd)	4) Reactor Steam Condensing Mode - steam bled off from reactor vessel, condensed through the RHR heat exchanger and directed to the RCIC pump suction or Suppression Pool.	RCS Liquid Space	Note A
Main Steam Supply	Stagnant steam from the reactor vessel terminates at the second MSIV.	RCS Steam Space	(2)
MSIVLCS	Steam bled off from main steam line, diluted and discharged into the SGT filter room.	RCS Steam Space	(2) Note B
SGT Filters	Process the halogens from primary containment leakage and MSIVLCS.	Primary Containment Atmosphere	(3)
CAC	Process the Primary Containment Atmosphere. (Hydrogen Recombination)	Primary Containment Atmosphere	(4)
Primary Containment	Primary Containment is isolated Post-Accident.	Primary Containment Atmosphere	(4)
Suppression Pool	The primary function of the suppression pool is to contain and condense the blowdown from the RCS post-accident.	Suppression Pool Liquid	(1)
Secondary Containment	The primary function of the Secondary Containment is to contain all the leakage from the Primary Containment Post-Accident.	Primary Containment Atmosphere	(5)

Table 3.4 cont'd

System	Operation Post-Accident	Contaminated Space	Source Term Assumptions
Sample Lines	Actuated to obtain Primary Containment Atmosphere samples per NUREG-0737	Primary Containment Atmosphere	(2)
Sample Lines	Actuated to obtain RCS liquid samples per NUREG-0737	RCS Liquid Space	(1)
Reactor Water Cleanup	Reactor water cleanup system isolated during post-LOCA. Liquid up to the second isolation valve is considered contaminated.	Reactor Coolant System Liquid	(1)
Reactor Recirculation	Suction from reactor recirculation system suction line and discharge into the reactor recirculation discharge line.	Reactor Recirculation Liquid	(1)
Floor Drains and Equipment Drains	Liquid from ruptured pipes or leaky seals discharged into the suppression pool.	RCS Liquid	(1)
Automatic Depressurization System	Automatic or manual depressurization of the reactor vessel by blowdown of the RCS into the suppression pool.	RCS Steam	(2)
Automatic Depressurization System	Alternate Shutdown cooling mode with reflood of reactor vessel and discharge into suppression pool.	Suppression Pool	(1)

Table 3.4 cont'd

Source Term Assumptions

(1) 50% Halogens and 1% Solid Fission Products diluted with Suppression Pool water plus RCS water.

(2) 100% Noble gases and 25% Halogens diluted with the RCS steam space.

(3) 25% Halogens leaked from the Primary Containment is assumed to be deposited in the SGT Filters at the rate of 0.73% per day. See section 5.5.4.1 for justification. 100% Noble Gases pass through also but are not absorbed.

(4) 100% Noble gases and 50% Halogens diluted with the Primary Containment air space. 47.5% Halogen plate-out inside Primary Containment was considered.

(5) Assumptions involved in the calculation of source terms for Secondary Containment Atmosphere is discussed in detail in Section 5.5.1.

(6) Based on a pressurized Reactor Coolant System.

NOTES: A According to accident mitigation procedures, this mode of operation is not used after a degraded core condition is identified.

B For the portion of system after the distribution header, credit is taken for dilution by clean air. See section 5.5.4.1 for justification.

4.0 ACCESS AND OCCUPANCY OF VITAL AREAS

NUREG-0578 initiated the requirement for a design review to identify the location of vital areas in which personnel occupancy may be unduly limited by the radiation fields during post-accident operations. It required that each licensee provide adequate access to vital areas by design changes, increased permanent or temporary shielding, or post-accident procedural controls. NUREG-0737 further makes the point that the purpose of this design review is to determine what actions can be taken over the short-term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident.

This shielding evaluation includes the calculation of gamma dose rates at selected locations outside the Reactor Building due to radioactive sources inside. The radioactive source terms obtained from ORIGEN computer calculations coupled with recommendations from Regulatory Guide 1.109 were the basis for the assumptions used in evaluating vital areas and access routes outside the Reactor Building.

4.1 Dose Rates Outside the Reactor Building

An analysis was conducted to determine the dose rates at selected locations outside the Reactor Building for personnel access purposes. The radiation level in the various areas outside the Reactor Building is defined by the following three radioactive sources:

1. Direct gamma ray dose from radioactive piping located inside the Reactor Building and attenuated through the walls of the Reactor Building.

2. Gamma shine dose from airborne activity inside the Reactor Building.

3. Gamma dose from airborne activity outside the Reactor Building.

Radiation levels outside the Reactor Building were determined by the zone dose method as discussed in Section 5.5. Representative zones were chosen at selected locations outside the Reactor Building such as ground level outside the railroad bay, sampling room, etc. The worst point in a zone was chosen to be the point directly outside the Reactor Building wall, at a height of six feet above floor elevation, at a lateral point determined by inspection to receive the highest dose along that wall.

The zones outside the Reactor Building are indicated by the letters Y and Z in the various elevations indicated in the radiation zone maps (shown in Figures 6.1 through 6.8). The result of the dose calculations are shown in Figures 6.19 through 6.24.

Appendix H presents the methodology used to calculate the radiation doses for the various vital areas and access routes.

4.2 Vital Areas and Access Routes Outside the Reactor Building

Radiation calculated for the access routes were based on the assumption that no individual would be in an access route longer than 30 minutes for the first 8 hours after the postulated LOCA before reaching the vital area of interest.

The assumption was also made that no individual would occupy an infrequent occupied vital area longer than 30 minutes for the first 8 hours after the postulated LOCA.

All integrated radiation doses calculated for the access routes were less than the guidelines presented in NUREG-0737.

All vital areas evaluated had radiation doses less than the guidelines presented in NUREG-0737 except for the security guardhouse and auxiliary security center. Security personnel will be relocated to the Technical Support Center per the WNP-2 Emergency Preparedness Plan if radiation levels approach those radiation guidelines presented in NUREG-0737 during the post-LOCA operations.

5.0 METHODS

The radiation dose assessment of safety-related equipment inside containment is being done by calculating the radiation dose to each component from all applicable radiation sources (contaminated liquid, steam, and airborne sources).

The Secondary Containment radiation dose assessment portion of the Shielding Evaluation was initiated by dividing the Reactor Building into radiation zones. Because of the large number of radioactive piping and safety-related equipment in the building, the division of the various regions of the Secondary Containment into radiation zones permits a precise, detailed calculation of the total integrated dose at the "worst target" location. The methods for performing the calculations are discussed in detail in the following sections.

The radiation dose assessment of safety-related equipment outside of the Reactor Building was done by calculating the radiation dose of each vital area where safety-related equipment was located. The assumptions and methodology used to perform these calculations are discussed in detail in the following sections and Appendix H.

5.1 The Use of Computer Codes

The two computer codes used in the Primary Containment shielding evaluation are ORIGEN 2 and QAD-CG. Descriptions of the two codes are found in References 5.1, 5.2, 5.3 and 5.10. ORIGEN 2 computes the radioactive source terms (inside containment) used by QAD-CG to calculate the radiation doses from piping and various pieces of equipment.

The two computer codes used in the Secondary Containment radiation shielding review were ORIGEN and QAD-P5A. Descriptions of the codes are found in References 5.4 and 5.3. ORIGEN computes the radioactive source terms used by QAD-P5A to compute the radiation from piping and other source configurations to pieces of equipment.

ORIGEN and ORIGEN 2 are a versatile fission product source term codes which solves the equations of radioactive growth and decay for large numbers of isotopes. ORIGEN 2 is being used to calculate the radioactivity of fission products and fuel materials that were assumed to be released from the reactor core during the postulated LOCA to become the Primary Containment source terms for the dose rate calculations.

5.2 Source Term Development For Primary Containment

The radiation level at any given location inside the Primary Containment of WNP-2 following the postulated LOCA such as that described in Section 3.1 is determined from the following major source contributors:

1. Gamma ray dose from airborne radioactive sources suspended in the drywell and wetwell inside Primary Containment (Airborne Gamma Dose).

2. Gamma ray dose from piping and/or equipment containing contaminated fluids which are recirculated inside Primary Containment (Direct Gamma Dose).

3. Beta ray dose emitted by airborne radioactive sources suspended in the

drywell and wetwell inside Primary Containment (Airborne Beta Dose).

4. Beta ray dose emitted by airborne radioactive sources inside Secondary Containment (Airborne Beta Dose).

The initial phase of this analysis was concerned with the determination of radioactive source terms for the liquids and gases inside Primary Containment. The ORIGEN 2 Computer Code was used for this calculation. The fission products at the end of fuel life (maximum burnup at power level of 3481 MWt for 1,000 days) was assumed to be available for release immediately following the accident. The concentrations of noble gases, halogens, and other fission products released to the gaseous and liquid sources were computed.

5.3 Source Term Development For Secondary Containment

The radiation level at a given location inside the Secondary Containment of WNP-2 following an accident such as that described in Section 3.1 is defined by the following major source contributors:

1. Gamma ray dose from airborne radioactive source inside Secondary Containment (Airborne Gamma Dose).

2. Gamma ray dose from radioactive sources suspended in the drywell and the wetwell inside Primary Containment (Containment Shine Dose).

3. Gamma ray dose from piping and/or equipment containing contaminated fluids which are recirculated outside Primary Containment (Direct Gamma Dose).

4. Beta ray dose emitted by airborne radioactive sources inside Secondary Containment (Airborne Beta Dose).

The initial phase of this analysis was concerned with the definition of radioactive source terms for the liquid and gas containing systems. The ORIGEN Computer Code was used for this calculation. The fission products at the end of fuel life (maximum burnup at power level of 3481 MWt for 1000 days) was assumed to be available for release immediately following the accident. The concentrations of noble gases, halogens, and other fission products released to the gaseous and liquid sources were computed. Subsequent fission product depletion and daughter product generation were then calculated for twenty time periods, covering a total period of one year. A detailed description of the method of analysis, including the assumptions used, as well as results of the source terms, is found in Appendix B and Reference 5.5.

5.3.1 Source Term Development for 1E/1M Equipment Outside the Reactor Building

The radiation level at any given location outside the Reactor Building of WNP-2 following the postulated LOCA as described in Section 3.1 is determined from the following major source contributors:

1. Direct gamma dose from radioactive piping located inside the Reactor Building and attenuated through the walls of the Reactor Building.

2. Gamma shine dose from airborne activity inside the Reactor Building.

3. Gamma ray dose from airborne activity outside the Reactor Building.

A detailed description of the method of analysis, including the assumptions used, as well as results of the source terms is found in Appendix H.

5.4 Parametric Studies for Direct Piping Dose in Secondary Containment

The purpose of the parametric study was to identify the parameters which have a significant effect on the radiation dose rates inside Secondary Containment. The computer code QAD-P5A was used to develop a correlation scheme for the significant parameters such that a simplified procedure for calculating radiation dose rates for complex source and receptor geometries can be developed. The dose rate at a target distance of 8 ft radially outwards from the centerline of an 8-inch schedule 40 pipe, infinitely long (standard pipe) was first calculated and defined as the standard dose rate. The results of this parametric study were then correlated as a set of correction factors to the standard dose rate. A simplified procedure was developed to calculate the dose rates and cumulate doses for complicated source-target configurations by using these correction factors. The development of these correction factors and the result of the parametric study inside Secondary Containment is discussed in detail in Appendix B.

5.5 Dose Rate and Cumulative Dose Calculation Procedure

The results of the source term calculations and those of the parametric study were used to generate and cumulate doses for complicated source target configurations inside Secondary Containment. The following steps were taken to define the radiation service conditions for the pieces of safety-related equipment:

1. Based on the accident scenarios, contaminated systems, and assumptions defined in Section 3.0, the radioactive source terms for liquid-containing and gas-containing systems were developed.

2. Radiation zones were selected and the radiation zone boundaries were carefully defined based on shield wall locations, contaminated piping locations, and locations of safety-related (1E/1M) equipment.

3. The radiation environment in each Secondary Containment zone (zone dose) was calculated (see Appendix B for detailed procedure). A zone dose is the radiation dose (gamma) that bounds the magnitude of dose received by all the pieces of safety-related (1E/1M) equipment located within that zone.

4. The zone dose as calculated in step 3 was used, as a first cut, to qualify all the pieces of safety-related (1E/1M) equipment located within that zone.

5. For the pieces of safety-related (1E/1M) equipment that could not be qualified for

the conservative radiation environment calculated in step 3, the integrated dose for that piece of equipment was redefined based on a more realistic and refined approach.

5.5.1 Calculation of Airborne Gamma Doses Inside Secondary Containment

The time-dependent post-LOCA activity levels as calculated by the ORIGEN computer code were used as input for the calculation of the airborne gamma dose rates and integrated doses inside the cubicles in the Secondary Containment. The assumptions used in this analysis are as follows:

1. Activity that leaks into the Secondary Containment is homogenously mixed with the Secondary Containment atmosphere prior to its removal from the atmosphere through the Standby Gas Treatment System (SGTS).

2. The minimum SGTS flowrate of 1100 SCFM was assumed to be the flowrate of the effluent air.

3. Air that leaks out of the Primary Containment flows directly and totally into the Secondary Containment. Bypass leakage was not considered.

4. Geometric factors were used to convert the semi-infinite cloud gamma dose to a finite gamma dose.

5. Primary Containment leakage rate of 0.5% volume/day was considered.

Justifications of the above assumptions are stated in Appendix B. The equations that

were used for the gamma dose calculations are described in Appendix B.

5.5.2 Methodology of Beta Dose Analysis

The source volume used for the beta dose analysis in Secondary Containment is a sphere surrounded by a shell of sufficient thickness to stop all outside beta particles from entering the source volume. This spherical source volume is conservative for any generalized source volume shape (the dose at the center of the sphere is higher than the dose at any point of any generalized source shape of equal total volume). The assumptions used in this analysis are as follows:

1. Atmosphere inside the equipment casing is identical to the atmosphere in the Reactor Building.

2. Doses will be calculated using an air dose as suggested by NUREG-0588, Revision 1.

3. The beta source term used was 100% of core noble gases and 25% of core halogens.

4. Daughter products of the airborne noble gases and halogens are included in the calculation of the airborne dose.

5. The primary to secondary leak rate is 0.5% of primary containment volume per day.

6. The SGT system operates at the minimum flow of 1100 scfm.

7. Primary to secondary leakage is homogeneously mixed in the secondary containment atmosphere.

8. No halogen plateout after release is assumed.

9. A spherical volume and equipment casing will be used.

The equations used for the beta dose calculations are described in Appendix E.

The beta dose to equipment is dependant on the internal volume size of the piece of equipment. The beta dose is determined through the use of an energy dependant geometry factor and a ratio of the internal equipment volume to an infinite cloud. The results of these factors are shown in Figure 5.1. Thus, the beta dose contribution to equipment can be determined from Figure 5.1 once the internal air volume of a piece of equipment is known.

5.5.3 Procedure For the Calculation of Radiation Zone Dose in Secondary Containment

As discussed previously, the gamma radiation level at a given location inside the Secondary Containment of WNP-2 following a Loss of Coolant Accident is determined for four types of radioactive source distributions:

- o Fission products suspended in the atmosphere of the Secondary Containment (Airborne Gamma Dose)
- o Gamma irradiation from the Primary Containment (Shine Dose)

- o Direct gamma irradiation from the radioactive fluid contained inside recirculating pipes (Direct Dose)
- o Airborne radioactive sources inside Secondary Containment (Airborne Beta Dose)

The dose contributed by each of these sources is determined by the location of the equipment, the time-dependent distribution of the source and the effects of shielding.

A step-by-step procedure for calculating radioactive zone doses is shown in Appendix C. The methods presented in that procedure make it possible to calculate the worst case gamma dose from the above-mentioned source contributors inside radiation zones of the Secondary Containment. In general, this procedure for determining zone doses consists of a correction factor method of calculating direct dose rates.

As discussed in Appendix B, the correction factor method for calculating dose rates provides a convenient and fairly precise way of determining direct dose rates due to generic pipe segments. For radioactive fluid contained within components of geometry other than generic pipe segments, such as Residual Heat Removal (RHR) Heat Exchangers, Standby Gas Treatment System (SGTS) filters, Hydrogen Recombiners, etc. special QAD-P5A computer modelling was performed to calculate the gamma dose contribution due to those systems. A brief description of the guidelines used in

modelling special components is found in Appendix B.

5.5.4 Calculation of Radiation Doses Due to Special Systems and Components Inside Secondary Containment

As discussed in Appendices B and C, the correction factor method for calculating gamma dose rates and integrated doses is involved with the application of the dose correction factors (pipe diameter, pipe length and radial distance correction factors) to a standard dose rate curve. A standard dose is defined as the gamma radiation measured at a target distance of 8 feet and emitted by radioactive sources contained within the suppression pool liquid and recirculated within infinitely long 8-inch schedule 40 piping. The systems that contain such radioactive fluids are the Reactor Coolant system, High Pressure Core Spray, Low Pressure Core Spray, and Residual Heat Removal systems. Other systems which contain fluids of different source terms and dilutions are considered special sources. The systems that need to be considered for special sources are:

- o SGTS filters
- o CAC system
- o Main Steam System
- o Main Steam Isolation Valve Leakage Control System

5.5.4.1 Source Term Assumptions In Secondary Containment

The assumptions for the calculations of source terms inside Secondary Containment for special source systems are listed as follows:

SGTS Filters

1. The SGTS Filters will be loaded by halogens at the rate of 0.73% Primary Containment free volume per day. This consists of 0.5% per day of Primary Containment leakage and 0.23% per day of leakage due to the MSIVLC system. No holdup of this activity in the Secondary Containment is assumed.

2. 25% of the total core halogen inventory is assumed to be released in the drywell free volume. This halogen fraction is assumed to be composed of 91% elemental, 4% organic and 5% particulate halogens.

3. The particulate halogens are assumed to be homogeneously distributed within the prefilters and the particulate filters, while the elemental and organic halogens are assumed to be homogeneously distributed within the charcoal filters.

Assumption 1 is consistent with the assumptions used in the Accident Analysis (Ref. 5.6, and Section 3.1).

Assumption 2 is the NRC recommended assumption for the distribution of halogen inventory (Ref. 5.7).

Assumption 3 is necessary because the time dependent distribution of activity within a filter is unknown. The homogeneous assumption, therefore, is considered appropriate and conservative for zone dose assessment.

CAC System

The function of the CAC system is to process the Primary Containment atmosphere to remove hydrogen after a LOCA accident. Therefore this system is assumed to be filled with gaseous source containing 25% halogens and 100% noble gases diluted with the Drywell free volume.

Main Steam System

The main steam lines are located inside and outside the Primary Containment; they include the Main Steam lines in the steam tunnel and the RCIC turbine supply and exhaust lines. The radioactive source term for this system is assumed to be composed of 100% noble gases and 25% halogens, distributed throughout the Reactor Coolant System (RCS) steam space.

Main Steam Isolation Valve Leakage Control System

The MSIVLC system of WNP-2 is a vacuum-type system which collects leakage between and downstream of the closed isolation valves and releases it to the atmosphere through the SGT system. Leakage through the valve stems (maximum leakage of 11.5 SCFH as described in Ref. 5.8) is directed to a distribution header or low pressure manifold where clean air is brought in to dilute the contaminated steam before exhausting to the SGTS filter unit at a rated flow rate of 50 SCFM. Thus the source term in the portion of piping system before the distribution header is conservatively assumed to be the same as that of the Main Steam system. For the portion of

the piping after the header, credit is taken for the dilution by the clean air. This assumption is consistent with that recommended in Reference 5.9.

5.5.4.2 Secondary Containment Analysis Method

The correction factor method is used for the calculation of the direct dose contribution due to the piping systems described in Section 5.5.4.1, with the exception of the SGTS filter system. A description of the analysis of the SGTS filter is documented in Appendix D. Generic piping dose rate and integrated dose (dose at a target distance of 8 feet away from the centerline of an infinitely long 8 inch schedule 40 pipe) for each system were developed using the source term assumptions discussed in section 5.5.4.1 and are shown in Appendix B. Parametric studies were also performed to investigate the variation of dose rates due to pipe diameter, pipe length and target distance for pipe segments containing gaseous source terms. The gaseous source term correction factors derived as a result of this parametric study (described in Appendix B), together with the generic dose rate curves generated for each system were used to calculate the direct gamma dose contribution on a target.

5.5.4.3 Calculation of Radiation Doses Inside Secondary Containment on Generic Mechanical Equipment

Table 5.1 is a sample list of generic mechanical equipment that are on the safety-related equipment list. For conservatism, the direct dose on the pieces of generic mechanical equipment is assumed to be the fluid contact dose. Figure 5.2 is an illustration of the point where the direct dose is calculated on a piping segment.

The Secondary Containment source term assumptions developed in section 5.5.4.1 are used for the calculation of radioactive source terms for different systems, and the fluid contact dose was calculated using QAD-P5A by following the guidelines set forth in Appendix C. Figures 5.3 through 5.5 are 6 month integrated fluid contact doses versus pipe diameter.

These curves are intended to give conservative, upper-bound direct gamma dose estimates for the qualification of the pieces of generic mechanical equipment and components in the various systems.

In order to use these curves to calculate the direct doses on generic mechanical equipment, the following steps should be taken:

1. Identify the system on which the equipment or component is located.

2. Identify the diameter of the contaminated pipe on which the equipment is located.

3. The six-month integrated dose for that piece of equipment or component can be determined by reading the appropriate curve.

Table 5.1

GENERIC MECHANICAL EQUIPMENT
Valve Packing
Lubricants
Seals
Expansion Joints
Pressure Relief Valve
Flow Element
Rupture Disc
Gasket Material
Conductivity Element
Valve
Strainers
Steam Traps
Filters (Piping)
Temperature Elements
Tanks
Moisture Separators
Evaporator
Heat Exchanger
Air Washer (Scrubber)
Pumps

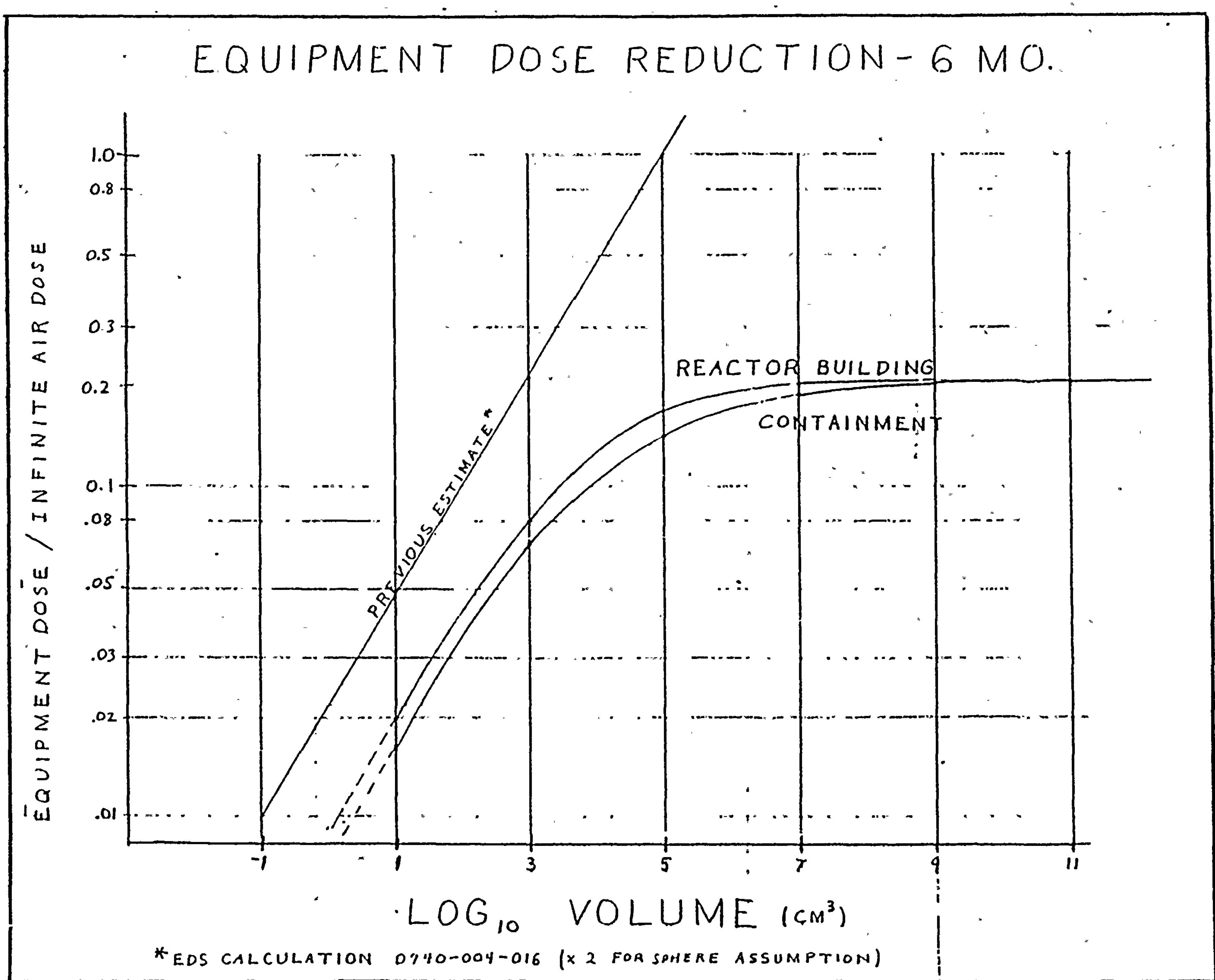
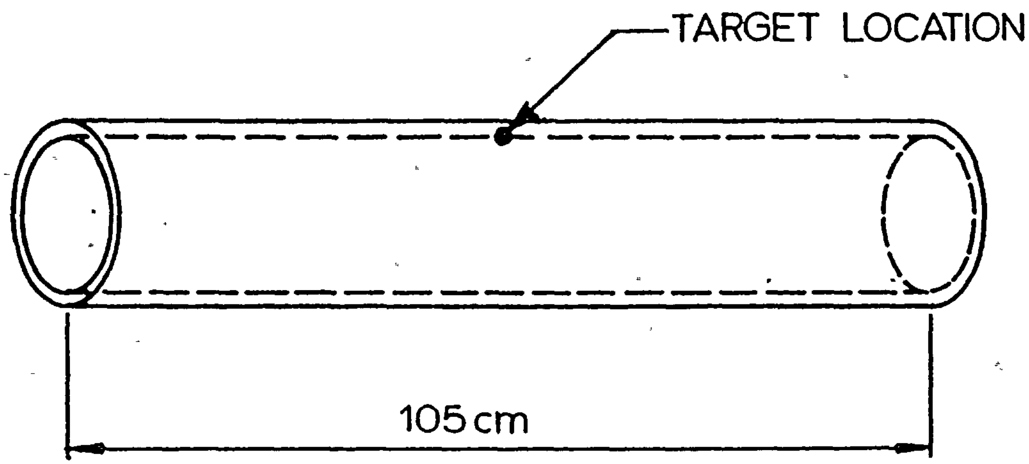

EQUIPMENT DOSE REDUCTION - 6 MO.
EQUIPMENT DOSE / INFINITE AIR DOSE
1.0
0.8
0.5
0.3
0.2
0.1
.08
.05
.03
.02
.01
PREVIOUS ESTIMATE*
REACTOR BUILDING
CONTAINMENT
-1
1
3
5
7
9
11
LOG10 VOLUME (CM³)
*EDS CALCULATION 0740-004-016 (x 2 FOR SPHERE ASSUMPTION)

Figure 5.2

DOSE MODEL
LIQUID SOURCE



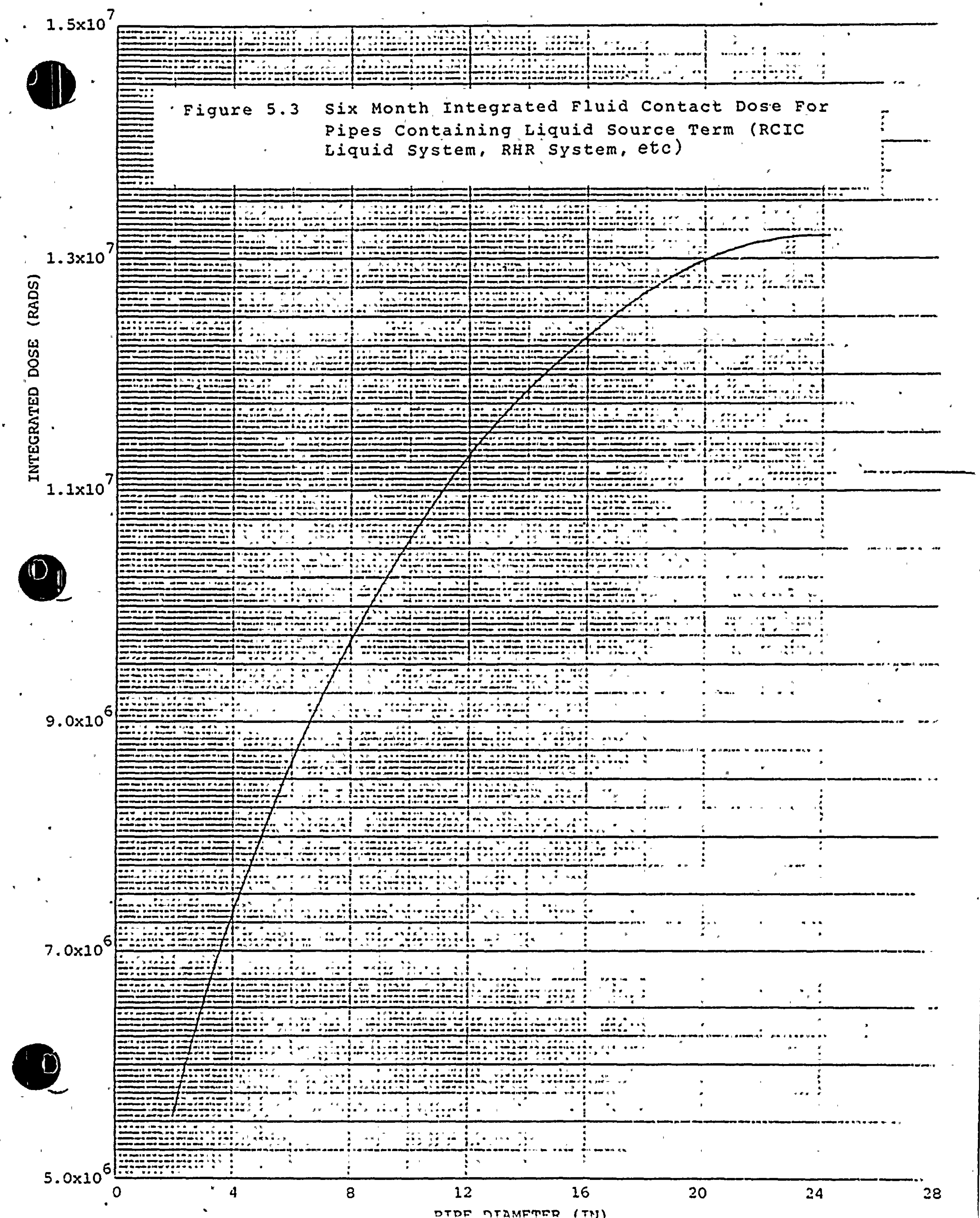


Figure 5.3 Six Month Integrated Fluid Contact Dose For Pipes Containing Liquid Source Term (RCIC Liquid System, RHR System, etc)
INTEGRATED DOSE (RADS)
1.5x10^7
1.3x10^7
1.1x10^7
9.0x10^6
7.0x10^6
5.0x10^6
0
4
8
12
16
20
24
28
PIPE DIAMETER (IN)

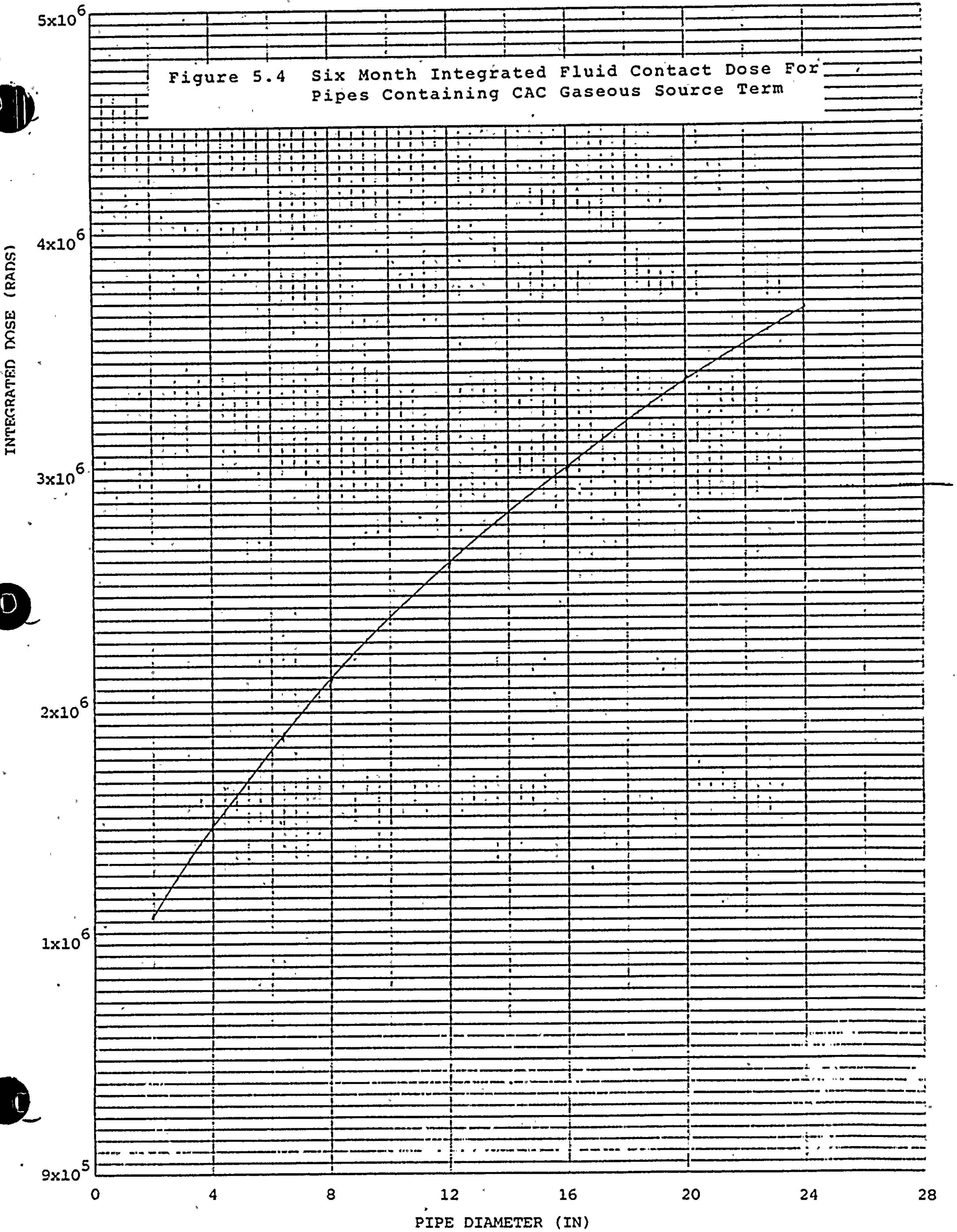


Figure 5.4 Six Month Integrated Fluid Contact Dose For Pipes Containing CAC Gaseous Source Term
INTEGRATED DOSE (RADS)
5x10^6
4x10^6
3x10^6
2x10^6
1x10^6
9x10^5
0
4
8
12
16
20
24
28
PIPE DIAMETER (IN)

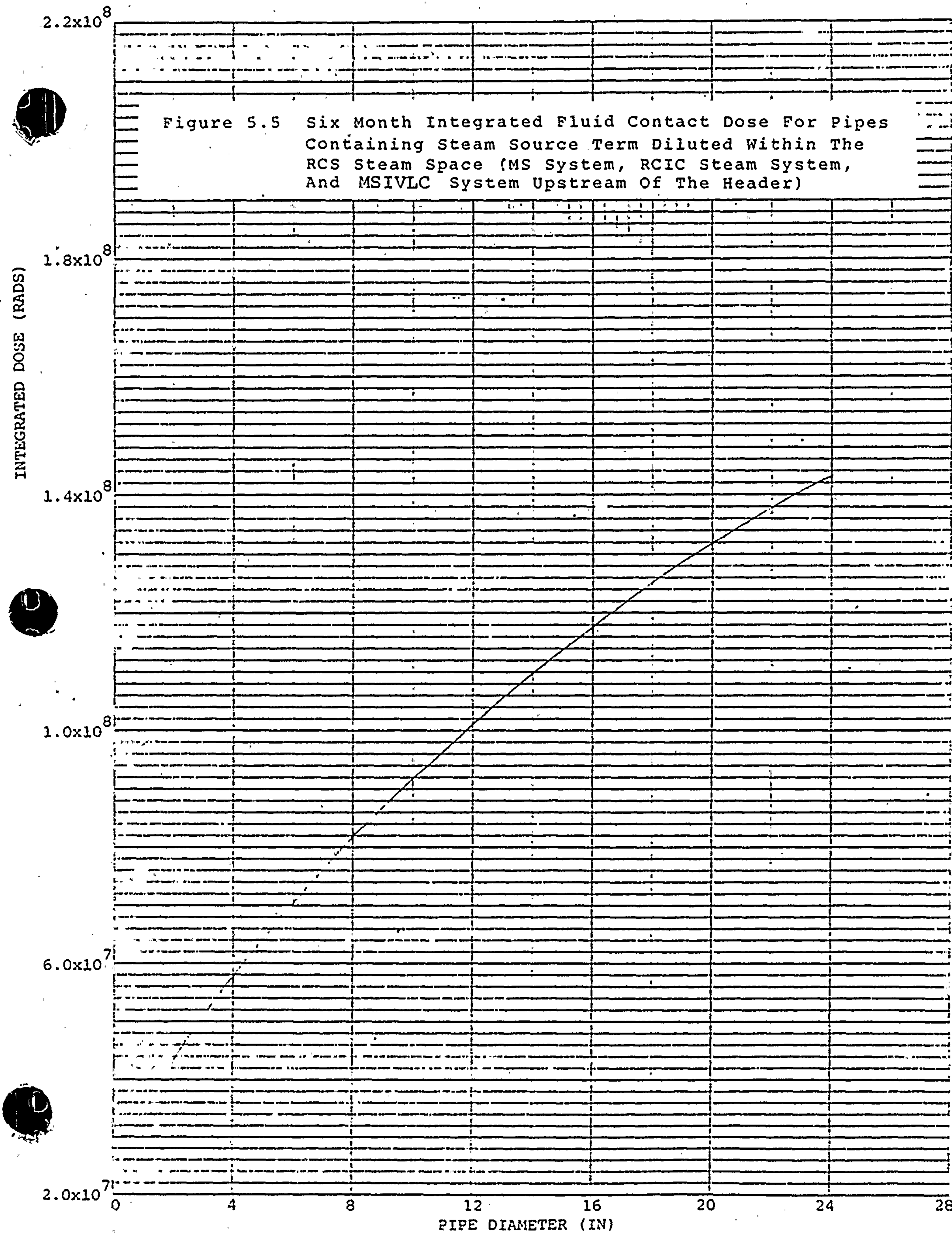


Figure 5.5 Six Month Integrated Fluid Contact Dose For Pipes Containing Steam Source Term Diluted Within The RCS Steam Space (MS System, RCIC Steam System, And MSIVLC System Upstream Of The Header)
INTEGRATED DOSE (RADS)
2.2x10^8
1.8x10^8
1.4x10^8
1.0x10^8
6.0x10^7
2.0x10^7
0
4
8
12
16
20
24
28
PIPE DIAMETER (IN)

6.0 RESULTS

All LOCA scenarios and accident conditions that could result in a limiting radiation environment for all the safety-related equipment in the initial 1E/1M list were reviewed and analyzed accordingly. Additional safety-related equipment subsequently identified will be evaluated with results presented in the Final Shielding Evaluation Report.

For this report, no attempt was made to verify the completeness of the safety-related equipment list. The safety-related equipment list will contain all equipment required to "mitigate the consequences of an accident, bring the plant to a safe shutdown condition, and provide long-term cooling capability." This effort is currently underway and will be addressed in the Final Shielding Evaluation Report.

Systems that could potentially contain radioactive material during and following the accident have been identified as listed in Section 3.2.1 and 3.2.2.

The accident radiation doses indicated in Table 6.3 and Figures 6.1 through 6.8, generated as a result of this analysis, are intended solely for the purpose of the qualification of safety-related equipment.

6.1 Primary Containment Radiation Results

Table 6.1 lists the 1E/1M (safety-related) equipment identified inside Primary Containment. The integrated direct gamma dose (40 yrs and 6 month LOCA -direct gamma and airborne gamma) is currently being evaluated for safety-related components

inside Primary Containment. The 40 year integrated gamma doses due to normal operation are taken from Reference 6.1. The direct gamma dose contribution inside Primary Containment is currently being evaluated.

Airborne beta dose inside Primary Containment was determined and is described in detail in Appendix G. The total beta dose contribution, due to Primary Containment atmosphere was calculated to be 3.4×10^9 Rads. The beta dose contribution inside Primary Containment due to plate-out of 25% halogens corresponds to a total dose of 2.3×10^5 Rads. Thus, the total beta dose contributions reported above should be used in conjunction with total integrated gamma radiation results to be presented in Table 6.1 for equipment qualification purposes. The radiation levels for Table 6.1 will be determined when the shadow shielding analysis and consideration of drywell plateout activities inside Primary Containment have been completed. These additional analyses efforts will be reported in the Final Shielding Evaluation Report.

6.2 Secondary Containment Radiation Results

Table 6.2 lists the 1E/1M (safety-related) equipment radiation zones and elevations inside Secondary Containment. Figures 6.1 through 6.8 present the radiation zone maps the Secondary Containment, and the "Worst Target Integrated Direct Gamma Dose" associated with all the 1E/1M equipment listed for each zone. The integrated direct gamma dose (40 yrs and 6 month LOCA - direct gamma, gamma shine, and airborne gamma) is evaluated for the worst target in each zone and is used for qualification (in conjunction with airborne beta doses) of all the other 1E/1M equipment in that zone. The 40 year integrated gamma doses (Figures 6.9 through 6.18) are taken from References 6.1 and 6.2. The gamma shine dose contribution outside Primary Containment due to sources inside the Primary Containment was investigated. No safety-related equipment was located in the direct shine path through the penetrations and the shine dose was much lower than the zone dose (less than 5% in all cases). Therefore, it was concluded that the Primary Containment shine dose contribution is negligible. For the evaluated equipment, this "negligible shine dose" was less than 100 rads for the 6 month LOCA integrated dose.

Airborne beta doses outside containment were evaluated per the methodology described in Section 5.5.2. The resultant curve produced from the beta analysis inside Secondary Contanment is shown in Section 5 (Figure 5.1) This curve will be used to determine the beta dose for equipment qualification purposes to a particular piece of equipment inside Secondary Containment once the internal air volume of the equipment is known.

6.3 Radiation Results Outside the Reactor Building

Figures 6.19 through 6.24 present the radiation zone maps for 1E/1M (safety-related) equipment located outside the Reactor Building. The doses indicated on each Figure are also the six month LOCA integrated gamma doses to be used for 1E/1M (safety-related) equipment qualification purposes. Table 6.3 also presents a summary of the six month LOCA integrated gamma doses on all 1E/1M equipment located outside the Reactor Building.

Radiation levels of vital areas and access routes were determined at selected locations outside the Reactor Building due to radioactive sources inside the Reactor Building and release of radiation activity from the Reactor Building elevated vent. The vital areas and access routes analysed are consistent with those discussed in NUREG-0737, Item II.B.2. The radiation levels determined for the vital areas and access routes identified in Figures 6.19 through 6.24 are summarized in Table 6.4. All of the vital areas and access routes have radiation levels less than the guidelines presented in NUREG-0737 except for the security guardhouse and the auxiliary security center.

Security personnel will be relocated to the Technical Support Center per the WNP-2 Emergency Preparedness Plan if guardhouse radiation levels approach those radiation guidelines presented in NUREG-0737 during the post-LOCA situation.

The analysis completed for vital areas and access routes assumed there would be no

access to equipment or areas located within the Reactor Building during the post-LOCA scenario.

Table 6.1

WNP-2 1E/1M Primary Containment
Equipment List of Total
Integrated Dose (40 Yr and LOCA)

Equipment Number	Equipment Elevation	Radiation Level 40 yr + Direct Gamma + Airborne Gamma (R)
MS-V-1	574'-3"	
MS-V-2	574'-3"	
MS-V-5	578'-5/8"	
MS-V-16	502'-2 13/16"	
MS-V-22A	505'-11 11/16"	
MS-V-22B	505'-11 1/2"	
MS-V-22C	505'-11 1/2"	
MS-V-22D	505'-11 11/16"	
RHR-V-9	509'-7"	
RHR-V-50A	509'-1 5/8"	
RHR-V-50B	509'-1 5/8"	
RHR-V-123A	509'-1 5/8"	
RHR-V-123B	509'-1 5/8"	to be provided at a later date
RRC-P-1A	507'-0"	
RRC-P-1B	506'-8"	
RRC-V-19	509'-10"	
RRC-V-23A	502'-6"	
RRC-V-23B	502'-6"	
RRC-V-60A	507'-0"	
RRC-V-60B	506'-8"	
RRC-V-67A	507'-0"	
RRC-V-67B	506'-8"	
RRC-V-85A	Drywell	
RCIC-V-63	551'-4"	
RCIC-V-76	552'-4"	
RWCU-V-1	540'-0"	
RWCU-V-100	500'-3"	
RWCU-V-101	514'-0"	
RWCU-V-102	500'-3"	
RWCU-V-106	500'-3"	
RCC-V-40	514'-0"	
RFW-LMS-10A	512'-0"	
RFW-LMS-10B	513'-0"	

Table 6.1 (continued)

Equipment Number	Equipment Elevation	Radiation Level 40 yr + Direct Gamma + Airborne Gamma (R)
MS-RV-1A	546'-5"	
MS-RV-1B	546'-11"	
MS-RV-1C	546'-11'	
MS-RV-1D	546'-5"	
MS-RV-2A	546'-5"	
MS-RV-2B	546'-11"	
MS-RV-2C	546'-11"	
MS-RV-2D	546'-5"	
MS-RV-3A	546'-5"	
MS-RV-3B	546'-11"	
MS-RV-3C	546'-11"	
MS-RV-3D	546'-5"	
MS-RV-4A	546'-5"	
MS-RV-4B	546'-11"	to be provided at a later date
MS-RV-4C	546'-11"	
MS-RV-4D	546'-5"	
MS-RV-5B	546'-11'	
MS-RV-5C	546'-11"	
MS-TE-4A	546'-5"	
MS-TE-4B	546'-5"	
MS-TE-4C	546'-5"	
MS-TE-4D	546'-6"	
MS-TE-4E	546'-5"	
MS-TE-4F	546'-6"	
MS-TE-4G	546'-6"	
MS-TE-4H	546'-6"	
MS-TE-4J	546'-4"	
MS-TE-4K	546'-4"	
MS-TE-4L	546'-5"	
MS-TE-4M	546'-6"	
MS-TE-4N	546'-7"	
MS-TE-4P	546'-6"	
MS-TE-4R	546'-6"	
MS-TE-4S	546'-6"	
MS-TE-4U	546'-7"	
MS-TE-4V	546'-5"	

Table 6.1 (continued)

Equipment Number	Equipment Elevation	Radiation Level 40 yr + Direct Gamma + Airborne Gamma (R)
MS-RPV-3	Drywell	
CMS-TE-21	515'-8 1/2"	
CMS-TE-22	515'-8 1/2"	
CMS-TE-23	515'-8 1/2"	
CMS-TE-41	450'-8"	
CMS-TE-42	492'-8"	
CMS-TE-43	450'-8"	
CMS-TE-44	Drywell	
LPCS-LMS-6	547'-3"	
SPTM-TE-1A	465'-5"	
SPTM-TE-1B	465'-5"	
SPTM-TE-2A	465'-5"	
SPTM-TE-2B	465'-5"	
SPTM-TE-3A	465'-5"	
SPTM-TE-3B	465'-5"	
SPTM-TE-4A	465'-5"	to be provided at a later date
SPTM-TE-4B	465'-5"	
SPTM-TE-5A	465'-5"	
SPTM-TE-5B	465'-5"	
SPTM-TE-6A	465'-5"	
SPTM-TE-6B	465'-5"	
SPTM-TE-7A	465'-5"	
SPTM-TE-7B	465'-5"	
SPTM-TE-8A	465'-5"	
SPTM-TE-8B	465'-5"	
SPTM-TE-9	447'-10"	
SPTM-TE-10	447'-10"	
SPTM-TE-11	447'-10"	
SPTM-TE-12	447'-10"	
SPTM-TE-13	447'-10"	
SPTM-TE-14	447'-10"	
SPTM-TE-15	447'-10"	
SPTM-TE-16	447'-10"	

Table 6.1 (continued)

Equipment Number	Equipment Elevation	Radiation Level 40 yr + Direct Gamma + Airborne Gamma (R)
LD-V-5A	508'-7"	
LD-V-5AA	501'-0"	
LD-V-5B	508'-7"	
LD-V-5BB	500'-10"	
LD-V-5C	508'-7"	
LD-V-5CC	546'-7"	
LD-V-5D	508'-7"	
LD-V-5DD	579'-5 1/2"	
LD-V-5E	503'-7"	
LD-V-5EE	503'-2"	
LD-V-5F	505'-4 1/2"	
LD-V-5G	507'-1"	
LD-V-5H	504'-5"	
LD-V-5L	507'-4"	
LD-V-5M	507'-5"	
LD-V-5N	508'-3"	to be provided at a later date
LD-V-5Q	556'-9"	
LD-V-5R	561'-11"	
LD-V-5S	556'-9"	
LD-V-5T	509'-2"	
LD-V-5U	506'-2"	
LD-V-5V	546'-8"	
LD-V-5W	548'-0"	
LD-V-5X	539'-7"	
LD-V-5Y	501'-3"	
LD-V-5Z	514'-11"	
LD-TE-16C1	549'-0"	
LD-TE-16C3	549'-0"	
LD-TE-16C4	549'-0"	
LD-TE-16C5	549'-0"	
LD-TE-16C6	549'-0"	
LD-TE-16C7	549'-0"	
LD-TE-16D1	549'-0"	
LD-TE-16E1	549'-0"	
LD-TE-16E2	549'-0"	
LD-TE-16F1	549'-0"	
LD-TE-16G1	549'-0"	

Table 6.1 (continued)

Equipment Number	Equipment Elevation	Radiation Level 40 yr + Direct Gamma + Airborne Gamma (R)
RHR-LMS-111A	562'-11"	
RHR-LMS-111B	562'-11"	
RHR-LMS-111C	562'-11"	
RHR-LMS-112A	509'-2"	
RHR-LMS-112B	509'-2"	
RHR-LMS-113	509'-7"	
TRM-CONN-01	Drywell	
TRM-CONN-02	Drywell	
TRM-CONN-03	Drywell	
TRM-CONN-04	Drywell	to be provided at a later date
TRM-CONN-05	Drywell	
TRM-CONN-06	Drywell	
TRM-CONN-07	Drywell	
TRM-CONN-08	Drywell	
RRC-TE-23A	Drywell	
RRC-TE-23B	Drywell	
RRC-TE-28A	Drywell	
RRC-TE-28B	Drywell	
RRC-TE-35A	Drywell	
RRC-TE-35B	Drywell	

Table 6.2

Individual Zone Sketches

of

Safety-Related Equipment Location

Elevations	Zones
422	C,D,E,I,J,L,M.
441	B,C,D,F,G,I,J.
471	A,B,D,E,F,H,I,J...480M.
501	B,F,I,F,M,O,P...510S.
522	B,C,D,F,G,H,J,K,N,O,P.
548	B,C,E,F,G,H,J,K,L,M,N,P,Q.
572	B,C,D,F,H,I,L,N.
606	A.

Table 6.3

WNP-2 1E/1M Equipment Vital Area List
Outside the Reactor Building For
Six Month Total Integrated
Dose (LOCA)

Vital Area Description	Radiation Level ** Direct Gamma Shine Gamma + Airborne Gamma (Rads)
Control Room (EL 501)	0.58
Technical Support Center	0.58
Sample Area (EL 487)	10.2
Nitrogen Supp'y to ADS Accumulators (EL 437)	7.6
Standby Service Water Pump Valves (cooling ponds)	3.5
Remote Shutdown Room (EL 467)	7.6
Switchgear Room #1 (EL 467)	7.6
Switchgear Room #2 (EL 467)	7.6
Radwaste Control Room (EL 467)	7.6
Battery Racks, DC Battery Chargers and 2MCC's* (EL 467)	7.6
3 MCC's* and 3 switchgears (EL 437)	7.6
DC Battery Charger and Rack (EL 437)	7.6
Diesel Oil Tanks (EL 437)	7.6
Solid Radwaste Control Panel and Decontamination Station Control Panel (EL 437)	7.6

* MCC's - Motor Control Centers

** Volume Correction Factors for a semi-infinite cloud were only applied to the Control Room and TSC due to the minimal exposure to the equipment. If the Volume Correction Factors were to be applied to all areas the integrated dose would be reduced by a minimum of five fold.

Table 6.4

WNP-2 Vital Areas and Access Route
List of Radiation Exposure to Personnel
During the Required Post-LOCA Operations

Vital Area Description	Radiation Exposure Gamma Whole Body (Rem)	Thyroid (Rem)(5)	Beta Skin (Rads)
*Control Room (EL 501)	0.58	0.58(1)	3.05
*Technical Support Center	0.58	0.58(1)	3.05
*Security Center	5.9	26.4(2)	9.4
*Auxiliary Security Center	3.5	15.3(2)	5.5
**Standby Service Water * Pump Valves (cooling ponds)	0.6	1.9(2)	0.9
*Sample Analysis Area (EOC)	0.0013	-	-
**All infrequenty occupied vital areas inside the Radwaste and Diesel Generator Buildings	0.25(4)	3.2(2)	0.95
**All access routes inside the Radwaste and Diesel Generator Buildings	0.25 (4)	3.2 (2)	0.95
**All access routes (3) outside the Radwaste and Diesel Generator Buildings	1.0	3.2(2)	1.5
**Post Accident Sample Area (487)	0.6	6.4(2)	2.0

* Area of continious occupancy

** Area occupied 0.5 hours at times after 1 hour into the LOCA

(1) Assumes self-contained respiratory equipment was used by personnel during 0 - 3 hours post-LOCA situation.

(2) No respiratory equipment was assumed.

(3) Extremely conservative analysis since the plane of airborne radioactivity cannot simultaneously cover all access routes.

Table 6.4 Continued

(4) A Volume Correction Factor for the semi-infinite cloud was included in the calculation

(5) If Self Contained Respiratory Equipment (SCBA) is used the thyroid dose will essentially equal the whoe body dose.

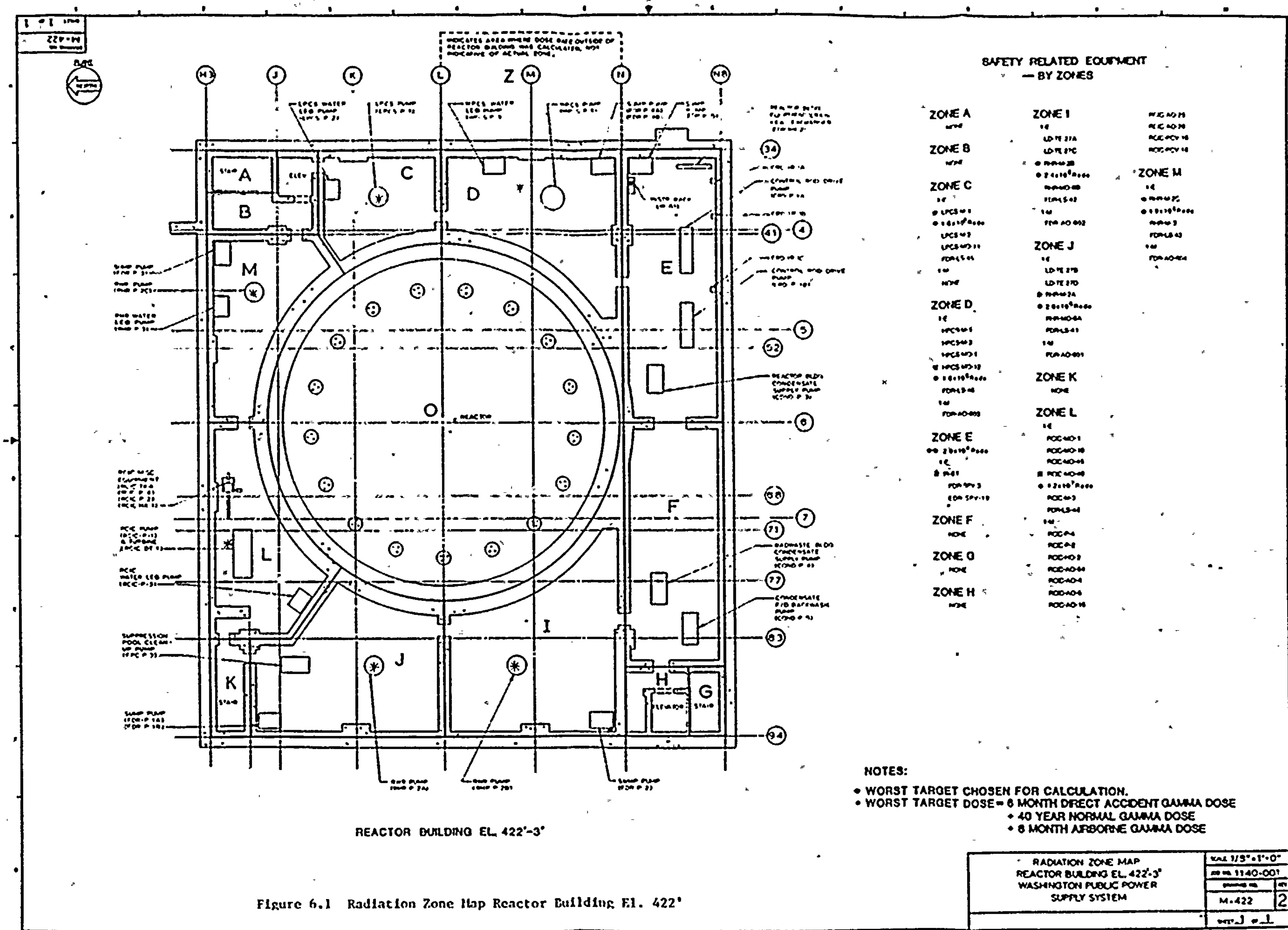


Figure 6.1 Radiation Zone Map Reactor Building El. 422'

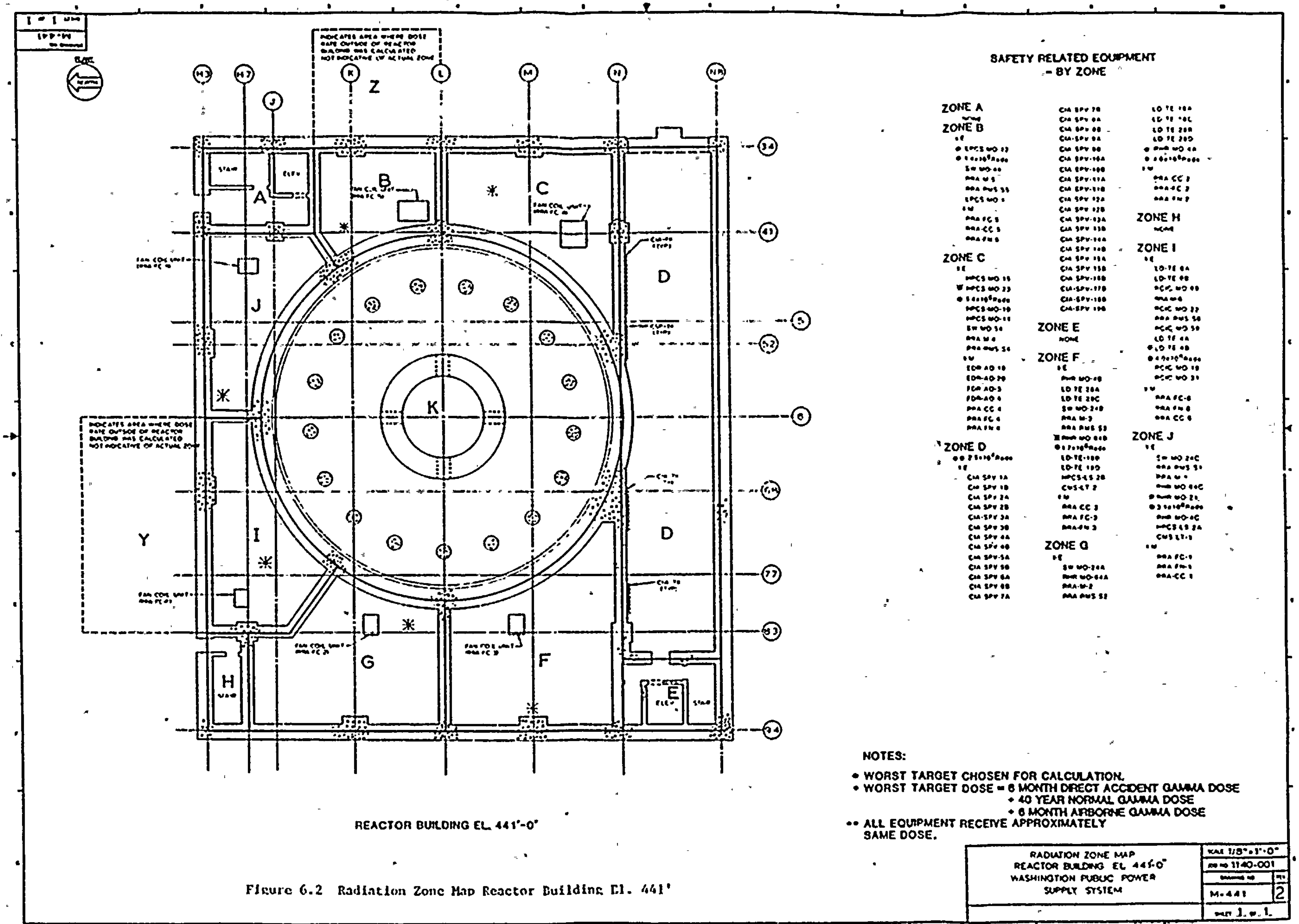
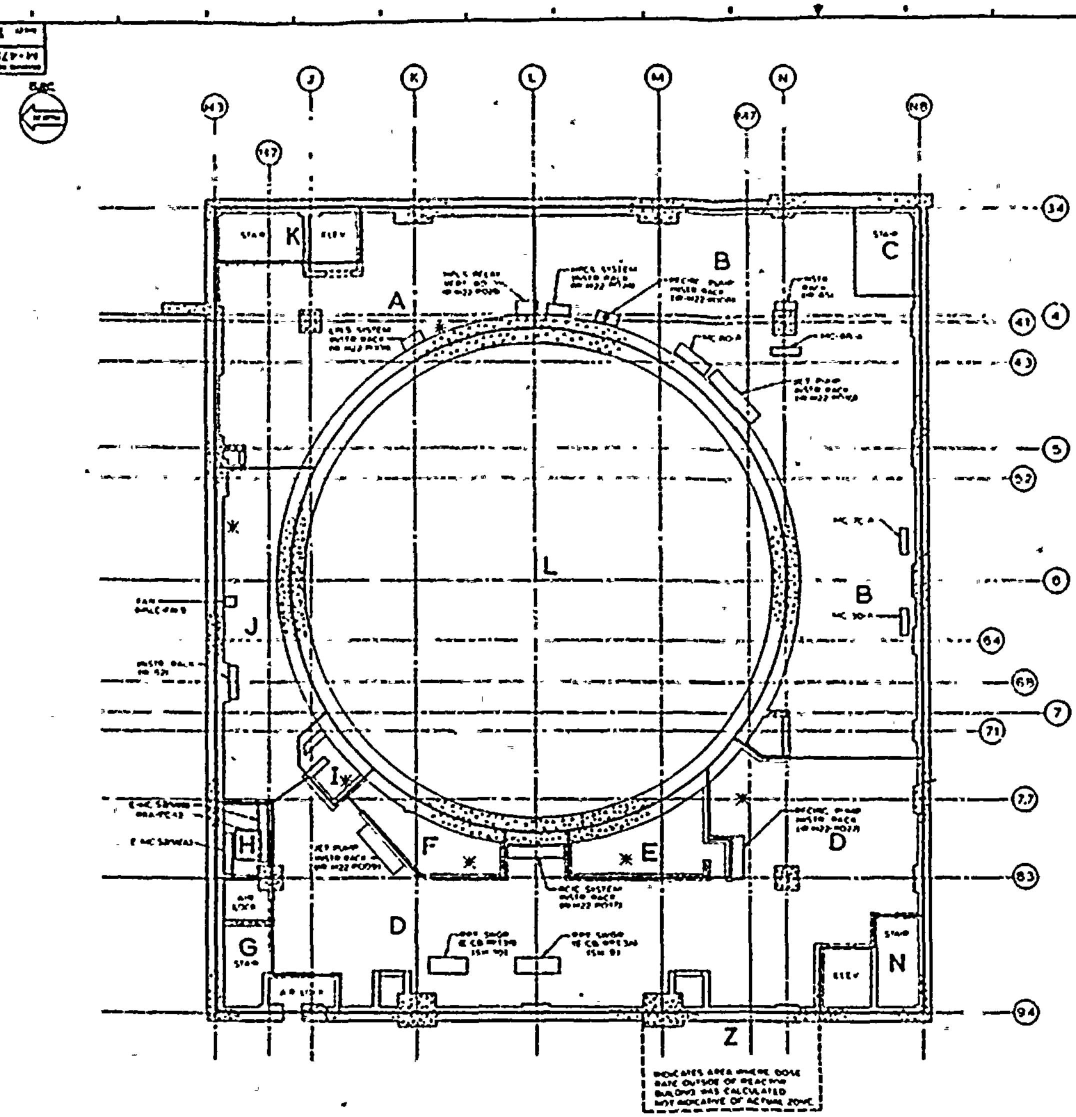


Figure 6.2 Radiation Zone Map Reactor Building El. 441'



SAFETY RELATED EQUIPMENT
-BY ZONES

ZONE A

ZONE B

ZONE C

ZONE D

ZONE E

ZONE F

ZONE G
NONE

ZONE H

ZONE I

ZONE J

ZONE K
NONE

ZONE L
NONE

ZONE M

ZONE N
NONE

NOTES:

- WORST TARGET CHOSEN FOR CALCULATION.
- WORST TARGET DOSE = 6 MONTH DIRECT ACCIDENT GAMMA DOSE
  + 40 YEAR NORMAL GAMMA DOSE
  + 6 MONTH AIRBORNE GAMMA DOSE
- •• ALL EQUIPMENT RECEIVE APPROXIMATELY SAME DOSE.

RADIATION ZONE MAP REACTOR BUILDING EL.471'-0" WASHINGTON PUBLIC POWER SUPPLY SYSTEM	SCALE 1/8" = 1'-0" 1140-C01 M-471 2 SHEET 1 of 1

Figure 6.3 Radiation Zone Map Reactor Building El. 471'

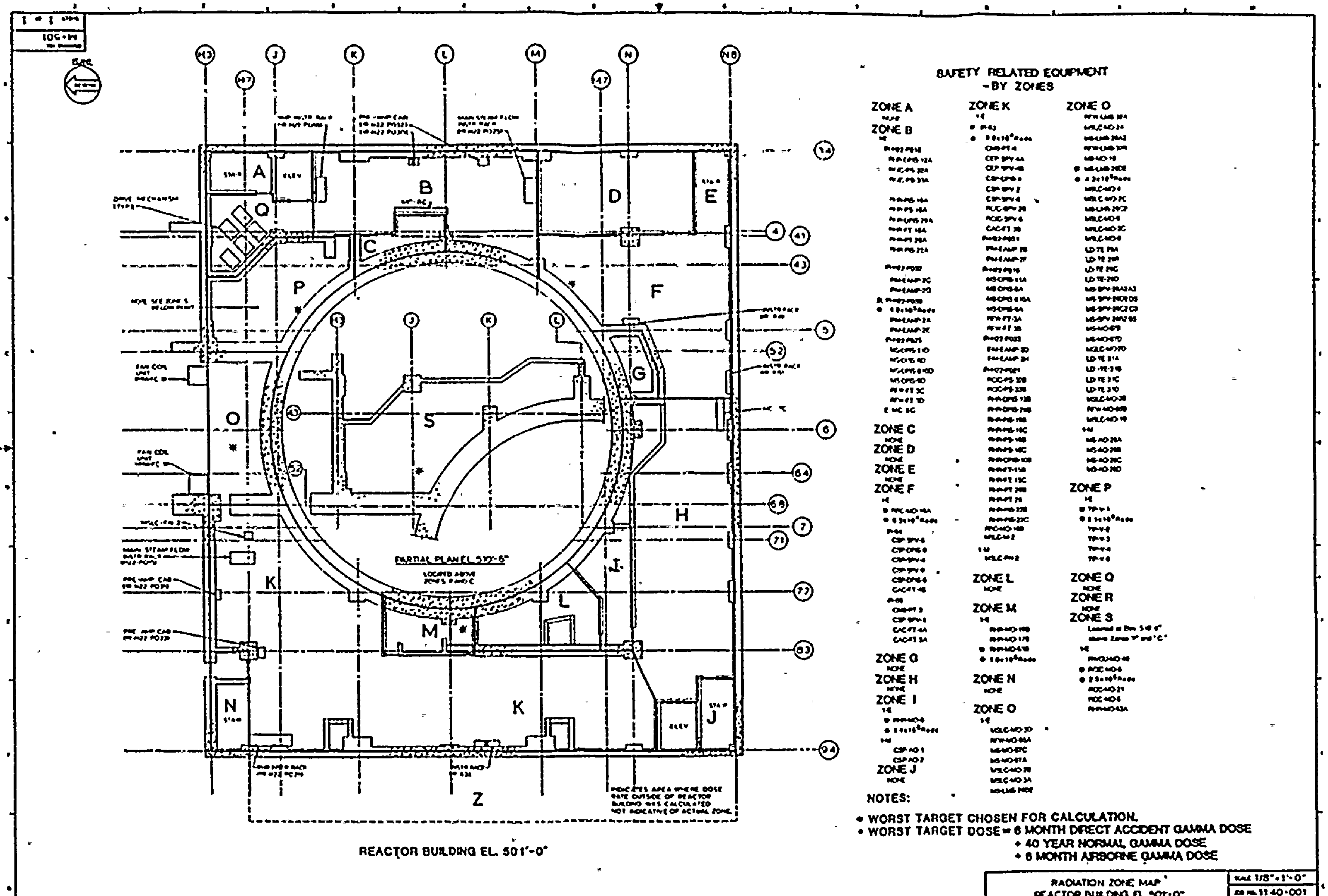
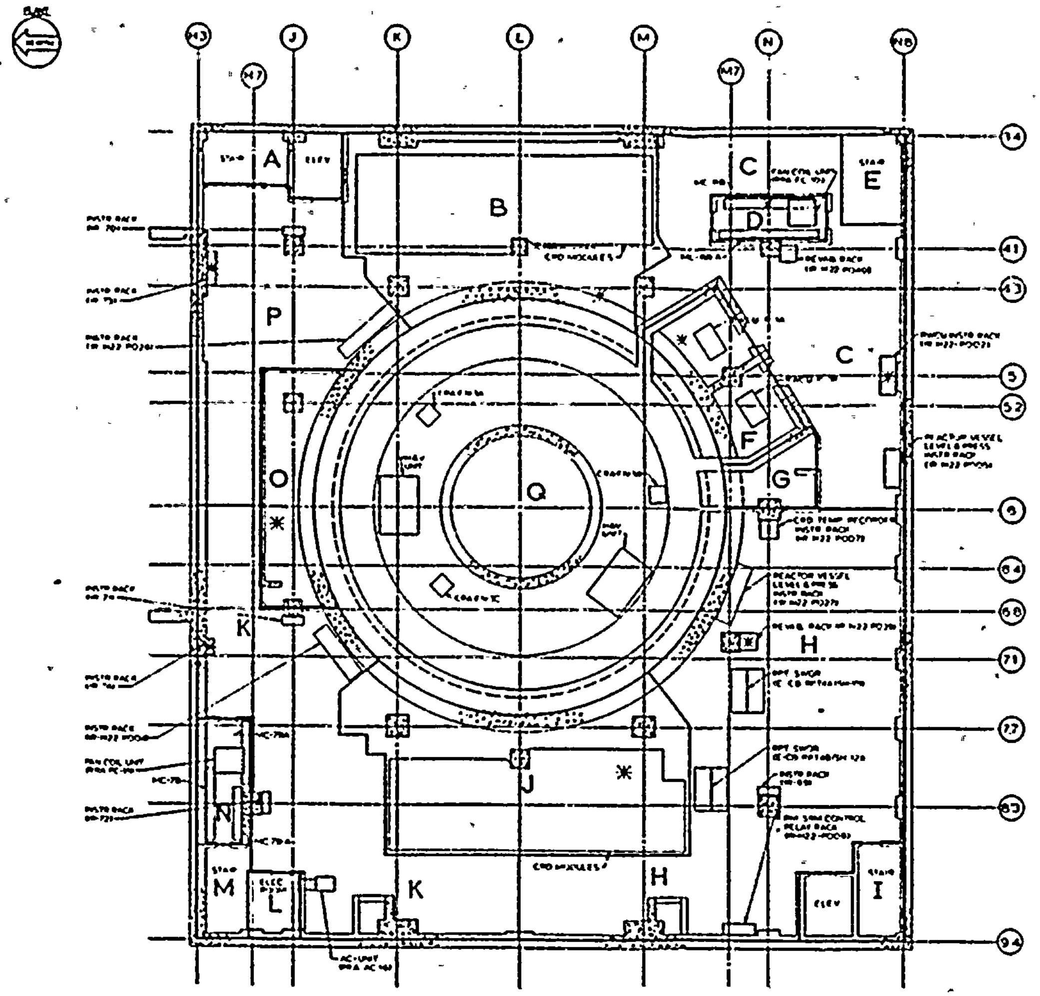


Figure 6.4 Radiation Zone Map Reactor Building El. 501'



REACTOR BUILDING EL. 522'-0"

SAFETY RELATED EQUIPMENT
- BY ZONES

ZONE A

ZONE B

ZONE C

ZONE D

ZONE E

ZONE F

ZONE G

ZONE H

ZONE I

ZONE J

ZONE K

ZONE L

ZONE M

ZONE N

ZONE O

ZONE P

NOTES:

* WORST TARGET CHOSEN FOR CALCULATION.
* WORST TARGET DOSE = 6 MONTH DIRECT ACCIDENT GAMMA DOSE
 + 40 YEAR NORMAL GAMMA DOSE
 + 6 MONTH AIRBORNE GAMMA DOSE

** ALL EQUIPMENT RECEIVE APPROXIMATELY SAME DOSE.

RADIATION ZONE MAP REACTOR BUILDING EL. 522'-0" WASHINGTON PUBLIC POWER SUPPLY SYSTEM	1/8"=1'-0" 1140-001 M-522 2

Figure 6.5 Radiation Zone Map Reactor Building El. 522'

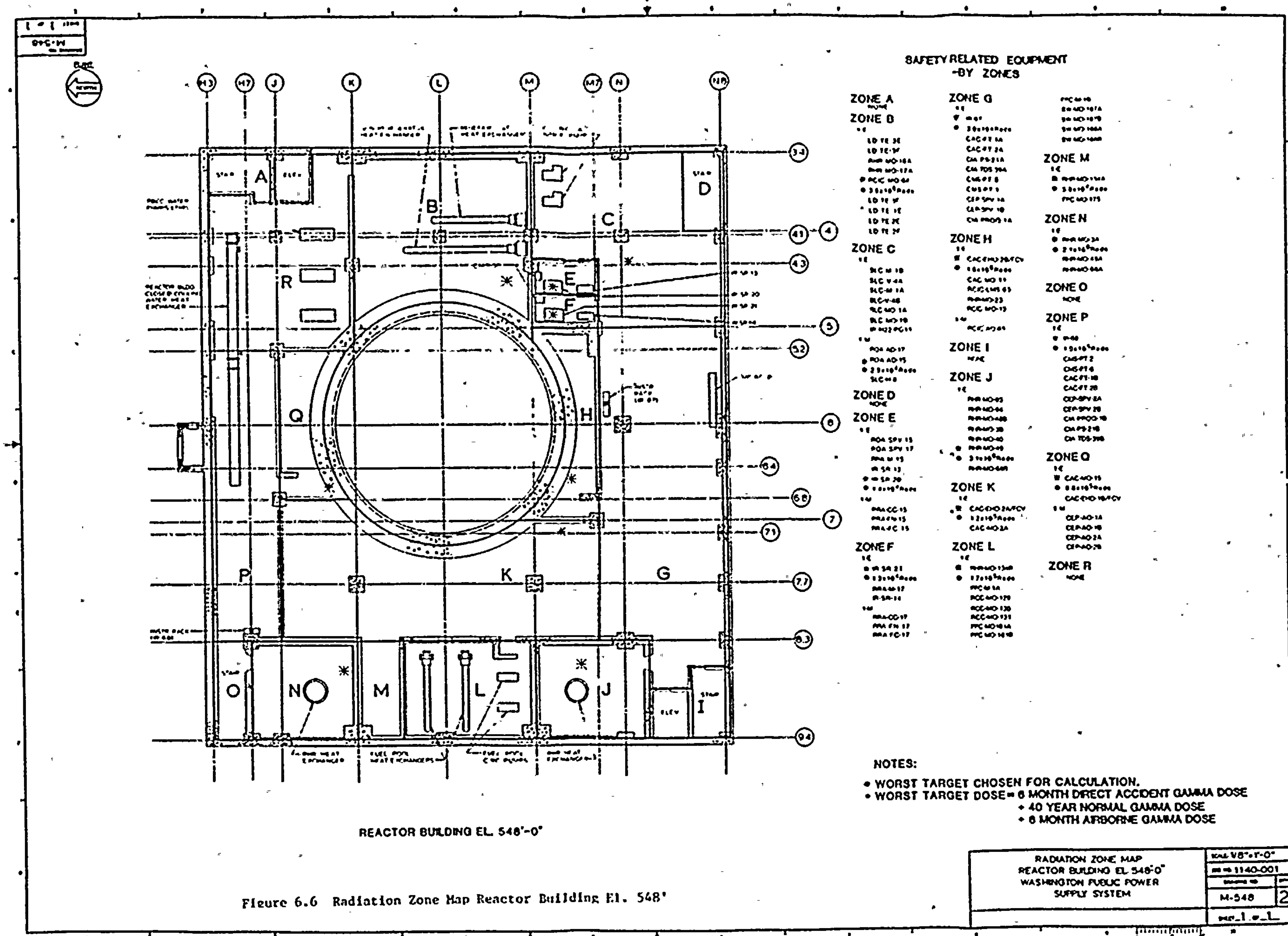


Figure 6.6 Radiation Zone Map Reactor Building El. 548'

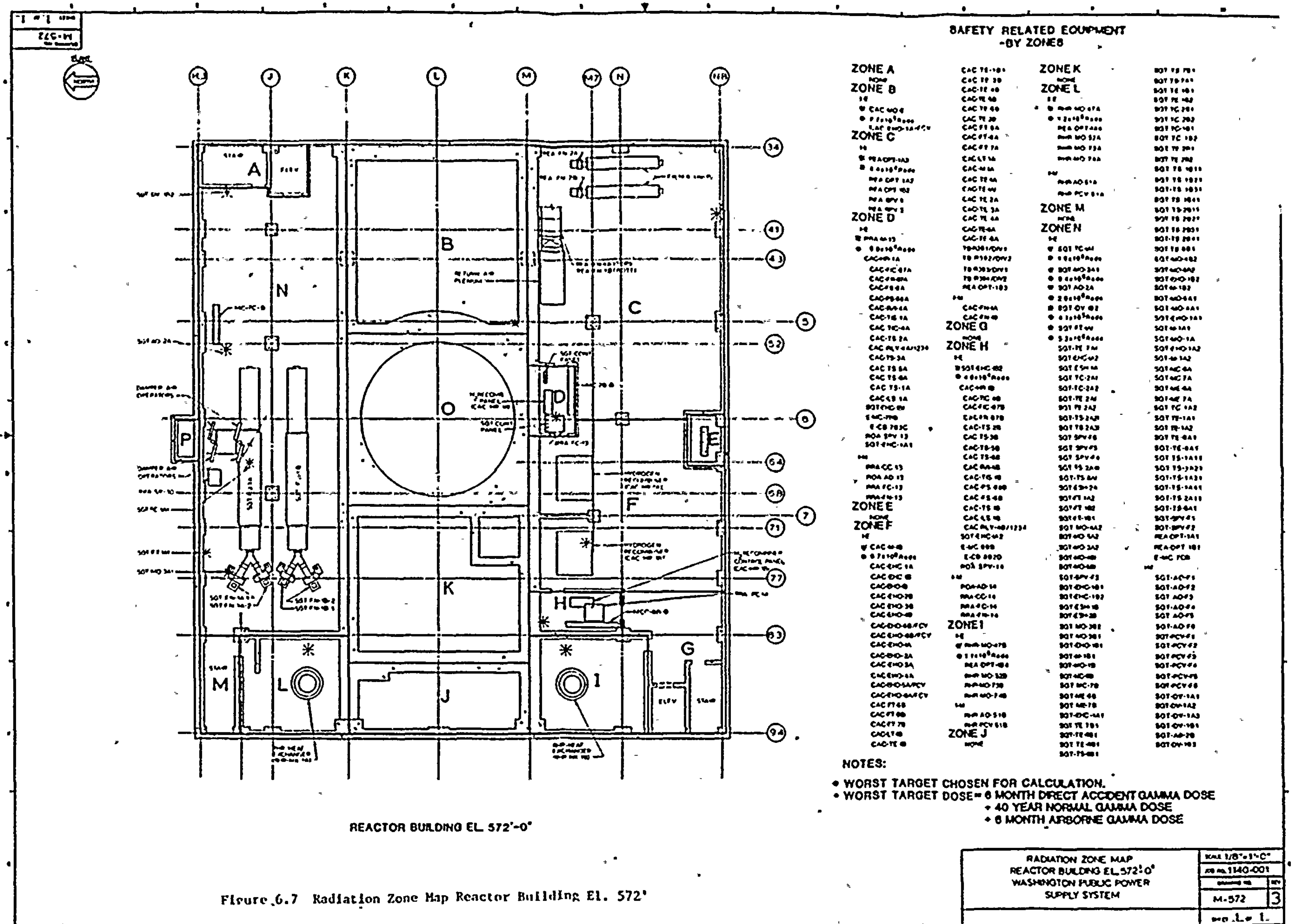


Figure 6.7 Radiation Zone Map Reactor Building El. 572'

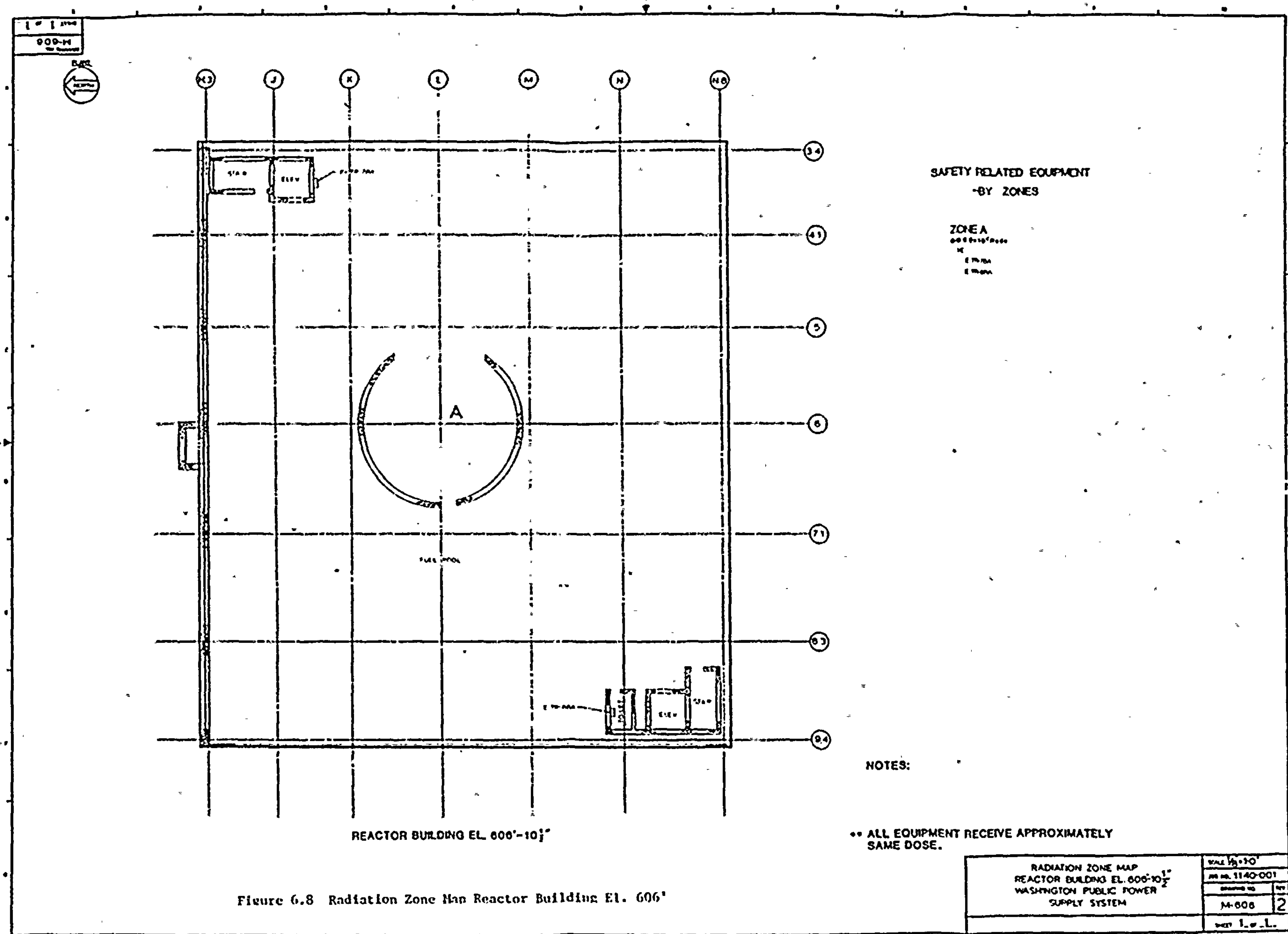


Figure 6.8 Radiation Zone Map Reactor Building El. 606'

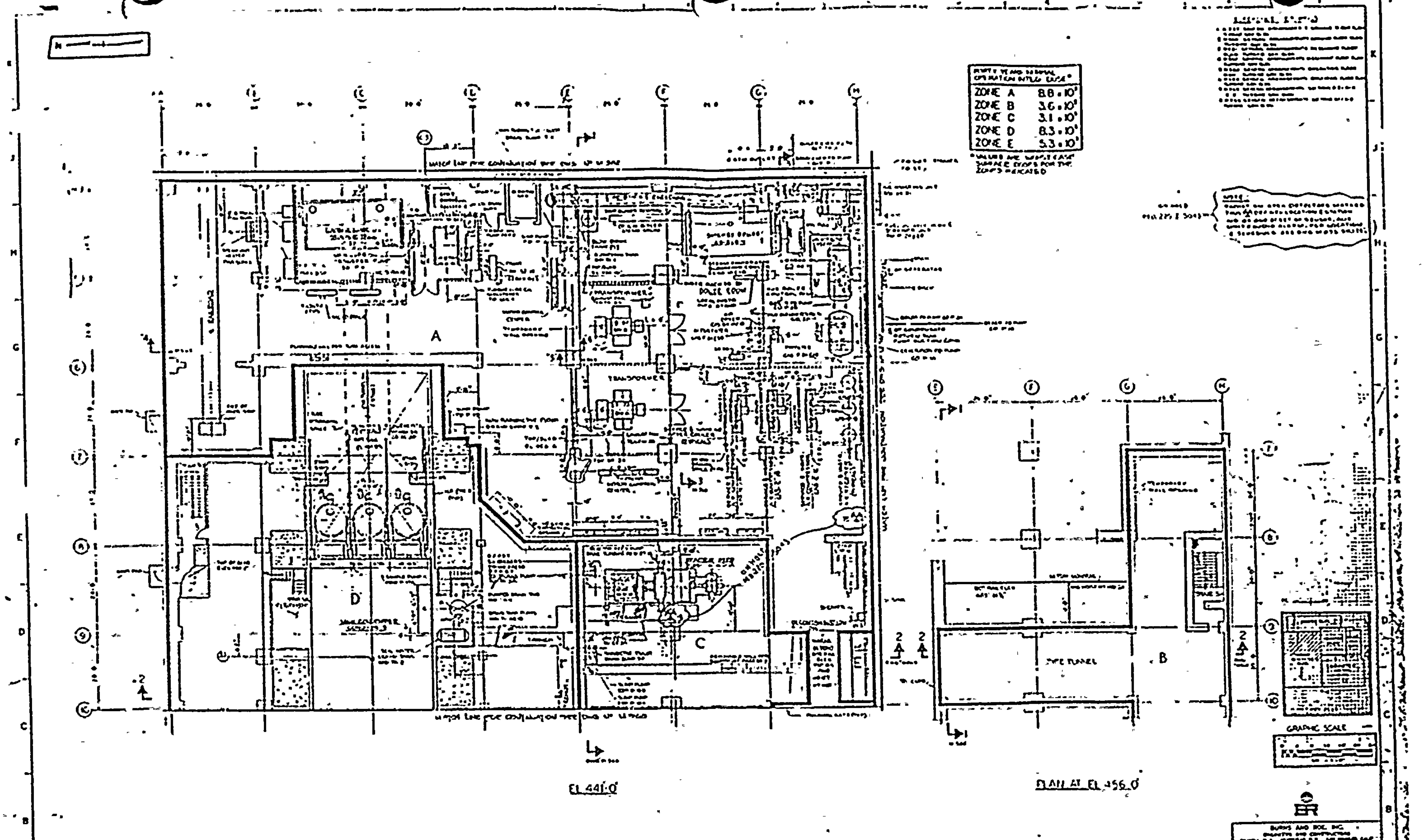


Figure 6.9 Forty-Year Integrated Dose Turbine Generator Bldg (El. 441')

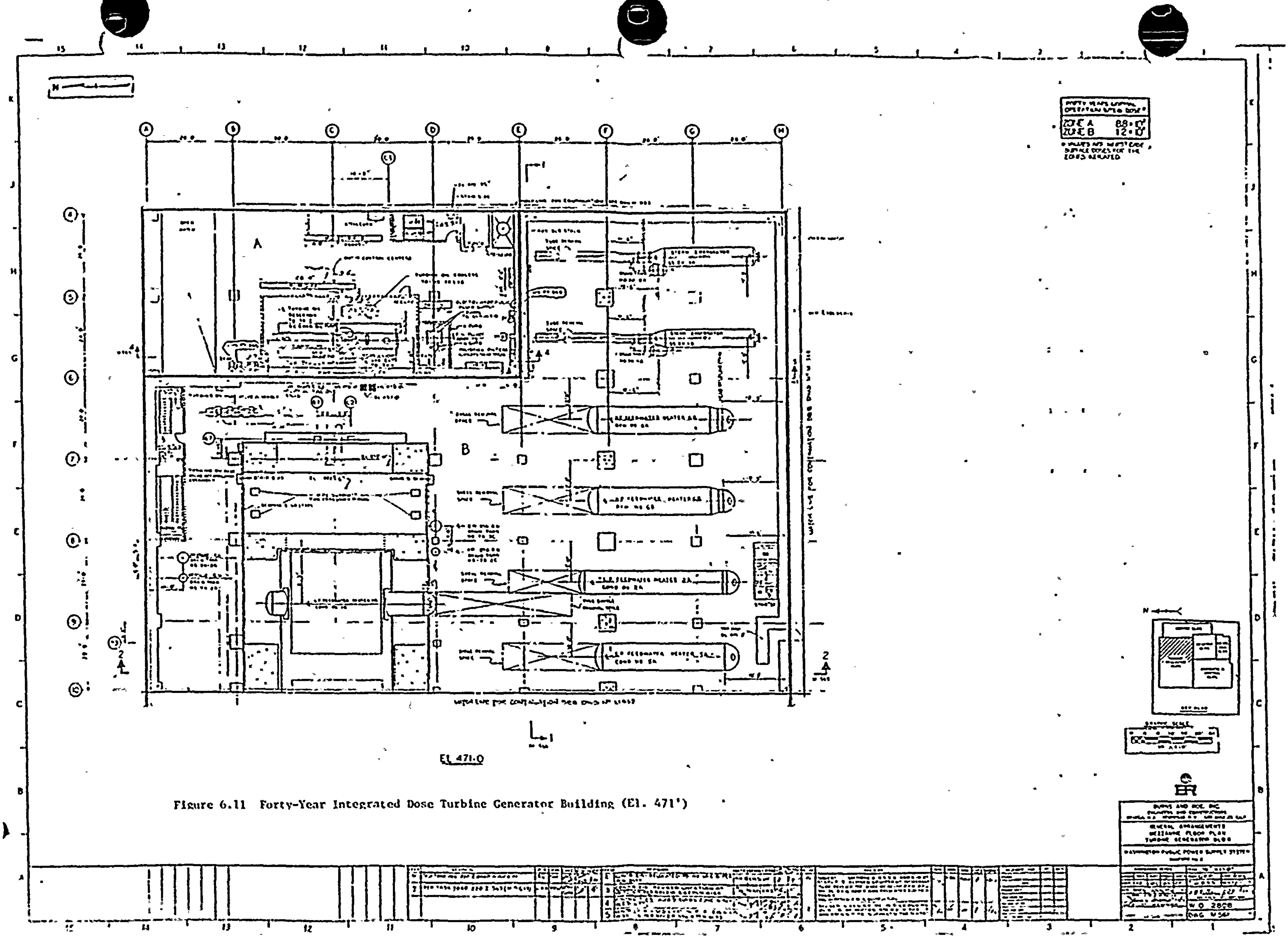
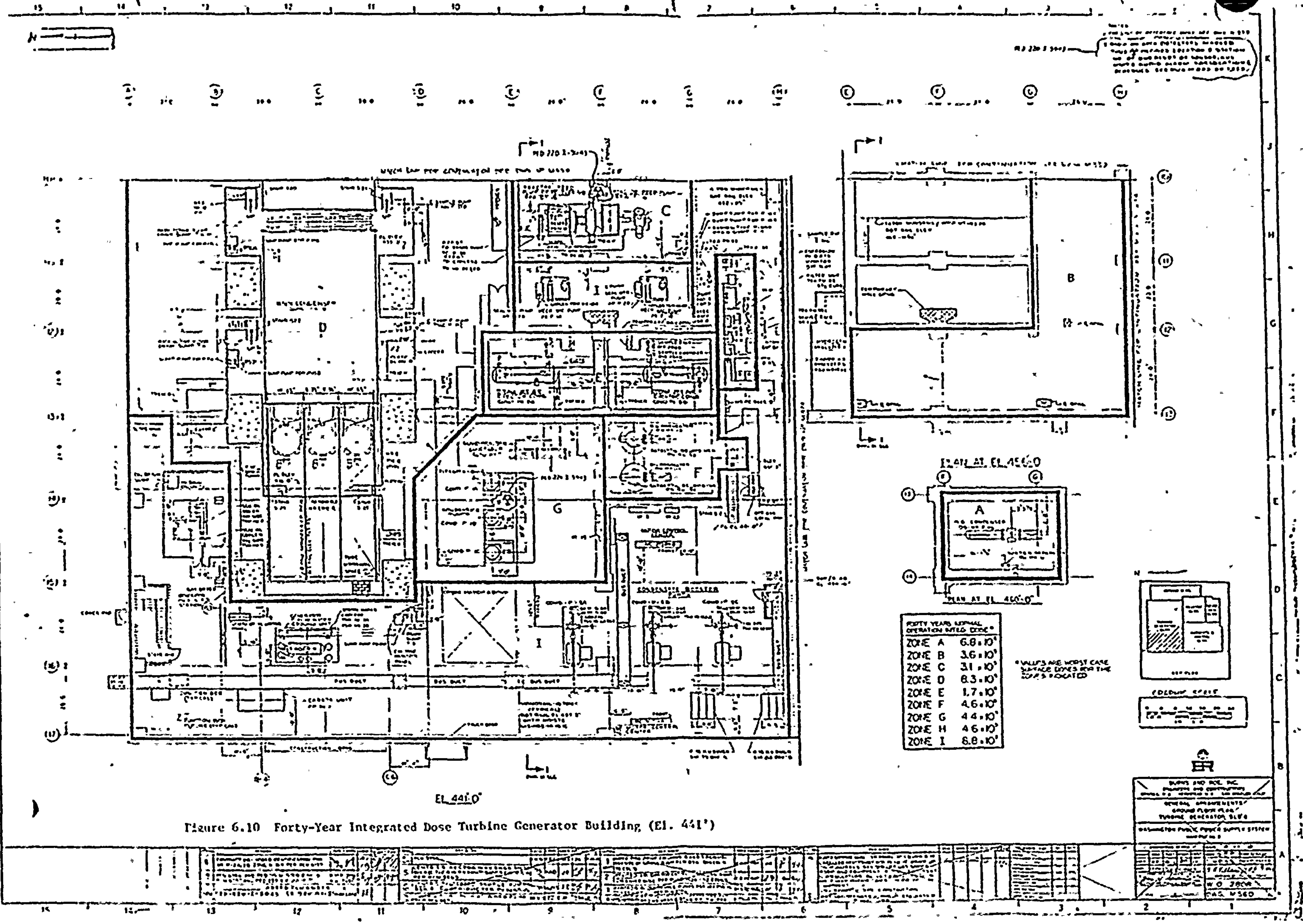


Figure 6.11 Forty-Year Integrated Dose Turbine Generator Building (El. 471')



FORTY YEARS NORMAL OPERATION INTEG DOSE*		
ZONE	A	6.8 x 10[illegible]
ZONE	B	3.6 x 10[illegible]
ZONE	C	3.1 x 10[illegible]
ZONE	D	8.3 x 10[illegible]
ZONE	E	1.7 x 10[illegible]
ZONE	F	4.6 x 10[illegible]
ZONE	G	4.4 x 10[illegible]
ZONE	H	4.6 x 10[illegible]
ZONE	I	8.8 x 10[illegible]

Figure 6.10 Forty-Year Integrated Dose Turbine Generator Building (El. 441')

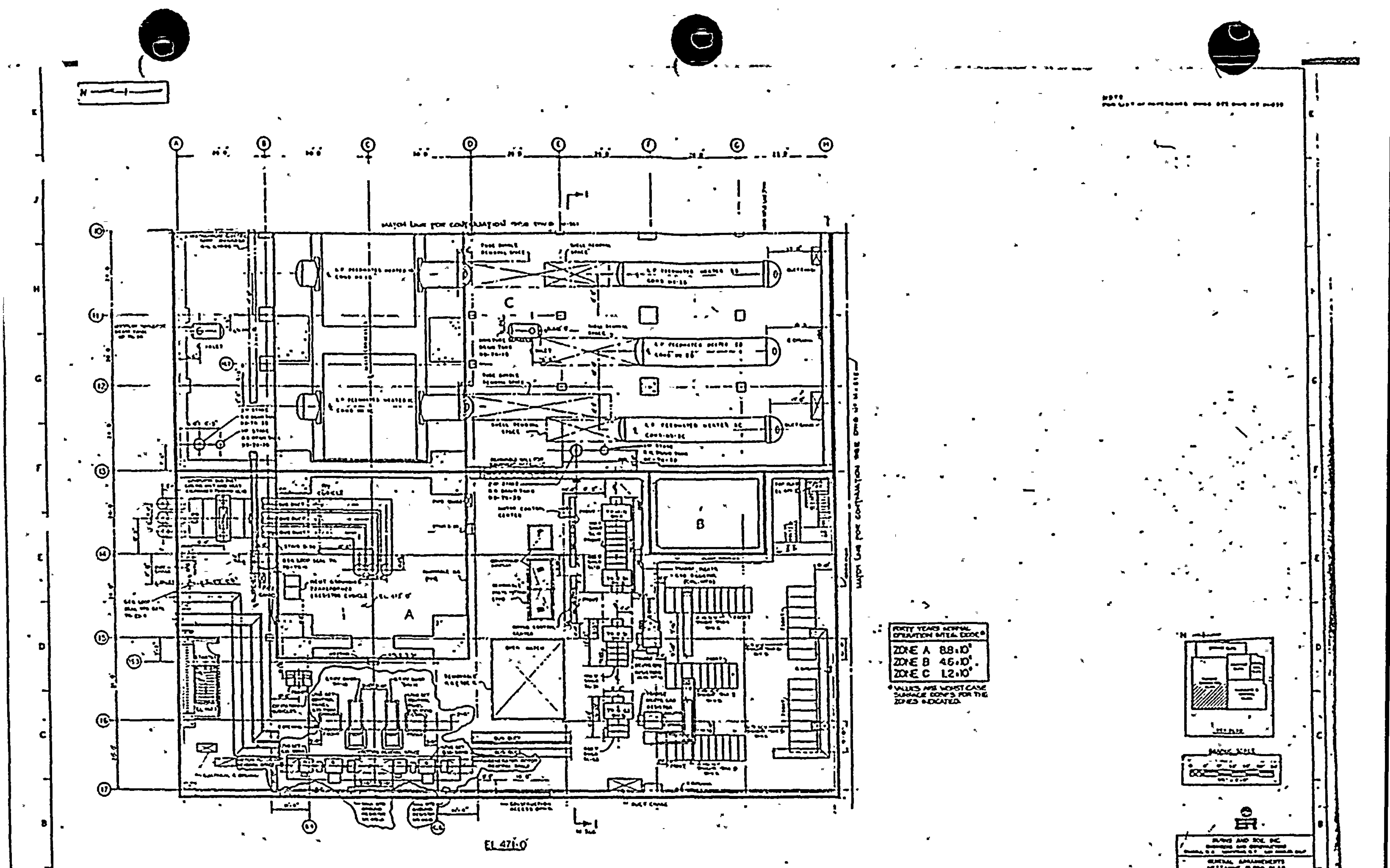


Figure 6.12 Forty-Year Integrated Dose Turbine Generator Building (El. 471')

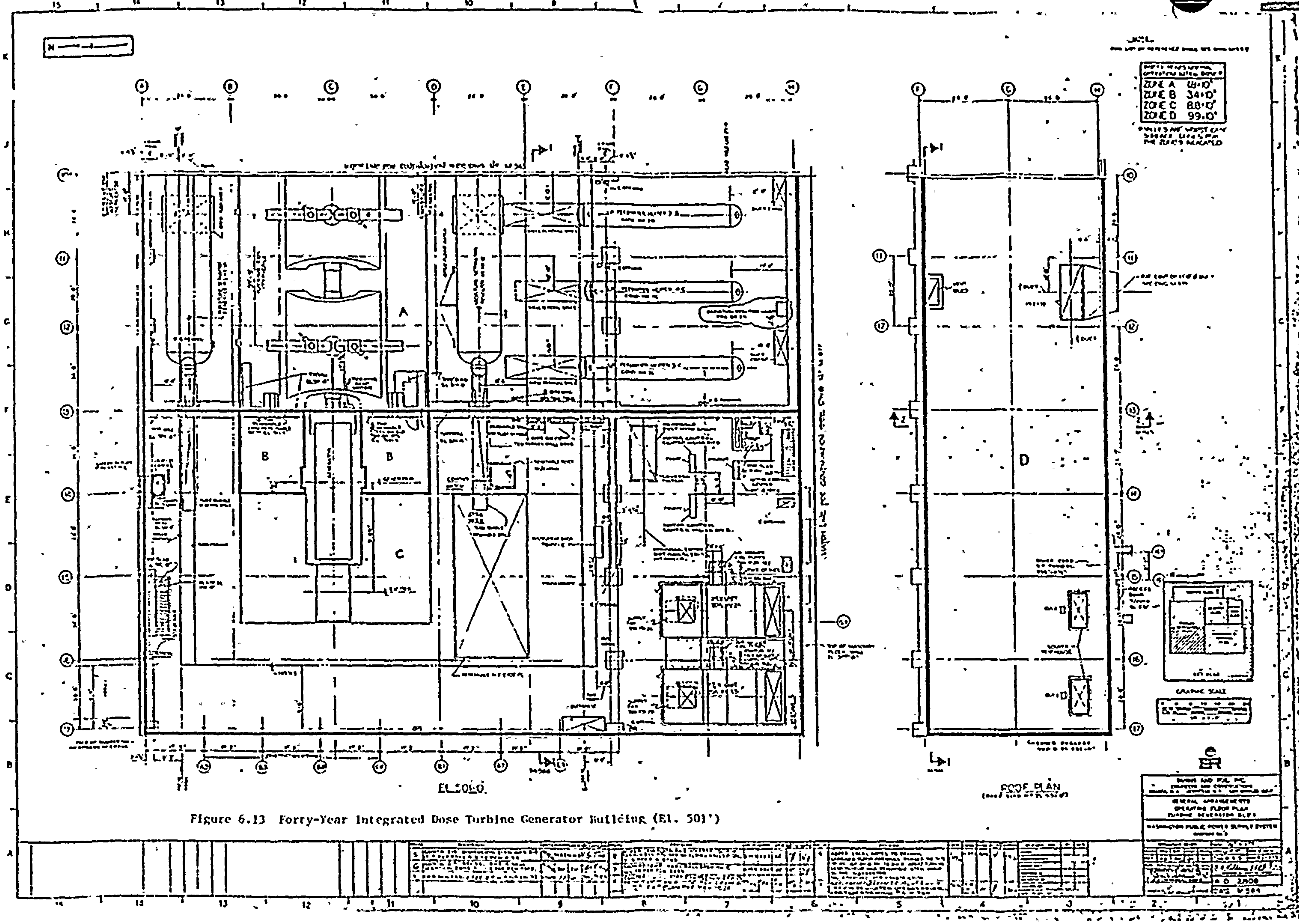


Figure 6.13 Forty-Year Integrated Dose Turbine Generator Building (El. 501')

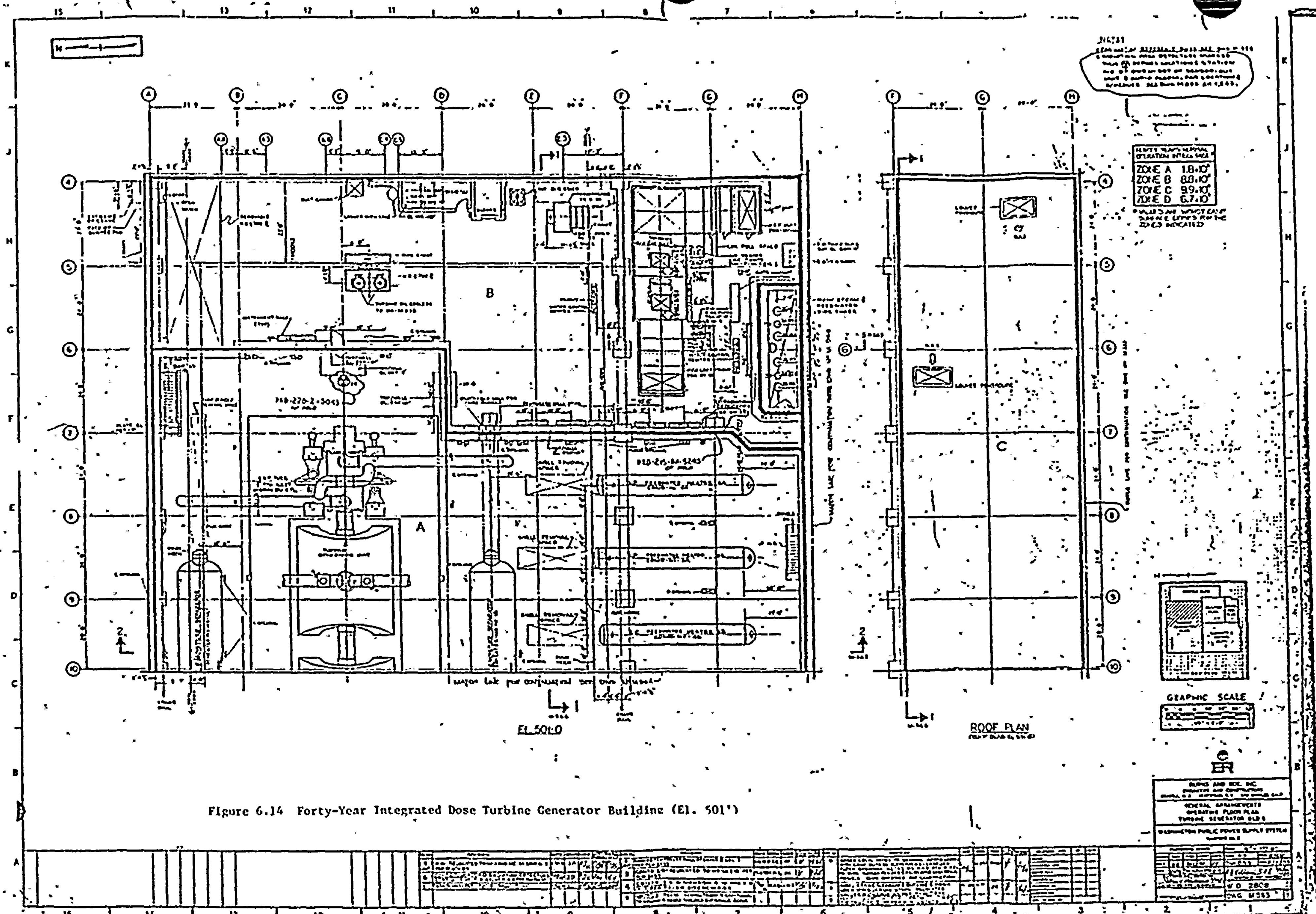


Figure 6.14 Forty-Year Integrated Dose Turbine Generator Building (El. 501')

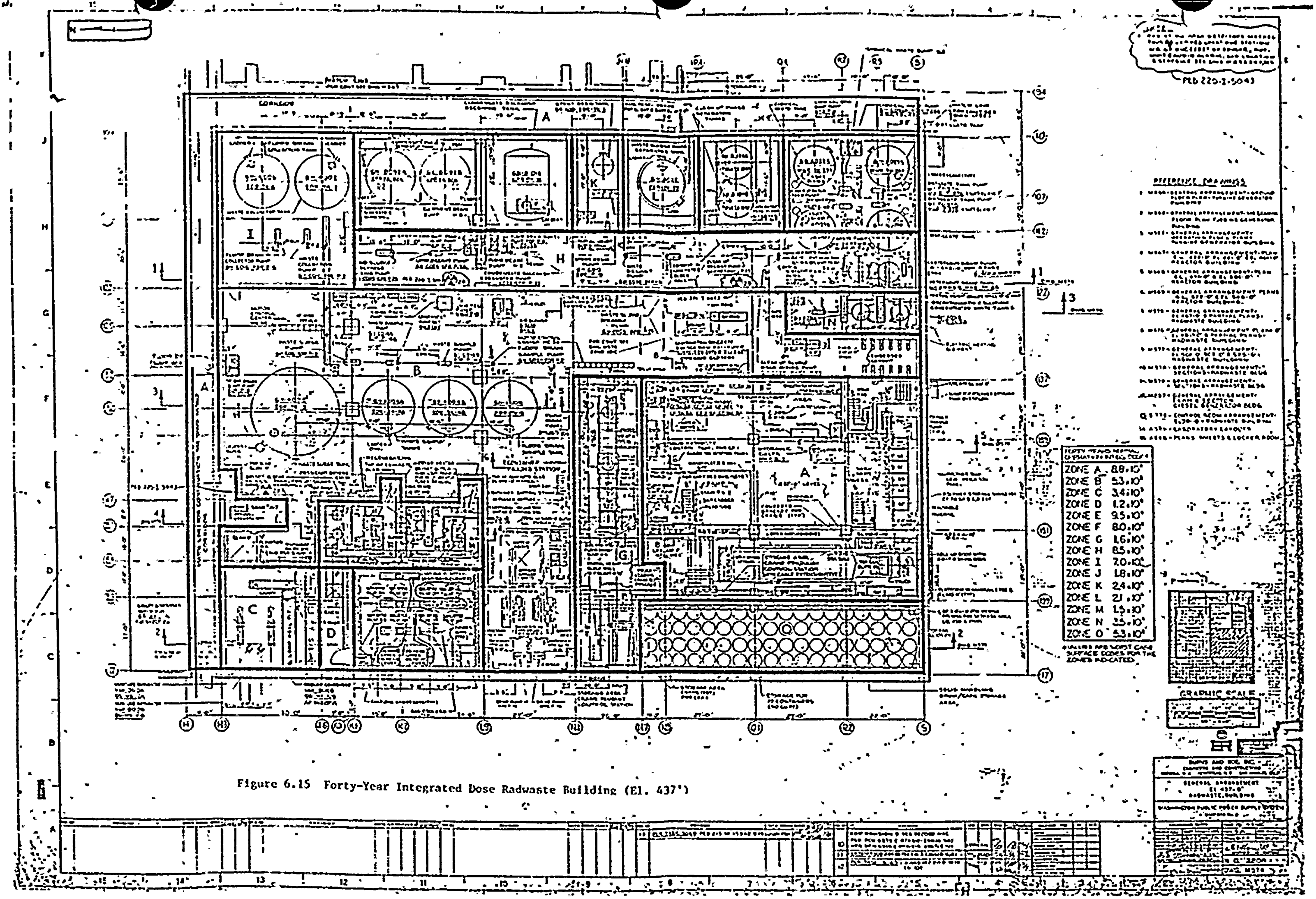


Figure 6.15 Forty-Year Integrated Dose Radwaste Building (El. 437')

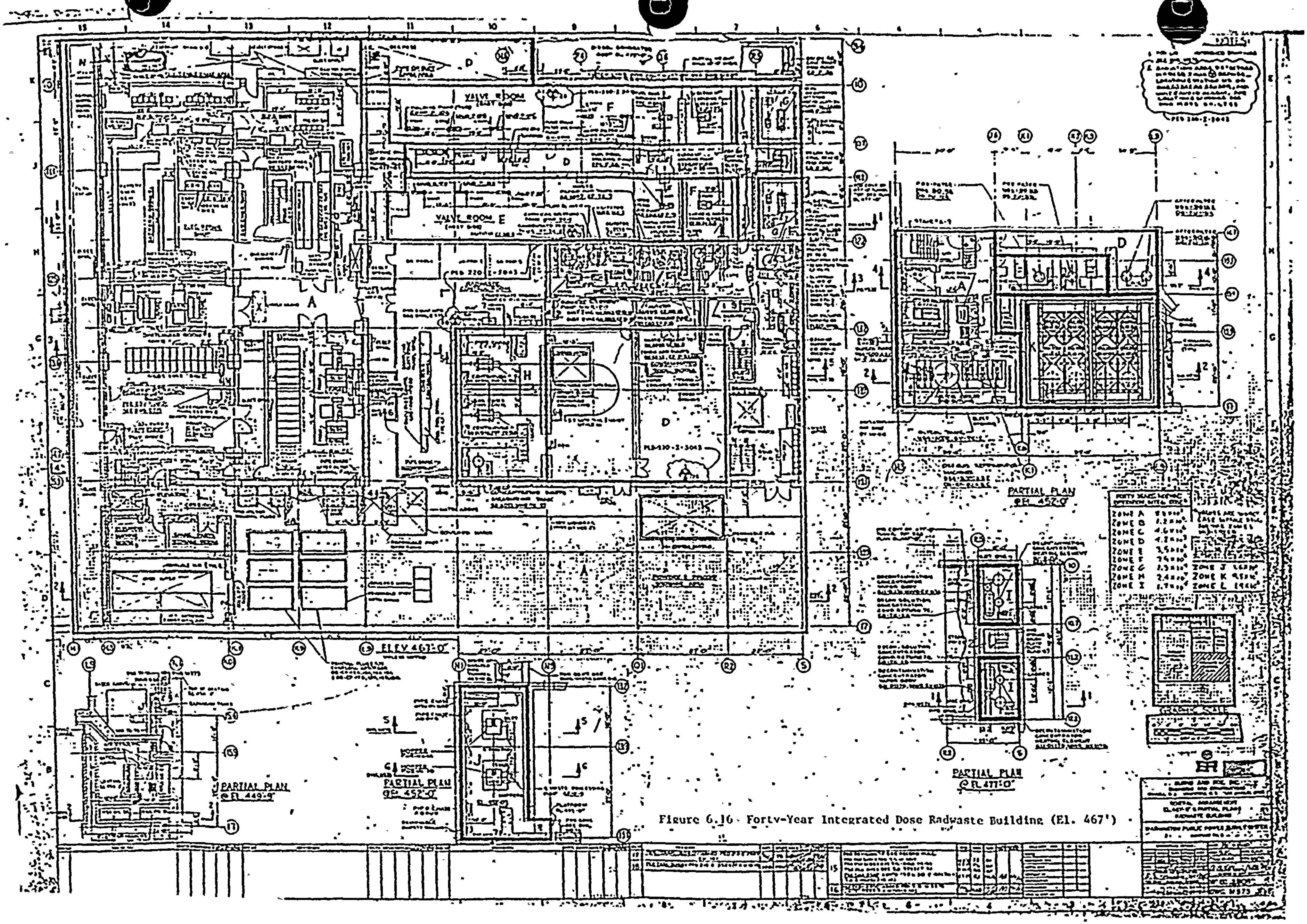


Figure 6.16 Forty-Year Integrated Dose Radwaste Building (El. 467')

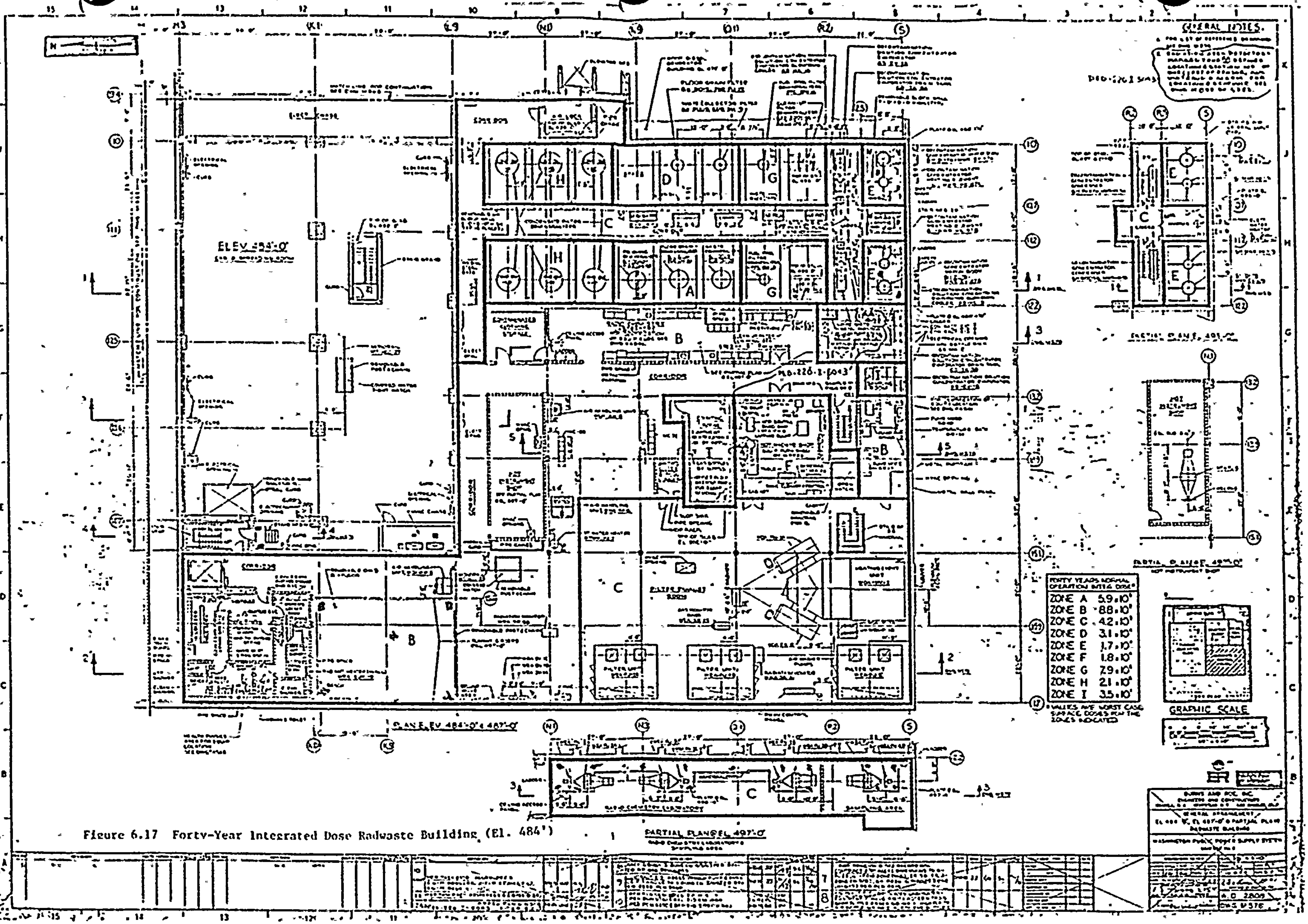


Figure 6.17 Forty-Year Integrated Dose Radwaste Building (El. 484')

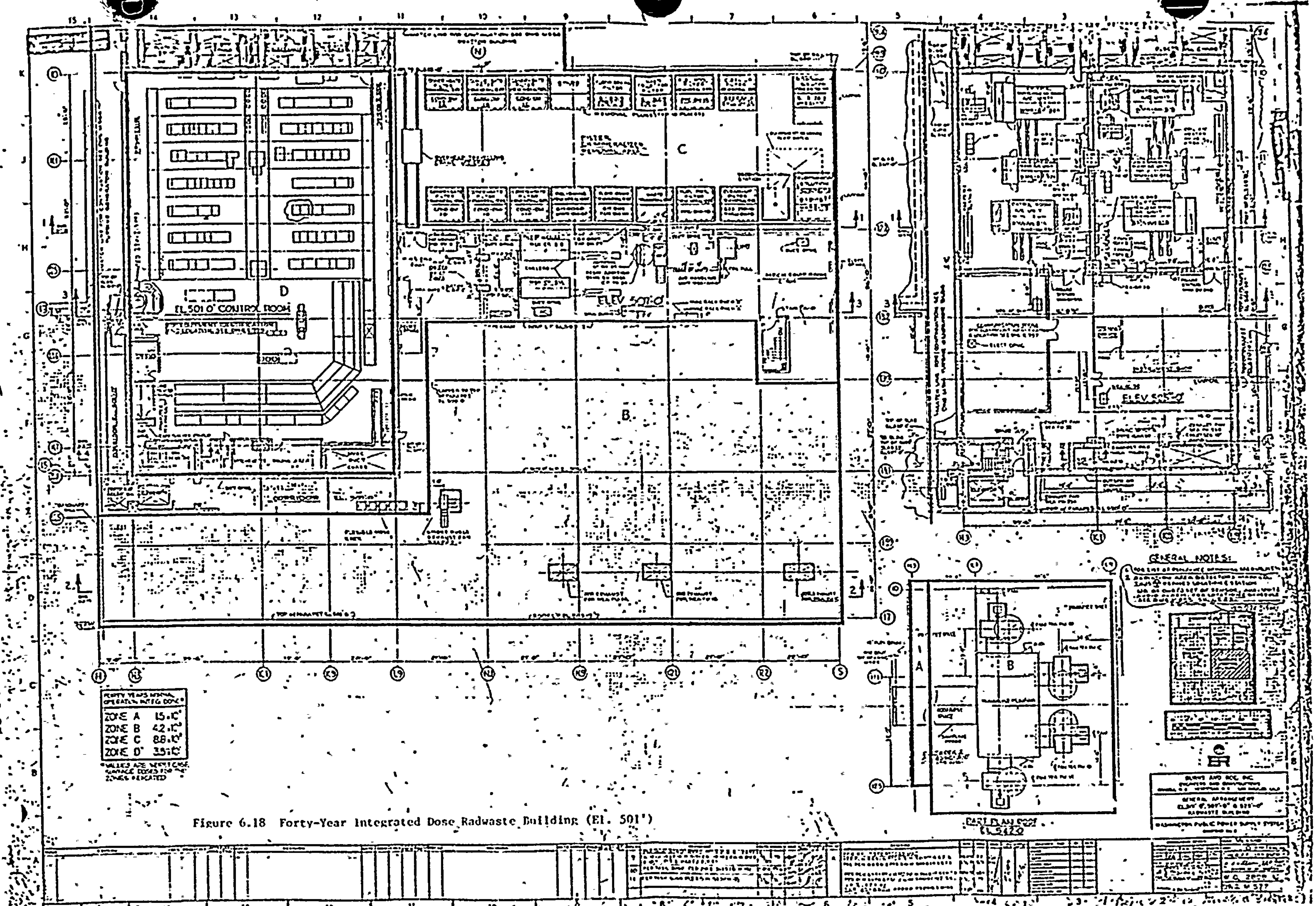


Figure 6.18 Forty-Year Integrated Dose Radwaste Building (El. 501')

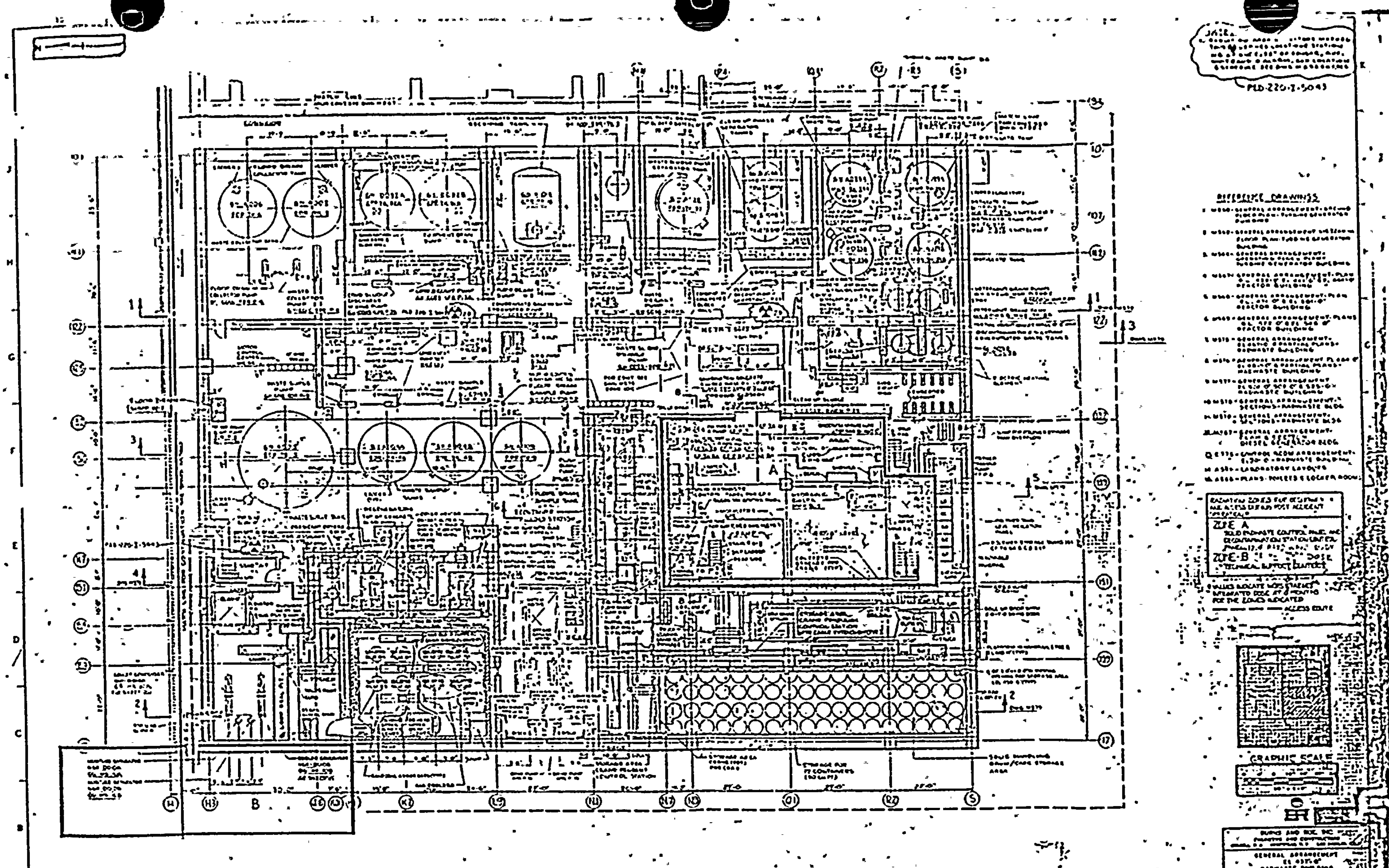


Figure 6.19 Vital Areas and Access Routes for Radwaste Building (El. 437')

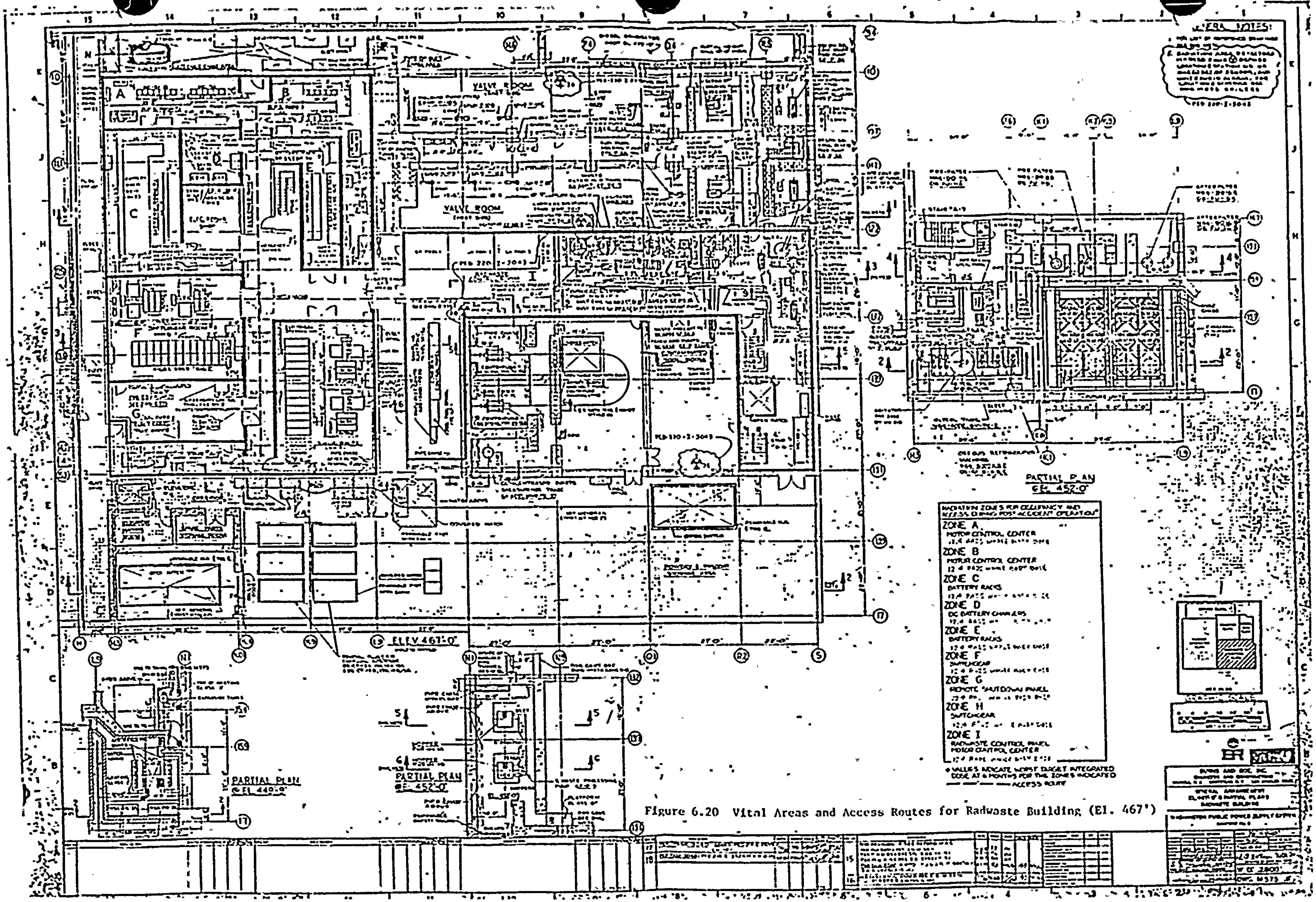


Figure 6.20 Vital Areas and Access Routes for Radwaste Building (El. 467')

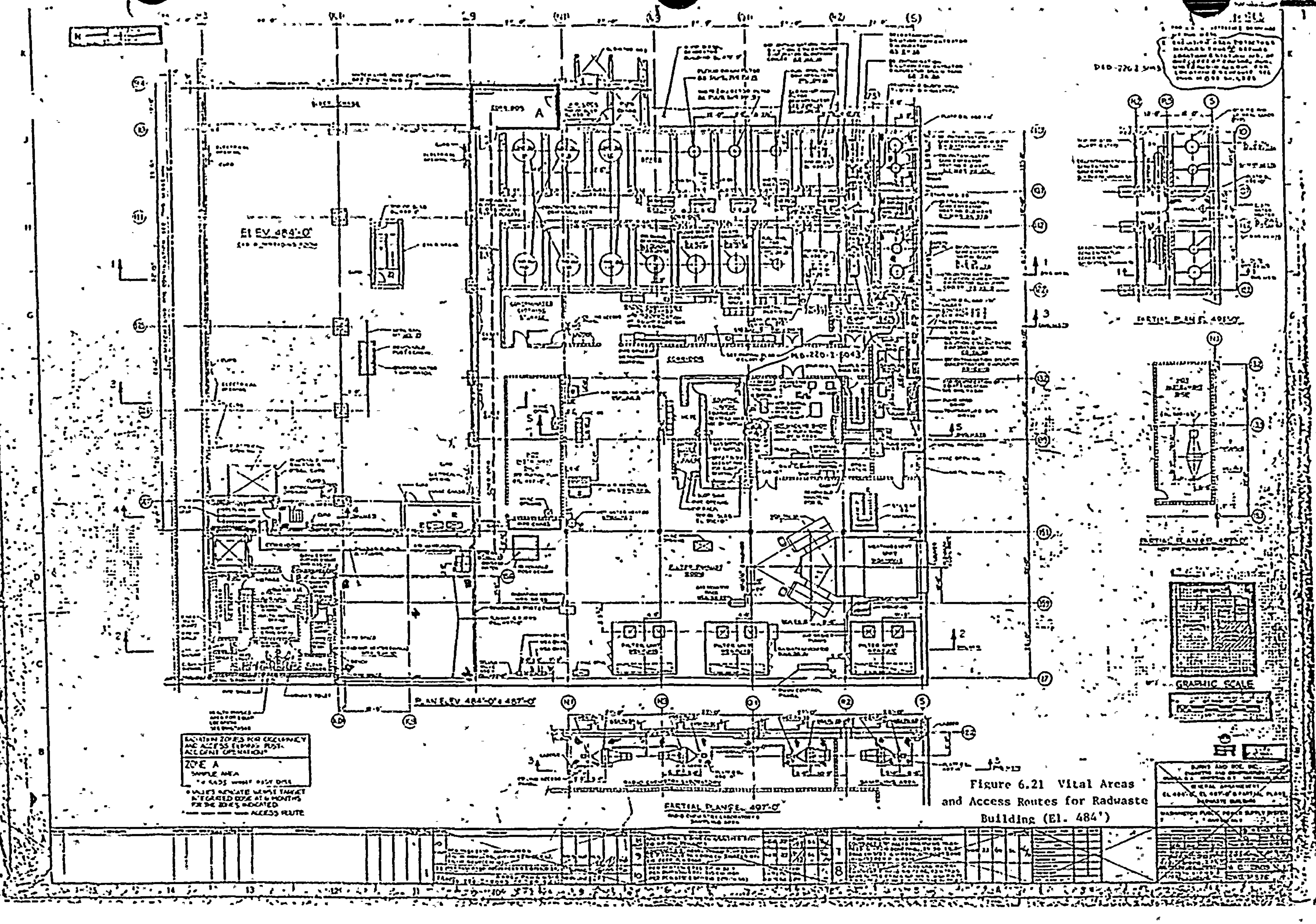


Figure 6.21 Vital Areas and Access Routes for Radwaste Building (El. 484')

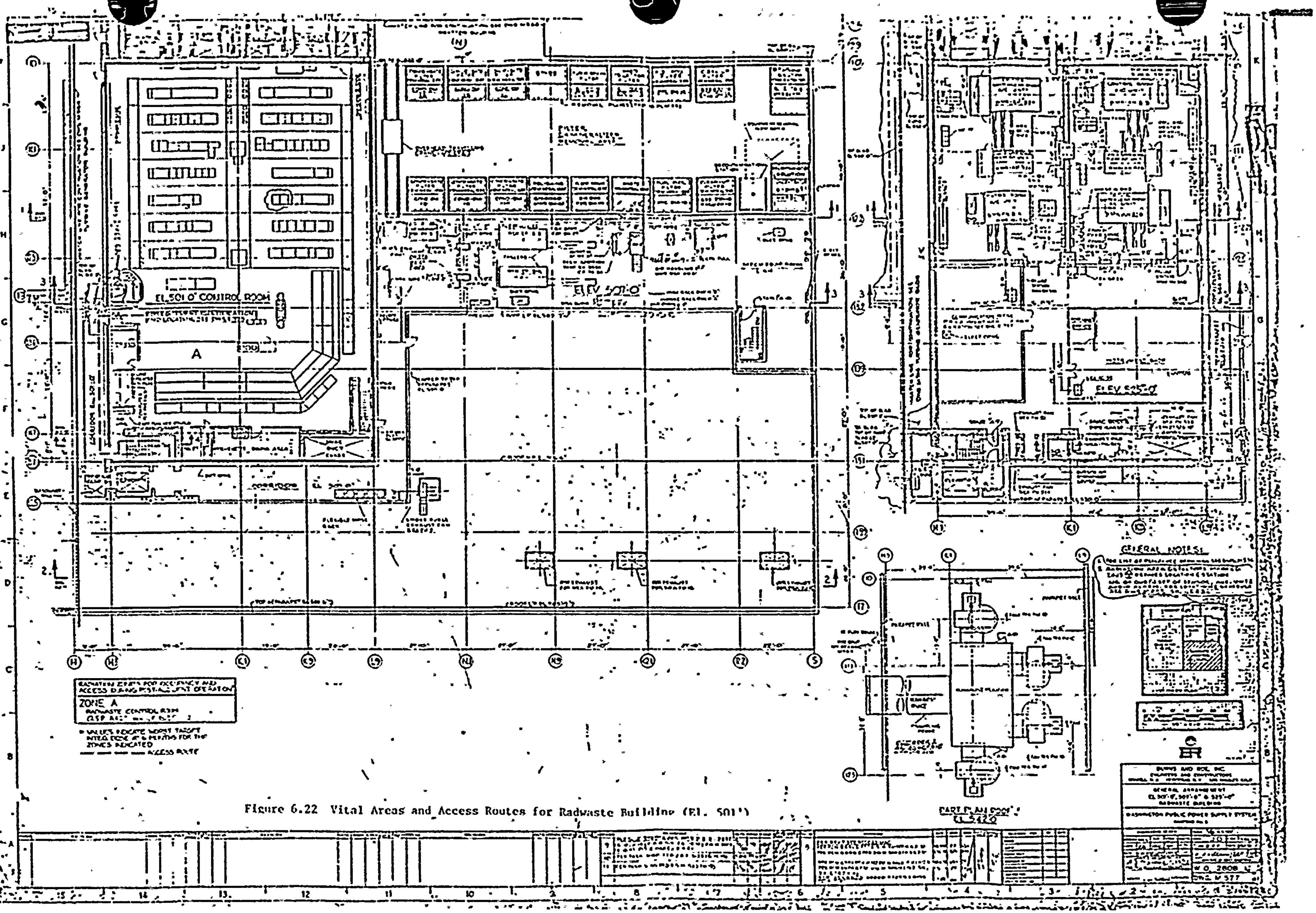


Figure 6.22 Vital Areas and Access Routes for Radwaste Building (El. 501')

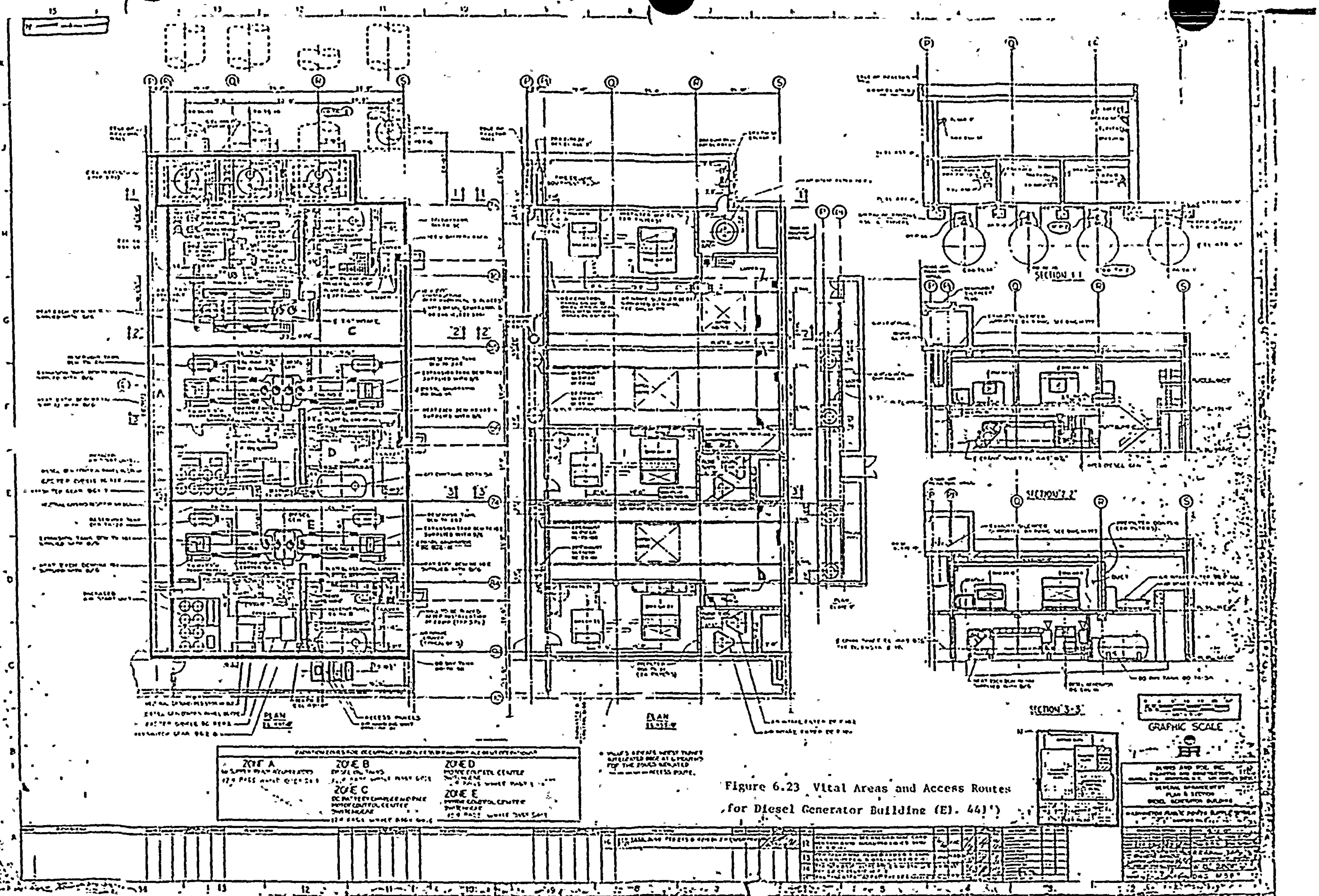


Figure 6.23 Vital Areas and Access Routes for Diesel Generator Building (El. 441')

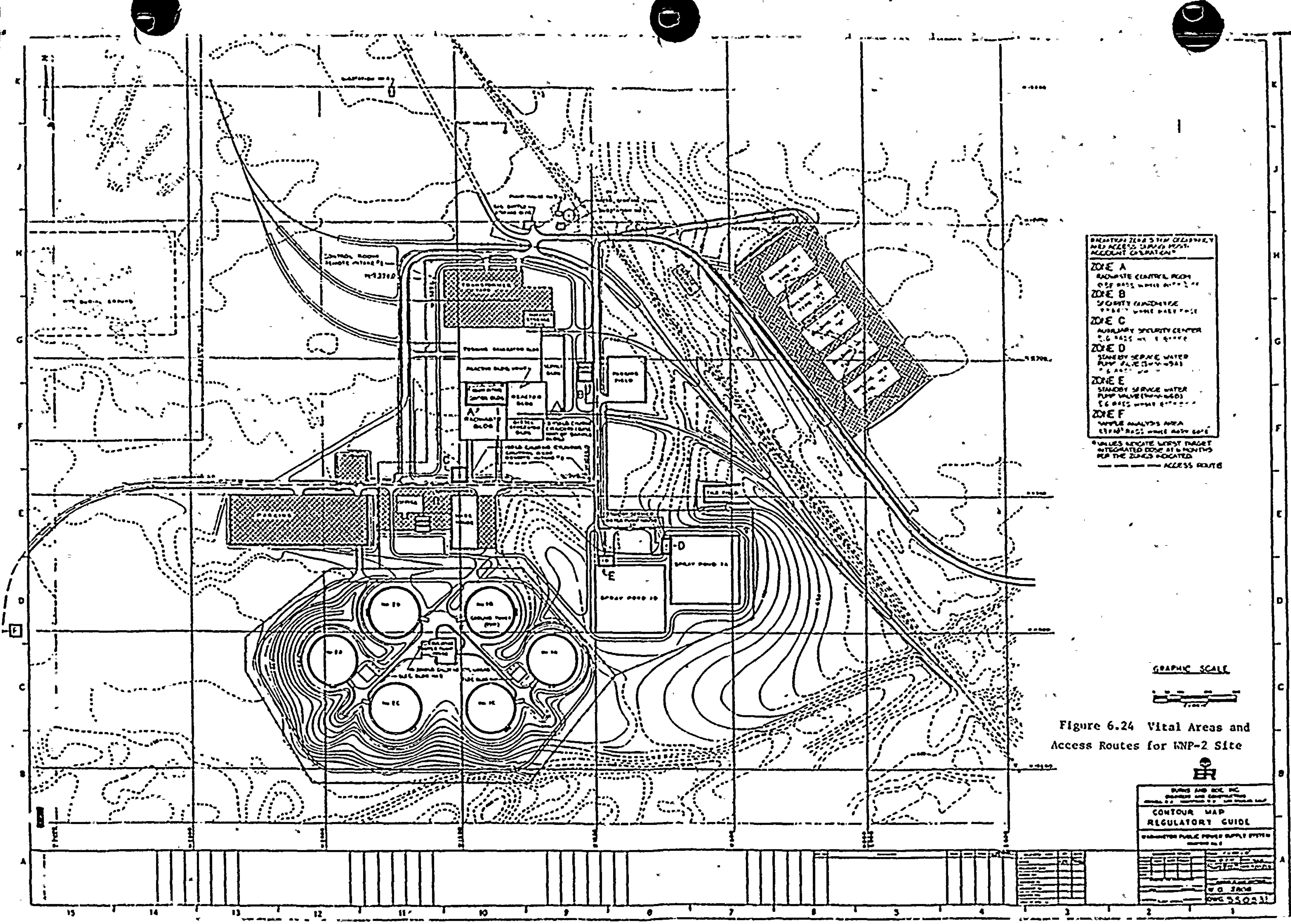


Figure 6.24 Vital Areas and Access Routes for WNP-2 Site

7.0 REFERENCES

2.1 NUREG-0578, "TMI Lessons Learned Task Force Status Report and Short-Term Recommendations".

2.2 NUREG-0588 Rev. 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment".

2.3 NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident".

2.4 Clarification letter to NUREG-0578, September 5, 1980.

2.5 NUREG-0737, "Clarification of TMI Action Plant Requirements".

2.6 IE Bulletin No. 79-01B, "Environmental Qualification of Class 1E Equipment".

2.7 Supplement No. 2 to IE Bulletin 79-01B, September 30, 1980.

5.1 ORNL, "ORIGEN2, Isotope Generation and Depletion Code - Matrix Exponential Method", ORNL Report No. CCC-371.

5.2 J.F. Perkins, U.S. Army Missile Command, Redstone Arsenal, Alabama, Report No. RR-TR-63-11 (July, 1963).

5.3 Oak Ridge National Laboratory, "Modifications of the Point-Kernel QAD-P5A", ORNL-4181, July 1968.

5.4 ORNL, "ORIGEN-79, Isotope Generation and Depletion Code - Matrix Exponential Method", ORNL Report No. CCC-217.

5.5 EDS Report No. 01-0740-1138 Revision 0, "Source Term Report", December 1980.

5.6 WPPSS, WNP-2 FSAR Section 15.6.5.5.1.1.

5.7 Regulatory Guide 1.3 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors".

5.8 WPPSS, WNP-2 FASR Sections 6.7.2 and 6.7.3.

5.9 Standard Review Plan SRP 15.6.5 "Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary".

5.10 ORNL, "QAD-CG, A Combinitorial Geometry Version of QAD-P5A, A Point-Kernal Code", ORNL Report No. CCC-307.

6.1 Letter BRWP-RO-81-288, dated July 29, 1981, "Forty-Year Integrated Dose for Radiation Zones in the Reactor Building".

6.2 Letter BRWP-RO-81-181, dated September 29, 1981, "Forty-Year Integrated Dose for Radwaste and Turbine Buildings".

B-1. The same as 5.4.

B-2. The same as 5.3.

B-3. The same as 2.2.

B-4. Letter EDSWP-81-015, dated February 18, 1981, "SGTS Filter Modeling Assumption".

B-5. Murphy and Campe "Nuclear Power Plants Control Room Ventilation System Design", 13th AEC Air Cleaning Conference.

B-6. Letter EDSWP-80-031, "Shielding Design Input Data".

B-7. Theodore Rockwell III, "Reactor Shielding Design Manual:, USAEC, TID-7004, March, 1956.

B-8. David A. Slade "Meteorology and Atomic Energy", USAEC, July, 1968.

B-9. Record of Conversation, D.A. Wert to J.A. Ogawa, dated 9/22/80, "Shielding Source Term".

C-1. The same as 2.2.

C-2. WPPSS, WNP-2 FSAR Section 15.6.5.5.1.1.

C-3. The same as 5.6.

C-4. The same as 5.7.

C-5. The same as 5.14.

C-6. The same as 5.3.

C-7. EDS Nuclear Inc. Calculation 0740-004-004, Rev. 0.

C-8. EDS Nuclear Inc. Calculation 0740-004-006, Rev. 0.

D-1. WPPSS, WNP-2 FSAR, Section 15.6.5.

D-2. Standard Review Plan, SRP-6.5.3, "Fission Product Control Systems", USNRC, June 1975.

D-3. Regulatory Guide 1.3, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" USAEC, Revision 2, June 1974.

D-4. The same as 5.3.

E-1. The same as B-6

E-2. The same as 2.2

E-3. The same as B-4

E-4. ORNL "ORIGEN-79, Isotope Generation and Depletion Code - Matrix Exponential Method", ORNL Report No. CCC-217.

H-1 David A. Slade "Meterology and Atomic Energy", USAEC, July, 1968.

APPENDIX A
Unisolated Leak Path Report

A basic assumption to the Plant Shielding Analysis is that the Reactor Building isolates such that there is no radiation leakage path to the outside. A leakage path investigation was done to verify the above assumption. While performing this investigation, the total number of lines (69) penetrating the RB boundary, the associated system components and interface systems were reviewed.

The assumption eliminating the consideration of leakage is consistent with NUREG 0737, Clarification 2. This investigation assumed that containment isolation occurred prior to the egress of highly radioactive fluid. Additionally, it assumed that all safety-related equipment was available, and that all safety systems were pressurized. Therefore, at any interface, such as a heat exchanger, no potential leakage was considered if the non-radioactive system was at a higher pressure than the radioactive system. This investigation has not considered leakage from equipment seals, closed valves or pipe rupture, except in the evaluation of the equipment and floor drain systems. The systems considered are tabulated by drawing number in Table A-1.

Table A-1

System Flow Diagrams Employed to Perform the Review

Drawing Number	Revision	Drawing Number	Revision
M501	10	M536	12
M502	17	M537	25
M503	5	M538	9
M504	25	M539	28
M505	14	M540	15
M506	23	M541	13
M507	27	M542	4
M508	25	M543	17
M509	10	M544	10
M510	30	M545	15
M511	15	M546	10
M512	8	M547	9
M513	33	M548	14
M514	13B	M549	14A
M515	17C	M550	9
M516	20	M551	8
M517	25	M552	12
M518	14	M553	10
M519	18	M554	11
M520	15	M555	7
M521	20	M556	10
M522	6	M557	4
M523	29	M607 Sheet 1	7
M524	19	M607 Sheet 2	5
M525	19	M607 Sheet 3	3
M526	25		
M527	18		
M528	15		
M529	21		
M530	18		
M531	24		
M532	20		
M533 Sheet 1	1		
M533 Sheet 2	1		
M533 Sheet 3	1		
M534	16		
M535 Sheet 1	26		
M535 Sheet 2	21		

APPENDIX B SOURCE TERM DEVELOPMENT AND PARAMETRIC STUDIES IN SECONDARY CONTAINMENT

The major tools used in the development of source terms and parametric studies inside Secondary Containment were the ORIGEN and QAD-P5A computer codes. Descriptions of the codes are in References B-1 and B-2. ORIGEN was used to compute the activities and energies of fission products released from the reactor core. The output of ORIGEN (the time-dependent energies and activity of radioactive fission products following LOCA) was used as input to calculate the airborne, shine, and direct doses for standard geometries as well as the basis of direct dose parametric studies.

B.1 Radioactive Source Terms in Secondary Containment

The ORIGEN Computer Code (Ref. B-1) was used to calculate the radioactive source terms inside Secondary Containment for liquid-containing and gas-containing systems. The fission products at the end of fuel life (maximum burnup at power level of 3481 MWt for 1000 days) were assumed to be available for release immediately following the accident. The concentrations of noble gases, halogens, and other fission products released to the gaseous and liquid sources were computed. Subsequent fission product decay and daughter product generation were then calculated for twenty time periods, covering a total period of one year.

The assumptions used in determining the initial distribution and leakage of radioactivity in the Primary Containment air and liquid space are as follows:

1. 100% of the noble gases and 25% of the halogens are distributed homogeneously

within the Primary Containment free volume immediately following the postulated accident.

2. 50% of the halogens and 1% of the remaining fission products contained in the core are mixed instantaneously and homogeneously with the Primary Containment liquid space. The Primary Containment liquid space is defined as the sum of the suppression pool liquid and the Reactor Coolant System (RCS) liquid.

3. The fission products available for release are defined as the total inventory generated in the equilibrium core after 1000 days at reactor power of 3481 MWt.

Assumptions 1 and 2 are NRC recommended assumptions for defining radioactivity release fractions for the qualification of safety-related equipment (Ref. B-3) and are detailed in Reference B-9.

Assumption 3 represents the maximum burnup level in the core prior to radioactivity release and is conservative.

Table B-1 shows the gamma ray activity concentration at selected time periods for the liquid-containing system, while Table B-2 shows the airborne gamma activity. The results of Table B-1 and B-2 were used as input in the dose parametric study, while the results of Table B-2 were used in airborne

dose calculation. Due to rapid decay of the high-energy isotopes, the average gamma ray energy for the gas-containing system varies from 0.8 Mev at the beginning of the accident to 0.3 Mev at 1 year after the accident

B.2 Airborne Dose in Secondary Containment

The time-dependent post-LOCA activity levels as calculated by the ORIGEN computer code were used as input in the calculation of the airborne beta and gamma dose rates and integrated doses inside the cubicles in the Secondary Containment. The assumptions used in this analysis are as follows:

1. Activity that leaks into the Secondary Containment is homogeneously mixed with the Secondary Containment atmosphere prior to its exhaust from the building with the Standby Gas Treatment System (SGTS).

2. The minimum SGTS flowrate of 1100 SCFM is assumed to be the flowrate of the effluent air.

3. Air that leaks out of the Primary Containment flows directly and totally into the Secondary Containment. Bypass leakage is not considered.

4. Geometric factors are used to convert the semi-infinite cloud gamma dose to a finite volume gamma dose.

5. Primary Containment activity leakage rate is 0.5%/day.

Assumption 1 is consistent with the NRC-recommended assumptions used for calculation of doses inside Primary Containment (Ref. B-3).

Assumption 2 is conservative because it represents the minimum flowrate of the SGT system (with the SGT system running and flow-balancing dampers set at the minimum flowrate) (Ref. B-4).

Assumption 3 is conservative when considering dosage in the Secondary Containment, since it maximizes the buildup of radioactivity in the Secondary Containment.

Assumption 4 is based on the assumption used in Reference B-5, and is based on an average gamma ray energy of 0.733 Mev. The effect of time dependence of average gamma ray energies has been proven to be negligible.

Assumption 5 is consistent with the assumptions established in Reference B-6.

A model of the Primary and Secondary Containment atmosphere is shown in Figure B-1. The activity concentration of a certain isotope inside the containment is changing due to the following three mechanisms:

1. Transport of activity due to air leakage.

2. Depletion of activity due to radioactive decay.

3. Increases in activity levels due to daughter product generation from radioactive decays of other isotopes.

Because activity inside the containment is assumed to be homogeneously distributed, the rate of change of radioactivity concentration due to daughter product generation and radioactive decay is independent of radioactivity transport. In other words, radioactivity would be transported at the same rate from the Primary Containment to Secondary Containment as if there were no decay. Therefore, the activity concentration inside the Primary and Secondary Containment can be expressed as:

$$C_{1i}(t) = F_{iR}(t)\ F_{1V}(t)\ C_{1i}(0) \qquad \text{(B-1)}$$

$$C_{2i}(t) = F_{iR}(t)\ F_{2V}(t)\ C_{1i}(0) \qquad \text{(B-2)}$$

where:

$F_{iR}(t)$ = Depletion factor of radioactivity concentration due to isotope decay and daughter product generation.

$F_{1V}(t)$ = Reduction factor of Primary Containment radioactivity due to transport of air through leakage and is constant for all isotopes.

$F_{2V}(t)$ = Reduction factor of Secondary Containment radioactivity due to transport of air through leakage and is constant for all isotopes.

$C_{1i}(0)$ = Airborne activity concentration in Primary Containment of a certain isotope at time (t=0).

$C_{2i}(0)$ = Airborne activity concentration in Secondary Containment of a certain isotope at time (t=0)

ORIGEN computer code calculates isotope decay and daughter product generation and is used to compute $F_{iR}(t) \cdot C_{1i}(0)$. The method of calculating $F_{1V}(t)$ and $F_{2V}(t)$ is developed as follows:

Ignoring activity decay and daughter product generation, the activity balance in Primary Containment is:

$$\frac{d}{dt}(C_{1i}V_1) = -Q_1C_{1i} \qquad \text{(B-3)}$$

Initial conditions:

$$\text{at } t = 0,\ C_{1i} = C_{1i}(0) \qquad \text{(B-4)}$$

The solution of equation (B-3) becomes :

$$C_{1i} = C_{1i}(0)\ \exp{-(Q_1/V_1)t} \qquad \text{(B-5)}$$

Total Activity balance in Secondary Containment:

$$\frac{d}{dt}(C_{2i}V_2) = Q_1C_{1i} - Q_2C_{2i} \qquad \text{(B-6)}$$

Initial conditions:

$$\text{at } t = 0,\ C_{2i} = 0 \qquad \text{(B-7)}$$

The solution to equation (B-6) becomes:

$$C_{2i} = \frac{Q_1/V_2}{(Q_2/V_2 - Q_1/V_1)} \left[\exp^{-(Q_1/V_1)t} - \exp^{-(Q_2/V_2)t} \right] C_{1i}(0) \qquad \text{(B-8)}$$

Defining:

$$F_{2V}(t) = C_{2i}/C_{1i}(0)$$

$$F_{2V}(t) = \frac{Q_1/V_2}{(Q_2/V_2 - Q_1/V_1)} \left[\exp^{-(Q_1/V_1)t} - \exp^{-(Q_2/V_2)t} \right] \qquad \text{(B-9)}$$

To calculate the airborne gamma dose rate inside the Secondary Containment, the method as described in Reference B-5 is used:

$$D_{\gamma\infty} = \sum_{i=1}^{N} 0.25\, \overline{E}_{\gamma i}\, X_{2i} \qquad \text{(B-10)}$$

$$X_{2i} = F_{2v}(t) . F_{iR}(t) . C_{1i}(0) \qquad \text{(B-11)}$$

$$D_{\gamma} = \frac{D_{\gamma\infty}}{GF} \qquad \text{(B-12)}$$

$$GF = \frac{1173}{V^{0.338}} \qquad \text{(B-13)}$$

Where:

$D_{\gamma\infty}$ = Semi infinite gamma cloud dose rate (Rads/sec)

$\bar{E}_{\gamma i}$ = Average gamma energy of the isotope (Mev)/dis

X_{2i} = Activity concentration inside Secondary Containment (Ci/m^3)

GF = Geometric factor used to scale the semi-infinite gamma cloud dose to a finite cloud dose

V = Volume of the finite cloud (ft^3)

D_γ = Gamma cloud dose rate in the center of the compartment (rad/sec)

F(t) = Dilution Factor

Q_1 = Rate of air leakage from Primary to Secondary Containment (m^3/sec)

t = Time after accident

V_1 = Primary Containment air volume (m^3)

V_2 = Secondary Containment air volume (m^3)

By taking $F_{iB}C_{1i}(0)$ from ORIGEN output and using equation (B-9) to calculate $F_{2v}(t)$, the total gamma dose in Secondary Containment can be computed by using equations (B-10) through (B-13).

The airborne semi-infinite cloud gamma dose rate is shown in Figure B-2. As can be observed from the figures, the gamma dose inside Secondary Containment reaches the peak at around three days after the accident, and decays slowly thereafter due to the decay of radioactivity by radioactive decays and removal by the SGTS exhaust system.

The geometric factor in equation (B-13) is developed in Reference B-5 for average gamma energies of 0.733 Mev. There has been a concern that this geometric factor may vary appreciably with time since the average gamma energy decreases with time due to the faster decay rate of the high energy isotopes. The average gamma energy during various time periods following the accident were computed and the results show that the average gamma energy varies from 0.3 Mev to 0.8 Mev. As discussed in Reference B-8, the geometric factor changes by less than 5% within that energy range. It is therefore concluded that the change in the geometric factor with time is negligible and that equation (B-13) can be used to calculate the finite cloud gamma dose inside the Secondary Containment.

B-3 Parametric Studies for Direct Piping Dose

The purpose of the parametric study was to identify the parameters which have a significant affect on the radiation dose rates. The computer code QAD-P5A was used to develop a correlation scheme for the significant parameters such that a simplified procedure for calculating radiation dose rates for complex source and receptor geometries can be developed. The dose rate at a target distance of 8 ft. radially outwards from the centerline

of an 8-inch Schedule 40 pipe, infinitely long (standard pipe) was first calculated and defined as the standard dose rate. A parametric study was then performed to investigate the effects of the variation of parameters such as pipe length, pipe diameter, shield thickness, and target locations on the dose rate. The results of this parametric study were then correlated as a set of correction factors to the standard dose rate. A simplified procedure was developed to calculate the dose rates and cumulate doses for complicated source-target configurations by using these correction factors.

B.3.1 Functional Dependence of Various Parameters on Secondary Containment Dose Rates

The Gamma ray energy flux from a line source "S_L" to a detector point "P" (see figure B-3) is shown in Reference B-7 as:

$$\phi = \frac{BS_L}{4\pi r} \int_0^{\theta_1} \exp^{-b_1 \operatorname{Sec}\theta} d\theta - \int_0^{\theta_2} \exp^{-b_1 \operatorname{Sec}\theta} d\theta \qquad (9)$$

where:

ϕ = uncollided gamma ray flux (photons/cm^2 - sec)

b_1 = total attenuation through shield

S_L = Source strength of line source (photons/cm sec)

B = Buildup factor

θ = Angle subtended by the length of the line source (see Figure B-3).

The source strength "S_L" is a function of the volume of liquid inside the pipe segments, which is also a function of the diameter and volume of the pipe. The angles "θ_1" and "θ_2" are also functions of "a/r" and "b/r", respectively (see Figure B-18 for definition of "a/r" and "b/r" respectively). Therefore, the functional dependence of gamma ray dose rates on the various parameters can be represented by the following equation:

$$\phi = \phi_0 \cdot F_D \cdot F_R \cdot \left[F_L\ (a/r, b_1) + F_L\ (b/r, b_1)\right] \tag{10}$$

where:

ϕ_0 = Base gamma ray flux for standard pipe

F_D = Pipe diameter Correction Factor

F_R = Radial Distance Correction Factor

F_L $(a/r, b_1)$ = Pipe Length Correction Factor

B.3.2 Parametric Study Procedures

The procedure for performing this parametric study is documented as follows:

1. Calculate the dose rate at a target distance of 8 ft. from the centerline of an 8-inch Schedule 40 pipe infinitely long (standard pipe).

2. Perform parametric studies on the variation of dose rates with:

    a. radial distance from the pipe centerline

    b. length of the pipe

    c. nominal pipe diameter

    d. time

    e. axial position along the pipe

3. Correlate the results of the parametric study by a set of geometric correction factors.

4. Develop a procedure for calculating dose rates by using the correction factors.

5. Verify the correlation scheme by calculating the dose rates at different target locations due to source piping of varied geometries through the use of QAD-P5A computer code, and compare the results to those obtained by using the procedure developed in step 4.

B.3.3 Direct Dose Parametric Study Results Inside Secondary Containment

The Standard Pipe gamma dose rate and integrated dose curves for the different systems having different source terms (source term assumptions defined in Section 5.5) are shown in Figures B-4 through B-11. The various correction factors were calculated by the following correlation.

$$F_R(r) = \frac{\text{Dose rate at a radial distance "r" from an infinitely long 8" sch 40 pipe}}{\text{Dose rate at a radial distance of 8 ft from an infinitely long 8" sch 40 pipe}}$$

$$F_L(\ell) = \frac{\text{Dose rate at a radial distance of 8 ft from an 8" sch 40 pipe of length "}2\ell\text{"}}{\text{Dose rate at a radial distance of 8 ft from an infinitely long 8" sch 40 pipe}}$$

$$F_D(d) = \frac{\text{Dose rate at a radial distance of 8 ft from an infinitely long sch 40 pipe of nominal diameter "d"}}{\text{Dose rate at a radial distance of 8 ft from an infinitely long 8" sch 40 pipe}}$$

The above mentioned correction factors for liquid system source terms are shown in Figures B-12, B-13 and B-14. The correction factor curves for gaseous source terms are shown in Figures B-15, B-16 and B-17.

B.3.4 Correction Factor Method of Determining Direct Doses in Secondary Containment

Using the parametric curves from Section B.3.3, one obtains dose rates at varied radial distances (between 2 ft to 40 ft) from varied pipe diameters (between 2 in. to 24 in.) of varied lengths (between 2 ft to infinity) at any given time period within one year. The step-by-step procedure for calculating direct dose is as follows:

a. Identify a/r, b/r parameters and obtain pipe length correction factor F_L from Figure B-13 or B-16, depending on the system being considered. (See Figure B-18 for definition of "a/r" and "b/r").

b. Obtain the standard dose rate from the standard dose rate curve for time "t" desired.

c. Obtain the pipe diameter correction factor $F_D(d)$.

d. Obtain radial distance correction factor $F_R(r)$.

e. The dose rate for the given pipe segment can be computed by:

$$\text{Dose Rate} = (\text{Standard Dose Rate}) \cdot F_R \cdot F_D \cdot F_L$$

Table B-3 compares the results for dose rate of 17 different pipe geometry and target locations as calculated using the Correction Factor Method to those calculated by using the computer code QAD-P5A. It was observed that the biggest difference in results between the two methods is less than 10%. It is concluded that the Correction Factor Method is adequate for calculating direct dose.

TABLE B-1

Gamma Energy Concentration in Liquid-Containing Systems

Mean Energy (Mev)	Gamma Energy Concentration (Mev/sec-cc)								
	0 hr	1 hr	9 hrs	24 hrs	72 hrs	720 hrs	2880 hrs	4320 hrs	8760 hrs
0.30	3.76E8	2.52E8	2.58E8	2.30E8	1.58E8	1.54E7	9.26E5	7.52E5	4.80E5
0.63	4.34E9	2.84E9	7.71E8	4.29E8	1.62E8	3.87E7	1.66E7	1.13E7	5.44E6
1.10	2.43E9	1.48E9	5.23E8	1.45E8	1.87E7	1.81E6	6.53E5	5.66E5	4.09E5
1.55	2.40E9	5.66E8	1.55E8	5.50E7	2.82E7	6.44E6	3.22E5	2.49E5	1.80E5
1.99	2.43E8	1.91E8	7.33E7	1.67E7	1.76E6	5.04E5	1.89E5	1.61E5	1.04E5
2.38	2.27E8	1.74E7	1.74E6	1.39E6	1.29E6	3.34E5	2.85E4	2.35E4	1.64E4
2.75	3.47E8	8.76E6	2.36E4	3.39E3	2.88E3	2.74E3	2.32E3	2.07E3	1.46E3
3.25	1.11E8	8.03E6	7.26E4	1.94E2	1.10E2	1.02E2	8.63E1	7.71E1	5.43E1
3.70	6.68E7	7.62E6	2.17E2	1.82E-2	1.82E-2	1.74E-2	1.54E-2	1.43E-2	1.24E-2
4.22	1.13E8	5.71E0	1.31E-2	1.31E-2	1.30E-2	1.25E-2	1.10E-2	1.03E-2	8.90E-3
4.70	2.05E8	1.63E0	6.90E-3	6.88E-3	6.88E-3	6.59E-3	5.81E-3	5.43E-3	4.69E-3
5.25	1.69E6	1.00E0	4.84E-3	4.84E-3	4.82E-3	4.63E-3	4.07E-3	3.82E-3	3.30E-3

TABLE B-2

Total Gamma Activity of the Airborne Fission Products

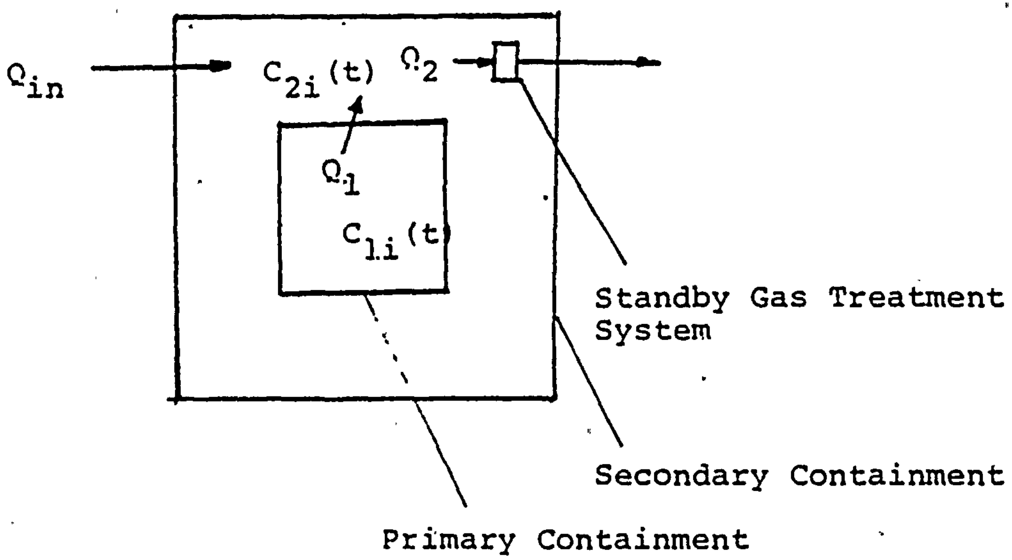
Mean Energy (mev)	Gamma Ray Activity (Photons/sec)						
	0.1 hr	1 hr	9 hrs	24 hrs	72 hrs	720 hrs	2160 hrs
0.30	6.37E18	5.13E18	4.08E18	3.25E18	2.22E18	1.20E17	9.6E14
0.63	1.62E19	9.46E18	1.96E18	9.38E17	2.49E17	9.26E15	1.83E14
1.10	3.86E18	2.69E18	8.08E17	2.09E17	1.45E16	1.04E12	4.79E10
1.55	2.42E18	1.81E18	1.70E17	2.97E16	4.36E14	1.20E13	4.65E11
1.99	1.14E18	8.25E17	1.53E17	1.48E16	8.85E13	negl	negl
2.38	1.04E18	9.43E17	7.75E16	1.89E15	1.12E12	negl	negl
2.75	5.31E17	3.02E17	8.79E15	1.35E14	9.28E8	negl	negl
3.25	3.44E16	7.08E15	4.09E11	1.74E8	negl	negl	negl
3.70	2.65E16	4.56E15	1.03E11	3.11E2	negl	negl	negl
4.22	4.50E16	1.11E11	negl	negl	negl	negl	negl
4.70	8.57E15	1.92E10	negl	negl	negl	negl	negl
5.25	6.28E15	1.56E10	negl	negl	negl	negl	negl

negl = negligible

TABLE B-3

Comparison of Direct Dose Rate Results

Target Location and Pipe Geometry					Dose Rate Results			
Pipe Diameter (cm)	Pipe Length (cm)	Target Location r (cm)	a (cm)	b (cm)	Time After Accident (hrs)	Correction Factor Method (Rads/hr)	Computer Result (Rads/hr)	Difference (%)
6	800	548.6	570	230	24	52.7	53.3	-1.1
6	800	91.4	720	80	24	484	479	+1.0
6	800	1066.8	650	150	24	15.3	16.1	+5.0
8	800	548.6	570	230	24	77.0	80.4	-4.23
8	800	391.4	720	80	24	105.0	110.0	-4.5
8	800	1066.8	650	150	24	22.4	24.4	-8.2
2	700	100.0	600	100	720	5.36	5.32	0.75
2	700	1066.8	600	100	720	0.159	0.146	8.9
2	700	100	-900	1600	720	0.0126	0.0123	2.3
2	700	1066.8	-200	900	720	0.128	0.124	3.2
12	400	1066.8	-400	800	720	1.14	1.21	-5.8
12	400	100	350	50	720	72.5	71.4	1.5
12	400	609.6	350	50	720	4.66	4.93	-5.5
12	400	1066.8	350	50	720	1.57	1.73	-9.3
10	600	304.8	-243.8	548.6	0.0333	554	539	2.7
10	600	121.9	450	150	0.0333	7617	7396	-3.0
10	600	1005.8	450	150	0.0333	258	280	-7.9



Q_2 = Air leakage rate from Secondary Containment (m^3/sec)

Q_1 = Air in-leakage rate from Primary Containment (m^3/sec)

V_2 = Volume of Secondary Containment (m^3)

V_1 = Volume of Primary Containment (m^3)

i = Nuclide index

$C_{2i}(t)$ = Activity concentration in Secondary Containment (Ci/m^3)

$C_{1i}(t)$ = Activity concentration in Primary Containment (Ci/m^3)

Q_{in} = Clean air in-leakage rate. (m^3/sec)

Figure B-1 Model of the Primary and Secondary Containment

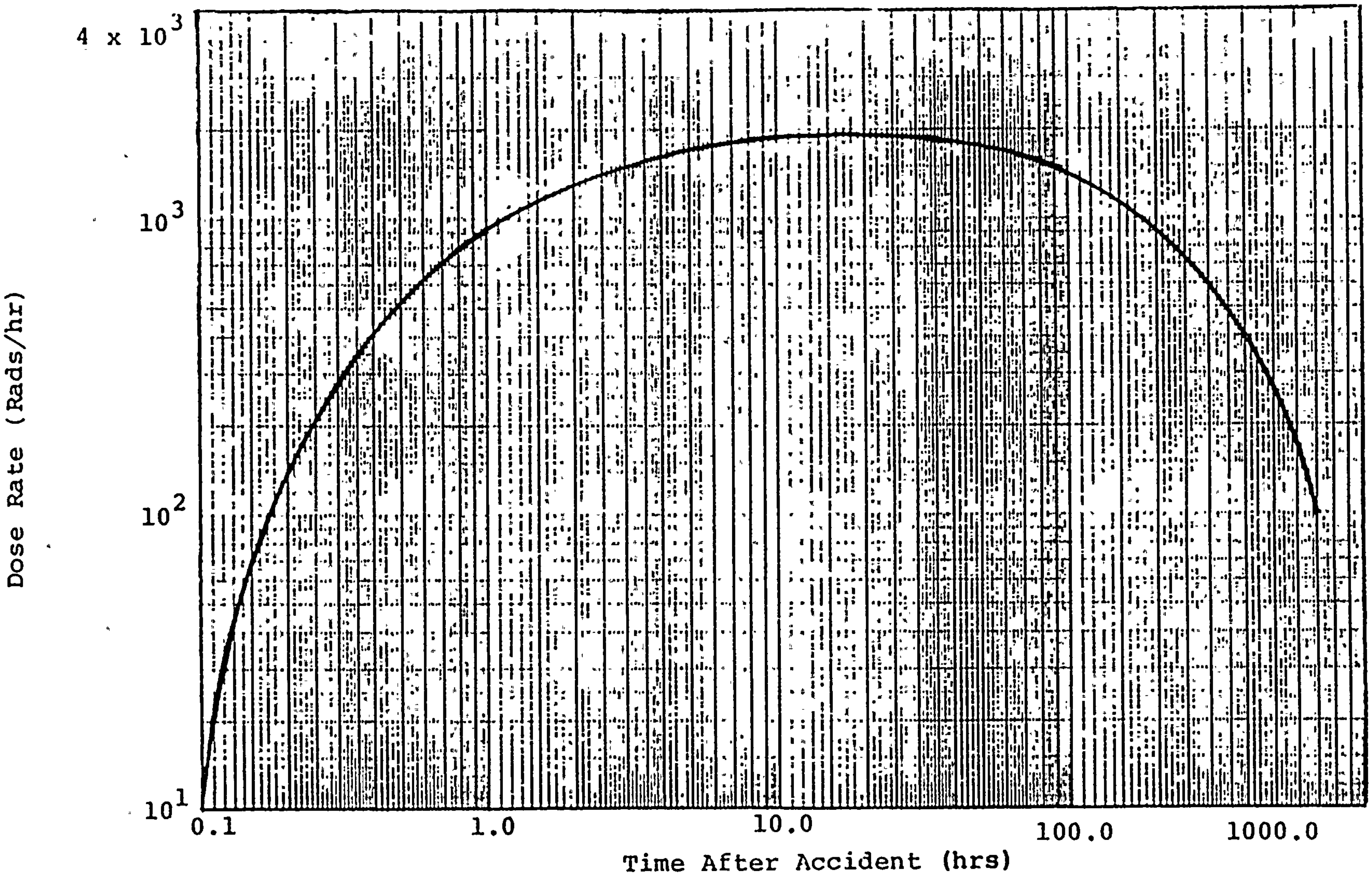
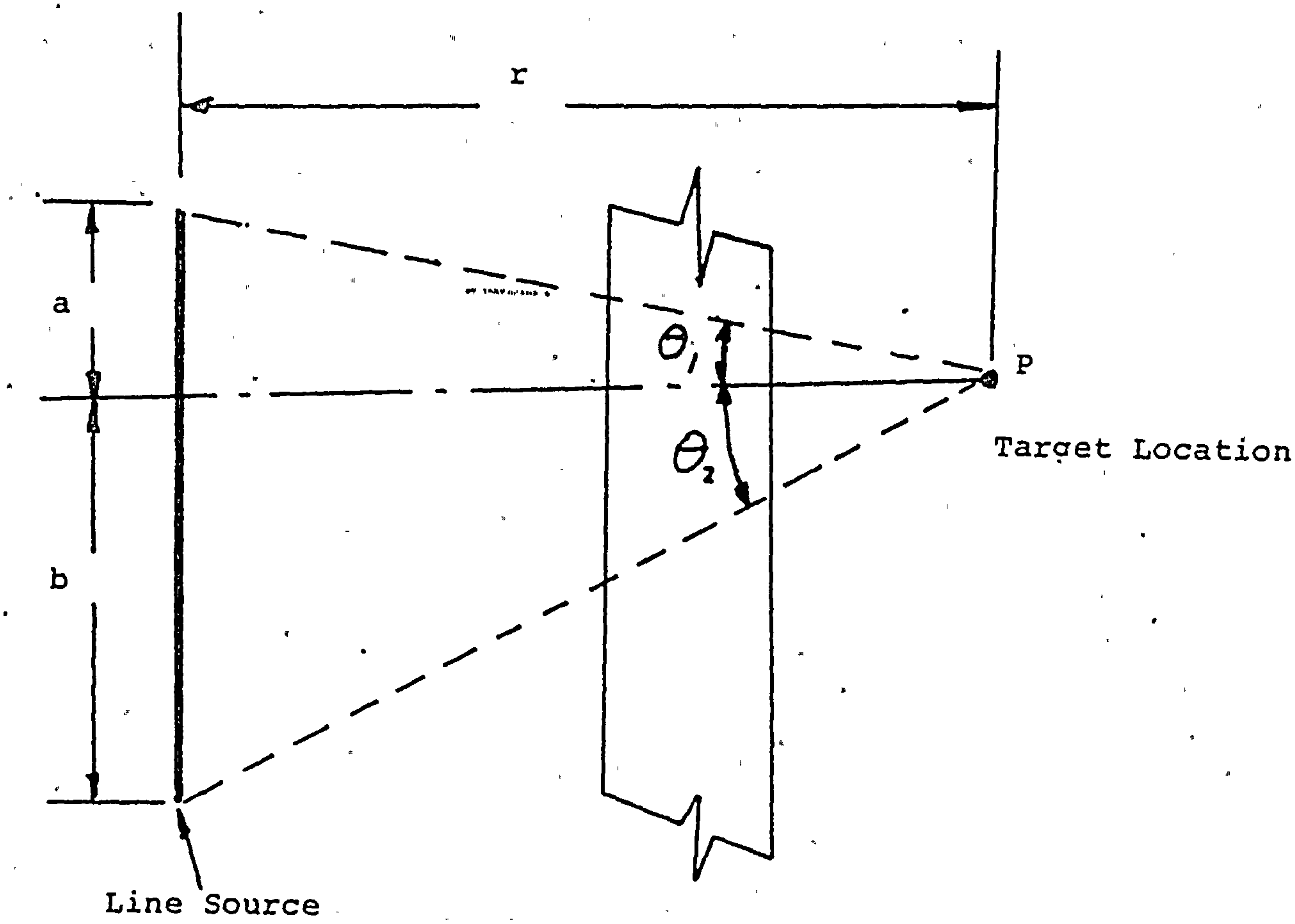


Figure B-2 Time Dependent Gamma Dose Rate For A Semi-Infinite Cloud of Fission Products At Secondary Containment Concentrations (0.5%/Day Primary Containment Leakage)



$$b_i = \sum_i \mu_i t_i$$

Figure B-3 Illustration of Parameters Used in the Shielding Equation

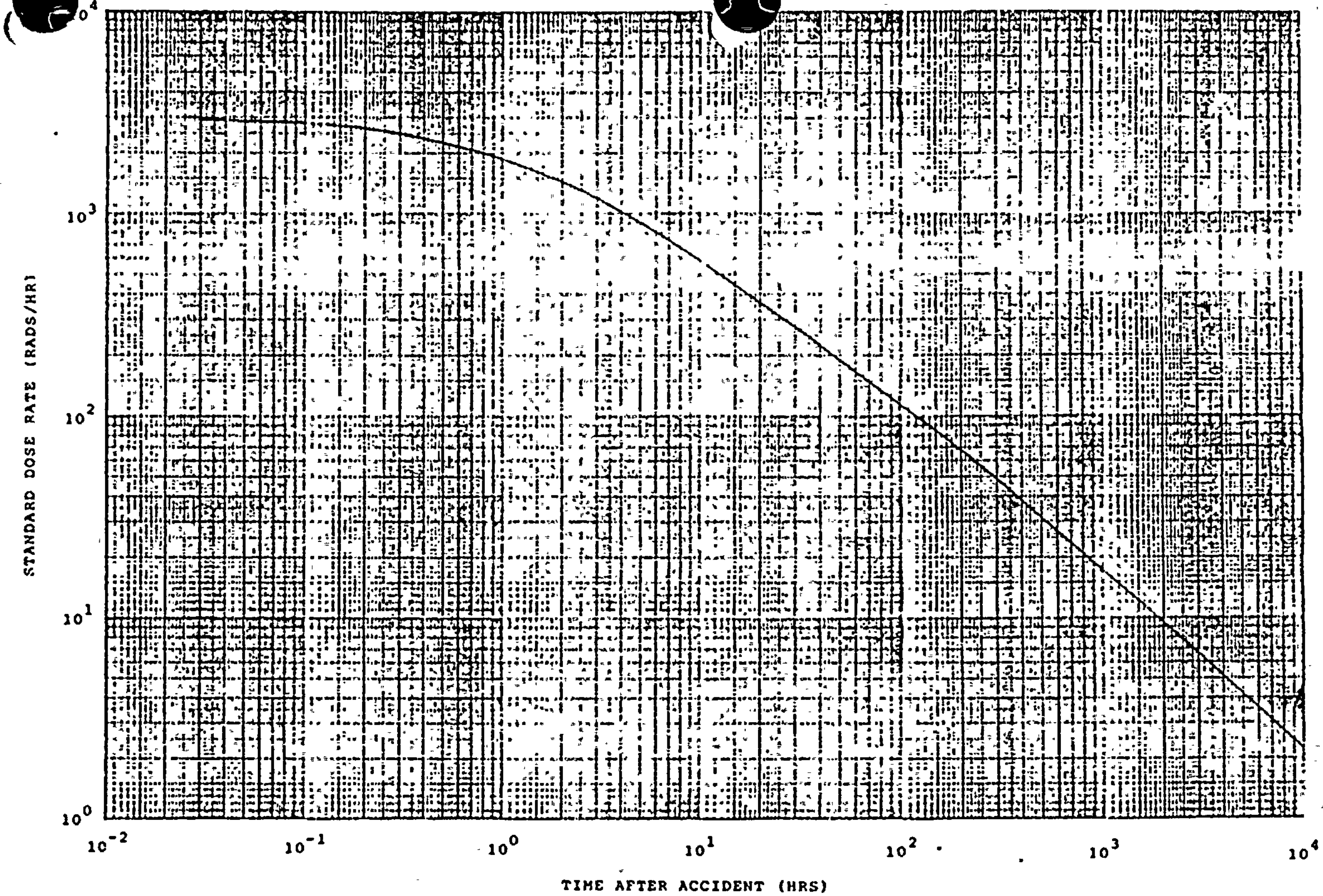


Figure B-4 Standard Gamma Dose Rate Curve for Liquid Containing Systems (RCIC Liquid System And RHR System)

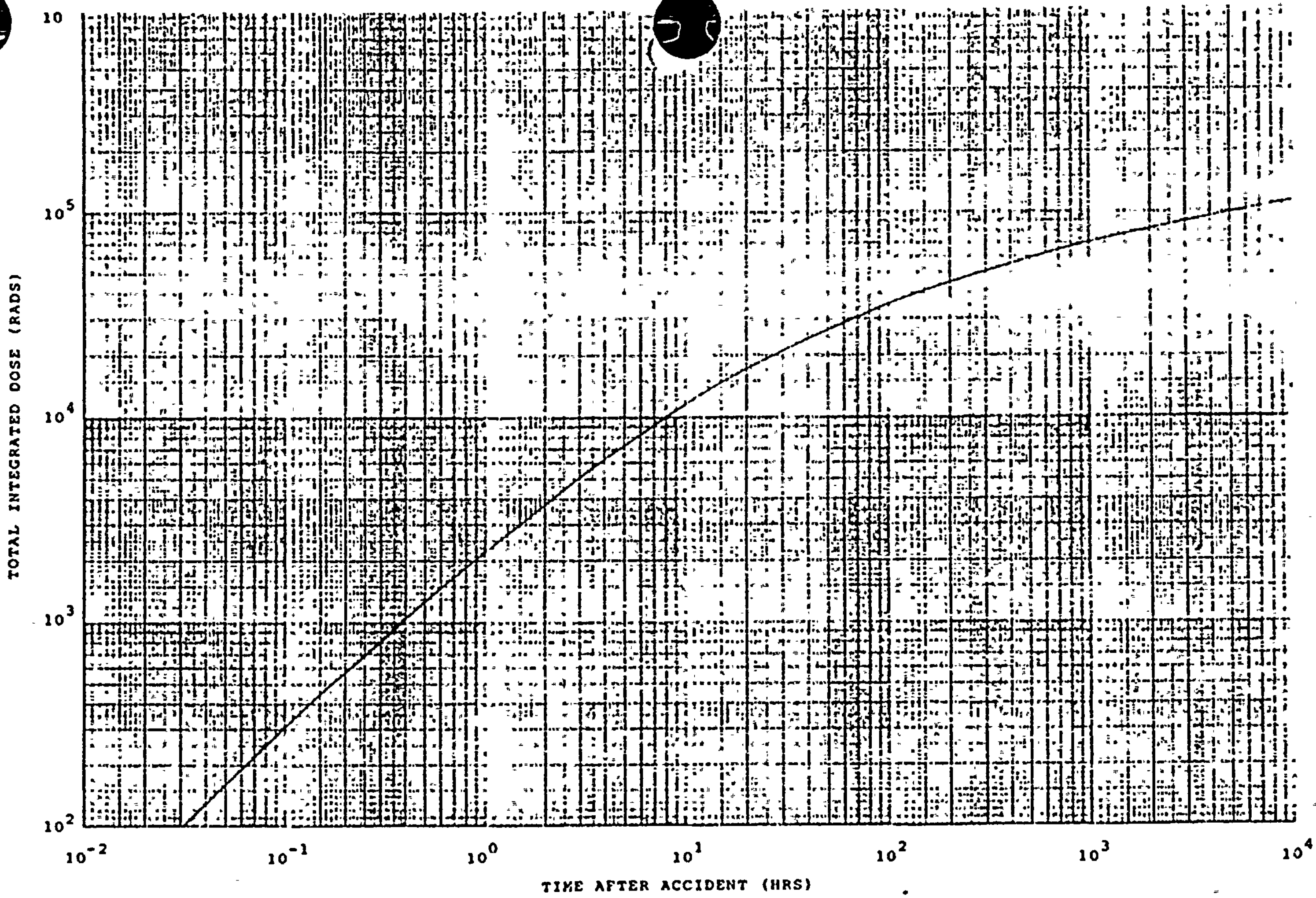


Figure B-5 Standard Integrated Gamma Dose Curve for Pipes in Liquid Containing Systems (RCIC Liquid System and RHR Systems)

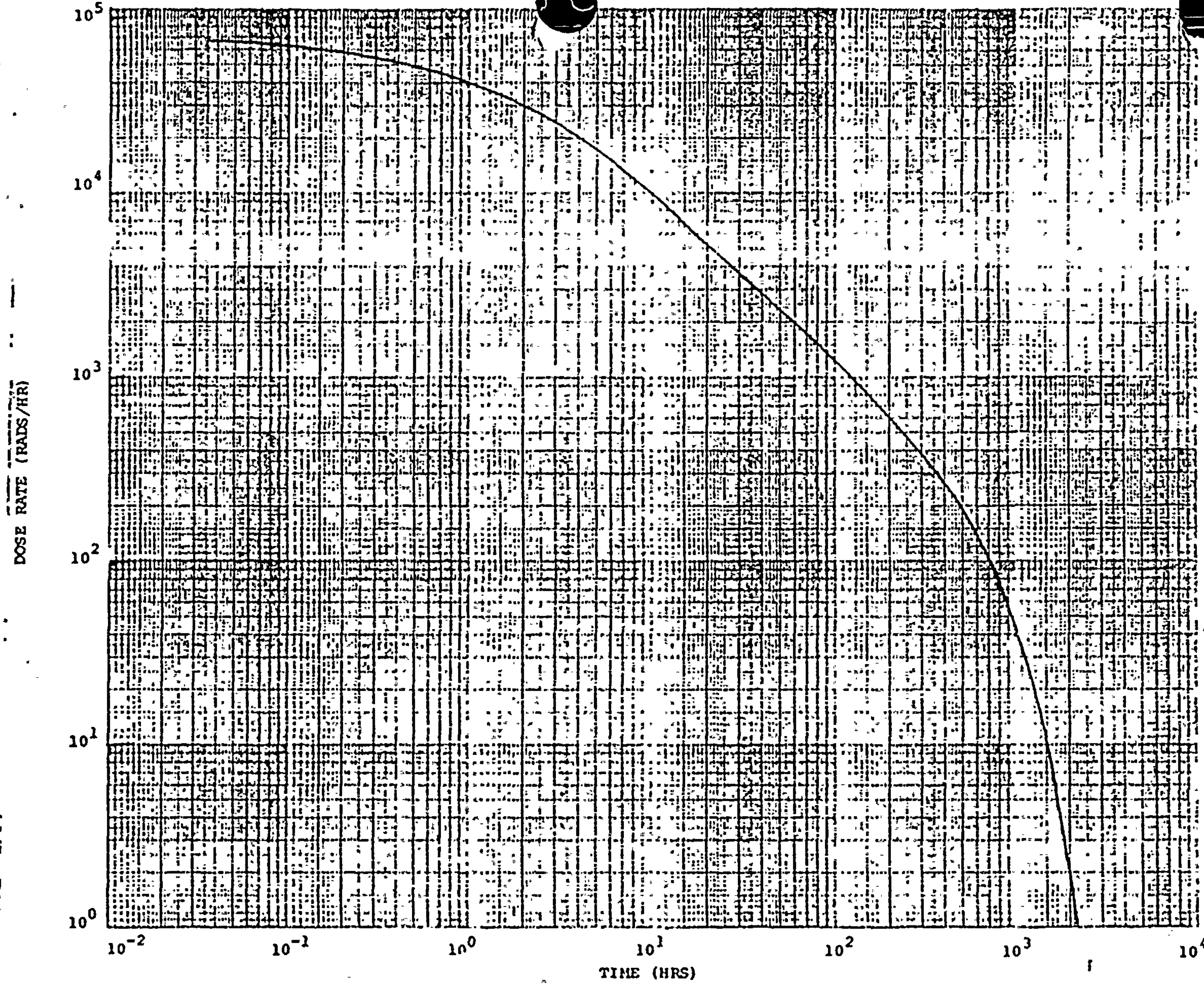


Figure B-6 Standard Gamma Dose Rate Curve For Pipes In The RCIC Steam System And MSIVLC Steam System Before The Header

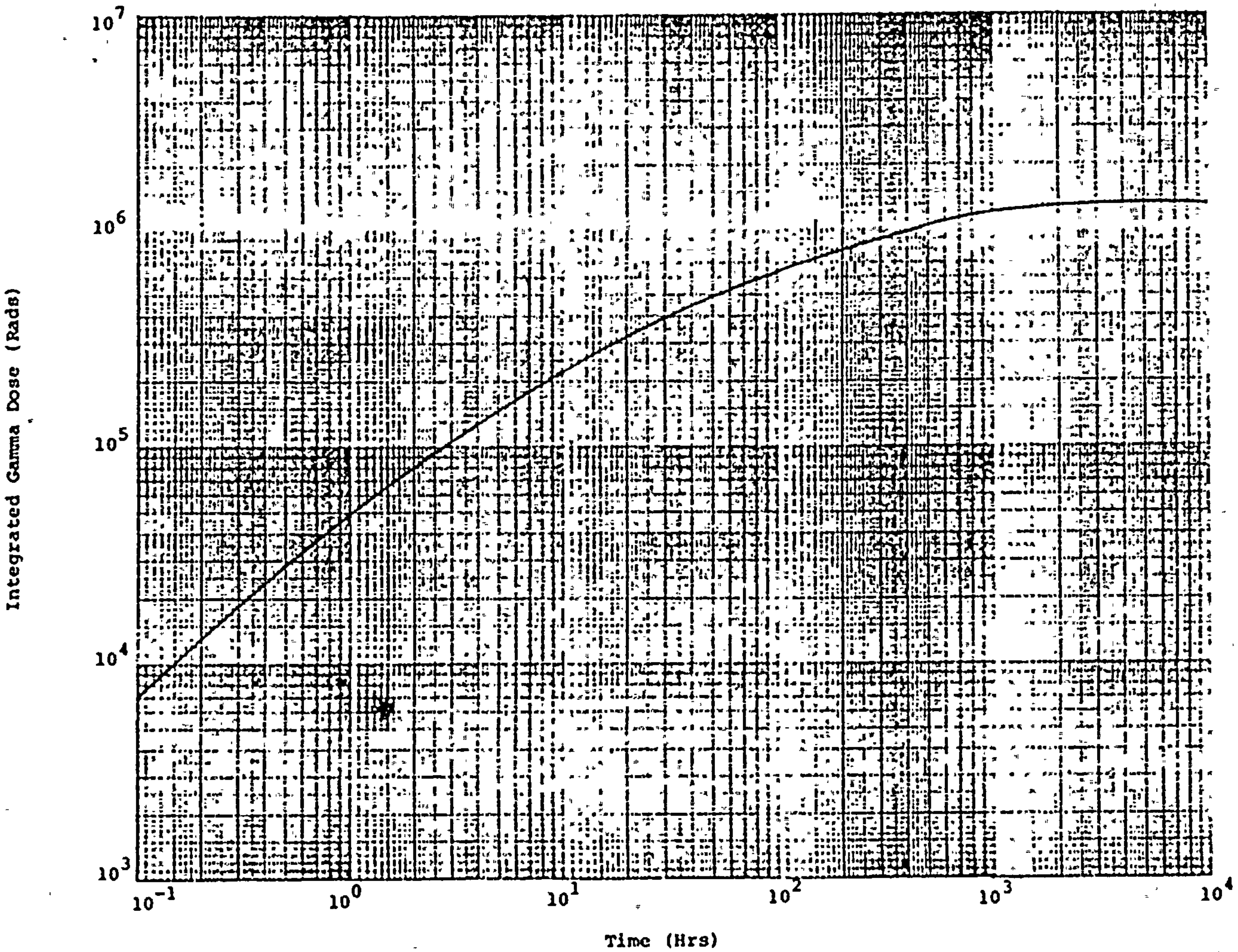


Figure B-7 Standard Integrated Gamma Dose Curve For Pipes In The RCIC Steam System And MSIVLC Steam System Before The Header

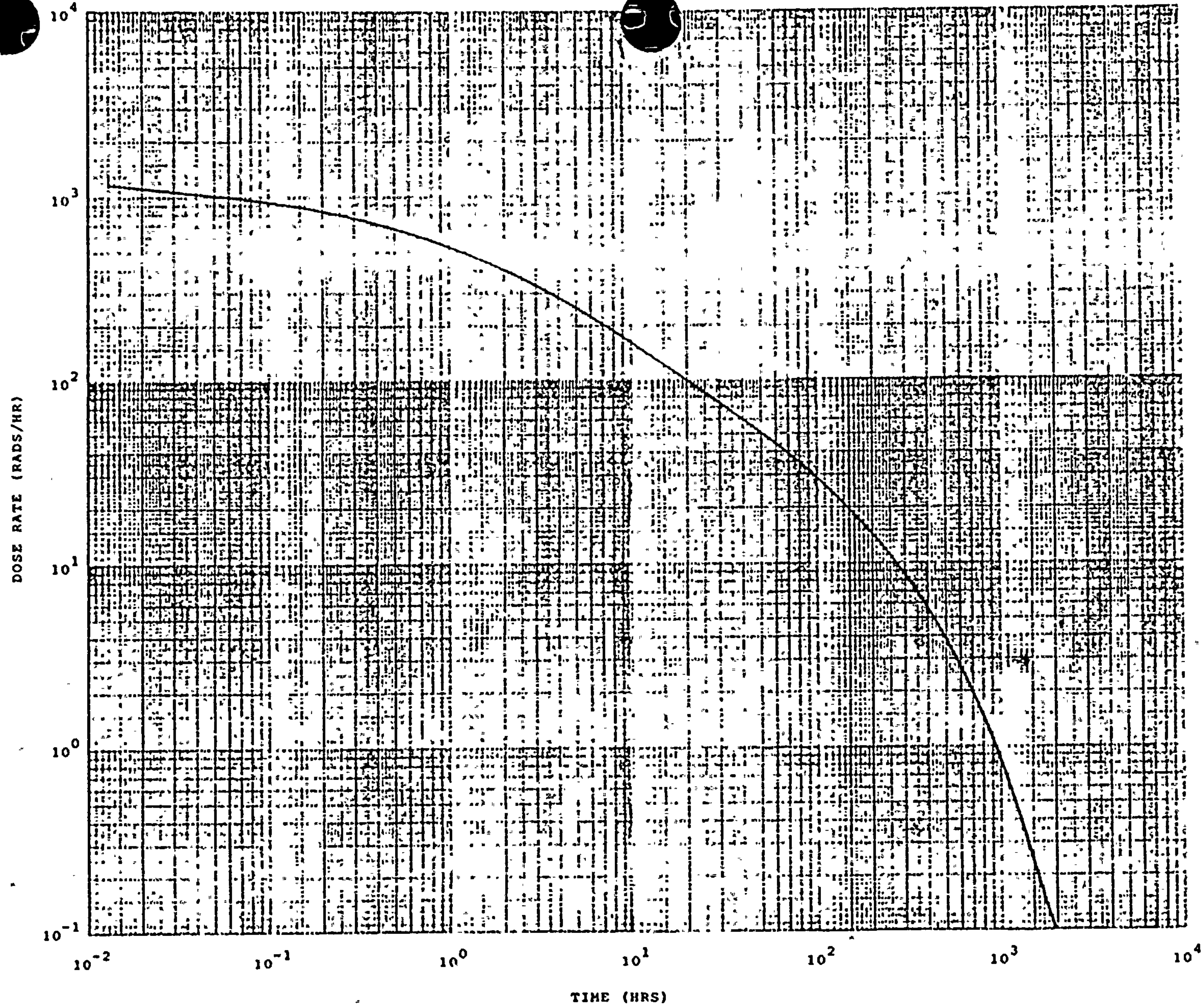


Figure B-8 Standard Gamma Dose Rate Curve For Pipes In The MSIVLC Steam System After The Header

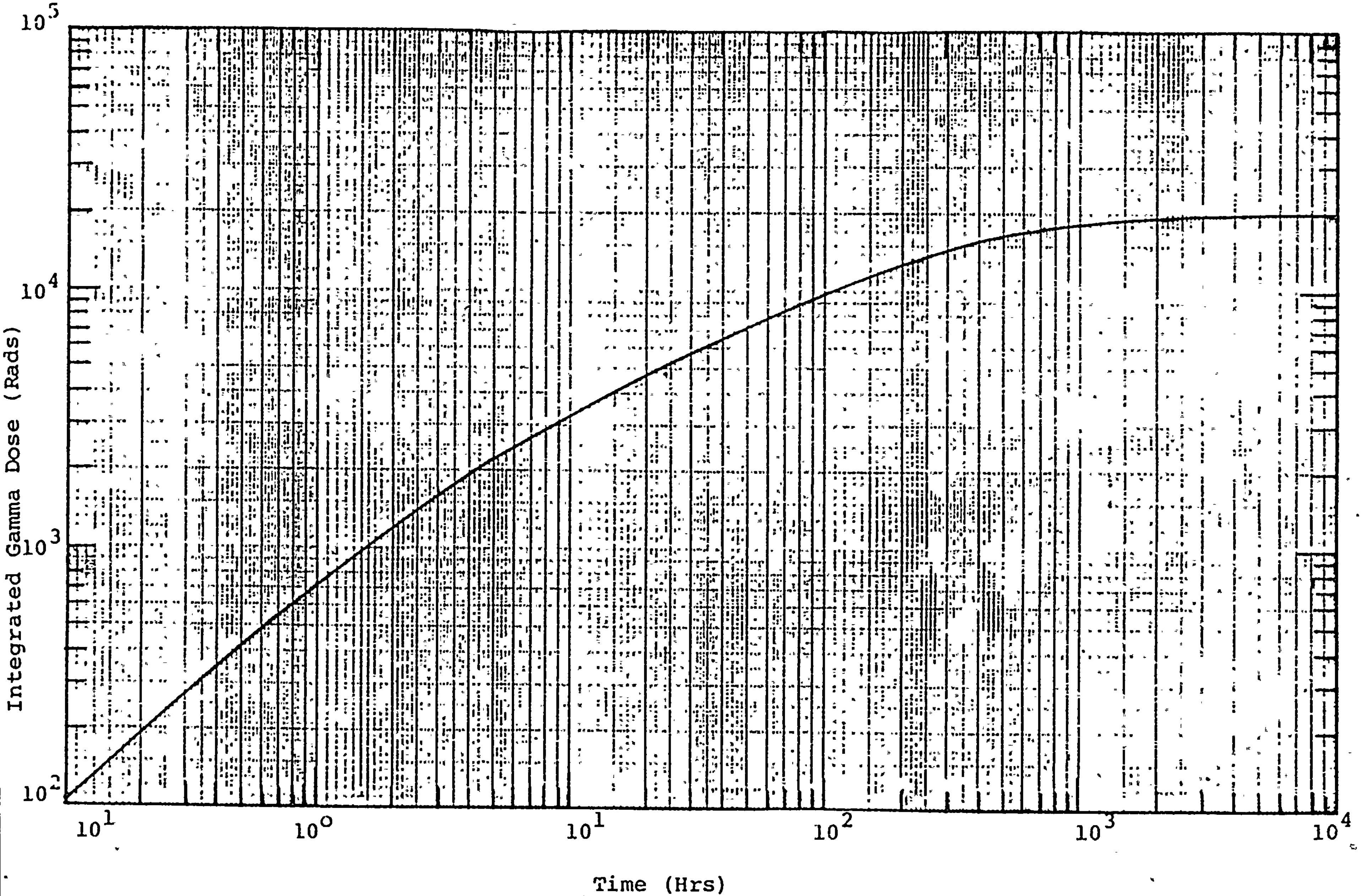


Figure B-9 . Standard Integrated Gamma Dose Curve For Pipes In The MSIVLC Steam System After The Header

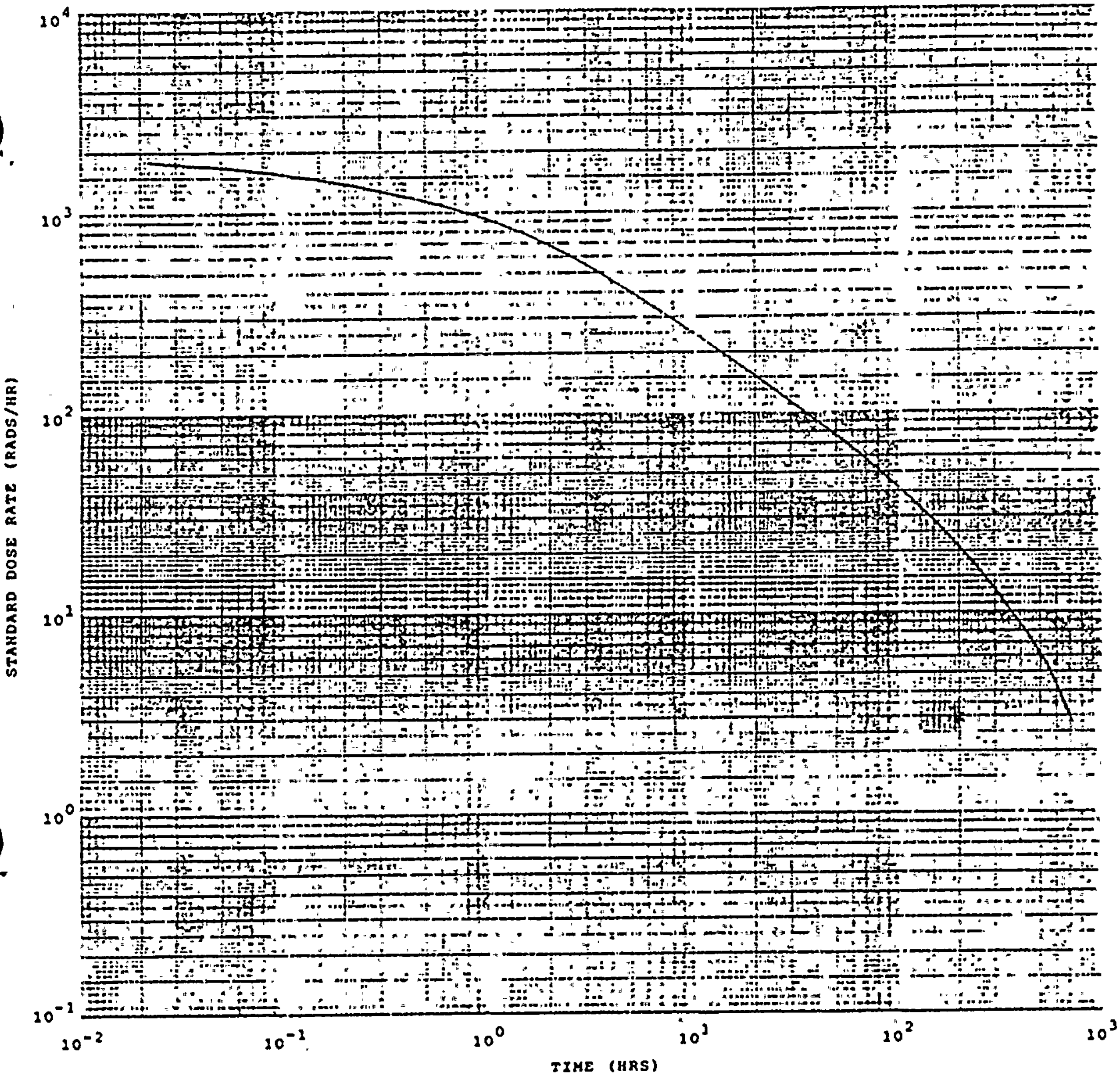


Figure B-10 Standard Gamma Dose Rate Curve For CAC System Gas Lines

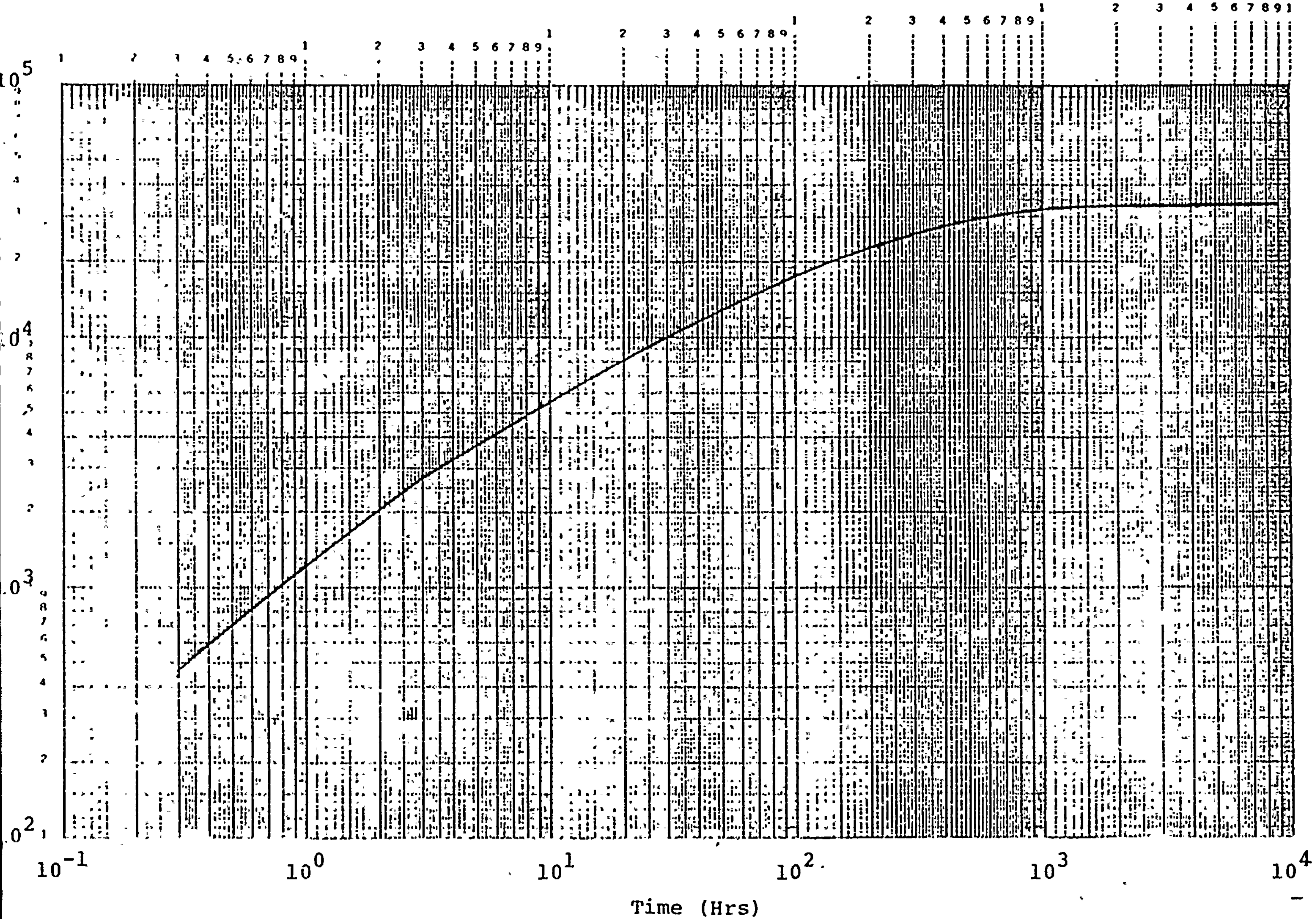


Figure B-11 Standard Integrated Gamma Dose Curve for CAC System Gas Lines

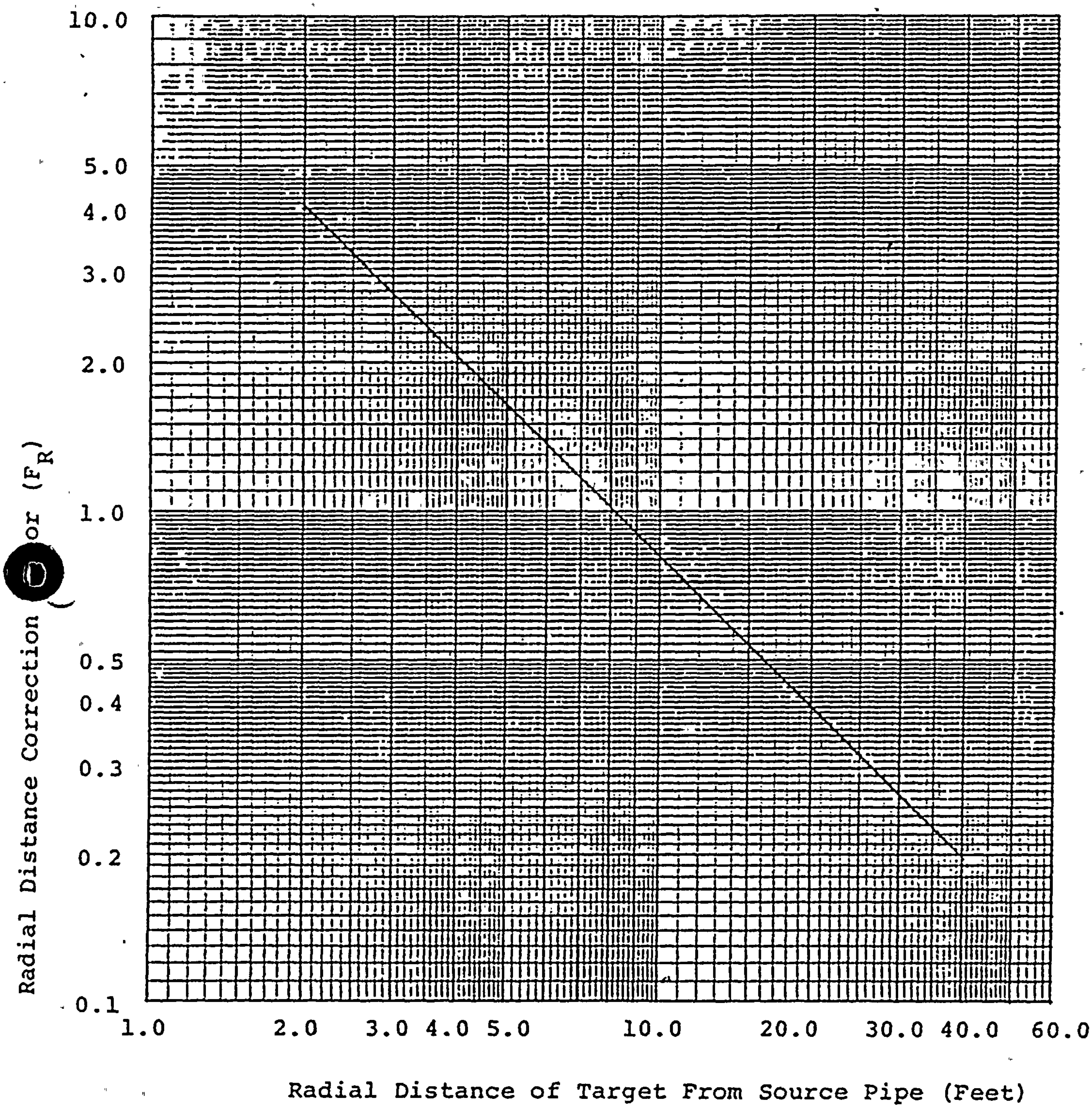


Figure B-12 Radial Distance Correction Factor For Liquid Sources

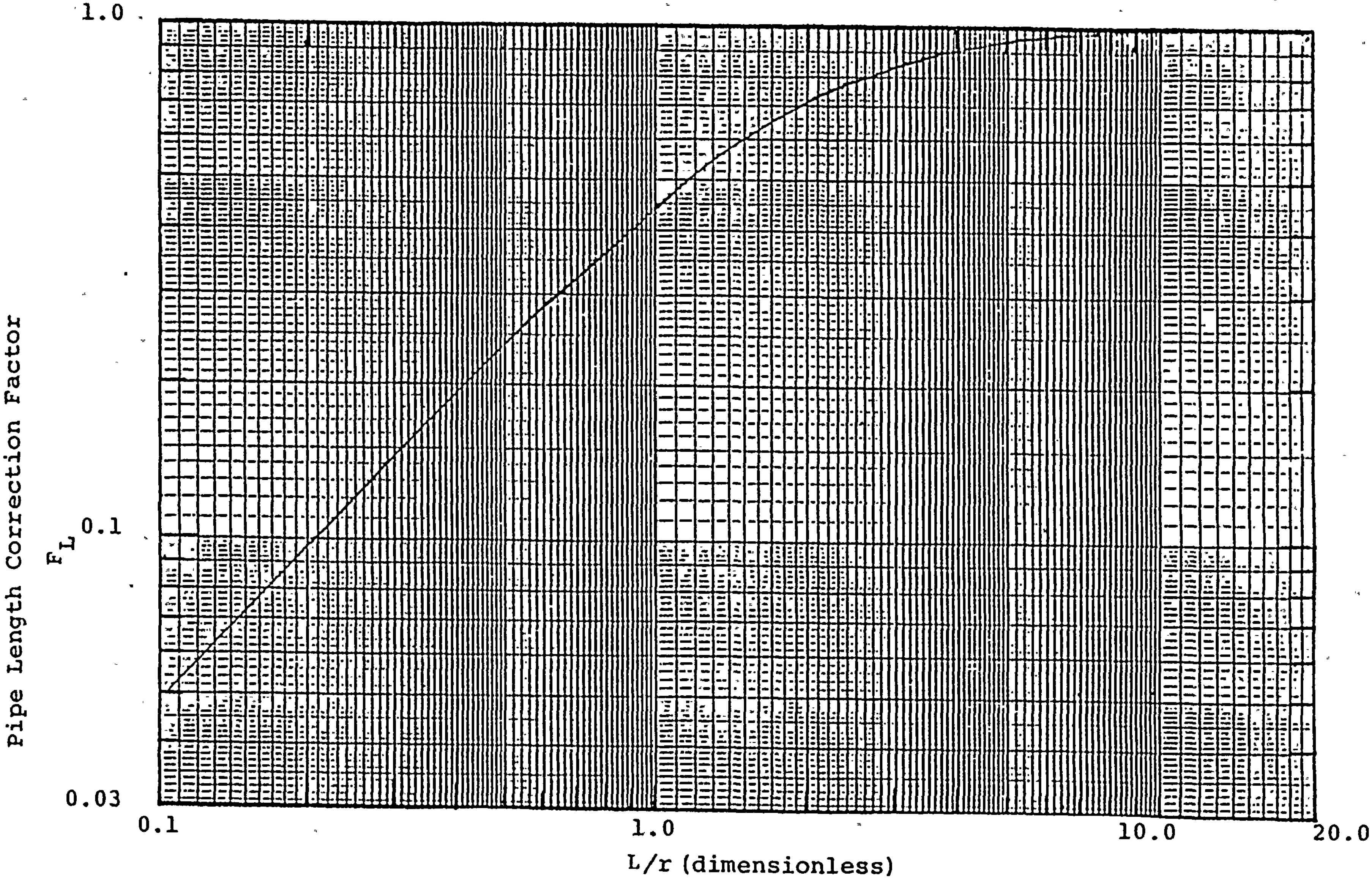


Figure B-13 Pipe Length Correction Factor For Liquid Sources

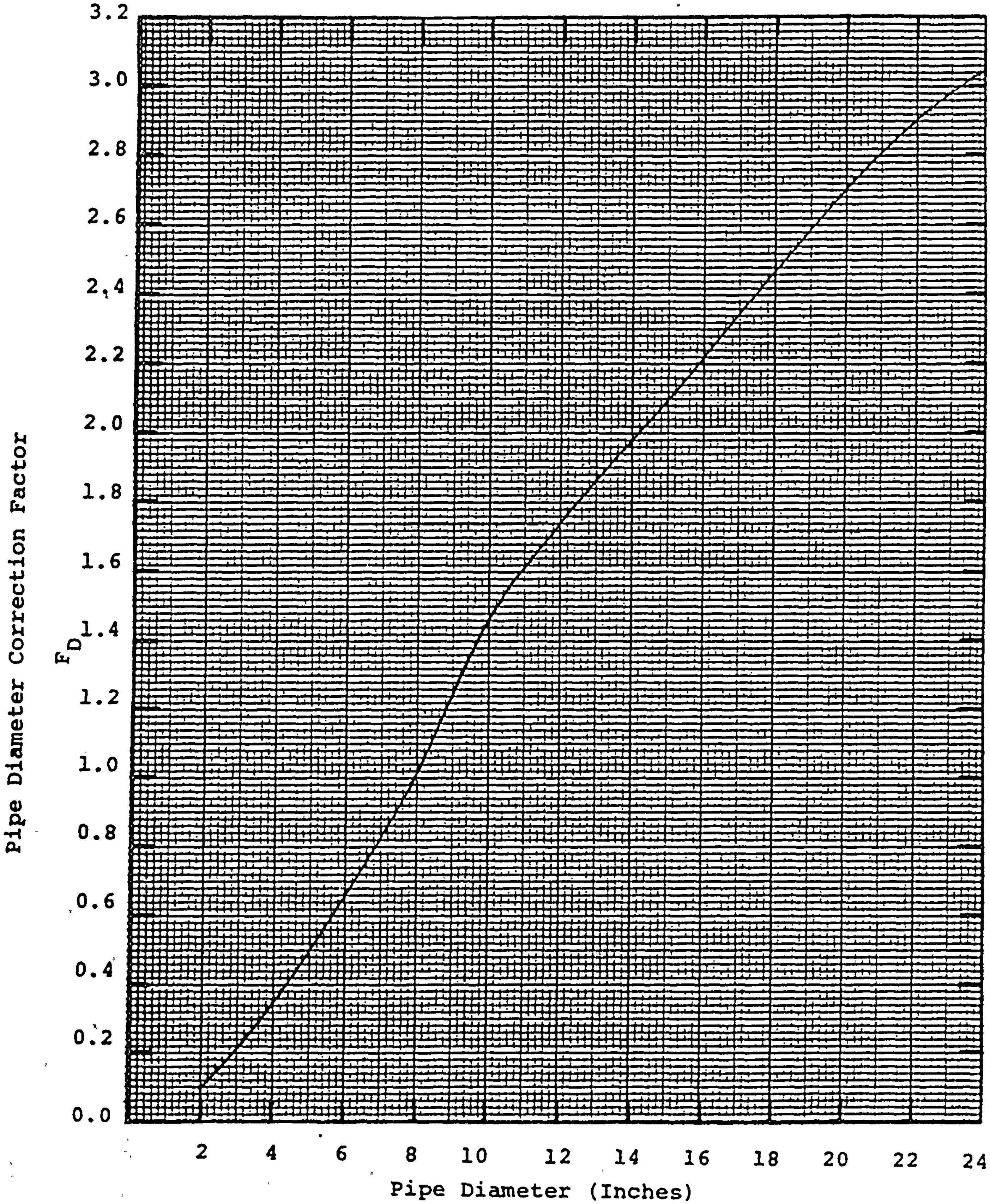


Figure B-14 Pipe Diameter Correction Factor For Liquid Sources

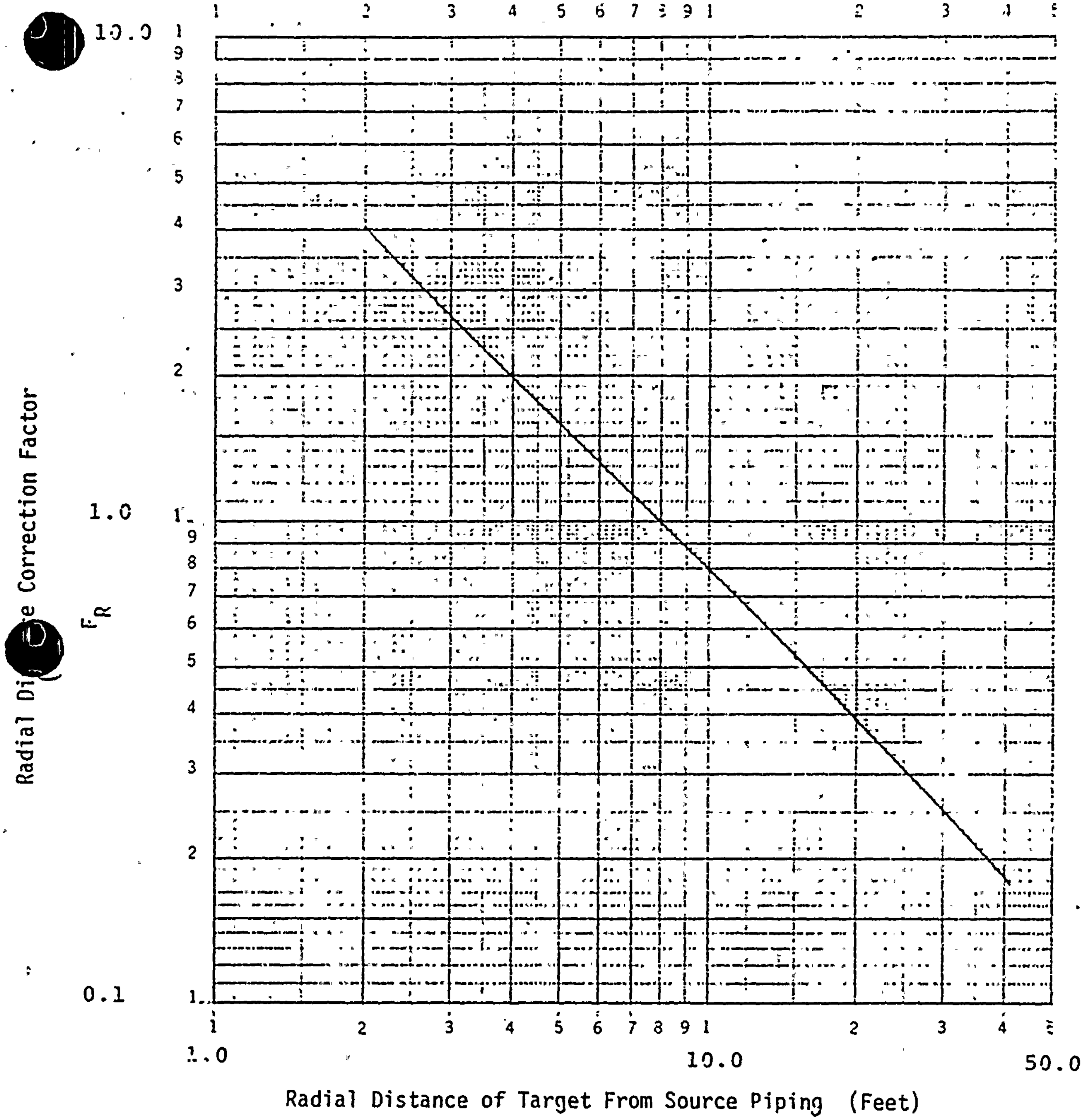


Figure B-15 Radial Distance Correction Factor For Gaseous Sources

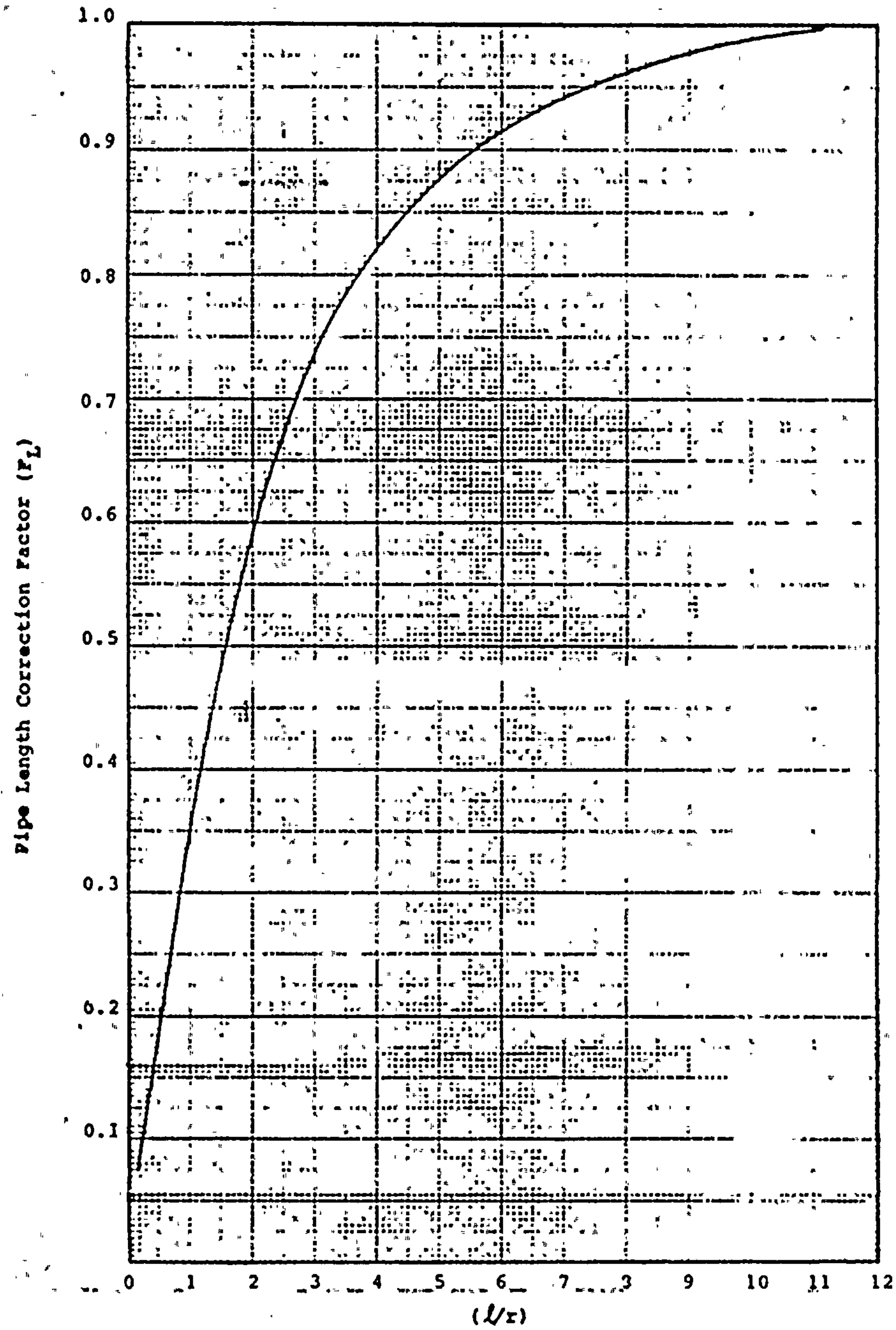


Figure B-16 **Pipe Length Correction Factor For Gaseous Sources**

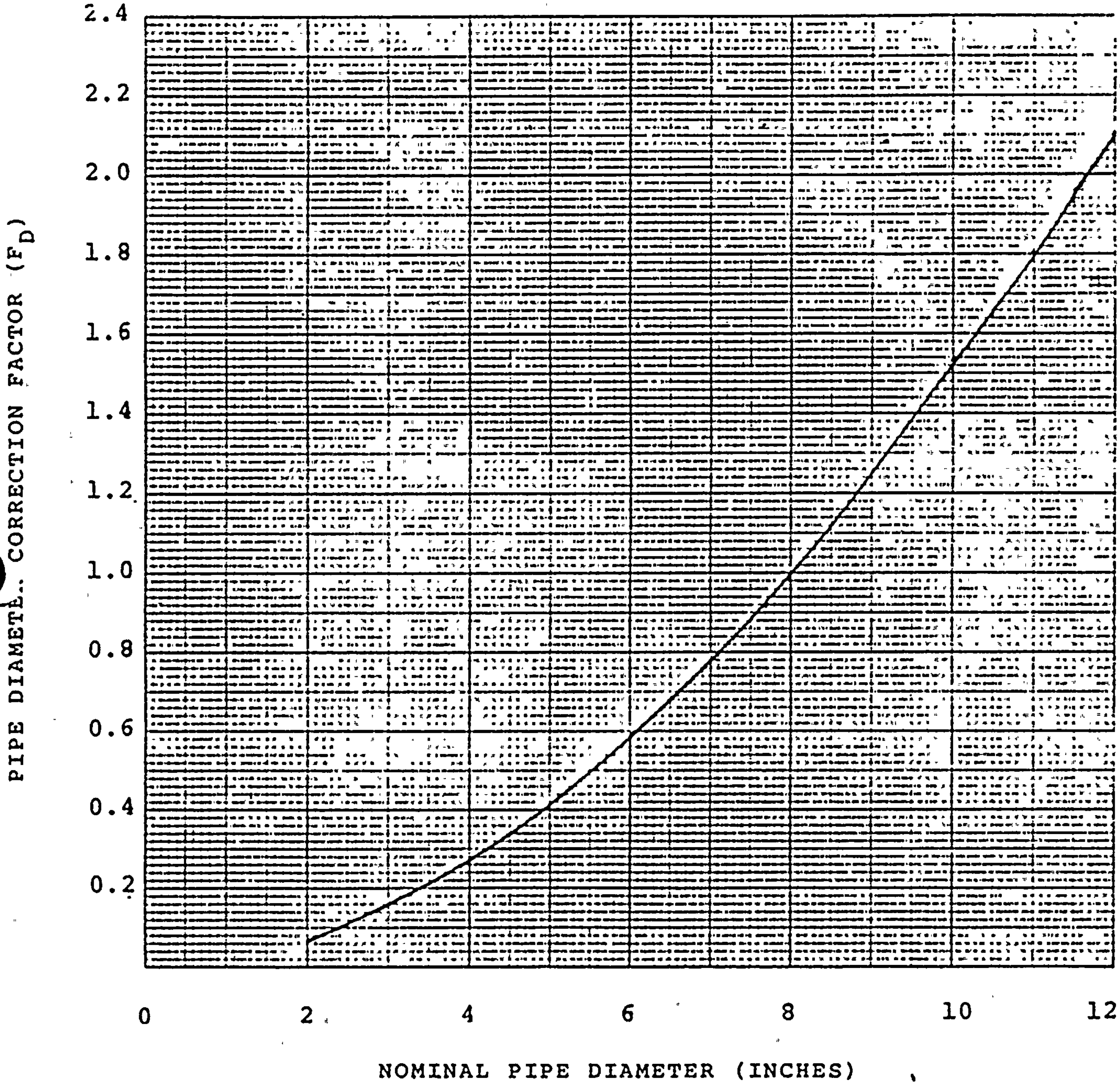
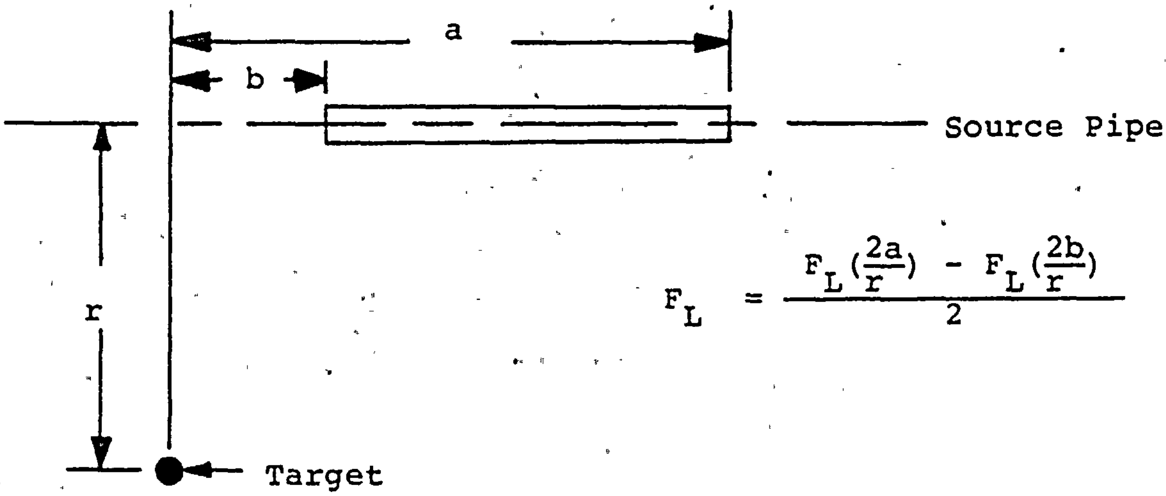


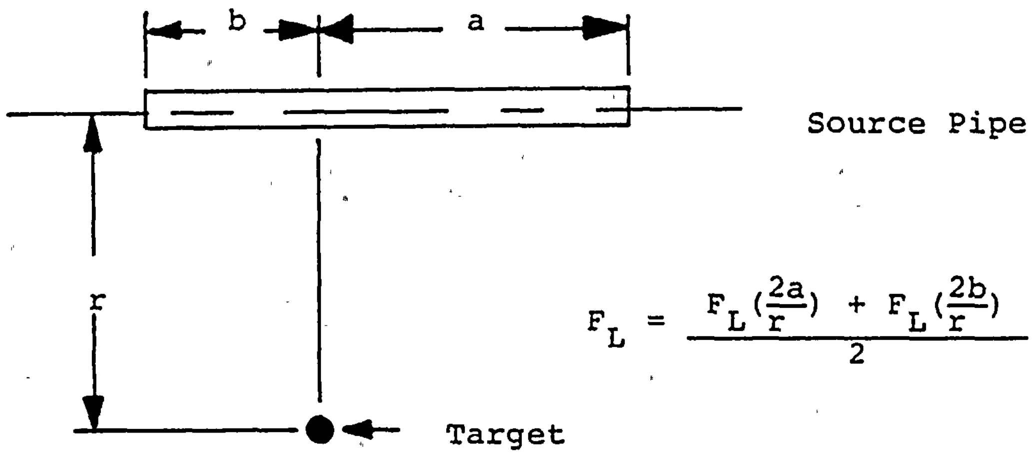
Figure B-17 Pipe Diameter Correction Factor For Gaseous Sources

Configuration 1



$$F_L = \frac{F_L\left(\frac{2a}{r}\right) - F_L\left(\frac{2b}{r}\right)}{2}$$

Configuration 2



$$F_L = \frac{F_L\left(\frac{2a}{r}\right) + F_L\left(\frac{2b}{r}\right)}{2}$$

Figure B-18 Parameters Used for the Calculation of Length Correction Factor

APPENDIX C: PROCEDURE FOR THE CALCULATION OF SECONDARY CONTAINMENT RADIATION ZONE DOSES

C.1 Introduction

Three Mile Island Lessons Learned Short Term Recommendations (NUREG-0578), Section 2.1.6.b, requires all nuclear power plant licensees to calculate post-LOCA accident environmental service conditions for all safety-related equipment. This procedure is specifically concerned with the definition of the Post-Accident radiological environments in the Secondary Containment of Washington Public Power Supply System Nuclear Project Unit 2 (WNP-2), a Boiling Water Reactor (BWR).

The assumptions used in this procedure are based on a non-mechanistic LOCA scenario in which core damage is experienced at the beginning of the accident and Primary Containment isolation is achieved prior to radiation transport.

The radiation level at a given location inside the Secondary Containment of WNP-2 during and following such an accident is defined by the following major source contributors.

1. Gamma ray dose from airborne radioactive sources inside Secondary Containment (Airborne Gamma Dose).

2. Gamma ray dose from radioactive sources suspended in the drywell and the

wetwell inside Primary Containment (Containment Shine Dose).

3. Gamma ray dose from piping containing recirculating fluids (Direct Gamma Dose).

4. Beta ray dose from airborne radioactive sources inside Secondary Containment (Airborne Beta Dose). (See Appendix E)

The methods presented in this procedure make it possible to calculate the worst case gamma ray dose due to the above-mentioned source contributors inside radiation zones (see Section C.2 for the definition of radiation zones) of the Secondary Containment of WNP-2. The radiation zone dose calculated by using this procedure is applicable solely for the purpose of environmental qualification of safety-related equipment.

The following sections of this procedure describe the nomenclature, assumptions, and methods used in calculating radiation dose rates and cumulative doses. Section C.2 defines the terms and nomenclature found in this procedure. The assumptions and approximations used in developing the dose rate calculation method, as well as limitations to this method, are stated in Section C.3. Section C.4 provides a step-by-step procedure for determining the worst case gamma dose rate and cumulative dose inside a particular radiation zone.

The calculation of airborne beta dose is defined in a separate calculation procedure and is not included in this procedure. (See Appendix E)

C.2 Definition of Terms

This section contains the definition of the terms and symbols as used in this procedure:

CIND: (Rads) — <u>Cumulative Integrated Dose</u>
Cumulative dose due to exposure to the decaying radioactive sources.

$\dot{D}_a$ (Rads/hr) — <u>Airborne Gamma Dose Rate</u>
Gamma Dose rate resulting from radioisotopes suspended in the atmosphere of the Secondary Containment.

$\dot{D}_d$ (Rads/hr) — <u>Direct Dose Rate</u>
Gamma dose rate resulting from the radioactive fluid contained inside recirculating pipes.

$\dot{D}_s$ (Rads/hr) — <u>Shine Dose Rate</u>
Gamma dose rate in the Secondary Containment resulting from radioisotopes suspended and deposited inside Primary Containment.

D_t: **Total Gamma Dose Rate**
Gamma dose rate contributed by the sum of airborne, direct, and shine components.

$$D_t = D_a + D_d + D_s$$

GF: **Geometric Factor**
Scaling factor used to convert semi-infinite airborne gamma dose to finite dose inside enclosed air spaces.

$$D_a = \frac{D_{a,\infty}}{GF}$$

$$GF = \frac{1173}{V^{0.338}} \qquad \text{(Ref. C-8)}$$

F_L: **Length Conversion Factor**
A scaling factor dependent upon the source pipe segment length and spatial orientation relative to a target (see Figure C-1 for the calculaton of this factor). F_L is used to convert the standard dose to the dose emitted by a pipe segment of finite length.

F_D: **Diameter Conversion Factor**
A scaling factor dependent upon the source pipe diameter. F_D is used to convert the standard dose to the dose emitted by a pipe of specified diameter.

F_R: **Radial Distance Conversion Factor**
A scaling factor dependent upon the radial distance of the target from the source piping. F_R is used to convert the standard dose to the dose at a target of specified radial distance from the source piping.

F_t: **Total Dose Contribution Correction Factor**
A scaling factor used to convert the standard dose to the dose at a target from a pipe segment of specified geometry and orientation.

$$F_t = F_D \cdot F_R \cdot F_L$$

F_s: **Sum of Dose Contribution Correction Factor**
A scaling factor used to convert the standard dose to the radiation zone dose due to all the significant pipe sources in the zone.

$$F_s = \sum_{i=1}^{n} F_{ti}$$

Radiation Zone: A region in the Secondary Containment defined to be such that gamma and beta radiation calculated in the zone bounds the magnitude of dose received by the pieces of safety-related equipment located in that zone.

Source Term: The total radiated energy (γ,β) associated with a specified quantity of radioactive material released from the reactor as the result of a postulated accident.

Special Sources: Radioactive source of such geometry or concentration that cannot be approximated by pipe segments of diameters 2 inches through 24 inches and containing contaminated liquid of activity concentration established in Section C.3.1. This can be a heat exchanger, standby gas treatment filter, pump, etc.

Standard Dose: Gamma dose at a target having a radial distance of 8 ft. from a source pipe centerline segment, infinitely long, of nominal pipe diameter 8 inches, schedule 40 piping.

Target: The point in space chosen to represent a location or object for which a dose rate and/or cumulative dose is being calculated.

Worst Target: Location of the piece of safety-related equipment inside a radiation zone which will experience the highest gamma dose among all the pieces of safety-related equipment in that zone.

C.3 Assumptions, Approximations, and Limitations

C.3.1 Basic Assumptions to be Used in the Analysis

Gamma doses and dose rates inside radiation zones will be determined for three types of radioactive source distributions:

- Isotopes suspended in the atmosphere of the Secondary Containment (airborne gamma dose).
- Gamma irradiation from the Primary Containment (shine dose).
- Direct gamma irradiation from the radioac tive fluid contained inside recirculating pipes (direct dose).

The dose contributed by each of these sources is determined by the location of the equipment, the time-dependent distribution of the source, and the effects of shielding.

The assumptions used in determining the initial distribution and leakage of radioactivity in the Primary Containment are as follows:

1. 100% of the noble gases and 25% of the halogens in the reactor core will be

distributed homogeneously within the Primary Containment free volume immediately following the postulated accident.

2. 50% of the halogens and 1% of the remaining fission products in the core will be mixed homogeneously with the Primary Containment liquid space instantaneously. The Primary Containment liquid space is defined as the sum of the suppression pool liquid and the Reactor Coolant System (RCS) liquid.

3. The core fission product source term is defined as the total product generated in the core after 1000 days at reactor power of 3481 MWt.

4. Primary Containment leakage of 0.50% volume/day was considered.

Justification of Assumptions

Assumptions 1 and 2 are NRC recommended assumptions for defining radioactivity release fractions for the qualification of safety-related equipment (Ref. C-1) and are con sistent with the accident analysis Reference C-2.

Assumption 3 represents the maximum burnup level in the core prior to radioactivity release and is conservative.

Assumption 4 is consistent with the assumptions established in Reference C-3.

C.3.1.1 Assumptions Used in the Calculation of Airborne Dose Rate Inside Secondary Containment

1. Activity that leaks into the Secondary Containment is homogeneously mixed with the Secondary Containment atmosphere prior to its removal from the atmosphere through the Standby Gas Treatment System (SGTS).

2. The minimum SGTS flowrate of 1100 SCFM is assumed to be the flowrate of the effluent air.

3. Air that leaks out of the Primary Containment flows directly into the Secondary Containment. Bypass leakage is not considered.

4. Geometric factors can be used to convert the semi-infinite cloud dose to a finite cloud dose.

Justification of Assumptions

Assumption 1 is consistent with the NRC recommended assumptions used for calculation of doses inside Primary Containment (Ref. C-1).

Assumption 2 is conservative because it represents the minimum flowrate of the SGTS system (Ref. C-4).

Assumption 3 is conservative when considering dosage in the Secondary Containment, since it maximizes the buildup of radioactivity in the Secondary Containment.

Assumption 4 is based on the results presented in Reference C-5 and based on average gamma ray energy of 0.733 Mev. The effect of variation of this parameter due to differences in gamma ray energies have been proven to

be negligible (see Appendix B for justification).

C.3.1.2 Assumptions Used for the Calculation of Shine Dose From Primary Containment

1. No depletion of activity due to leakage is assumed.

2. The airborne source is assumed to be uniformly distributed in the drywell and in the wetwell air space. The effect of the plate-out of iodine on the walls is not considered in Secondary Containment.

3. Activity in the wetwell water volume is assumed to be uniformly distributed in the sump water.

4. The dosage at a point inside the region closest to the source is considered to be representative of the gamma dose in the region.

Justification of Assumptions

Assumption 1 maximizes the source activity and is conservative.

Assumptions 2 and 3 are necessary because plate-out mechanisms are unknown. These assumptions are consistent with that considered in Reference C-1.

Assumption 4 maximizes the gamma ray dose at the region and is conservative.

C.3.1.3 Assumptions and Approximations Used in the Calculation of Direct Doses

1. No valve leakage is assumed.

2. Schedule 40 piping is assumed.

3. Heat exchangers and pumps can be approximated as pipe systems. The volume of radioactive liquid in the component and its length are used to determine an equivalent volume of liquid.

4 Radioactive piping with diameters 2-1/2 inches or less was not modelled unless it was determined that such a pipe was a major source contributor. A major source contributor is defined as the only radioactive pipe in a target area or the radioactive pipe of closest proximity to the target.

Justification of Assumptions

Assumption 1 is consistent with Reference 2.5, Item II. B.2, Clarification (2).

Assumption 2 is a conservative simplification of the calculation process. Because the majority of the pipe segments considered are schedule 40 piping, and because increases in pipe schedule can only decrease the dose rates at the targets, this approximation is considered to be conservative and appropriate.

Assumption 3 is a crude approximation for dose rates contributed by complex geometries. Because the pump and heat exchanger walls are thicker than the pipe walls of schedule 40 piping, this assumption is conservative.

Assumption 4 is made because the dose contributions due to pipe segments of diameters

less than 2-1/2 inches are generally negliglible, unless they are major source contributors.

C.3.2 Limitations

The following limitations apply to the use of this procedure for the calculation of radiation zone doses.

1. This procedure is only applicable to the calculation of radiation zone doses in the Secondary Containment of WNP-2.

2. The assumptions stated in Section C.3.1 are basic to the methodology used in this procedure. Changes in any of the assumptions will affect the accuracy of the results generated using this procedure.

3. The calculation of direct doses using the generic curves in this procedure is limited to liquid sources in schedule 40 pipe segments or equivalent pipe segments with nominal pipe diameters ranging from 2 inches to 24 inches. Any deviation from these pipe geometries should be modeled as special cases. Note: Schedule 40 piping is used because the majority of the pipe segments to be considered are standard pipes (sch 40). Increases in the pipe schedule only introduces conservatism in the results.

4. The results for direct dose calculated using the generic curves were found to be accurate to within 10% (see Ref.C-8 for error study).

5. Source piping located 40 ft or further from the target is generally an insignificant dose contributor. If its contribution is not found to be negligible, it should be considered as a special source.

C.4 Procedures For The Calculation of Secondary Containment Radiation Zone Doses

This procedure describes the method used in calculating the radiation doses inside radiation zones.

For equipment located inside a zone, the following three sources contribute to the total dose level.

° Airborne dose (gamma and beta).

° Direct gamma dose from sources within pipes.

° Direct gamma shine dose from drywell and wetwell.

A step-by-step procedure is discussed in the following sections for the calculation of the maximum total gamma dose and dose rates for each zone. The calculation of the airborne beta dose and dose rate is discussed in a separate calculation and is not included in this procedure. (See Appendix E)

C.4.1 <u>PROCEDURE A:</u> Radiation Zone Dose Calculation

The first step in preparing a zone dose calculation is to identify all the parameters to be used. This includes the identification of all the potential sources and targets, both inside and outside the zone, and the identification of the dimensions of the zone. Figure C-2 is a step-by-step flowchart of the

calculation procedure. When identifying sources outside the zone, sources at the upper and lower elevations in the review process are included. Some rough dose estimates are used to determine whether sources outside a zone are significant source contributors. For example, if the closest pipe segment is a few feet away from a target, calculations will show that pipe segments outside the room thirty feet away are insignificant source contributors. Conversely, if a target is located near a wall with several pipes on the other side of a wall, then those pipes may become significant source contributors.

C.4.2 PROCEDURE B: Airborne Dose Calculation in Secondary Containment

Because the semi-infinite airborne dose and dose rates are already calculated and shown in Figures C-6 and C-7, the only calculation involved in determining the airborne dose is the conversion of the semi-infinite cloud dose at Reactor Building concentrations to a finite cloud dose inside the cubicles in which the radiation zones are defined. The first step in this calculation is to determine the volume which defines the air space (or zone) of interest. An enclosed air space is defined as a cubicle, at least 95% shielded by concrete (or equivalent shielding) at least 1 foot thick.

To convert a semi-infinite cloud dose (calculated in Ref C-7) to a finite cloud dose, a geometric factor is used.

$$D_a(t) = \frac{D_{a,\infty}(t)}{GF} \qquad (4-1)$$

where $GF = \frac{1173}{V^{0.338}}$ (Ref. C-8)(4-2)

GF = geometric factor (dimensionless)
V = Volume of the enclosed air space (ft^3)

Similarly,

$$CIND_a(t) = \frac{CIND_{a,\infty}(t)}{GF} \qquad (4\text{-}3)$$

Figure C-3 is a step-by-step flowchart of the procedure for calculating airborne gamma doses.

C.4.3 PROCEDURE C: Primary Containment Shine Dose Calculation

Containment shine doses are calculated using the QAD-P5A computer code. Guidelines for preparing input parameters are documented in Procedure E and Reference C-6. The modelling procedure and the accuracy of the results are highly dependent on the geometry to be modeled, specification of the source volume, and the selection of a buildup factor. Figure C-4 is a step-by-step procedure for calculating containment shine doses.

C.4.4 PROCEDURE D: Direct Dose Calculation

The first step in the direct dose calculation (from Ref C-8) is the identification of the "worst" target. Normally, the worst target is the piece of equipment that is closest to the major source piping and can be selected by inspection. However, if situations arise such that the worst case target cannot be chosen by simple inspection, order-of-magnitude calculations are performed for each potential worst case target in the zone. These calcu-

lations are illustrated in Steps 3a through 3c of Figure C-5.

The next step is to identify special sources. Special sources are defined as source geometries that cannot be represented by liquid pipe segments between 2 and 24 inches in diameter. Example special sources are: SGTS filters, RCIC steam pipe, turbines, and heat exchangers larger than 24 inches diameter. Other components such as pumps and small heat exchangers should be modelled as pipes. The pipe cross-sectional area is calculated by dividing the total fluid volume by the effective length of the component.

The contribution due to sources with shield walls is investigated next. Figure C-13 is used for this evaluation. If these sources are determined to be significant contributors, special QAD-P5A modeling procedures as described in Procedure E are followed.

It is unlikely that all sources under consideration will contribute significantly to the dose at a specific target. If all source contributions were to be calculated, the time involved in performing the calculation would be unnecessarily long without making a substantial improvement in the accuracy of the results.

Hence, as the sources are being identified, good judgment is used to distinguish between sources which contribute significantly to the target dose and those sources which do not.

An insignificant source is determined by comparing its dose contribution to the source making the largest dose contribution. The comparison is facilitated by arranging sources in decreasing order of importance and assigning rank numbers to the sources. The largest dose contributor is given a ranking number of 1. The largest dose contributor is determined by inspection of the sketches and drawings being used. The largest dose contributor is generally the longest segment with the largest pipe diameter and the least amount of intervening shielding between the target and source. All sources which are in the radiation zone and have been assumed to be insignificant contributors are listed as such to indicate that those sources have been considered.

Equations Used in the Calculation of Dose Rates

The following procedure is followed for calculation of correction of Dose Rates factors of Dose Rates (Step 9 through Step 12 of Figure C-5):

1. Identify the radial distance of the pipe segment from the target; read F_R from Figure C-11.

   If the target is in contact with the source piping, read F_D from Table C-1 and set F_R and F_L equal to 1. (Note: dose rate is not a function of pipe length and radial distance).

   If the target is geometrically in line with the source pipe segment, as shown in configuration 3 of Figure C-1, set F_L=1 and read F_D and F_R from Figures C-14 and

C-15, respectively. (Note: F_L is defined here because dose rate is not sensitive to pipe length variation.)

2. Identify the pipe diameter; read F_D from Figure C-10.

3. Determine F_L from Figure C-12; use equations in Figure C-1 to calculate this factor.

4. The total dose contribution factor for a given pipe segment (I) is given as

$$F_t(I) = F_D(I) \cdot F_R(I) \cdot F_L(I)$$

5. When all the significant contributions have been calculated, sum the total dose contribution factors.

$$F_s = \sum_{n=1}^{n} F_t\ (I)$$

6. To determine if a source is negligible, the following test should be performed:

   When N source segments are being considered and the dose contribution of ranking I is less than 1/10 of the dose rate calculated from the largest source divided by (N-I), the sources remaining should not contribute more than 10% to the total source contribution. This level of accuracy should be adequate for most calculations.

The total integrated direct dose and dose rate can be calculated.

$D_D(t) = D_{Do}(t) \cdot F_s + D_{D(t)}$ (Special Sources) .

$CIND_D(t) = CIND_{Do}(t) \cdot F_s + D_D(t)$ (Special Sources)

where $D_{Do}(t)$ and $CIND_{Do}(t)$ are dose rates and cumulative doses for standard pipe segments and are found on Figures C-8 and C-9.

Table C-1 Diameter Correction Factor (F_D) for Targets in Contact with the Source Piping

Nominal Pipe Diameter (in)	Pipe Diameter Correction Factor (F_D)
2	18.4
4	24.4
6	54.6
8	33.3
10	35.3
12	35.3
14	35.5
16	33.7
20	32.0
24	29.6

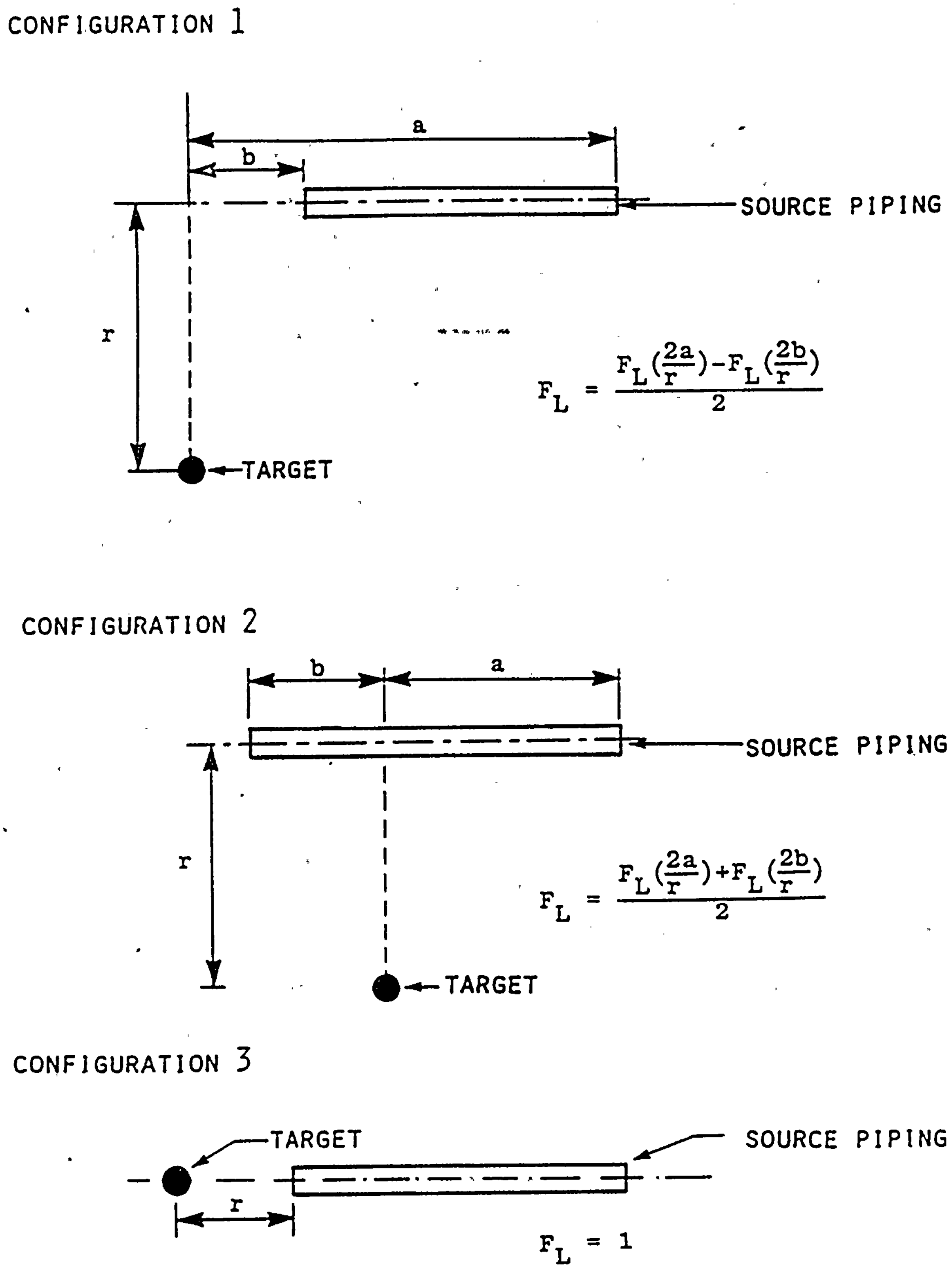


Figure C-1 Calculation of Length Correction Factor

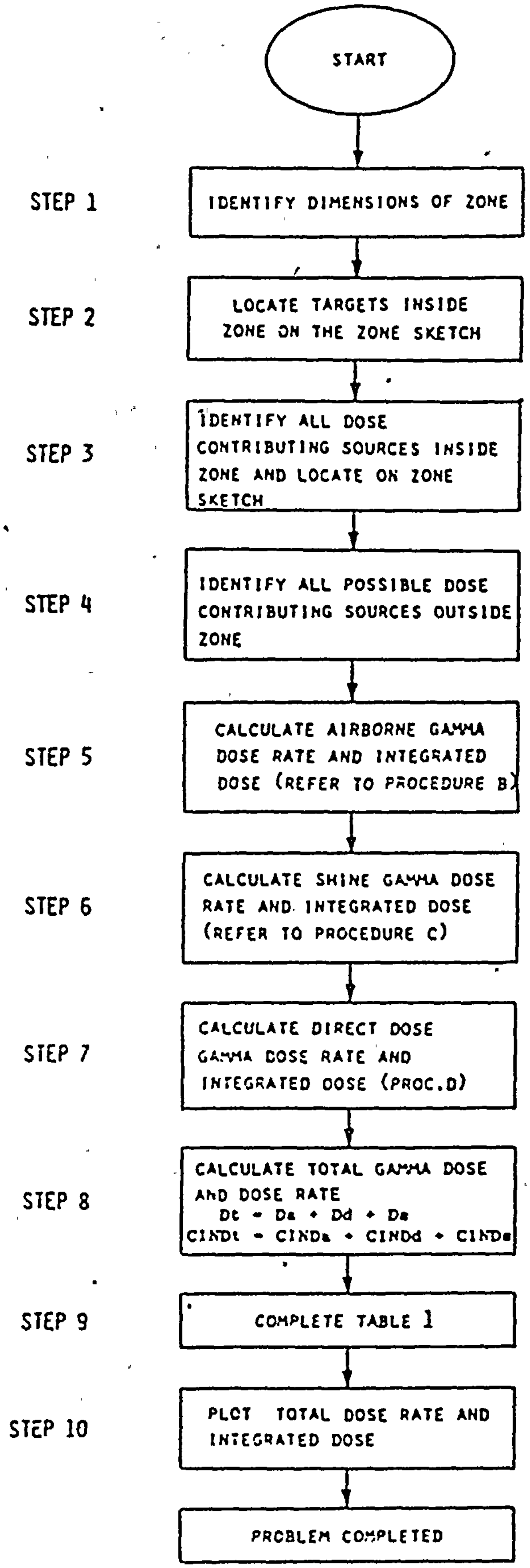


Figure C-2 PROCEDURE A: PROCEDURE FOR CALCULATING RADIATION ZONE DOSES

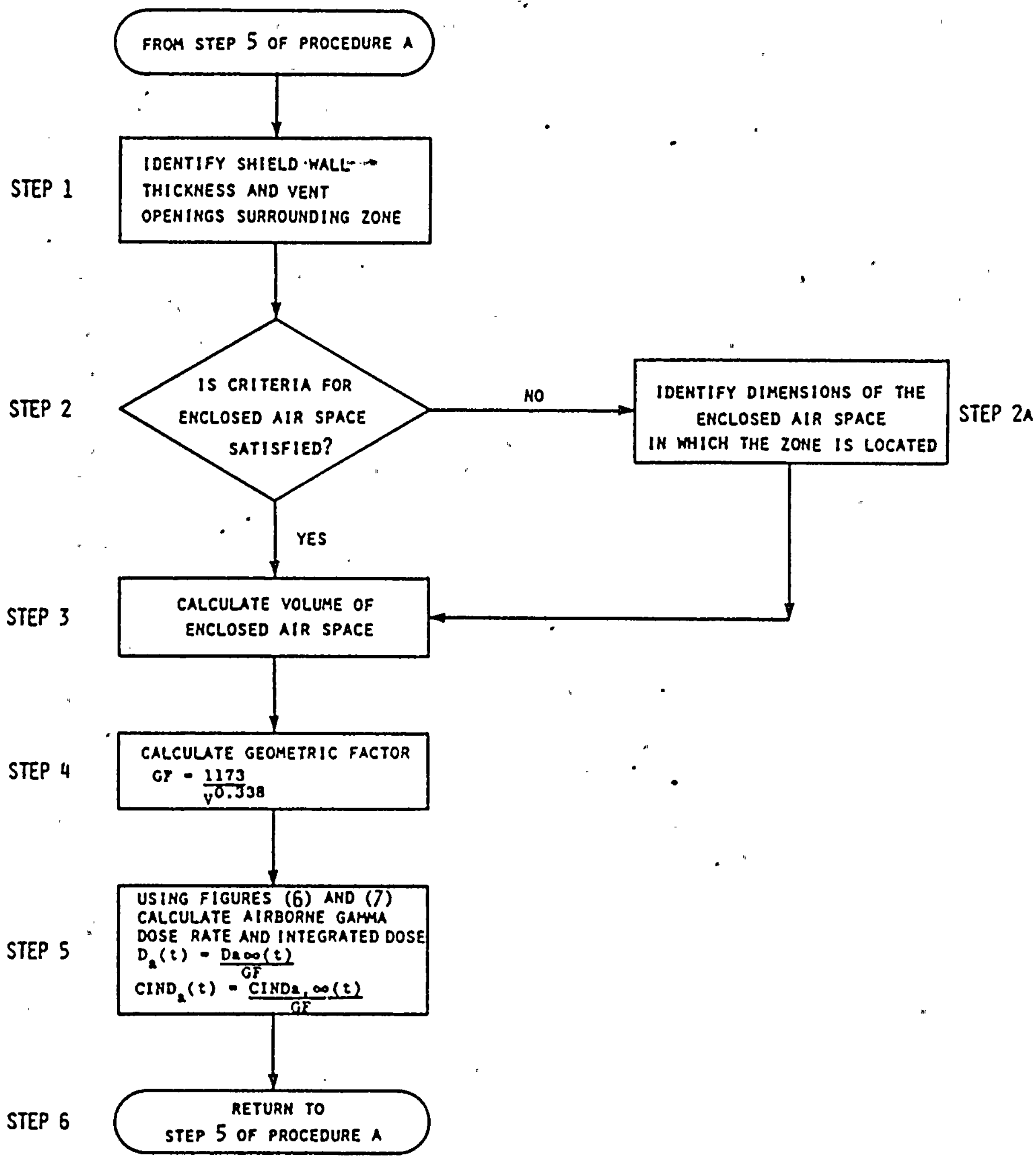


Figure C-3 PROCEDURE B: PROCEDURE FOR CALCULATING AIRBORNE GAMMA DOSE RATE AND INTEGRATED DOSES

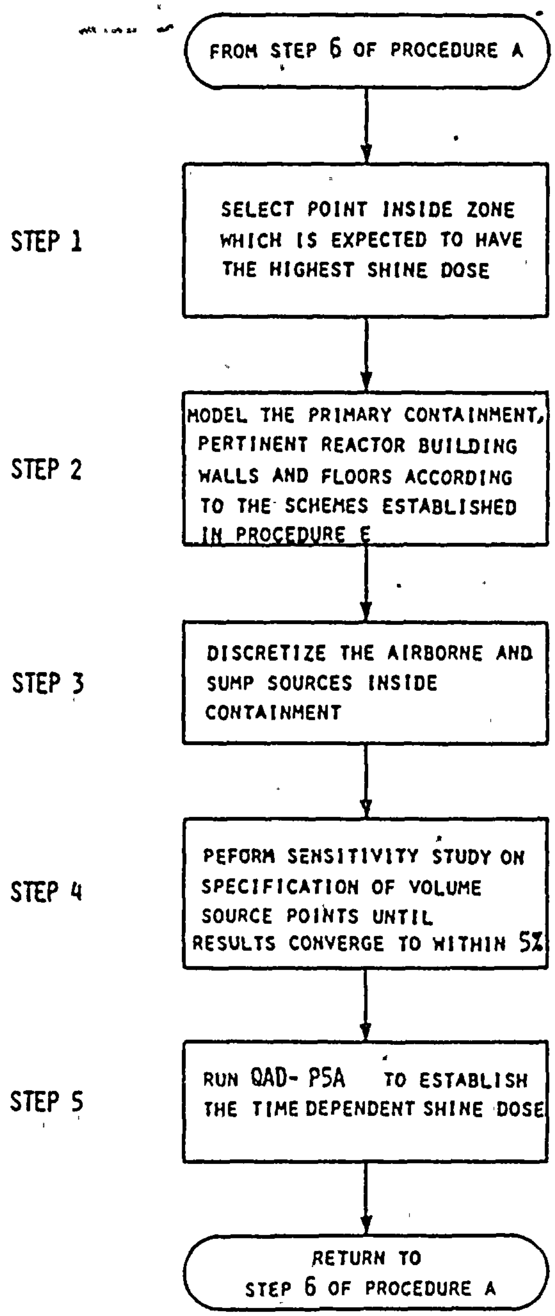


Figure C-4 Procedure C: Procedure for the Calculation of Containment Shine Dose

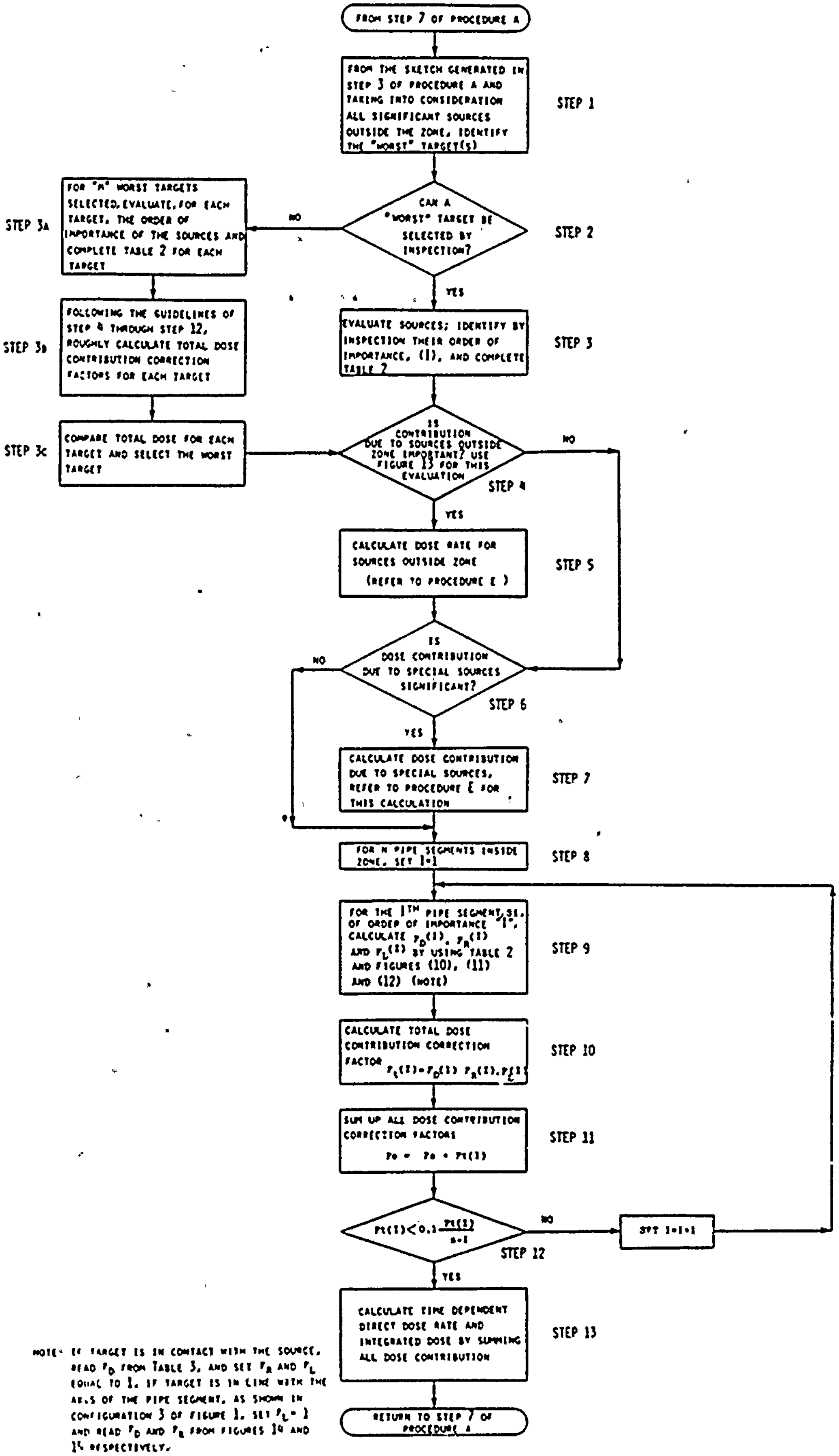


Figure C-5 Procedure D: Procedure for the Calculation of Direct Dose Rate and Integrated Dose

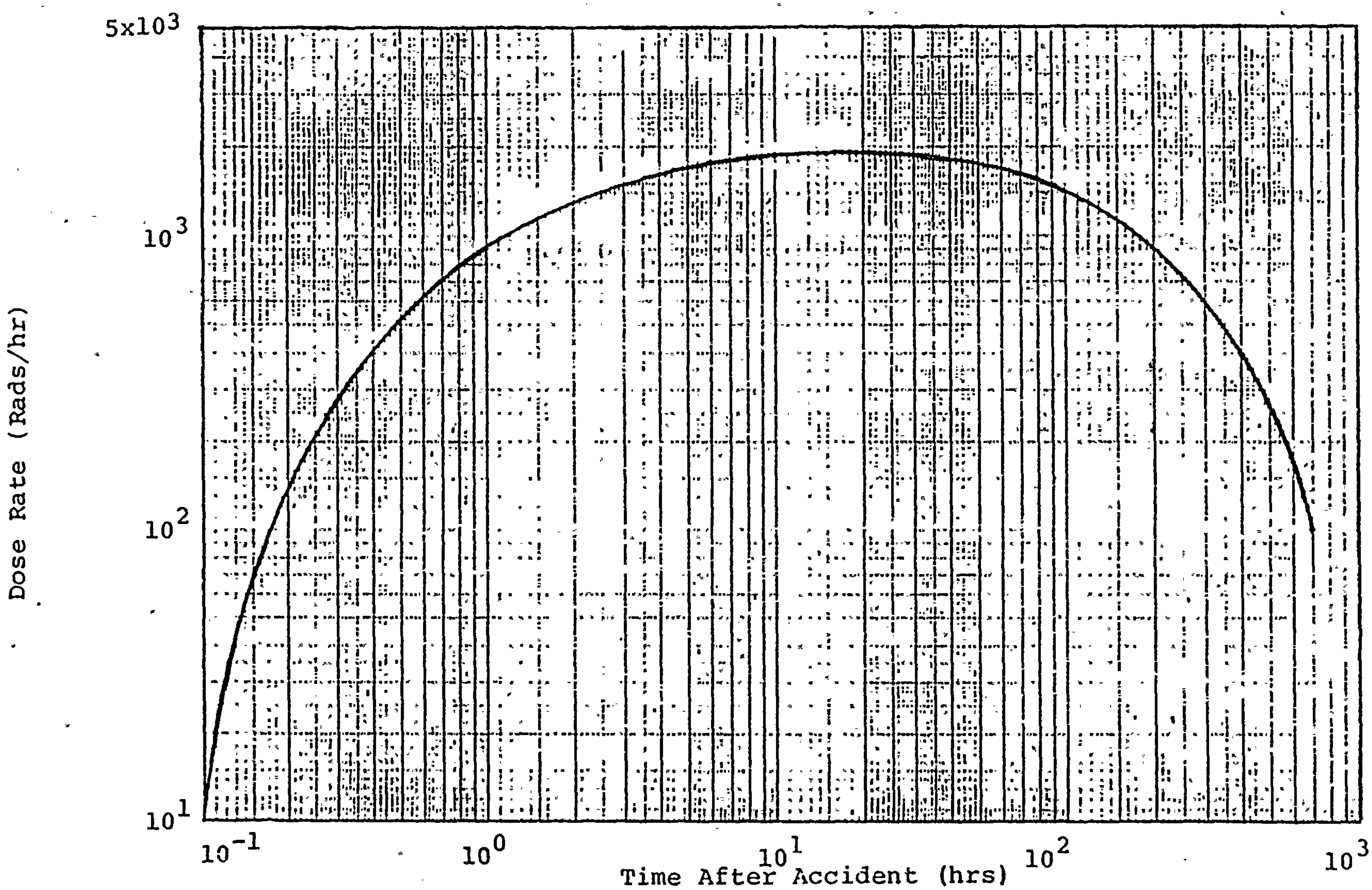


Figure C-6 Time Dependent Gamma Dose Rate For A Semi-Infinite Cloud of Fission Products At Secondary Containment Concentrations (0.5%/Day Primary Containment Leakage)

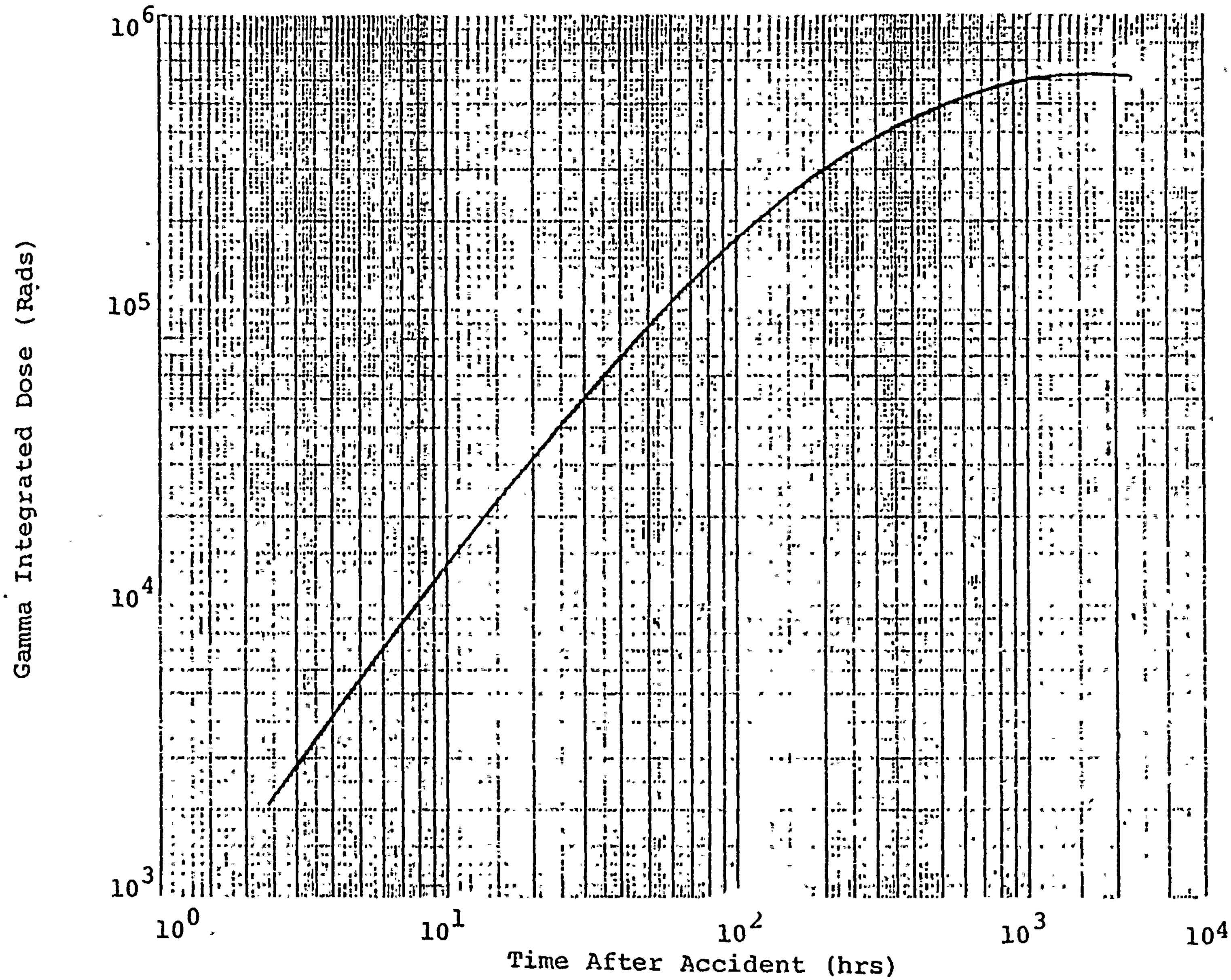


Figure C-7 Time Dependent Gamma Integrated Dose For a Semi-Infinite Cloud Of Fission Products At Secondary Containment Concentrations (0.5%/Day Primary Containment Leakage)

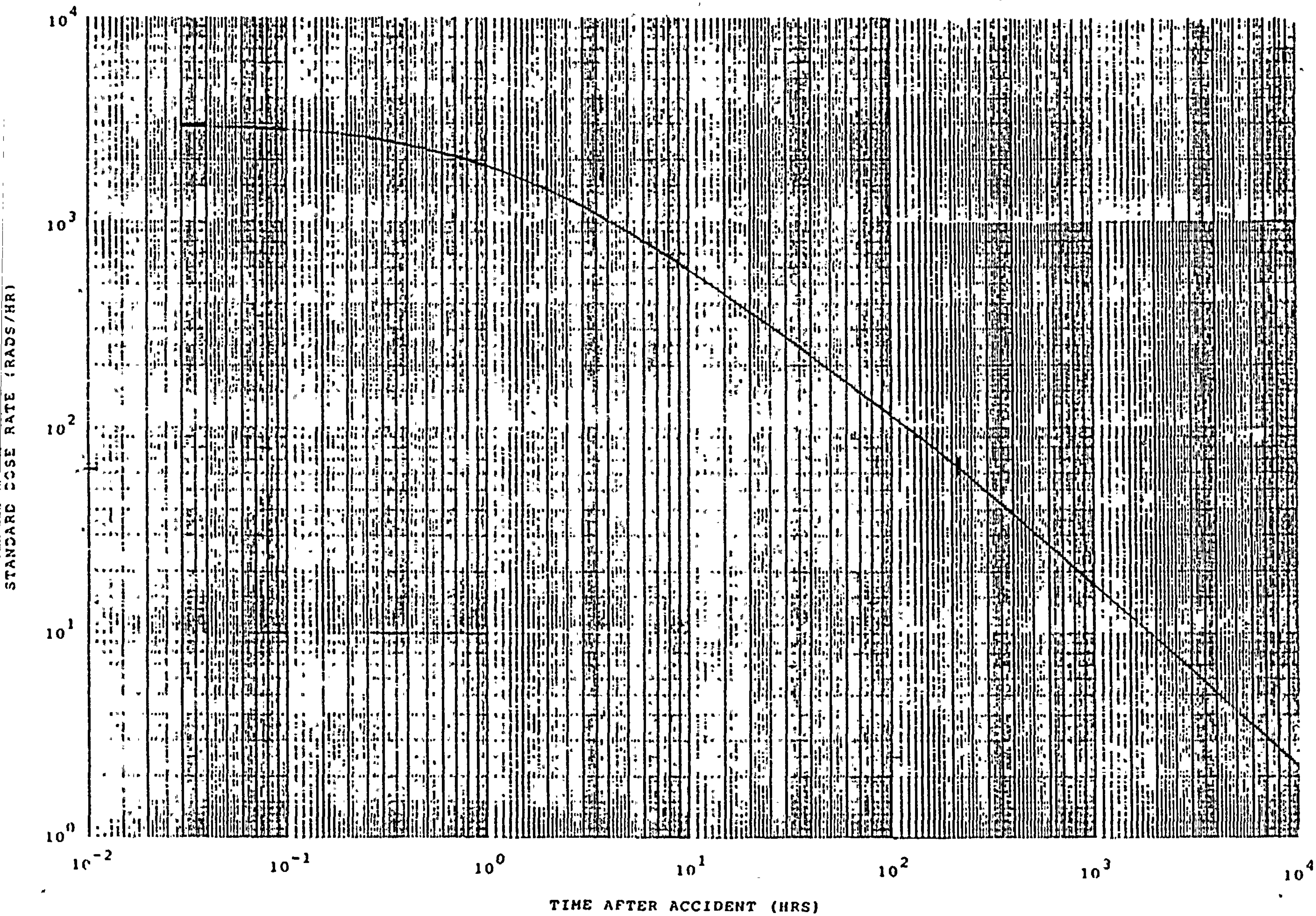


Figure C-8 Gamma Dose Rate at a Target 8 Feet Away From Standard Pipe

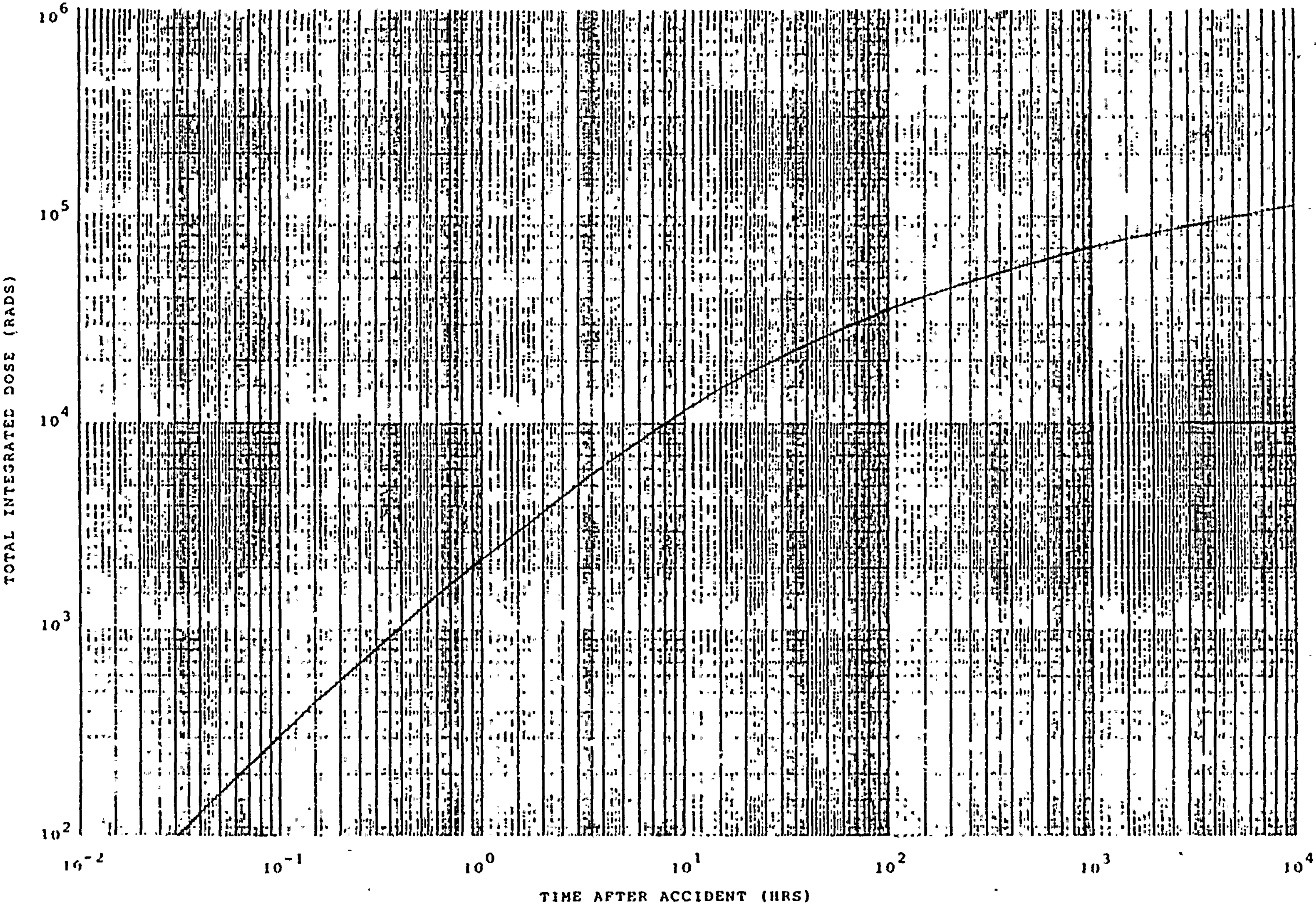


Figure C-9 Gamma Integrated Dose at a Target 8 Feet Away From Standard Pipe

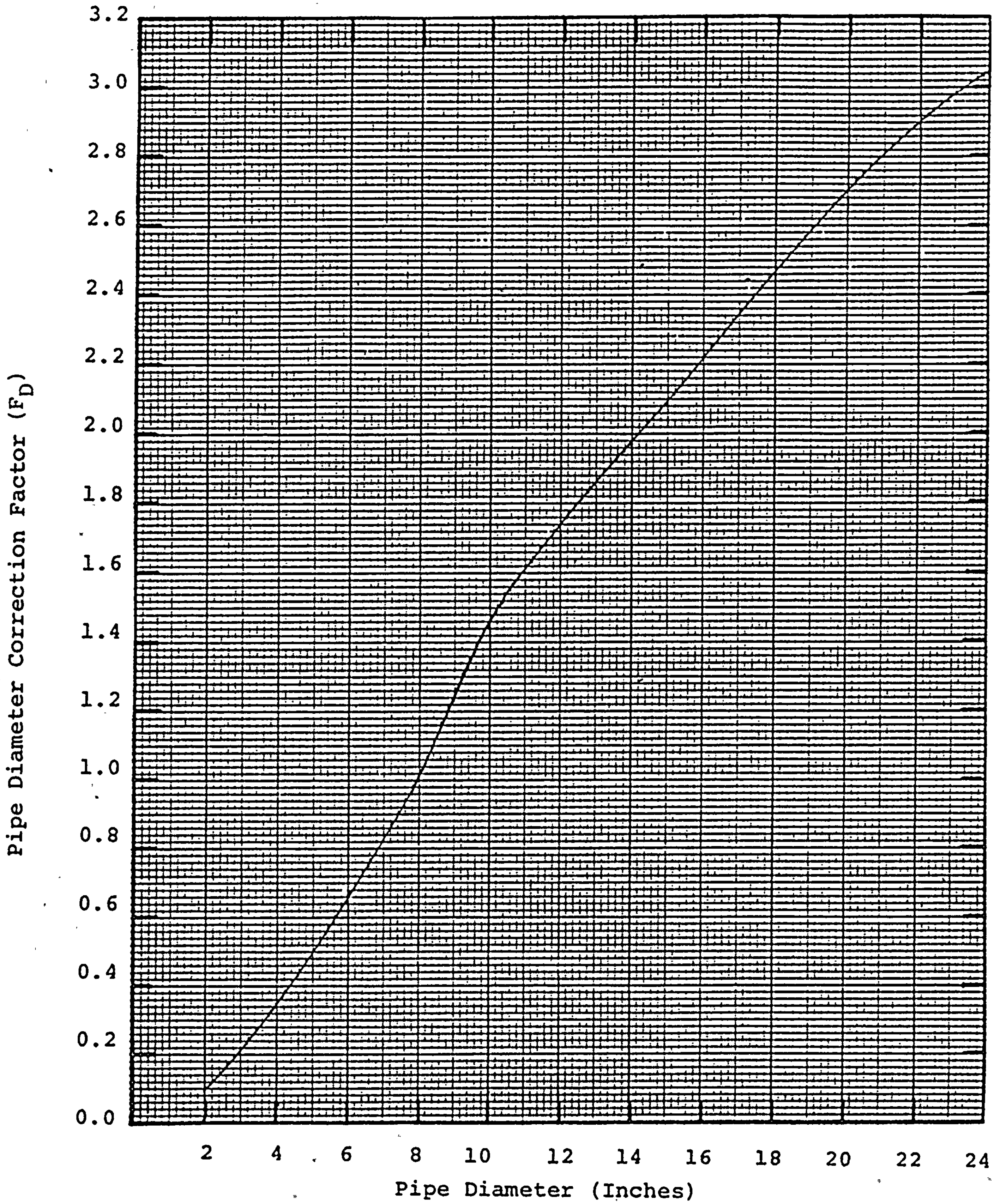


Figure C-10 Pipe Diameter Correction Factor

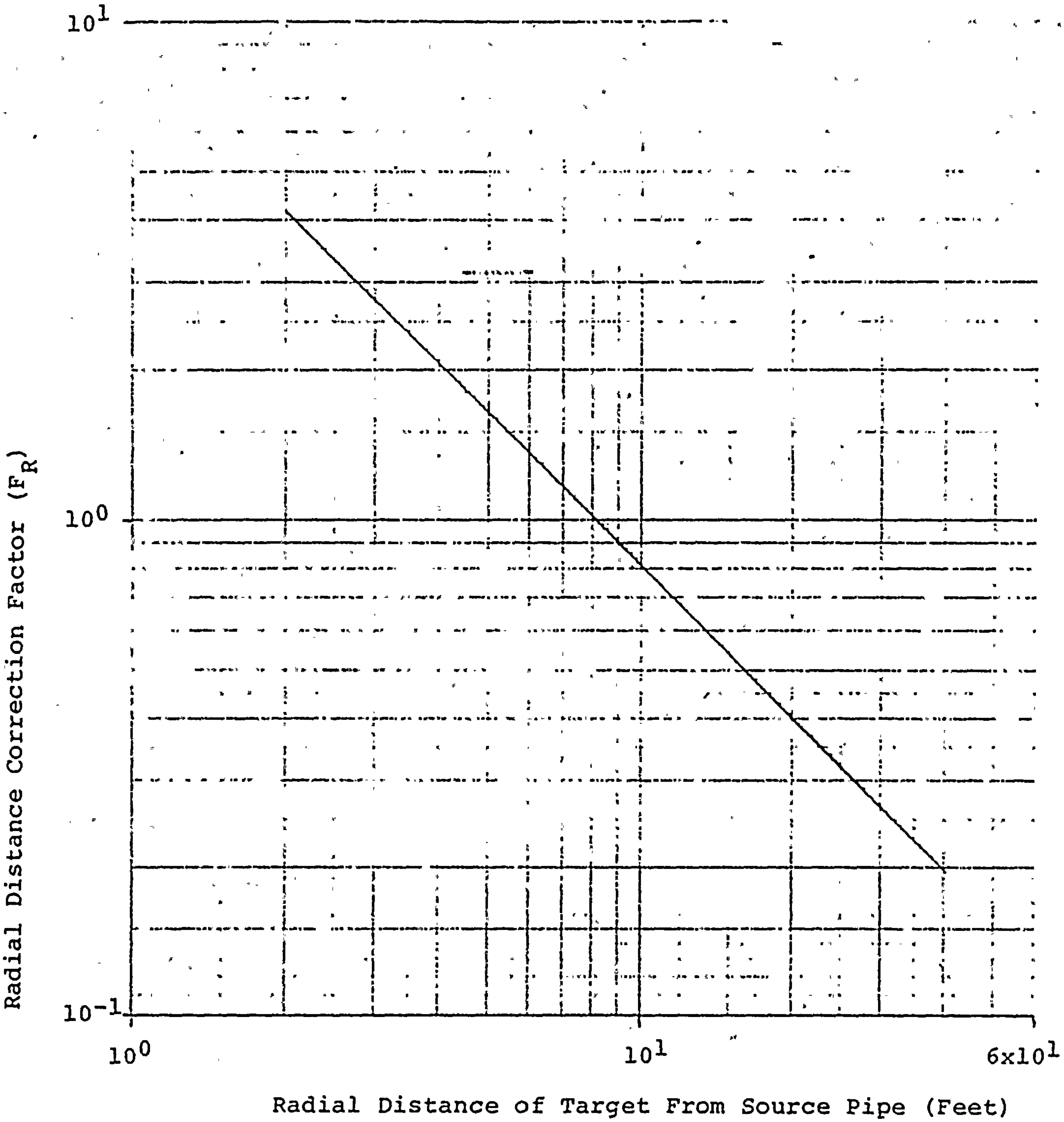


Figure C-11   Radial Distance Correction Factor

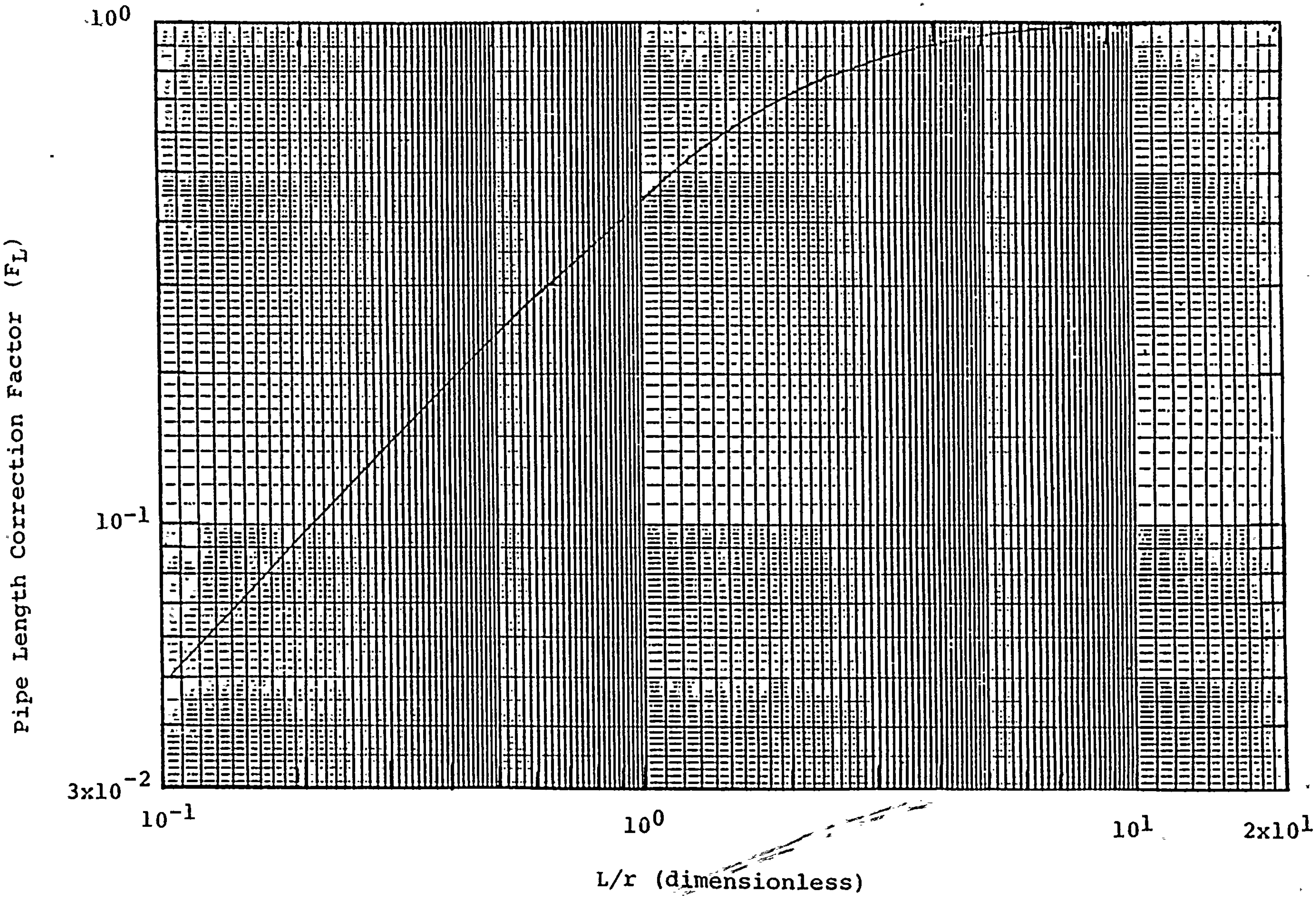


Figure C-12 Pipe Length Correction Factor

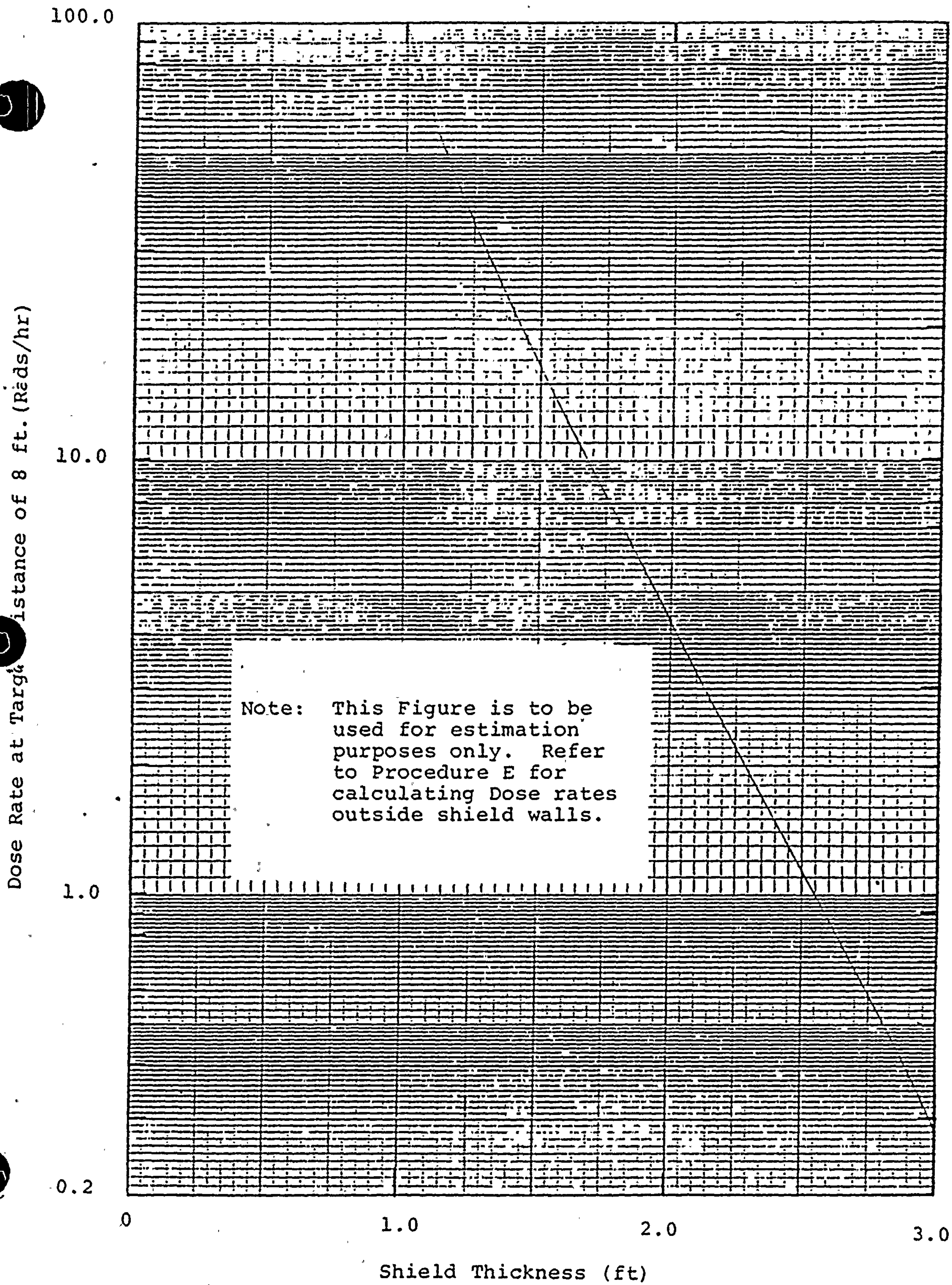


Figure C-13 Dose Rate vs. Concrete Shield Thickness for Standard Pipe (8" Sch 40)

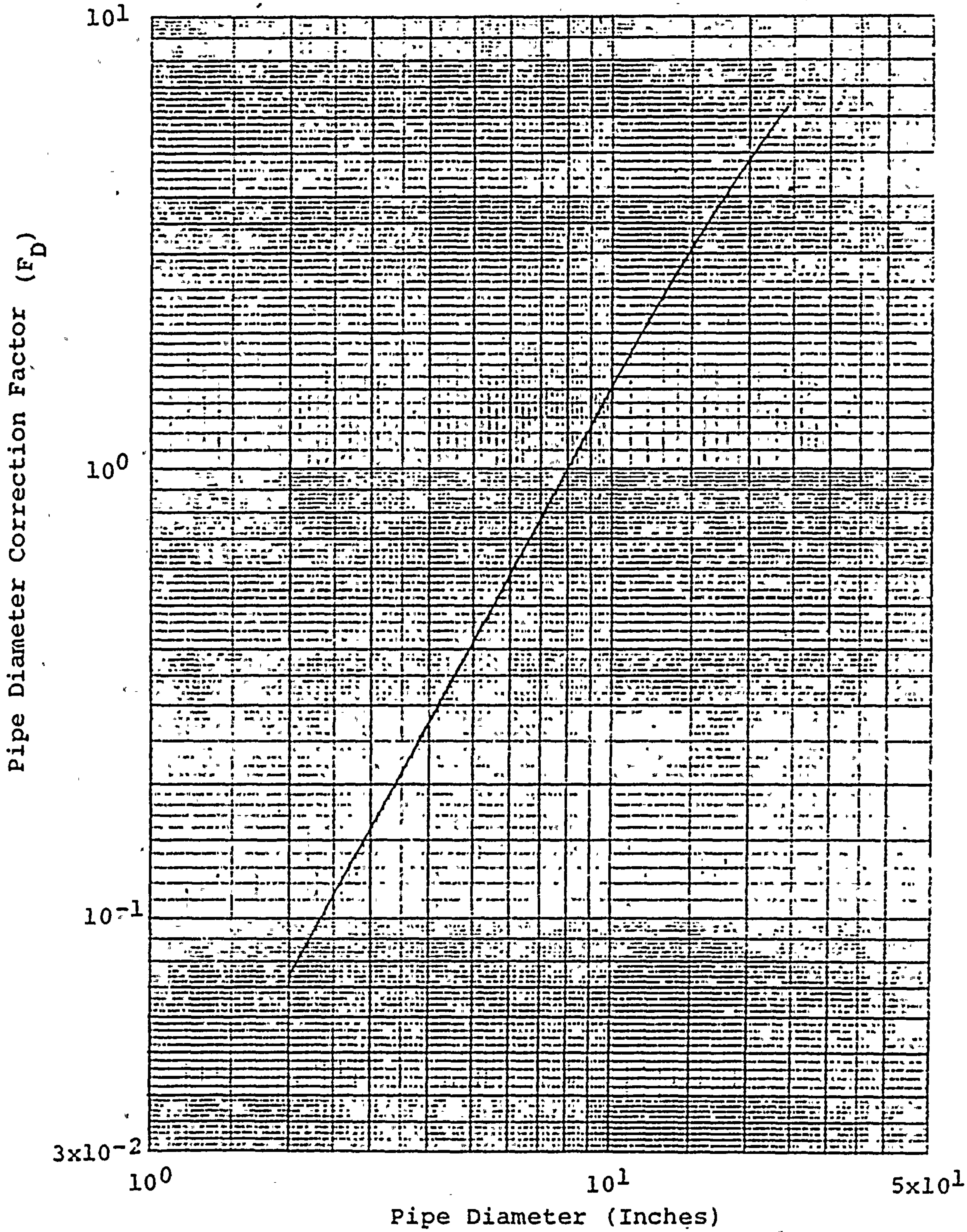


Figure C-14 Pipe Diameter Correction Factor for Targets Located Axially in Line with Source Piping (Configuration 3 Of Figure C-1)

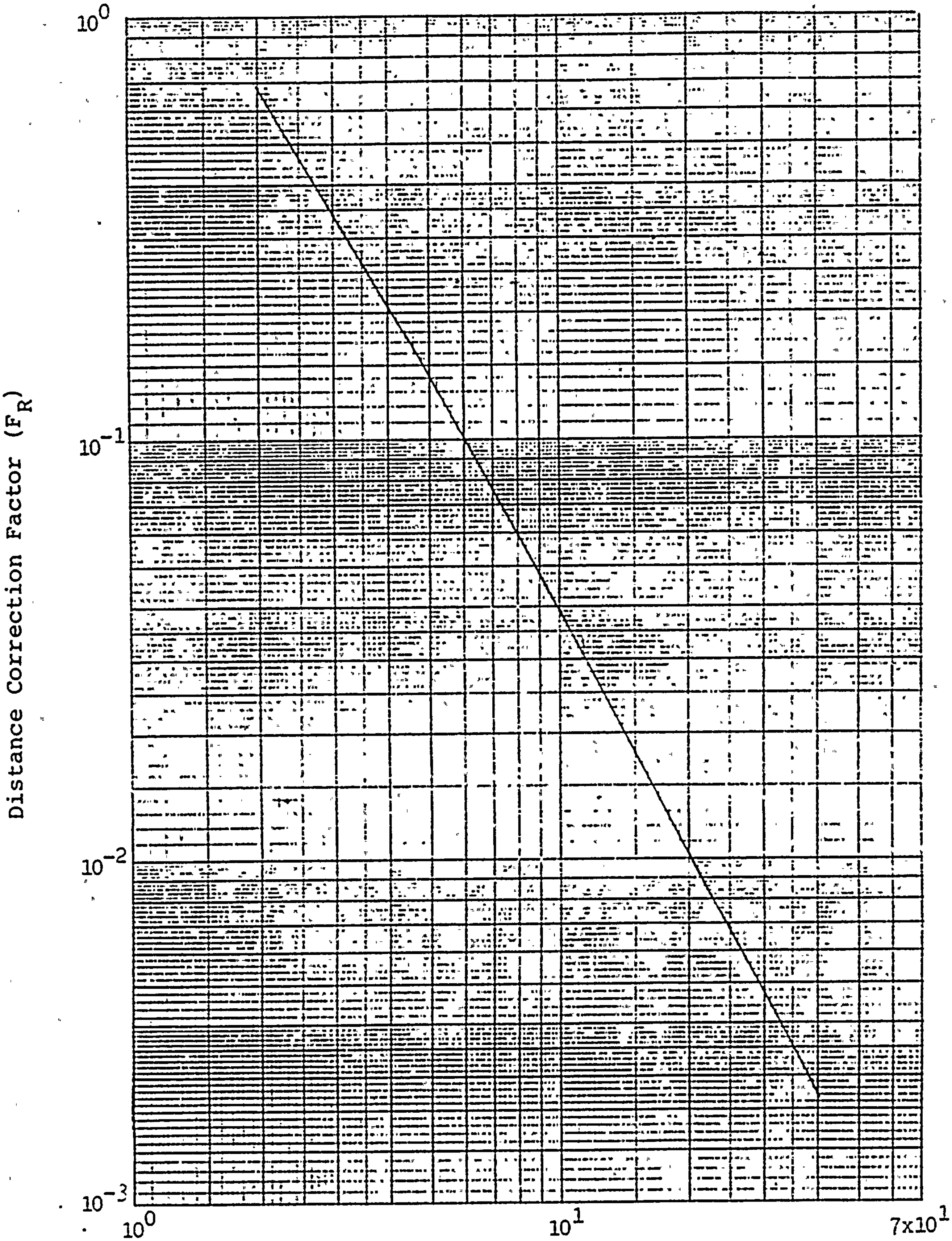


Figure C-15 Distance Correction Factor for Targets Located Axially in Line with Source Piping (Configuration 3 of Figure C-1)

APPENDIX D Calculation of the Secondary Containment Radiation Doses Due to the Standby Gas Treatment System

The SGTS filters are located in the Reactor Building (Elevation 572' -0") of WNP-2 and function to process the radioactive gaseous effluent from the Primary and Secondary Containment. In the event of a LOCA in the Primary Containment, the SGT system will be actuated, and the gaseous iodine, the methyl iodine as I-131 that leak out of the Primary Containment, will be absorbed in the charcoal absorber filters while the particulates will be absorbed in the prefilters and HEPA filters. Depending on the radioactive source distribution and the Primary Containment leakage rate, the radioactive iodine concentration in the filters will increase with time.

The purpose of this study is to evaluate the time dependent gamma radiation level exposing safety-related equipment located near the SGTS filters and in adjacent rooms following a LOCA.

The time-dependent buildup of activity in each of the filters is first calculated. The time and energy-dependent gamma activity levels on the SGTS filters is developed by a combination of computer runs and hand calculations and is used as input to the QAD-P5A computer code to calculate the gamma radiation levels for the pieces of safety-related equipment located in the room. A discussion of the analysis is given as follows.

D.1 Description of the SGTS Filters

Figure D-1 is a drawing of the SGTS filter train. The SGT system consists of two fully redundant filter trains, each of which consists of the following components in series:

a) A demister to remove entrained water particles in the incoming air stream.

b) Two banks of electrical coil heaters designed to limit the humidity of the incoming air to 70% at design flow during post-LOCA conditions.

c) A bank of prefilters to remove large particulates from the air stream.

d) A bank of high efficiency particulate air (HEPA) filters to remove 99.9% particulates from the airstream.

e) Two four inch deep banks of charcoal absorbers arranged as shown in Figure D-2, are designed to absorb the gaseous elemental and organic halogens from the airstream. The dimensions of the charcoal filters are shown in Figure D-3.

f) A second bank of HEPA filters, identical to that described in d) above. The function of this second HEPA filter bank is to capture charcoal dust which may escape from the charcoal filters.

Both SGTS filter units are located on elevation 572 of the Reactor Building and are automatically actuated and become fully operational within 34 seconds in the event of any of the three isolation signals.

a. High radiation in the Reactor Building ventilation exhaust duct.

b. High drywell pressure.

c. Low water level in the reactor vessel.

D.2 Calculation of Time-Dependent Filter Activity Concentration

The analysis of the time-dependent transport of the radioactivity from the Primary Containment to the SGTS filters and the activity concentration on each filter is based on the following assumptions:

1. The SGTS filters are assumed to be loaded by halogens at the rate of 0.73% Primary Containment free volume per day. This is composed of 0.5% from Primary Containment leakage and 0.23% from MSIVLCS.

2. Straight exhaust through the filters, with no mixing or holdup in the Secondary Containment atmosphere, is assumed.

3. The released halogen fraction is 25% of the core halogen inventory. This halogen fraction is assumed to be composed of 91% elemental, 4% organic and 5% particulate halogens.

4. The particulate halogens will be homogeneously distributed within the Prefilters and the HEPA filters, while the elemental and organic halogens will be homogeneously distributed within the two charcoal filters of the filter train.

5. MSIVLCS discharges directly to the inlet of the operating SGTS filter unit. Therefore, there is no secondary cloud dose associated with the MSIVLCS discharge.

Assumption 1 is based on the Primary Containment rated leakage flowrate and is consistent with the assumptions used in the radiological consequence analysis (Ref. D-1).

Assumption 2 is a NRC recommended assumption for the analysis for Fission Product Control Systems (Ref. D-2).

Assumption 3 is recommended by the NRC for use in radiological consequence analysis (Ref. D-3).

Assumptions 4 and 5 are necessary because the time dependent absorption and leakage of halogens in the filters is unknown. The homogeneous assumption is considered appropriate and conservative.

The time and energy dependent gamma activity concentration in the SGTS filters was first investigated. As discussed in Section 5.5.4, this analysis was performed by a combination of computer analysis and hand calculations. The activity concentration of a halogen isotope inside a SGTS filter is changing with time due to the following three mechanisms:

1. Transport of activity from the Primary Containment and deposition on the filters due to air leakage.

2. Depletion of activity due to radioactive decay.

3. Increases in activity levels due to daughter product generation from radioactive decay of other isotopes.

Because activity is assumed to be transported directly to the SGTS filters and that activity inside the Primary Containment and inside a filter is assumed to be homogeneously distributed, the rate of change of radioactivity concentration due to daughter product generation and radioactive decay is independent of radioactivity transport. In other words, radioactivity would be transported at the same rate from the Primary Containment to the SGTS filters as if there were no decay. Therefore, the activity concentration inside the filters can be expressed as:

$$C_{1i}(t) = F_{iR}(t)\ F_{1V}(t)\ C_{1i}(0) \qquad \text{(D-1)}$$

$$C_{ni}(t) = F_{iR}(t)\ F_{nV}(t)\ C_{1i}(0) \qquad \text{(D-2)}$$

where:

$F_{iR}(t)$ = Depletion factor of radioactivity concentration due to isotope decay and daughter product generation and is independent of transport.

$F_{1V}(t)$ = Reduction factor of Primary Containment radioactivity due to transport of air through leakage and is constant for all isotopes.

$F_{nV}(t)$ = Buildup factor of radioactivity in the n-th SGTS filter due to Primary Containment leakage and is constant for all isotopes.

$C_{1i}(0)$ = Airborne activity concentration in Primary Containment of a certain isotope at time (t=0).

ORIGEN computer code calculates isotope decay and daughter product generation and is used to compute $F_{iR}C_{1i}(0)$. The method of calculating $F_{1V}(t)$ and $F_{3V}(t)$ is developed as follows:

Ignoring radioactivity decay and daughter product generation, the activity balance in Primary Containment:

$$\frac{d}{dt}(C_{1i}V_1) = -Q_1C_{1i} \qquad \text{(D-3)}$$

Initial conditions:

$$\text{at } t=0,\ C_{1i} = C_{1i}(0) \qquad \text{(D-4)}$$

The solution of equation (D-3) becomes:

$$C_{1i}(t) = C_{1i}(0)e^{-(Q_1/V_1)t} \qquad \text{(D-5)}$$

Total activity balance on SGTS filters (Ignoring daughter product generation and radioactivity decay):

$$\frac{d(C_{Ni}V_N)}{dt} = I_nQ_1C_{1i} \qquad \text{(D-6)}$$

Initial conditions:

$$\text{at } t=0,\ C_{ni}=0 \qquad \text{(D-7)}$$

The solution to equation (D-6) becomes

$$C_{ni} = \frac{I_n V_1}{V_n} C_{1i}(0) \left[1-e^{-(Q_1/V_1)t} \right] \qquad \text{(D-8)}$$

Defining:

$$F_{NV}(t) = \frac{C_{Ni}}{C_{1i}(0)}$$

$$F_{nv}(t) = \frac{I_n V_1}{V_n} \left[1-e^{-(Q_1/V_1)t} \right] \qquad \text{(D-9)}$$

Where $C_{1i}(t)$ = Gamma activity concentration in Primary Containment of the i-th energy level (photons/sec-cm^3).

$C_{ni}(t)$ = Gamma activity concentration of the i-th energy level in n-th SGTS filter (photons/sec-cm^3).

I_n = Halogen fraction to be deposited in the n-th SGTS filter.

Q_i = Air leakage rate from Primary Containment (cm^3/sec).

t = Time after accident (sec).

V_1 = Volume of Primary Containment (cm^3).

V_n = Volume of n-th SGTS filter segment (cm^3).

Substituting equation (D-9) into equation (D-2) and defining

$$X_{1i}(t) = F_{iR}(t)\, C_{1i}(0) \qquad \text{(D-10)}$$

Equation (D-2) becomes:

$$C_{Ni}(t) = \frac{I_N V_1}{V_N} \left[1\text{-}\exp^{-(Q_1/V_1)t}\right] X_{1i}(t) \qquad \text{(D-11)}$$

The ORIGEN computer code was used to calculate $X_{1i}(t)$. The ORIGEN run result is shown in Table D-1.

D.3 Calculation of Activity Concentration in the SGTS Filters

After the activity concentration in each filter segment is determined, the gamma radiation dose for safety-related equipment located in the SGTS filter room is determined by the use of Computer Code QAD-P5A (Ref. D-4). The QAD-P5A modelling procedure as described in Appendix C is followed for this analysis. The following modelling assumptions were used:

1. Self shielding of the filters are conservatively neglected.

2. Shielding due to the sheet metal filter housing are conservatively neglected.

Assumption 1 is made because the density of the charcoal dust or the wire mesh (Prefilter and HEPA filters) in the filters is low.

Neglecting the self shielding effect of the filters will not add too much conservatism to the results.

Assumption 2 is made because of computer code stability considerations. The shielding effect of the thin sheet metal filter housing is negligible.

Five representative targets were chosen for direct dose analysis in the SGTS filter room. The safety-related equipment chosen are:

1. SGT-TC-1A1 (TC-2-1)
2. SGT-MO-3A1
3. SGT-AO-2A
4. SGT-DV-1B2
5. SGT-FT-1A1

These targets are chosen according to their proximity to the SGTS filters. The relative locations of the targets are shown in Figure D-1.

The time and energy dependent gamma ray activity concentration as calculated using the method described in Section D.2 was used as input to the QAD-P5A model described above. The dose rate results as an output of this analysis were integrated numerically to give time-dependent integrated dose. Table D-2 shows the direct gamma dose rate and integrated dose results of the five targets.

Table D-1

Total Gamma Activity of the Released Airborne Halogens

Mean Energy (Mev)	Gamma Activity (Photons/sec)												
	0 hr	0.0333 hr	0.1 hr	0.333 hr	1.0 hr	3.0 hr	9.0 hr	24 hr	72 hr	216 hr	1440 hr	4320 hr	8760 hr
0.30	1.71E18	1.55E18	1.42E18	1.36E18	1.28E18	1.27E18	1.38E18	1.25E18	8.61E17	5.00E17	6.06E15	4.22E11	5.48E07
0.63	1.14E19	1.13E19	1.10E19	9.92E18	7.54E18	4.05E18	1.83E18	9.26E17	2.49E17	5.71E16	6.90E14	6.02E10	3.73E10
1.10	3.59E18	3.16E18	2.90E18	2.69E18	2.30E18	1.62E18	8.08E17	2.09E17	1.45E16	1.24E14	2.58E07	5.21E06	4.42E05
1.55	2.56E18	9.15E17	8.50E17	7.65E17	5.91E17	3.41E17	1.39E17	2.76E16	4.01E14	1.27E13	4.63E02	0	0
1.99	1.87E17	1.87E17	1.85E17	1.78E17	1.63E17	1.25E17	6.21E16	1.27E16	8.83E13	3.00E07	0	0	0
2.38	1.36E17	4.70E16	6.90E15	1.21E15	7.80E14	1.81E14	2.87E13	6.98E11	4.82E06	1.59E-9	0	0	0
2.75	2.15E17	7.69E16	2.27E16	1.22E16	5.39E15	5.94E14	1.13E13	5.21E10	3.43E05	1.13E-10	0	0	0
3.25	2.86E16	1.05E16	1.42E15	1.20E13	9.68E12	1.04E12	4.61E08	1.78E00	0	0	0	0	0
3.70	2.97E16	1.66E16	1.19E16	8.62E15	3.59E15	2.62E14	1.03E11	3.11E02	0	0	0	0	0
4.22	4.16E16	9.52E15	4.63E14	1.19E10	8.54E-4	0	0	0	0	0	0	0	0
4.70	7.29E16	1.67E16	8.09E14	2.08E10	1.48E-3	0	0	0	0	0	0	0	0
5.25	0	0	0	0	0	0	0	0	0	0	0	0	0

Table D-2

Direct Gamma Dose Rate and Integrated
Dose Results For Targets In The SGTS Room

Time (hr)	TARGET TC 2 - 1 Dose Rate (Rad/hr)	TARGET TC 2 - 1 Integrated Dose (Rads)	TARGET MO-3A1 Dose Rate (Rad/hr)	TARGET MO-3A1 Integrated Dose (Rads)	TARGET SGT-AO-2A Dose Rate (Rad/hr)	TARGET SGT-AO-2A Integrated Dose (Rads)	TARGET SGT-DV-1B2 Dose Rate (Rad/hr)	TARGET SGT-DV-1B2 Integrated Dose (Rads)	TARGET SGT-FT-1A1 Dose Rate (Rad/hr)	TARGET SGT-FT-1A1 Integrated Dose (Rads)
0.0	0	0	0	0	0	0	0	0	0	0
0.1	6.40E3	3.27E2	3.25E2	1.66E1	6.96E1	3.57E0	1.33E1	6.82E-1	1.85E2	9.45E0
1.0	4.62E4	2.52E4	2.34E3	1.28E3	5.02E2	2.74E2	9.60E1	5.24E1	1.33E3	7.27E2
3.0	8.47E4	1.56E5	4.30E3	7.93E3	9.23E2	1.70E3	1.76E2	3.25E2	2.45E3	4.51E3
9.0	1.31E5	8.03E5	6.65E3	4.08E4	1.43E3	8.74E3	2.73E2	1.67E3	3.78E3	2.32E4
24.0	1.67E5	3.03E6	8.46E3	1.54E5	1.81E3	3.30E4	3.47E2	6.32E3	4.81E3	8.76E4
72.0	1.83E5	1.14E7	9.30E3	5.81E5	1.99E3	1.24E5	3.81E2	2.38E4	5.29E3	3.30E5
216.0	2.37E5	3.94E7	1.20E4	2.00E6	2.58E3	4.29E5	4.93E2	8.20E4	6.84E3	1.14E6
120.0	1.17E5	1.29E8	5.95E3	6.53E6	1.27E3	1.40E6	2.44E2	2.68E5	3.38E3	3.71E6
1440.0	1.60E4	1.76E8	8.14E2	8.97E6	1.74E2	1.92E6	3.33E1	3.67E5	4.62E2	5.09E6
2160.0	1.71E3	1.83E8	8.68E1	9.29E6	1.86E1	1.99E6	3.56E0	2.81E5	4.93E1	5.28E6
4320.0	2.40E0	1.85E8	1.22E-1	9.38E6	2.61E-2	2.01E6	5.00E-3	3.84E5	6.93E-2	5.33E6
8760	0	1.85E8	0	9.38E6	0	2.01E6	0	3.84E5	0	5.33E6

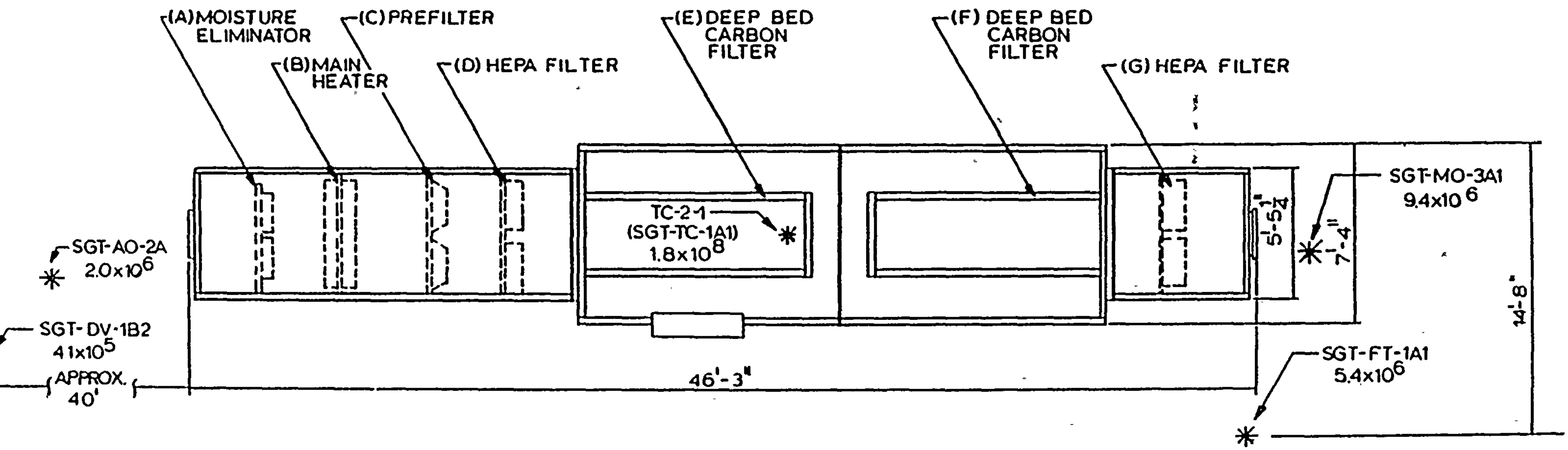
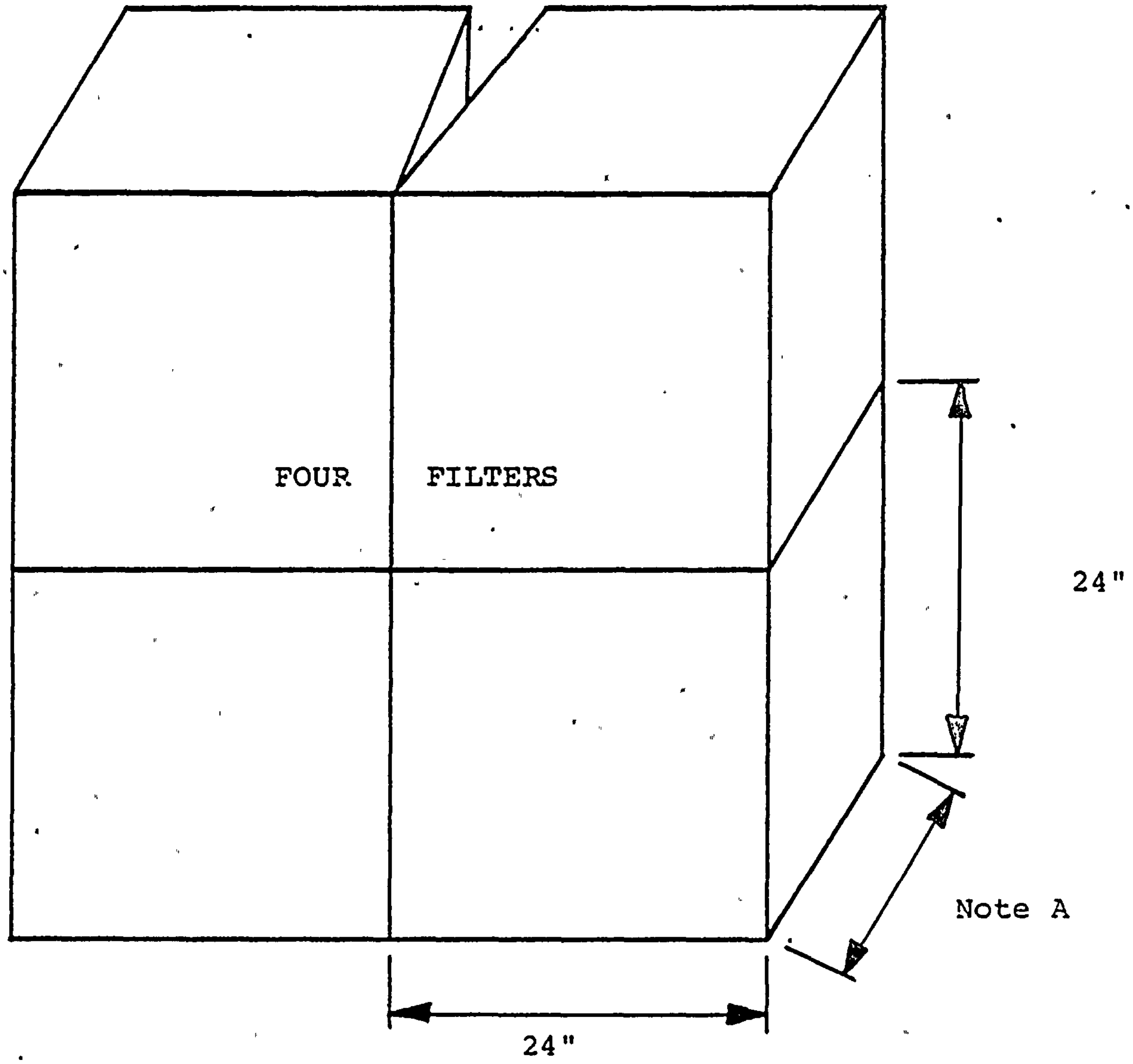
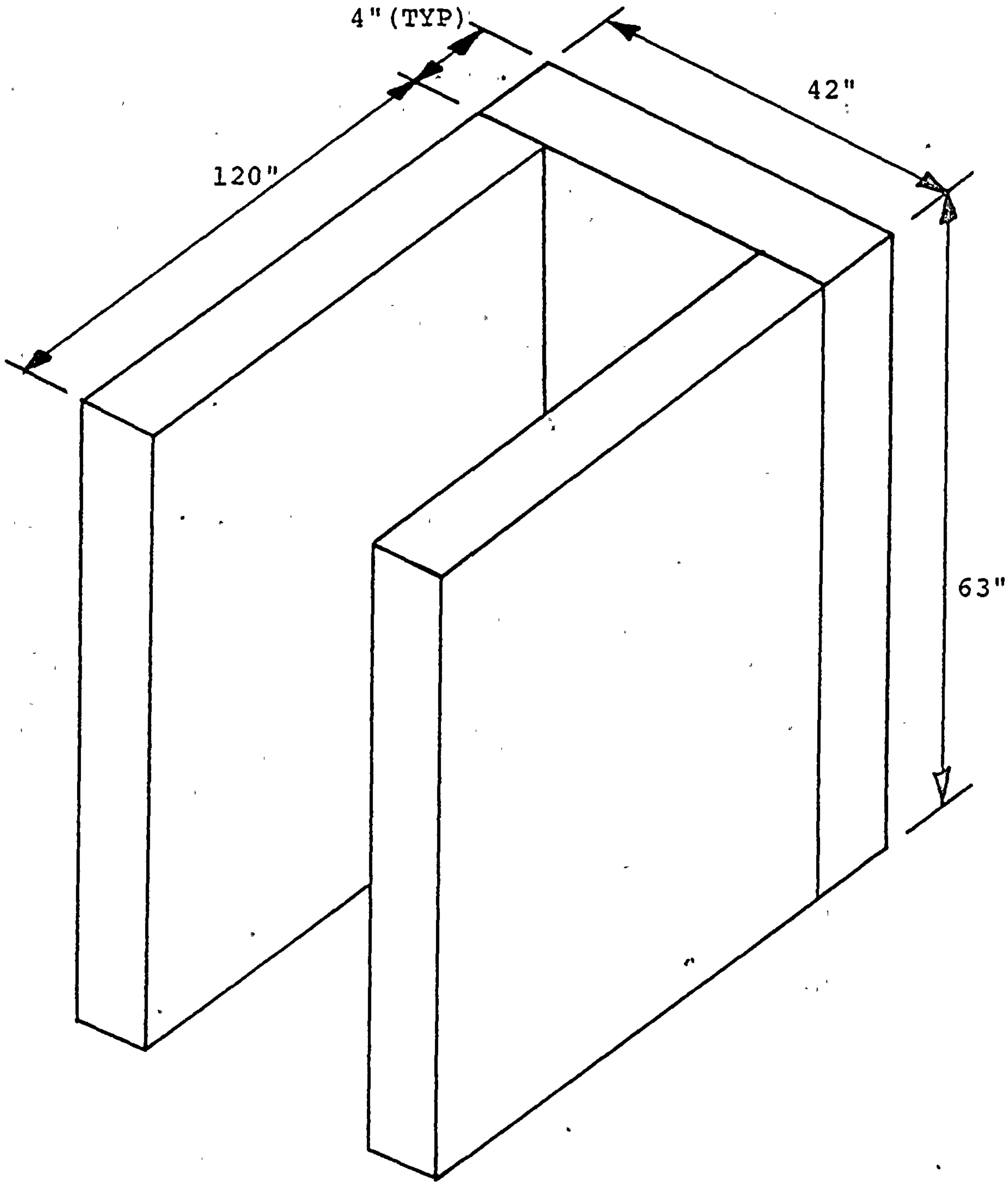


Figure D-1 Standby Gas Treatment Filter



NOTE A: Prefilter 8"
HEPA Filter 11½"

Figure D-2 Geometry Of Prefilters And HEPA Filters



Not to Scale
Figure D-3 Geometry of Charcoal Filters

APPENDIX E:

The source volume used for the beta dose analysis in Secondary Containment is a sphere surrounded by a shell of sufficient thickness to stop all outside beta particles from entering the source volume. This spherical source volume is conservative for any generalized source volume shape (the dose at the center of the sphere is higher than the dose at any point of any generalized source of equal total volume).

Justification of Assumptions

The assumptions used for the beta analysis in Secondary Containment are presented in Section 5.5.2.

Assumption 1 is conservative because there will be some actual delay in transport of the gaseous fission products into the equipment.

Assumptions 2 and 3 are based on NUREG-0588, Revision 1. (Ref. E-1).

Assumption 4 is conservative and was required by the use of ORIGEN as a source code.

Assumption 5 is consistent with the assumptions established in Reference E-2.

Assumption 6 is conservative because it represents the minimum flowrate of the SGT system (with the SGT system running and the flow-balancing dampers set at the minimum flowrate) (Ref. E-3).

Assumption 7 is consistent with the NRC-recommended assumptions used for the calculation of doses inside Primary Containment (Ref. E-4).

Assumptions 8 and 9 are conservative.

The first portion (Step 1) of the calculation is the determination of the air dose at the center of the spherical source as a function of the volume of the sphere. The second portion (Step 2) of the calculation is the determination of the equipment dose as a function of the air dose.

Step 1:

The variation of beta dose rate from a typical beta energy distribution in a one-dimensional absorbing medium can be approximated by the formula:

$$D(X) = A \exp(-\mu_E X) \qquad \text{(E-1)}$$

where

$D(X)$ is the dose at a point X
A is a constant

X is the position in the material
μ_E is a parameter that depends on beta energy.

This relationship holds approximately up to the point where all beta particles are absorbed. This point is called the range of the beta particles. The range of a beta particle is dependent upon the energy of the beta particle and is denoted r_E.

Both of the parameters μ_E and r_E may be determined by empirical formulas given below (based on the maximum energy of the beta

particles) and approximately independent of the absorbing medium:

$$\mu_E = 17\rho(E\ max)^{-1.14} \qquad (E-2)$$

and

$$r_E = (0.412/\rho)\ E^n \text{ for } .01 \leq E \leq 3 \qquad (E-3)$$

$$r_E = (0.530E - .106)/\rho \text{ for } 2.3 \leq E \leq 20 \qquad (E-4)$$

where

ρ is material density (in g/cm3)
E is energy of beta particle (in MeV) 0
μ_E is in cm^{-1}
r_E is in cm
n is defined as 1.265 - .0954 LnE

The dose at a given point from a single beta source is now transformed into a dose from a uniform concentration of airborne sources which extend from radius zero to radius r. This relation is found to be:

$$D(r) = K(1-\exp\ (\mu_E r)) \qquad (E-5)$$

where K is a constant

This relationship is valid for $r \leq r_E$. At $r \geq r_E$, none of the beta particles originating beyond r_E reach the target point. Hence, at this radius, an effective infinite medium for

airborne beta radiation has been reached. The dose from a volume such that $r \geq r_E$ is equal to the dose from an infinite volume, which is denoted D_∞ .

The dose as a function of volume radius is thus found to be given by the dual relation:

$$D(r) = D_\infty \frac{(1-\exp(-\mu_E r))}{(1-\exp(-\mu_E r_E))} \quad 0 \leq r \leq r_E \qquad \text{(E-6)}$$

This relation may be transformed to a function of volume by noting that $V = 4\pi r^3/3$.

Since μ_E and r_E vary for each beta energy, this equation cannot be solved analytically for the case of a mixture of many beta energies - which is the case at hand. However, since D_∞ for each beta energy is known (from the calculation of the semi-infinite source), $D_{E(v)}$ for each beta energy at a given volume may be determined. All contributions to the total dose at a given volume are then added together.

The largest volume to be considered in this calculation was determined by finding the volume of sphere of radius equal to the beta particle with the largest range. This led to a maximum volume of 10^{11} cc. A minimum volume of 10 cc was arbitrarily chosen as the lower limit to the volumes to be calculated. Volumes of 10^3, 10^5, 10^7, and 10^9 cc were also chosen. The dose rates and integrated six-month doses for volumes of these sizes were calculated, and the results are plotted in Figures E-1 through E-6. These figures

are the reduction in air dose from the semi-infinite medium air dose.

Step 2:

The absorbed beta dose within a physical target is not always equal to the beta dose at a mathematical point in air at the surface of that piece of equipment. The beta ionization energy (dose) deposited on the surface of a solid object is distributed in a thin surface layer to a depth equal to the beta range in the material. The absorbed dose rate within the affected layer is equal to the average dose rate within the layer. The ratio of the absorbed dose to the air dose at the equipment surface is found to be:

$$D(abs)/D(0) = (1-e^{-\mu_E r_E})/r_E \mu_E$$

The value of this ratio is constant for a given energy group. For the beta energy distribution found in this calculation, this ratio is found to lie within the values .137 to .204. Thus, air doses (Figures E-7 and E-8) are conservatively multiplied by the factor 0.204 to obtain beta doses to equipment. Airborne beta doses to equipment enclosed in small volumes are plotted as a function of volume size in Figure E-9.

The dose at a given point from a single beta source is now transformed into a dose from a uniform concentration of airborne sources which extend from radius zero to radius r.

This dose at a given point from a differential source volume will be:

$$dD(r) = C \frac{Sdr}{4\pi r^2} \left[\frac{D(r)}{D(o)}\right] \tag{E-8}$$

Where C is a constant conversion factor to be defined later. The first term indicates the dispersion of beta energy from a point source without absorption. The second term (D(r)/D(0)) indicates the effect of distance for shielding.

Integrating gives (for a point at the center of a spherical volume) with $dV=r^2 dr d\Omega$:

$$D(r) = C\ Si \int_0^{4\pi} \int_0^{r} \frac{1}{4\pi r^2} \left[\frac{D(r)}{D(0)}\right] d\Omega r^2 dr$$

Substituting the formula for D(r) gives:

$$D(r) = C\ Si \int_0^{4\pi} \frac{d\Omega}{4\pi} \int_0^{r} \frac{A \exp(-\mu_E r)}{A \exp(-\mu_E o)} dr$$

$$D(r) = C\ Si \int_0^{4\pi} \frac{d\Omega}{4\pi} \int_0^{r} \exp(-\mu_E r)\, dr \tag{E-9}$$

Solving gives:

$$D(r) = \frac{C\ Si}{\mu_E} (1-\exp(-\mu_E r))$$

$$D(r) = K (1-\exp(-\mu_E r)) \tag{E-10}$$

where K is a constant and r is the extent of the source volume.

The absorbed beta dose within a physical target is not always equal to the beta dose at a mathematical point in air at the surface of that piece of equipment. The beta ionization energy (dose) deposited on the surface of a solid object is distributed in a thin surface layer to a depth equal to the beta range in the material. The absorbed dose rate within the affected layer is equal to the average dose rate within the layer. The ratio of the absorbed dose to the air dose at the equipment surface is derived as follows:

$$E_i = \overline{D}_i r_i \qquad \overline{D}_i = D_i \text{ (abs)} \tag{E-11}$$

where:

E_i = total energy deposited in the affected layer

$\overline{D}_i$ = average dose rate at a point in the affected layer

r_i = beta range in the material

i = beta energy group

D_i (abs) = absorbed dose within the target

We also know that:

$$E_i = \frac{D_i(0)}{\mu_i} \quad (1-e^{-\mu_i r_i}) \qquad \text{(E-12)}$$

where:

$D_i(0)$ = dose rate at the air-equipment boundary
$D_i(0)$ = dose rate in air

Substituting the first equation into the second gives:

$$D_i(\text{abs})\ r_i = (D_i(0)/\mu_i) \quad (1-\exp(-\mu_i r_i)) \qquad \text{(E-13)}$$

and

$$D_i(\text{abs})/D_i(0) = (1-\exp(-\mu_i r_i))/\mu_i r_i \qquad \text{(E-14)}$$

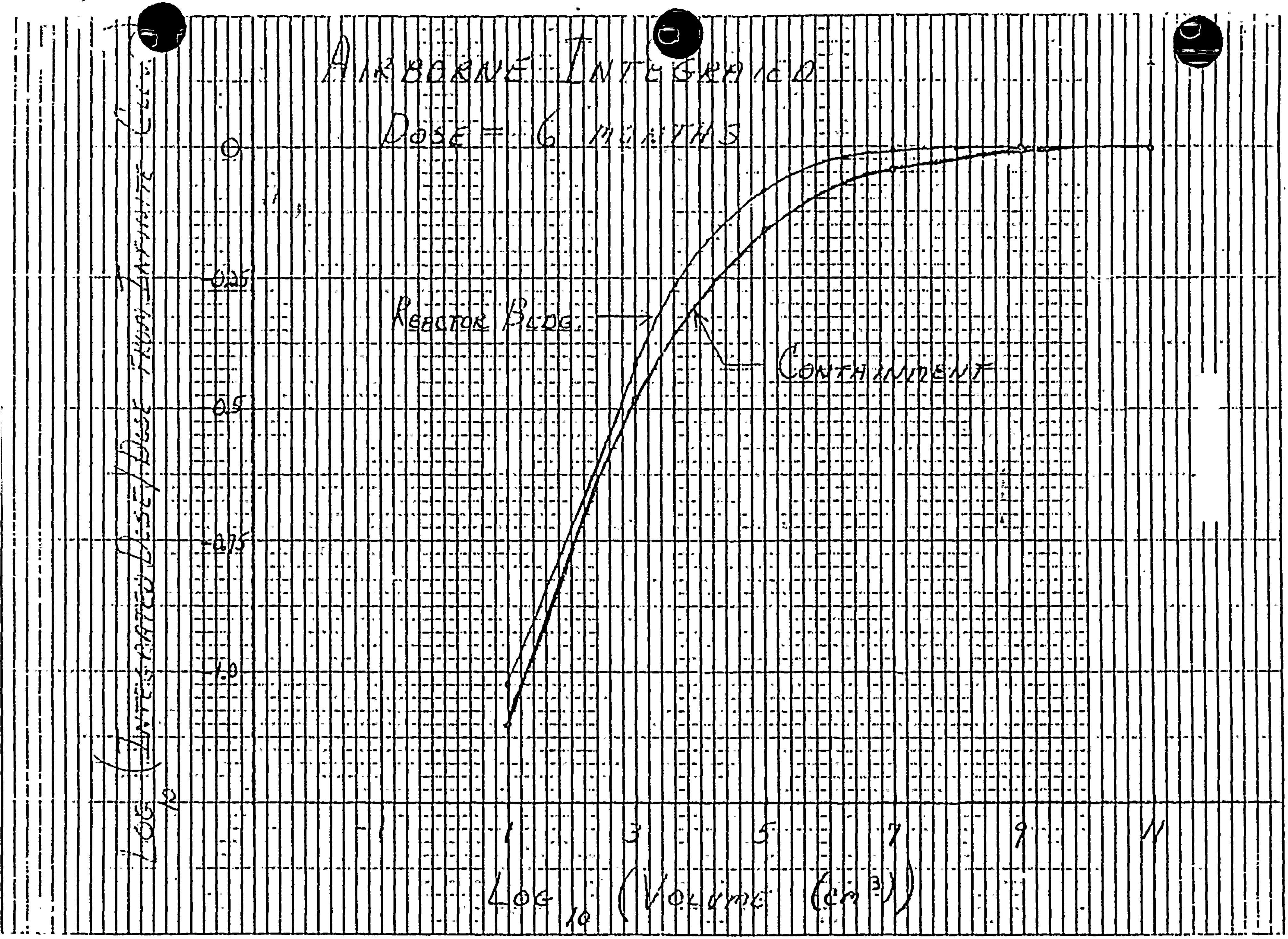
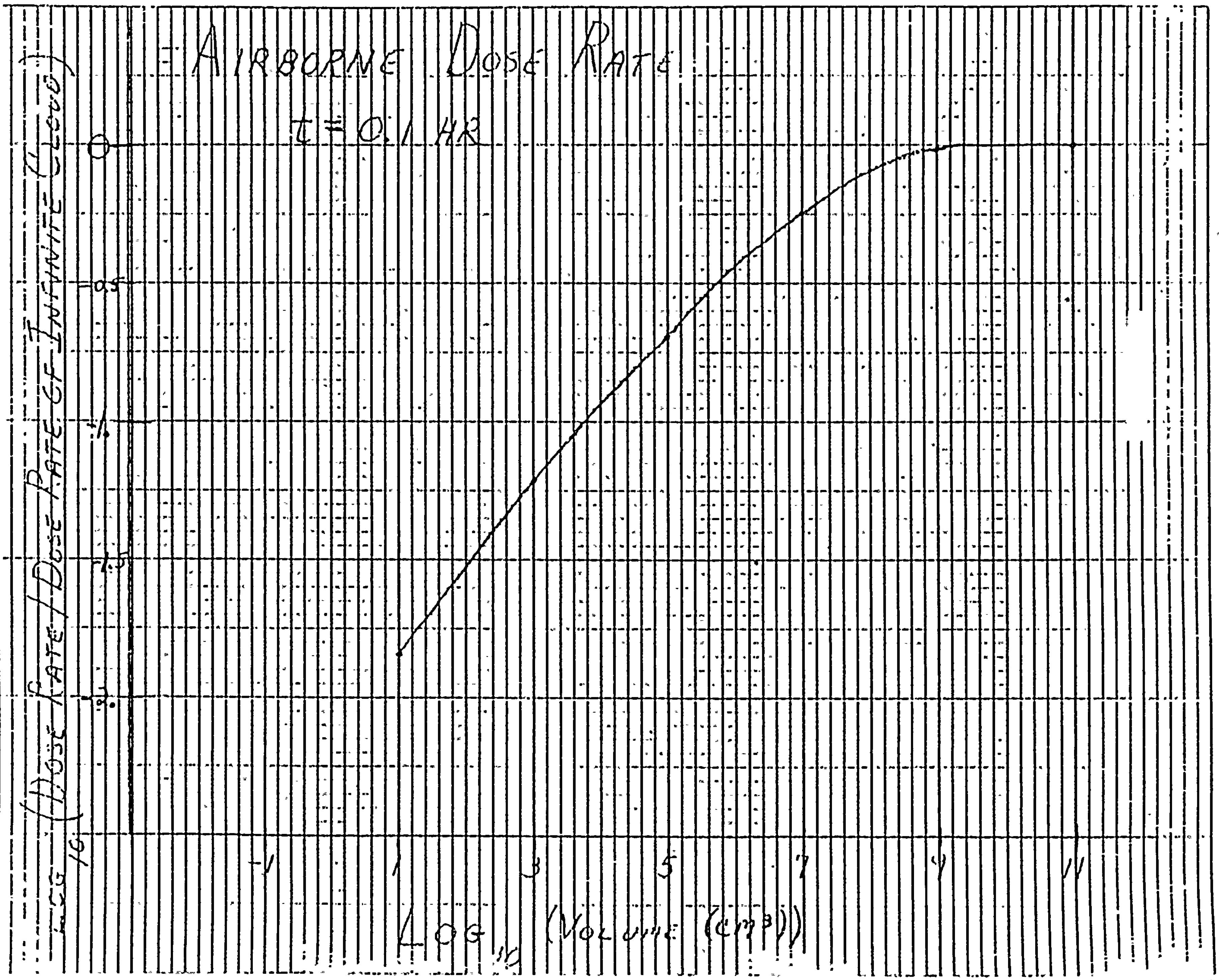
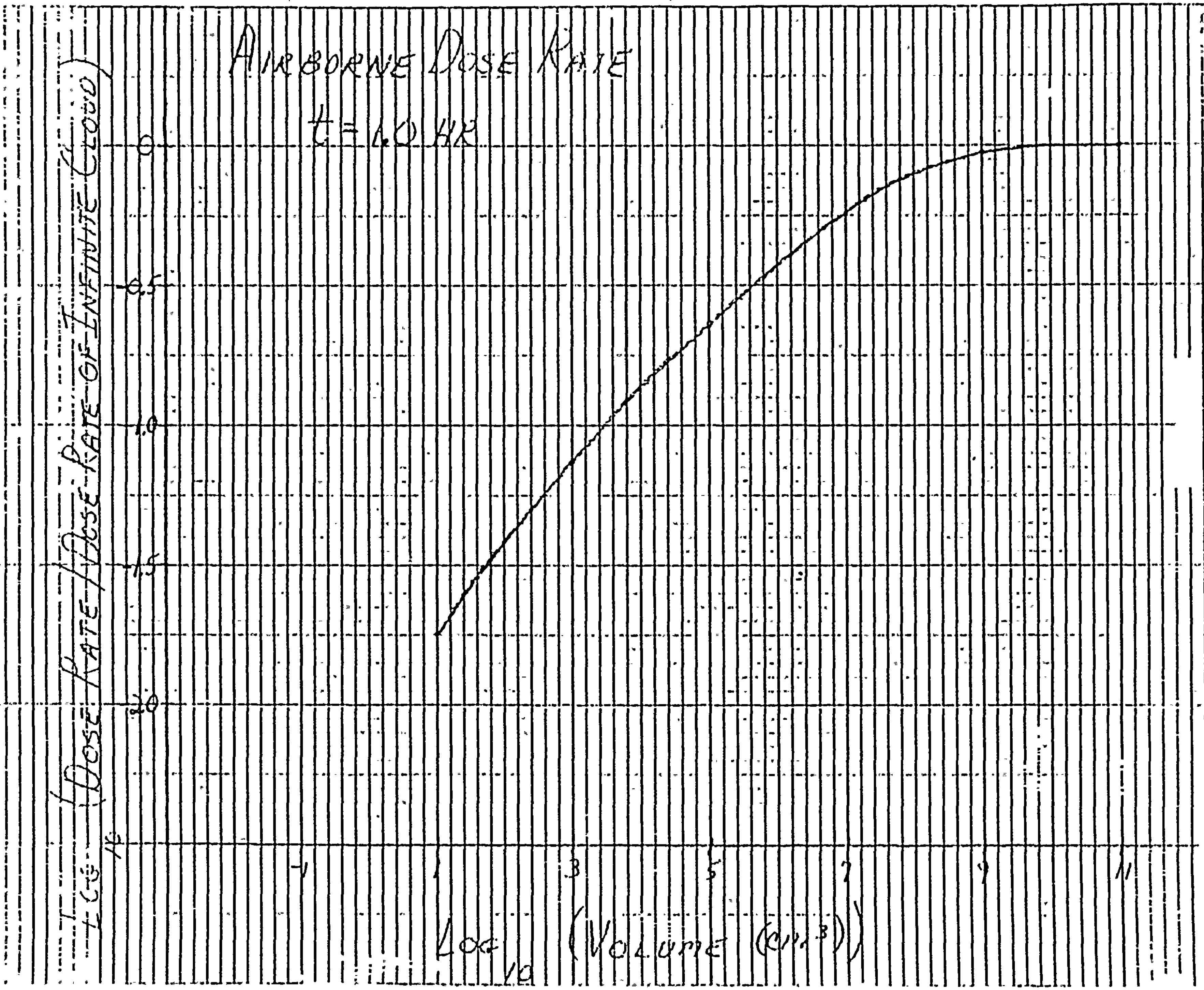


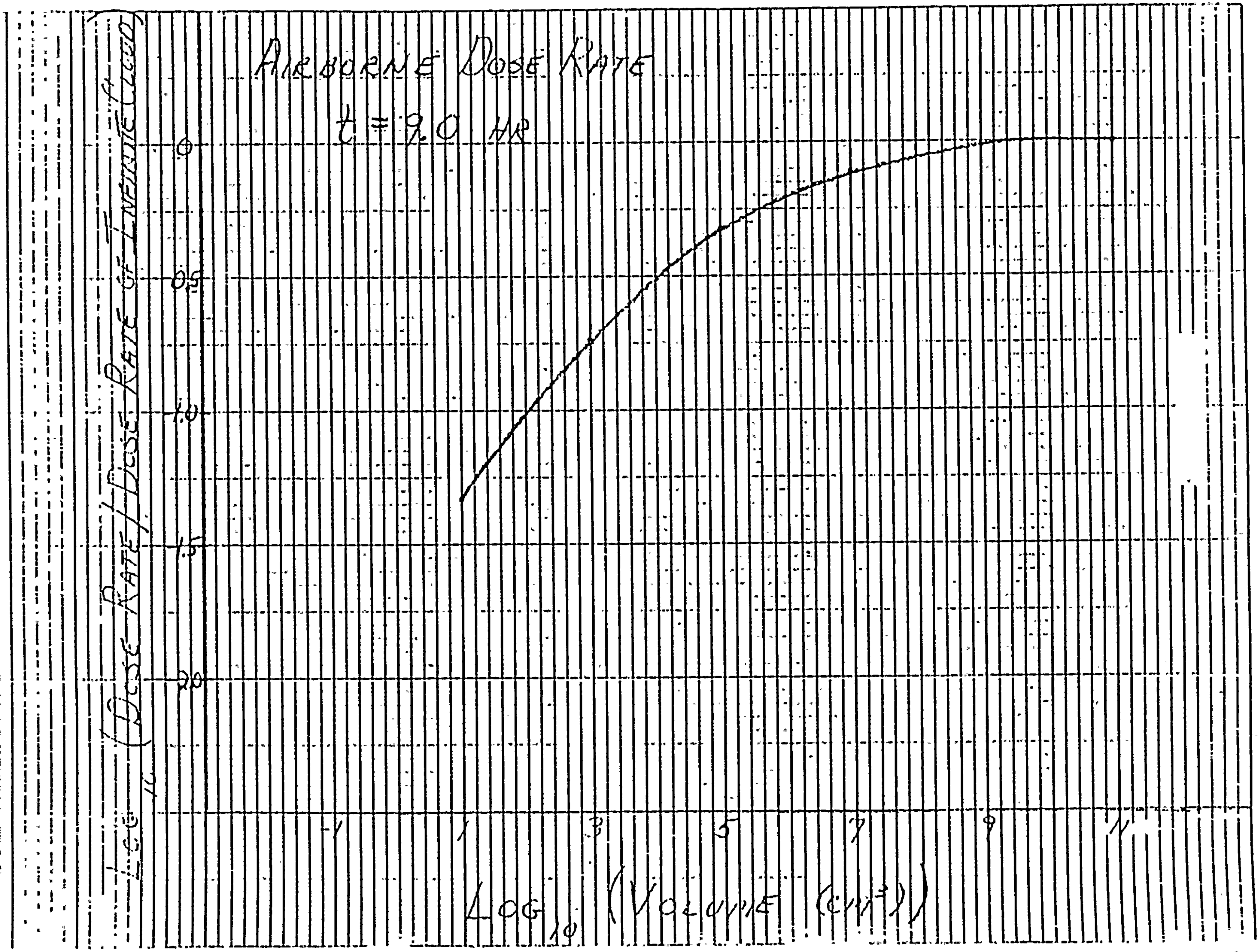
Figure E.1 Six Month Airborne Integrated Beta Dose

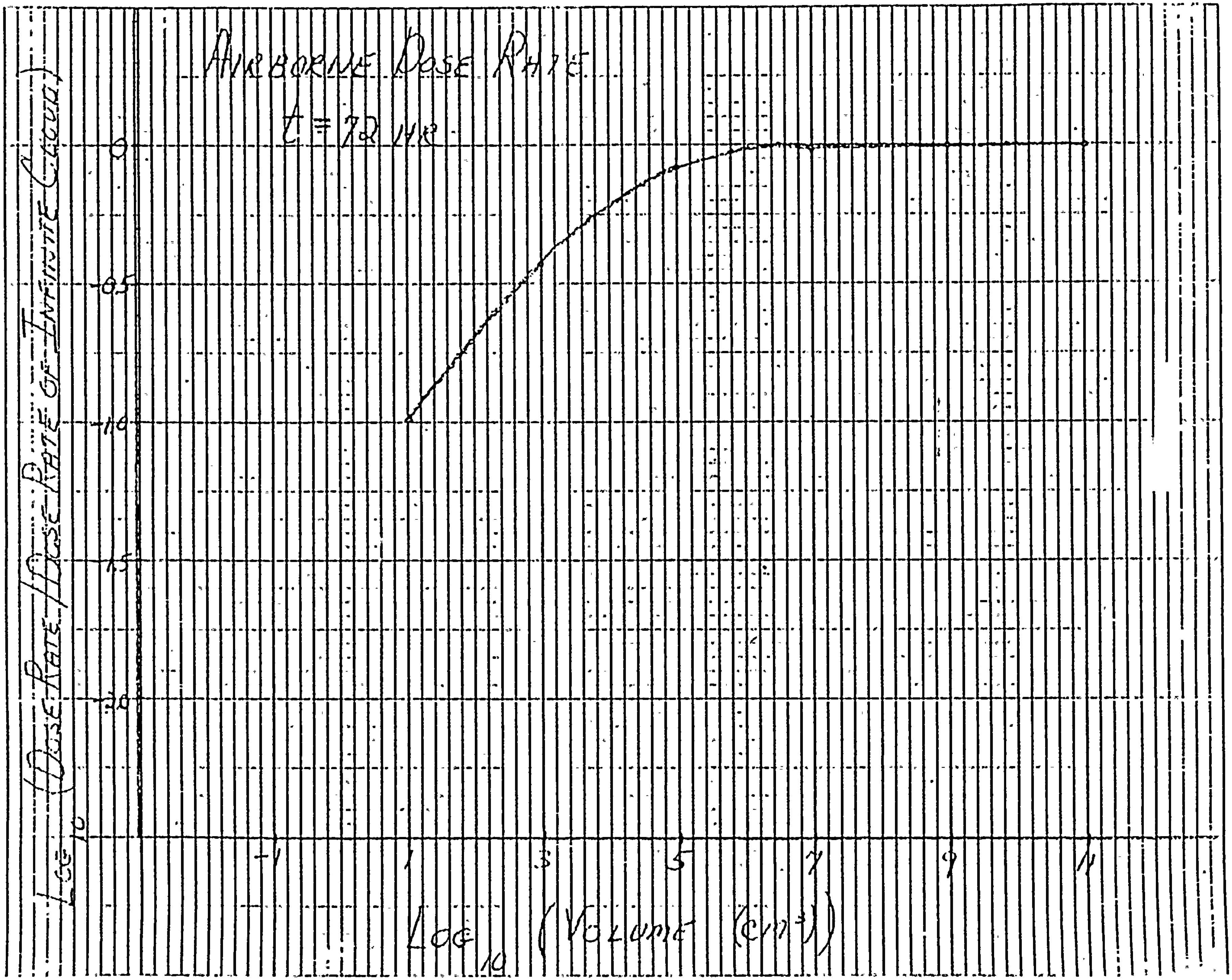
K+E 10 X 10 TO THE INCH • 7 X 10 INCHES KEUFFEL & ESSER CO.

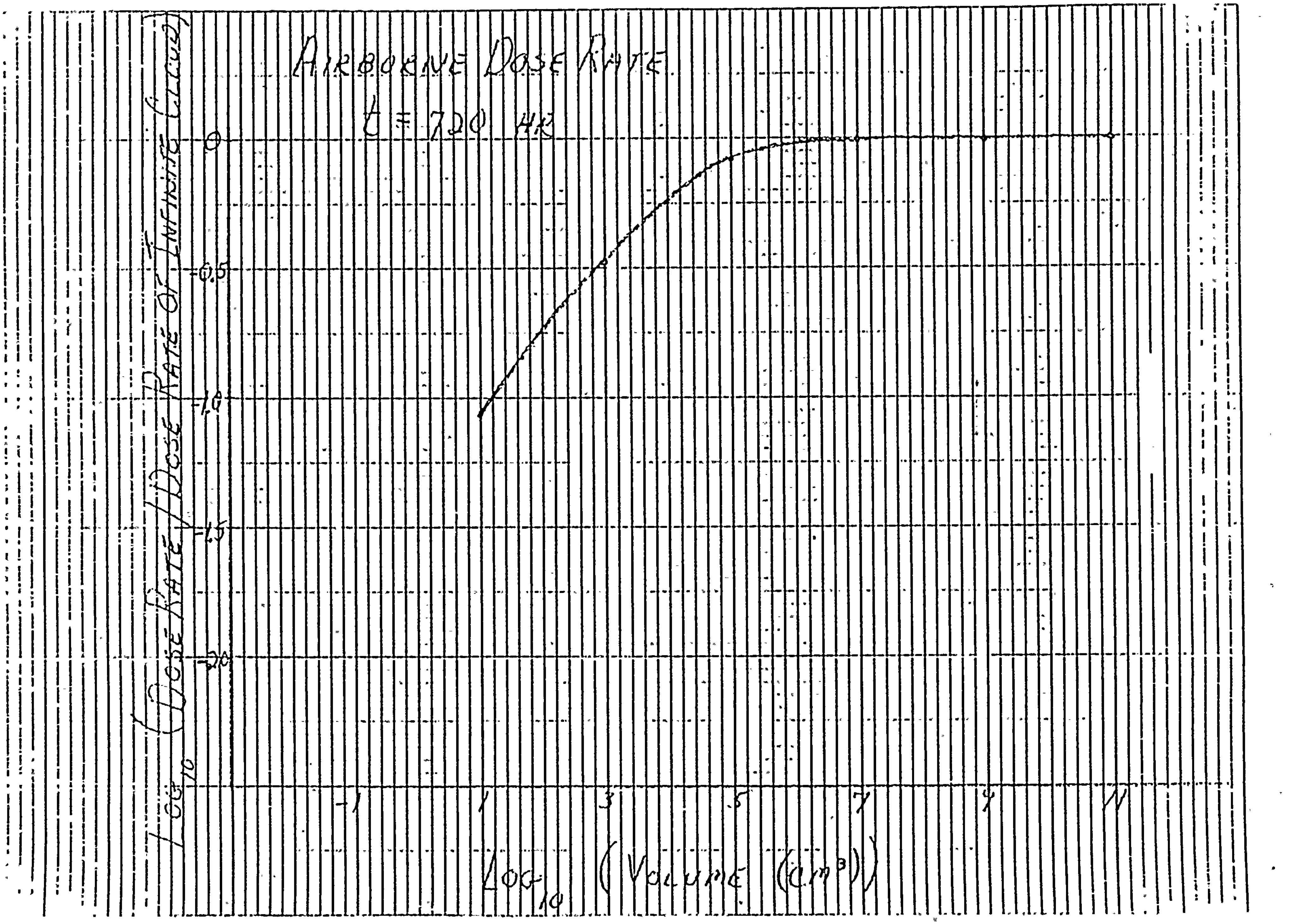
46 0782

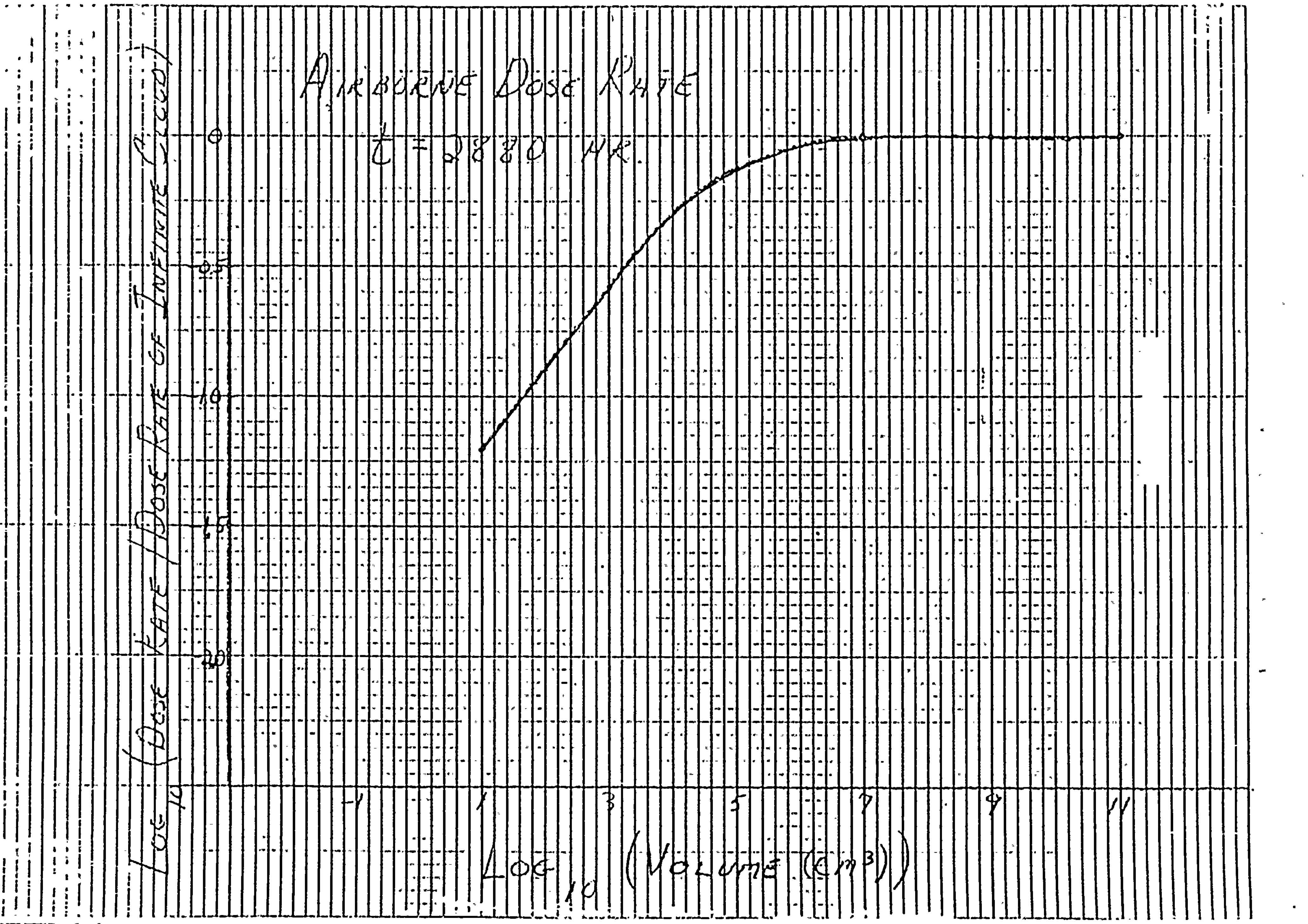



AIRBORNE DOSE RATE
t = 6.0 HR
0
-0.5
-1.0
-1.5
-2.0
1
3
5
7
9
11
LOG10 (DOSE RATE / DOSE RATE OF INFINITE CLOUD)
LOG10 (VOLUME (cm.3))

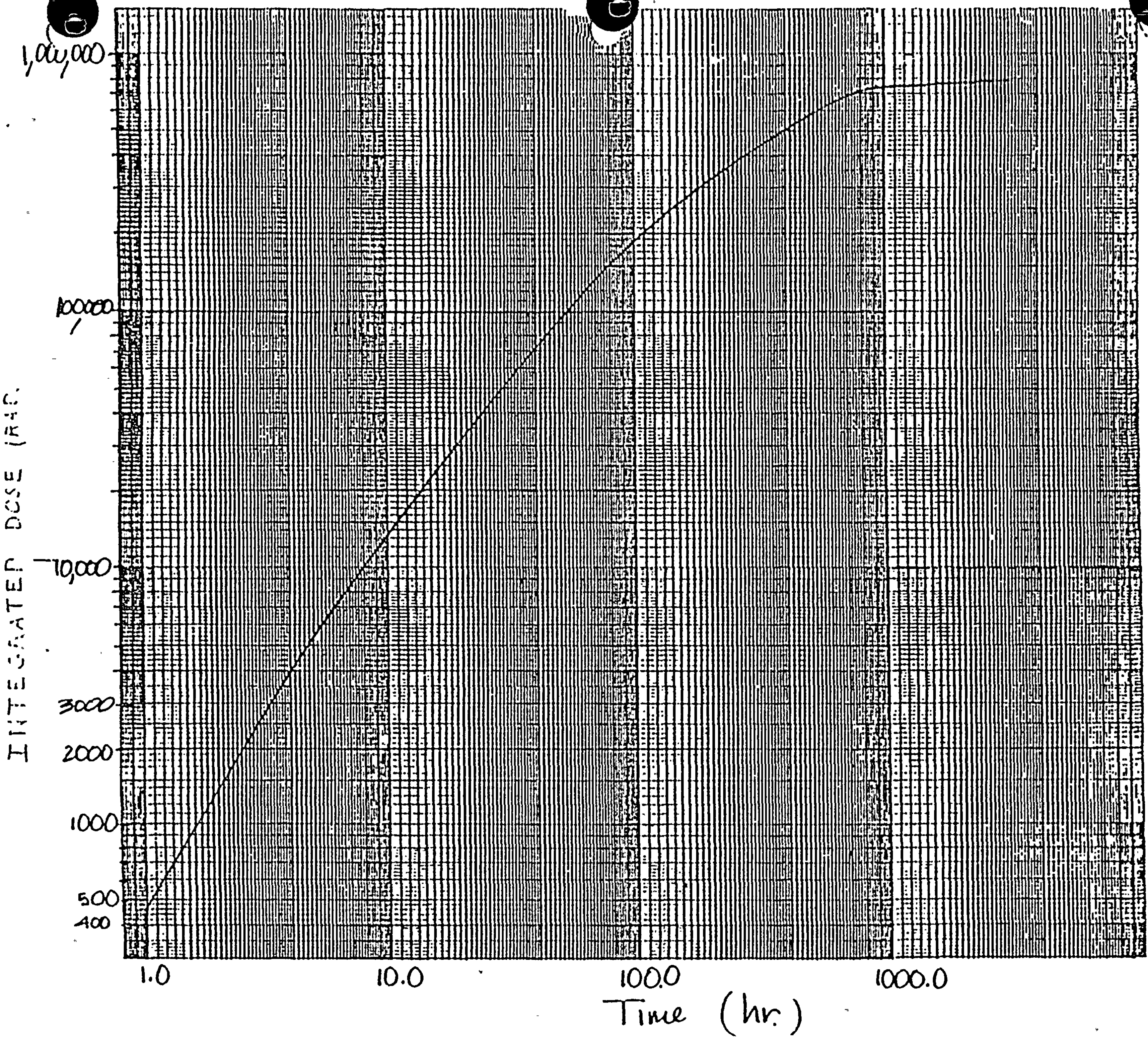

AIRBORNE DOSE RATE
t = 20 HR
0
.5
1.0
1.5
2.0
-1
1
3
5
7
9
11
LOG10 (DOSE RATE / DOSE RATE OF INFINITE CLOUD)
LOG10 (VOLUME (cm³))


AIRBORNE DOSE RATE
t = 72 HR
0
-0.5
-1.0
-1.5
-2.0
LOG10 (DOSE RATE / DOSE RATE OF INFINITE CLOUD)
-1
1
3
5
7
9
11
LOG10 (VOLUME (cm³))


AIRBORNE DOSE RATE
t = 720 HR
0
-0.5
-1.0
-1.5
-2.0
LOG10 (DOSE RATE / DOSE RATE OF INFINITE CLOUD)
-1
1
3
5
7
9
11
LOG10 (VOLUME (cm³))


AIRBORNE DOSE RATE
t = 2880 HR
0
-.5
-1.0
-1.5
-2.0
-1
1
3
5
7
9
11
LOG (DOSE RATE / DOSE RATE OF INFINITE CLOUD)
LOG10 (VOLUME (cm3))

Infinite Air Dose - [illegible]ctor Building β



Infinite Air Dose - Containment β

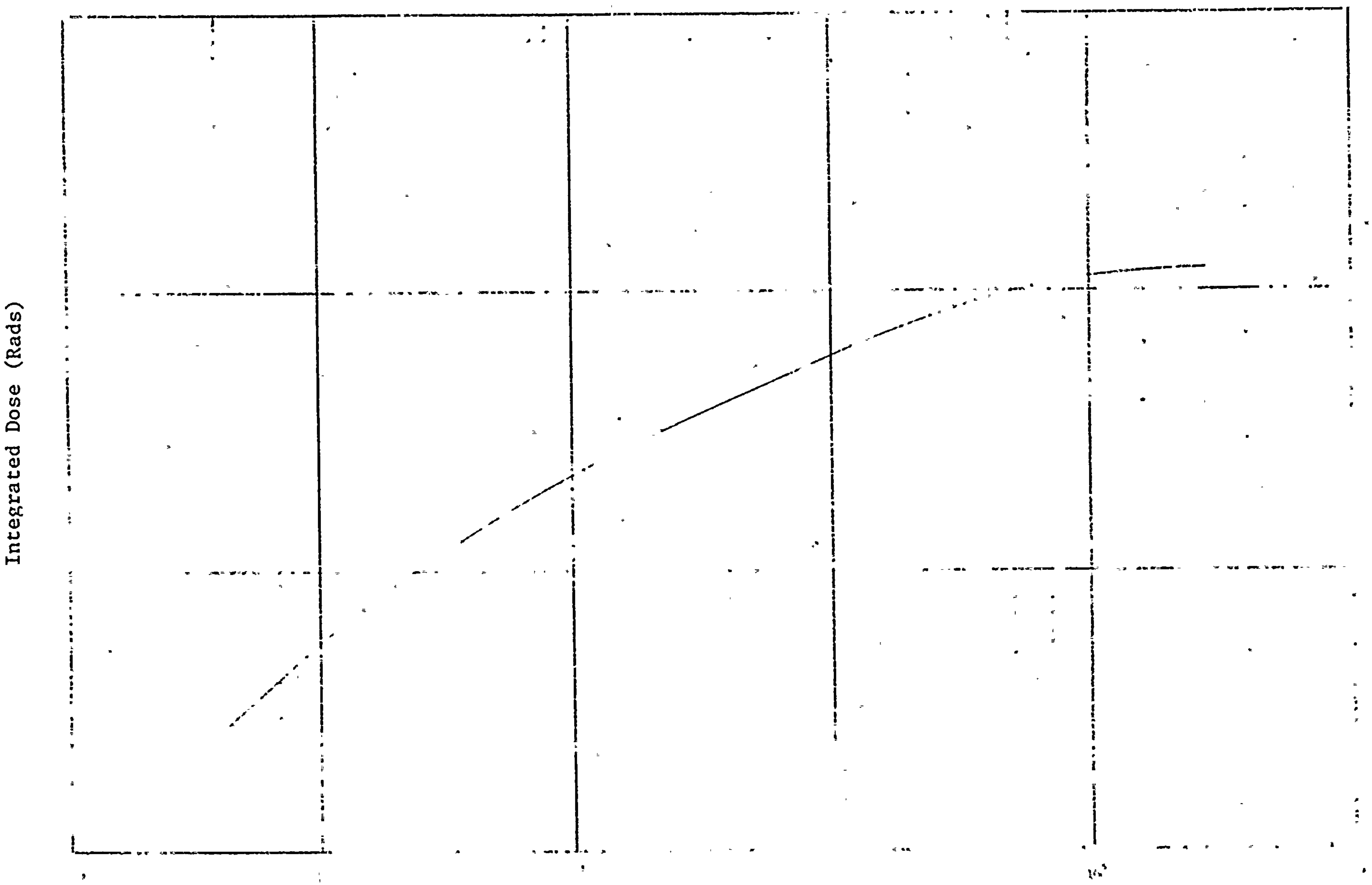


Figure E.9 Integrated Air Beta Doses Inside Containment

APPENDIX F: PRIMARY CONTAINMENT ANALYSES

For safety-related equipment located inside the primary containment, the most severe post-accident radiation environment is a postulated LOCA which completely depressurizes the primary system. Such an event maximizes the integrated dose due to the primary containment atmosphere and plate out source terms. This is added to the 40 year operation dose.

F.1 Core Source Term

The basic radiation source term, i.e., core inventory at time of the accident and during the following six month time period is being calculated with ORIGEN 2. A maximum nuclide core inventory was calculated using 105% full power (3481 MWt) and an operating time of 1,000 days.

Calculation of the basic source term will be done by generating the decay rate spectrum for solids, halogens and noble gases as a function of time following the postulated LOCA event. The fine time mesh interval spacing shown in Table F.1 was used. The decay rates, for each of seven energy groups, were integrated and an average value determined. In addition, a beta decay rate, averaged over six months, was calculated for each source type, i.e., solids, halogens, and noble gases. The results are presented in Table F.2.

F.2 Radiation Dose From Liquid Systems Due to Normal Operation

It was assumed that the primary radiation source in liquid systems, in the drywell, is due to the N-16 activity in the reactor water coolant.

From Table II.1-4 of the FSAR, the N-16 activity (max.) equals 40 Ci/gm. For a pipe segment of length L, the average source strength, $\overline{S}$, is given by

$$\overline{S} = S_i \left[1-\exp(-\lambda aL)\right]/(\lambda aL)$$

where L = length of pipe section (ft)
a = coolant transit rate (in pipe section) (Sec/Ft)

λ = N-16 decay constant = 0.09627 sec^{-1}

S_i = activity at entrance to pipe section

with

$$S_i = S_o \exp(-\lambda \Sigma_i aL)$$

where
S_o = initial N-16 activity
$\Sigma_i aL$ = sum over all previous pipe sections.

F.2.1 Liquid Systems Considered

The liquid systems which contribute the major radiation dose, inside primary containment, are

- RWCU and connecting RRC piping
- MS lines, and
- RHR lines during their period of operation.

For the RWCU, RRC and MS lines the analysis neglected the contributions from

fission products, corrosion products and other sources since these dose contributions are small compared to the N-16 contribution, during normal operation. For the RHR lines the radioactive source was the coolant source terms given in the FSAR Table 12.2-5. For conservatism, no decay was assumed beyond 4 hours after reactor shutdown.

F.2.2 Results

Dose rates from various RWCU and RRC pipe segments are shown in Figures F-1 to F-14. Dose rates are presented for different pipe lengths, since different detector (equipment) locations will "see" different pipe lengths.

Figures F-15 to F-17 present the dose rate from RHR piping. In calculating the integrated dose during the expected 40 year plant life, it is assumed that the RHR lines contribute 20% of the time.

Figure F-18 presents the dose rate from various lengths of a 26' Main Steam line due to N-16 at a level of 50 μCi/gm. Due to the relatively short transit time, decay was neglected, resulting in a conservative value.

F.2.3 Dose Contribution due to General Radiation Environment in Primary Containment

In addition to the dose rate from the various systems, liquid and gaseous, containing radioactive sources there is a general radiation environment due to normal reactor operation and the resultant neutron leakage from the reactor core. A fraction of the neutron

core leakage flux manages to penetrate the reactor pressure vessel, escaping into the reactor cavity. Some will traverse vertically while others will manage to penetrate the sacrificial shield wall. In addition to the resultant neutron dose rate, the neutrons interact with the material along their path generating secondary gamma rays.

The ANISN one-dimensional discrete ordinates computer code was used to calculate the transport of neutrons, generation of secondary gamma rays and their transport. The material and geometric configuration was that at core mid-plane.

The total dose rate* just beyond the outer steel liner of the sacrificial shield wall is 79 Rads/hour; 5 Rads/hour due to neutrons and 74 Rads/hour due to capture, prompt and fission product gammas. An estimate, based on geometric and material attenuation factors, was made** to determine the axial variation of the dose rate. The approximate dose rate reduction factor as a function of distance from core mid-plane is shown in Table F.3.

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* B&R Calc. 5.01.20

** B&R Calc. 5.01.97A

F.3 Dose Rate due to LOCA (Depressurized Primary System)

This section presents the methodology of analysis used to calculate the six-month integrated dose following a postulated loss-of-coolant accident (LOCA) in which the primary system is depressurized. In this scenario the radiation from the core was assumed to be immediately released into the containment atmosphere.

A detail description of the source and geometric analytical model assumptions follows in the next sections.

The sources considered, were

° containment atmosphere (gamma ray & beta)

- drywell
- wetwell air space

° plate-out (gamma ray & beta)

- drywell surfaces
- piping & equipment

° suppression pool water

F.3.1 Source Assumptions

To insure the maximum radiation environment for all relevant positions within containment, a non-mechanistic accident scenario was postulated.

Conservative source terms, as per the appropriate Reg. Guides & NUREG's, were

chosen for each contributing system, independently.

However, in calculating the radiation dose at a particular location, it was not necessary to assume that all source distribution assumptions were conservative simultaneously. Instead, a set of mutually compatible assumptions were used which gave the maximum dose for the location being considered. Thus the contribution of 50% core iodines in containment atmosphere plus 50% core halogens in the suppression pool were not added - when only a maximum grand total of 50% of the core halogens is released following a LOCA, (TID-14844, NUREG-0737 & II.B.2.(4)a).

F.3.2 Containment Atmosphere Source

(1) A nonmechanistic instantaneous release from the core inventory into the Primary Containment free volume of 100% noble gases and 50% halogens was assumed.

(2) When considering drywell and wetwell atmosphere sources, it was assumed that the activity released (in the drywell) was uniformly mixed between drywell plus wetwell air volumes.

(3) It was assumed, per NUREG 0588 Rev 1, that the iodine released to containment atmosphere begins diffusing to be deposited on

containment surfaces (plate-out source). This iodine plate-out source was used in containment atmosphere dose calculations.

(4) Assumed that 1% remaining solids are immediately contained within the suppression pool water.

(5) Containment atmosphere source contains 100% of the core noble gases and 50% of the core halogens.

(6) The maximum nuclide core activity was calculated using an operating time (operation at full power) of 1,000 days.

F.3.3 Suppression Pool Source

(1) Liquid in the suppression pool was assumed to contain the following percents of core inventory

0% noble gases,
50% iodines,
1% solid fission products

This assumption is from Reg. Guide 1.7 (Table 1.7 (Table D-1) based on TID-14844 (See NUREG-0588, Appendix D, §6).

(2) Source of (1) was assumed to be released into suppression pool.

(3) Uniform source distribution in suppression pool volume plus liquid volume of the reactor coolant system (RCS) was assumed.

F.3.4 Plate-Out Source

(1) It is assumed that 95.5% of released halogens diffuses to the primary containment surfaces.

(2) Plate-out occurs by diffusion in accordance with NUREG/CR-0009.

(3) All plate-out sources are retained, through the 6 month time period.

(4) Source density (Mev/cm^2-sec) was given by total source of (1) divided by plate-out surface, (5).

(5) Plate-out occurs on containment surfaces within drywell (i.e., incl. top and bottom of drywell, inner surface of biowall, inner and outer surfaces of sacrificial wall and outer surface of reactor pressure vessel insulation) uniformly. (See Figure F-19).

(6) For dose locations near piping (within drywell), plate-out on piping surface was considered. Source strength, per unit area, as given by (4) was used.

F.3.5 Drywell Atmosphere Dose Analysis

The dose rate inside the drywell, due to containment atmosphere, was calculated using the source terms and weighted by source type percentages, i.e., 100% noble gases, and 50% halogens. These were input to the QAD point-kernel computer code.

The truncated cone geometry is being mocked up by ~420 - 7X7X7 cubicles with the dose contribution from each section calculated independently, as a function of source energy.

For calculational purposes the drywell volume within the sacrificial shield was considered to be a solid steel cylinder. Scattering out this volume was neglected, as was albedo from sources outside the region. Material inside the drywell is being considered. To approximate the shadow shielding effect of the steel in the drywell, its mass in each 7X7X7 cubical is homogenized with the air to give an averaged density.

F.3.6 Plate-Out Dose Analysis

The iodines that plate-out are assumed to be on drywell walls, dome, floor, the inside and outside surfaces of the sacrificial shield wall, and the external surfaces of the RPV insulation. The plate-out dose to equipment is calculated from the surface activities taking credit for the intervening shadow shielding as described in Section F.3.5 above.

F.3.7 Wetwell Dose Analysis

Dose rate contributions are being calculated for wetwell atmosphere and suppression pool water sources for detector locations inside the drywell and wetwell. The general configuration is shown in Figure F-22.

F.4 Dose Due to Systems After a LOCA Condition

This section presents the methodology and results of analyses used to calculate the six-month integrated dose following a postulated LOCA, in which the primary system is depressurized, due

to various systems. The systems considered are:

- RHR
- HPCS
- LPCS
- RCIC
- RWCU
- MS
- RRC

The accident source term, averaged over six months, as given in section F.1, Table F.2, was input to the QAD point kernel code. A multitude of computer calculations were made for each radioactive system as a function of pipe diameter. Figures F-23 through F-28 show the dose rate vs. distance from pipe surfaces for various pipe diameters, for the RWCU, RRC and RHR systems. Figures F-29 through F.30 show results for the LPCI system. Results for the LPCS/HPCS systems are presented in Figure F.31 through F-36. Figure F-37 shows results for the MS system (26") piping.

F.5 Dose to Specific Safety-Related Equipment

The total gamma dose, i.e., the integrated 40 year normal operation dose plus the integrated six month dose following a postulated LOCA will be determined for all of the 1E/1M (safety-related) equipment

located within the Primary Containment. The equipment list and their elevation inside Primary Containment are presented in Table 6.1. The results will be shown in Table 6.1 for gamma ray dose contributions due to

- drywell atmosphere (LOCA)
- plate-out (LOCA)
- wetwell atmosphere (LOCA)
- suppression pool water (LOCA)
- systems (LOCA)
- systems normal operation

TABLE F.1

TIME MESH SPACING USED IN SOURCE CALCULATION

Time (min.)*	Time (min.)	Time (min.)
0	1.12 + 03	1.080 + 05
2.0 + 01**	1.28 + 03	1.296 + 05
4.0 + 01	1.44 + 03	1.512 + 05
6.0 + 01	2.16 + 03	1.728 + 05
8.0 + 01	2.88 + 03	2.160 + 05
1.0 + 02	3.60 + 03	2.592 + 05
1.2 + 02	4.32 + 03	
1.8 + 02	5.04 + 03	
2.4 + 02	5.76 + 03	
3.0 + 02	1.44 + 04	
3.6 + 02	2.88 + 04	
4.2 + 02	4.32 + 04	
4.8 + 02	5.76 + 04	
6.4 + 02	7.20 + 04	
8.0 + 02	8.64 + 04	
9.6 + 02		

* After LOCA

** read 2.0 + 01 as 2.0×10^1

TABLE F.2

BETA AVERAGE DECAY RATE (MeV/sec), 0 - 6 MONTHS AFTER LOCA

BETA DECAY RATE (MeV/sec)

Solids	1.29 + 19
Halogens	1.84 + 17
Noble Gases	1.54 + 17

TABLE F.3

APPROXIMATE DOSE RATE REDUCTION FACTOR

VS. DISTANCE FROM CORE MID-PLANE

Distance (ft.)	Reduction Factor
0	1.0
5	0.5
10	0.01
15	0.001

TABLE F.4: to be issued at a later date

TABLE F.5:
to be issued at a later date

TABLE F.6:

to be issued at a later date

TABLE F.7:

to be issued at a later date

TABLE F.8:

to be issued at a later date

TABLE F.9:

to be issued at a later date

TABLE F.10:

to be issued at a later date

TABLE F.11: to be issued at a later date

TABLE F.12:

to be issued at a later date

TABLE F.13:

to be issued at a later date

TABLE F.14:

to be issued at a later date

TABLE F.13: WETWELL ATMOSPHERE DOSE

Detector Location R (cm)*	Z (cm)**	Dose Rate (R/hr)
In Wetwell:		
10	632	2.44 + 04
650	632	2.10 + 04
650	900	2.81 + 04
650	1200	2.90 + 04
650	1555	2.12 + 04
In Drywell:		
815	1642	6.04
815	2000	6.20
730	2500	4.42
610	3000	3.34
610	3500	2.69
500	4000	2.20

* from core center

** from bottom of pool

TABLE F.14: SUPPRESSION POOL WATER DOSE

Detection Location R (cm)*	Z (cm)**	Dose Rate (r/hr)
In Wetwell:		
10	632	2.77 + 03
650	632	5.12 + 02
650	900	2.13 + 03
650	1200	1.51 + 03
650	1555	1.02 + 03
In Drywell:		
815	1642	2.74 - 01
815	2000	2.35 - 01
730	2500	1.68 - 01
610	3000	1.21 - 01
610	3500	9.93 - 02
500	4000	8.28 - 02

* from core center
** from bottom of pool

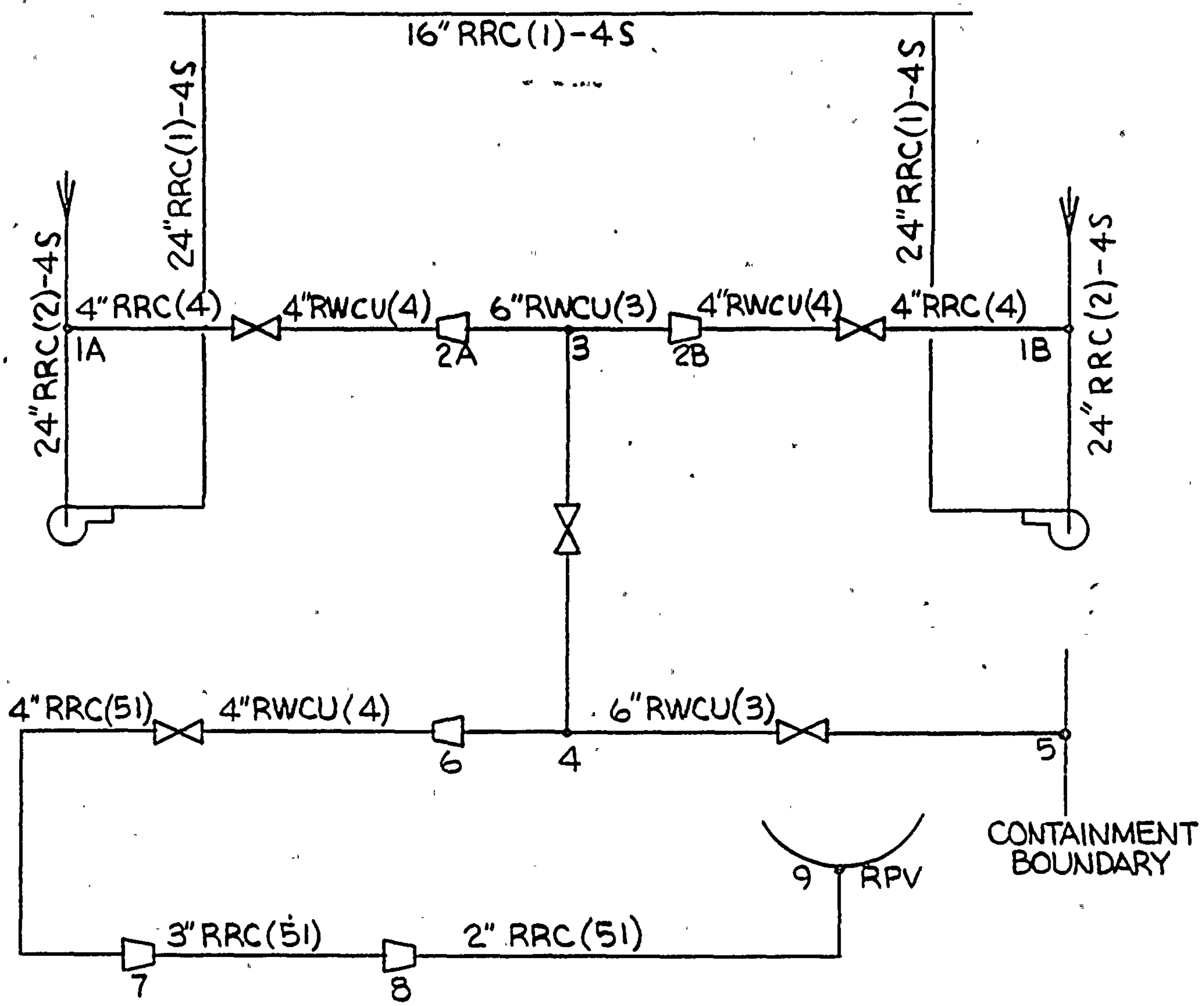

Figure F.1 Node Point and Line Identification
RWCU and RRC Systems

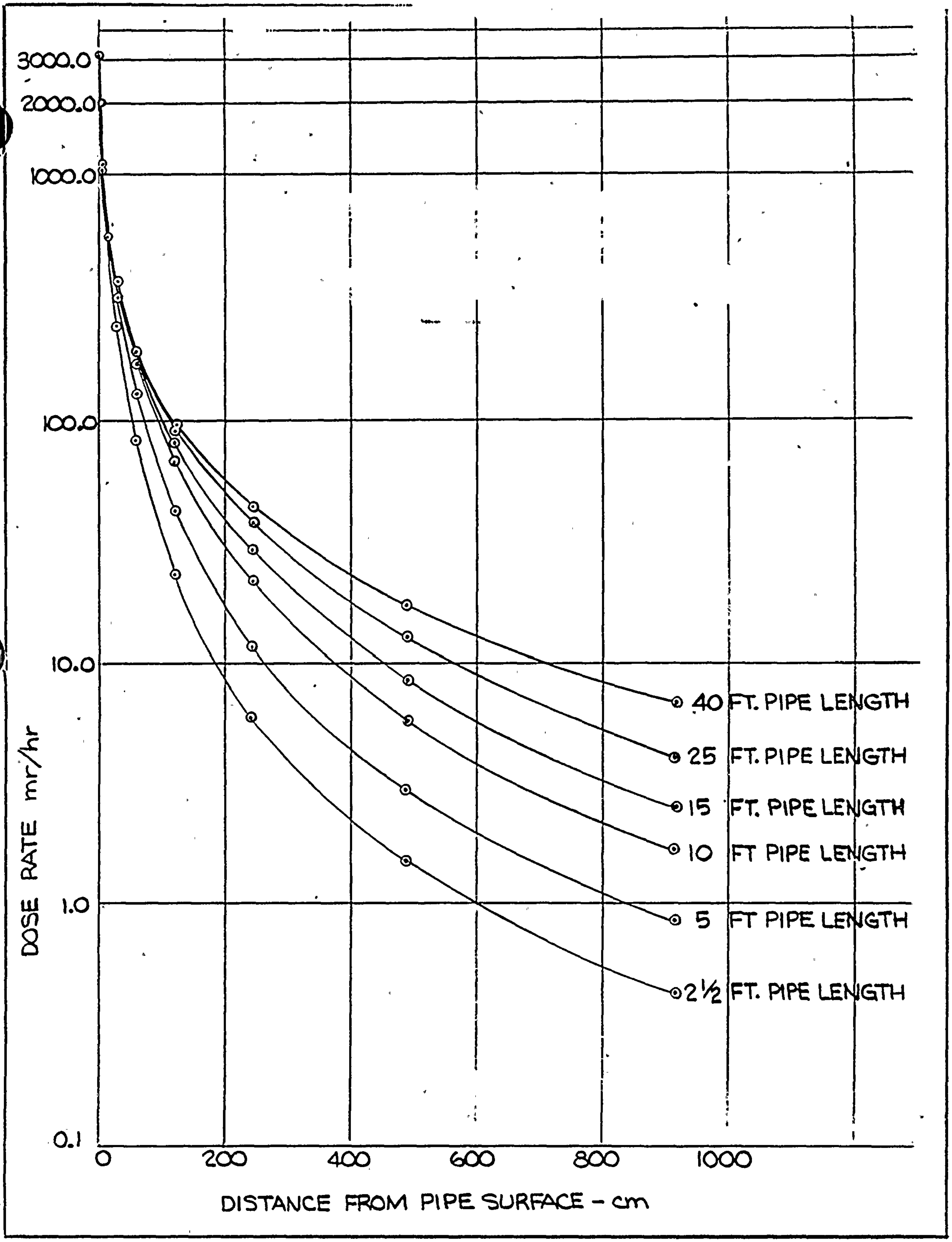

Figure F.2 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
2 in. RRC (51)-4 Node 9→8 4.56 x 10^6 MeV/cc-sec

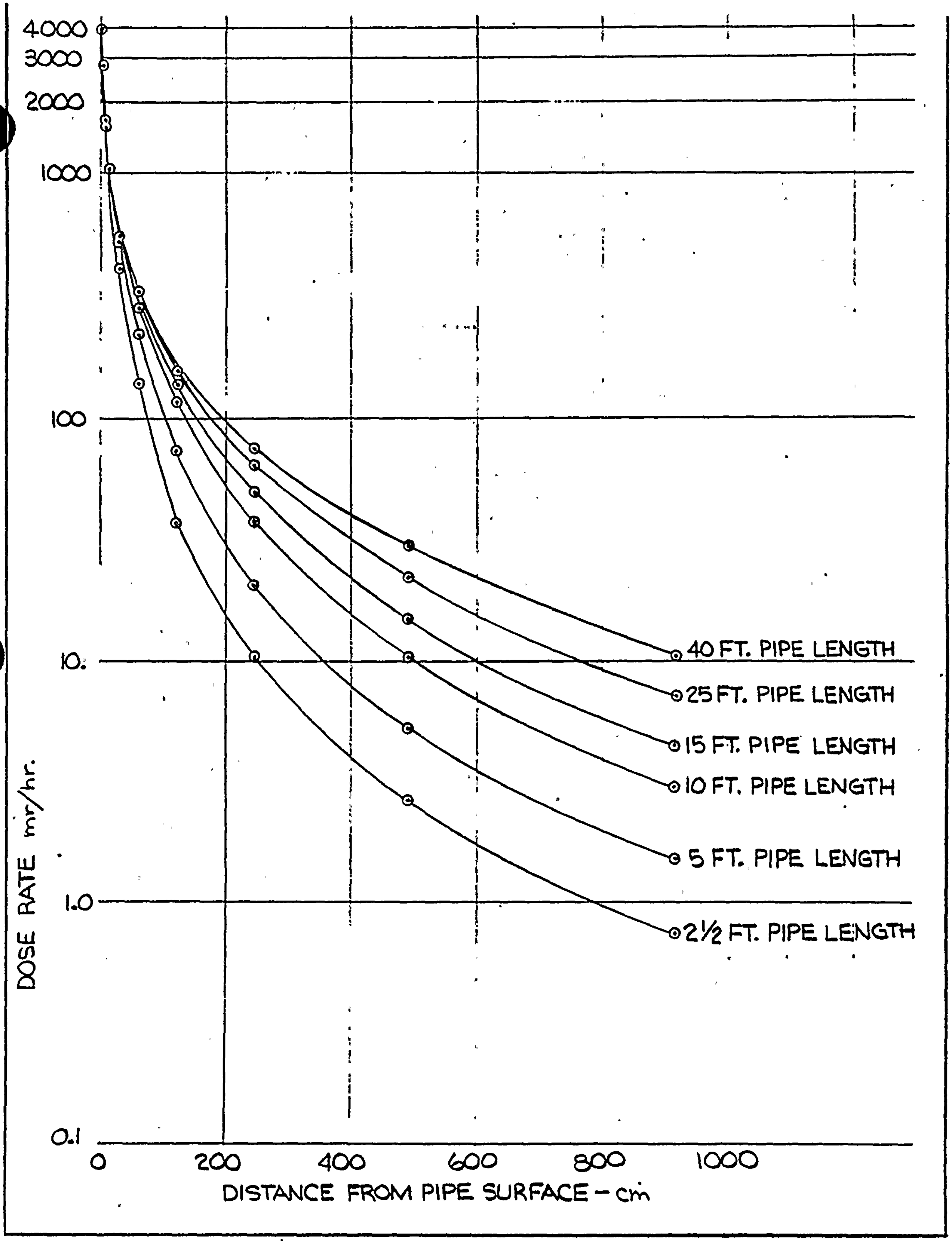

Figure F.3 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface

3 in. RRC(51)-4 Node 8 – 7 3.51 x 10^6 MeV/cc-sec

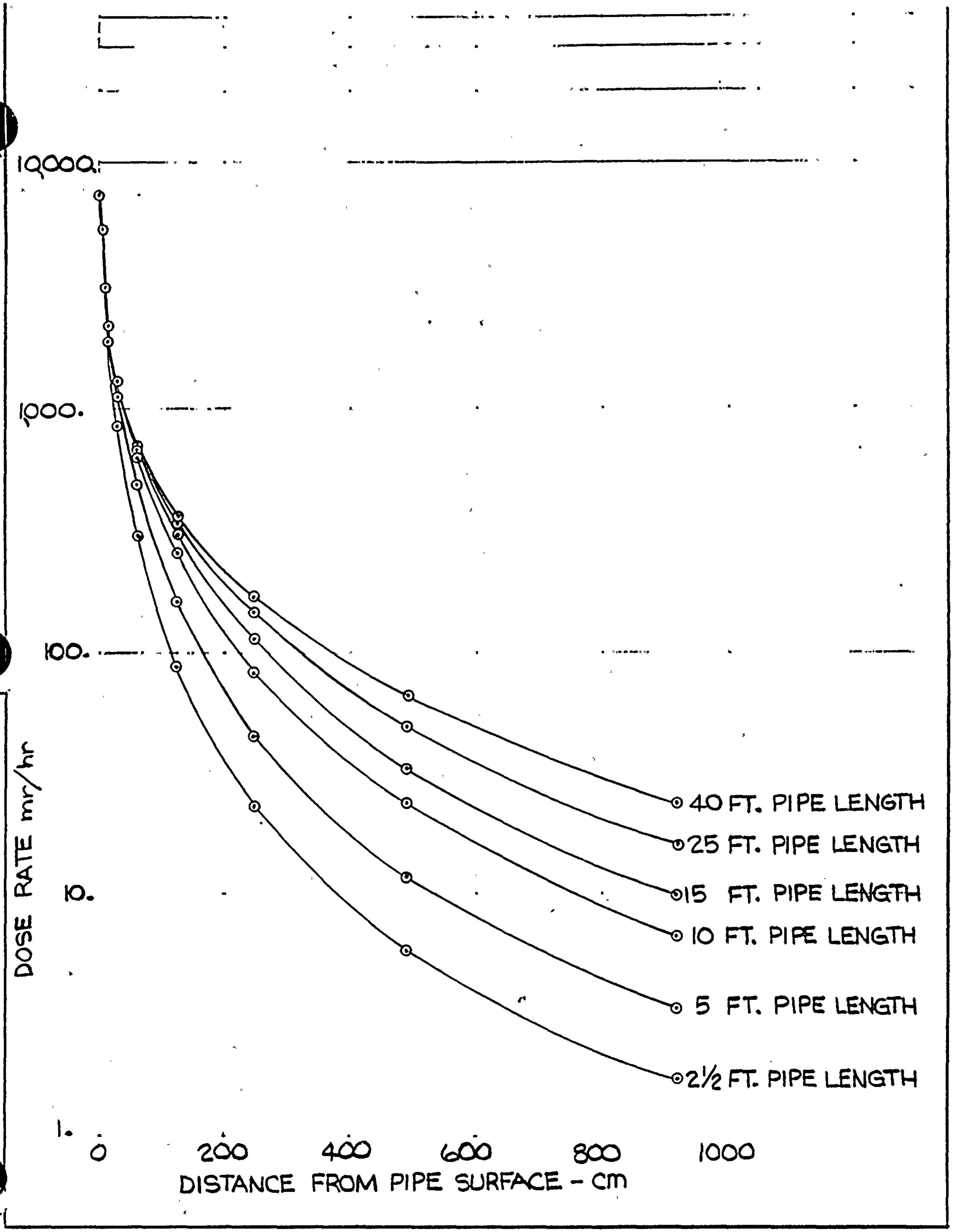

Figure F.4 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
4in. RRC(4)-4S Node 1B--2B 3.64×10^6 MeV/cc-sec

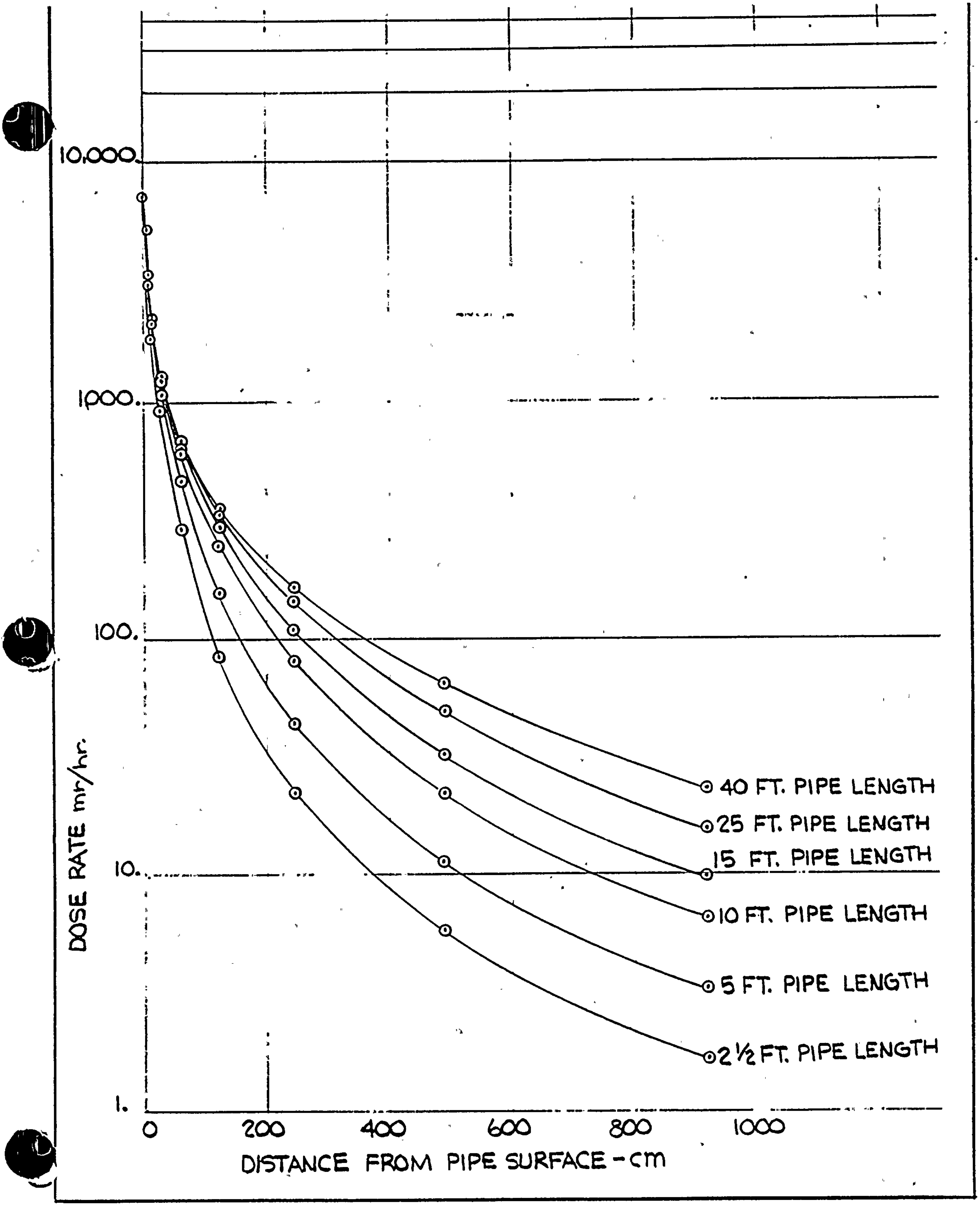

Figure F.5 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
4 in. RRC(4)-4S Node 1A ➝ 2A 3.50 x 10^6 MeV/cc-sec

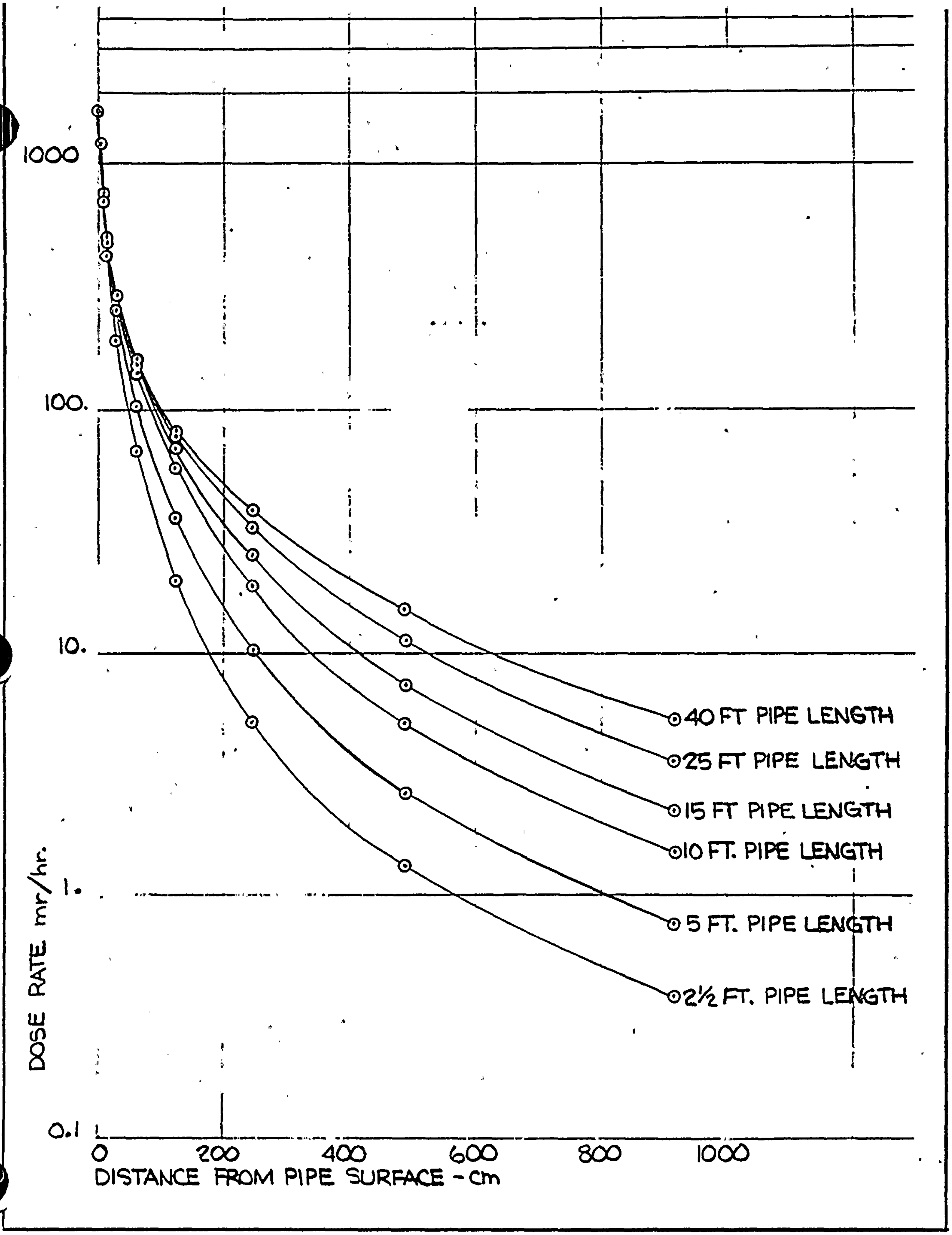

Figure F.6 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
4 in. RRC(51)-4S and 4 in. RWCU(4)-4 Node 7 > 5 8.09 x 10^5 MeV/cc-sec

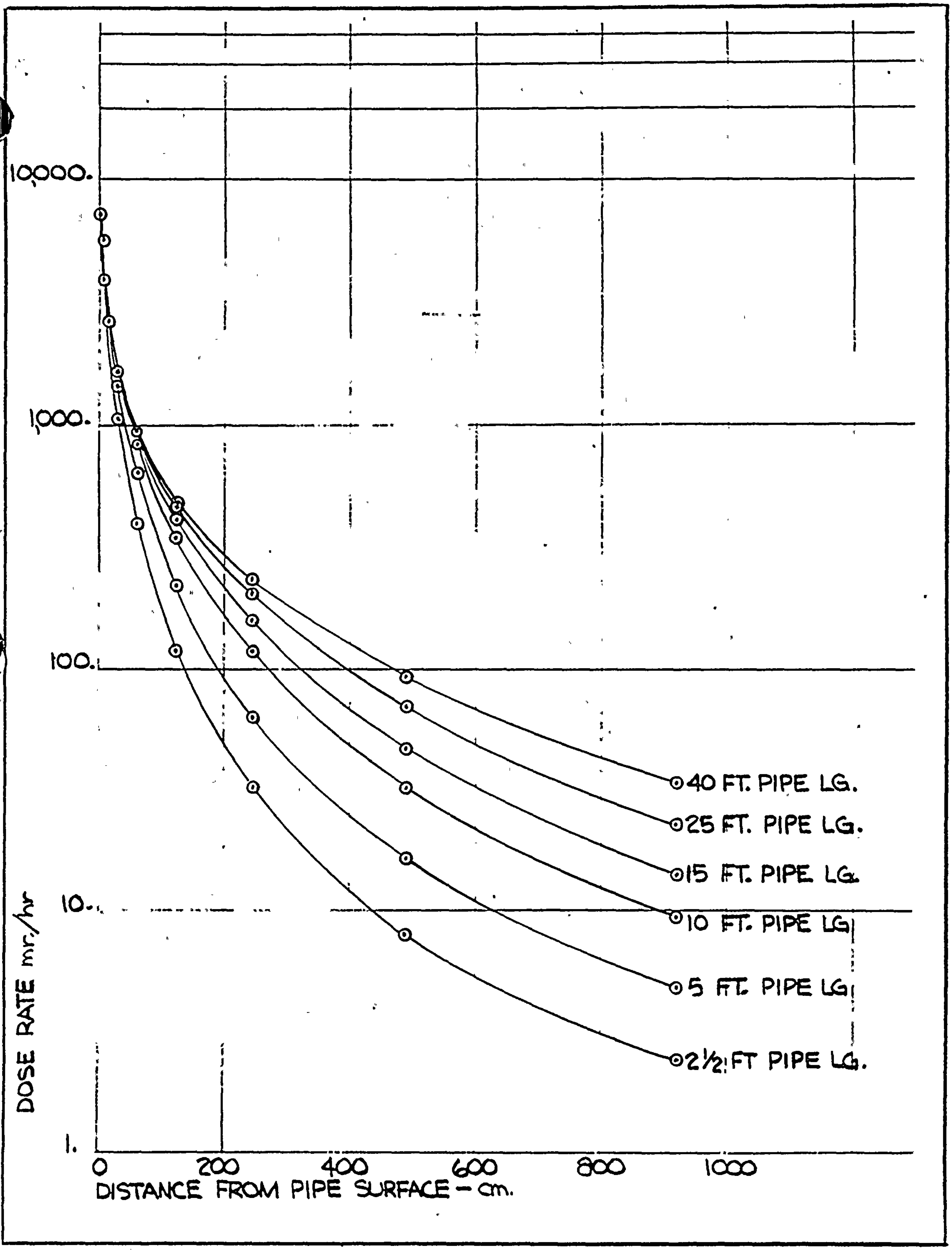

Figure F.7 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface

6 in. (RWCU(3)-4 Node 2B→3 2.42×10^6 MeV/cc-sec

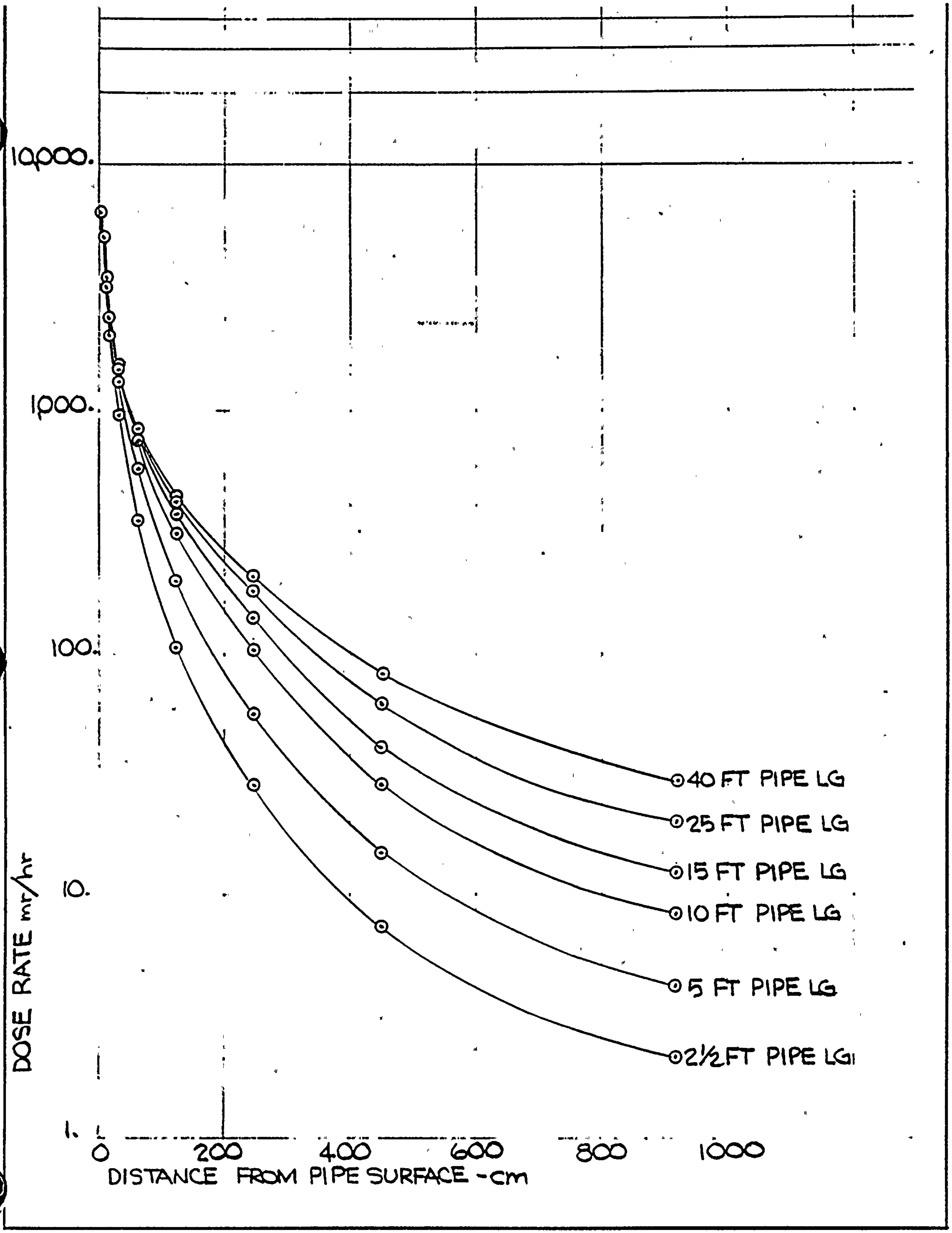

Figure F.8 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
6 in. RWCU(3)-4 Node 2A -3 2.22 x 10^6 MeV/cc-sec

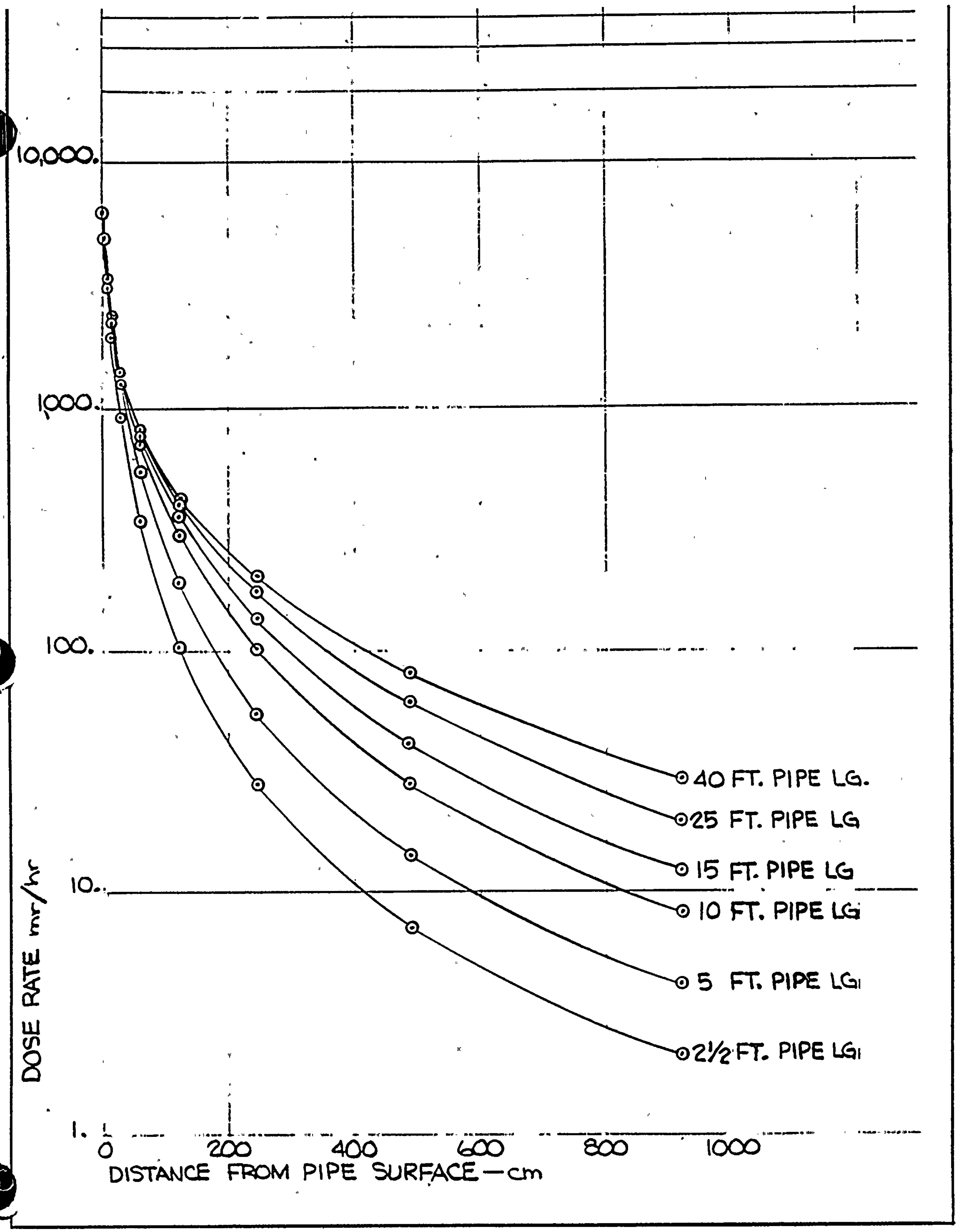

Figure F.9 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
6 in. RWCU(3)-4 Node 3 -4 2.12 x 10^6 MeV/cc-sec

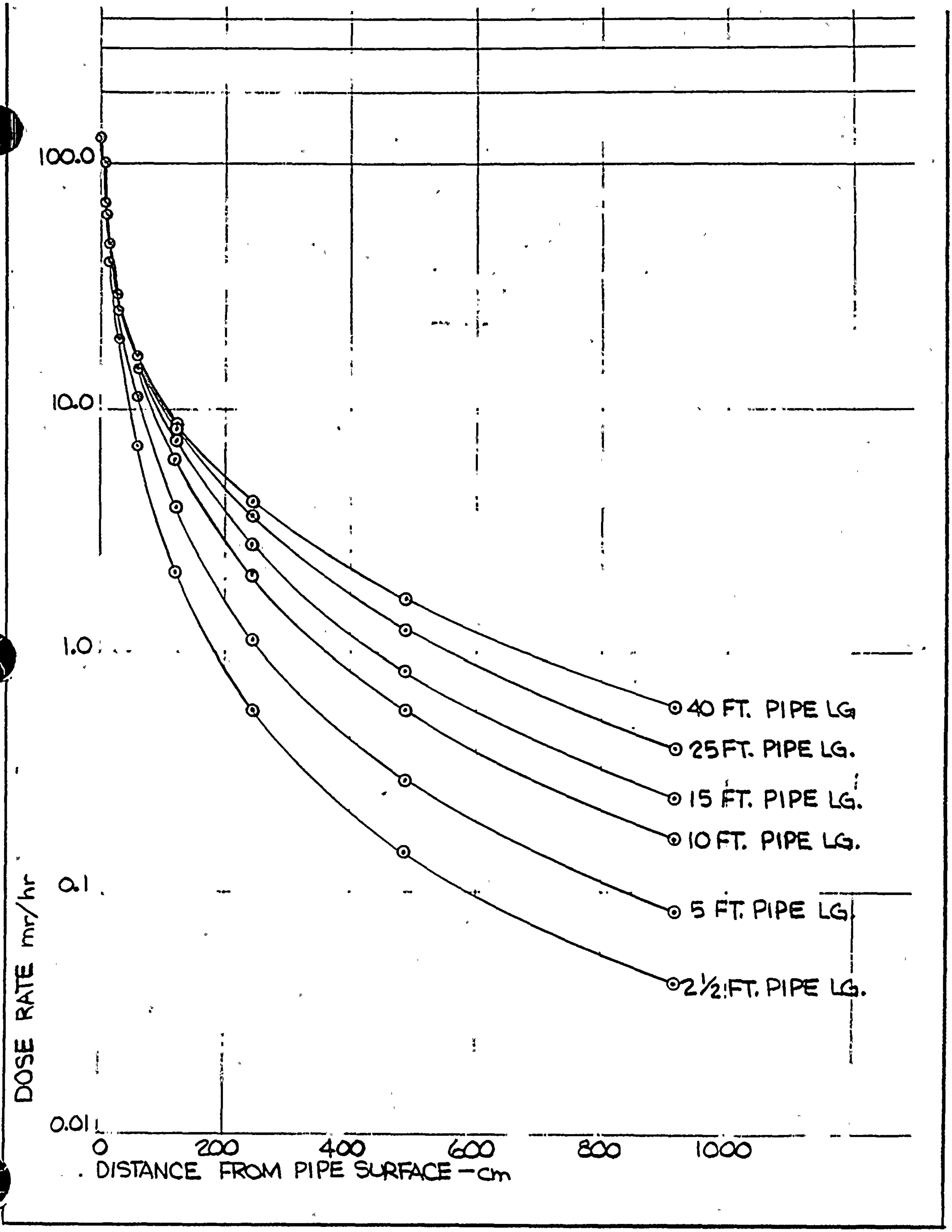

Figure F.10 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
6 in. RWCU(3)-4 Node 6 → 4 4.31 x 10^4 MeV/cc-sec

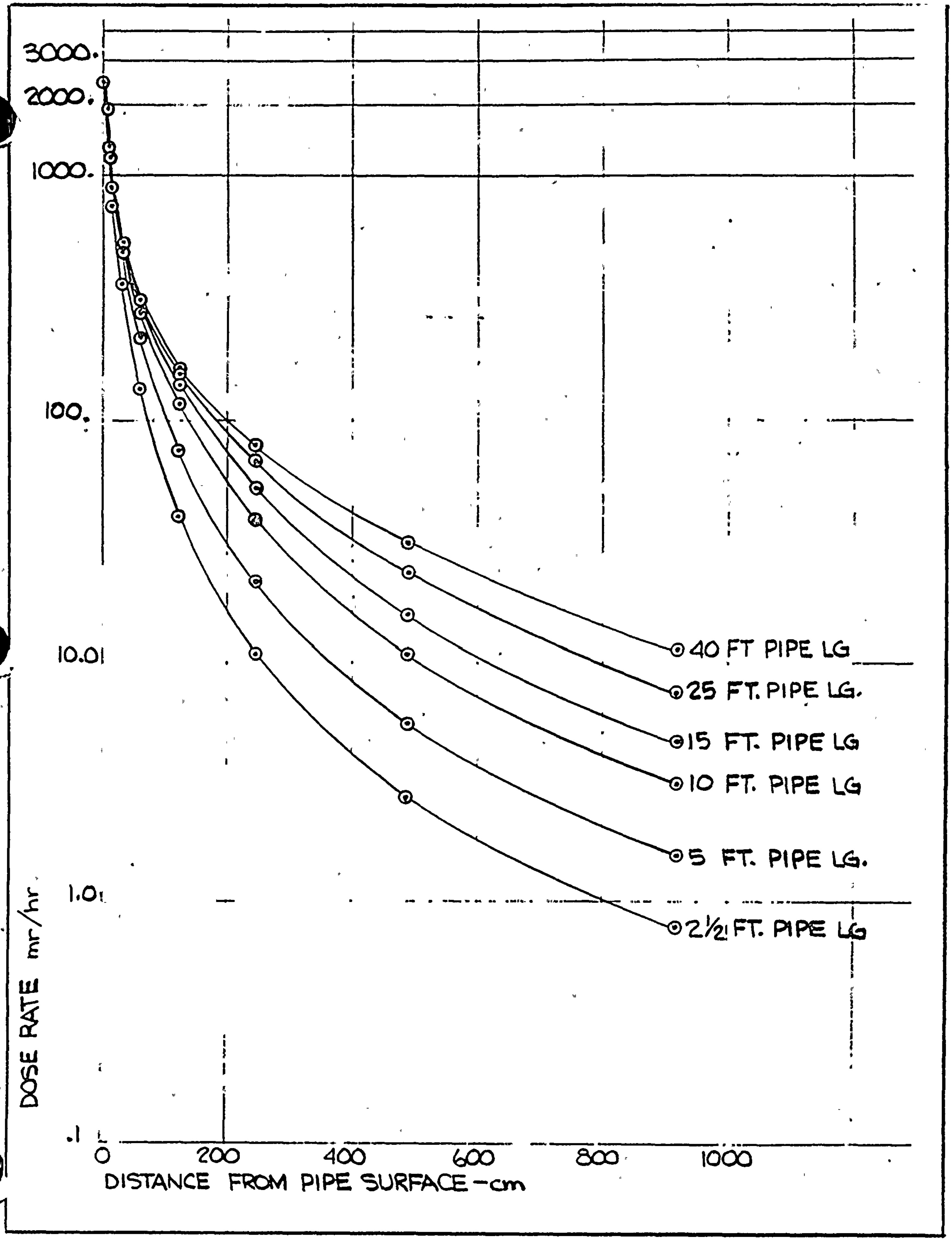

Figure F.11 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
6 in. RWCU(3)-4 Node 4→5 8.22 x 10^5 MeV/cc-sec

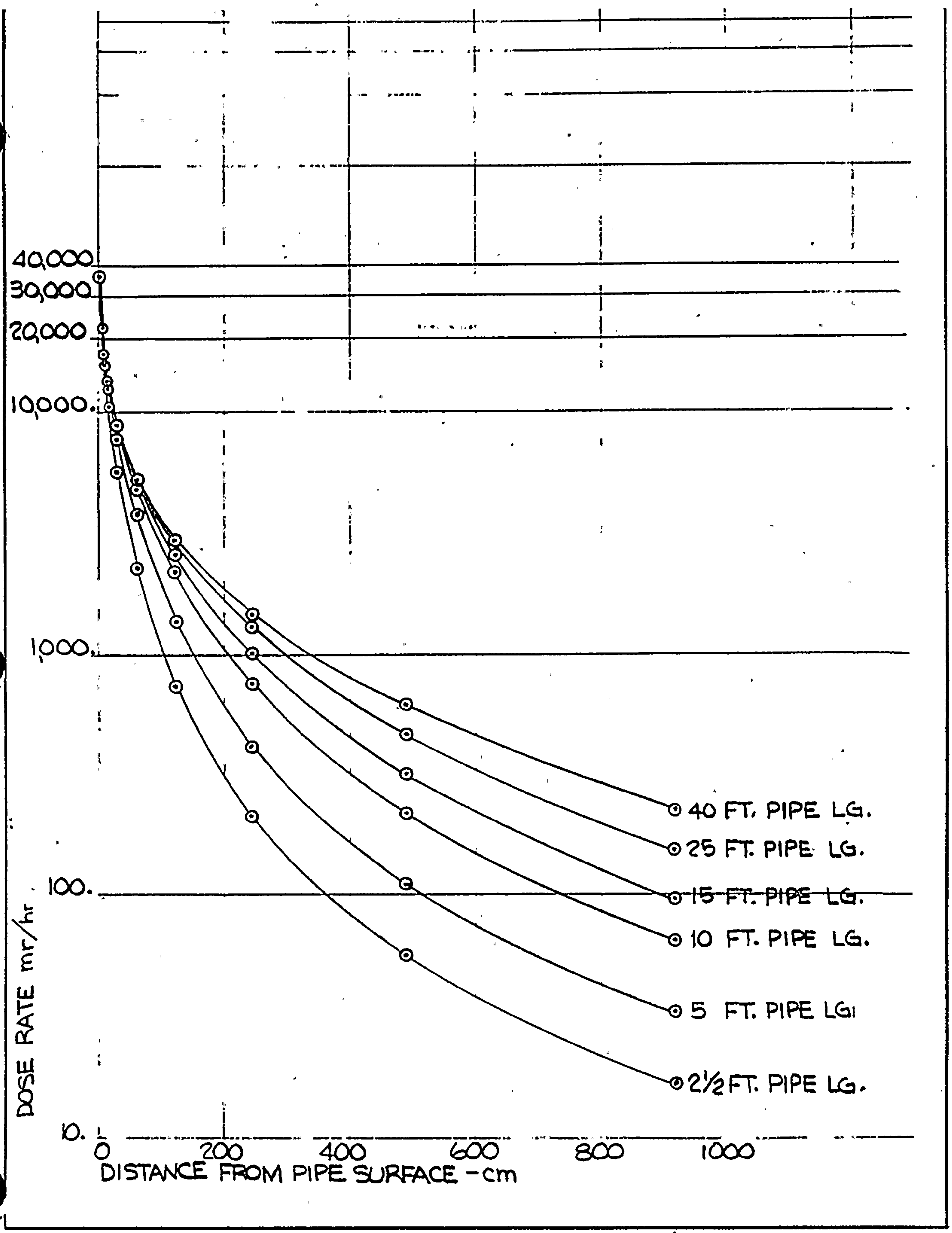

Figure F.12 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface

12 in. RRG(1)-4S 5.03×10^6 MeV/cc-sec

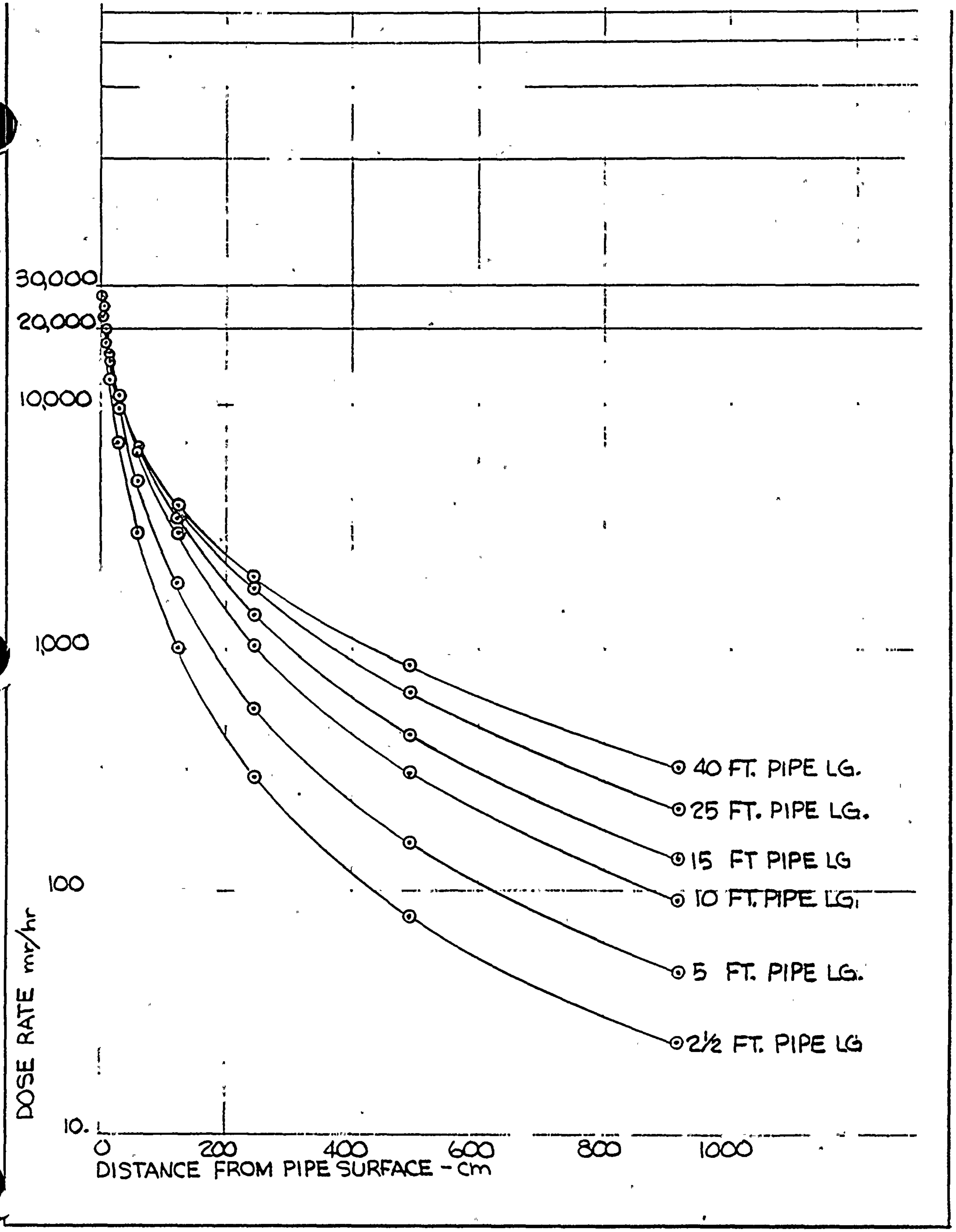

Figure F.13 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
16 in. RRC(1)-4S 5.03 x 10^6 MeV/cc-sec

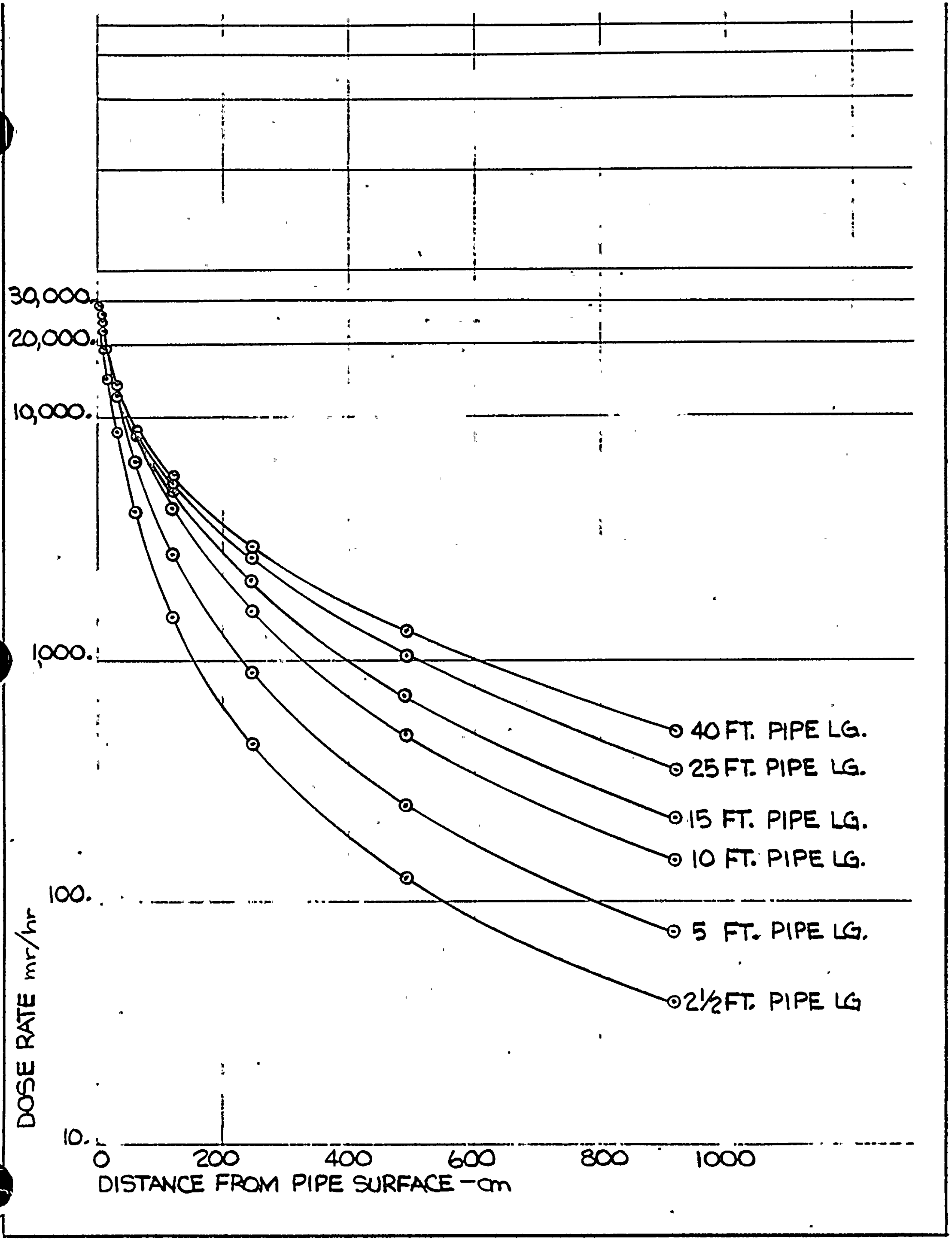

Figure F.14 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface

24 in. RRC(1)-4S and 24 in. RRC(2)-4S 5.03 x 10^6 MeV/cc-sec

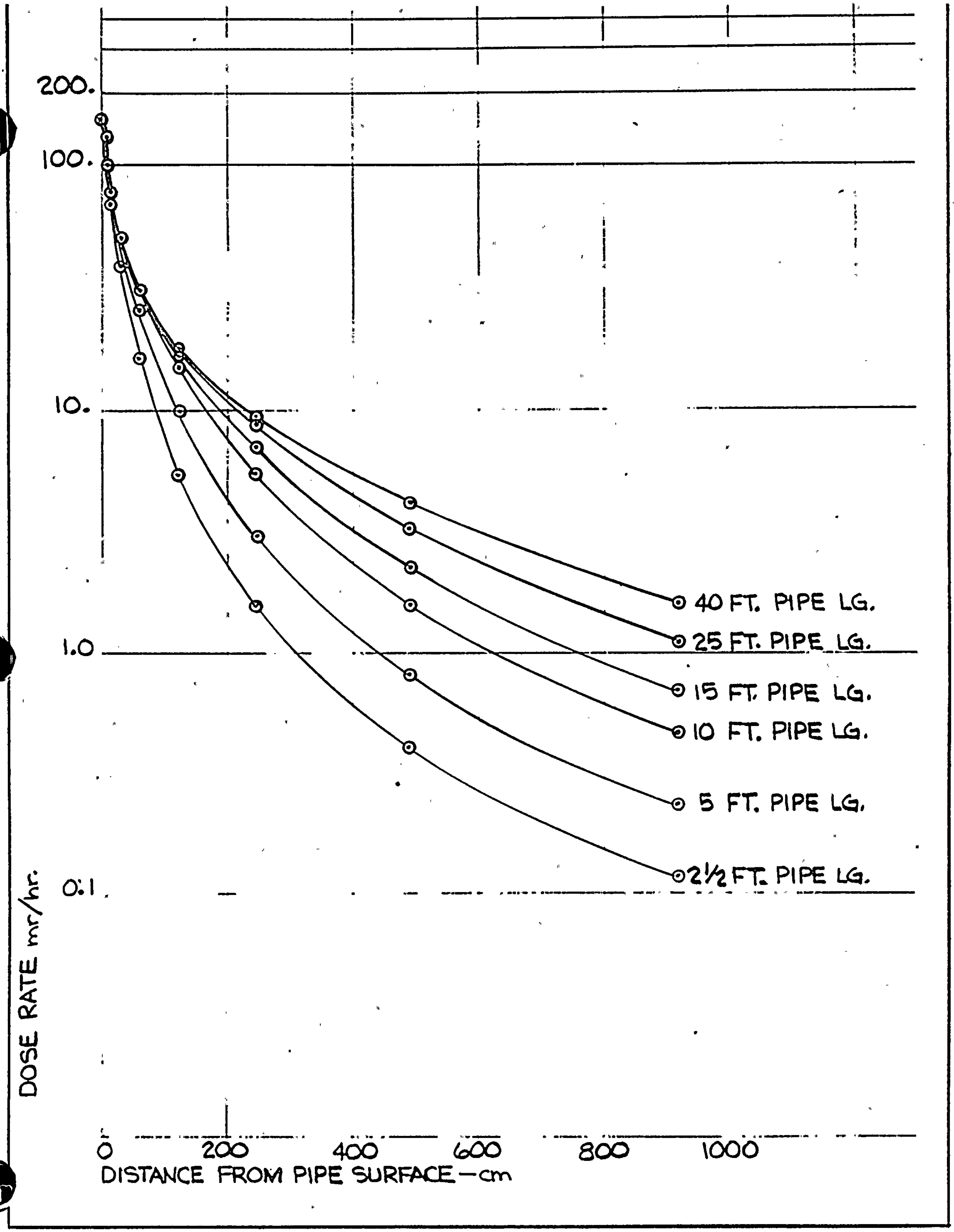

Figure F.15 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
12 in. RHR(1)-4 or 12 in. RHR(1)-45

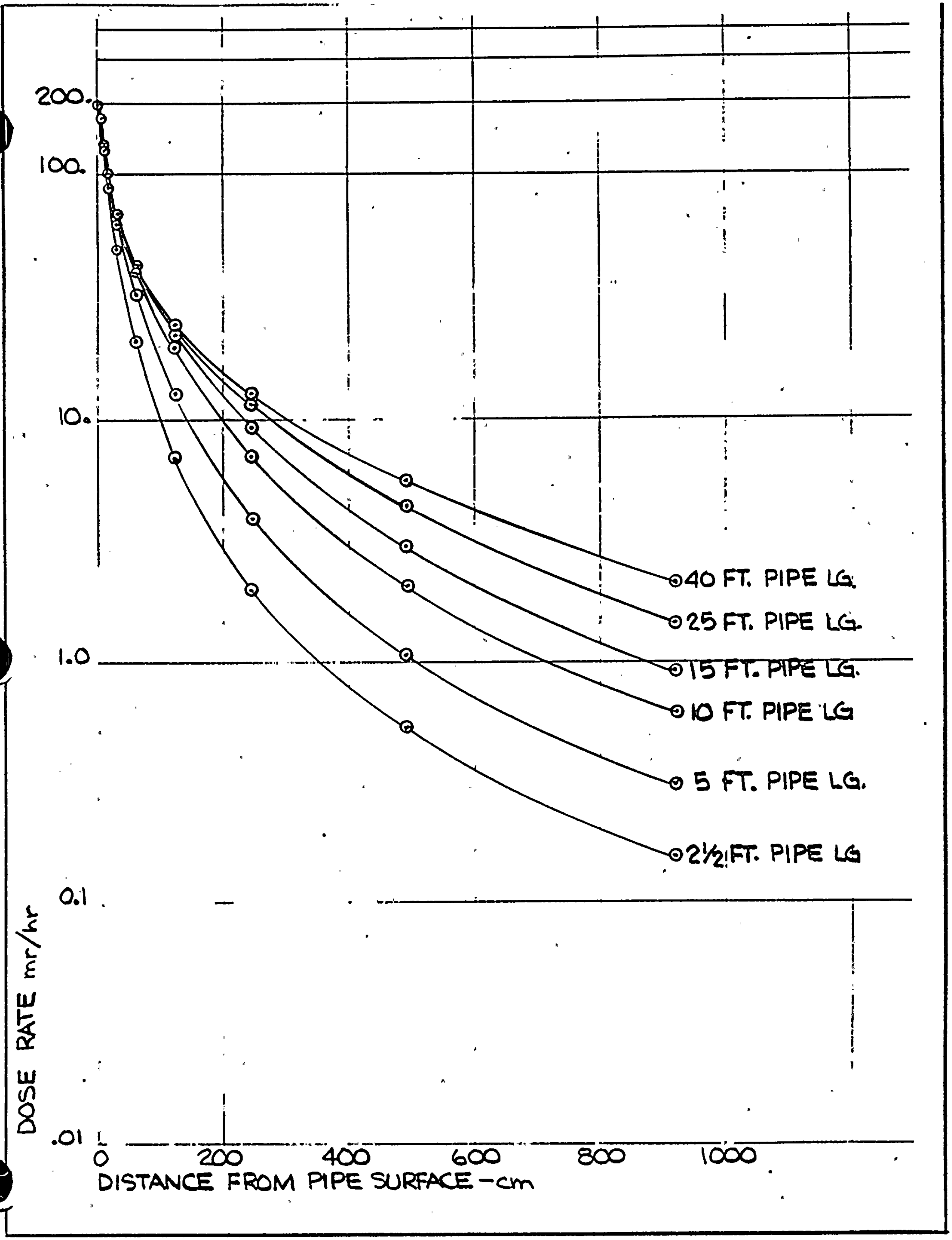

Figure F.16 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
14 in. RHR(1)-4 or 14 in. RHR(1)-4S

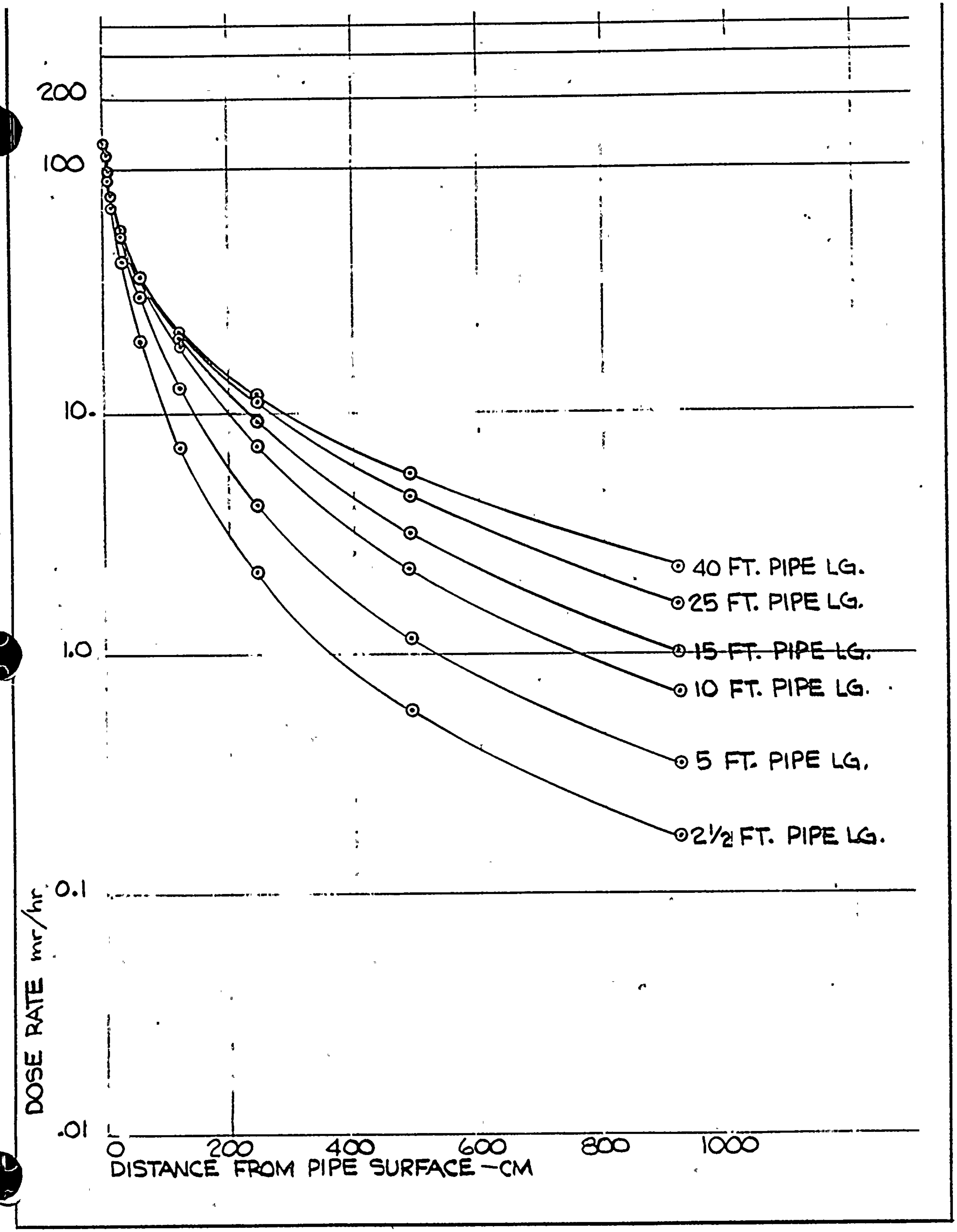

Figure F.17 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
20 in. RHR(2)-4 or 20 in. RHR(2)-4S

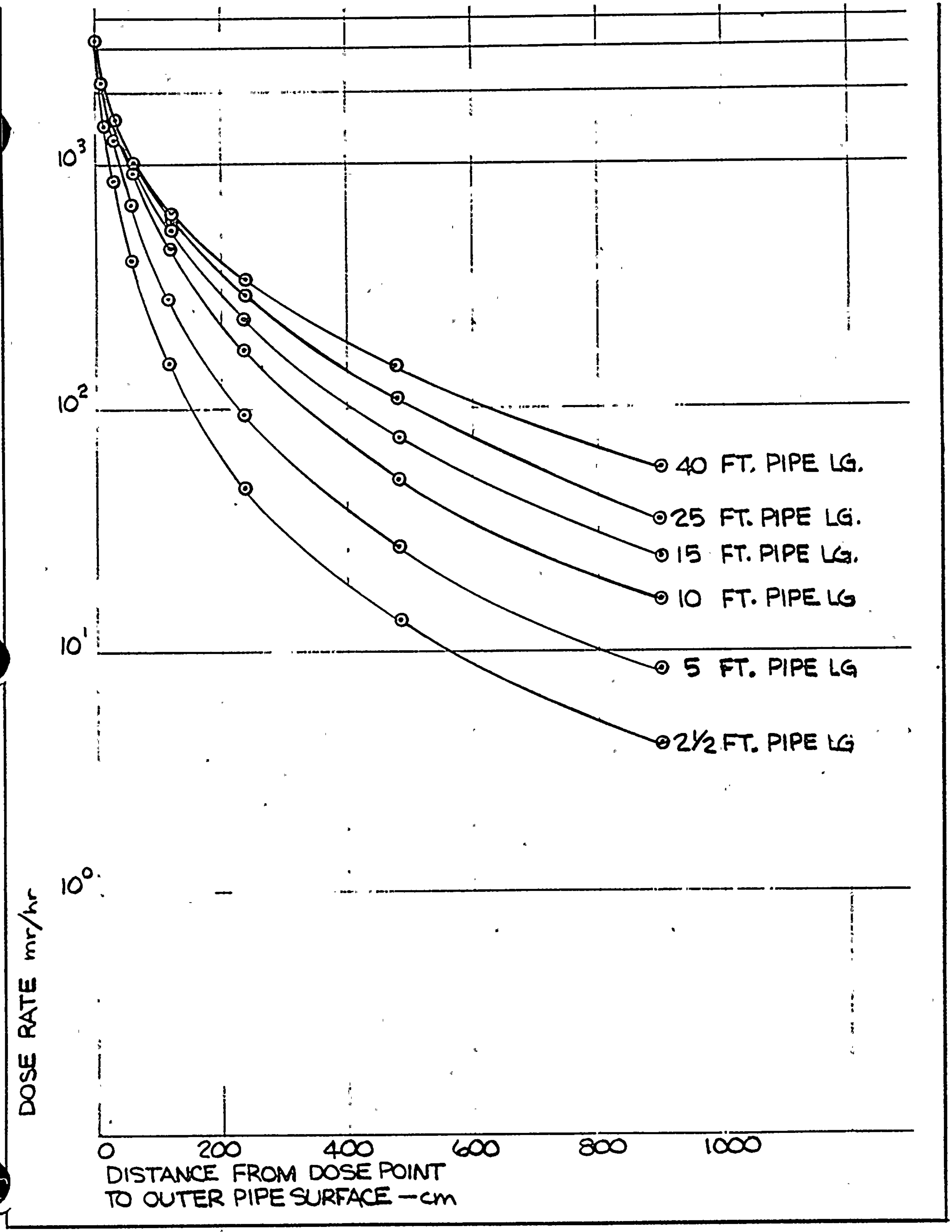

Figure F.18 Dose Rate vs. Distance from Surface of 26 in. Main Steam Pipe

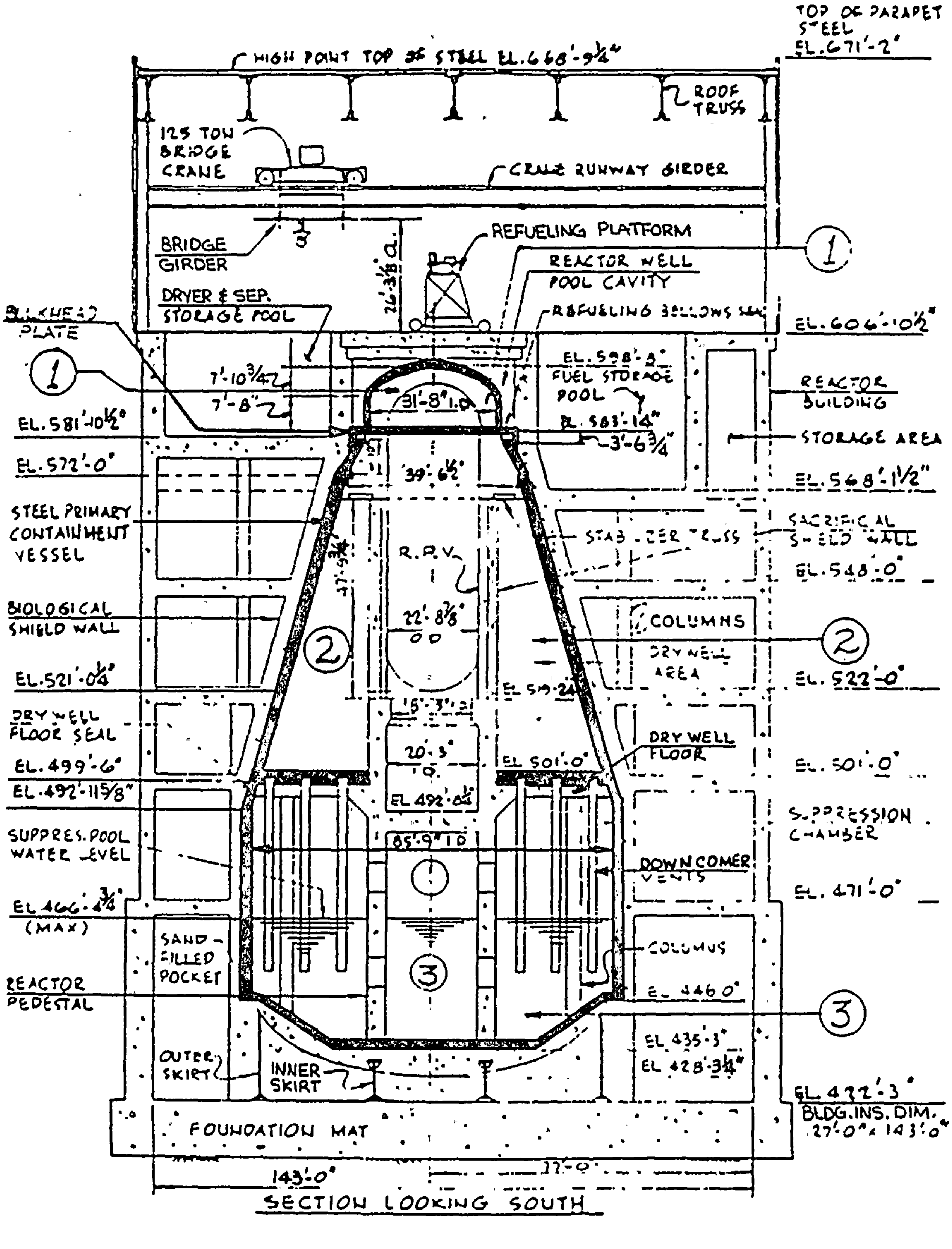

Figure F.19 Containment Cross Section

FIGURE F.20

to be issued at a later date

FIGURE F.21

to be issued at a later date

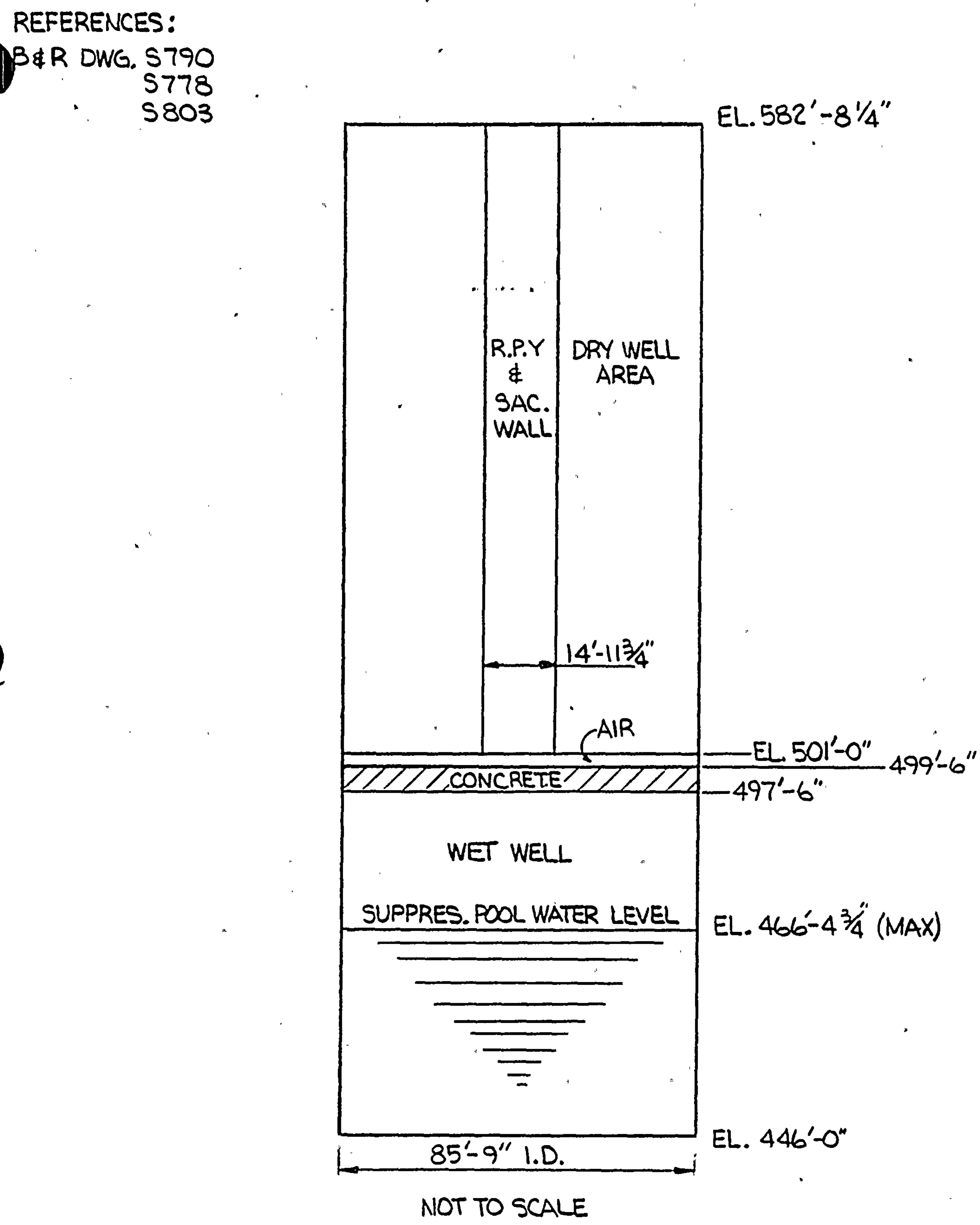

Figure F.22 Wetwell Zone Model

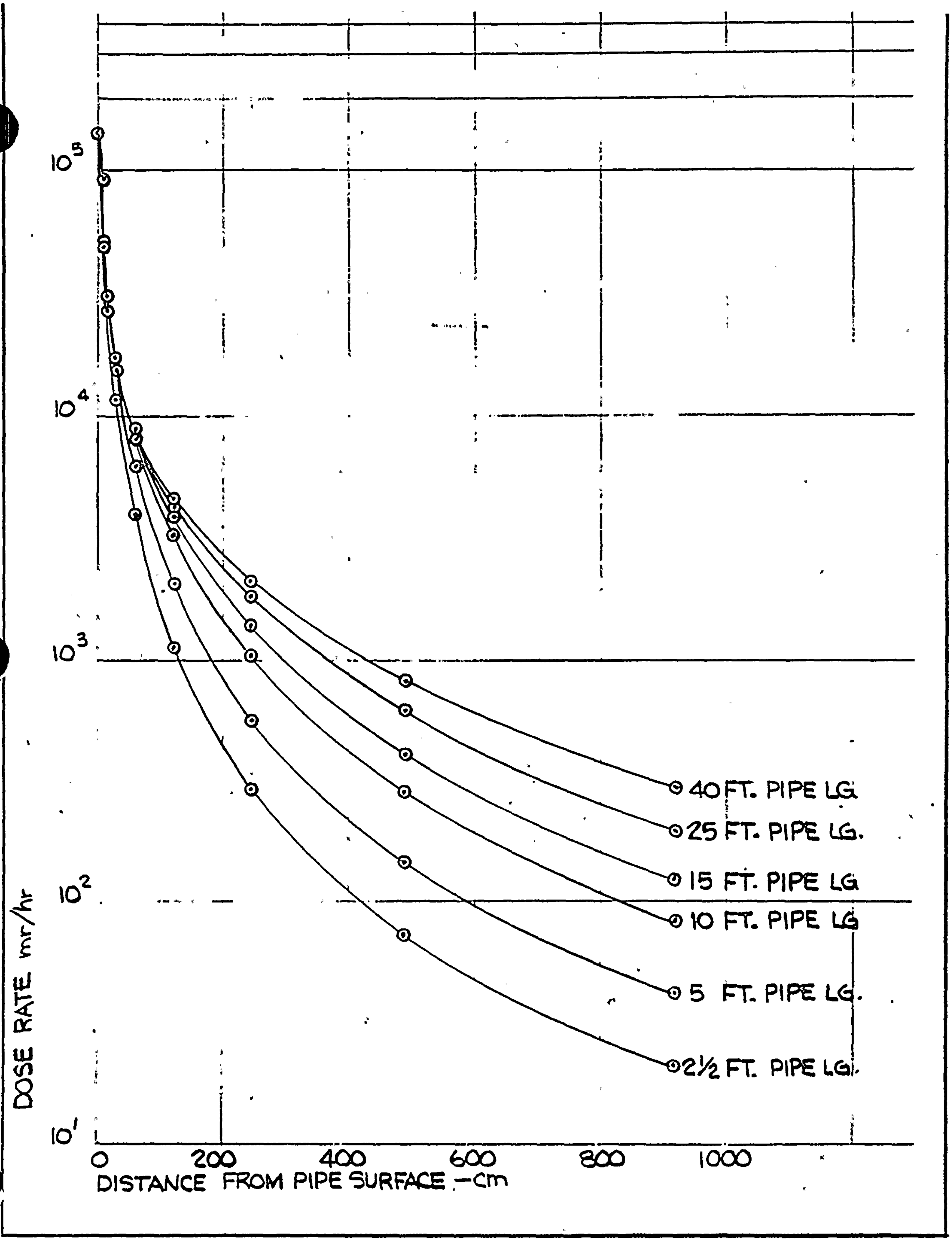

Figure F.23 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
RWCU, RRC and RHR Systems

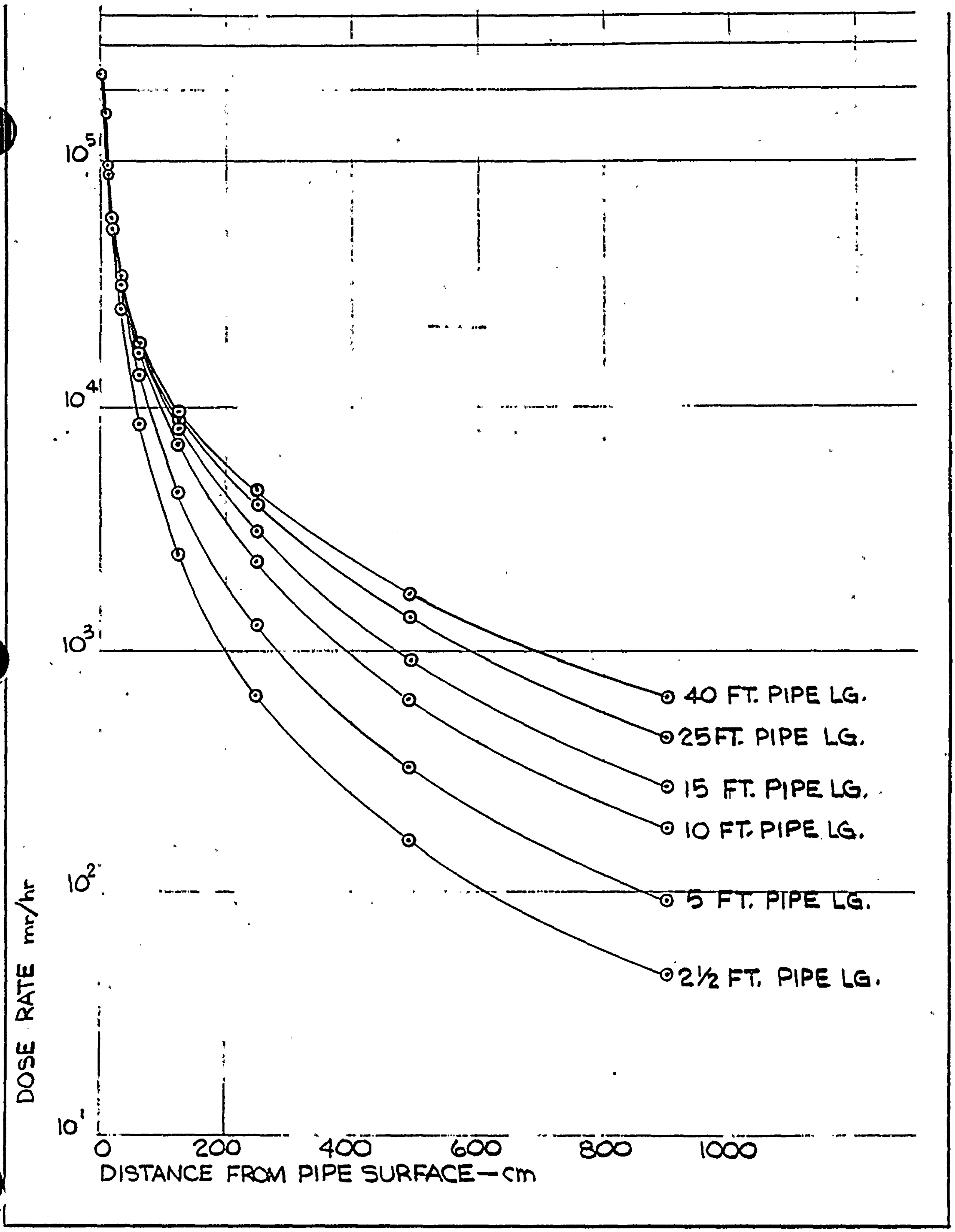

Figure F.24 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
RWCU, RRC and RHR Systems
3 in. Pipes-Sched. 160

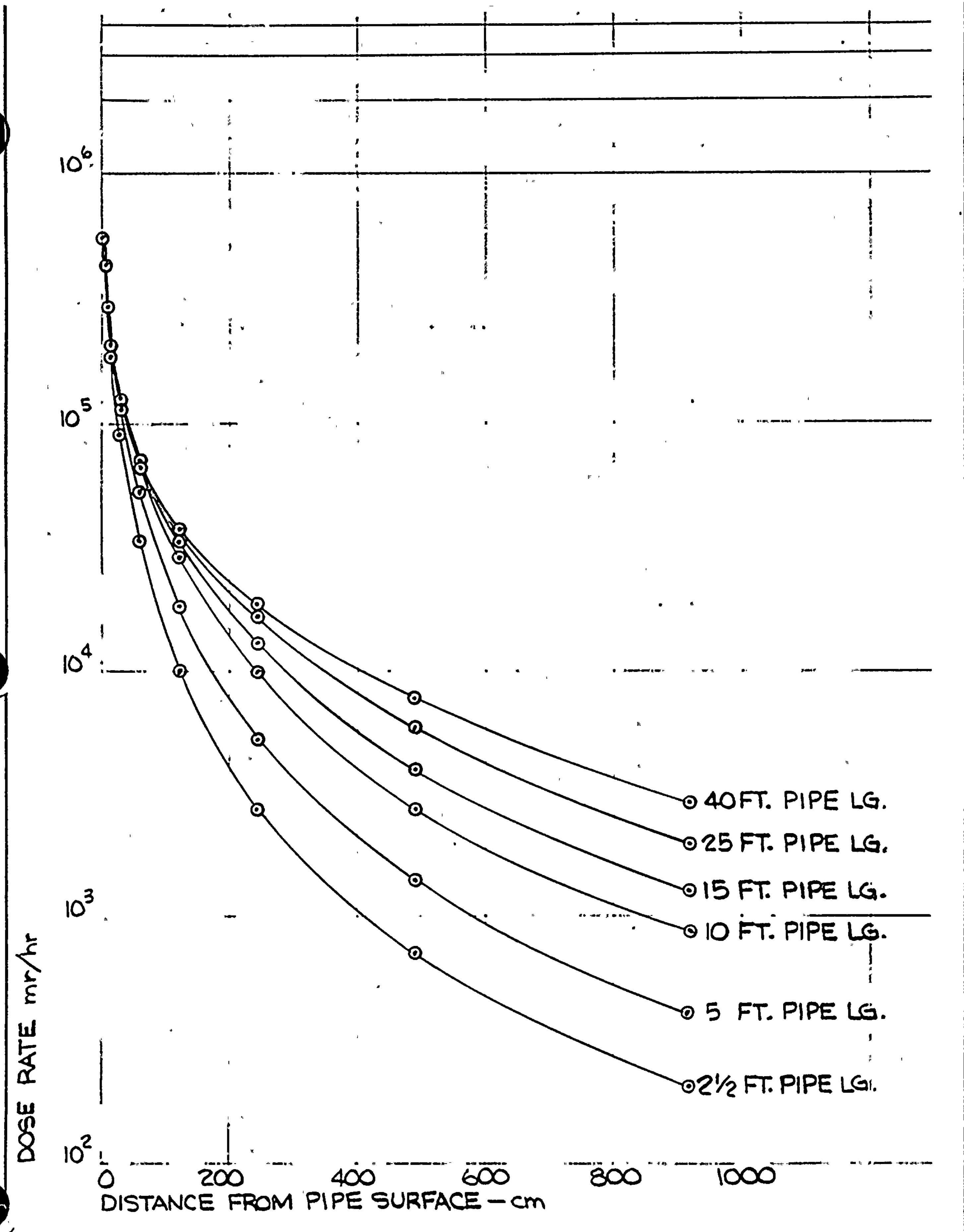

Figure F.25 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
RWCU, RRC and RHR Systems
6 in. Pipes-Sched. 80

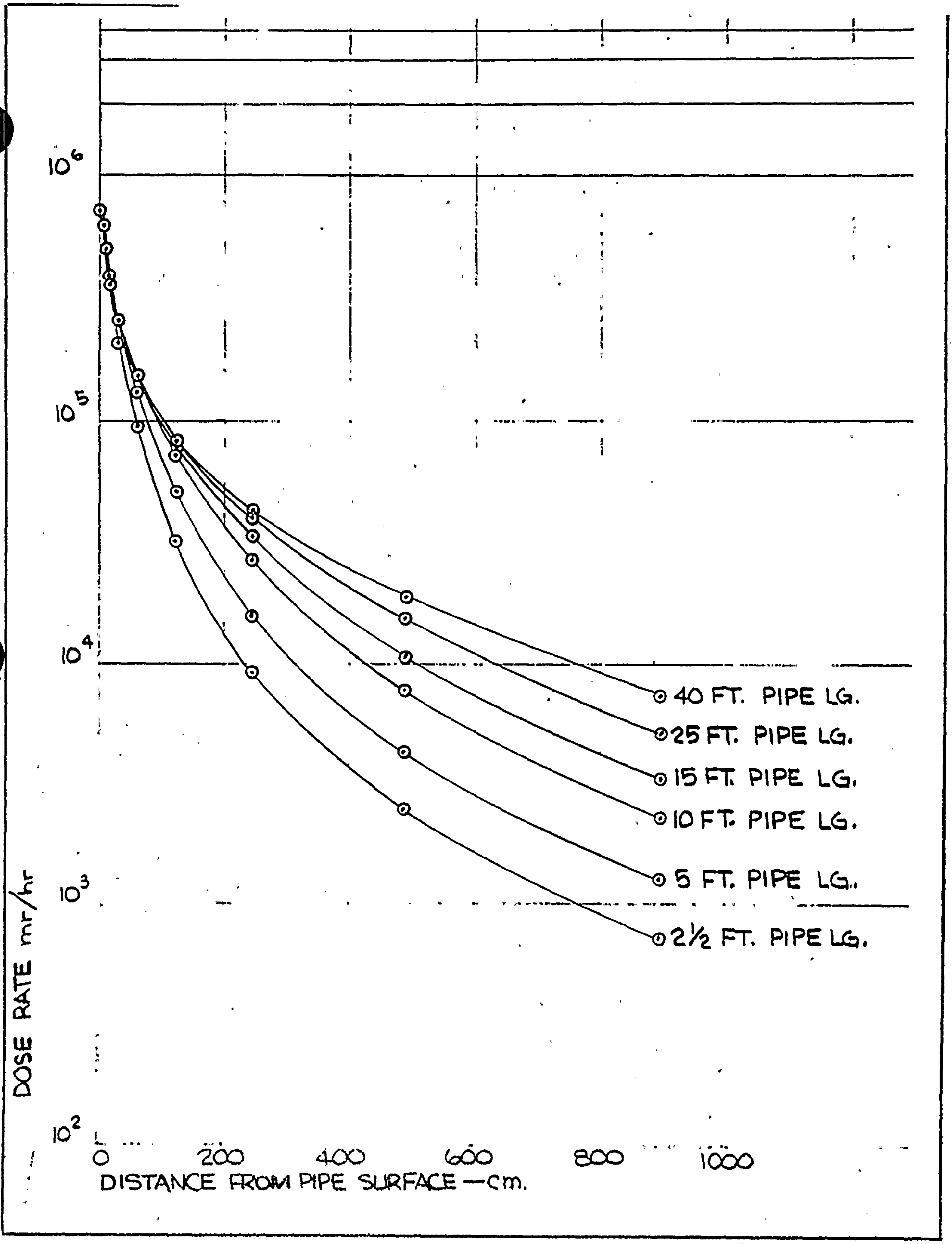

Figure F.[illegible] Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
RWCU, RRC and RHR Systems
[illegible] in. Pipes-Sched. 80

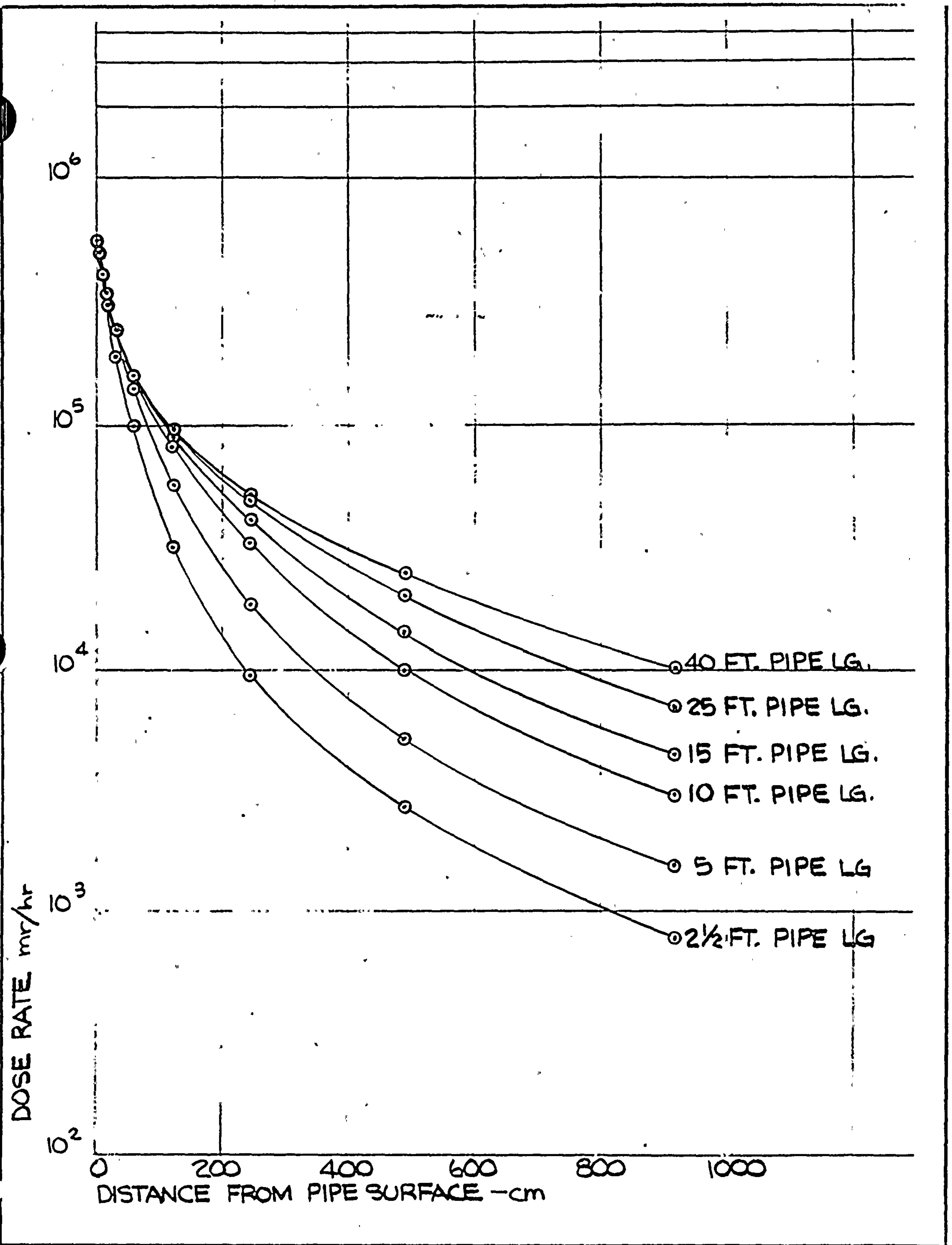

Figure F.27 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
RWCU, RRC and RHR Systems

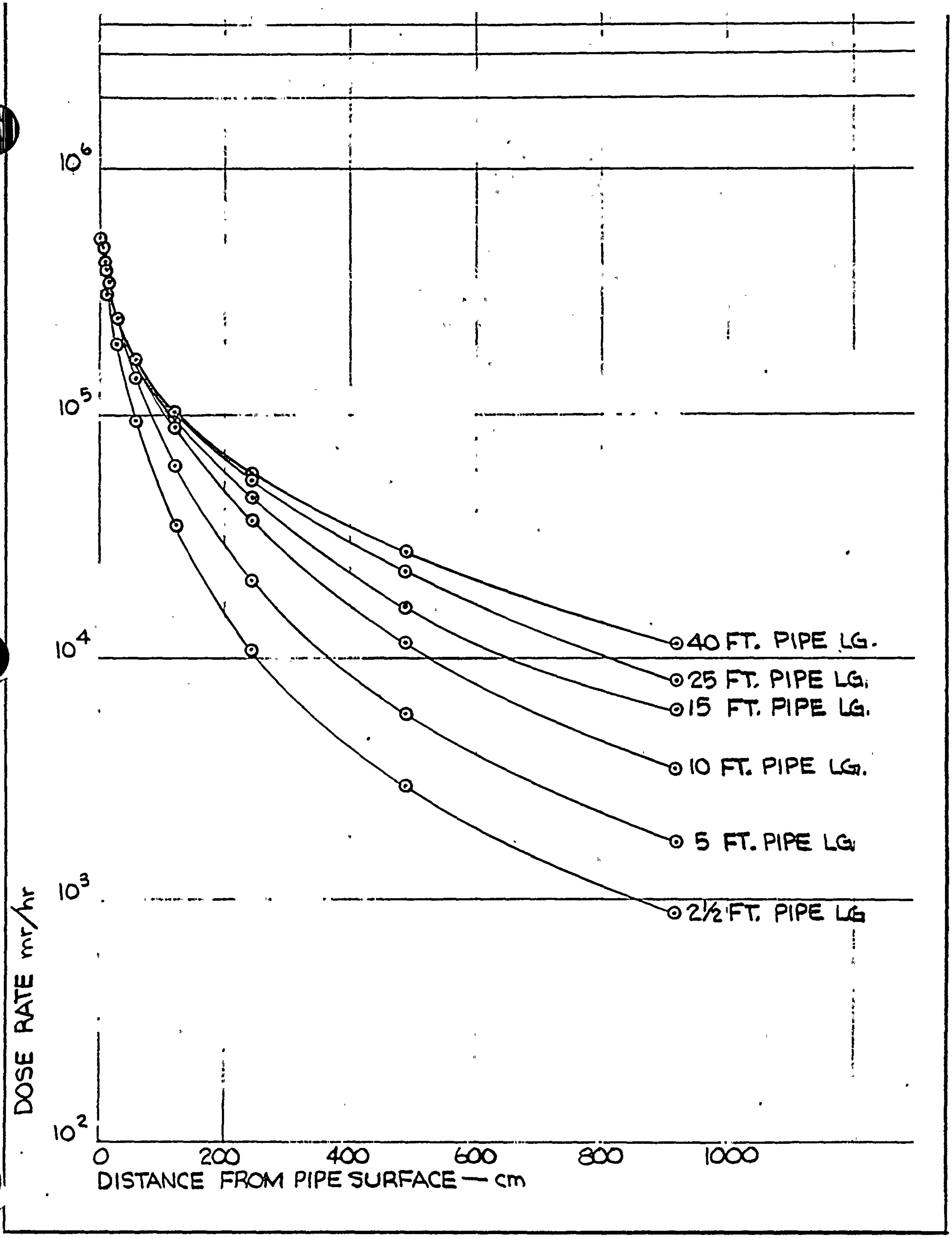

Figure F.28 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
RWCU, RRC and RHR Systems
24 in. Pipes-Sched. 80

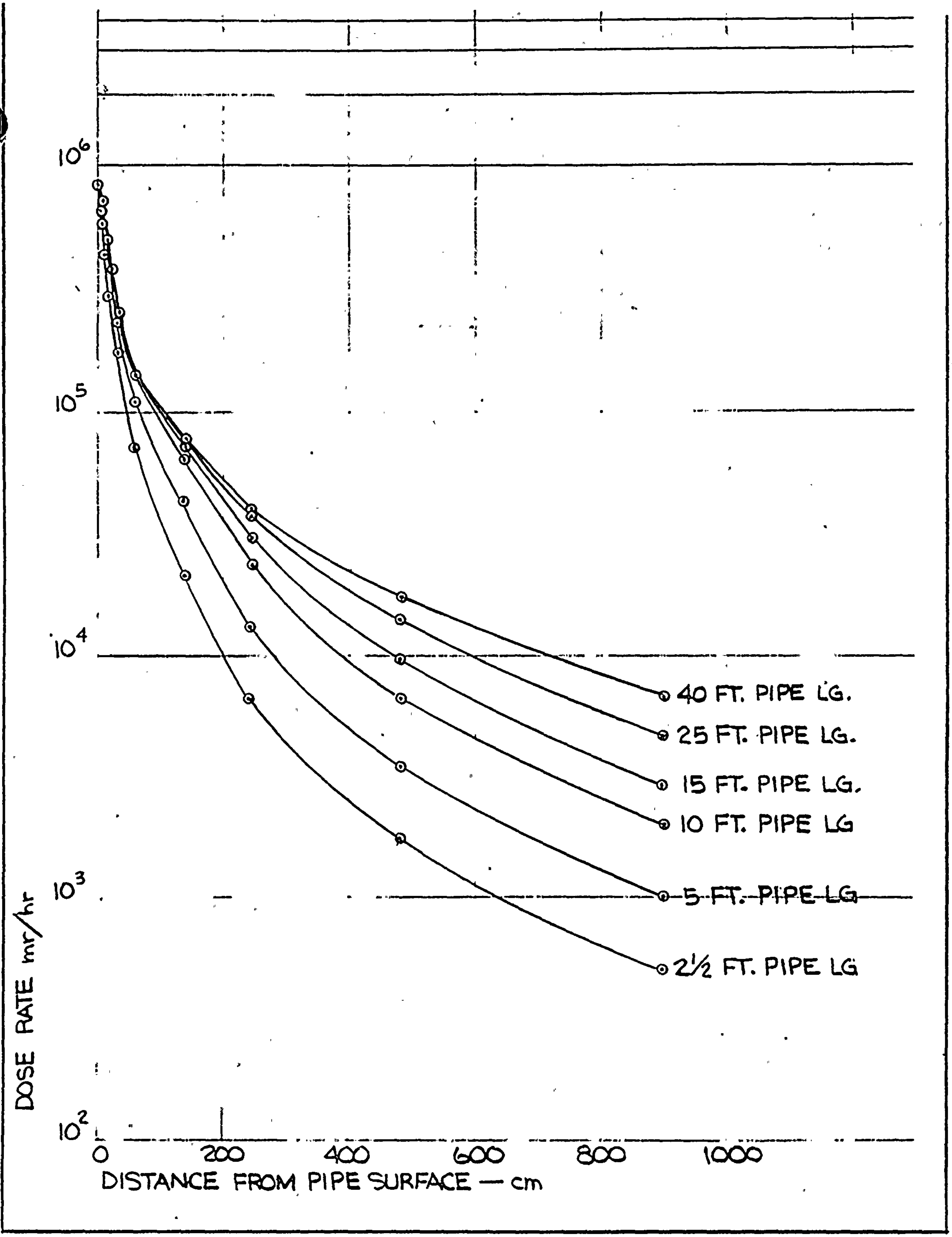

Figure F.29 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
LPCI System Schedule 80, D=12 in.
O.D=12.75 in. I.D=11.376 in.

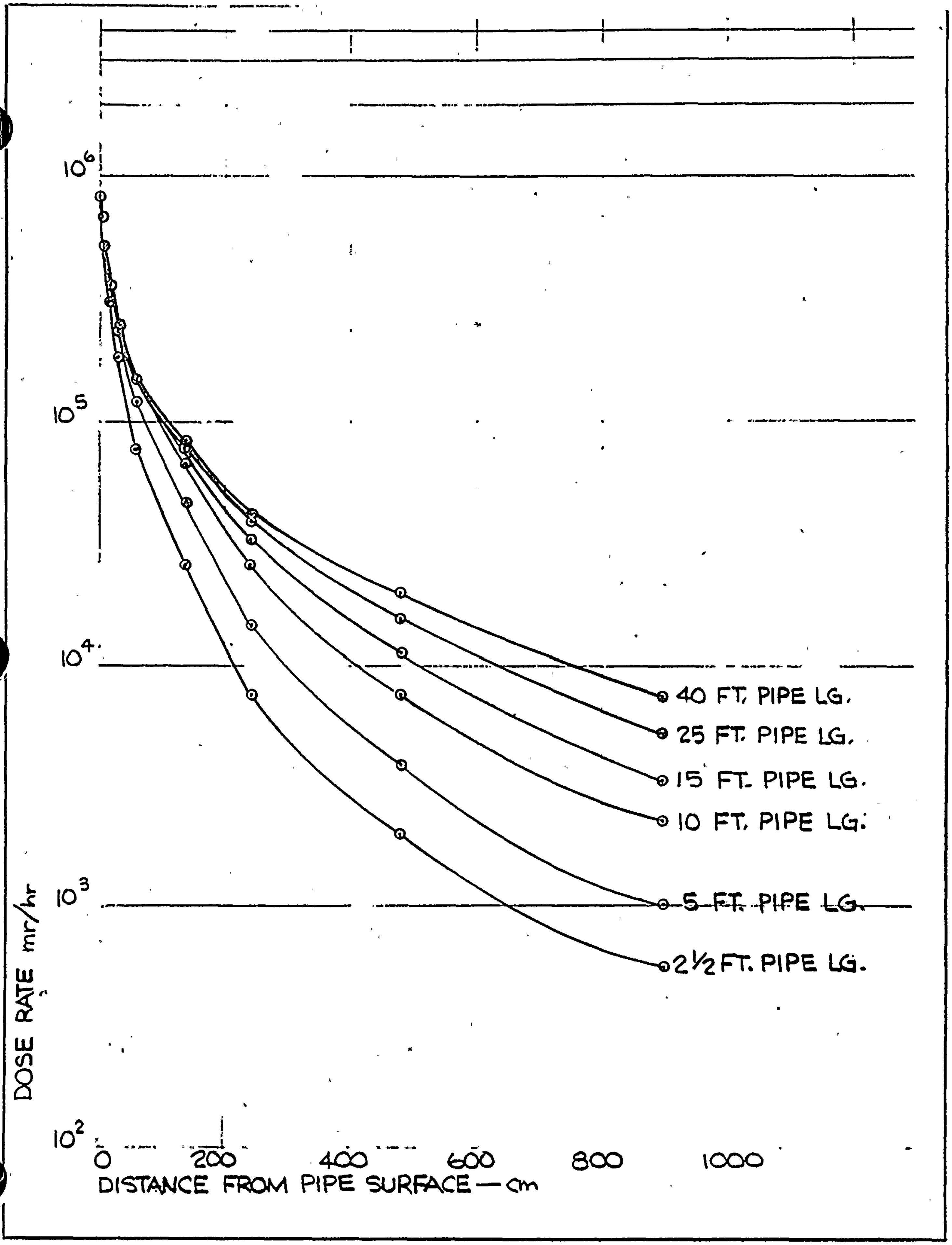

Figure F.30 Dose Rate at Pipe Mid-Plane vs. Distance from Pipe Surface
LPCI System, Schedule 80, D=14 in.

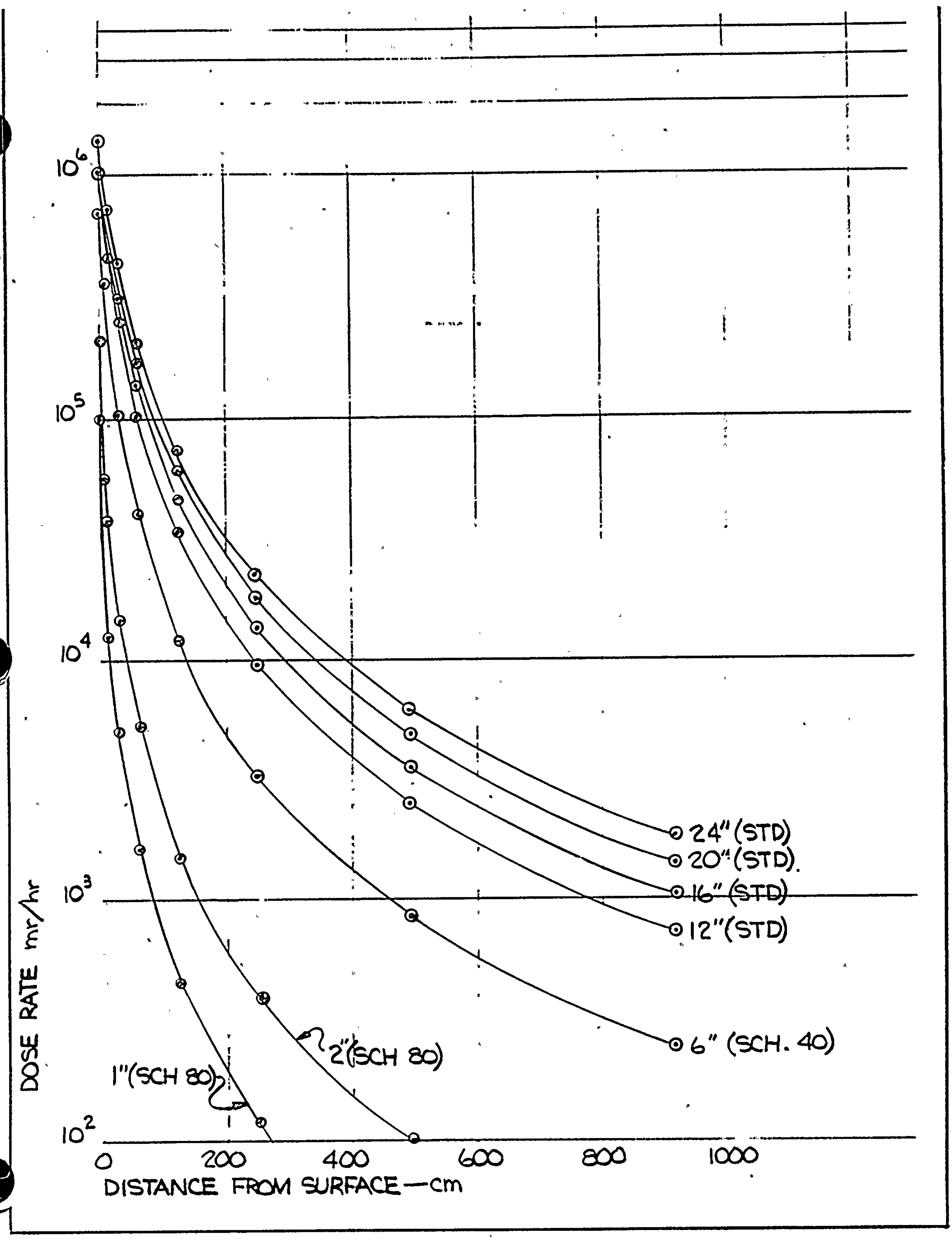

Figure F.31 Dose Rate vs. Distance from Pipe Surface
Pipe Length = 2.5 ft.

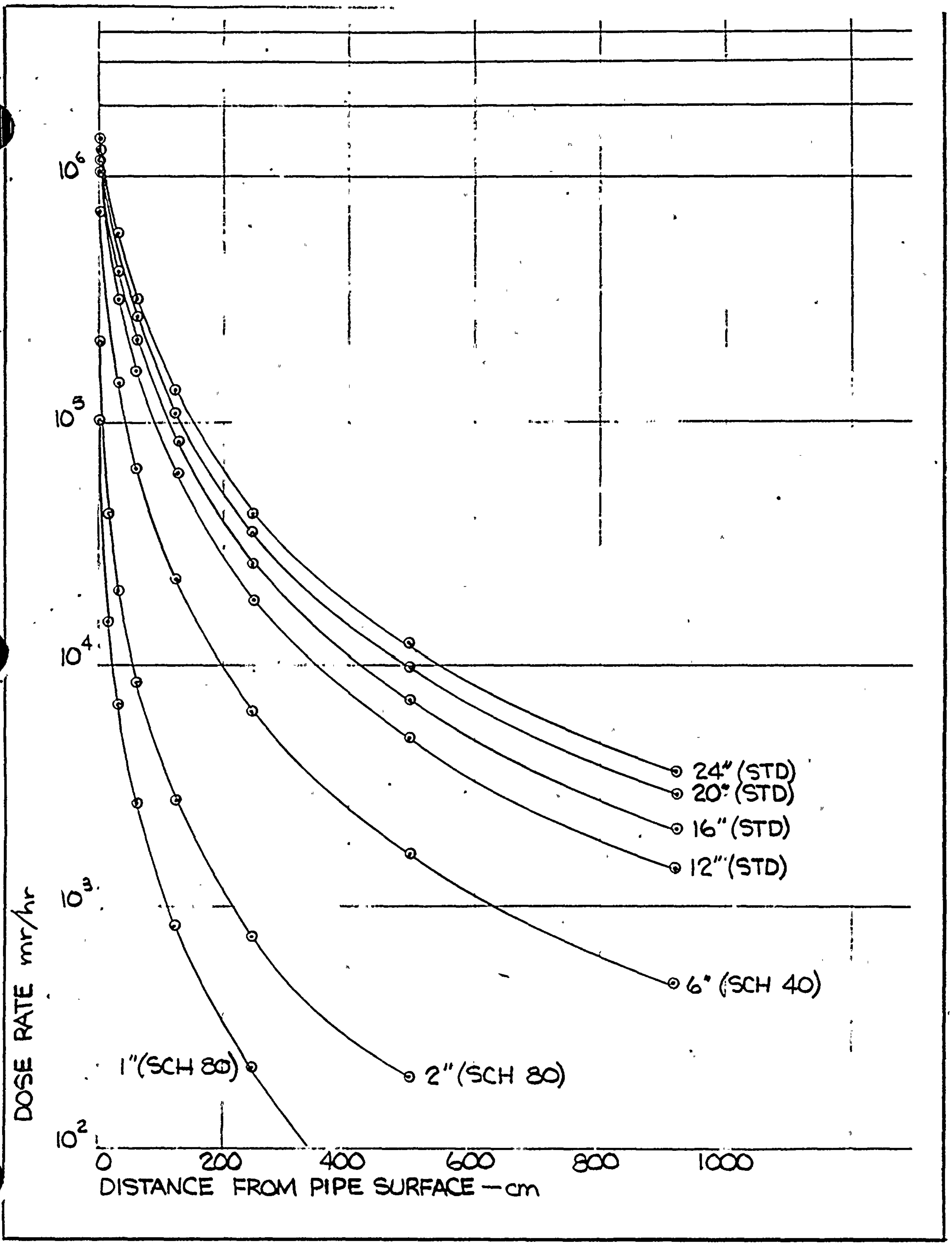

Figure F.32 Dose Rate vs. Distance from Pipe Surface

Pipe Length = 5 ft.

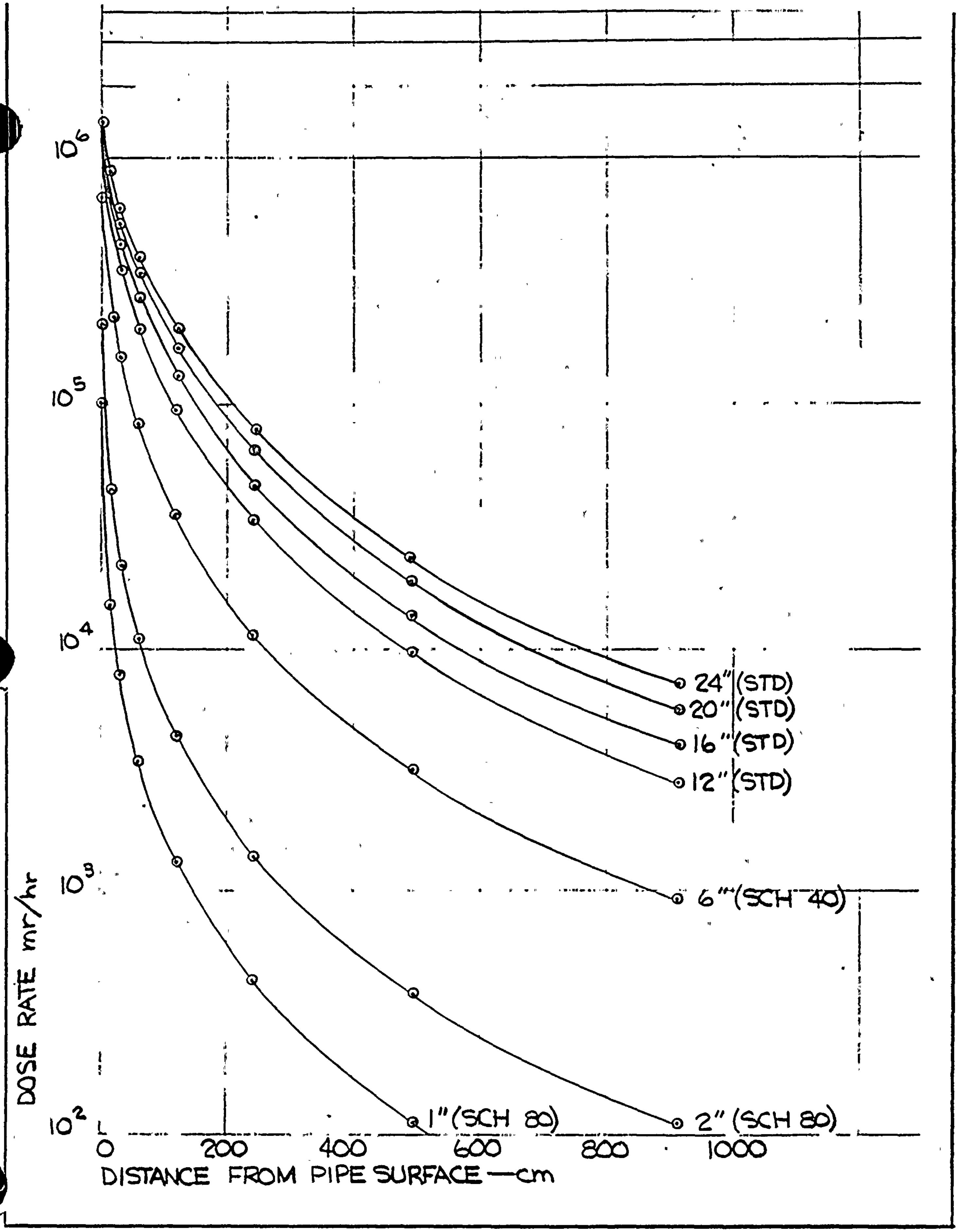

Figure F.33 Dose Rate vs. Distance from Pipe Surface

Pipe Length = 10 ft.

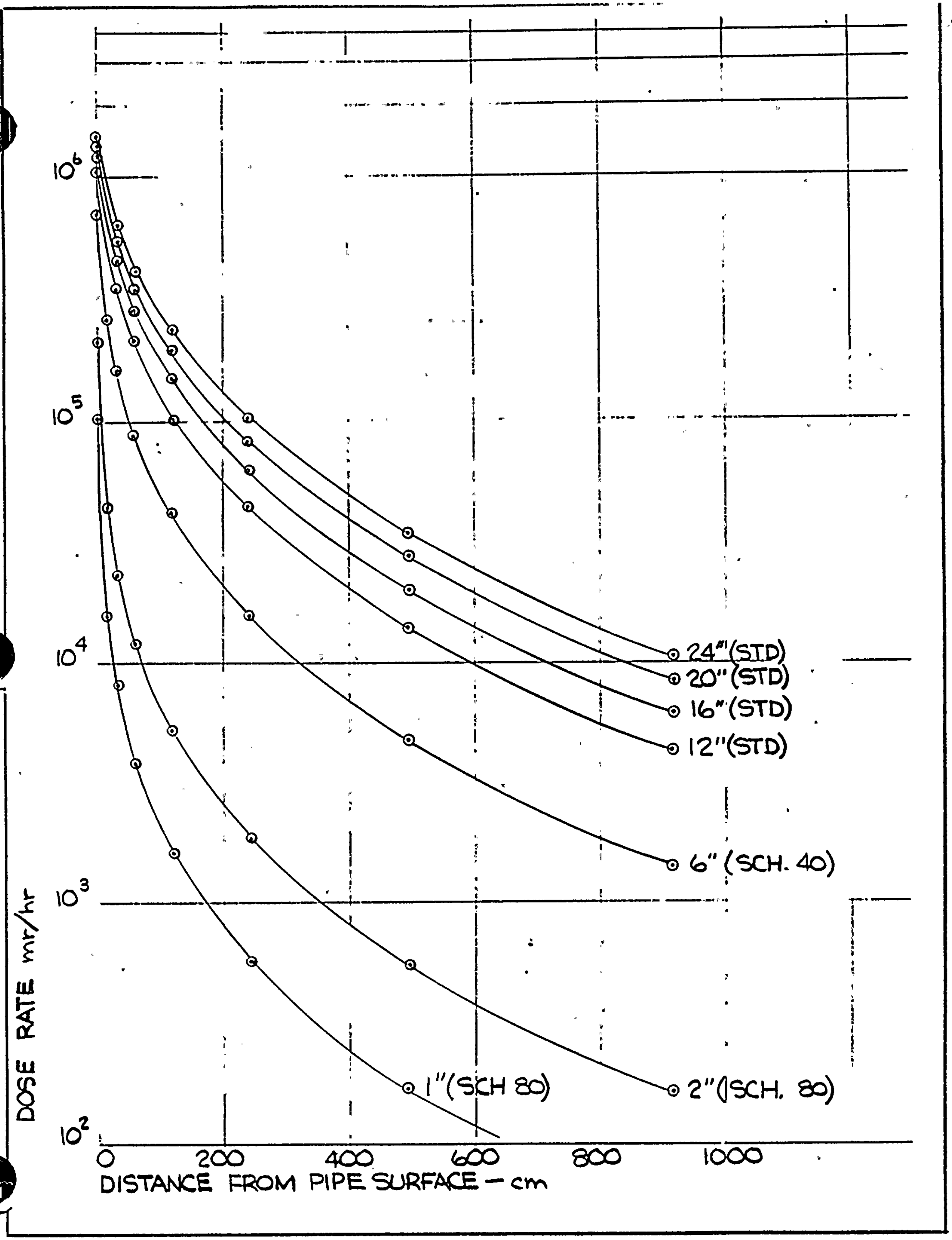

Figure F.34 Dose Rate vs. Distance from Pipe Surface

Pipe Length = 15 ft.

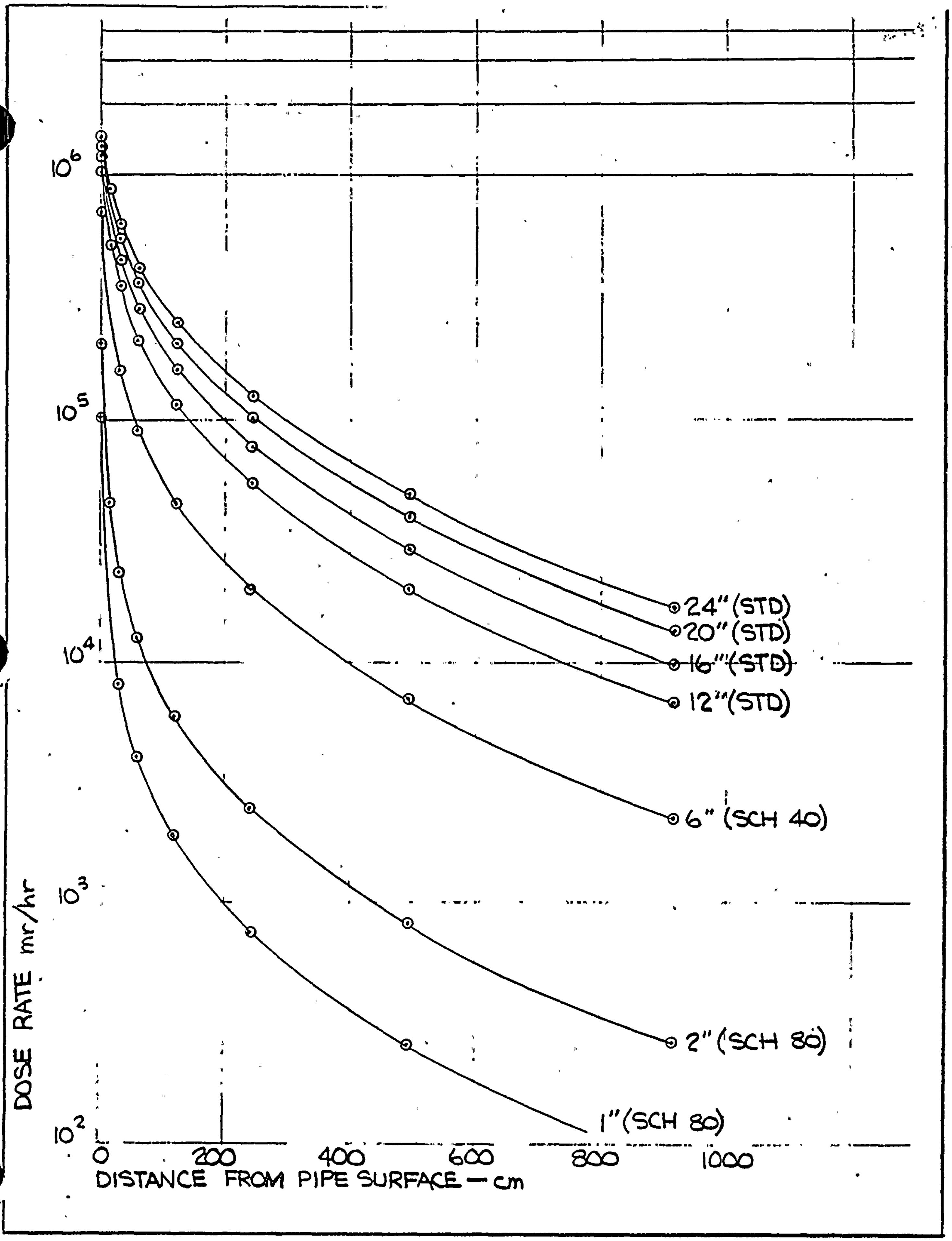

Figure F.35 Dose Rate vs. Distance from Pipe Surface
Pipe Length = 25 ft.

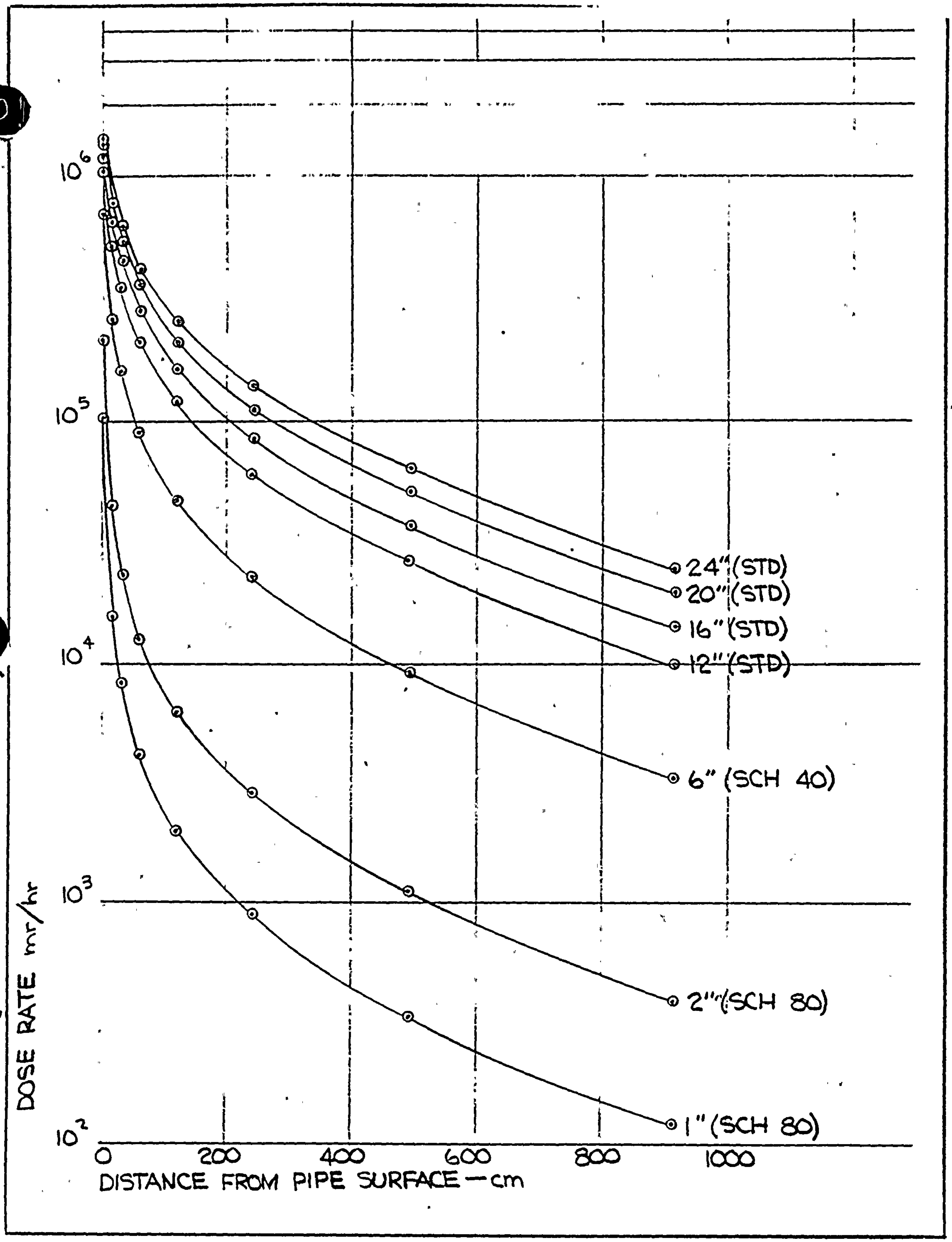

Figure F.36 Dose Rate vs. Distance from Pipe Surface
Pipe Length = 40 ft.

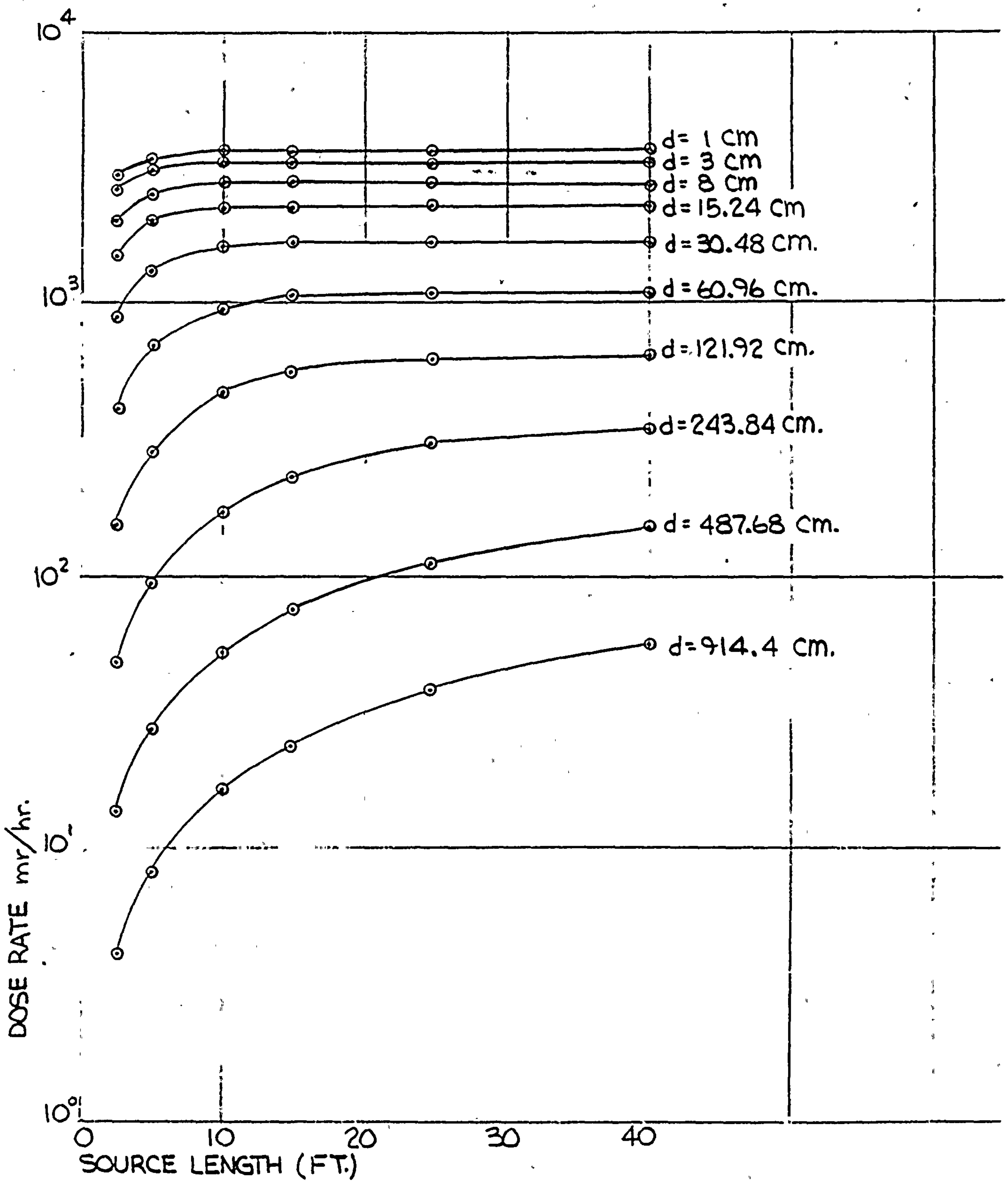

Figure F.37 Dose Rate vs. Pipe Source Length at Various Distances from Outer Pipe Surface

APPENDIX G: Beta Dose Contribution in Primary Containment

This appendix presents the calculated beta dose contribution due to a postulated LOCA. Table G-1 gives the average six-month source, Mev/sec, for the three fission product source types.

For a containment atmosphere source containing 100% noble gases plus 25% halogens, the source equals 2.0×10^{17} Mev/sec.
Using an infinite cloud model the beta dose rate is given by

$$D_\beta = 0.23\ \bar{E}_\beta \lambda \text{ (Rad/Sec)}$$

where,

$\bar{E}_\beta$ = average beta energy, and

λ = activity concentration, Ci/m^3

The average beta dose rate of 7.84×10^5 Rad/hr corresponds to a total beta dose contribution due to containment atmosphere of 3.4×10^9 Rads.

To determine the beta dose contribution due to plate-out of 25% halogens, a plate-out surface area of 3.3×10^7 cm^2 was conservatively assumed.

For $\bar{E}_\beta \sim 1$ MeV, this corresponds to a six month total beta dose contribution of 2.3×10^5 Rads.

TABLE G-1: BETA SOURCE STRENGTH

Radiation Source	Source Strength (Mev/Sec)
100% Particulates	1.29×10^{19}
100% Halogens	1.84×10^{17}
100% Noble Gases	1.54×10^{17}

APPENDIX H VITAL AREAS AND ACCESS ROUTES ANALYSED FOR POST-LOCA OPERATIONS

This appendix presents the methodology and assumptions used to determine the integrated dose to equipment and personnel for vital areas and access routes outside the Reactor Building during post-LOCA operations. The source term is the Reactor Building elevated vent with gaseous effluents being filtered by the Standby Gas Treatment System prior to discharge to the atmosphere.

H.1 Source of Radioactivity to the Reactor Building Elevated Vent

Two contributions were considered as the source of the radioactivity to the Reactor Building Elevated Vent.

- o Leakage from the drywell to the Reactor Building and discharged via the SGTS to the Reactor Building Elevated Vent was assumed at a rate of:

  0.5%/day = 2.1E-4 Hr^{-1}

- o Leakage from the assumed leaks on the Main Steam Isolation Valves in the Main Steam Tunnel was assumed at a rate of:

  0.22%/day = 9.2E-5 Hr^{-1}

Thus, the total leakage rate of activity from the primary system is assumed to be

0.72%/day = 3.0E-4 Hr^{-1}

H.1.1 Reactor Building Air Discharge Rate

All radioactivity considered outside the Reactor Building is assumed to discharge via the Reactor Building Elevated Vent.

The removal rate of the Reactor Building ventilation can be determined as follows:

$$\text{Removal Rate} = \frac{\text{SGTS discharge rate}}{\text{Reactor Building volume}}$$

SGTS discharge flow of 2430 Ft^3/Min was assumed which represents one volume change per day.

Reactor Building volume = 3.5E+6 Ft^3
Thus, the Removal Rate is:

$$\text{Removal Rate} = \frac{(2430\ Ft^3/Min)(60\ Min/Hr)}{3.5E{+}6\ Ft^3}$$

Removal Rate = 4.2E-2 Hr^{-1}

This removal rate was used in the determination of radiation levels outside the Reactor Building.

H.2 Calculation Methodology

A small computer program was written to complete the calculations for the 18 nuclides and five time periods and sum the results. The equation used to determine the dose is as follows:

$$\text{Dose(Rads)} = DF(j)\left(\frac{X}{Q_1} * TF * \frac{Q1j + Q2j}{3600}\right) \qquad \text{(H-1)}$$

Doseji..	=	Rads from jth nuclide for a ith time period
DF_j	=	Gamma Dose Factors for semi-infinite cloud $\left(\frac{\text{Rad m}^3}{\text{Ci - Hr}}\right)$ for jth nuclide.
X/Q_1	=	sec/m^3 for gaseous releases from the Reactor Building Vent to the atmosphere for the th time period.
RF	=	Removal fraction of activity via the standby gas treatment
TF	=	Transmission Fraction (1-RF)
TF	=	0.01 for particulates and iodines (99% efficiency or RF)
TF	=	1.0 for noble gases (WNP #2 FSAR Section 6.5).
Q1j	=	Integrated activity of jth nuclide over ith time period that was released via leaks in the main steam isolation valves (Curies/Hour).
Q2j	=	Integrated activity of jth nuclide over the ith time period that was released via leakage from the primary to secondary containment (Curies/Hour).

3600 = Conversion from hours to seconds

H.2.1 Assumptions Used in Calculation Methodology

The following equation from "Meterology and Atomic Energy" (Reference E-1) was used to determine the X/Q values shown in Table H.1

$$\text{Dilution} = 2.22(M)\left(3.16 + 0.1\frac{S}{(Aex)^{\frac{1}{2}}}\right)^2 \frac{V_{mea}}{Vex}$$

$= F_B$ (Building wake factor)

M = 1 if Intake and Exhaust same level

M = 2 if Intane and Exhaust separated by one floor

M = 4 if Intake is in Building wate cavity

S = Shortest Intake Exhaust arc length

Aex = Exhaust area

V mean = mean approach flow

Vex = mean exhaust flow

The intake was assumed to be for category F Conditions with a V mean = 1 meter/sec.

Then

$$\chi/Q = \frac{1}{F_B\ R_R}$$

F_B = Building wake factor

R_R = Release Rate from Reactor Building Vent (m^3/sec)

Concentration in reactor vent $C_V = Q/R_R$

Q = Curies/sec. released.

Concentration at intake $C_I = \frac{C_V}{F_B}$.
C_I also = $Q(\chi/Q)$.

Therefore:

$$C_i = \frac{C_V}{F_B} = Q(\chi/Q) = \frac{Q}{(F_B)(R_R)}$$

$$(\chi/Q) = \frac{1}{(F_B)(R_R)} = \text{total dilution factor } (D_F).$$

An F class stability was assumed for atmospher conditions. Five percent meteorology was then applied for time periods from 0 to 180 days. The dilution factors decrease by the following ratios for the time periods indicated.

Time(Hrs)	0-2	2-8	8-24	24-96	96-4320
Ratio	1.0	0.35	0.04	0.02	0.01

The dilution factors were multiplied by the fi percent meteorology ratios to determine the actual χ/Q values used in these computations. (See Table H.1)

H.2.2 Integrated Activity Equations used in this Analysis

The time dependent activity of each nuclide being released from the MSIV was analysed as follows:

$$\frac{dA1}{dt} = P\ A_o e\ (-\lambda + \frac{.0072}{24}t)$$

(H-3)

where

P = Fractional leak from MSIV per hour (9.2 E-5 H^{-1})

A_o = Initial activity of th nuclide in primary containment at = 0 hrs.

Thus, the activity concentration over a time period of t_1 to t_2 is:

(H-4)

$$Q1 = \int_{t_1}^{t_2} PA_o e^{-(\lambda + 3.0E-4)t}$$

or

$$Q1 = \frac{PA_o}{(\lambda + 3.0E-4)} \left[e^{-(\lambda + 3.0E-4)t1} - e^{-(\lambda + 3.0E-4)t2} \right]$$

The integrated activity concentration from the primary to secondary containment leakage, Q2, was calculated as follows:

$$\frac{dA_2}{dt} = KA_1 - L_2C_2 - \lambda A_2$$

(H-5)

where

K = Fractional leak rate from primary containment

$$\frac{.005}{24hr} = 2.1E-4 \ Hr^{-1}$$

A_1 = Activity in primary containment

$$A_1 \doteq A_o \ e^{-(\lambda + \frac{.0072}{24})t}$$

A_o	=	Initial activity (ci) at t = 0
$\frac{.0072}{24}$	=	Leakage removal rate from primary containment per hour
$\frac{.0072}{24}$	=	3.0E-4 Hr^{-1}
L2	=	Discharge rate from Reactor Building Vent via standby gas treatment = 2430 ft^3/min(60 min/hr)

L2 = 1.46E + 5 Ft^3/hr

C2	=	Activity concentration in secondary containment
A2	=	Curies in secondary containment
V2	=	Volume in secondary containment

$$C2 = \frac{A2}{V2} = \frac{A2}{3.5E + 6Ft^3}$$

Rearranging

$$\frac{dA2}{dt} = KA_o e^{-(\lambda+3.0E-4)t} - \frac{L2}{V2}A2 - \lambda A2$$

or

$$\frac{dA2}{dt} = KA_o e^{-F1t} - (\frac{L2}{V2} + \lambda)A2 \quad \text{(H-5A)}$$

$$dA2 = \left[KA_o e^{-F1t} - F_2 A2\right]dt$$

where

$$F_2 = \lambda + \frac{L2}{V2} \quad \text{and} \quad F_1 = (\lambda + 3.0E-4)$$

$$\dot{A2} = KA_1 - F_2A2$$

$$\dot{A2} + F_2A2 = r(t),\ r(t) = KA_o e^{-F_1 t} \tag{H-6}$$

solving

$$A2 = e^{-F_2 t}\left[\left(\frac{KA_o}{F_2 - F_1}\right) e^{(F_2-F_1)t} + c\right] \tag{H-7}$$

at $t=0$, $A_2=0$

$$C = \frac{-KA_o}{F_2-F_1} = \frac{-2.1E-4}{4.2E-2} A_o$$

$$C = -.005A_o$$

Thus,

$$A2(t) = .005A_o e^{-\lambda t}(1-e^{-(.042)t}) \tag{H-8}$$

and,

$$Q2 = \frac{L2A2(t)}{V2}$$

or

$$Q2 = \frac{1.4E + 5\ ft^3/hr}{3.5E + 6\ ft^3}\left[.005A_o e^{-\lambda_j t}(1-e^{-C2T})\right]$$

where

$$C2 = 0.042$$

thus,

$$Q2 = 2.11E\text{-}4\ A_o e^{-\lambda_j t}(1-e^{-C2t}) \qquad (H\text{-}9)$$

To determine the integrated concentration:

$$Q2(t) = 2.1E\text{-}4A_o \int_{t_1}^{t_2} \left[e^{-\lambda t} - e^{-(\lambda+C2)t}\right] dt \qquad (H\text{-}10)$$

Solving,

$$Q2 = 2.1E\text{-}4A_o \left[e^{-\lambda t_1} - e^{-\lambda t_2}\right] \frac{(e^{-C2t_1} - e^{-C2t_2})}{(\lambda + C2)} \qquad (H\text{-}11)$$

The values of Q1 and Q2 are substituted in for each Nuclide and each time period. Then using equation H-1, the dose commitment for each nuclide and each time period may be calculated, These results are presented in Section 6.3.

TABLE H.1: POST-LOCA X/Q VALUES USED FOR CALCULATIONS OF INTEGRATED DOSES OUTSIDE THE REACTOR BUILDING

Area	Time (in hours)				
	0-2	2-8	8-24	24-96	96-4320 (180 days)
Security Center	4.1-4	1.45-4	1.6-5	8.2-6	4.1-6
Auxiliary Security Center	2.4-4	8.4-5	9.6-6	4.8-6	2.4-6
Sample Analysis Area (EOC)	1.5-4	5.3-5	6.0-6	3.0-6	1.5-6
Nitrogen Supply to Accumulators	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Standby Service Water Pump Valves	2.4-4	8.4-5	9.6-6	4.8-6	2/4-6
Remote Shutdown Room	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Switchgear Room #1	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Switchgear Room #2	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Radwaste Control Room	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Battery Racks					
DC Battery Charger	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Motor Control Center	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Three Motor Control Centers/ Three Switchgears	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
DC Battery Charger and Rack	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Diesel Oil Tanks	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6
Solid Radwaste Control Panel	5.3-4	1.8-4	2.1-5	1.1-5	5.3-6

The standby service water pump valves are approximately 700 feet from the release point. This distance is too great to calculate a dilution based solely on a building wake factor. However, the conservative assumption will be made that the dilution at the valves is the same as at the auxiliary guard house which is only 420 feet.

* Read as 4.1×10^{-4} etc.