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Y/OWI/TM-36/22

TECHNICAL SUPPORT FOR GEIS: RADIOACTIVE WASTE ISOLATION IN GEOLOGIC FORMATIONS

Volume 22

Nuclear Considerations for Repository Design

April 1978

Prepared By

Science Applications, Inc.
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800 Oak Ridge Turnpike
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UNION CARBIDE

OFFICE OF WASTE ISOLATION
OAK RIDGE, TENNESSEE

prepared for the U.S. DEPARTMENT OF ENERGY *under*
U.S. GOVERNMENT *Contract W-7405 eng 26*

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Volume No.	Volume Title	Prepared by*
TM-36/1	Executive Summary	SAI
TM-36/2	Commercial Waste Forms, Packaging and Projections for Preconceptual Repository Design Studies	SAI
TM-36/3	Stratigraphies of Salt, Granite, Shale, and Basalt	D&M
TM-36/4	Baseline Rock Properties-Salt	D&M
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TM-36/21	Ground Water Movement and Nuclide Transport	D&M
TM-36/22	Nuclear Considerations for Repository Design	SAI
TM-36/23	Environmental Effluent Analyses	SAI

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*These documents were prepared by Science Applications, Inc. (SAI), 800 Oak Ridge Turnpike, Oak Ridge, Tennessee 37830; Dames & Moore (D&M), 20 Haarlem Avenue, White Plains, New York 10603; and Parsons Brinckerhoff Quade & Douglas, Inc. (PBQD), 1 Penn Plaza, New York, New York 10001.

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ABSTRACT

This volume, Y/OWI/TM-36/22, "Nuclear Considerations for Repository Design," is one of a 23-volume series, "Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations," Y/OWI/TM-36, which supplements the "Contribution to Draft Generic Environmental Impact Statement on Commercial Waste Management: Radioactive Waste Isolation in Geologic Formations," Y/OWI/TM-44. The series provides a more complete technical basis for the preconceptual designs, resource requirements, and environmental source terms associated with isolating commercial LWR wastes in underground repositories in salt, granite, shale and basalt. Wastes are considered from three fuel cycles: uranium and plutonium recycling, no recycling of spent fuel and uranium-only recycling.

Included in this volume are baseline design considerations such as characteristics of canisters, drums, casks, overpacks, and shipping containers; maximum allowable and actual decay-heat levels; and canister radiation levels. Other topics include safeguard and protection considerations; occupational radiation exposure including ALARA programs; shielding of canisters, transporters and forklift trucks; monitoring considerations; mine water treatment; canister integrity; and criticality calculations.

PREFACE

Project Background

One of the major problems related to the production of electricity by light-water nuclear reactors is the management of radioactive wastes generated by the use of nuclear fuel. However, the subject is considered amenable to a rational solution, and the technology involved is considered to be well within the capabilities of our present-day technological base.

An important step toward the realization of an effective waste management program is the preparation of the generic environmental impact statement for commercial waste management. A pivotal issue in waste management is the means of providing long-term, permanent storage of these wastes in a manner that best assures their isolation from the biosphere. Analyses spanning two decades have generated the widely supported concept for providing final isolation of these nuclear wastes in deep geologic formations. Therefore, the Office of Waste Isolation* was assigned the responsibility for preparation of those sections of this generic statement dealing with deep geologic waste isolation.

The original concept for deep geologic disposal was first advanced in 1957 when a National Research Council Advisory Committee of the National Academy of Sciences suggested the burial of solid radioactive wastes in salt deposits. To date, the majority of the research, development, and demonstration (RD&D) activities have been in salt. The current United States Department of Energy (DOE) National Waste Terminal Storage (NWTS) program calls for the selection of two sites overlying suitable salt formations by 1979, followed by the construction and start-up in 1985 of one NRC-licensed repository designed for the permanent disposal/isolation of commercial nuclear wastes in a salt formation at one of these two sites. In addition to this activity in salt, vigorous RD&D programs have been initiated to determine the appropriateness of various hard rocks as host media for a repository.

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*Operated by Union Carbide Corporation-Nuclear Division for the Department of Energy.

The deep-geologic-isolation portion of the generic statement considers repositories located in salt, granite, shale, or basalt. The repositories are designed to handle wastes from the nuclear fuel cycle in which the spent fuel is considered a waste (no reprocessing), or from either of two fuel cycles that include reprocessing--the cycle with uranium and plutonium recycle and that with only uranium recycle. To prepare this contribution, the Office of Waste Isolation contracted with Dames & Moore, Parsons Brinckerhoff Quade & Douglas, Inc., and Science Applications, Inc. In order to prepare this description, generic sites were defined, preconceptual repository designs completed, and resource requirements and effluents from the repositories identified. The preconceptual repository designs for the salt host formation were based on more than two decades of analysis and in situ experimentation. The data base upon which the repository design for the non-salt host formations were based is more more sparse since repository-oriented analyses of these formations have been proceeding only for the last couple of years. For each of the host rocks additional analyses were performed during the conduct of these studies. Details of this additional technical work are described within the twenty-three volumes of this report.

For an overview of these preconceptual repository design studies, the study objectives and scope, and the major study assumptions, the reader is referred to Volume 1 of this series, the "Executive Summary" (Y/OWI/TM-36/1).

Volume Summary (Y/OWI/TM-36/22)

This volume is a collection of independent solutions of problems that arose during the preconceptual design studies. These problems are quite diverse, although each has to do with nuclear material considerations.

For repository design purposes, it is necessary to know the types of wastes expected to be received, the characteristics of the waste canisters and shipping containers, e.g. the size, weight, decay-heat load, radiation levels, contamination levels, and how they are to be handled, so that the designer can begin the preliminary design of the receiving, handling, and storage facilities. This volume sets forth the baseline design considerations for the waste materials expected to be received at the repository, for shipping casks and shipping containers, and for methods of receiving the canisters and drums at the repository. The design considerations presented here are preliminary. There are ongoing programs, both at OWI and at other installations, intended to better define optimum canister materials and sizes.

Special nuclear material (SNM) must be safeguarded to prevent its diversion and to prevent the sabotage of equipment whose failure could result in its dispersal and endangerment of the public. Safeguard requirements for nuclear facilities have been developed by both the DOE and NRC but, to date, detailed safeguard requirements have been developed primarily for commercial nuclear power plants and little attention has been given specifically to waste repositories. However, the NRC is currently strengthening the safeguards requirements for both nuclear power plants and for fuel-cycle facilities. Therefore, in view of the current state of the waste-repository effort and the future uncertainty of NRC regulations, it appears impractical to discuss detailed safeguards requirements. Hence, this volume discusses safeguards considerations as they apply to a waste repository only in a general way.

Since a National Waste Terminal Storage facility will handle large quantities of radioactive material, its radiation-control program must ensure that radiation doses to occupational personnel are within acceptable limits. A radiation-protection program (defined in 10CFR20) must limit the maximum dose to individual operating personnel and, additionally, must ensure that the cumulative dose to the operations staff is as low as reasonably achievable (ALARA). Radiation-exposure requirements should be factored into the layout and design from the beginning. Early emphasis should also be placed on equipment reliability and good decontamination procedures.

Projected dose problems could occur in the cask-handling area, canister-storage feed room, canister-hoist shaft, canister-transfer galleries and transporter facilities. The hoist-shaft and transporter problems would likely be peculiar to mining facilities. The design of a shielded transporter for emplacing HLW canisters has been recognized as a problem and has been undergoing study in an ongoing program. In contrast, drums containing LLW-TRU or LLW have been defined as safe for surface contact, and handling with normal forklift trucks having unshielded cabs has been assumed to be an appropriate procedure. However, the specifications by BPNL of more radioactive drummed wastes have required shielding calculations to be made for drum handling. Calculations described indicate that for a 4-pallet x 4-pallet x 2-pallet storage configuration, and for a 0.25 mrem/hr dose rate to the forklift operator, the cab would require lead shielding of 7.8 cm for 5-year wet waste from a fuel reprocessing plant (FRP), 6.8 cm for 10-year FRP wet waste and 8.7 cm for material that has been contaminated with small amounts of 10-year PWR spent fuel.

A program of measuring and monitoring is described which will verify that the in-plant effluent control of radioactive materials is in accordance with 10CFR20 which restricts off-site doses to absolute values and to values that are as low as reasonably achievable (ALARA). This program will measure repository contributions to off-site exposure by direct radiation, radioactivity in breathing air, deposited radioactivity, and radioactivity in consumed materials such as water, milk, plants, meat, and fish.

Because of the geologic characteristics of granite, shale, and basalt, repositories built in these formations have a potential to allow water to enter the mine by permeation through the layers of rock constituting the formation. The water may enter the repository as a result of permeation either downward through the overlaying strata or by upflow through the lower layers. Since this water must be removed from the mine area in order to perform the normal underground activities of transporting and embedding waste canisters, waste repositories in granite, shale, or basalt formations must include provisions for handling this waste and for possible treatment of the water before discharge. The chief concern in this volume is that the water may become contaminated with radioactive isotopes before being pumped out of the mines and, hence, could require radwaste treatment before discharge. This report considers only the radioactive contamination of groundwater and does not address the issue of contamination by dissolved solids. For salt, water permeation is an extremely unlikely event (Class 8 or 9) and a water treatment system, therefore, does not seem appropriate.

Containment of the radioactive wastes by the canisters or drums in which they are packaged is essential to smooth repository operations since a breaching of one of these containers while it is being handled within the repository leads to operational difficulties and possibly to undesirable releases to the environment. Additionally, the baseline repository design assumed a period during which the stored canisters must remain in a readily retrievable state. Therefore, the preconceptual GEIS designs are based on a requirement of canister integrity throughout the retrievability period (five years).

In addition, analysis of potential nuclide transport after decommissioning of the facilities indicates that in the case of the hard-rock (granite, shale or basalt) repositories it could be desirable to extend the canister integrity to allow more time for radioactive decay of some of the more soluble isotopes. This reduces potential concentrations that could be transported to the biosphere by groundwater circulation from the repository. Canister integrity is less important for the repository designs in salt, since flooding of a salt repository is an extremely unlikely event (Class 8 or 9).

Since the assurance of canister retrievability is the driving consideration for canister integrity in salt, the canister integrity requirements are dictated by the length of the retrievability period. If retrievability is not required, or the retrievability phase of repository operation has been terminated, canister integrity in salt is not required. Chapter 6 discusses the primary sources of canister stresses, the severity of corrosion as it would impact canister retrievability, HLW stainless steel corrosion, and the corrosion of carbon steel drums in air.

Studies were made of criticality problems associated with the temporary storage of eight HLW-glass-containing canisters in one shipping cask. Following verification of computational methods, calculations indicated that the system of 8 canisters is very subcritical ($k_{eff} < 0.3$). However, a refined calculation is recommended which includes a more accurate representation of the actual geometry and a more accurate determination of the SiO_2 content. These calculations should include the fissionable actinides above plutonium and should account for reflector effects provided by the cask itself.

1.0 BASELINE DESIGN CONSIDERATIONS

1.1 INTRODUCTION

For preliminary repository design purposes, it is necessary to know the types of wastes expected to be received, the characteristics of the waste canisters (e.g. the size, weight, decay-heat load, radiation levels, contamination levels, and methods of handling), and similar information on the shipping containers for the packages. Once the material to be received has been defined, the designer can begin the preliminary design of the receiving, handling, and storage facilities for the repository. The purpose of this section is to set forth the baseline design considerations for the waste materials expected to be received at the repository. Baseline design considerations for shipping casks and shipping containers are also presented to cover the methods of receiving the canisters and drums at the repository. It should be emphasized that the design considerations presented here are preliminary and that there are ongoing programs, both at OWI and at other installations, to better define optimum canister materials and sizes.

1.2 FUEL-CYCLE AND WASTE-FORM CONSIDERATIONS

In this study, three alternative fuel cycles are considered. They are as follows:

Cycle No.	Fuel Cycle
I	Total recycle
II	Spent unreprocessed fuel (SURF) cycle*
III	U-only recycle

For purposes of this study, the following burn-ups were assumed for each recycle mode:

PWR: 33,000 MWD/MTHM at 30 MW/MTHM
BWR: 27,500 MWD/MTHM at 20.7 MW/MTHM

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*Also referred to as the "throwaway" or once-through fuel cycle.

The HLW and spent fuel are assumed to be 10 years old when received at the repository. All other waste is assumed to be 5 years old.

The high-level wastes, the intermediate-level wastes, and the cladding hulls from Cycles I and III, and the spent fuel assemblies from Cycle II are assumed to be packaged in canisters. Furthermore, the intermediate-level wastes and the cladding hulls from Cycles I and III are assumed to be packaged in the same size canisters as the high-level wastes in the glass form from Cycles I and III.

A separate set of criteria was developed for packaging the low-level transuranic (LL-TRU) wastes from Cycles I and III. These will be packaged in drums and can be handled by contact means. Since it is expected that low-level non-transuranic wastes (LLW-nonTRU) from these two cycles will be handled independently of transuranic contaminated wastes (as they are currently handled), these wastes are not considered in this study. However, the small amount of low-level wastes assumed to arrive at a spent fuel repository from Cycle II are assumed to be handled in the same manner as the LL-TRU wastes from the reprocessing cycles.

The waste forms considered for this study, as a function of the fuel cycle, are as follows:

o <u>High-Level Wastes in the Glass Form, Intermediate-Level Wastes and Cladding Hulls</u>

	Fuel Cycle	Type of Waste
I	Total Recycle	PWR-HLW (Glass)
		BWR-HLW (Glass)
		ILW-TRU
		CW
III	U-only Recycle (Pu with HLW)	PWR-HLW (Glass)
		BWR-HLW (Glass)
		ILW-TRU
		CW

o <u>High-Level Wastes in the Calcine Form</u>

	Fuel Cycle	Type of Waste
I	Total Recycle	PWR-HLW (Calcine)
		BWR-HLW (Calcine)
III	U-only Recycle (Pu with HLW)	PWR-HLW (Calcine)
		BWR-HLW (Calcine)

- LWR Spent Fuel Assemblies

	Fuel Cycle	Type of Assembly
II	Spent Unprocessed Fuel (SURF) Cycle	PWR or BWR

- Low-Level Wastes

Fuel Cycle	Source
I and III (LLW-TRU)	MOX fuel fabrication and reprocessing
II (LLW at lesser amounts than I and III)	Reactor operations and other sources

1.3 GLASS- AND CALCINE-CANISTER DESIGN CONSIDERATIONS

1.3.1 Canister Descriptions

1.3.1.1 HLW Glass Canisters

The HLW glass canister chosen for this study is one foot in diameter (ID) by 10 feet in length with an active length of 8 feet (Figure 1-1). The glass waste canister is a standard 12-inch pipe with a 3/8" wall thickness. The baseline concentrated glass waste has a specific volume of about 3 ft^3/metric ton of heavy metal reprocessed. Since the glass waste volume is 6.28 ft^3/ canister, the resulting waste density is 2.09 MTU equivalent*/canister. The weight of a filled canister is 1,720 pounds.

1.3.1.2 HLW-Calcine Canisters

The HLW calcine canister chosen for this study is 8 inches in diameter by 10 feet in length with an active length of 8 feet (Figure 1-1). The calcine-waste canister is a standard 8-inch pipe with a 5/16" wall thickness.

The calcine waste has a specific volume of 1.5 ft^3/MTU. Therefore, the equivalent fuel density is 1.85 MTU equivalent/canister and the weight of a filled canister is 660 pounds.

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* MTU equivalent/canister refers to the amount of spent reactor fuel that must be reprocessed to fill one waste canister.

1.3.2 Canister Decay-Heat Levels

The maximum allowable decay-heat level of HLW canisters is dictated by the allowable canister temperature[1] and thermomechanical stress considerations for the rock form chosen for disposal.[2] The maximum decay heat levels are as follows:

MAXIMUM ALLOWABLE DECAY HEAT LEVELS

Rock Type	HLW Glass Canisters	HLW Calcine Canisters
Salt	3.2 kW/canister	2.6 kW/canister
Granite	1.8 kW/canister	1.5 kW/canister
Shale	1.3 kW/canister	1.1 kW/canister
Basalt	1.2 kW/canister	1.1 kW/canister

The standard glass and calcine canisters (2.09 and 1.85 MTU equivalent/canister, respectively) have decay-heat power levels of 2.8 and 2.5 kw/canister when filled with PWR HLW. For BWR HLW, the values are 2.3 and 2.0, respectively, for glass and calcine. To achieve the lower heat loads for granite, the waste is uniformly mixed in a ratio of 40 percent inert material and 60 percent waste. For shale and basalt, the ratio is 57 percent inert material and 43 percent waste. This information is used to generate the actual decay-heat levels, as summarized below:

ACTUAL DECAY HEAT-LEVELS

Rock Type	PWR HLW Glass Canisters kW/Canister	PWR HLW Calcine Canisters kW/Canister	BWR HLW Glass Canisters kW/Canister	BWR HLW Calcine Canisters kW/Canister
Salt	2.8	2.5	2.3	2.0
Granite	1.7	1.5	1.4	1.2
Shale	1.2	1.1	1.0	0.8
Basalt	1.2	1.1	1.0	0.8

1.3.3 Canister Radiation Levels

The vertical mid-plane canister radiation levels for HLW glass and

HLW calcine canisters in salt are given in Tables 1-1 and 1-2. For granite and shale canisters, it is assumed that the radiation levels will decrease by an amount proportional to the inert-material volume fraction. The radiation levels used for repository design purposes are higher than the expected radiation levels and are given in Table 1-3.

1.4 SPENT-FUEL CANISTER DESIGN CONSIDERATIONS

1.4.1 Canister Descriptions

For Cycle II with no recycle of spent-fuel, the spent-fuel assemblies are encased intact in a spent-fuel-assembly canister and are assumed to be shipped to the repository 10 years after removal from the reactor.

1.4.1.1 PWR Assembly Canisters

The repository was designed to accept PWR fuel-assembly canisters with a 14" outside diameter and an overall length of 16 feet. The wall thickness of the canister was taken to be 3/8". The weight of a standard 17x17 PWR assembly is approximately 1,500 pounds[3] and the calculated weight of an empty canister is approximately 900 pounds. The repository is designed to accept assembly canisters weighing up to 3,000 pounds.

1.4.1.2 BWR Assembly Canisters

The repository was designed to accept BWR assembly canisters with a 10 3/4" outside diameter and an overall length of 16 feet. The wall thickness was taken to be 1/4". The weight of an 8x8 BWR assembly is approximately 600 pounds[3] and an empty canister weighs approximately 500 pounds.

1.4.2 Canister Decay-Heat Levels

The decay-heat power levels of assemblies irradiated as described in Section 1.2.2 and cooled for 10 years are as follows:

PWR 0.55 kW/assembly
BWR 0.18 kW/assembly

1.4.3 Canister Radiation Levels

For the burn-up and cool-down assumptions used in this study, the canister radiation levels were calculated and are given in Table 1-4. Also shown in Table 1-4 are the values used for the repository design purposes.

1.5 DESIGN CONSIDERATIONS FOR LLW-TRU OR LLW DRUMS

1.5.1 Drum Description

For the purposes of this study, it was assumed that the LLW-TRU from Cycles I and III or LLW from Cycle II will be stored in DOT type 17C drums. These drums have a 23 1/2" outer diamter, and 18 gauge wall, and are 34 1/2" high. The weight of a filled drum is assumed to be 900 pounds.

1.5.2 LLW-TRU or LLW Drum Decay Heat Level

The average decay heat level of an LLW-TRU or LLW drum was calculated to be 1.2×10^{-3} kW. For repository design purposes a maximum value of 3.0×10^{-3} kW was used.

1.5.3 LLW-TRU or LLW Drum Radiation Levels

The average radiation levels of the waste drums when received are shown in Table 1-5. Also shown are the values used for repository design.

1.6 GENERAL CANISTER AND DRUM DESIGN CONSIDERATIONS

1.6.1 Surface Contamination Levels

Per 49CFR173.397, the maximum canister and drum contamination levels were assumed to be:

220 dis/min/cm^2 beta-gamma emitting nuclides
22 dis/min/cm^2 alpha emitting nuclides

However, the maximum canister and drum contamination levels by transferable radioisotopes were assumed to be:[4]

100 dis/min/cm^2 beta-gamma emitting nuclides
3 dis/min/cm^2 alpha emitting nuclides

1.6.2 Materials of Construction

All canisters are assumed to be fabricated from 304L stainless steel. All drums are assumed to be fabricated from carbon steel.

1.6.3 Handling Considerations

It is assumed that all waste canisters will be equipped with a standardized lifting and handling device. For purposes of this study the device was assumed to be less than 12" high and located at the top and center of the canister.

It is also assumed that all drums will be handled by conventional drum-handling techniques.

1.6.4 Backfilling of Canisters and Drums

All canisters are assumed to be backfilled with 5 psig of helium to provide a controlled storage environment. LLW-TRU and LLW drums are not pressurized with gas.

1.6.5 Overpacks

Canisters and drums that leak will be overpacked to provide additional containment. The overpacks are assumed to be fabricated from carbon steel. Overpacked canisters and drums will be handled by the same methods as regular canisters and drums (Section 1.6.3).

1.7 DESIGN CONSIDERATIONS FOR SHIPPING CASKS

1.7.1 Shipping Cask Description

It is assumed that all canisters will be shipped in a shipping cask. The repository has been designed to accept loaded shipping casks up to 22' in length and 10' in diameter that weigh up to 125 tons (shipping cask plus lifting yoke weight).

Currently, the NLI 10/24 rail cask is the only cask that has been referenced for the transportation of waste that requires shielding. The NLI 10/24 cask has been NRC certified for use by rail and meets all the design criteria required for shipping waste canisters from any of the fuel cycles considered. The cask is rated at 70 kW heat load unventilated and 34,100 lbs. payload.

The cavity diameter is 45" and length is 179.5". Design work is currently underway at NL Industries, Inc. to modify the cavity length to approximately 200", which will then accommodate 16'-long spent-fuel canisters.

The shipping-cask loading assumptions used for repository design are as follows:

HLW - Cycles I & III	Glass:	8 Canisters per cask
	Calcine:	12 Canisters per cask
ILW, Cladding - Cycles I & III		8 Canisters per cask
Spent Fuel -	PWR:	7 Canisters per cask
	BWR:	12 Canisters per cask

Heat-load, payload, and criticality questions have been considered and approved for the configurations listed above. Sufficient room in the cask for a canister rack or basket was also taken into account.

1.7.2 Canister Positioning in Shipping Cask

The canisters are assumed to be placed in the shipping cask in the vertical position for removal by a lifting device. If the canisters are held in place by some device inside the shipping-cask to prevent movement and damage during shipping, it is assumed that the device does not inhibit vertical canister removal. The shipping-cask lifting device is assumed to be suitable for crane-hook operations.

1.7.3 Shipping Cask Handling

The shipping cask is assumed to be shipped in a horizontal position. It is assumed that equipment is available to raise the cask to a stable vertical position with a crane hook. Lifting and positioning devices that are compatible with the repository crane and special tools needed for tie-down removal are assumed to be supplied by the shipper.

After the tie-downs have been removed and the shipping cask placed in a vertical position, the shipping cask is assumed to be capable of being moved, using a crane hook with a yoke, and placed in a stable vertical position on a flat concrete floor.

1.8 DESIGN CONSIDERATIONS FOR DRUM-SHIPPING CONTAINERS

1.8.1 Drum Shipping Container Description

For purposes of this report, it is assumed that all LLW-TRU or LLW drums are shipped in shipping containers (regardless of the fuel cycle chosen). The repository has been designed to accept loaded shipping containers up to 8'x8'x20' long with a maximum weight of 44,300 lbs when filled with 42 drums. Normal shipments are assumed to utilize the DOT type 8B cargo containers.

1.8.2 Drum Positioning in Shipping Containers

The drums are assumed to be placed in the vertical position inside the shipping container and to be loaded so that unloading can be accomplished with the use of forklifts.

1.8.3 Shipping Container Handling

The shipping container is assumed to be shipped in a horizontal position and to be capable of being removed from the shipping vehicle by the repository crane.

1.9 GENERAL SHIPPING CASK AND SHIPPING CONTAINER CONSIDERATIONS

1.9.1 Atmosphere

All shipping cask and container atmospheres are assumed to be dry when received at the repository. Air or inert atmosphere in the shipping container will be acceptable at the repository.

1.9.2 Shipping Vessels

The repository is capable of receiving shipping casks and shipping containers via railroad car or truck. The maximum car or truck length is assumed to be 66 ft. All rail cars are assumed to meet American Association of Railroads standards for unrestricted service.

1. REFERENCES

1. Office of Waste Isolation/Union Carbide Corporation, Nuclear Division, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations - Thermal Analyses, Y/OWI/TM-36/19, Oak Ridge, TN, March, 1978.

2. Office of Waste Isolation/Union Carbide Corporation, Nuclear Division, Technical Support for GEIS: Radioactive Waste Isolation in Geologic Formations - Thermomechanical Stress Analysis and Development of Thermal Loading Guidelines, Y/OWI/TM-36/20, Oak Ridge, TN, March, 1978.

3. Kisner, R. A.; Turner, D. W., and Marshall, J. R.; OWI Waste Projections and Source - Term Data for FY 1977, Office of Waste Isolation/Union Carbide Corporation, Nuclear Division, Y/OWI/TM-34, Oak Ridge, TN, 1978.

4. Atlantic Richfield Hanford Company, Commercial Waste Engineering Progress Report, April-June 1975, ARH-LD-113.

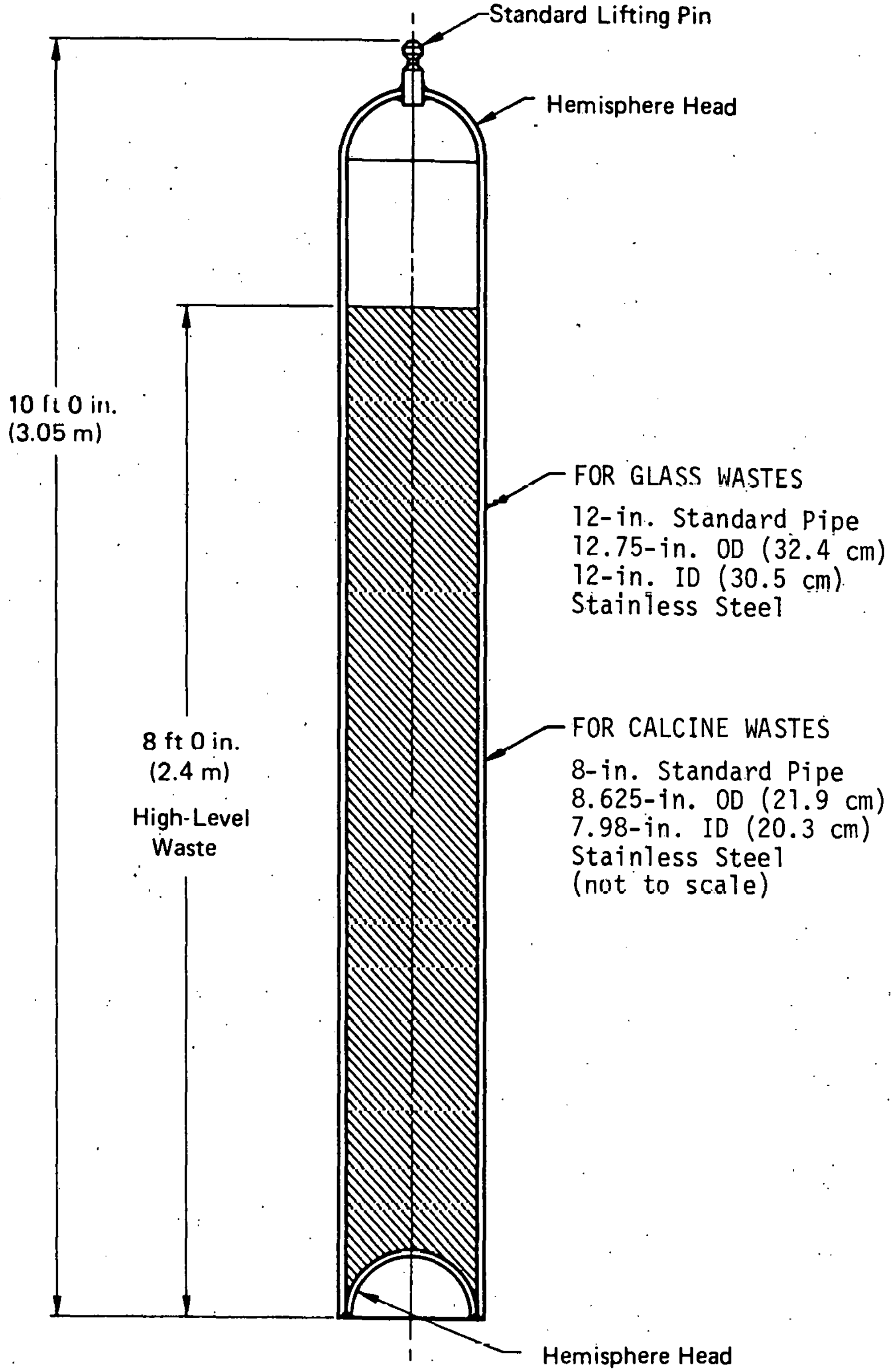

Figure 1-1. **HIGH-LEVEL WASTE CANISTER.**

TABLE 1-1

CANISTER RADIATION LEVELS*--TOTAL RECYCLE

Type of Waste	Waste Form	Average Heat per Canister kW	Average Gamma Radiation Intensity rem/hr		Average Neutron Radiation Intensity rem/hr	
			At Surface	10' From Centerline	At Surface	10' From Centerline
PWR-HLW	Glass	2.8	8.8×10^4	4.0×10^3	3.8×10^1	1.7×10^0
BWR-HLW	Glass	2.3	7.1×10^4	3.2×10^3	3.9×10^1	1.9×10^0
PWR-HLW	Calcine	2.5	8.1×10^4	7.1×10^3	8.0×10^1	6.8×10^0
BWR-HLW	Calcine	2.0	5.3×10^4	4.6×10^3	4.7×10^1	4.0×10^0
ILW	No Solids	1.2×10^{-3}	4.7×10^1	2.2×10^0	1.3×10^{-2}	5.4×10^{-4}
CW	Metals	1.5×10^{-2}	1.1×10^2	4.8×10^0	2.7×10^{-2}	1.3×10^{-3}

*For canisters in a salt repository. For canisters in granite and shale repositories these values should be multiplied by factors of approximately 0.60 and 0.43, respectively.

TABLE 1-2

CANISTER RADIATION LEVELS*--U-ONLY RECYCLE (Pu with HLW)

Type of Waste	Waste Form	Average Heat per Canister kW	Average Gamma Radiation Intensity rem/hr		Average Neutron Radiation Intensity rem/hr	
			At Surface	10' From Centerline	At Surface	10' From Centerline
PWR-HLW	Glass	2.6	8.5×10^4	3.9×10^3	1.4×10^0	6.2×10^{-2}
BWR-HLW	Glass	2.[illegible]	6.8×10^4	3.1×10^3	1.7×10^0	7.9×10^{-2}
PWR-HLW	Calcine	2.3	5.9×10^4	5.1×10^3	2.0×10^0	1.7×10^{-1}
BWR-HLW	Calcine	1.6	4.2×10^4	3.6×10^3	1.5×10^0	1.3×10^{-1}
ILW	Solids	1.2×10^{-3}	4.6×10^1	2.2×10^0	4.7×10^{-4}	2.1×10^{-5}
CW	Metals	1.5×10^{-2}	1.1×10^2	4.7×10^0	1.1×10^{-3}	4.7×10^{-5}

*For canisters in a salt repository. For canisters in granite and shale repositories these values should be multiplied by factors of approximately 0.60 and 0.43, respectively.

TABLE 1-3

RADIATION LEVELS USED FOR REPOSITORY DESIGN PURPOSES

		Average Gamma Radiation Intensity rem/hr		Average Neutron Radiation Intensity rem/hr	
Type of Waste	Waste Form	At Surface	10' From Centerline	At Surface	10' From Centerline
PWR-HLW	Glass or Calcine	2.0×10^5	1.0×10^4	1.0×10^2	1.0×10^1
BWR-HLW	Glass or Calcine	2.0×10^5	1.0×10^4	1.0×10^2	1.0×10^1
ILW	Solids	1.0×10^2	1.0×10^1	1.0×10^{-1}	5.0×10^{-3}
CW	Metals	1.0×10^3	1.0×10^1	1.0×10^{-1}	5.0×10^{-3}

TABLE 1-4

RADIATION LEVELS FOR PWR AND BWR SPENT FUEL ASSEMBLIES

	Average Gamma Radiation Intensity rem/hr		Average Neutron Radiation Intensity rem/hr	
Type of Assembly	At Surface	10' From Centerline	At Surface	10' From Centerline
PWR-Calculated Value	3.5×10^4	1.8×10^3	3.0×10^{-1}	1.8×10^{-2}
PWR-Design	7.0×10^4	3.5×10^3	1.0	5.8×10^{-1}
BWR-Calculated Value	2.6×10^4	1.3×10^3	2.2×10^{-1}	1.3×10^{-2}
BWR-Design	5.0×10^4	2.5×10^3	1.0	5.8×10^{-1}

TABLE 1-5

AVERAGE RADIATION LEVELS OF THE LLW-TRU OR LLW DRUMS*

	Average Gamma Radiation Intensity rem/hr		Average Neutron Radiation Intensity rem/hr	
Fuel Cycle	At Surface	10' From Centerline	At Surface	10' From Centerline
U & Pu Recycle	4.7×10^{-3}	2.2×10^{-4}	1.3×10^{-2}	5.4×10^{-4}
Spent Fuel Cycle	9.4×10^{-3}	4.4×10^{-4}	1.3×10^{-2}	5.4×10^{-4}
U-Only Recycle (Pu with HLW)	4.7×10^{-3}	2.2×10^{-4}	4.7×10^{-4}	2.1×10^{-5}
Repository Design Values	2.0×10^{-2}	1.0×10^{-3}	2.0×10^{-2}	1.0×10^{-3}

*LLW-TRU wastes from Cycles I and III; LLW only from Cycle II.

2.0 SAFEGUARD CONSIDERATIONS

2.1 INTRODUCTION

2.1.1 Background

Since the acquisition of strategic quantities of special nuclear material (SNM) by terrorist groups could lead to endangerment of the public health and safety, SNM is safeguarded to prevent its diversion and to prevent the sabotage of equipment whose failure could result in the dispersal of SNM. Safeguard requirements for nuclear facilities have been developed by both the DOE and NRC. A waste repository for disposal of waste generated by commercial nuclear plants must meet the NRC requirements set forth in 10CFR70 and 73 and any future safeguards requirements developed by NRC for waste repositories.*

To date, detailed safeguard requirements have been developed primarily for commercial nuclear power plants. Less consideration has been given to the development of criteria for other fuel cycle facilities and little attention has been given specifically to waste repositories. However, the NRC is currently strengthening the safeguards requirements for both nuclear power plants and other fuel cycle facilities.

As a result, the NRC has indicated that requirements for fuel-cycle facilities will be reviewed and, in all probability, be further strengthened in the future.[1] Therefore, given the current state of the waste repository effort and the future uncertainty of NRC regulations, it would be impractical to discuss detailed safeguards requirements at this time. The purpose of this section, then, is to discuss general safeguards considerations as they apply to a waste repository.

2.1.2 Environmental Impact of Safeguards

Safeguards considerations involve such things as the number and strength of physical barriers (walls, fences, doors, etc), the number

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*If the DOE were to construct and operate the repository, DOE requirements could also apply. For purposes of this report, only NRC requirements were considered.

and location of entry ports, the number of guards, and the extent of searches and escorts. While the actual level of required safeguards may vary depending upon waste form, fuel cycle or rock type chosen, it is not expected that the relative environmental impact of safeguards will be significantly different for the different waste-form, fuel-cycle or rock-type concepts. Also, the incremental environmental impact of future changes in regulations is not expected to be significant.

2.2 WASTE FORM CONSIDERATIONS

When developing safeguards criteria, consideration must be given to the type and form of material to be safeguarded. 10CFR73.6 states that special nuclear material which is not readily separable from other radioactive material, and which has a total external radiation dose rate in excess of 100 rems per hour at a distance of three feet from any accessible surface without intervening shielding, is exempt from the requirements of 10CFR73.30 through 73.36 and of 73.60, 73.70 and 73.72. HLW from Cycles I and III and spent fuel assemblies in shipping casks meet the requirements 10CFR73.6. Therefore, it can be expected that, because of the waste form, safeguard requirements for a waste repository will be less stringent than for nuclear power plants and reprocessing plants.

2.3 ITEM ACCOUNTABILITY

Because of the form of the waste being received and the large quantities of waste to be received, the repository can be expected to develop rigorous item-accountability requirements. A recommended item-accountability plan is described in this section. This accountability plan is in accordance with 10CFR70 and consistent with the intent of NRC Regulatory Guides 5.1, 5.26, 5.28, 5.29, and 5.49.

2.3.1 Facility Organization

Responsibility for nuclear-waste accountability should be assigned to a single individual at an organizational level sufficient to provide independence of action. The nuclear-waste custodian service should be separated in a manner that ensures that the activities of one organizational unit (or individual) controls and checks on the activities of another organizational unit (or individual).

An up-to-date manual of approved accountability procedures should be maintained and reflected in the facility process specifications, packaging instructions, and standard operating procedures. A formal program for the training and periodic requalification of personnel assigned to nuclear-waste accountability functions should be developed and documented.

2.3.2 Facility Operation

Item Control Areas (ICAs) should be established for physical and administrative control of nuclear waste. The number of ICAs established at a plant should be sufficient to localize nuclear-material-inventory discrepancies. The custody of all nuclear waste within any ICA should be the responsibility of a single individual. Each ICA should be kept separate from assemblies assigned to any other area. The quantity of SNM moved into or out of such an ICA is represented by a number of discrete units. In other words, nuclear waste is inventoried and moved into or out of ICAs by unique item identity and count.

2.3.3 Identification Controls

The repository will be required to determine that all individually identifiable waste assemblies received at the facility in irradiated or non-irradiated form are still on hand or have been shipped to other facilities. This should be done by identifying each waste package by a unique number. If one or more packages are missing and not accounted for, an investigation may be required. A system of management audits and reviews is required to ensure the adequacy of these controls.

2.3.4 Storage and Internal Transfers

A documented system of control over the nuclear material within a facility should be maintained. All transfers of nuclear waste between ICAs should be documented and validated for individual assembly by identity, count, and a previously measured valid SNM content (at time of fabrication). A centralized accounting system using double entry bookkeeping with subsidiary accounts for each ICA should be established and maintained. Storage- and internal-handling controls should be established, maintained, and followed to provide information on a timely basis related to the identity, quantity, and location of all waste packages within a facility. A unique item-identification system should be established to ensure that no two waste packages can have the same number. Records showing the identity, source, and disposition of all assemblies should be maintained.

2.3.5 Shipping and Receiving

It is recommended that shipments and receipts be independently accounted for by both the shipper and receiver. Shipper/receiver differences should be reviewed and evaluated on an individual container or lot basis, on a shipment basis, and on a cumulative basis. Appropriate investigative action should be taken on any shipper/receiver differences. The detection of missing material and, in turn, the discovery of diversion or theft should be timely. Receipts should be piece-counted and item-identified for comparison with the shipment bill-of-lading as soon as possible. The integrity of the tamper-saving devices on the shipping casks should be verified. Records of shipper/receiver difference evaluations, investigations, and corrective actions must be maintained on file at the facility for at least five years.

2.4 PHYSICAL PROTECTION REQUIREMENTS

Because of the form of the waste, the radiation levels, and the size and weight of the canisters, it is expected that no additional physical protection will be required for the repository buildings other than that required for confinement of the radioactive material. Therefore, preliminary protection requirements were developed which emphasize the area around the repository. For protection requirements, the site can be divided into three areas:

Area A -- the 200 acres which contain all the surface facilities,

Area B -- the 2,000 acres of the inner controlled area surrounding all surface facilities which encloses the maximum extent of underground development, and,

Area C -- 16,000 acres of the outer controlled area.

The actual level of protection required (i.e. the number of guards, the size of the isolation zone, and so on) must be developed at the detailed design phase of the project. Detailed plans must be submitted to the NRC and approved before repository operation begins. It is expected, however, that the detailed plan would include the items discussed in Sections 2.4.1 and 2.4.2.

2.4.1 Area "A" Requirements

Precautions for Area A are the most extensive and cover perimeter and portal safeguards. Perimeter safeguards include:

- Isolation zones - permits the assessment system to provide accurate information for all intrusion contingencies.
- Physical barriers - such as barbed wire fences and buildings constructed of stone, brick, cinder block, concrete, steel or comparable material.
- Illumination - sufficient for the monitoring and observation of outdoor areas adjacent to the physical barrier at the perimeter of the protected area and for the detection of penetration or attempted penetration of the protected area on the isolation zone adjacent to the protected area.
- Surveillance - at least one member of the physical security force would have the capability of observing activities in the isolation zone at all times; of clearly viewing the isolation zone before the intruder has time to leave the area; and of transmitting equivalent surveil-lance data to the central and secondary alarm stations.

- Security locks - manipulation resistant, with an effective system for managing all locks, keys, combinations, and other related equipment employed in securing the perimeter of the protected area.
- Intrusion detectors - tamper proof and self checking, providing high-assurance detection of all penetrations into the protected areas, with sufficient emergency electrical power for operation during the attack.
- Guard force - including fixed guards positioned at security posts at the perimeter of the protected area and guards patrolling the barrier at the perimeter.

Portal safeguards covering personnel and vehicle portals include:

Physical-structure requirements
Security locks
Guard force
Detection probabilities
Intrusion detectors of men and material
Access authorization
Communications

2.4.2 Area "B" and "C" Requirements

Safeguards for Area B consist of fencing with periodic signs indicating NO DRILLING OR MINING ALLOWED and daily patrols by a guard force around the periphery. For Area C a patrol road would be constructed around the perimeter. In addition, periodic signs would be placed around the periphery stating that the government owns subterranean rights to the area and that permission must be obtained before commencing drilling or mining within the respective area, and weekly patrols will be conducted around the periphery by a guard force.

2. REFERENCES

1. NUREG-0116, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle", October 1976, Section 4.10.

3.0 OCCUPATIONAL RADIATION EXPOSURE

3.1 INTRODUCTION

Since a National Waste Terminal Storage facility will handle large quantities of radioactive material, a radiation control program must be established to ensure that radiation doses to occupational personnel are within acceptable limits. The NRC requirements for radiation protection are contained in 10CFR20. A radiation protection program as defined in 10CFR20 must limit the maximum dose to individual operating personnel and, additionally, it must ensure that the cumulative dose to the operations staff is as low as reasonably achievable (ALARA). These concepts play an integral role in determining the facility design and maintenance requirements.

Control of occupational radiation exposure begins with the repository layout. Radiation exposure requirements should be factored into the design from the beginning. Early emphasis should also be placed on equipment reliability and good decontamination procedures. Additionally, large dose commitments can come about as the result of maintenance activities in highly active cells and steps should be taken to ensure that the doses from maintenance activities are ALARA.

3.2 ALARA CONSIDERATIONS

The concept of ALARA implies that there is no fixed dose to the operating staff which satisfies the requirements of ALARA (i.e. ALARA requirements cannot be satisfied by stating that the average staff radiation dose will be limited to an arbitrary fraction of, say, 1/10 of 10CFR20 limits). As NRC Regulatory Guide 8.8 states:

> "The concept of maintaining occupational radiation exposures ALARA does not embody a specific numerical guideline at the present time. Rather it is a philosophy that reflects specific objectives for radiation dose management in 1) establishing an ALARA Program; 2)

designing facilities and selecting equipment; 3) establishing a radiation control program, plans, and procedures; and, 4) making supporting equipment, instrumentation, and facilities available."

While there have been no numerical cost/benefit guidelines proposed for limiting occupational radiation exposures, numerical guidelines have been set forth for ensuring that off-site doses are ALARA.* However, it is possible to calculate a numerical cost/benefit number for staff radiation doses which could serve as a guide to aid repository design efforts. This approach is discussed in 3.2.5.

3.2.1 Establishing an ALARA Program

The establishment of an ALARA Program requires a management policy for implementing the program and an independent staff to ensure that the ALARA requirements are continually applied to design, operation, inspection, modification, and maintenance activities. Additionally, a radiation-protection training program must be established for all personnel who either work in radiation areas or work with radioactive materials and for all personnel who supervise those who work in radioactive areas or with radioactive materials. More explicit details on establishing an ALARA program are discussed in NRC Regulatory Guide 8.8.

3.2.2 Designing the Facility and Selecting Equipment

Arriving at a design for the repository that reflects the incorporation of radiation protection design features to allow operation without unreasonable dose exposure requires careful planning and review of the facility and procedures from the initial design phase to the actual operating phase. Figure 3-1 illustrates the type of feedback and cooperation needed to achieve the goal of designing and running the

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*The requirements are stated in 10CFR50, App. I., and Regulatory Guide 1.110 discusses suggested methods for achieving those guidelines.

repository economically with an acceptable dose commitment. The input to the initial design requires personnel familiar with radiation protection such as health physicists or shielding engineers, along with the system and equipment designers.

Once the preliminary design has been completed, all sources of radiation in the repository should be quantified (see 1.0, Baseline Design Considerations, Tables 1-1 thru 1-3, for canister radiation levels). As the design progresses, these calculations should be expanded to include contamination of cells by transferable radionuclides and contamination by leaking canisters.

In order to control unnecessary or inadvertent exposure of personnel to radiation, all locations within the repository should be zoned as to the level of radiation expected when the repository is operating (see 3.4). This zoning should be done during the design phases of the repository. After operation begins, actual dose rates in the different locations should be measured routinely to determine the actual or current radiation levels. Zones associated with high dose rates should be kept as small as reasonably achievable, while still allowing any tasks required in the zone to be performed. Radiation zones to be occupied a large amount of time by personnel should be designed to the lowest practicable dose rates.

Design of the repository should include measures to control access to radiation areas and to provide adequate shielding to keep doses within radiation areas ALARA. For areas of high radiation potential (>100 mrem/ hr), 10CFR20 requires design features and administrative controls that provide effective ingress control, ease of egress, and appropriate warning devices and notices. According to Reg. Guide 8.8 access control of radiation areas should reflect the following:

(1) Extraordinary design features to avoid any potential dose to personnel large enough to cause acute biological effects for a short exposure period.

(2) Administrative controls such as standard operating procedures which are effective in preventing inadvertent exposures of personnel and the spread of radiocontaminants during the transportation from one location to another onsite.

(3) Features such as platforms, walkways, stairs, or ladders that permit prompt accessibility for servicing or inspecting components located in higher radiation zones.

Engineering controls shall be provided for protection against airborne radioactive material as part of the repository design features. Use of individually worn respirators for routine tasks by repository personnel shall not be part of the standard operating procedure. Instead, engineered design features shall be a part of the overall design. For non-routine or emergency situations, the use of respirators may be appropriate if the design of engineered controls is not feasible. Reg. Guide 8.8 suggests that design features of a radiation facility should include:

(1) Use of air pressure gradients and air flows from areas of low potential airborne contamination to areas of higher potential airborne contamination. Periodic checks to ensure maintainability should be part of the standard operating procedures.

(2) Auxiliary ventilation systems that augment the permanent system for providing local control of airborne contaminants where equipment or containers with potential airborne sources are opened to the atmosphere.

Monitoring of airborne radioactive material concentrations and dose rates in various areas of the repository is a desirable feature that should be implemented as part of the original design of the repository. The design of such a monitoring system should include the following:

(1) Readout capability at the main radiation-protection access-control point,
(2) Optimum placement of detectors,
(3) Ability to detect failed instrumentation,
(4) Local readouts and indicators,
(5) A clear and unambiguous readout,
(6) Instrument ranges capable of giving accurate readings over all anticipated radiation levels and concentrations,

(7) Ability to keep a "hard" copy or record of all readouts.

Additional factors to be considered during the design are provisions for isolation and decontamination of radioactive equipment and the selection of appropriate materials for use in high radiation fields.

3.2.3 Radiation Protection Program

A radiation protection program must be established. The basic purpose of a radiation protection program is to control the number of personnel who enter high radiation areas, the amount of time that the personnel spend in the high radiation areas, and the magnitude of the potential dose to personnel.

3.2.4 Radiation Protection Facilities, Instrumentation, and Equipment

A well-trained and equipped radiation-protection staff is an essential element of a radiation-control program. Details can be obtained from Regulatory Guide 8.8.

3.2.5 ALARA Cost/Benefit Analysis

Cost/benefit analyses lend themselves to ALARA calculations since there is a direct cost involved in radiation reduction (e.g. in providing additional shielding) while there is a direct benefit from lowering radiation doses. For off-site doses, the NRC has chosen a value of $1,000/man-rem as the basis for cost/benefit analyses. This value was calculated conservatively as the health cost associated with low-level radiation exposure. From an ALARA standpoint, this number is applied by calculating the off-site dose and then determining if design changes to reduce that dose are less expensive than $1,000/man-rem of dose reduction. A bounding calculation can also be made by calculating the maximum amount of money that should be spent to reduce the dose to 0. This bounding value is also referred to as the Total Direct Cost (TDC) and Appendix B discusses how a TDC analysis is performed.

For occupational radiation exposures, consideration must also be given to the replacement cost of the operating personnel (i.e., salary, training costs, cost of extra personnel, etc.) in addition to the $1,000/man-rem cost of health effects. Replacement costs vary depending on the geographical location of the plant, the level of training required for facility operation, the back-up personnel capabilities of the operating organization and so on. Because the replacement value can vary greatly depending upon these factors, no requirements have been developed for a replacement value. However, this should be taken into account for on-site ALARA calculations, and a value of about $5,000/man-rem would provide an accurate assessment in the absence of detailed calculations.

For purposes of ALARA calculations, an initial staff dose "budget" can be calculated and a cost/benefit analysis performed as for the off-site doses. A convenient starting point for calculating staff dose budgets is to assume that the average dose will be a specified fraction of the 10CFR20 value of 5 rem/year. This budgeted dose is used for preliminary design purposes. Then a cost/benefit analysis is performed after the preliminary design phase is completed to determine if a further dose reduction is justifiable.

3.3 MAINTENANCE CONSIDERATIONS

Because of the high radiation fields associated with the waste canisters, and because of the projected throughput of canisters, maintainability of equipment becomes an important element in the calculation of personnel doses. Determination of the repository budget dose is very dependent upon the amount of time spent maintaining "hot" systems. For this reason, identified problem areas in the repository are basically areas where equipment reliability is crucial.

Areas identified as projected dose-commitment problems are as follows:

(1) Cask handling area,
(2) Canister storage feed room,
(3) Canister hoist shaft,
(4) Canister transfer galleries, and
(5) Transporter facilities.

3.3.1 Cask Handling Area

Two problems exist in the cask handling area, (1) possible high dose rates at the cask surface (200 mrem/hr of non-transferable radionuclides), and (2) possible contamination of surfaces due to failed seals, ports, etc. Therefore, care must be taken in setting up procedures for cask handling and sampling. Other than decontamination problems, good administrative control in the handling of the casks should result in acceptable dose commitments in the cask-handling area. Because of the possible high dose rates at the cask surface, administrative controls will be vital.

Contamination of a cask on external surfaces does not appear to be a problem that will be encountered frequently. However, when such an occurrence is encountered, two areas of concern are important. Because of possible high dose rates at the surface, care must be taken by personnel who contact the cask during decontamination. The second and more important concern is the possibility of airborne activity resulting from the surface contamination. Inhalation of such activity can result in high dose commitments to personnel. Good administrative procedures and engineered safeguards are required to limit the possible exposures.

3.3.2 Canister-Storage Feed Room

The canister-storage feed room must be able to handle any surge capacity due to various reasons such as the hoists being out of service for a period of time. Storage, on a temporary basis, of high-level waste canisters presents a problem due to the accumulation of these high-level radiation sources. Depending upon the design of the room, access of personnel may be impossible due to the intense radiation fields. Temporary storage, such as in the floor with cover plugs, may help solve some of the maintenance problems. The equipment associated with the feed/storage system will require occasional maintenance. In order to avoid lengthy repository shutdowns, design of the facility must allow for quick maintenance of all items either directly or remotely. It is felt that without careful design and planning the canister-storage feed room could represent a sizable part of the repository budget dose. Careful design should include consideration of the following:

(1) High-reliability equipment with simplicity of design,
(2) Provision for backup equipment,
(3) Ease of maintainability,
(4) Use of materials rated for high radiation fields, and
(5) Suitable surveillance equipment, such as windows, corner mirrors, and television cameras.

3.3.3 Canister-Hoist Shaft

Of the problem areas listed here the canister-hoist shaft would present the most severe problem since a stuck or jammed hoist with a canister somewhere in the shaft could require delicate maintenance procedures. The dose commitments required to work on and resolve such a problem could cause unacceptable dose budgets. Time requirements to resolve the problem might cause the shaft to be out of service for a long period of time, thus, shutting down the facility. Therefore, design considerations of the hoist/ shaft system should include the following:

(1) High-reliability equipment,
(2) Use of materials rated for high radiation fields,
(3) Shielding considerations for both direct exposure and streaming, such as through the hoist housing above the shaft, and
(4) Provisions to repair a hoist with a canister jammed in the confines of the shaft.

3.3.4 Canister-Transfer Galleries

The canister-transfer galleries require remote handling of casks and canisters because of the radiation fields found within the galleries. For this reason, the surveillance of the galleries is important. Operators of the galleries are expected to be in their working environments up to 40 hours per week. Therefore, the shielding of the gallery walls is crucial in the reduction of dose commitments. Figure 3-2 gives a rough estimate of the amount of dose reduction for various concrete thicknesses. Care must be used, since Figure 3-2 is only for one canister in contact with a concrete wall. Possible problems dealing with potential dose commitments for the galleries are strictly due to maintenance of the facilities. Replacement of windows and repair of remote equipment are examples. Access by maintenance personnel could be required. Therefore, good decontamination procedures are a necessity. A requirement would be the running of "clean" decontamination procedures as standard operating procedures. Thus, if personnel access is required, a minimal delay in access would be achieved. Design considerations of the galleries should include the following:

(1) High-reliability equipment,
(2) Suitable surveillance equipment, such as windows, corner mirrors, etc.,
(3) Use of materials rated for high radiation fields (such as stainless steel),
(4) Curved corners,
(5) Floor drains with slanting floors,
(6) Accessible decontamination equipment,
(7) Design to limit streaming problems through ports, etc.,
(8) Wall shields to limit operating-personnel dose commitments well under accepted guidelines, and
(9) Access of personnel to the galleries via hatches controlled by administrative procedures and designed features.

3.3.5 Transporter Facilities and Storage Rooms

Remote handling of all canisters is required in the transporter facilities in a manner similar to all other canister manipulation operations. Important considerations include keeping the dose commitments to drivers as low as is reasonably achievable. This requires both engineered design features and administrative control. Transfer of canisters from the lateral exchange room to the transporters may be a problem because of clearances of the transporter to the travel surface. Possible high dose rates may be incurred as the canister is hoisted into the transporter while it travels the gap between the surface and the transporter. Similar problems exist during and after canister placement into the storage well, but before a shield plug is added. Because of the number of operations taking place in the storage areas and exchange areas, the dose received by transporter drivers and other storage-facility personnel may be high. Careful consideration of all designs and operating procedures must be made in order that successful operation with acceptable dose commitments takes place.

3.4 ACCIDENT CONSIDERATIONS

Accidents may occur such that facility personnel are exposed to radiation unintentionally.* Accidents are broadly classified into two main categories as follows:

(1) Unintentional access to an area with a high radiation field, and

(2) Accidental release of activity, such as from a dropped canister.

In each case, facility personnel can receive unexpected dose commitments.

Accidents in which personnel enter into high level radiation fields unintentionally can generally be prevented through engineered safety features and appropriate administrative controls. In repository areas, such as the canister-storage feed room, which have personnel-access ways, interlocks must be provided along with standard operating procedures which would preclude entrance into such areas when high radiation fields are present. Potential lethal doses could be encountered, so it is imperative that careful thought be given to these particular features.

Calculation of facility personnel doses due to accidents can be calculated from postulating the location and quantity of leaking radioactive materials. Inhalation of radionuclides could result in dose commitments to various organs of the body. Air concentrations must be calculated over the time of an accident from knowledge of the repository ventilation system. Dose commitments can be calculated directly from the determination of the quantity inhaled by personnel. External exposure can be calculated from all air and surface concentrations.

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*ALARA type considerations do not apply to accidents. NRC requirements for accident conditions are contained in 10CFR100.

Cleanup or decontamination after an accident is a source for accumulation of high doses by facility personnel. Procedures for decontamination must be determined ahead of any cleanup attempts in order that the dose commitments are kept to a minimum. The health-physics staff of the repository would control the dose commitment aspects of such a cleanup operation.

3.5 SHIELDING CALCULATIONS

During the preconceptual repository design studies, a number of specific questions arose concerning the shielding requirements associated with the handling of the radioactive material being received at the repository. The calculations to resolve these questions are described in this section. In most cases, design recommendations were required in a very short time; consequently detailed calculations were not performed. In these cases, recommendations for more accurate calculations are listed.

For preliminary design purposes the following radiation levels for radiation areas were used as design criteria:

Type of Area	Exposure Time	Whole Body Dose Rate Limit
Continuous occupancy	>2 hrs. per day	<2.5 mrem/hr
Limited occupancy	<2 hrs. per day	<10.0 mrem/hr
Occasional occupancy	<0.5 hrs. per day	<40.0 mrem/hr
Non-routine occupancy	To be determined administratively	
Unrestricted area	---	As low as is reasonably achievable with a goal of less than 10 percent of the limits of 10CFR20

Shielding calculations were performed accordingly and are discussed in the following subsections. Note that as design work evolves the adequacy of these radiation limits must be verified by ALARA calculations.

3.5.1 Single PWR Spent-Fuel-Canister Dose Rates

The objective was to calculate the radiation field associated with a single canister containing 0.461 MTHM (plus cladding) of 5-year PWR spent fuel. This calculation was also performed for a canister containing 10-year PWR spent fuel. Results will be presented as dose* rates at several distances from the canister surface. These dose rates will serve as source terms for the analyses of various shield design problems.

Solution

The analysis was performed with the discrete ordinates code ANISN[1] using S_8 quadrature and P_3 22-18 group (22 neutron groups and 18 gamma-ray groups) CASK[2] cross sections. The infinite-cylinder option was used to model the cylindrical canister. This procedure is reasonably accurate near the surface but could overestimates the dose rates by factors of about 2 to 4 as the distance from the source increases. Table 3-1 presents the ANISN-calculated dose rates for both 5-year and 10-year PWR spent fuels.

Recommendations

1) The surface dose rate distribution associated with the cylindrical canister can better be described with a 2-dimensional analysis, i.e. a DOT calculation. The 1-dimensional ANISN calculation may be too conservative.
2) Consider other fuel cycles and years after emplacement.

3.5.2 Attenuation Curves for Lead and Stainless Steel, 10-Year Spent PWR Fuel

The objective was to prepare dose-rate-attenuation curves for

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*The word dose in this and all following sections should be interpreted as "dose equivalent".

various shielding materials with a canister containing 10-year-old PWR spent fuel as the radiation source.

Purpose

These curves would be used to perform preliminary hand-calculated estimates of shielding requirements. These estimates would usually be followed by machine calculations utilizing more accurate models.

Solution

Dose rates from a single canister containing 10-year PWR spent fuel were calculated at a fixed point outside various thicknesses of stainless steel (304) and lead annular shields. The spent-fuel canister is a stainless steel (304) tube with a 12.75-in. I.D. and 0.375-in.-thick walls. All dose-rate calculations used in determining the attenuation factors were performed with the discrete-ordinates code ANISN[1] using S_8 quadrature and the P_3 22-18 group CASK[2] cross-section library. The resulting attentuation curves are presented in Figures 3-3 and 3-4.

Recommendations

Attentuation factors should also be calculated for iron, concrete, and repository media such as salt, granite, and shale for 10-year PWR spent fuel. These same curves should also be generated for other waste forms which have significantly different energy spectra from that of 10-year PWR spent fuel.

3.5.3 Individual Spent-Fuel-Canister Carrier

The objective was to determine the dose rate at the surface and at 3 feet from the surface of a shielded carrier containing a single 10-year PWR spent-fuel canister. A cross-sectional view of the canister-carrier arrangement is shown in Figure 3-5. This information would assist in the design of surface handling facilities and in the determination of surface-facility operational procedures.

Solution

The radiation field and associated dose rates were previously obtained by an ANISN analysis of a canister containing 0.461 MTHM (plus cladding) of 10-year PWR spent fuel. Hand calculations were performed using broad-beam transmission curves[3] for monoenergetic gamma rays. Subsequently, an ANISN analysis of the canister-carrier arrangement was also made, thus eliminating the uncertainties associated with the use of the monoenergetic transmission curves. As in the other calculations, the ANISN calculation was done using S_8 quadrature and P_3 22-18 group CASK cross sections. The hand-calculated method gives a dose rate of 95 mrem/hr at the surface, while the computer method gives a dose rate of 28 mrem/hr at the surface and 9.5 mrem/hr at 3 feet from the surface.

Recommendation

The hand calculations were performed to satisfy an urgent request for design guidance. As would be expected, due to the use of conservative assumptions, the hand-calculated results were quite conservative when compared with ANISN calculations. Further, because of the relative simplicity of the ANISN calculations, hand calculations should be used for actual design only when absolutely necessary. Finally, a two-dimensional DOT calculation, although more costly, would provide the most realistic (least conservative) results.

3.5.4 Spent-PWR-Fuel Shielded Transporter

During the course of this study, a preconceptual design for a shielded transporter capable of carrying up to 8 spent-fuel canisters had been proposed by Parsons Brinckerhoff Quade and Douglas, Inc., (one of the participants in this effort) as a possible solution to large flow volumes and long route times for the transporters. The task here was to evaluate this design with respect to its adequacy from a radiation protection viewpoint. Since another solution was subsequently developed for the canister handling problems, this transporter concept has not been incorporated into the final preconceptual repository design.

Solution

The radiation protection provided by the materials of this carrier as shown in Figure 3-6 was calculated by considering only one canister and its immediate material environment. The material was modeled as a 3-in. stainless steel annular shield, Figure 3-7.

The dose rate on the surface of the 3-in. stainless steel shield was calculated by the discrete-ordinates code ANISN[1] using S_8 quadrature and P_3 22-18 group (22 neutron groups and 18 gamma-ray groups) CASK[2] cross sections and infinite-cylindrical geometry. The surface dose rate consisted of a 90 mrem/hr component due to neutrons[a] and a 250 rem/hr component due to gamma rays. Hand calculations were performed to estimate the actual shielding requirements for this carrier. A monoenergetic gamma-ray source was estimated from the energy flux distribution provided by the base ANISN calculation described in section 3.5.1. The attenuation provided by additional thicknesses of stainless steel was estimated from the Haywood et al.[3] attenuation curves for iron. Assuming the Department of Transportation[b] maximum allowable dose rate at three feet as the surface dose rate (200 mrem/hr), the overall shield thickness required was estimated to be 10.8 inches.

Recommendations

It should be noted that another solution to the transporter-route times problem was developed. However, if the cartridge design concept should ever be considered feasible with the 10.8-in. shield thickness estimated here, a more refined calculation would be required to determine the actual thicknesses and arrangement of materials necessary for the complete carrier. The surface dose rate used in these preliminary calculations was based on the Department of Transportation maximum-allowable surface-dose-rate for shipping containers. Future analyses should involve the use of a more realistic surface-dose rate.

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[a] The neutron component was found to be negligible in all of the shielding calculations.

[b] 49CFR173.393

3.5.5 Spent Fuel Storage Room, 10 Canisters (10-Year PWR Spent Fuel)

The shielding requirements for a surface-facility temporary storage room containing an arrangement of ten 10-year PWR spent fuel canisters was calculated. The ten canisters were assumed to have a 6-inch surface-to-surface spacing and hanging on the interior surface of a concrete wall, the thickness of which must attenuate the emergent radiation on the exterior surface to a dose rate of 0.25 mrem/hr.

Solution

A hand calculation was performed to provide a preliminary estimate of the required concrete thickness. The effective source plane was estimated by summing the contributions from each of the ten individual assemblies, taking into account the 1/r geometric attentuation. A characteristic source energy was selected from the energy flux distribution given by the ANISN analysis of a bare canister. Transmission curves[3] corresponding to this source energy for ordinary concrete were used to obtain the concrete thickness. The effect of geometric attentuation was estimated by assuming a disk source and applying configuration dimensions into a standard handbook formula. The geometric attentuation was found to be negligible.

An ANISN[1] calculation was also performed which consisted of S_8 quadrature and P_3 22-18 group (22 neutron groups and 18 gamma-ray groups) CASK[2] cross sections. Slab geometry was used and the source term was an infinite plane source having a surface intensity equal to that of the ten canisters smeared out over their projected area. The hand-calculated concrete thickness was 4.4 feet compared to 4.06 feet as calculated by the ANISN method.

Recommendations

Future design calculations, if required, should include a more accurate representation of the source canisters and other geometric considerations of the problem. This increased detail would be most effective when used in conjunction with a two-dimensional DOT calculation.

3.5.6 Shield Plug for HLW Storage in Salt

The gamma-ray attentuation properties of the shield plug designed for use in the storage of HLW canisters in holes in the floor were calculated.

Solution

The storage configuration for HLW canisters in salt is shown in Figure 3-8. The analysis of the shield plug consisted of four steps:

1) The dose rate within the 1/2-in. clearance gap (nominal gap width) was calculated with the discrete-ordinates code ANISN[1] using S_8 quadrature and the P_3 22-18 group CASK[2] cross-section library. The calculation was performed in infinite-cylindrical geometry and resulted in a dose rate within the gap of 12,000 rem/hr (when the canister is modeled as shown in Figure 3-8 including the surrounding salt).

$$D_{gap} = 12{,}000 \text{ rem/hr}$$

2) The streaming through the gap is estimated according to the single-scattering analysis performed by Haworth, et al.[4] These results are presented in graphical form and will underestimate the actual transmitted dose; however they are generally considered adequate for preliminary design usage. The ANISN-calculated dose rate of step 1 can be reduced by another factor of one-half since only one-half of the gamma rays emergent into the clearance gap are forward directed (along or up the gap) - the backward component experiences a much greater attentuation and can be neglected (contributes only about one percent). The forward component is given by

$$\begin{aligned} D_{forward} &= D_{gap} \times 1/2 \\ &= 12{,}000 \times 1/2 = 6{,}000 \text{ rem/hr.} \end{aligned}$$

By this procedure, the dose rate at the step which corresponds to the 18-in. thickness of the plug is determined to be

$$D_{step} = 220 \text{ mrem/hr}$$

3) The dose rate emergent at the top of the canister is assumed to equal the dose rate within the gap for an infinite cylinder:

$$D_{top} = D_{gap} = 12{,}000 \text{ rem/hr}$$

4) The material attenuation by the shield plug is determined by ANISN-calculated attenuation factors (slab-geometry) shown in Figure 3-2. The 18 in. of concrete at the step provides an attenuation factor of about 4×10^{-3} and results in a surface dose of 0.44 mrem/hr due to D_{step}.

$$D_{\substack{surface \\ step}} \cong 110 \times (4 \times 10^{-3}) = 0.44 \text{ mrem/hr}$$

The other component of the surface dose is due to that transmitted from the top of the canister through the full thickness of the plug. The required attenuation factor corresponding to a surface dose rate of 25 mrem/hr is

$$\frac{D_{surface}}{D_{top\ can}} = \frac{25 \text{ mrem/hr}}{12000 \times 10^{3} \text{ mrem/hr}} = 2.1\ (10)^{-6}$$

According to the attenuation curve (Figure 3-2) this would require about 3 ft. of concrete. Thus the 10-ft thickness specified in this design for the overall plug thickness is much too conservative. A shield plug having an overall thickness in the range of 3-4 ft. would provide the attenuation required to achieve a surface dose rate of 25 mrem/hr.

Recommendations

A more complete analysis will be required if this concept moves toward advanced design stages. In that event the following analyses should be performed:

1) A DOT calculation to better define the flux distribution on the canister surface. The attenuation characteristics of the shield plug could be determined by the same or an additional DOT calculation.

2) A more precise method should be applied to the streaming problem around the canister and shield plug. This would probably require a special quadrature with the DOT analysis to more accurately treat the streaming along the annular gaps.

3.5.7 Shielding Considerations During Repository Operation

During the conceptual stages of the NWTS Repository, calculations of dose rates anticipated during operation are, at best, only estimates of the actual dose rates to be encountered during operation. Areas identified as possibly having high dose rates include all areas where high-level (HLW) canisters are in close proximity to personnel. Although the baseline designs assume all HLW and spent-fuel wastes are 10 years old when they arrive at the repository, the alternative BPNL case allows younger wastes. Therefore, as a worst case (strongest source), a 5-year PWR HLW in glass with uranium and plutonium recycle is considered, and therefore, all calculations here have been done for this case. The calculations were performed with the discrete-ordinates code ANISN.[1] The group-photon spectrum per canister is shown in Table 3-2.

Figure 3-2 shows the transmittal dose rates for various thicknesses of normal-density concrete due to one canister of PWR HLW, assuming that the concrete shield begins at the canister surface and that the dose rate at the canister surface is $1.4(10)^5$ rem/hr.* As can be seen, 5 ft. of concrete reduces the dose rate to an acceptable 0.1 mrem/hr.

Figure 3-9 shows the calculated dose rates along the surface of the shielded elevator-hoist cage. The shielding consists of 2-1/4 in. of steel, plus 5 in. of lead. If three canisters are handled in such a configuration, the maximum dose rate at the surface would be about 106 mrem/hr.

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*This dose rate at the surface differs by a factor of 7 from that calculated in Section 3.5.1. However this value is a representative value and is not expected to significantly change the results.

Figure 3-10 shows the geometry for storing HLW PWR canisters in salt using 7-1/2 ft. concrete plugs. The calculated dose rate above the plug is 2.05×10^{-6} mrem/hr., which is insignificant. Since the densities of salt and concrete are nearly the same, the dose rate at the surface of the storage area with a full matrix of buried canisters should be acceptable for full-time occupancy.

3.5.8 Dose Rates and Forklift Shielding Requirements for ILW and LLW-Drum Transfer Operations

Introduction

The main objective of these analyses is to determine the shielding requirements for a forklift which would handle 55-gallon drums containing 5-year FRP[a] wet-waste (cementation products) materials during routine repository operations. Some basic shield-design data were also calculated for the handling of drums containing 10-year FRP wet waste and 10-year PWR spent-fuel materials[b] - these materials correspond to less-severe and more-severe alternative design situations respectively. The study was conducted parametrically considering surface dose rates of 1 mrem/hr, 10 mrem/hr, 0.2 rem/hr, and 1.0 rem/hr. The drums would be handled by the forklift as 1-pallet (6 drums) or 2-pallet (12 drums) arrangements. The 1-pallet and 2-pallet drum arrangements are schematically shown in Figures 3-11 and 3-12 respectively. In addition, the shielding requirements associated with stacking the pallets in their final storage location were calculated considering a 4-pallet-wide x 4-pallet-high x 2-pallet-deep arrangement as shown in Figure 3-13.

Analysis

The simple nature of the shielding problems permitted independent treatments of the geometric-attenuation and the material attenuation

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[a] The acronym FRP designates Fuel Reprocessing Plant.

[b] PWR spent-fuel materials are those materials that may have been contaminated by PWR spent-fuel wastes.

effects. Further, the drums in relation to the forklift operator's cab were considered as point sources and the dose rate due to more than one drum was obtained by superposition.

For sufficiently large distances, the geometric attentuation for any source of finite extent will closely approximate the inverse square attenuation law of the point source

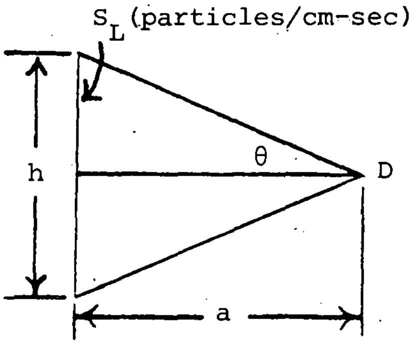
$$D(r) = D(r_0)\left(\frac{r_0}{r}\right)^2$$

The error involved in the point source approximation can be estimated from the following relation for a line source[5]

$$\frac{\phi(\text{point})}{\phi(\text{exact})} = \frac{S_L h/4\pi a^2}{S_L F(\Theta,0)/2\pi a} = \frac{\tan\Theta}{\Theta}$$

$$\tan\Theta = \frac{h/2}{a}$$

$$F(\Theta,b) = \int_0^{\Theta} e^{-b\sec\Theta'}\,d\Theta'$$



The parameter b, which is the mean-free-path thickness of intervening shields, is zero in this case and the expression for F(Θ,b) becomes

$$F(\Theta,0) = \Theta \text{ (radians).}$$

The estimates of the errors as a function of the angle Θ (one half of the subtended angle) are presented in Table 3-3. It is noted that the inverse-square law underestimates the true attentuation thus contributing to a conservative result. Further, the largest value of Θ encountered in the present work is about 15 degrees, which would produce a 2.4 percent error. In view of the preliminary nature of these calculations, it is concluded that the small and conservative errors associated with the point source approximation would be consistent with good design practice.

The material-attentuation analyses were performed with the discrete-ordinates code ANISN[1] using S_8 quadrature and P_3 22-18 group (22 neutron groups and 18 gamma-ray groups) CASK[2] cross-section library. The energy spectra of the radioactive materials contained in the drums were determined on the basis of the specified constituents of the waste materials[6] and the (11/3/76) spent fuel ORIGEN run. ANISN calculations of the dose rate at a fixed detector position were made for various thicknesses of lead shielding (0-12 cm Pb). The simple ratio of dose rates

$$\frac{D(t \text{ cm Pb})}{D(0 \text{ cm Pb})}$$

provides the material-attenuation factor for t cm of lead corresponding to a particular type of radioactive waste. Lead attenuation curves were determined for 5-year and 10-year FRP wet waste (cementation products) and 10-year PWR spent fuel.

Emergent Gamma-Ray Spectra

The spectra of the gamma rays emergent from the 55-gallon drums were determined for 5-year and 10-year FRP wet-waste (cementation products) materials. Transport calculations were performed using the discrete-ordinates code ANISN[1] and the CASK[2] cross-section library. The activities of the constituents (activation products, fission products, and actinides) of the 10-year FRP wet waste together with the ORIGEN-calculated energy spectrum of these constituents were used to establish a 10-year FRP source spectrum in terms of the ORIGEN energy-group structure. This spectrum was then transformed to the CASK group structure on the basis of simple energy-group weighting of the group activities. The 5-year FRP spectrum was obtained in a similar fashion - the activities of the 5-year constituents were first obtained by correcting the activities of the 10-year constituents for decay using ORIGEN-calculated 5-year and 10-year activities.

The spectra of the 5-year and 10-year constituents in ORIGEN energy-group format are presented in Tables 3-4 and 3-5 respectively and the CASK-format spectral data for the 5-year and 10-year constituents are presented in Table 3-6 and 3-7 respectively. Finally, the source spectra for the 5-year and 10-year FRP wet wastes are presented in Tables 3-8 and 3-9 respectively. The ANISN-calculated energy spectra of the gamma rays emergent from the 55-gallon drums are presented in Table 3-10 and also plotted in Figure 3-14 for the 5-year and 10-year FRP wet-waste (cementation products) materials and for 10-year PWR spent fuel. An examination of these spectra clearly shows that the 10-year FRP case would constitute a less severe shielding problem while the 10-year PWR case would, with its more energetic source spectrum, involve a more serious shielding problem. Thus, these two cases could provide useful off-design information.

Gamma-Ray Attenuation Curves

In order to facilitate the consideration of many design configurations and because the method of analysis permits the separate treatments of geometric and material attenuation effects, lead attenuation curves were determined for three kinds of waste materials: 5-year and 10-year FRP wet-waste (cementation products) materials and 10-year PWR spent fuel. The ANISN calculation described in the previous section was employed to determine the material-attenuation factors for various thicknesses of lead (0-15 cm). These results are presented in Table 3-11 and as attentuation curves in Figure 3-15. It is noted that the 10-year FRP wastes would require about 1 cm less lead shielding than the 5-year FRP wastes and the 10-year PWR spent fuel wastes would require about a 1 cm thicker lead shield. Also, after about the first 2 cm of lead shielding, the three attenuation curves are nearly parallel and additional thickness of lead would provide about the same attenuation regardless of the kind of waste material.

Design Calculations

The requirements for a shielded forklift based on a maximum dose rate to the operator of 0.25 mrem/hr were determined for three basic arrangements of the 55-gallon drums: 1 pallet (6 drums), 2 pallets (12 drums), and 32 pallets (4-pallets-wide x 4-pallets-high x 2-pallets-deep). A series of calculations were performed to identify the major design considerations of the shielded forklift as well as providing guidelines for the operational limits of a given design.

Operator dose rates were first calculated assuming no shielding of the cab and no self-shielding* by the drums for 1-pallet and 2-pallet arrangements. Dose rates to the operator were calculated for drums having surface activities ranging from 1 mrem/hr up to 1 rem/hr and are presented in Table 3-12. It is noted that the dose rate to the operator is less than 0.25 mrem/hr for only the lowest surface dose rate considered.

The calculations were repeated including the self-shielding effect provided by the front row of drums. These drums were represented as a homogeneous slab of concrete having a thickness equal to the drum diameter and a concrete density adjusted according to the volume change - a reduction in the concrete density by a Π/4 factor. These results are presented in Table 3-13.

The dose rates were reduced by about one third with the self-shielding included. However, again only the dose rates due to the drums having the lowest surface dose rate (1 mrem/hr) were below the desired dose rate to the operator of 0.25 mrem/hr. No attempt was made to include the effects of streaming between adjacent drums - the dose rates with streaming will be intermediate to those presented in Tables 3-12 and 3-13, probably very close to the case including self-shielding effects. Since drums having surface dose rates in excess of 1 mrem/hr must be handled, some shielding of the operator's cab will be required.

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*Self-shielding here refers to the shielding provided by the front row of drums to the radiation emitted by the back row. This effect could be important for drums which contain concrete or other relatively dense matrix material.

The forklift cab shielding requirements to reduce the operator dose rate in the operator's cab to 0.25 mrem/hr were determined in a parametric fashion by applying the gamma-ray attenuation factors shown in Figure 3-15 to the 1-pallet and 2-pallet results presented in Tables 3-12 and 3-13. The amount of shielding required to achieve a given material attenuation is dependent on the energy spectrum of the gamma rays emergent from the drums. Results are presented in Table 3-14 for the drums containing 5-year FRP wet waste - the design basis. Off-design shielding requirements are presented in Tables 3-15 and 3-16 considering drums containing 10-year FRP wet waste and 10-year PWR spent fuel respectively. An examination of these results reveals that no shielding is required for drums having surface dose rates of 1 mrem/hr. The shielding requirements increase to about 6 cm of lead for drums having surface dose rates of 1 rem/hr. Further, less than 1 cm of additional thickness of lead is required for the two pallet configurations as compared with the 1-pallet configuration. The observation can be made that a nominal 2-in. thick (about 5 cm) lead shield will provide adequate shielding for most drum-handling situations that might be encountered during presently envisioned repository operation.

The cab shielding requirements associated with the 55-gallon drums in their final storage configuration were also studied assuming a 4-pallet x 4-pallet x 2-pallet arrangement as shown in Figure 3-13. The dose rates to the operator in an unshielded cab were calculated by summing the geometrically attenuated contributions from the individual drums. Results are presented in Table 3-17 with and without self shielding by the drums. Comparisons of the unshielded dose rates to the operator associated with the final storage configuration (Table 3-17) with the dose rates produced by the 1-pallet and 2-pallet arrangements (Tables 3-12 and 3-13) would reveal that significant increases in the operator's dose rate do occur. The increase in the dose rates are about a factor of six if self-shielding by the drums is neglected and by a factor of four with self-shielding.

The shielding requirements to reduce the operator's dose rate to 0.25 mrem/hr were determined by applying the material-attenuation factors for lead (presented in Table 3-11 and Figure 3-15) to the unshielded results presented in Table 3-17. Results were obtained for the drums containing 5-year FRP wet waste and for the two off-design cases, drums containing 10-year FRP wet waste and drums containing 10-year PWR spent fuel. These required lead thicknesses are presented in Tables 3-18, 3-19 and 3-20 for the three types of waste materials - with and without self-shielding by the drums.

Finally, the off-design performance of the shielded forklift was investigated considering lead thicknesses of 2 cm and 5.3 cm and drums containing 10-year PWR spent fuel. The 2-cm thick lead shield provides adequate shielding for 2 pallets of 10 mrem/hr drums containing any of the three kinds of waste forms and the 5.3-cm thick lead shield provides the required operator protection when handling 2 pallets of 0.2-rem/hr drums. The forklift shielded with 2 cm of lead can also safely handle one 0.2-rem/hr drum containing 5-year FRP wet waste or two 0.2-rem/hr drums containing 10-year FRP wet waste. The forklift shielded with 5.3 cm of lead also provides adequate shielding for two 1-rem/hr drums containing 10-year PWR spent fuel, five 1-rem/hr drums containing 5-year FRP wet waste, or twelve 1-rem/hr drums (2 pallets) containing 10-year FRP wet waste. With some additional conservatism in the amount of lead used to shield the cab - for example, considering a total shield thickness of 3 in. (7.62 cm) of lead in the forward direction and with occasional exposure rates to the operator being permitted which are slightly in excess of 0.25 mrem/hr - a given shielded forklift should provide adequate service for a broad range of off-design conditions while remaining consistent with the safety criteria set forth in the NRC and OSHA (Occupational Safety and Health Act) regulations and ALARA (As Low As Reasonably Achievable) principle.

3.5.9 Shielding Requirements for the Thermosyphon Configuration

A shielding analysis was performed on the ventilated PWR spent-fuel canister storage configuration shown in Figure 3-16. During the preconceptual repository designs, this configuration was considered as an emplacement technique for spent-fuel-canister storage from which some of the decay energy could be removed by thermosyphon-mode of ventilation. The canister is stored in a 3/8-in. thick stainless steel liner which is convectively cooled by the upward flow of air in the inner annulus which is formed by the stainless steel liner and the concrete sleeve. The coolant air is drawn from the main storage room through the outer annulus formed by the concrete sleeve and the excavated cylindrical hole in the salt. This arrangement is expected to produce satisfactory "thermosyphon" operation (Volume 19, TM-36/19 THERMAL ANALYSIS, APPENDIX B), but does provide streaming paths for the gamma radiation emitted by the stored wastes. Unfortunately, the design requirements imposed on the configuration by heat-transfer and fluid-flow considerations are generally inconsistent with the material arrangements that would normally be employed in strictly shield design.

Methods of Analysis

The one-dimensional discrete-ordinates code ANISN[1] was used to calculate the radiation fields and associated dose rates in the inner and outer annuli of the thermosyphon configuration shown in Figure 3-16. The analysis was performed in infinite cylindrical geometry with a S_8 quadrature set and P_3 22-18 group CASK[2] cross sections. The dose rate in the inner annulus on the surface of the stainless steel liner was found to be 4140 rem/hr and the dose rate in the outer annulus on the outer surface of the concrete sleeve was found to be 120 rem/hr.

The streaming of gamma rays through the inner and outer annuli and around the stepped plug were estimated using the calculated data of R. Haworth et al.[6] based on a single-scatter treatment of the streaming component. These design data are simple to use; however, they will always underestimate the true leakage dose because the multipli-scattered component is completely neglected and a significant buildup of the scattered component may occur when a moderate-Z shielding material such as concrete is present.

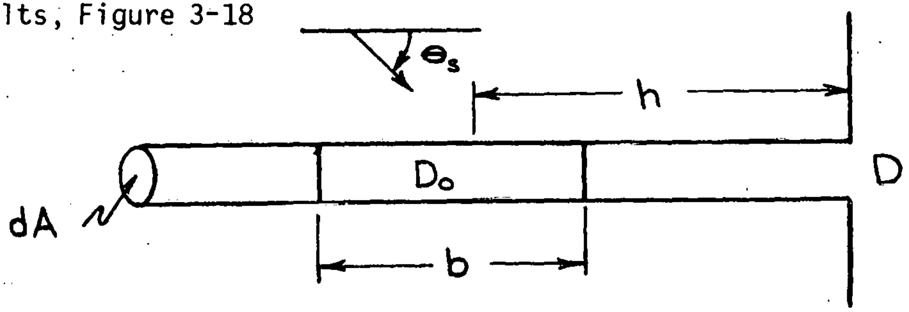
The transmission (material attenuation) of gamma rays through concrete was estimated from the correlations presented by Jones and Haywood[3] as well as through one-dimensional discrete-ordinates calculations using ANISN. Preliminary estimates of the material attenuation by concrete were made by using the Jones and Haywood correlations; however, final results were based on the ANISN calculations.

ANISN Calculated Dose Rates in the Inner and Outer Annuli

An S_8 ANISN[1] calculation using the CASK[2] cross sections was performed in infinite-cylindrical geometry. The dose rate at the stainless steel liner in the inner annulus was found to be 4140 rem/hr, and the dose rate in the outer annulus was obtained for concrete sleeve thicknesses ranging from the initial design thickness of 8 in. to 32 in. These results are presented in Figure 3-17. The dose rate in the outer annulus was found to be 120 rem/hr for an 8-inch thick concrete sleeve, 5.67 rem/hr for a 16-in. thick sleeve, and 0.0159 rem/hr for a 32-in. thick sleeve. The semi-log plot of these results (dose rate versus sleeve thickness) is essentially linear, thus permitting a simple graphical interpolation for other sleeve thicknesses.

Dose Rates at the Surface Due to Streaming through the Inner and Outer Annuli

Streaming calculations were based on R. Haworth's[4] single-scatter results, Figure 3-18



$$D/D(\text{annulus}) = C_2 b\,dA/h^3$$

where:

$D(\text{annulus}) \equiv D_i$ for the inner annulus
$D(\text{annulus}) \equiv D_o$ for the outer annulus
$b = 192$ in.
$h = 150$ in.
$dA \equiv dA_i = 260.8\ \text{in}^2$ (inner annulus)
$dA \equiv dA_o = 1988.6\ \text{in}^2$ (outer annulus)
C_2 = single-scatter coefficient, Fig. 3-18.

Assumptions:

1) ANISN-calculated dose rates (D_i and D_o corresponding to inner and outer annuli respectively) consist of a forward component (along or up the gap) D/2 and a backward component (along or down the gap) D/2.
2) The single-scatter factor C_2 was estimated from Figure 3-18. to be about 0.1 for the forward component and about 0.001 for the backward component. These values were conservative, since the constituents of concrete have atomic weights less than iron, the C_2 for the forward component was taken at a scatter angle of 0°, and the C_2 has an essentially uniform value for the backward directions.
3) Because of the low value of C_2 for the backward component, this contribution to D is neglected (about a one percent effect) and the total contribution as presented consists entirely of the forward component.

Inner Annulus Calculation

The dose rate at the storage-room floor due to gamma-ray streaming through the inner annulus is given by the following expression

$$D = \frac{D_i}{2} \ C_2 b d A_i / h^3$$

$$D = \frac{4140}{2} \times 0.10 \times 192 \times 260.8/132^3$$

$$D = 4.50 \text{ rem/hr}$$

Outer Annulus Calculation

Following the same calculational procedure used in the inner annulus calculation, the dose rate at the storage-room floor due to gamma-ray streaming through the outer annulus is given by the following expression

$$D = \frac{D_o}{2} \ C_2 b d A_o / h_3$$

Where D_o, the dose rate in the outer annulus corresponds to various concrete sleeve thicknesses.

1) 8-in. sleeve thickness

$$D = \frac{120}{2} \times 0.10 \times 192 \times 1988.6/132^3$$

$$D = 1.0 \text{ rem/hr}$$

2) 12-in. sleeve thickness

$$D = \frac{26.3}{2} \times 0.10 \times 192 \times 1988.6/132^3$$

$$D = 0.218 \text{ rem/hr}$$

3) 16-in. sleeve thickness

$$D = 0.048 \text{ rem/hr}$$

4) 32-in. sleeve thickness

$$D = 0.132 \text{ mrem/hr}$$

The results are presented in Figure 3-19 as a semi-log plot of dose rate versus concrete thickness for several depths of burial. Then, the concrete sleeve thickness required to reduce the dose rate at the storage-room floor to some specified tolerance (i.e., 0.25 mrem/hr, 25 mrem/hr etc.) can be read directly from Figure 3-19, i.e.:

$$t\ (D = 0.25 \text{ mrem/hr}) = 29.5 \text{ inches (for } h = 12 \text{ ft.)}.$$

Dose Reduction Due to Increased Depths of Burial

Consider the h=150 in. as the standard depth of burial (center of canister to surface) and define a dose reduction factor $\frac{D(h)}{D(150)}$ according to the relation

$$\frac{D(h)}{D(150)} = \frac{150^3}{h^3}$$

Figure 3-20 is a plot of the results for several depths of burial. This effect produces only a nominal reduction in dose - for example, an increased depth of burial by 12 ft. (h=299 in.) produces only about a factor of ten reduction in dose rate at the surface.

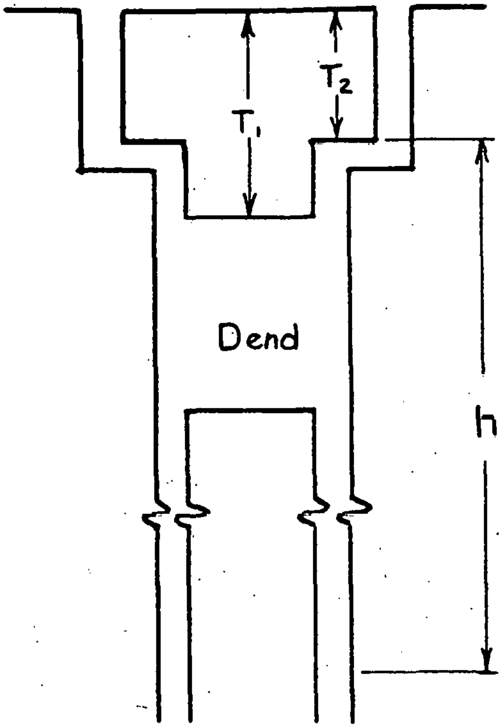
Plug Design

The overall plug design involves three phases: 1) calculation of the overall thickness, T_1; 2) calculation of the thickness at the step, T_2; 3) calculation of the dose rate due to streaming around the plug. Even though the three phases involve separate calculations, the total dose rate above the plug is affected by all three effects. Also, adjustments in design to reduce the contribution to the total dose rate from one effect may affect the contribution from another effect, thus, the final overall plug design may require iteration between the three phases.

Overall Thickness, T_1

Assume that the same dose rate exists at the end of the canister as that in the inner annulus.

$D_{end} = 4140$ rem/hr



The shield-plug thicknesses required to achieve storage-room-floor dose rates of 25 mrem/hr, 2.5 mrem/hr, and 0.25 mrem/hr were determined according to the ANISN-calculated broad-beam slab attenuation factors for concrete slabs (Figure 3-21).

1) Surface dose rate of 25 mrem/hr

$$\frac{D}{4.14(10)^6} = \frac{25}{4.14(10)^6} \cong 0.6(10)^{-5}$$

According to Figure 3-21, a plug thickness of $T_1 \cong 2.5$ ft. will provide an attenuation factor of about 10^{-5}.

2) Surface dose rate of 2.5 mrem/hr

$$\frac{D}{4.14(10^6} = \frac{2.5}{4.14(10)^6} = 0.6(10)^{-6}$$

According to Figure 3-21, a plug thickness of $T_1 \cong 3.0$ ft. will provide an attentuation factor of about 10^{-6}.

3) Surface dose rate of 0.25 mrem/hr

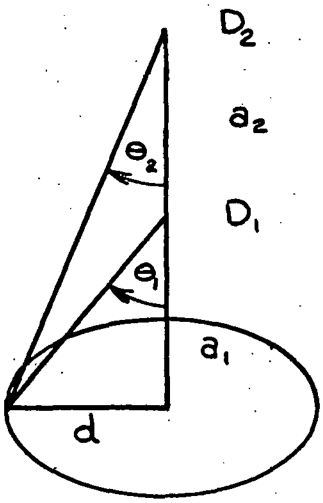
$$\frac{D}{4.14(10)^6} = \frac{0.25}{4.14(10)^6} = 0.6(10)^{-7}$$

According to Figure 3-21, a plug thickness of $T_1 \cong 3.6$ ft. will provide an attenuation factor of about 10^{-7}.

The above calculations only considered material attenuation and thus overestimate the surface dose rate. The inclusion of geometric attenuation will result in a more realistic estimate of the surface dose rate.

Geometric attenuation for disk source

The ratio of the dose rates at different distances from a disc source is given by the following equation taken from Rockwell's shielding manual[4]



$$\frac{D_2}{D_1} = \frac{\ln \sec \theta_2}{\ln \sec \theta_1}$$

where:

$$\theta_1 = \tan^{-1} \left(\frac{d}{a_1}\right)$$

$$\theta_2 = \tan^{-1} \left(\frac{d}{a_2}\right)$$

For the case of a 12-in. diameter canister and a shield-plug thickness of 2.5 ft.,

$d = 6$ in.

$a_1 = 3$ in.

$a_2 = 36$ in.

$\Theta_1 = \tan^{-1}(6/3) = 1.11$ radians

$\Theta_2 = \tan^{-1}(6/36) = 0.165$ radians,

and the geometric-attenuation factor is given by

$$\frac{D2}{D1} = \frac{\ln(\sec 0.165)}{\ln(\sec 1.11)} = \frac{0.0137}{0.810} = 0.017.$$

The total attenuation factor for the 2.5 ft. thick shield plug is given by the product of the material-attenuation and geometric-attenuation factors

$$1.1(10)^{-5} \times 1.7(10)^{-2} = 1.87(10)^{-7},$$

and the dose rate at the storage-room floor becomes

$$4140 \text{ rem/hr} \times 1.87\ (10)^{-7} = 0.77 \text{ mrem/hr}.$$

Thus, when geometric attenuation is included, a 2.5-ft. thick shield plug will result in a surface dose rate of 0.77 mrem/hr.

Plug thickness (T_2) at the step:

The dose rate at the step calculated using Haworth's equation for narrow gaps[4] (Figure 3-18) is given by

$$D(T_2) = 4140 \times \frac{1}{2} \times 0.10 \times 192 \times 260.8/114^3 = 7.0 \text{ rem/hr}$$

where the effective value of h is 114 in. If a surface dose of 2.5 mrem/hr is desired the following attentuation factor is required

$$\frac{D}{7.0} = \frac{2.5(10)^{-3}}{7.0} = 0.71(10^{-3} \simeq (10)^{-4}.$$

Then according to the transmission data of Jones,[3] Figure 3-22, a concrete thickness (T_2) at the step of 24 in. is required:

$$T_2(h=114 \text{ in}) = 24 \text{ in. } (2 \text{ ft.}).$$

The dose rate at the step for an effective value of h = 122 in. is given by

$$D(T_2) = 4140 \times \frac{1}{2} \times .01 \times 192 \times 260.8/122^3 = 5.71 \text{ rem/hr}$$

and the required attenuation factor to a surface dose of 2.5 mrem/hr is given by

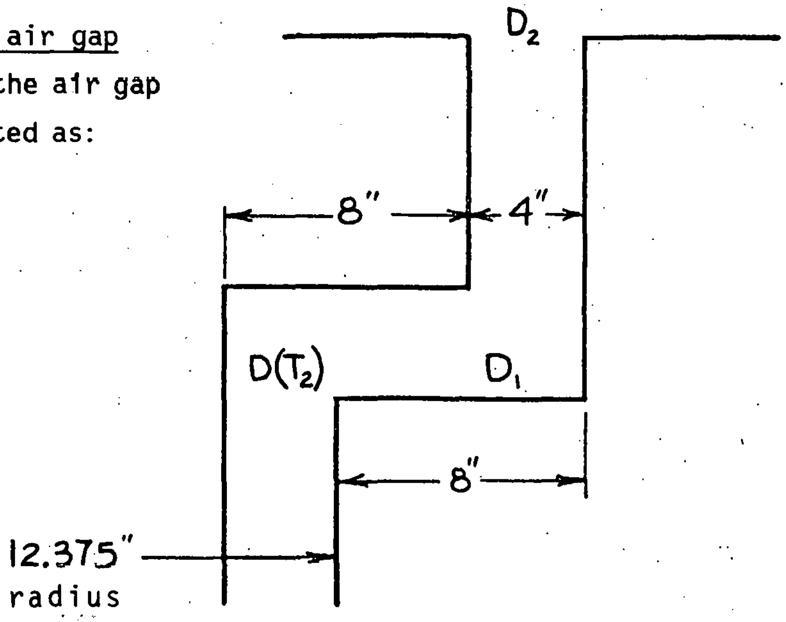
$$\frac{D}{7.0} = \frac{2.5(10)^{-3}}{7.0} = 0.71(10)^{-3} \simeq (10)^{-4}.$$

and again a concrete thickness (T_2) at the step of about 24 in. (2 ft.) is required.

Streaming through the 4-in. air gap

The dose rate $D(T_2)$ in the air gap at the step has been calculated as:

$$D(T_2) = 5.71 \text{ rem/hr}$$



There are three major contributions to the $D_1/D(T_2)$ transmission; 1) scatter from bottom of the shield plug; 2) scatter from the side of the plug; 3) scatter from a small volume inside the plug. The analysis used for each contribution is basically that given by Haworth in reference 4.

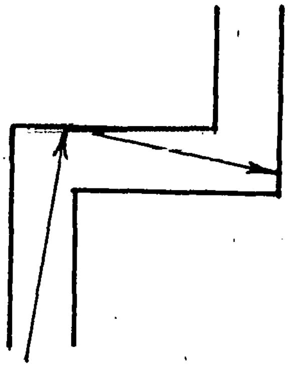
1) Scatter from bottom of step:

$$\frac{D_3^1}{D(T_2)} = C_3\ W\ dA/h^3$$

where the single-scatter factor C_3 for grazing diffuse reflection into a narrow plane gap is shown in Figure 3-23.

$$\frac{D_3^1}{D(T_2)} = 0.1 \times 4 \times \pi\ (12.375^2 - 8.375^2)/10^3$$

$$= 1.04(10)^{-2} \qquad \text{(Eq. 1)}$$

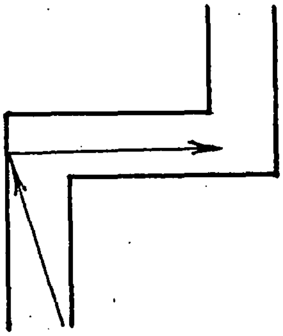


2) Scatter from side of small section:

$$\frac{D_3^2}{D(T_2)} = C_3\ W\ dA/h^3$$

$$\frac{D_3^2}{D(T_2)} = 0.01 \times 4 \times [2\ \pi\ (8.375)4]/12^3$$

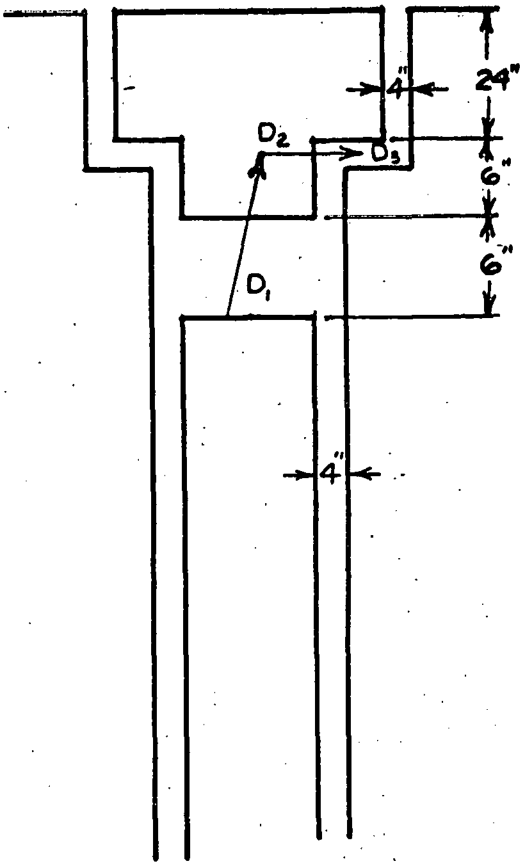
$$= 4.87(10)^{-3} \qquad \text{(Eq. 2)}$$



3) Scatter from inside small section of plug:

The dose rate (D_2) in the plug at a position near the step which is primarily due to the transmission of gamma rays associated with D_{end} = 4140 rem/hr will contribute to the dose (D_3). D_{end} experiences a material attenuation of about a factor 0.5 (Figure 3-22) due to about 4 in. of concrete (assume an $\bar{E}$ = 900 Kev and, from Table 3-21, $\mu = 0.157\ cm^{-1}$) so that

$$D_2 = 4140 \times 0.5 = 2070 \text{ rem/hr}$$



The contribution from D_2 to D_3 (as schematically shown in the figure) is given by the Haworth[4] relationship (Figure 3-24)

$$D_3^3 = C_1 \, dA \, D_2/h^2$$

where

$C_1 = 0.02$ (average scattering angle of about 90°
$h = 12$ in.
$dA = 2\pi(12.375)(4) = 311 \text{ in}^2$

Then

$$D_3^3 = 0.02 \times 311 \times 2070/12^2 = 89 \text{ rem/hr}$$

Finally the total contribution to D_3 is obtained by summing the three contributions

$$D_3 = 89 + 5.71(4.87\text{x}10^{-3} + 4.02\text{x}10^{-2}) \cong 89 \text{ rem/hr}$$

The above analysis does not take into account geometric attenuation due to increasing cross sectional area.

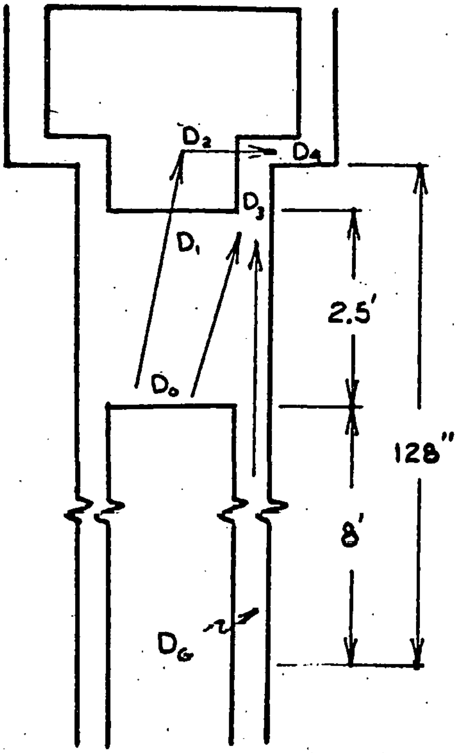
Sinking canister an additional 2 ft

The dose rate may be further reduced by sinking the canister to take advantage of geometric attenuation.

$D_1 = 1.7(10^{-2})\ 4140$ rem/hr $= 70.38$ rem/hr
Geometric attenuation

$D_2 = 70.38\ (0.5) = 35.2$ rem/hr
Concrete attenuation

D_4^3 = contribution to dose rate 4 from contributing pathway 3, previous page.



$D_4^3 = 2(10^{-2})(311/144)\ 35.2 =$
1520 mrem/hr (Eq. 3)

The D_3 has contributions from the end of the canister in addition to that streaming up the gap. Further, assume contributions from the end of the canister to be same as

D_1 = 70.4 rem/hr.

Contribution from streaming:

The contribution to the dose rate D due to streaming is denoted by $D^{\S}$ and is calculated according to the Haworth4 relationship

$$D_3^S = D_G C_2 bdA/h^3$$

where

C_2 = 0.1
b = 192 in.
dA = 260.8 in^2
h = 128 in.

$$D_G = \frac{4140}{2} - 2070 \text{ rem/hr}$$

$$D_3^S = 2070 \times 0.1 \times 192 \times 260.8/128^3 = 4.94 \text{ rem/hr}$$

Total D_3:

$$D_3 = D_1 + D_3^S$$
$$= 70.4 + 4.94 = 75.4 \text{ rem/hr}$$

D_4 has four contributors (assuming no direct contribution) 1) scatter from inside small section of plug, 2) entering side of duct, 3) scatter from side of small section of plug, and 4) scatter from bottom of large section of plug.

1) Scatter from inside small section of plug:

The contribution to D_4 due to scatter from inside the small section of the plug was previously calculated to be 1520 mrem/hr (see Eq. 3).

2) Contribution through side of duct:

This contribution was calculated using Haworth's method[4] (Figure 3-18).

$$\frac{D_4}{D_3} = C_2 \frac{b\ dA}{h^3}$$

b = 8 in. and the dose rate D_3 was assumed to be uniformly distributed along b.

An average cross sectional area was used for dA.

$$dA = d\bar{A} = 2\pi\bar{r}(4\ \text{in.})$$

$$\bar{r} = 12.375''$$

$$d\bar{A} = 2\pi(12.375)(4) = 311\ \text{in.}^2$$

The fraction of D_3 which contributes to D_4 is then

$$\frac{D_4}{D_3} = \frac{0.1(8)(311)}{8} = 0.49$$

3) Scatter from side of small section of plug:

The $\frac{D_4}{D_3}$ ratio for this pathway was previously calculated to be $4.87(10)^{-3}$ (see Eq. 2).

4) Scatter from bottom of large section of plug:

As in 3) above this ratio was previously calculated to be $1.04(10)^{-2}$ (see Eq. 1).

The total fraction of D_3 contributing to D_4 by pathways 2, 3, and 4 is therefore,

$$\frac{D_4}{D_3} = 0.49 + 4.87(10)^{-3} + 1.04(10)^{-2} \simeq 0.50$$

Dose rate D_4 is then given by

$$D_4 = 0.5D_3 + 1520 \text{ mrem/hr}$$

$$D_4 = 37.7 \text{ rem/hr} + 1.52 \text{ rem/hr}$$

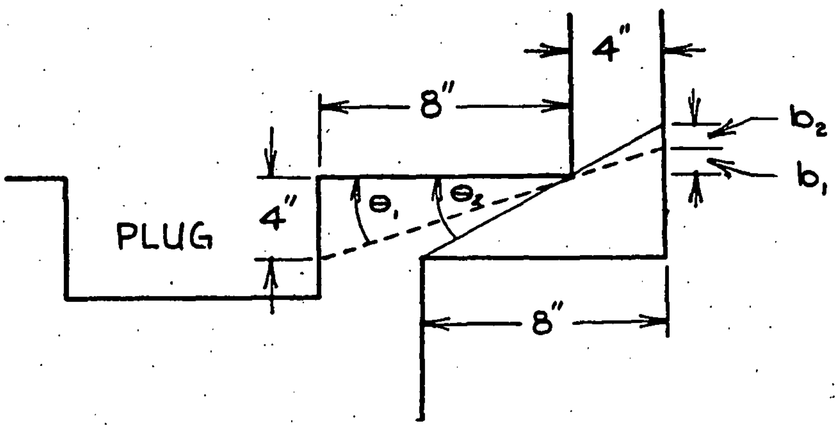
$$D_4 = 39.2 \text{ rem/hr} \qquad \text{(Eq. 4)}$$

This example thus demonstrates that sinking the canister an additional 2 feet results in a reduction of the dose rate at the second bend by approximately 0.44.

Streaming up the vertical gap

As in the previous sections, this analysis was performed by the method described by Haworth.[4] There are three major pathways which contributed to the dose rate, D, above the vertical gap: 1) through the side of the duct, D(side); 2) scatter from the end of the horizontal duct, D(end); 3) scatter from the bottom of the vertical duct, D(bottom).

1) Contribution through the side of the duct:



From the preceding D_4 calculation the only pathways contributing to D(side) are 1), 2), and 3). D(side) may then be written as:

$$D(\text{side}) = \sum_{i=1}^{3} C_2 \frac{b_i \, dA}{h^3} D_4^i$$

where,

D_4^i is the contribution to D_4 by pathway i
b_i is b for the i^{th} pathway

Since C_2, dA, and h are independent of the path D(side) becomes

$$D(\text{side}) = \frac{C_2 dA}{h^3} \sum_{i=1}^{3} b_i \, D_4^i$$

For i = 1 and i = 3 b_i is a maximum value and is equal to b_1.
For i = 2 b_i has an average value b_2.

This method accounts for scatter off both sides of the duct. Therefore not accounting for attenuation through the corner on the left-hand side of the duct results in a conservative estimate.

From the previous section:

$D_4^1 = 1520$ mrem/hr

$D_4^2 = 36.9$ rem/hr

$D_4^3 = 75.4$ mrem/hr

The expression for D(side) can be written as

$$D(\text{side}) = C_3\, dA\ \frac{b_1}{h_1^3}\ (1520 + 75.4)\ \text{mrem/hr} + \frac{b_2}{h_2^3}\ (36.9\ \text{rem/hr})$$

where

$$\frac{b_1}{4\ \text{in.}} = \frac{4\ \text{in.}}{8\ \text{in.}} \qquad \text{thus} \qquad b_1 = 2\ \text{in.}$$

$$\frac{b_2}{4\ \text{in.}} = \frac{4\ \text{in.}}{4\ \text{in.}} \qquad \text{thus} \qquad b_2 = 4\ \text{in.}$$

$C_3 = 0.1$ (Fig. 3-18)

$h_1 = 23$ in. $\qquad h_2 = 22$ in.

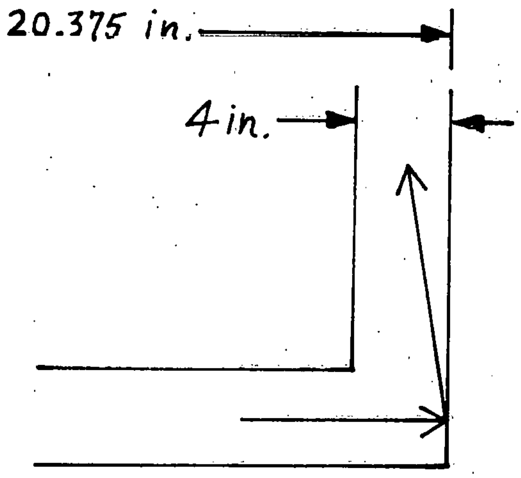
$dA = \pi(20.375^2 16.375^2) = 462\ \text{in}^2$

The dose rate D(side) is then given by

$$D(\text{side}) = 4.62 \left[\frac{2}{23^3}\ (1.60) + \frac{4}{22^3}\ (36.9)\right]\ \text{rem/hr}$$

$$D(\text{side}) = 0.653\ \text{rem/hr}$$

2) Scatter from end of horizontal duct:



The ratio $\frac{D(end)}{D^4}$ was determined by the relation

$$\frac{D(end)}{D_4} = C_3 \, W \, \frac{dA}{h^3},$$

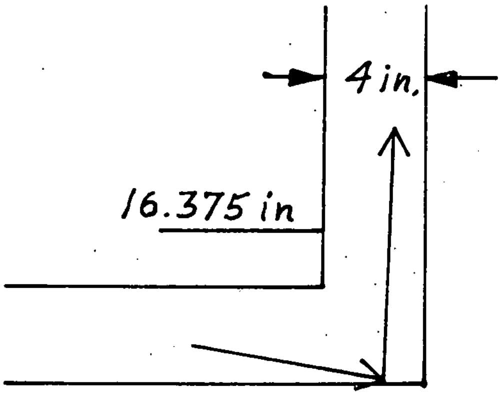
where

C_3 = single-scatter factor, Fig. 3-23
W = gap width
dA = duct surface area
$C_3 = 0.01$ $W = 4$ in. $h = 26$ in.
$r = 20.375$ in. (conservative value)

then $$\frac{D(end)}{D_4} = 0.01\,(4)\,\Pi\,(20.375)(2)(4)/26^3$$

$$\frac{D(end)}{D_4} = 0.0012$$

3) Scatter from bottom of vertical duct:



As in Eq. 2) the ratio $\frac{D(bottom)}{D_4}$ was determined by the relation

$$\frac{D(bottom)}{D_4} = C_3 \quad W \, \frac{dA}{h^3}$$

Where dA is now the cross-sectional area of the vertical duct and

$C_3 \doteq 10^{-3}$ $W = 4$ in. $h - 28$ in.

Then

$$\frac{D(\text{bottom})}{D_4} = 10^{-3}(4)\ \pi\ [(20.375)^2 - (16.375)^2]/28^3$$

$$\frac{D(\text{bottom})}{D_4} = 8.4 \times 10^{-5}$$

The dose rate, D, above the vertical gap is given by:

$D = D(\text{side} + D(\text{end}) + D(\text{bottom})$

or

$D = 0.653$ rem/hr $+ D_4(0.0012 + 8.4 \times 10^{-5})$

From Eq. 4) $D_4 = 39.2$ rem/hr

therefore,

$D = 0.653 + 1.28\text{x}10^{-4}(39.2)$ rem/hr

$D = 682$ mrem/hr

Figure 3-25 presents the dose rate, D, as a function of canister depth.

Conclusions on Thermosyphon and Plug Design

The streaming of gamma-ray radiation along the annuli constitutes a serious shielding problem. It was shown here that a stepped plug 2.5-ft. thick will attenuate the gamma leakage through the inner annulus to levels at the surface of 50 mrem/hr. When the distance between the canister top and plug bottom is approximately 8.5 ft., a concrete sleeve thickness of about 15 inches is required to achieve comparable dose levels at the surface due to gamma rays streaming through the outer annulus (Figure 3-26).

The concrete thickness of 8 inches results in an unacceptable dose rate at the surface of about 197 mrem/hr. For allowable dose rates above the inner annulus significantly less than 50 mrem/hr, the plug design will have to be altered rather than sinking the canister further.

These calculations are not conservative in the tradional sense and the shielding problems are probably more severe than these calculations indicate.

3. REFERENCES

1. Engle, Ward W. Jr., "ANISN - One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693, ORNL (1967).

2. "CASK-40 Group Coupled Neutron and Gamma-Ray Cross Section Data" (1973); available as DLC-23/CASK from RSIC, ORNL.

3. Jones, T.D., and Haywood, F.F., "Transmission of Photons through Common Shielding Media, "ORNL-TM-4728.

4. Haworth, R., Liquorish, A.C., and Taylor, C.B., "Radiation Leakage from Narrow Gaps in Shields," Nuclear Engineering and Design 32 (1975) 428-437.

5. Rockwell, T., "Reactor Shielding Design Manual," D. Van Nostrand Publishing Co., Inc., 1956.

6. Battelle - September 28, 1977 letter to C. D. Zerby.

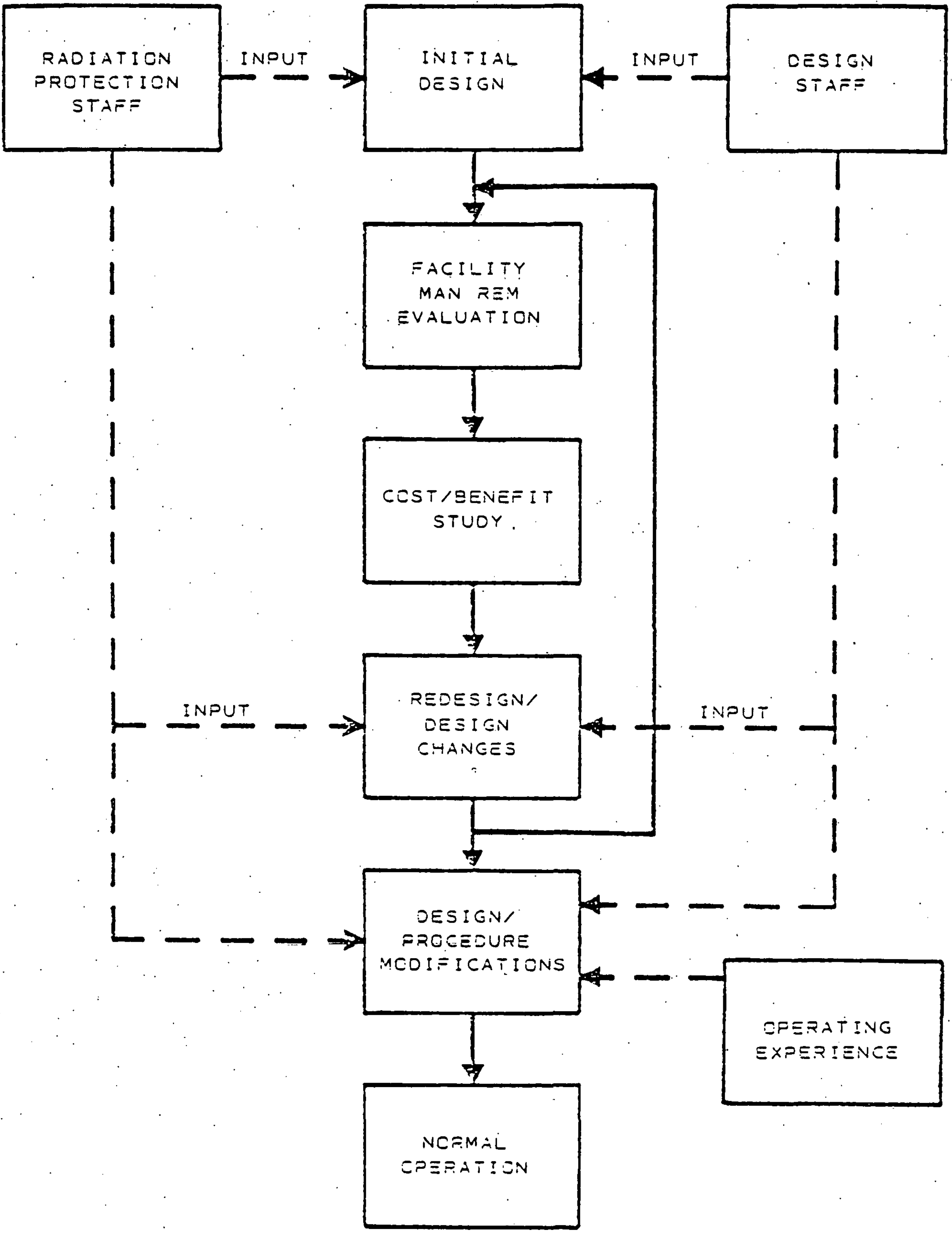


Figure 3-1 RADIATION PROTECTION DESIGN PHILOSOPHY FOR NWTS REPOSITORY

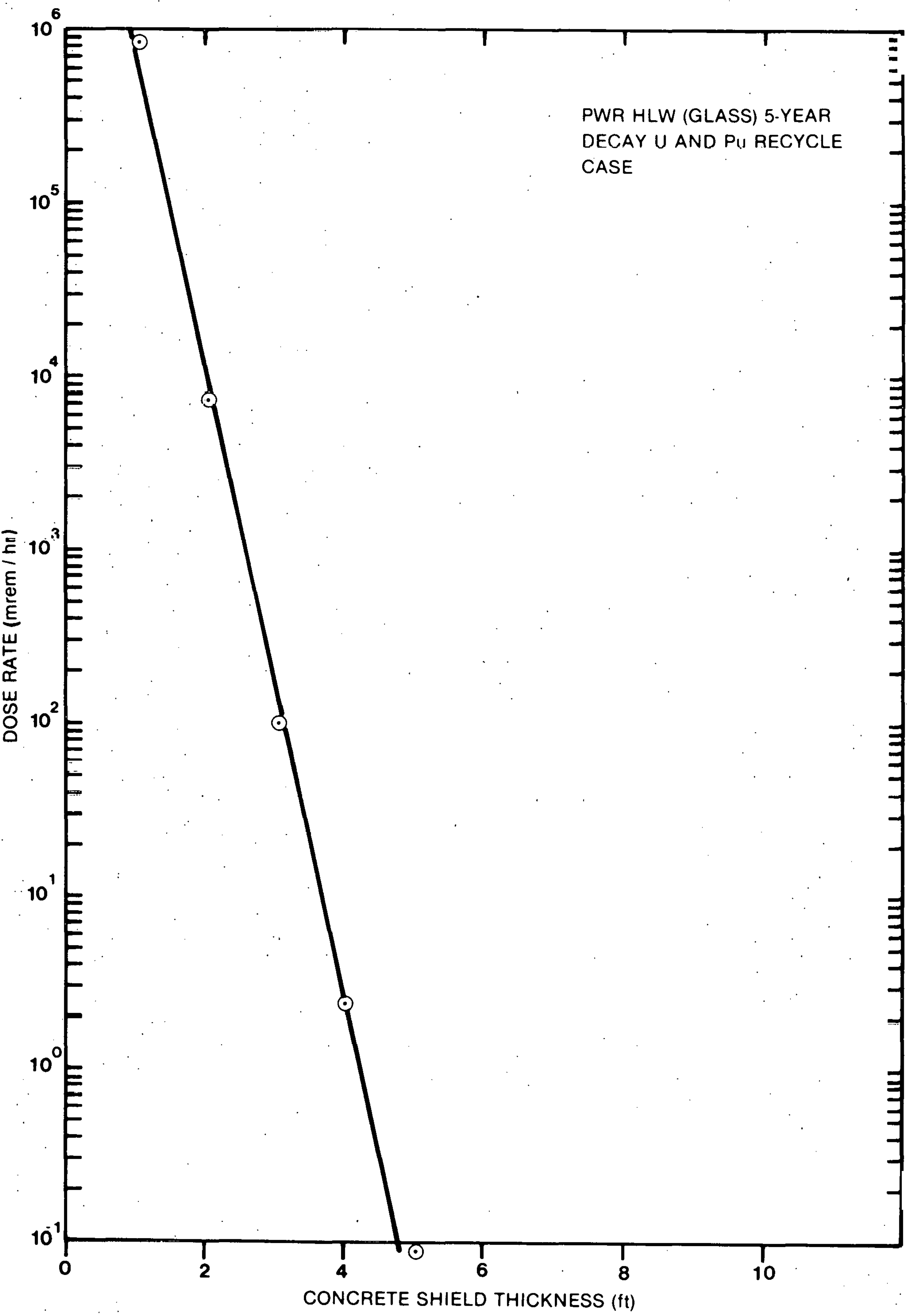


Figure 3-2 ATTENUATION CURVE FOR CONCRETE

PA1089-17-1

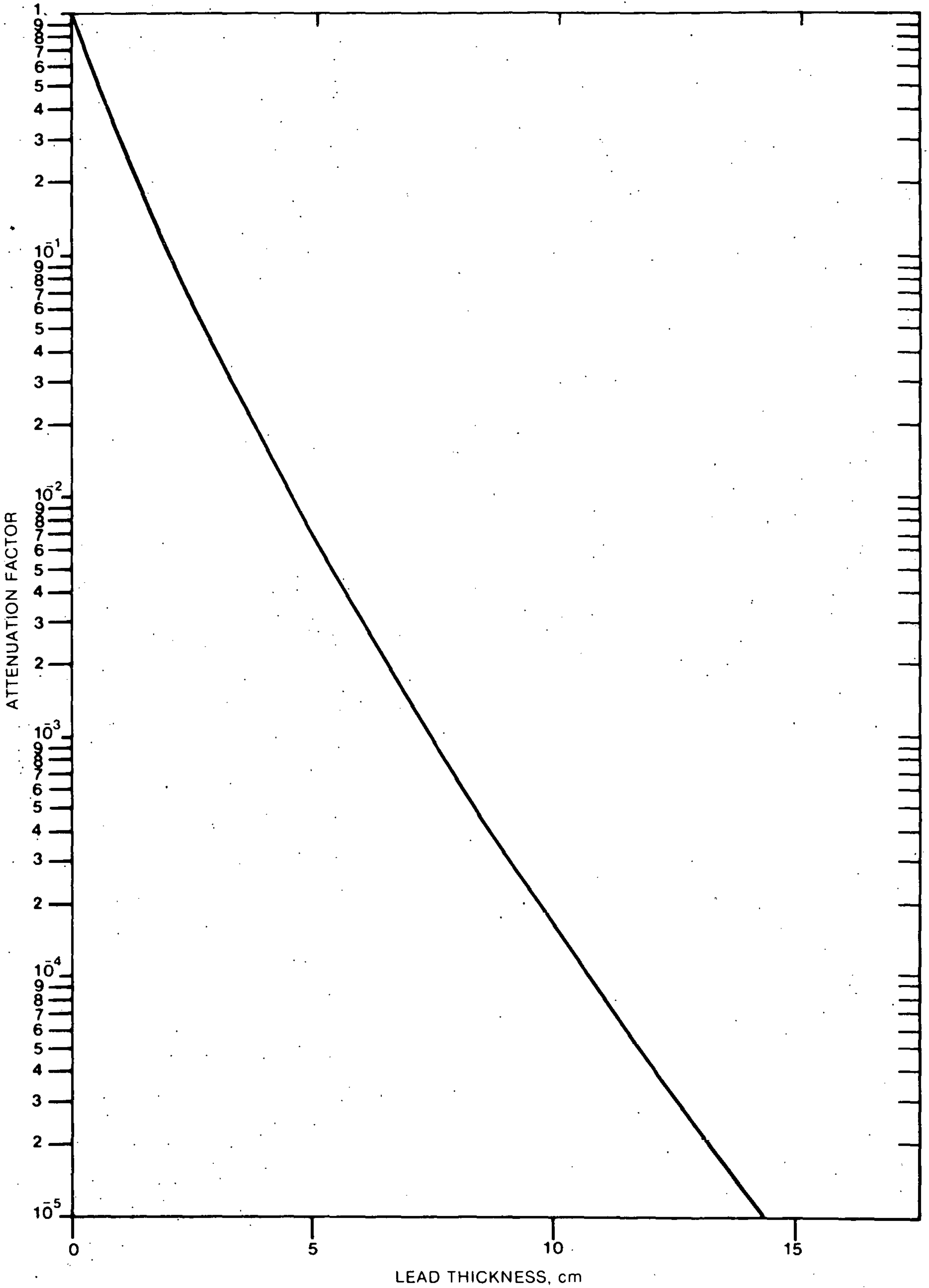


Figure 3-3 LEAD ATTENUATION CURVE FOR 10 YR PWR SPENT-FUEL GAMMA RAYS

PA1089-13-1

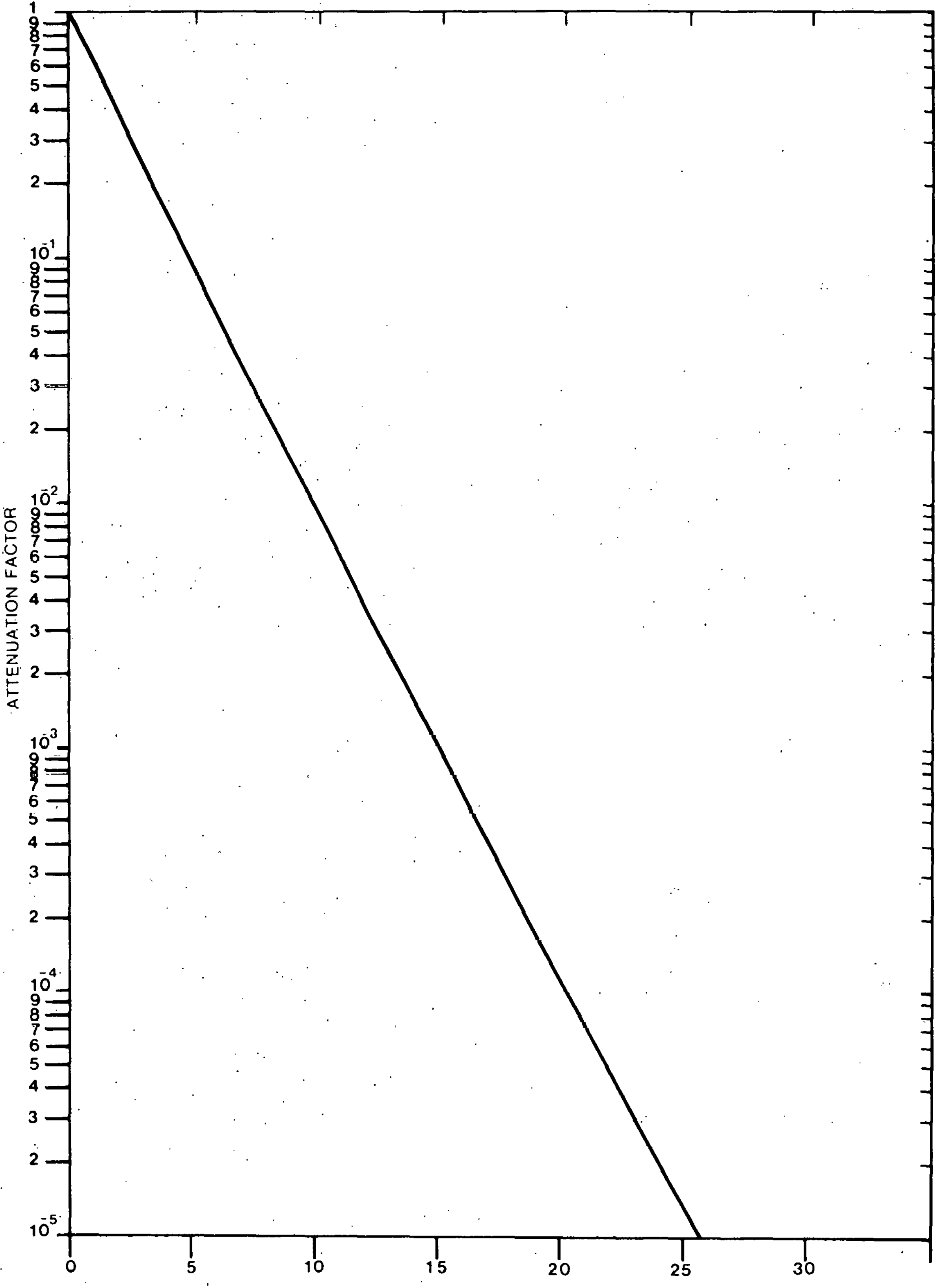


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Figure 3-4 STAINLESS STEEL (304) ATTENUATION CURVE FOR 10 YR PWR SPENT-FUEL GAMMA RAYS

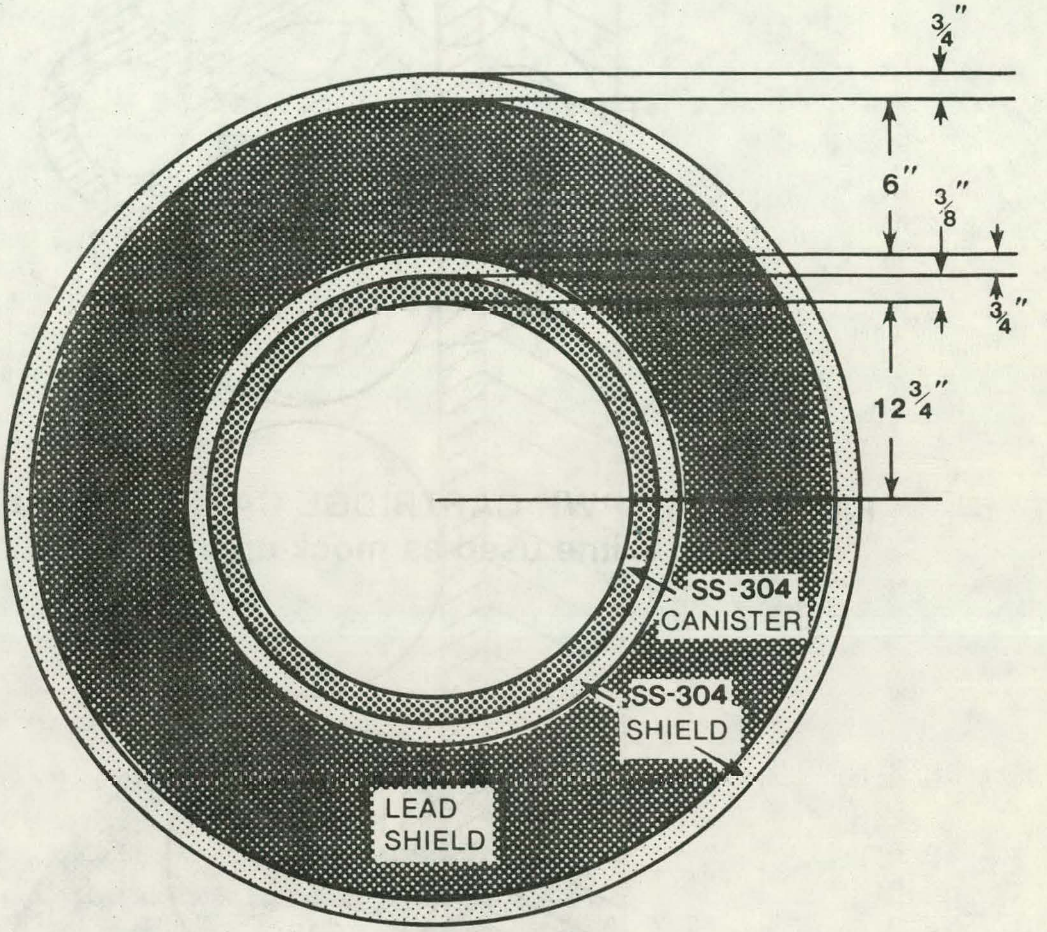

Figure 3-5 INDIVIDUAL SPENT-FUEL CANISTER CARRIER

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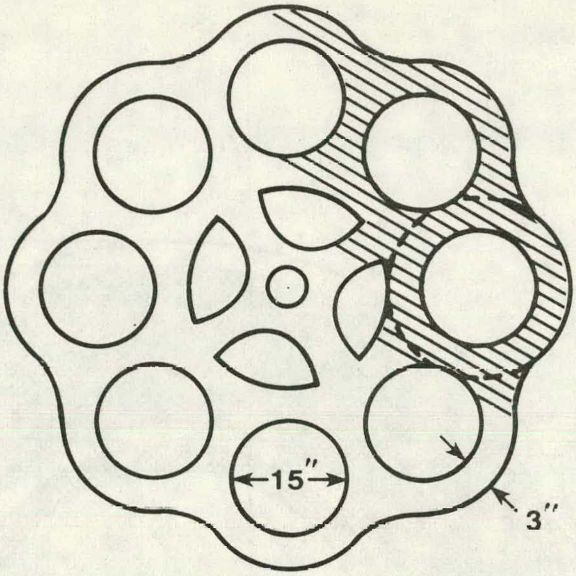

Figure 3-6 PWR CARTRIDGE CARRIER (area within dotted line used as mock-up in ANISN)

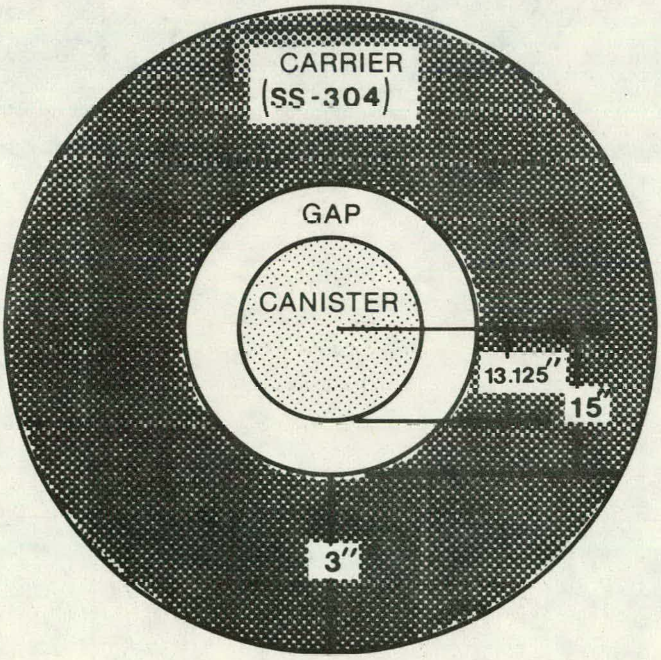

Figure 3-7 MOCK-UP OF PWR CARTRIDGE CARRIER AS USED IN ANISN

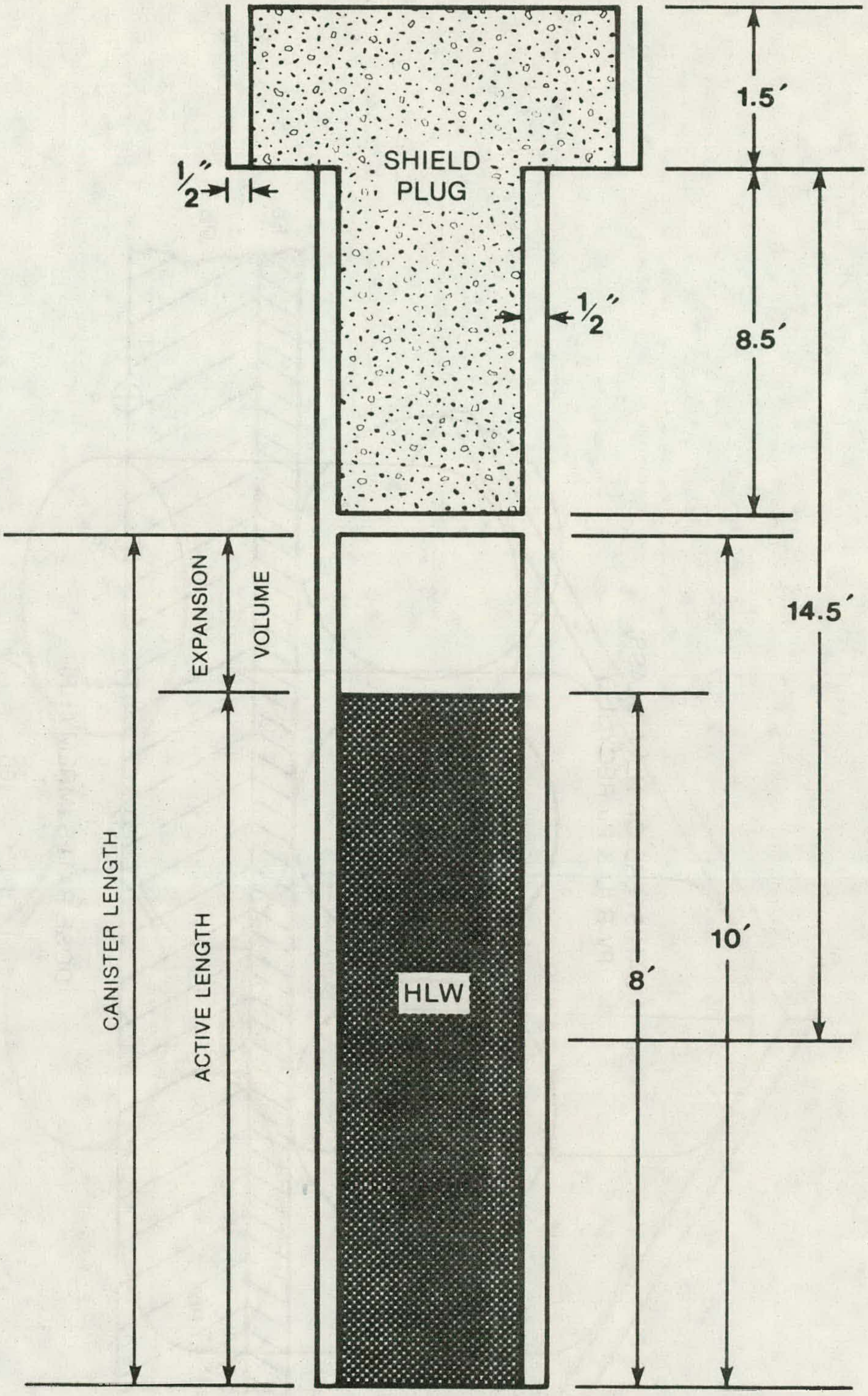

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Figure 3-8 STORAGE CONFIGURATION FOR HLW CANISTER IN SALT

PA1089-18-1

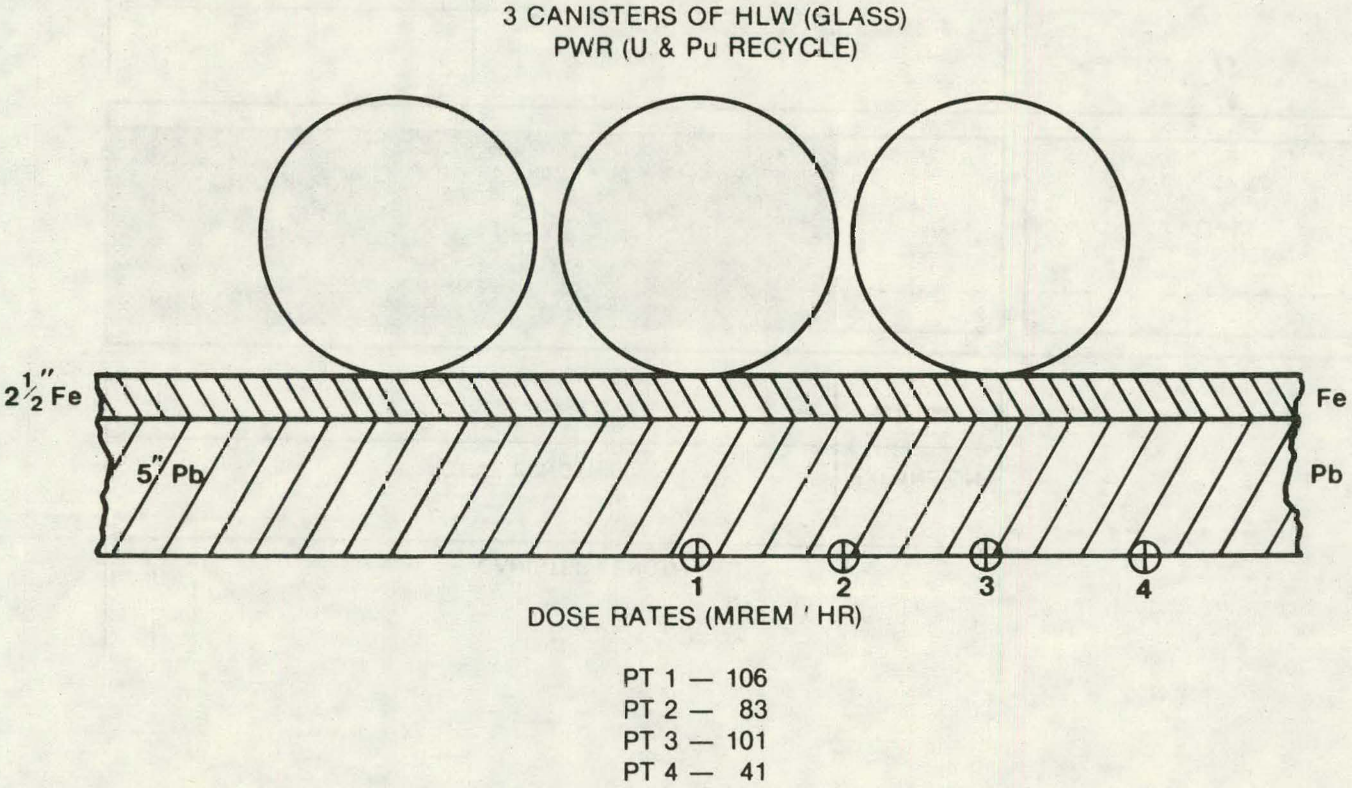

Figure 3-9 WALL SECTION, SHIELDED ELEVATOR

PA1089-20-1

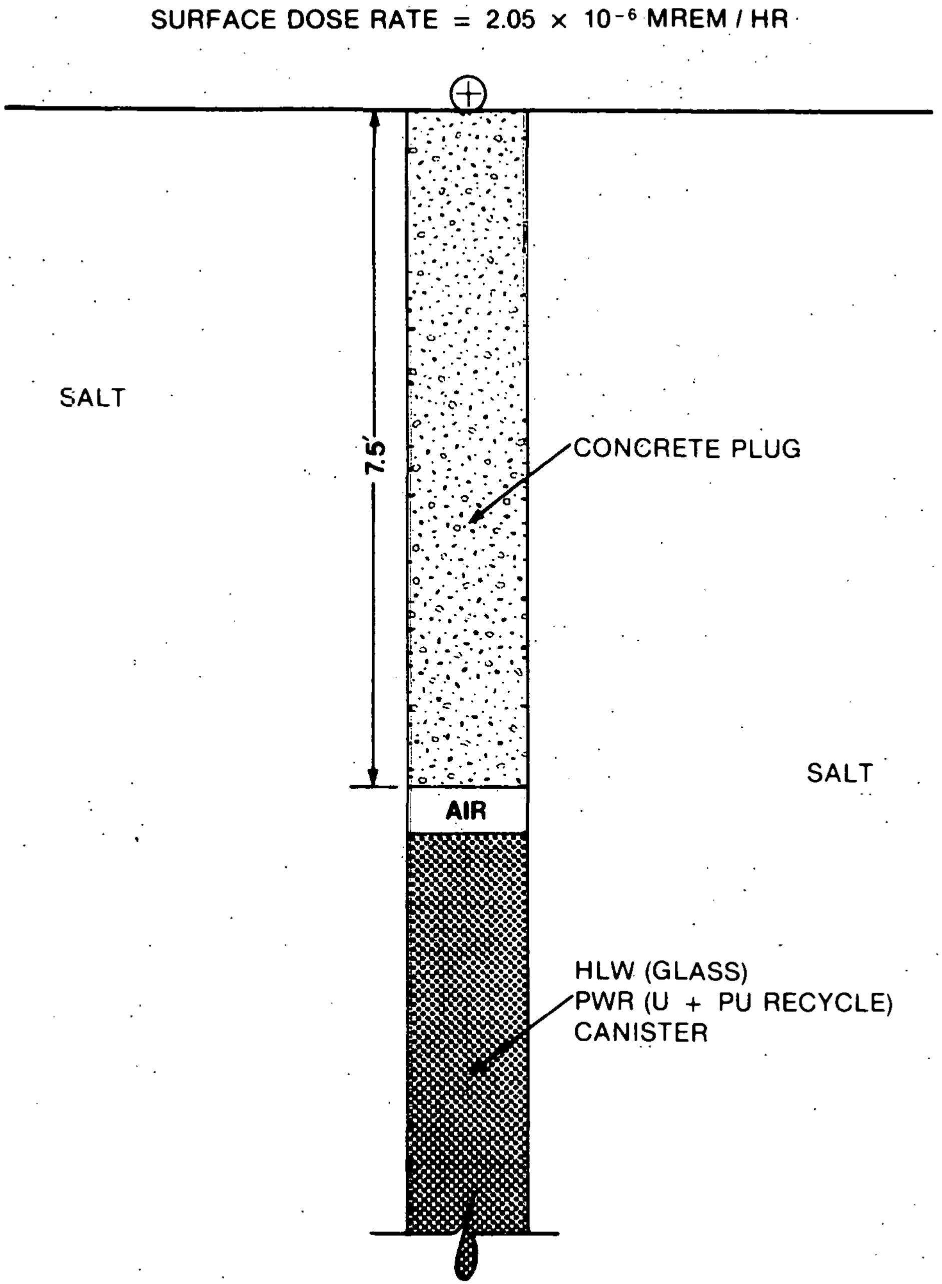

Figure 3-10 MOCK-UP OF HLW CANISTER IN SALT WITH 7.5′ CONCRETE PLUG

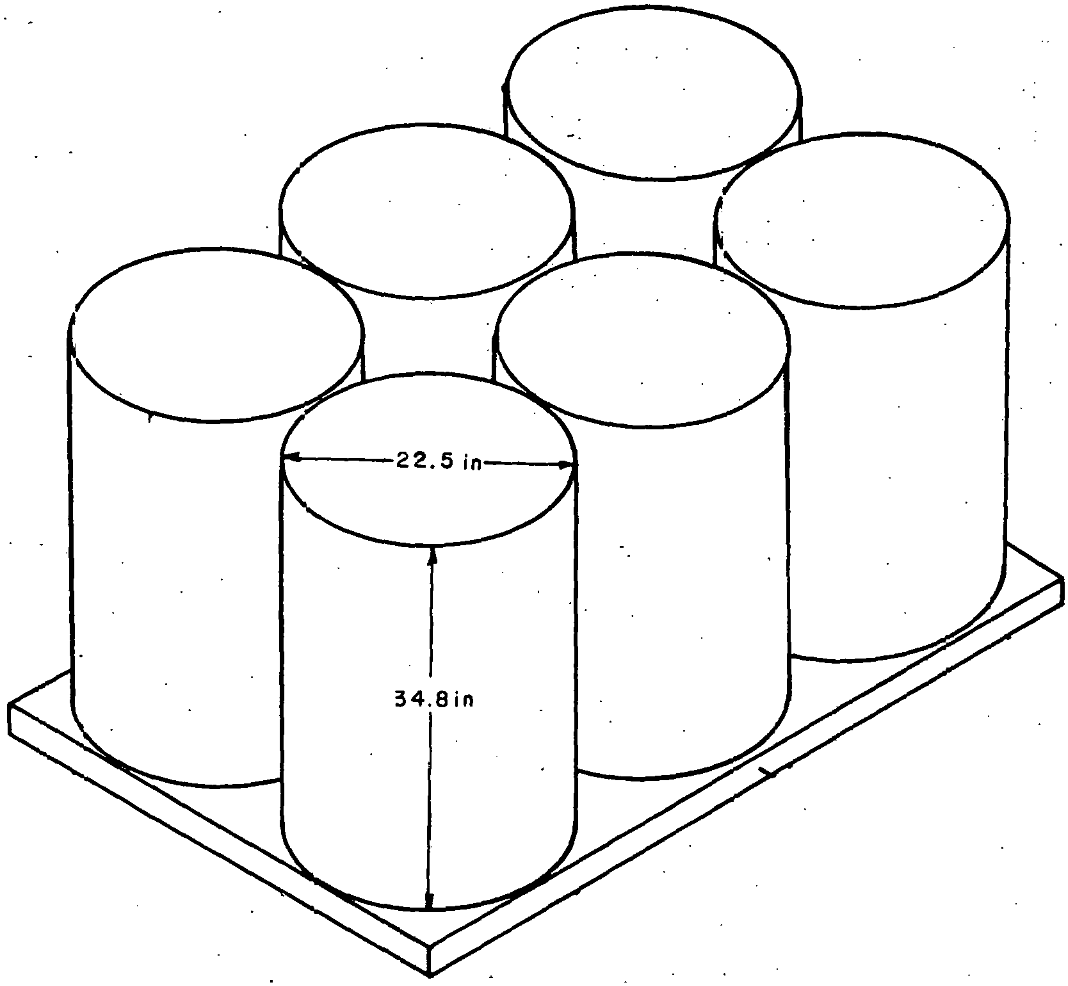

FIGURE 3-11 SINGLE PALLET CONFIGURATION

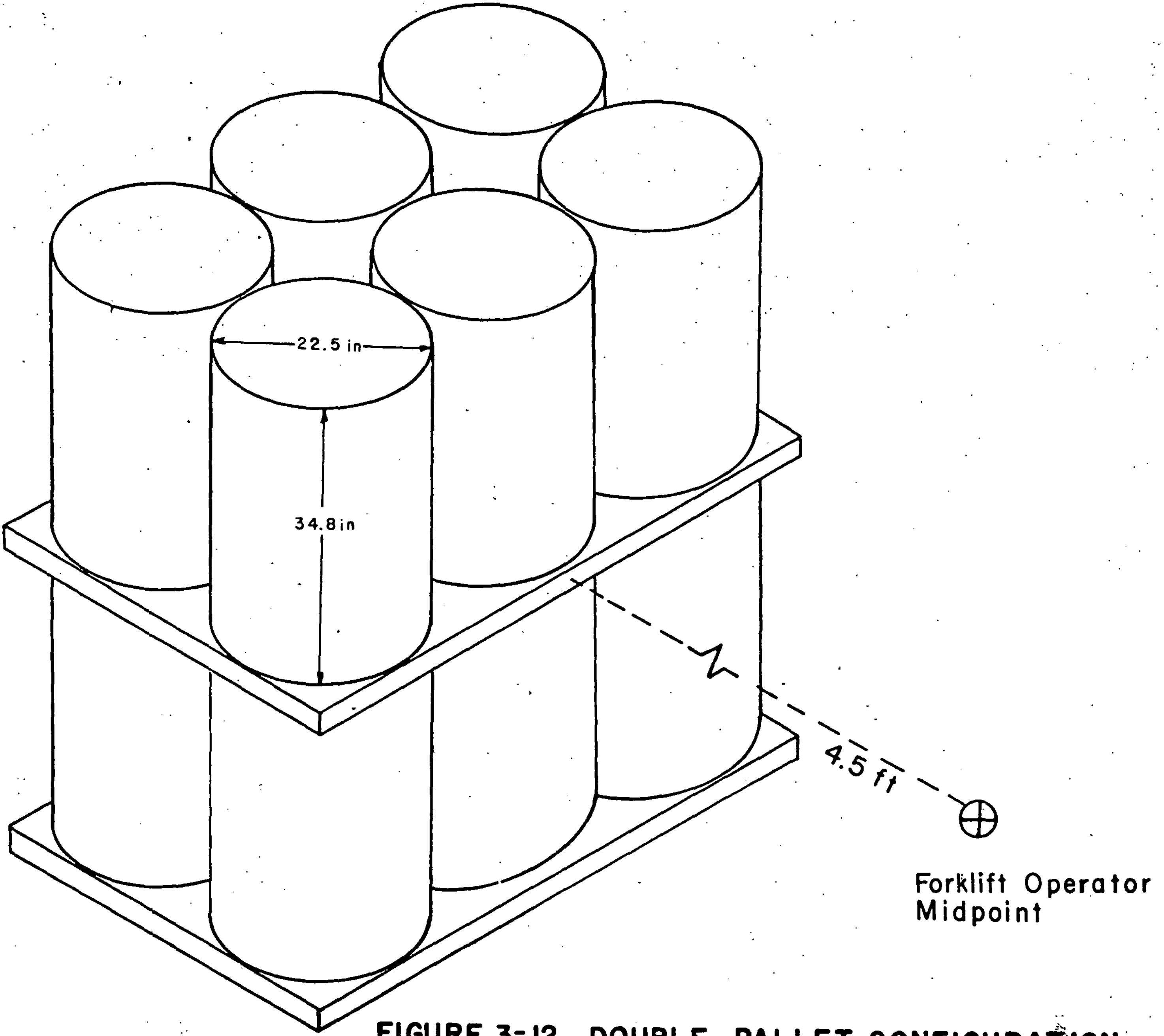

FIGURE 3-12 DOUBLE PALLET CONFIGURATION

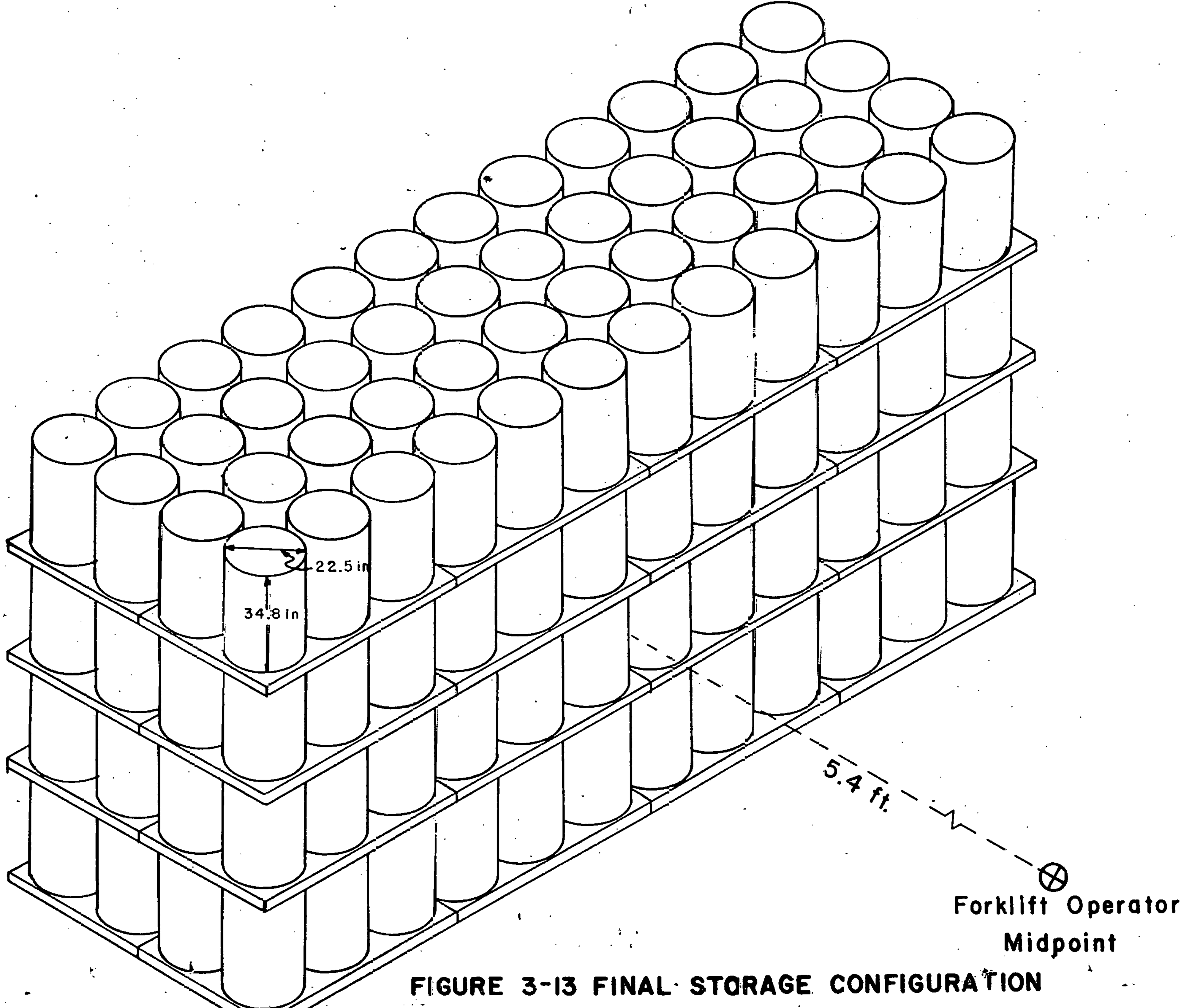

FIGURE 3-13 FINAL STORAGE CONFIGURATION

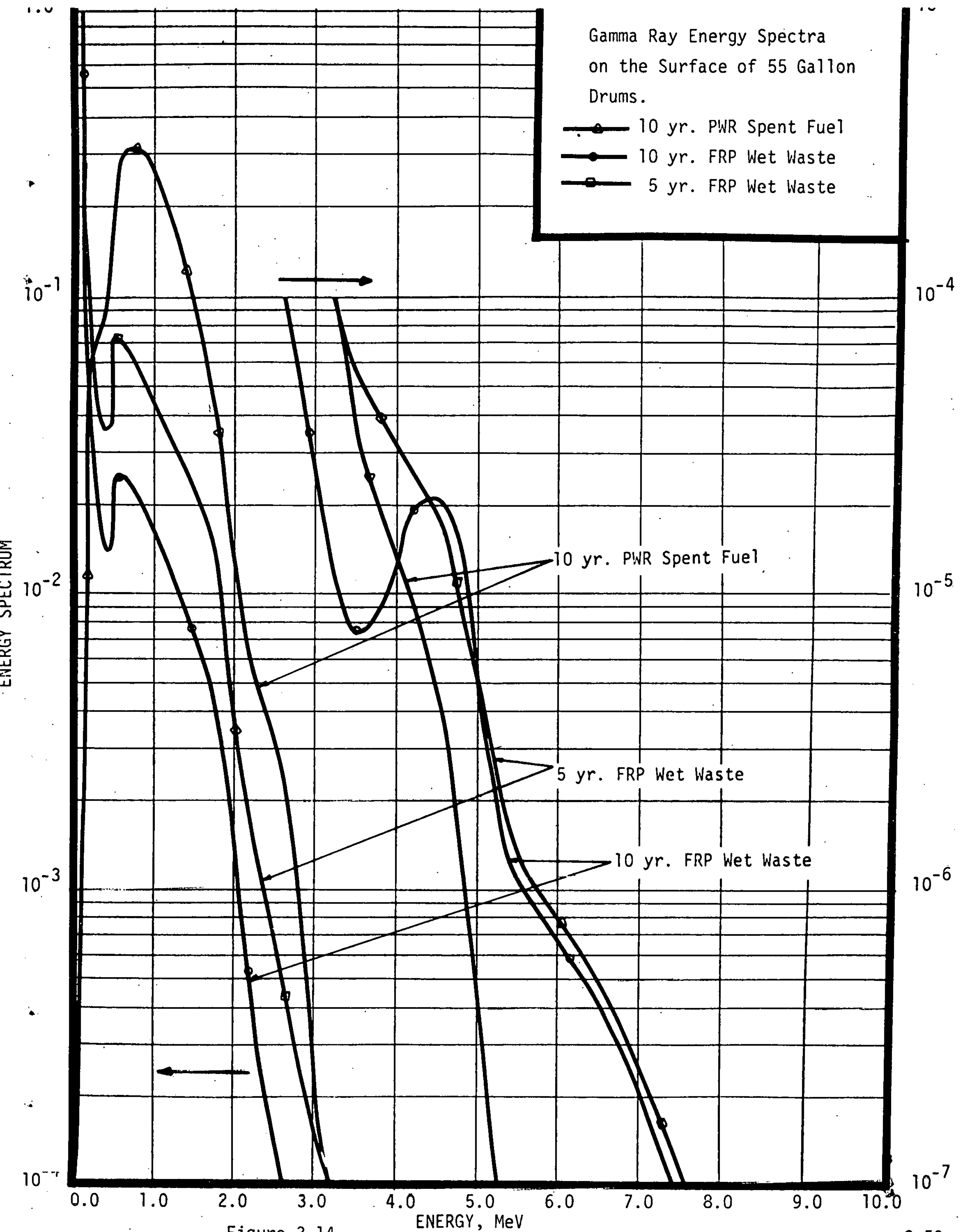
Gamma Ray Energy Spectra on the Surface of 55 Gallon Drums.
10 yr. PWR Spent Fuel
10 yr. FRP Wet Waste
5 yr. FRP Wet Waste
10 yr. PWR Spent Fuel
5 yr. FRP Wet Waste
10 yr. FRP Wet Waste
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
0.0
1.0
2.0
3.0
4.0
5.0
6.0
7.0
8.0
9.0
10.0
ENERGY, MeV
ENERGY SPECTRUM

Figure 3-14

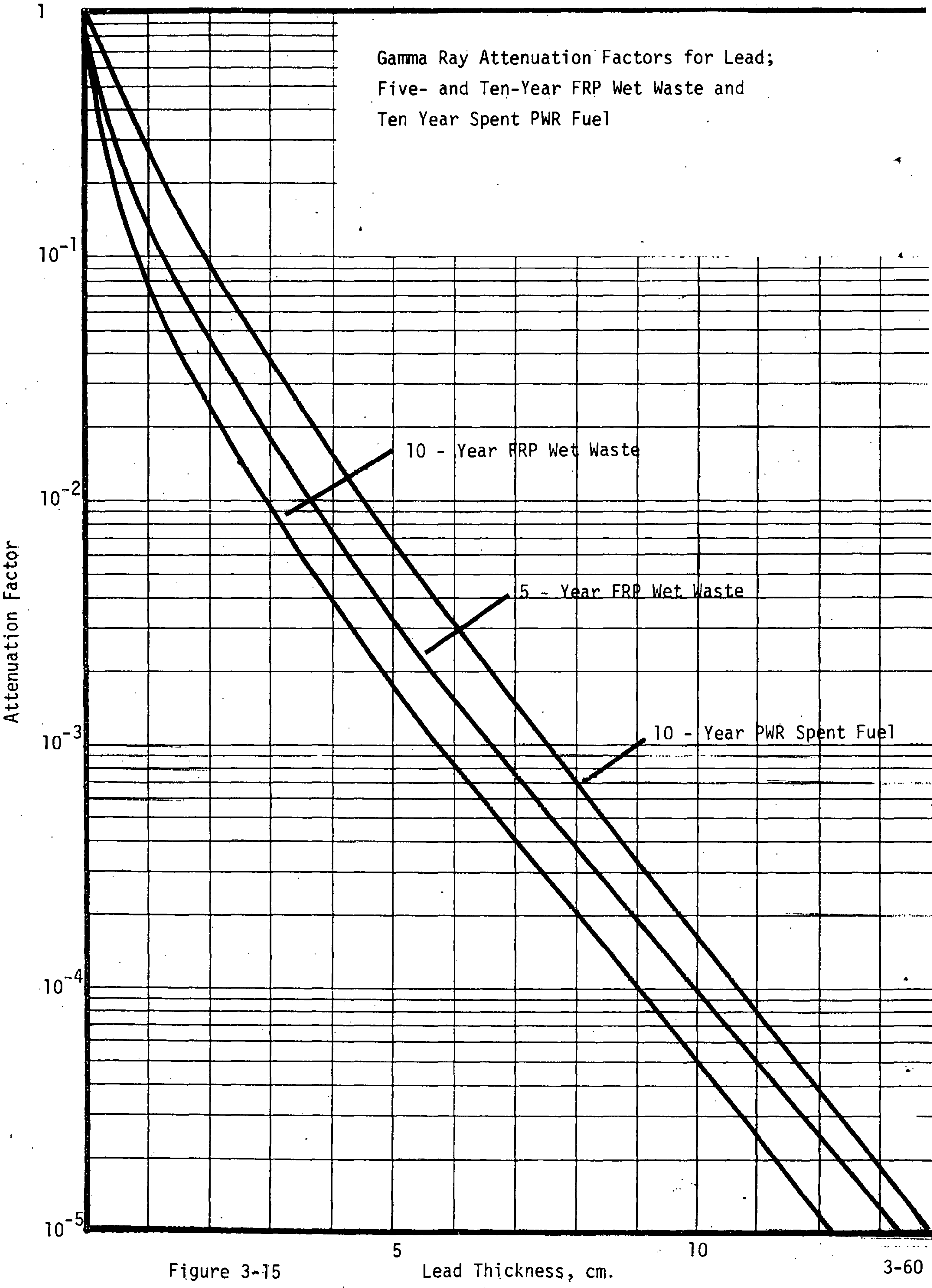
Gamma Ray Attenuation Factors for Lead; Five- and Ten-Year FRP Wet Waste and Ten Year Spent PWR Fuel

Figure 3-15

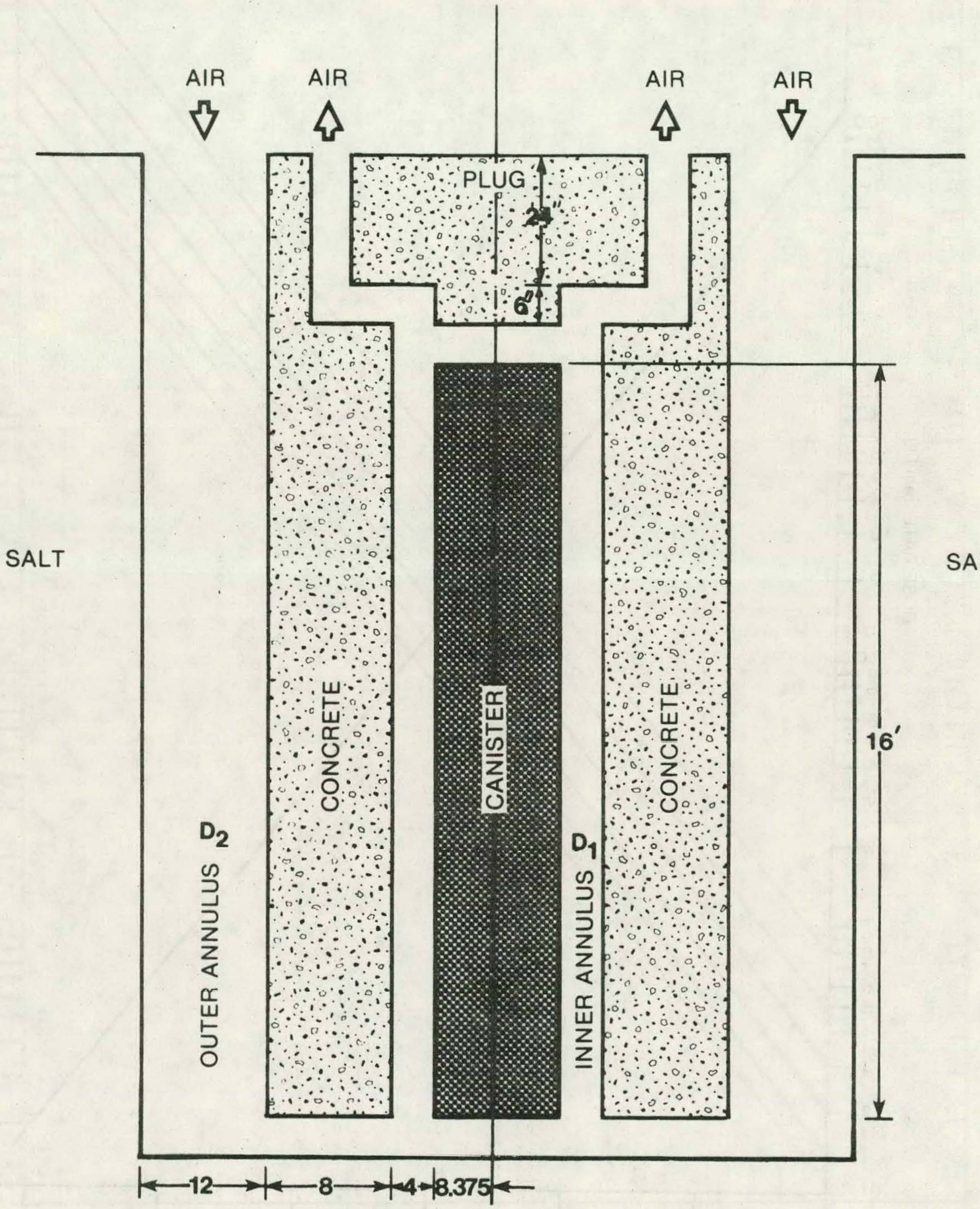

Figure 3-16 PWR SPENT-FUEL STORAGE CONFIGURATION FOR SALT

1089-11-1

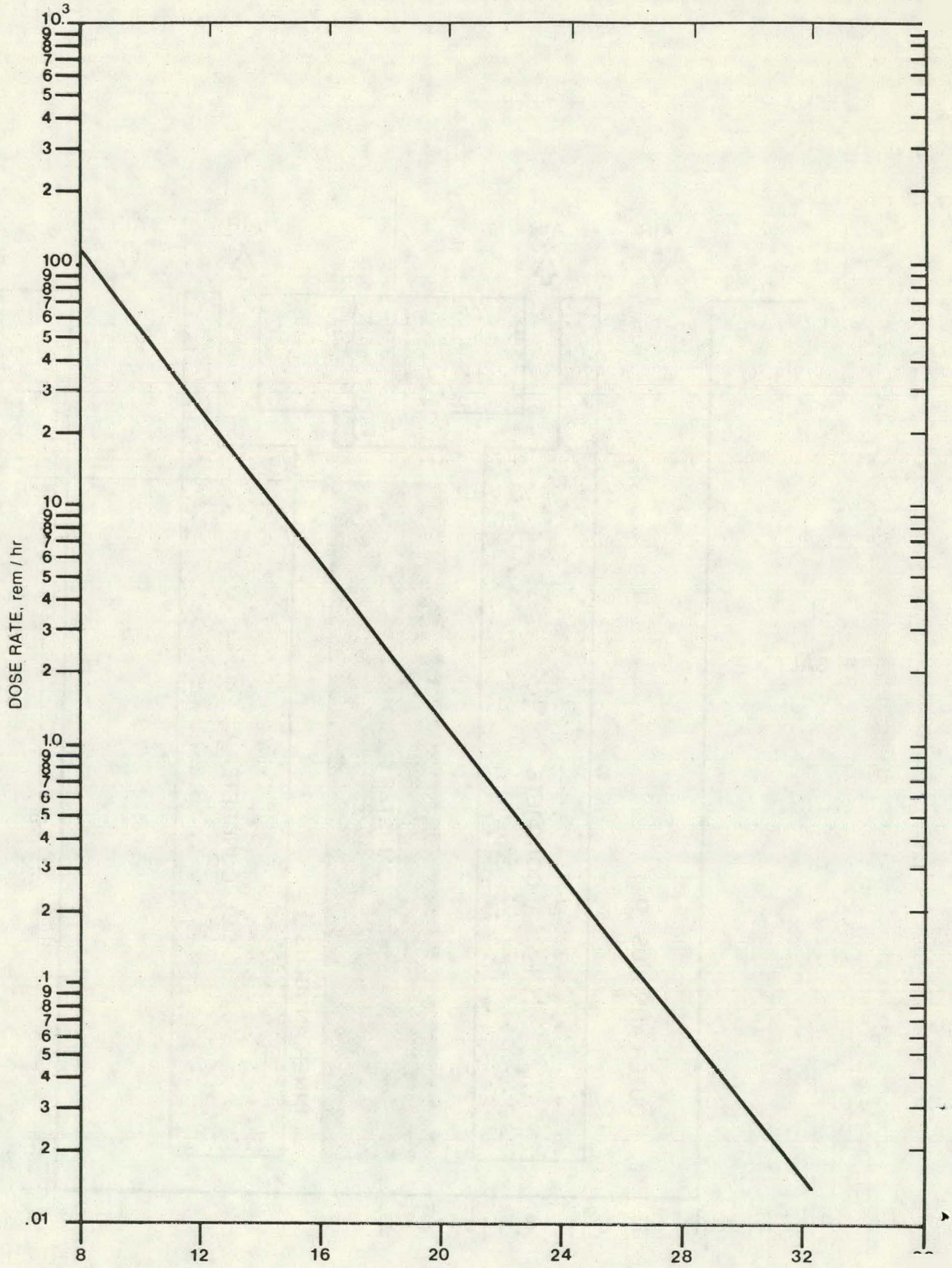

Figure 3-17 DOSE RATE IN THE OUTER ANNULUS AS A FUNCTION OF CONCRETE THICKNESS

PA1089-10-1

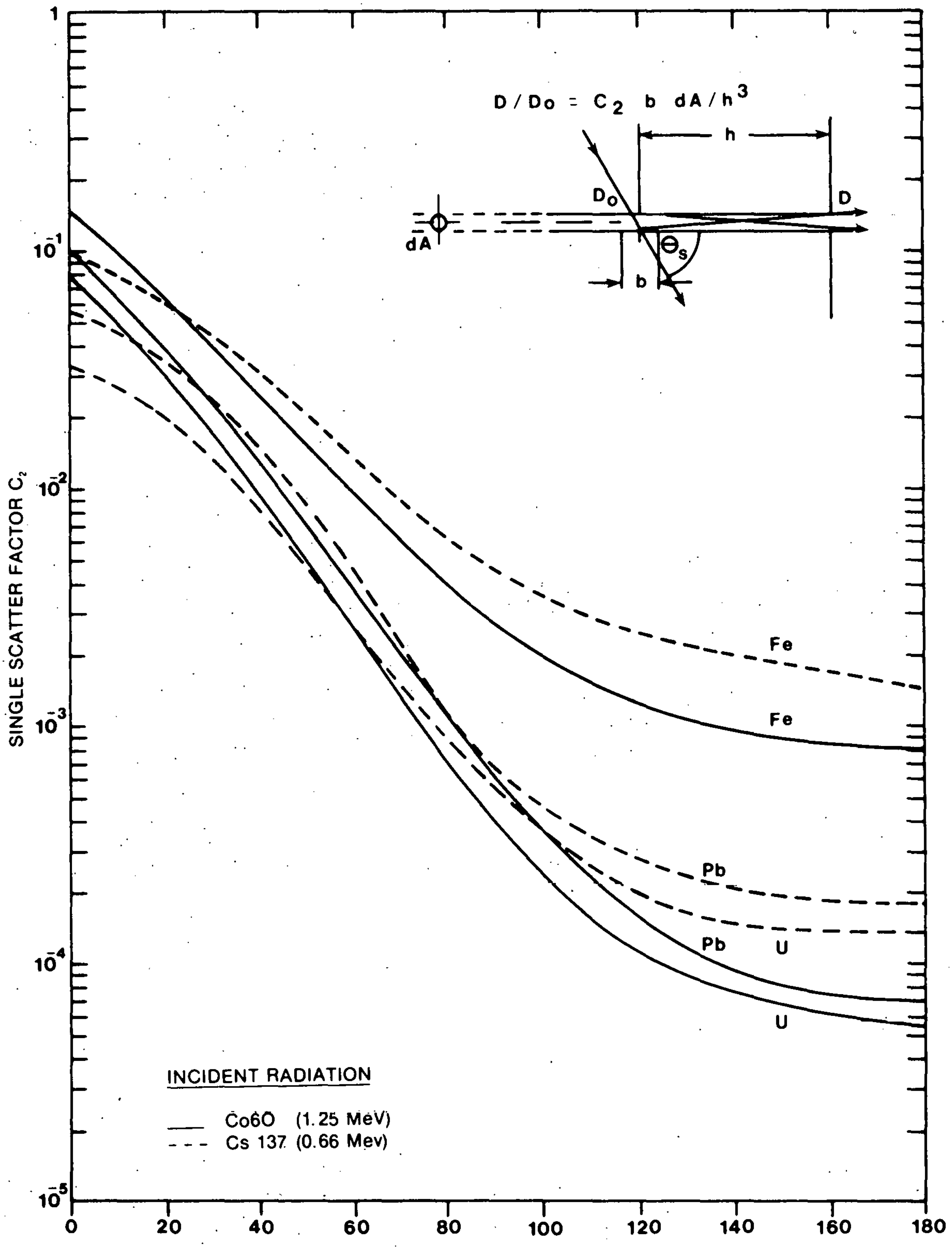

Figure 3-18 SINGLE-SCATTER FACTOR C_2 FOR LEAKAGE INTO SIDE OF NARROW HOLE

PA1089-9-2

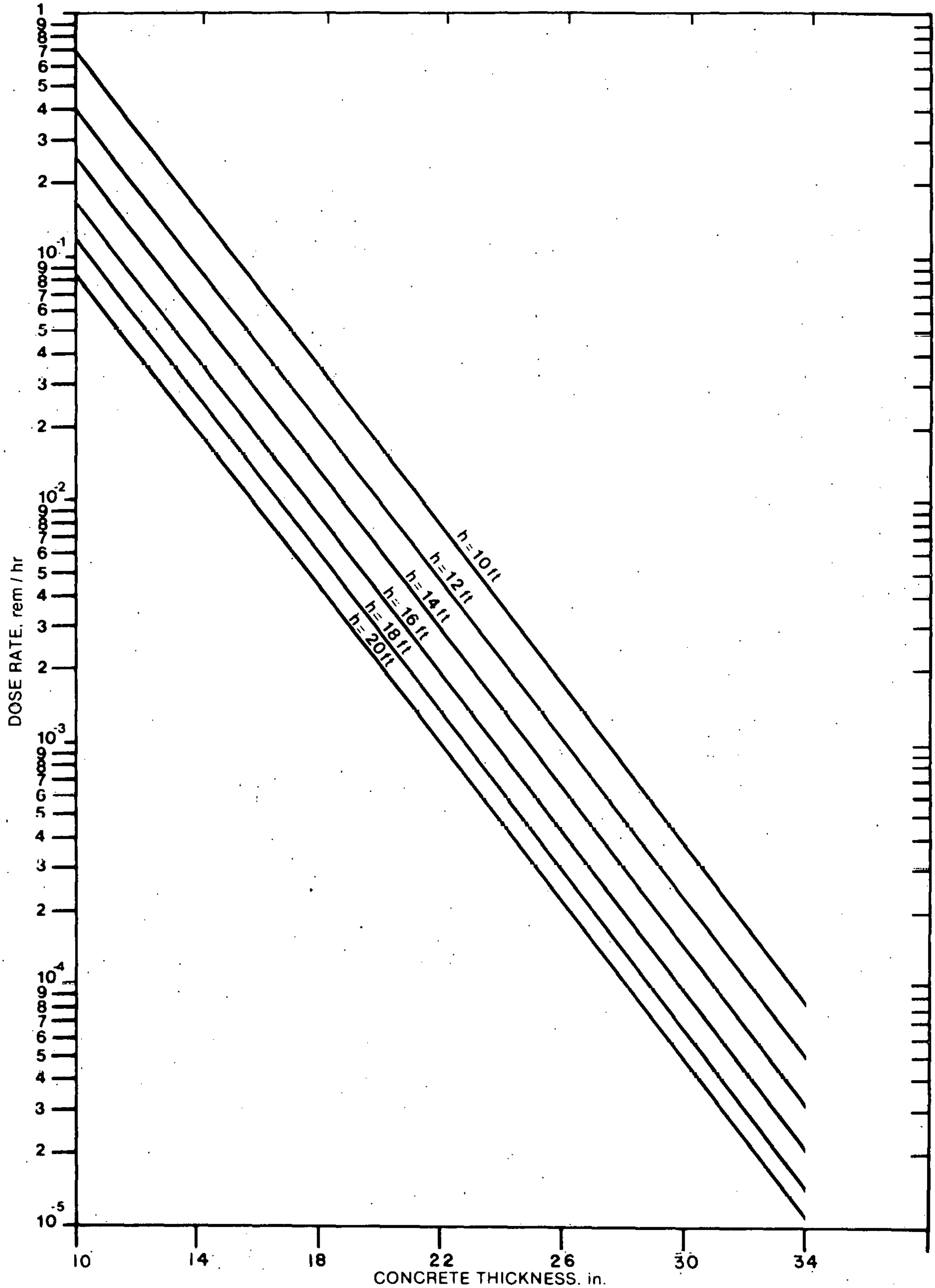

Figure 3-19 DOSE RATE ABOVE OUTER ANNULUS AS A FUNCTION OF CONCRETE THICKNESS FOR VARIOUS CANISTER DEPTHS

PA1089-8-1

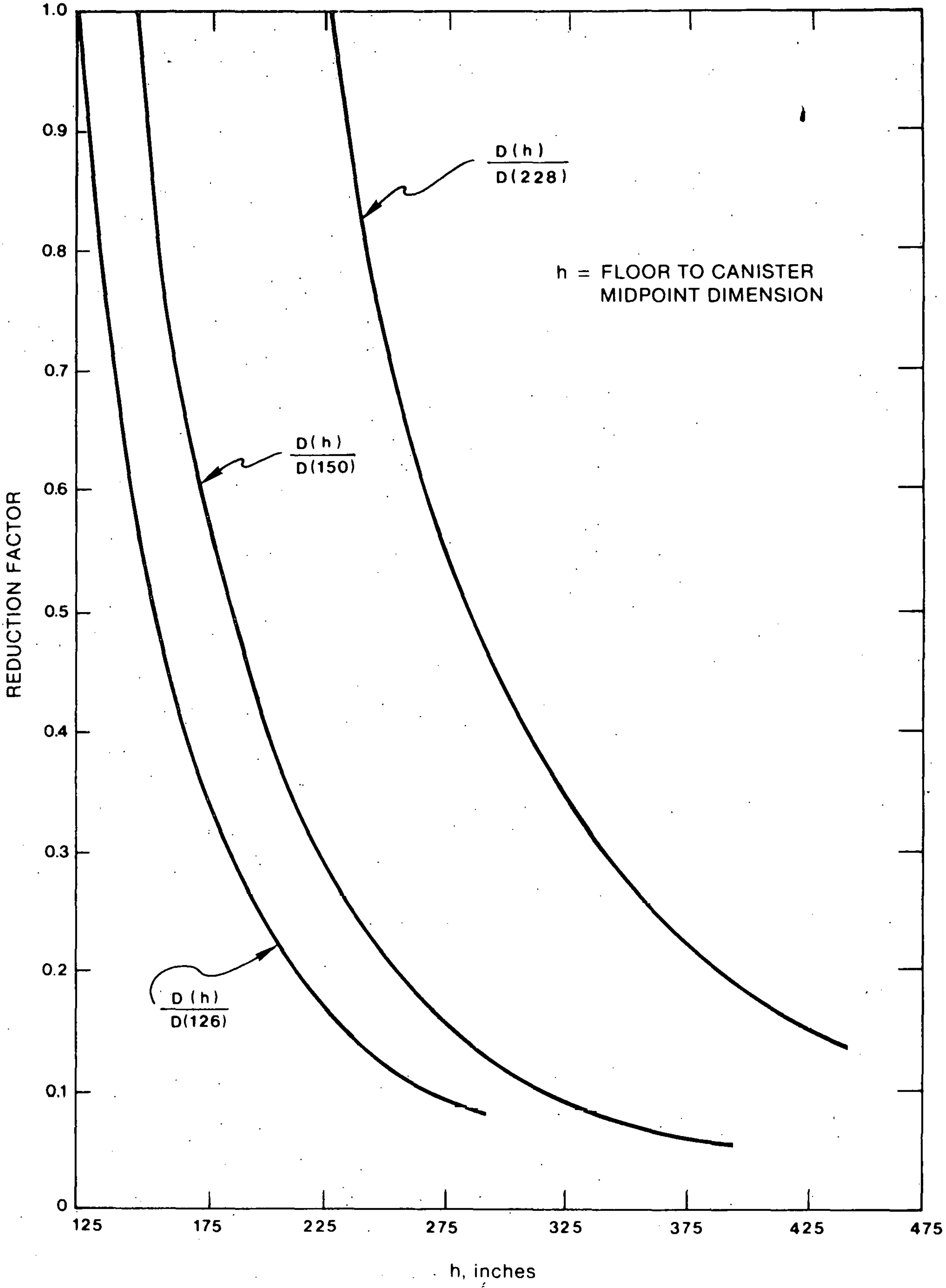

Figure 3-20 DOSE REDUCTION FACTOR AS A FUNCTION OF CANISTER DEPTH

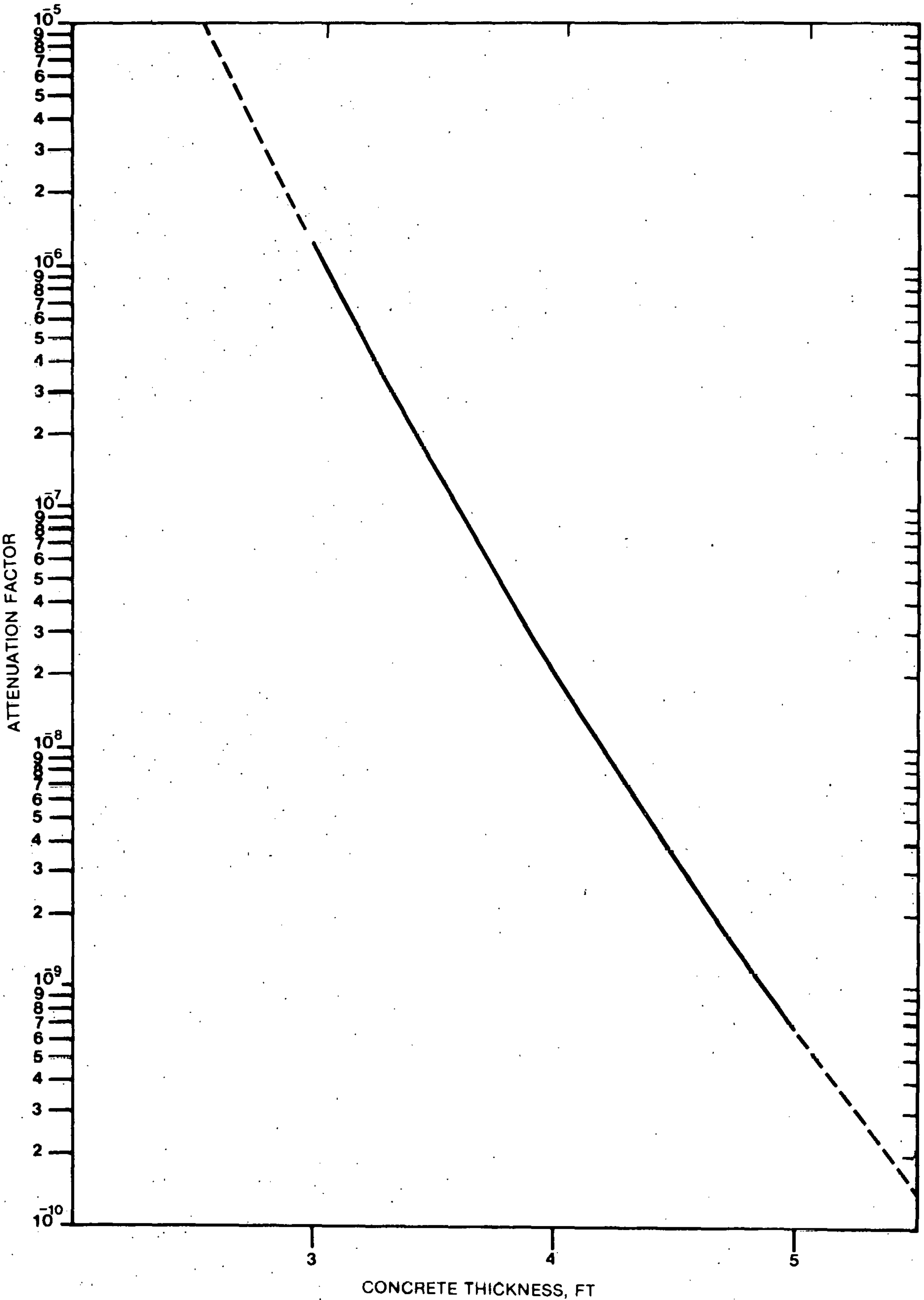

Figure 3-21 CONCRETE ATTENUATION FACTOR

PA1089-6-1

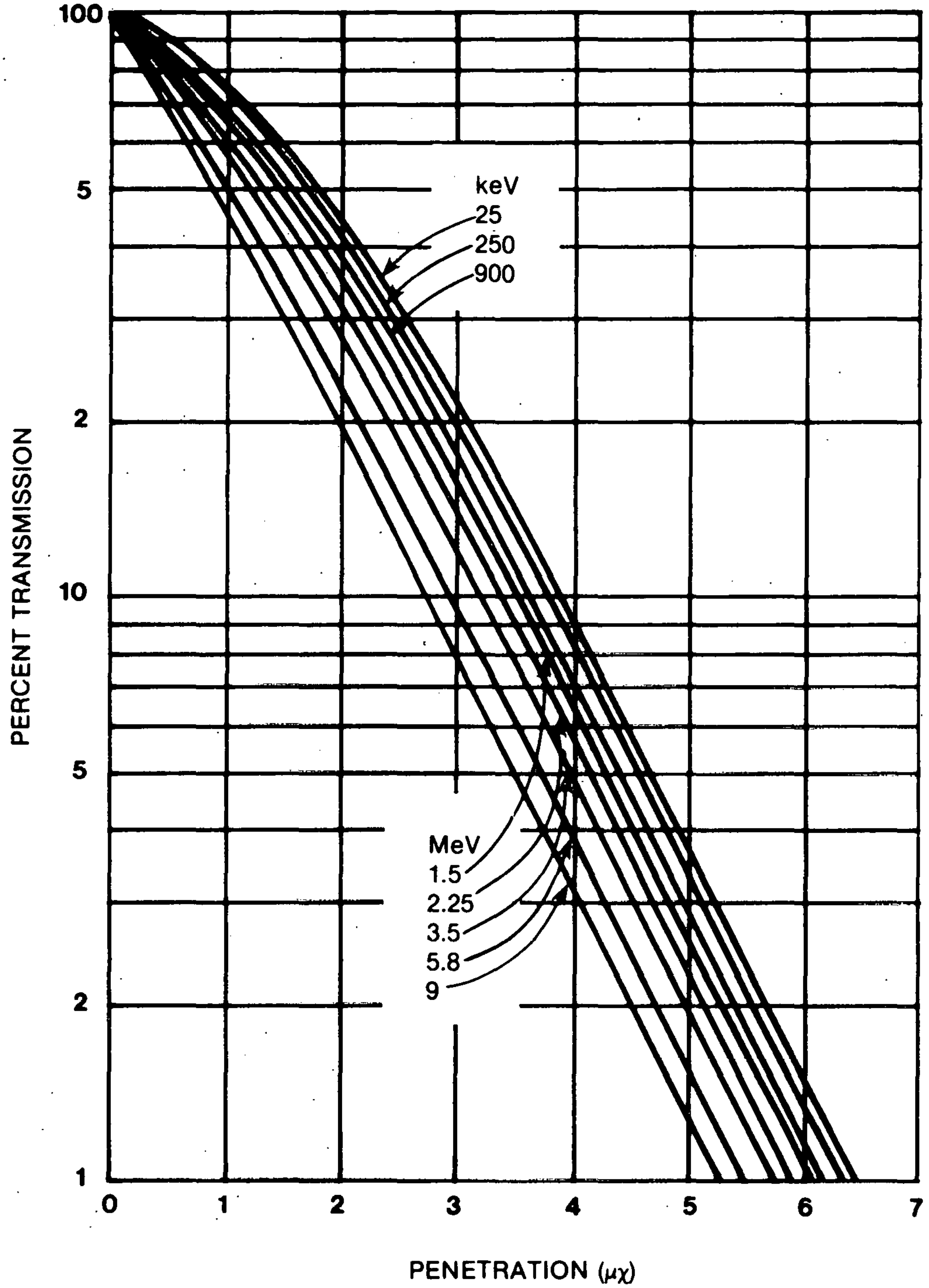

Figure 3-22 TRANSMISSION OF PHOTON KERMA THROUGH ORDINARY CONCRETE

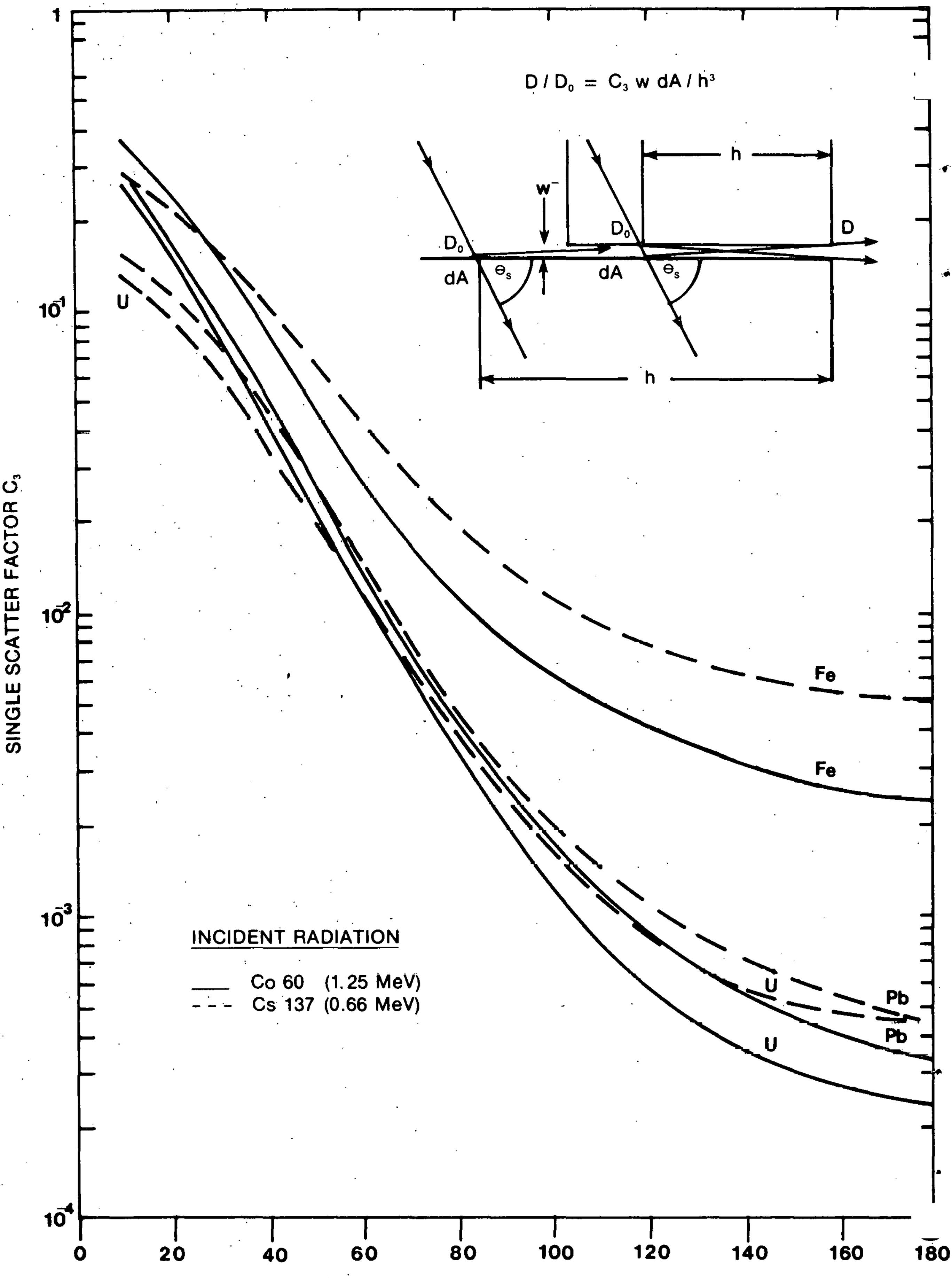

Figure 3-23 SINGLE—SCATTER FACTOR C_3 FOR GRAZING DIFFUSE REFLECTION FROM FLAT SURFACE OR LEAKAGE INTO NARROW PLANE GAP

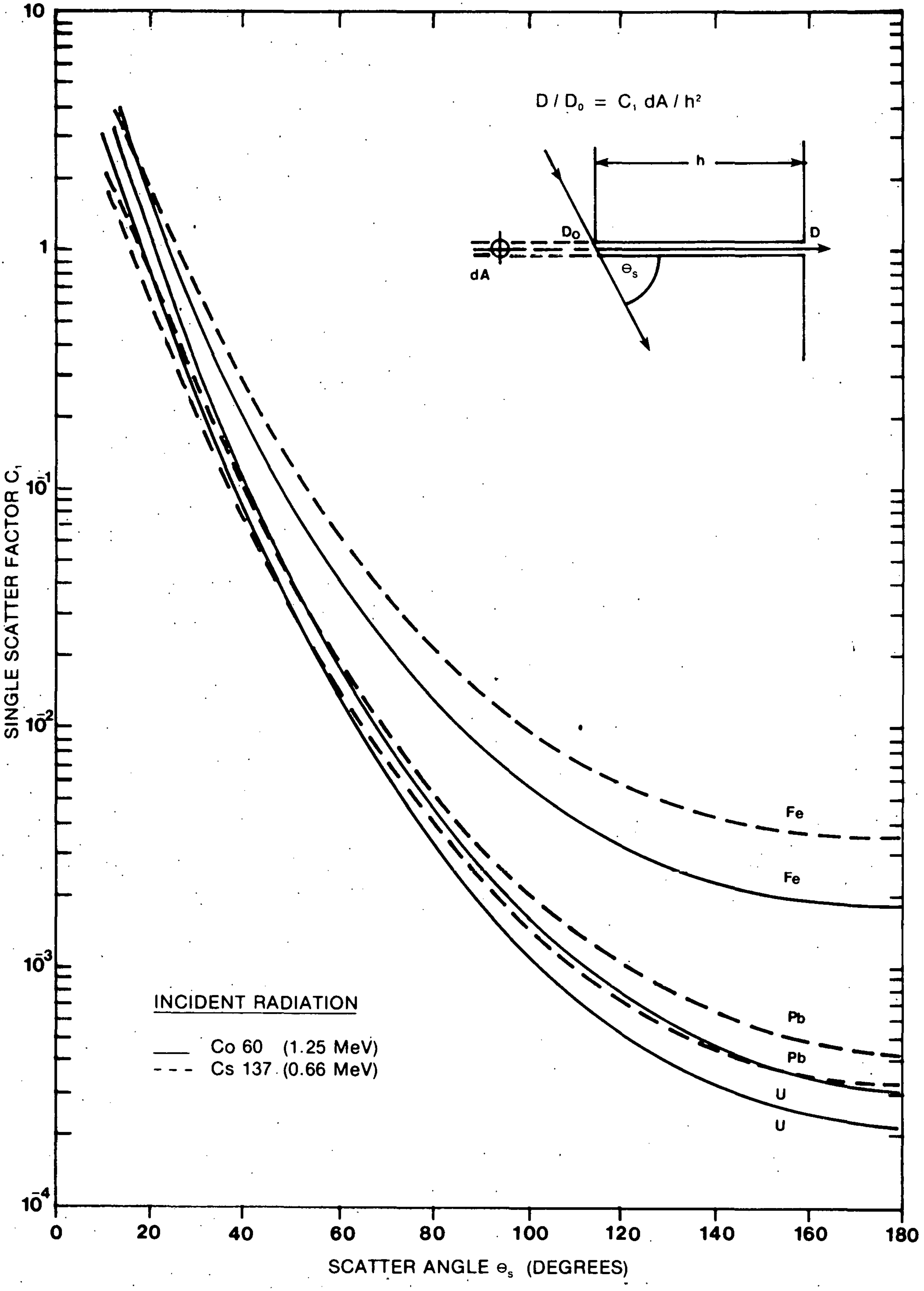

Figure 3-24 SINGLE-SCATTER FACTOR C_1 FOR LEAKAGE INTO BOTTOM OF NARROW OPENING

8

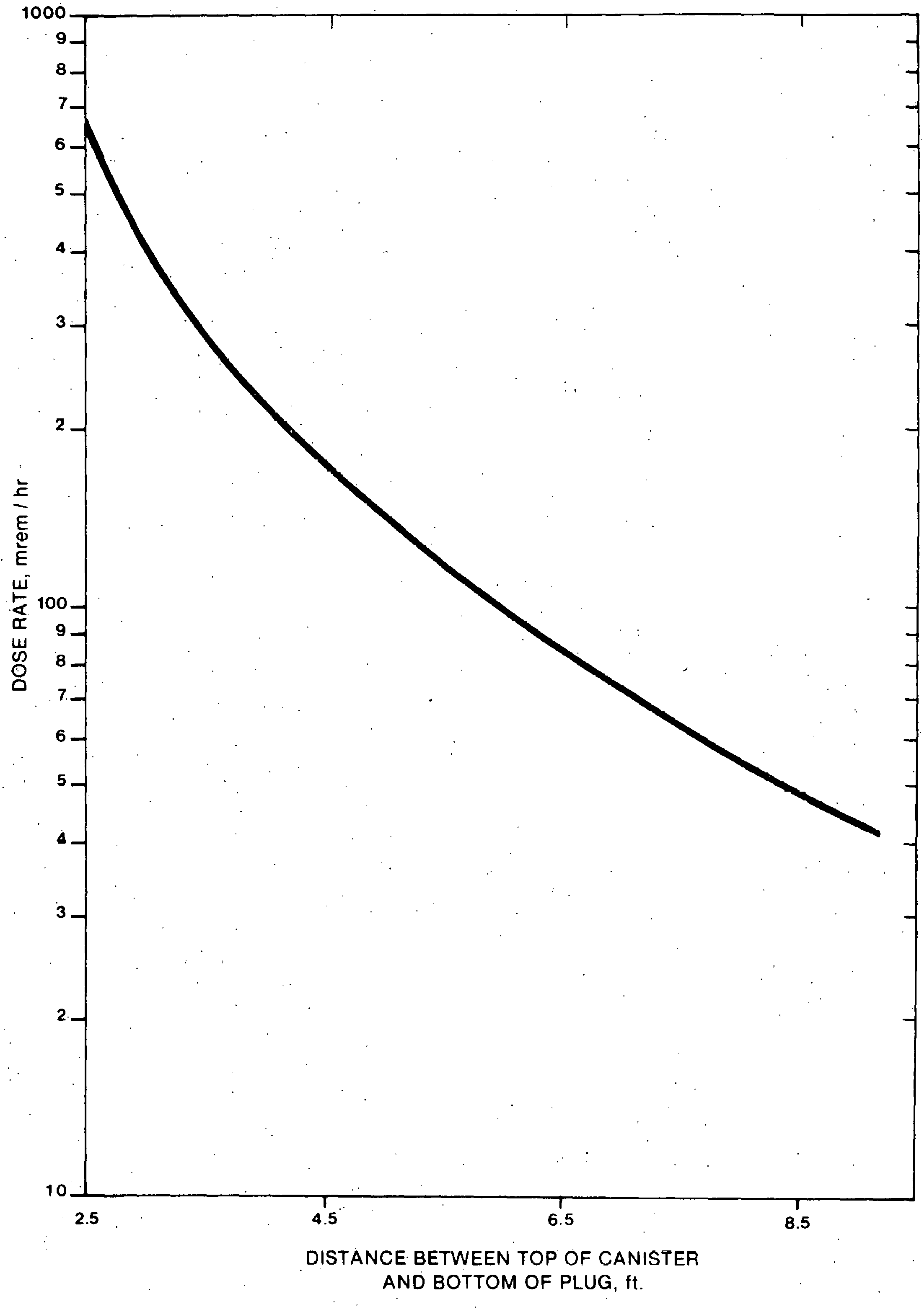

Figure 3-25 DOSE RATE AT TOP OF PLUG ABOVE INNER ANNULUS

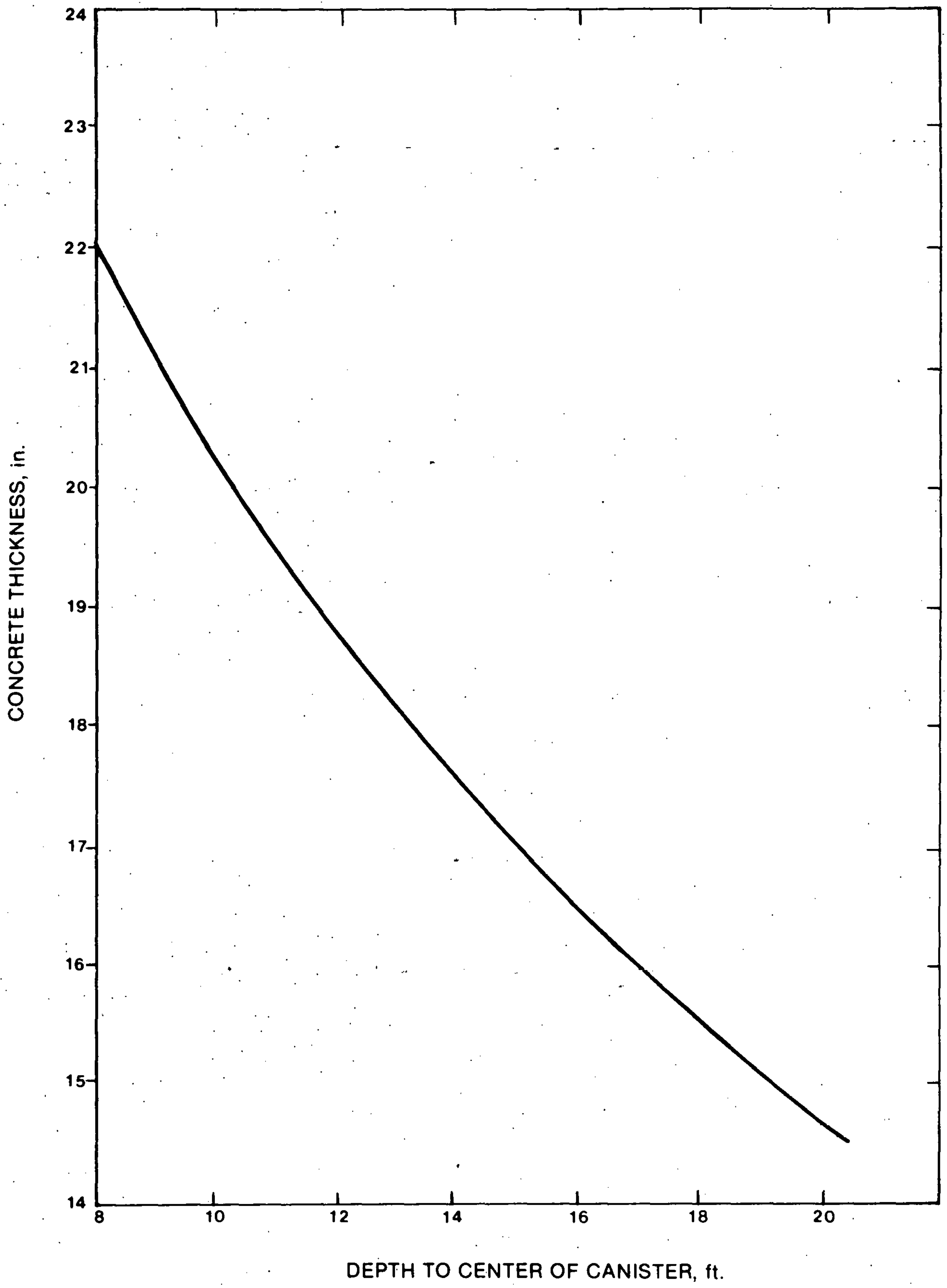

Figure 3-26 CONCRETE REQUIRED FOR 25 mrem / hr ABOVE OUTER ANNULUS

TABLE 3-1
DOSE RATES FOR 5 AND 10-YEAR PWR SPENT FUELS

Fuel	Distance from Centerline	Dose Rate (rem/hr)
5-Year PWR	7.1 in. (surface)	20,078
	24 in.	5,459
10-Year PWR	7.1 in. (surface)	8,260
	24 in.	2,245

TABLE 3-2
GROUP PHOTON SPECTRUM PER CANISTER (5-YEAR PWR HLW GLASS)

Group Mean Energy (MeV)	Intensity (Photons/Sec)
0.30	8.14E14
0.63	1.77E16
1.10	8.80E14
1.55	1.30E14
1.99	1.20E13
2.38	3.66E12
2.75	2.86E11
3.25	9.07E9

TABLE 3-3
ESTIMATED ERRORS IN LEAD ATTENUATION CURVES VERSUS SUBTENDED ANGLES

θ (degrees)	ϕ(point)/ϕ(exact)	Percent Error
1	1.0001	0.01
5	1.003	0.3
10	1.010	1.0
15	1.024	2.4
20	1.043	4.3
25	1.069	6.9
30	1.103	10.3

TABLE 3-4

SPECTRUM* OF THE 5-YEAR CONSTITUENTS, ORIGEN GROUP FORMAT

Average Group Energy, MeV	Clad & Structural Material	Fission Products	Actinides
5.25	----	----	3.06E-8
4.7	----	----	4.86E-8
4.22	----	----	1.03E-7
3.7	----	----	1.63E-7
3.25	----	2.53E-7	2.75E-7
2.75	----	7.99E-6	5.52E-6
2.38	----	1.03E-4	8.77E-7
1.99	2.39E-12	4.66E-4	1.86E-6
1.55	1.15E-10	5.94E-3	4.06E-6
1.1	1.00	2.81E-2	5.00E-3
0.63	2.51E-3	9.22E-1	1.95E-2
0.3	8.36E-5	4.31E-2	4.56E-3
0.2			6.84E-3
0.15			1.16E-2
0.1			1.90E-3
0.06			6.37E-1
0.04			2.60E-1
0.03			5.14E-2
$\frac{S_T(5\text{ yr.})}{S_T(10\text{ yr.})}$	1.94	1.83	0.71

*Fraction of source gamma rays emitted per group.

TABLE 3-5

SPECTRUM* OF THE 10-YEAR CONSTITUENTS, ORIGEN GROUP FORMAT

Average Group Energy, MeV	Clad & Structural Material	Fission Products	Actinides
5.25	----	---	2.50E-8
4.7	----	---	3.98E-8
4.22	----	---	8.40E-8
3.7	----	---	1.33E-7
3.25	----	1.93E-8	2.07E-7
2.75	----	6.09E-7	7.46E-6
2.38	----	7.77E-6	7.18E-7
1.99	3.19E-11	2.72E-5	1.57E-6
1.55	1.05E-9	2.52E-3	3.03E-6
1.1	9.92E-1	3.66E-2	6.65E-6
0.63	8.81E-3	9.21E-1	4.29E-4
0.3	5.03E-4	3.89E-2	3.39E-3
0.2			4.95E-3
0.15			5.49E-3
0.1			1.69E-3
0.06			7.49E-1
0.04			1.77E-1
0.03			5.99E-2

*Fraction of source gamma rays emitted per group.

TABLE 3-6

SPECTRUM* OF THE 5-YEAR CONSTITUENTS, CASK GROUP FORMAT

Cask Group	Energy Interval, MeV	Clad & Structural Material	Fission Products	Actinides
23	8 - 10	----	---	---
24	6.5 - 8	----	---	---
25	5.0 - 6.5	----	---	3.06E-8
26	4.0 - 5.0	----	---	1.52E-7
27	3.0 - 4.0	----	2.53E-7	4.38E-7
28	2.5 - 3.0	----	3.37E-5	5.74E-6
29	2.0 - 2.5	1.20E-12	3.10E-4	1.59E-6
30	1.66 - 2.0	3.70E-11	2.08E-3	2.19E-6
31	1.33 - 1.66	4.44E-2	5.34E-3	2.25E-4
32	1.0 - 1.33	7.33E-1	2.06E-2	3.67E-3
33	0.8 - 1.0	2.23E-1	1.91E-1	5.03E-3
34	0.6 - 0.8	1.00E-3	3.69E-1	7.84E-3
35	0.4 - 0.6	1.00E-3	3.69E-1	7.84E-3
36	0.3 - 0.4	2.09E-5	1.08E-2	3.04E-3
37	0.2 - 0.3	2.09E-5	1.08E-2	6.08E-3
38	0.1 - 0.2	2.09E-5	1.08E-2	1.49E-2
39	0.05 - 0.1	1.04E-5	5.39E-3	6.38E-1
40	0 - 0.05	1.04E-5	5.39E-3	3.11E-1

*Fraction of source gamma rays emitted per group.

TABLE 3-7

SPECTRUM* OF THE 10-YEAR CONSTITUENTS, CASK GROUP FORMAT

Cask Group	Energy Interval, MeV	Clad & Structural Material	Fission Products	Actinides
23	8 - 10	----	---	---
24	6.5 - 8	----	---	---
25	5.0 - 6.5	----	---	2.50E-8
26	4.0 - 5.0	----	---	1.24E-7
27	3.0 - 4.0	----	1.93E-8	3.4E-7
28	2.5 3.0	----	2.55E-6	7.64E-6
29	2.0 - 2.5	1.59E-11	1.94E-5	1.32E-6
30	1.66 - 2.0	3.43E-10	7.98E-4	1.73E-6
31	1.33 - 1.66	4.41E-2	3.36E-3	2.38E-6
32	1.0 - 1.33	7.27E-1	2.68E-2	4.88E-6
33	0.8 - 1.0	2.22E-1	1.92E-1	8.73E-5
34	0.6 - 0.8	3.52E-3	3.68E-1	1.72E-4
35	0.4 - 0.6	3.52E-3	3.68E-1	1.72E-4
36	0.3 - 0.4	1.26E-4	9.72E-3	2.26E-3
37	0.2 - 0.3	1.26E-4	9.72E-3	4.43E-3
38	0.1 - 0.2	1.26E-4	9.72E-3	8.08E-3
39	0.05 - 0.1	6.29E-5	4.86E-3	7.50E-1
40	0 - 0.05	6.29E-5	4.86E-3	2.37E-1

*Fraction of source gamma rays emitted per group.

TABLE 3-8

SOURCE SPECTRUM FOR 5-YEAR FRP WET WASTES

CASK Group	Activation Products (γ/sec) 5.18E10 γ/sec/drum*	Fission Products (γ/sec) 9.657E10 γ/sec/drum*	Actinides (γ/sec) 3.51E12 γ/sec/drum*	Total (γ/sec)
23	---	---	---	---
24	---	---	---	---
25	---	---	8.79E4	8.79E4
26	---	---	4.36E5	4.36E5
27	---	1.87E3	1.20E6	1.20E6
28	---	2.46E5	2.69E7	2.71E7
29	8.24E2	1.87E6	4.64E6	6.51E6
30	1.77	7.70E7	6.08E6	8.31E7
31	2.29E8	3.24E8	8.37E6	5.61E8
32	3.76E9	2.59E9	1.72E7	6.37E9
33	1.15E9	1.85E10	3.07E8	2.00E10
34	1.82E7	3.55E10	6.05E8	3.61E10
35	1.82E7	3.55E10	6.05E8	3.61E10
36	6.53E5	9.41E8	7.94E9	8.88E9
37	6.53E5	9.41E8	1.56E10	1.65E10
38	6.53E5	9.41E8	2.84E10	2.93E10
39	3.25E5	4.69E8	2.64E12	2.64E12
40	3.26E5	4.69E8	8.33E11	8.33E11

*Table A5. Ci/MTHM x $\frac{1}{1.263}\frac{\text{MTHM}}{\text{Drum}}$ (C. D. Zerby Letter of September 28, 1977; ORIGEN run 11/3/76.)

TABLE 3-9

SOURCE SPECTRUM FOR 10-YEAR FRP WET WASTES

CASK Group	Clad (γ/sec) 1.02E10 γ/sec/drum*	Fission Products (γ/sec) 1.77E11 γ/sec/drum*	Actinides (γ/sec) 2.50E12 γ/sec/drum*	Total (γ/sec)
23	---	---	---	---
24	---	---	---	---
25	---	---	7.65E4	7.65E4
26	---	---	3.80E5	3.80E5
27	---	4.48E4	1.09E6	1.13E6
28	---	5.96E6	1.43E7	2.03E7
29	1.23E-2	5.49E7	3.97E6	5.89E7
30	3.79E-1	3.68E8	5.47E6	3.73E8
31	4.54E8	9.45E8	5.62E8	1.96E9
32	7.50E9	3.65E9	9.17E9	2.03E10
33	2.28E9	3.38E10	1.26E10	4.89E10
34	1.02E7	6.53E10	1.96E10	8.49E10
35	1.07E7	6.53E10	1.96E10	8.49E10
36	2.14E5	1.91E9	7.60E9	9.51E9
37	2.14E5	1.91E9	1.52E10	1.71E10
38	2.14E5	1.91E9	3.72E10	3.91E10
39	1.06E5	9.54E8	1.60E12	1.60E12
40	1.06E5	9.54E8	7.77E11	7.78E11

*Table A5. Ci/MTHM x $\frac{1}{1.263}\frac{\text{MTHM}}{\text{Drum}}$ (C. D. Zerby Letter of September 28, 1977; ORIGEN run 11/3/76.)

TABLE 3-10

GAMMA RAY ENERGY SPECTRA* ON THE SURFACE OF 55-GALLON DRUMS

Cask Group	10-Yr PWR SF	10-Yr FRP WW	5-Yr FRP WW
40	3.80E-7	7.66E-3	6.83E-3
39	9.67E-4	8.38E-1	6.32E-1
38	3.16E-2	5.37E-2	1.20E-1
37	7.20E-2	2.48E-2	5.88E-2
36	8.40E-2	1.42E-2	3.58E-2
35	2.80E-1	2.65E-2	7.65E-2
34	3.06E-1	2.03E-2	5.85E-2
33	1.83E-1	1.07E-2	3.21E-2
32	3.62E-2	3.69E-3	1.43E-2
31	5.17E-3	3.52E-4	1.50E-3
30	1.26E-3	5.68E-5	3.03E-4
29	3.40E-5	7.12E-6	5.36E-5
28	4.67E-6	2.10E-5	1.90E-5
27	4.27E-8	1.08E-6	1.23E-6
26	2.12E-9	4.10E-7	4.31E-7
25	5.39E-10	8.79E-8	9.23E-8
24	7.91E-11	---	---
23	3.01E-11	---	---

*Fraction of source gamma rays emitted per group.

TABLE 3-11

GAMMA RAY ATTENUATION FACTORS FOR LEAD

Lead Thickness, cm	10-Yr PWR	5-Yr FRP WW	10-Yr FRP WW
0	1.0	1.0	1.0
.5	0.48	---	---
1.0	0.27	0.134	7.54E-2
1.5	0.16	7.63E-2	---
2.0	9.7E-2	4.56E-2	---
2.5	6.0E-2	---	---
3.0	3.8E-2	1.79E-2	---
3.5	2.4E-2	---	---
4.0	1.6E-2	---	---
4.5	1.0E-2	---	---
5.0	6.9E-3	3.35E-3	1.70E-3
7.5	1.0E-3	5.18E-4	---
10.0	1.7E-4	---	---
12.0	----	2.55E-5	1.19E-5
15.0	6.8E-6	---	---

TABLE 3-12

DOSE RATE TO FORKLIFT OPERATOR--NO SHIELDING OF THE CAB OR SELF-SHIELDING*

	Operator Dose Rate (mrem/hr)	
Surface Dose Rate	Single Pallet	Two Pallets
1 mrem/hr	0.12	0.24
10 mrem/hr	1.2	2.4
0.2 rem/hr	23.6	47.2
1 rem/hr	118.	236.

*Self-shielding here refers to the shielding provided by the front row of drums to the radiation emitted by the back row. This effect could be important for drums which contain concrete or other relatively dense matrix material.

TABLE 3-13

DOSE RATE TO FORKLIFT OPERATOR*--NO SHIELDING OF THE CAB

	Operator Dose Rate (mrem/hr)	
Surface Dose Rate	Single Pallet	Two Pallets
1 mrem/hr	0.0765	0.153
10 mrem/hr	0.765	1.53
0.2 rem/hr	15.3	30.6
1 rem/hr	76.5	153.

*Self-shielding by the front row of drums included.

TABLE 3-14

FORKLIFT SHIELDING REQUIREMENTS TO REDUCE OPERATOR'S DOSE RATE TO 0.25 mrem/hr--5-YEAR FRP WET WASTE

Surface Dose Rate	Cab Shielding Requirements (cm Lead)			
	No Self-Shielding		Self-Shielding Included	
	1-Pallet	2-Pallets	1-Pallet	2-Pallets
1 mrem/hr	0	0	0	0
10 mrem/hr	0.7	1.3	0.5	0.9
0.2 rem/hr	3.6	4.4	3.2	4.0
1 rem/hr	5.6	6.5	5.2	5.9

TABLE 3-15

FORKLIFT SHIELDING REQUIREMENTS TO REDUCE OPERATOR'S DOSE RATE TO 0.25 mrem/hr--10-YEAR FRP WET WASTE

Surface Dose Rate	Cab Shielding Requirements (cm Lead)			
	No Self-Shielding		Self-Shielding Included	
	1-Pallet	2-Pallets	1-Pallet	2-Pallets
1 mrem/hr	0	0	0	0
10 mrem/hr	0.5	0.75	0.25	0.5
0.2 rem/hr	2.8	3.6	2.4	3.25
1 rem/hr	4.7	5.6	4.25	5.0

TABLE 3-16

FORKLIFT SHIELDING REQUIREMENTS TO REDUCE OPERATOR'S DOSE RATE TO 0.25 mrem/hr--10-YEAR PWR SPENT FUEL

Surface Dose Rate	Cab Shielding Requirements (cm Lead)			
	No Self-Shielding		Self-Shielding Included	
	1-Pallet	2-Pallets	1-Pallet	2-Pallets
1 mrem/hr	0	0	0	0
10 mrem/hr	1.3	2	0.8	1.5
0.2 rem/hr	4.4	5.3	3.9	4.8
1 rem/hr	6.4	7.3	5.9	6.8

TABLE 3-17

DOSE RATE TO FORKLIFT OPERATOR--FINAL STORAGE CONFIGURATION (4 x 4 x 2 PALLETS)--NO SHIELDING OF THE CAB

	Operator's Dose Rate (mrem/hr)	
Surface Dose Rate	No Self-Shielding	Self-Shielding Included
1 mrem/hr	1.48	0.584
10 mrem/hr	14.8	5.84
0.2 rem/hr	296	116.8
1 rem/hr	1,480	584

TABLE 3-18

FORKLIFT CAB SHIELDING REQUIREMENTS TO REDUCE OPERATOR'S DOSE RATE TO 0.25 mrem/hr--FINAL STORAGE CONFIGURATION--5-YEAR FRP WET WASTE

	Cab Shielding Requirements (cm Lead)	
Surface Dose Rate	No Self-Shielding	Self-Shielding Included
1 mrem/hr	0.8	0.3
10 mrem/hr	3.0	2.1
0.2 rem/hr	6.8	5.6
1 rem/hr	9.2	7.8

TABLE 3-19

FORKLIFT CAB SHIELDING REQUIREMENTS TO REDUCE OPERATOR'S DOSE RATE TO 0.25 mrem/hr--FINAL STORAGE CONFIGURATION--10-YEAR FRP WET WASTE

	Cab Shielding Requirements (cm Lead)	
Surface Dose Rate	No Self-Shielding	Self-Shielding Included
1 mrem/hr	0.5	0.20
10 mrem/hr	2.3	1.5
0.2 rem/hr	5.9	4.75
1 rem/hr	8.1	6.80

TABLE 3-20

FORKLIFT CAB SHIELDING REQUIREMENTS TO REDUCE OPERATOR'S DOSE RATE TO 0.25 mrem/hr--FINAL STORAGE CONFIGURATION--10-YEAR PWR SPENT FUEL

	Cab Shielding Requirements (cm Lead)	
Surface Dose Rate	No Self-Shielding	Self-Shielding Included
1 mrem/hr	1.5	0.65
10 mrem/hr	3.9	2.85
0.2 rem/hr	7.6	6.50
1 rem/hr	10.0	8.70

TABLE 3-21

LINEAR ATTENUATION COEFFICIENTS FOR H_2O, Fe, CELLULOSE, ORDINARY CONCRETE, AND SOIL (Ref.3)

Energy (MeV)	H_2O[a] (cm-1)	Fe[b] (cm-1)	Wood[c] (cm-1)	Ordinary Concrete[d] (cm-1)	Soil[e] (cm-1)
10.0 - 8.0	0.0225	0.237	0.0140	0.0555	0.0375
8.0 - 6.5	0.0250	0.237	0.0154	0.0592	0.0400
6.5 - 5.0	0.0275	0.245	0.0170	0.0638	0.0433
5.0 - 4.0	0.0315	0.253	0.0196	0.0711	0.0483
4.0 - 3.0	0.0360	0.269	0.0222	0.0797	0.0544
3.0 - 2.5	0.0410	0.284	0.0254	0.0899	0.0614
2.5 - 2.0	0.0460	0.308	0.0284	0.0995	0.0682
2.0 - 1.66	0.0520	0.332	0.0315	0.110	0.0750
1.66 - 1.33	0.0575	0.363	0.0351	0.121	0.0829
1.33 - 1.00	0.0650	0.411	0.0399	0.139	0.0956
1.00 - 0.8	0.0740	0.482	0.0453	0.157	0.108
0.8 - 0.6	0.0840	0.553	0.0509	0.177	0.121
0.6 - 0.4	0.0970	0.648	0.0591	0.204	0.140
0.4 - 0.3	0.112	0.790	0.0680	0.238	0.163
0.3 - 0.2	0.127	0.988	0.0771	0.275	0.187
0.2 - 0.1	0.150	1.62	0.0910	0.336	0.224
0.1 - 0.05	0.183	5.53	0.110	0.545	0.333
0.05 - 0	0.510	790.	0.246	5.57	2.69

[a] $\rho = 1$

[b] $\rho = 7.9$

[c] $\rho = 0.64$ which corresponds to longleaf pine

[d] $\rho = 2.33$

[e] $\rho = 1.60$

4.0 RADIATION MONITORING PROGRAM

To ensure radiological protection for both on-site personnel and the off-site population, a program of measuring and monitoring will be provided at the repository to verify that the effluent control equipment's proper operation for controlling the release of radioactive materials is in accordance with 10CFR20. 10CFR20 restricts off-site losses to absolute values and to values that are as low as reasonably achievable (ALARA).* This program will measure repository contributions to off-site exposure by direct radiation, radioactivity in breathing air, deposited radioactivity, and radioactivity in consumed materials (water, milk, plants, meat, fish, etc.).

A specific radiological effluent-monitoring system will be designed, implemented, and maintained to provide data on measurable levels of radiation and concentrations of radioactivity in the site environs to assure compliance with 10CFR50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Practicable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," and to meet the intent of Criterion 64 of Appendix A, "Monitoring Radioactivity Releases." This section describes a radiation monitoring program proposed to meet the intent of these guides as they may apply to a generic repository.

4.1 MONITORING

4.1.1 Baseline Data

It is normal practice that two years prior to the start of plant operations, a baseline study be conducted to determine population distributions, local agricultural practices, prevailing meteorological conditions, local food-consumption patterns, and other variables required for definitive dosimetric calculations. Data collected would include background-radiation measurements and sampling and analysis of

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*See 3.0 for a more detailed discussion of the radiation control program.

air, milk, meat, agricultural products, wildlife flora and fauna in the human food chain, soils, and water. These data would be compared later with subsequent data collected during operation.

The preoperational measuring and monitoring program would be extended into the operational phase to assure continuity and directly comparable data.

4.1.2 Program Scope

During the first three years from the time waste is brought onto the site the measuring and monitoring program will be sufficiently comprehensive to verify any projected correlations between radioactive effluents and levels in environmental media.

After three years, if actual demonstration shows insignificant contributions of radioactivity to any specific pathway from plant operations, the monitoring and measuring of that pathway may be reduced to effect operating efficiency. Under no circumstances will the monitoring and measurement of a possible pathway be eliminated entirely during the operational and decommissioning phases of the repository. Results of all individual measurements will be retained, including such pertinent data as sampling location, sample count rate, sample size, sampling and analytical procedures, units of data presentation, and precision and accuracy associated with individual measurements. Explanation of anomalous measurements will be provided to the appropriate Regulatory Agencies.

The above statements apply during the operation and decommissioning phases of the repository. After all emplacements have been made, after the retrievability periods have expired, and after all backfilling has taken place and the repository has been sealed, it is anticipated that only a few rudimentary surface monitoring stations will continue to exist.

4.1.3 Analysis and Quality Control

Samples will be analyzed for specific radionuclides released, as well as gross monitoring of radioactivity attributable to plant operations.

Written procedures will be provided for the taking of samples and performance of analyses. Verification of proper procedures being used and the results will be documented.

Control checks and tests will be applied to the analysis of samples by the use of blind duplicate analyses of selected samples and probably by cross checking by an independent laboratory. Controls will be applied to the entire sample-collection procedure to assure that representative samples are obtained and that the samples are not changed, cross contaminated, or otherwise affected, prior to their analysis, or during handling or storage.

4.1.4 Detection Capabilities

The detection capability requirements will be established primarily on the basis of potential human dose. The most sensitive measurement techniques will be used, consistent with the state-of-the-art, for determination of radioactive concentrations and radiation levels. The objective of the measuring and monitoring techniques will be to detect levels corresponding to a few percent of concentration guides required by 10CFR20. In addition, the dynamic range of the instrumentation and techniques used for routine monitoring will be such that they will remain on scale and functioning under the worst-case postulated radioactive release.

4.2 MONITORING NETWORK

During the active life and decommissioning of the repository the measurement of radioactivity and radiation throughout the repository mine, surface facilities, and site environs will be accomplished by the systematic coverage of all potential pathways of release. The network provides redundancy by having monitoring stations in series along the potential pathway from the waste itself to the outermost perimeter of the site.

4.2.1 Storage Rooms

The canister-storage rooms will be monitored on a continuous basis. A continuous sample of air will be drawn from the room and routed to a particulate/gas on-line monitor through a manifold system. The particulates will be monitored for α, β and γ activity, and the gases will be monitored for β activity. Helium detectors can be used to locate areas with potential activity releases. The monitoring arrangement is shown in Figure 4-1.

All activity levels and alarms will be displayed locally at the monitoring stations, as well as being transmitted to the central control station.

4.2.2 Corridors and Vertical Shafts

All corridors and vertical shafts will be continuously monitored with on-line monitors. An αβγ particulate and β-gas monitor will be located in the main corridors near the waste receiving station. In addition, identical monitors will be located at intervals throughout the corridor network.

Area monitors measuring ambient γ radiation will be located throughout the tunnels and vertical shafts. The area-monitoring network will provide a means of locating and tracking radiation movement throughout the mine by the central-control-station operator.

All activities and alarms will be displaced locally at the monitoring stations, as well as being transmitted to the central control station. Figure 4-1 shows the locations of underground radiation monitors.

4.2.3 Underground Transport Vehicles

Each canister and drum underground transport vehicle will have its own self-contained monitoring station. Each vehicle's monitoring station will include an αβγ-particulate and β-gas monitor. The air sample for these monitors will be taken inside the occupied control cab. Also inside the cab will be a γ and neutron area monitor. A γ and neutron area monitor detector will also be mounted externally to the shielded cab with readout on the operator's console. In addition to the data displays in the cab, a communications link, radio or hardware if available, will be maintained for all data transmissions to the central control station (Figure 4-2).

4.2.4 Central Area Surface Facilities

Each interior separate space envelope of the surface facilities will have a continuous on-line αβγ-particulate and β-gas monitor. A similar monitoring station will be mounted on the roof of the highest building. Area γ and neutron monitors will be located throughout the surface handling facility. All liquid drains will be continuously monitored for αγ activity (Figures 4-3 and 4-4).

Gamma area monitors will be installed in the hallways of all significant buildings of the facility and on exterior walls. Within the central facility αβγ-particulate and β-gas monitors will be placed at several outdoor locations.

All of these monitors will have local displays and displays in the gate guard house, as well as data transmission to the central control center (Figure 4-5).

All ventilation systems and stacks will be monitored by redundant αβγ-particulate and β-gas monitors. Redundancy will extend to and include the primary power source used by the monitor.

All access gates to the central area will be continuously monitored for αβγ-particulate, β-gases, and ambient γ radiation.

4.2.5 Inner Controlled Area

The tentative monitoring arrangement for the inner and outer controlled areas is shown in Figure 4-6. The inner controlled area lies directly above the 2000-acre mine and since it excludes the 200-acre central area, covers about 1800 acres. Each quadrant of the inner controlled area perimeter will be monitored by αβγ-particulate, β-gas, and Hi-Lo area monitors. The location of sampling stations and frequency of sample collection will be determined by site-specific factors such as population distribution, agricultural activity, meteorology, hydrology, and topography. An additional monitoring station of similar design will be adjacent to the access route.

The perimeter monitoring stations will be self-contained, all-weather enclosures of tamper-proof design. Continuous data communications with the central control station will be maintained.

4.2.6 Outer Controlled Area

The outer controlled area forms a 14,000-acre annulus surrounding the inner controlled area and is bounded by a peripheral road passable at least by 4-wheel-drive vehicles. The area will be monitored in the same manner and configuration as the inner controlled area.

4.2.7 Mobile Monitoring Station

An all-terrain vehicle complete with $\alpha\beta\gamma$-particulate, β-gas, Hi-Lo, γ, and neutron monitors will be on-site and available for dispatch at all times. The vehicle will be equipped with a power generator and two-way radio communications with the central control station. The vehicle will contain a laboratory with basic equipment required for radiological analysis of environmental samples. Portable survey and sampling equipment will be included.

4.3 RADIATION MONITORING INSTRUMENTATION

All on-line monitors will be designed for continuous unattended operation for at least three months under normal operating conditions. All on-line monitors will be linked to the data acquisition and control consoles in the central control center located in the Mine Operations Building. In addition, each monitor will be capable of independent stand-alone operation. Monitors will be packaged in weatherproof enclosures with self-contained environment control. Tamper-proof access to controls and for maintenance will be provided. Local data displays and indicators will be in the same units as those located in the central control center for clarity of communications.

Output from the programmed monitor electronics will provide complete operating status to the data acquisition and control system including background compensation, multiple radiation alarms, failure alarms, radiation levels, and other pertinent data as needed to fulfill the operating software program.

Monitors with alarms that require corrective action should have the alarms in 2-of-3 logic to minimize false alarms. This logic requires at least 2 out of 3 monitors at the same location to be in an alarm state before the external alarm activates. Shielding around detectors will be provided as necessary to allow the sensitivities required. Each monitor will be provided with means of self-checking and self-aligning drift and calibration deviations. All monitors will be calibrated with nuclides and fully documented. In-situ recalibration will be provided with all monitors. All wetted surfaces of effluent monitors will be stainless steel. Provisions will be made for quick positive decontamination. The on-line monitors will be designed, manufactured, and operated in accordance with the applicable regulations.

4.3.1 On-Line Monitors

The on-line gas monitors will be designed for continuous flow-through service. The effluent to be monitored will be filtered prior to measurement, and a means of purging for decontamination will be provided. The scintillation or solid-state detectors will be sized to provide the sensitivity and range listed in Table 4-1. Gross β activity will be monitored.

The liquid monitors will be designed for continuous flow-through service. A system for purging will be provided. Solid-state or scintillation detectors can provide the sensitivity and range listed in Table 4-1. Gross α and γ radioactivity in the liquid will be monitored. Pulse height analyses of the γ activity will also be provided.

Area monitors will operate on a continuous basis. Range and sensitivity will be as listed in Table 4-1.

All particulate monitors will use a standard size filter with filter characteristics selectable from commercial suppliers. Proper effluent flow through the monitor will be assured by a self-regulating flow control system with redundant air movers. The detectors used in these monitors will be solid-state with simultaneous but separate α and β measurement. A γ detector will provide gross γ activity, as well as pulse-height analysis, on command.

4.3.2 Analytical Instrumentation

An adequately instrumented laboratory will be maintained to measure the effluent samples. The following types of instruments are required for adequate measurement.

(1) Low-level planchet counters capable of analyzing α, β, and γ activities.
(2) Well counter for analyzing γ activities in liquids.
(3) Counter for α activities in liquids.
(4) Supporting spectrometers, computers, records, and other electronic equipment as necessary to properly display and record the data from the counting equipment. Data acquisition, reduction, and storage must be compatible with the data acquisition and control system.

4.4 DATA ACQUISITION AND CONTROL (DAC) SYSTEM

The radiation monitoring network will transmit its data to a DAC system and be controlled by a DAC system (Figure 4-7). The DAC system will analyze the plant's radiological effluent data for reporting and decision-making purposes and should minimize the potential for analytical and clerical errors. In addition to servicing the monitoring network, the DAC will receive input from radiochemistry analyses and the site meteorology system. The DAC will combine these data with rate-of-release data from the monitors to provide total assessment of environmental dose impact on a real time basis. Predictive evaluation, averages, and baseline data will be in the DAC retrievable storage to facilitate reporting procedures. Control exercised by the DAC will include programmed filter changes, decontamination purges, alarm testing, calibration checking, and extensive diagnostics.

4.5 APPLICABLE REGULATIONS

The NRC is currently preparing the requirements for the design, construction, and operation of a National Waste Terminal Storage Facility. Because of the current lack of detailed guidelines, the radiation monitoring requirements should be based on the intent of the requirements used in the other segments of the nuclear industry. Table 4-2 contains a list of NRC regulatory guides that are generally applicable to a waste repository monitoring program.

Other applicable federal, state, and local statutes will be complied with, depending on site locations.

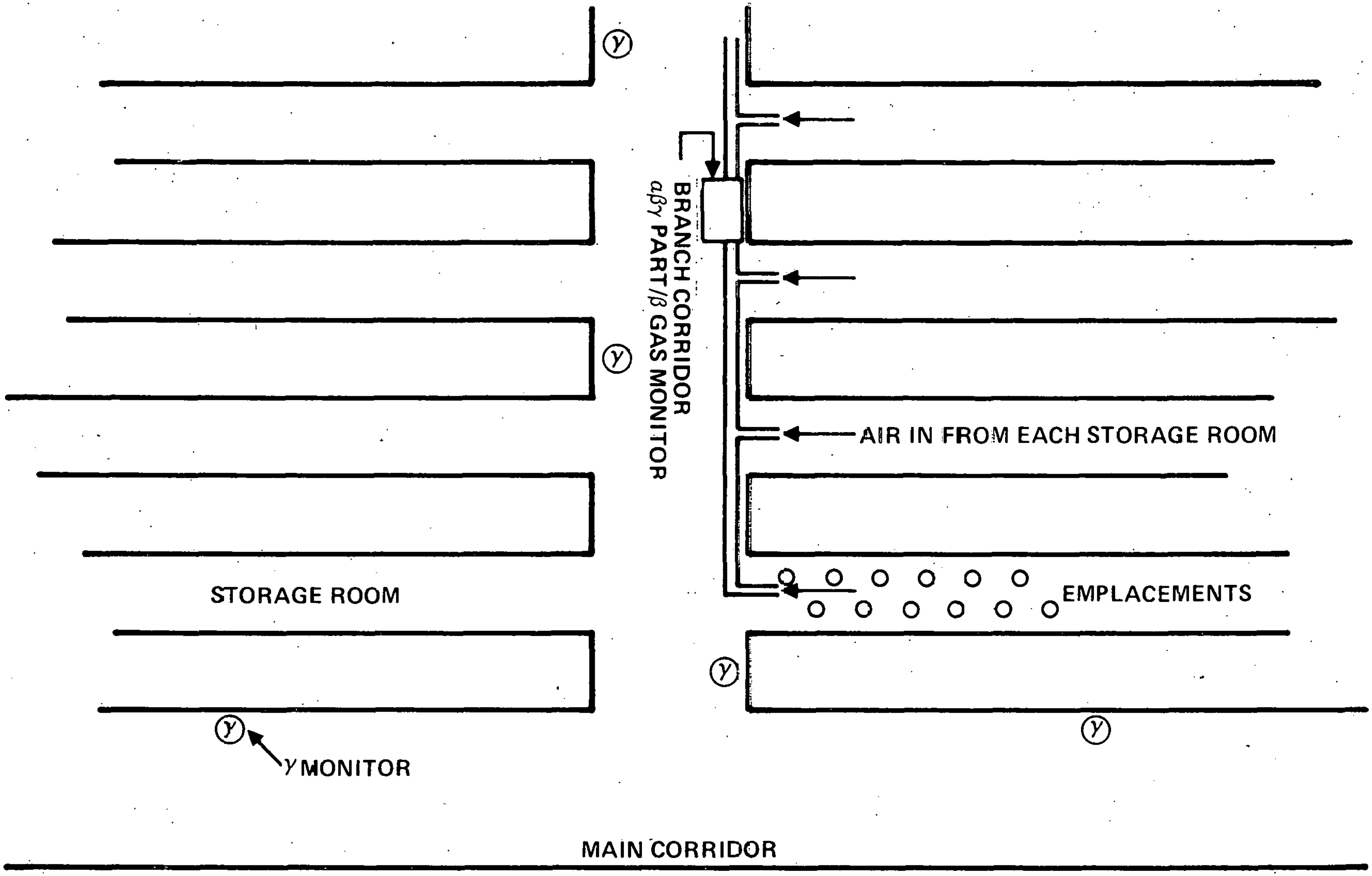
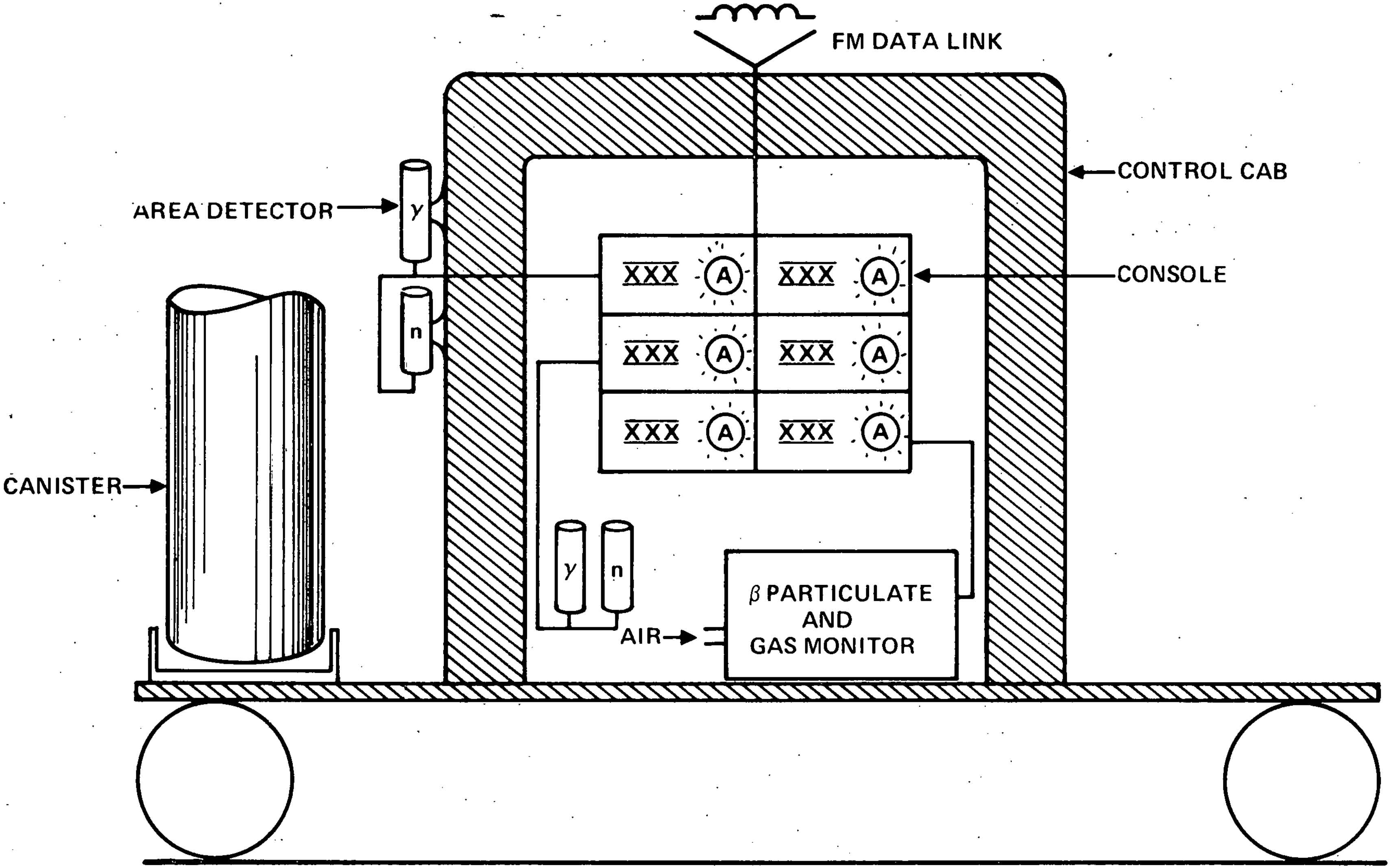


Figure 4-1. LOCATING UNDERGROUND RADIATION MONITORS



THE SAME READING WILL BE DUPLICATED ON ELEVATOR, IF ELEVATOR IS MANNED.

Figure 4-2. CONTINUOUS MONITORING OF UNDERGROUND CANISTER TRANSPORT VEHICLE

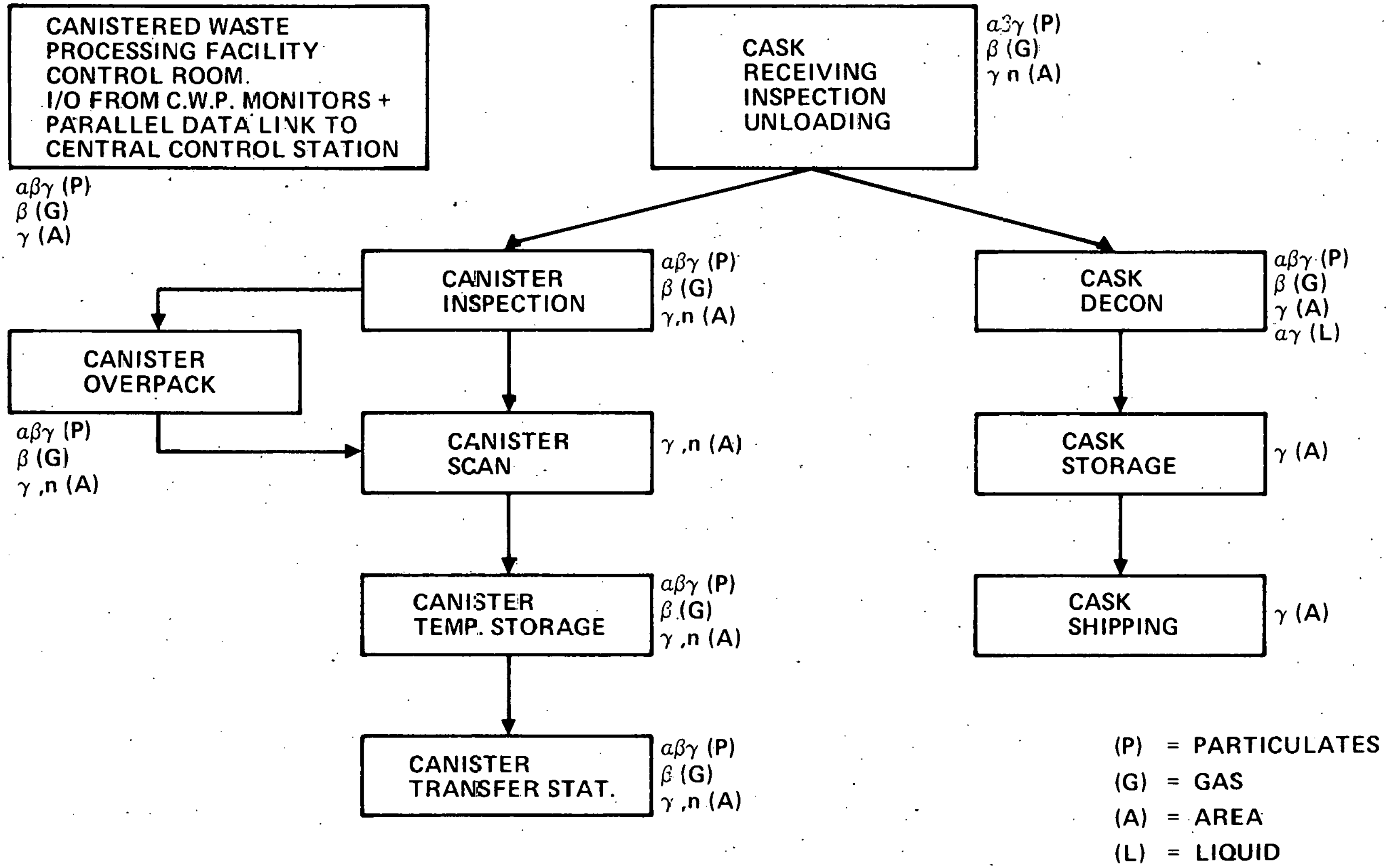


Figure 4-3. MONITORING CANISTERED WASTE PROCESSING STATIONS

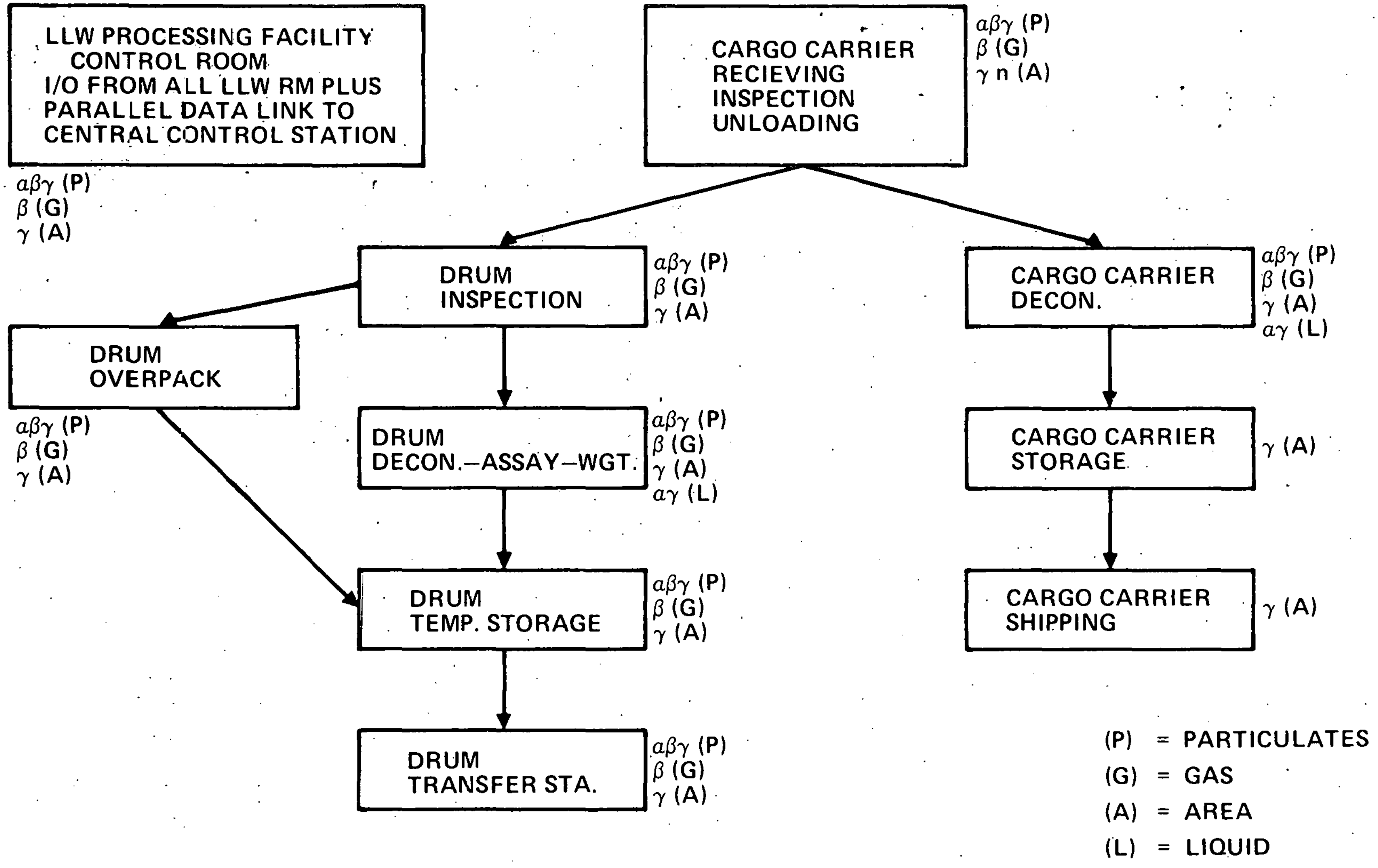


Figure 4-4. MONITORING LOW-LEVEL-WASTE PROCESSING STATIONS

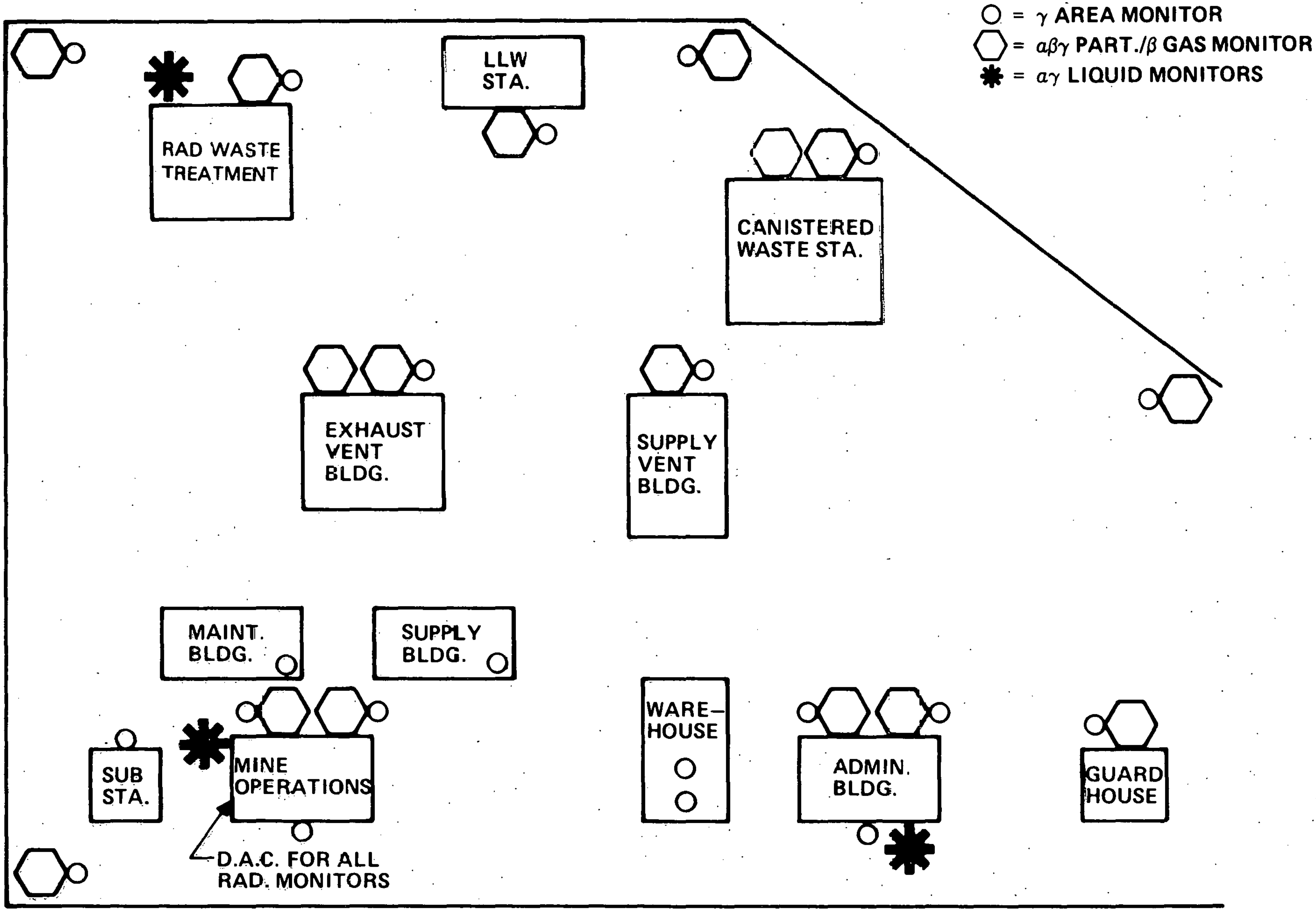


Figure 4-5. MONITORING CENTRAL FACILITY

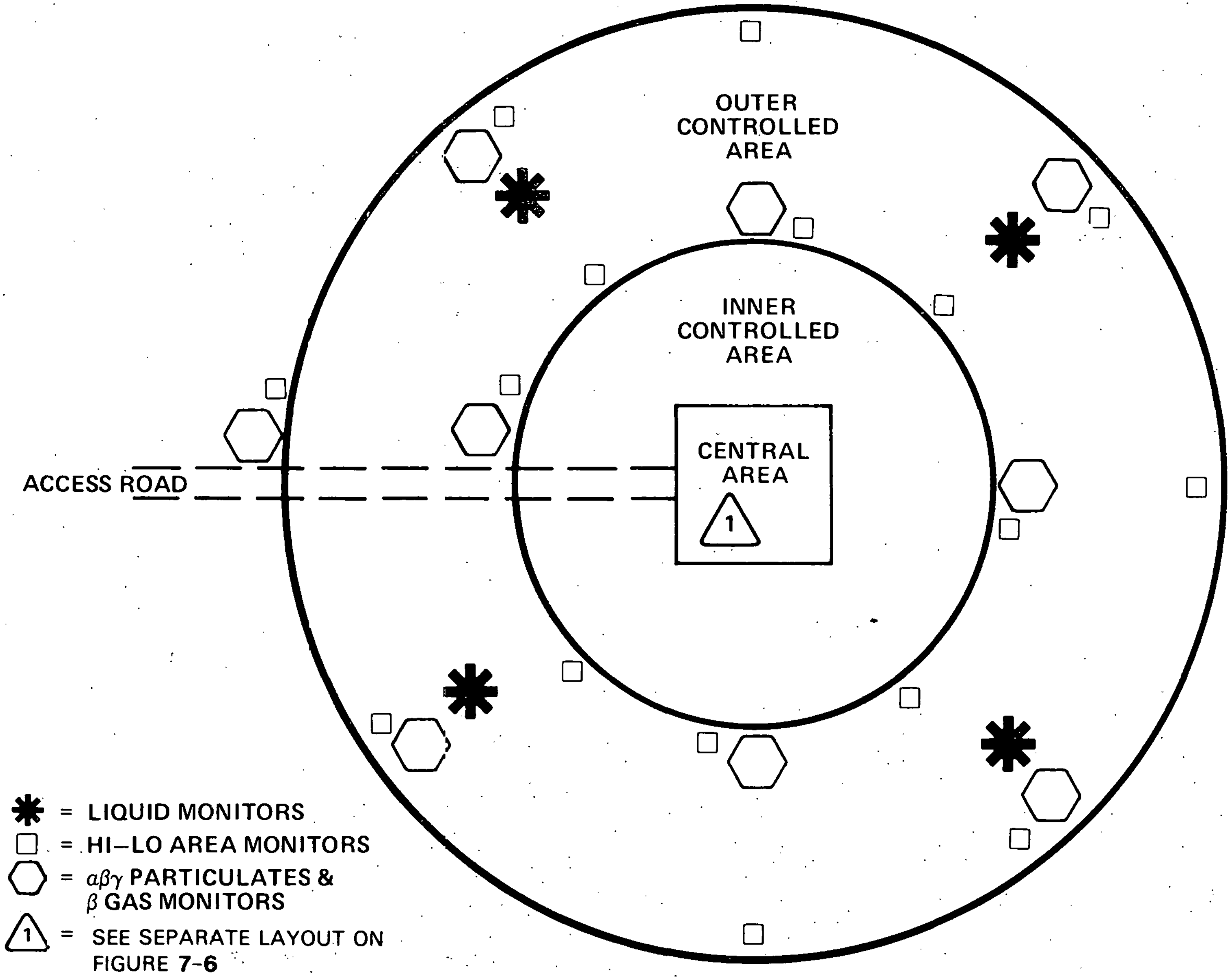


Figure 4-6. MONITORING INNER & OUTER CONTROLLED AREA

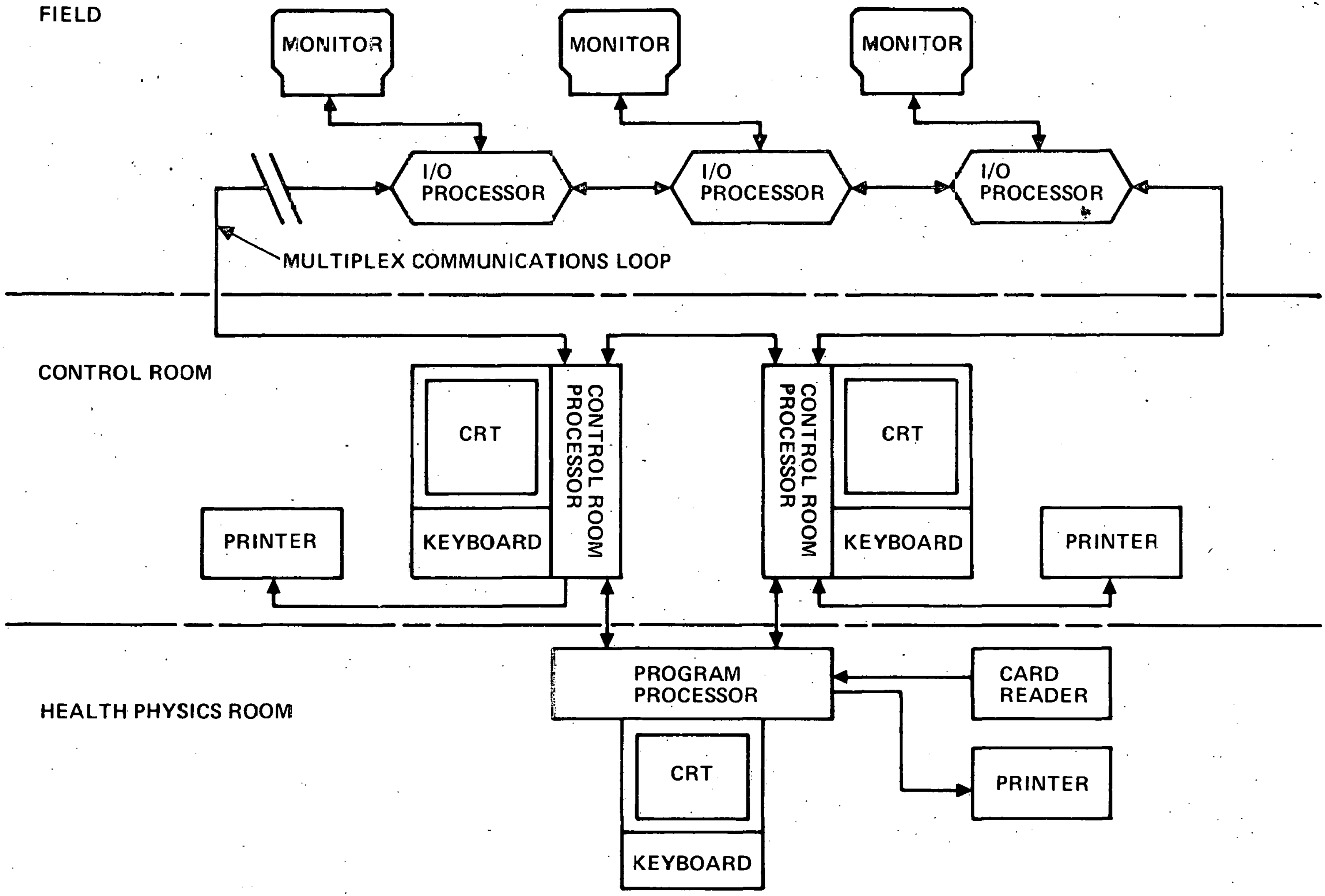


Figure 4-7. DATA ACQUISITION AND CONTROL SYSTEM (DAC)

TABLE 4-1

PRELIMINARY ESTIMATES OF SENSITIVITY AND RANGE FOR ON-LINE RADIATION MONITORS

Conditions

External backgrounds; 1.0 mR/hr with energy equivalent to Co-60 from all directions.

Internal Background; instruments characteristics.

Sensitivity; 95% confidence level with maximum of one minute counting time.

Accuracy; ±40% over entire dynamic range.

Particulates: Maximum of one hour collection time of ≥ 0.3 μ particulates.

Nuclide	Lower end of range (μCi/cc)	Upper end of range (μCi/cc)$^{(1)}$
Particulates (β channels)		
Sr-89	1.2×10^{-11}	10^4
Sr-90	1.6×10^{-11}	10^4
Co-58	8.0×10^{-11}	10^4
Co-60	3.2×10^{-11}	10^4
Cs-134	1.6×10^{-11}	10^4
Rb-88	8.0×10^{-11}	10^4
Particulates (α channels)		
Ra-226	8×10^{-12}	10^3
Pu-239	8×10^{-12}	10^3
Po-210	8×10^{-12}	10^3
Ra-224	8×10^{-12}	10^3

TABLE 4-1 (Continued)

Particulates (γ channels)		
Co-60	2×10^{-10}	10^{4}
Cs-137	2×10^{-10}	10^{4}
Ba-133	2×10^{-10}	10^{4}
Kr-85	2×10^{-10}	10^{4}
Gases		
Xe-133	5×10^{-7}	10^{5}
Xe-138	2×10^{-7}	10^{5}
Kr-85	3×10^{-7}	10^{5}
Kr-85m	2×10^{-7}	10^{5}
Kr-87	2×10^{-7}	10^{5}
Kr-88	2×10^{-7}	10^{5}
A-41	2×10^{-7}	10^{5}
Liquids		
Cs-134	2×10^{-7}	10^{5}
Cs-137	4×10^{-7}	10^{5}
Ba-140	6×10^{-7}	10^{5}
Mo-99	1×10^{-7}	10^{5}
Co-60	3×10^{7}	10^{5}
Co-58	9×10^{-7}	10^{5}
Fe-59	5×10^{-7}	10^{5}
Pu 241	2×10^{-4}	10^{2}
Cm-244	7×10^{-6}	10
Am-244	4×10^{-6}	10

Area (direct γ, 80 Kev to 3.5 Mev, 4π)

Plant Area - 10^{-1} mR/hr to 10^{7} mR/hr

Perimeter - 10^{-1} μR/hr to 10^{4} mR/hr

(1) Instruments will remain at full scale reading at levels 10 times maximum range specified.

TABLE 4.2

GENERALLY APPLICABLE REGULATORY GUIDES

Regulatory Guide	Title
1.16	"Reporting of Operational Information Appendix A Technical Specifications," Rev. 5, August 1975
1.21	"Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactivity Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Rev. 1, June 1974
1.23	"On-Site Meteorological Programs" (Safety Guide 23, February 17, 1972)
1.97	"Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant Conditions During and Following an Accident," December 1975
1.105	"Instrument Spans and Setpoints," November 1975
1.109	"Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, APP, I," March 1976
1.111	"Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," March 1976 Y14H
4.1	"Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants," Rev. 1, April 1975
4.2	"Preparation of Environmental Reports for Nuclear Power Stations," Rev. 1, January 1975
4.5	"Measurements of Radionuclides in the Environment-Sampling and Analysis of Plutonium in Soil," May 1974
4.6	"Measurements of Radionuclides in the Environment-Strontium-89 and Strontium-90 Analyses," May 1974
8.8	"Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as is Reasonably Achievable," Rev. 2, March 1977 Y20H

5.0 RADIOACTIVE MINE WATER TREATMENT FOR GRANITE, SHALE, AND BASALT REPOSITORIES

5.1 INTRODUCTION

Because of the geologic characteristics of granite, shale, and basalt, repositories built in these formations have a potential to allow water to enter the mine by permeation through the layers of rock constituting the formation. The water may enter the repository as a result of permeation either downward through the overlaying strata or by upflow through the lower layers. Since this water must be removed from the mine area in order to perform the normal underground activities of transportation and emplacement of the waste canisters, waste repositories in granite, shale, or basalt formations must include provisions for handling this water and for its possible treatment before discharge.* The chief concern in this section is that the water may become contaminated with radioactive isotopes before being pumped out of the mines and, hence, could require radwaste treatment before discharge. This section considers only the radioactive contamination of mine water and does not address the issue of contamination by dissolved solids.

5.1.1 Quantity of Mine Water to be Treated

The quantity of mine water that may enter a repository is dependent upon the rock form and upon the type of geologic strata that surround the mined areas. For the purposes of this section, the quantity of water inflow was calculated from generic geologic strata for each rock formation as described in Reference 1. From Reference 1, a treatment system for the mine effluent water should have a capacity of about 5,200,000 gallons of water per day if the repository were to be located in a shale bed and a capacity of about 400,000 gallons per day if the repository were located in either basalt or granite formations.

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*For salt, water permeation is an extremely unlikely event (Class 8 or 9) and a routine water treatment system is, therefore, not appropriate.

These quantities of water to be treated neglect any possible measures that could be taken within the mine to prevent mixing of downflow and upflow waters. If the downflow water, which would not be contaminated upon entering the mine as the radioactive materials are to be stored in or on the mine floor, could be prevented from mixing with the upflow water, then the water treatment plant size could be reduced to about 2.9 x 10^6 gallons per day for locations in shale and to about 50,000 gallons per day for locations in basalt or granite.

5.1.2 Radioactive Contamination Potential

The principle means of water contamination is water coming in contact with high- and intermediate-level waste canisters placed in holes in the mine floor. This water could then become contaminated with the transferable radionuclides on the canister surfaces. The maximum amounts of transferrable radioisotopes on the outside of the canisters are given in Section 1.6.1. The amount of water that could contact the canisters depends on many factors such as the water flow rate, temperature in the vicinity of the canisters (particularly for the higher heat producing canisters), and the condition of the barriers between the rock and the waste. (For canister and sleeve integrity, see Section 6.) During the retrievable phase of repository operations, the holes will be lined with carbon steel sleeves that could serve as barriers to prevent water from contacting the canister surfaces as long as the sleeves remain intact. However, it is likely that some sleeves will not remain water-tight throughout the operational phase of the repository (~30 to 40 years), and, further, any canisters emplaced after the retrievable phase will be placed into holes without the attendant sleeves. Therefore, groundwater will contact at least some of the canisters during repository operations and some of the canister surface contamination will enter the water. Additionally, the possibility exists that some of the containers may corrode or otherwise be damaged sufficiently to allow water to contact the waste itself. However, because the solidified waste forms are fairly impervious to leaching (see OWI/TM-36-21) and because most canisters should remain intact through repository operations (see Section 6), the potential for this mechanism of contamination of mine water is not considered to be as great as by direct contamination by external transferable nuclides.

5.2 BASIS FOR THE WATER TREATMENT SYSTEM

The primary factors to be considered in the design of a water treatment processing system are the required purity of the effluent water, the types and quantities of the contaminants in the source water and the quantity of water to be treated. The use of these factors to obtain an efficient, cost-effective treatment system is discussed in the following sections.

5.2.1 Required Decontamination Factor of the Treatment System

In order to determine the need for a treatment system and the required decontamination factor (DF) if a system was required, a bounding calculation was performed in the following manner. First, it was assumed that each canister arriving at the repository has the maximum allowable transferable radionuclide (β and γ)concentration of 10,000 dpm/100 cm^2. It was further assumed that all the transferable isotopes are transferred to mine water in the first year that they are emplaced in the repository. For the maximum number of HLW, ILW and CW canisters emplaced in a granite repository in 1 year (40,000), the total assumed activity transferred is then:

$$C_{transferred} = (40{,}000 \text{ canisters/yr})(3.1 \times 10^4 \text{ cm}^2\text{/canister})(100 \text{ dpm/cm}^2)$$

$$C_{transferred} = 1.24 \times 10^{11} \text{ dpm/yr} = 0.056 \text{ Ci/yr} = 56{,}000 \ \mu\text{Ci/yr}$$

Using the granite mined water pumping rate of 400,000 gpd,[1] and allowing for a factor of 10 to account for uncertainties and the peak/average contamination ratio, yields:

$$C = \frac{(56{,}000 \ \mu\text{Ci/yr})(10)}{(400{,}000 \text{ gal/day})(3{,}785 \text{ cm}^3\text{/gal})(365 \text{ day/yr})} = 1 \times 10^{-6} \ \mu\text{Ci/cm}^3$$

Therefore, to meet the 10CFR20 and EPA National Interim Primary Drinking Water Regulation (EPA-570/9-76-003) of about 10^{-8} or 10^{-9} $\mu Ci/cm^3$, a treatment system with a DF of 10^3 would be required.

5.2.2 Need for Pretreatment of the Mine Water

The mine water may contain significant amounts of non-radioactive ionic contaminants including Ca and Mg. In order for any final treatment to be cost effective in reducing the concentrations of the radionuclides to the range required by the NRC and EPA standards, most of the Ca and Mg must be removed before final treatment can be undertaken, since sufficient quantities of these contaminants can require frequent regeneration of the ion-exchange medium.

In general, the capacity of a system for removal of radioactivity is inversely proportional to the amount of non-radioactive cations present. For this reason, an effective pretreatment to reduce Ca and Mg concentrations is necessary regardless of the final processes selected. The choice of a pretreatment system will be site specific and dependent primarily on the concentrations of Ca and Mg ions to be removed. Normally these cations are referred to as "the hardness" and are present in water as the bicarbonates $Ca(HCO_3)_2$ and $Mg(HCO_3)_2$. A standard treatment using lime, $Ca(OH)_2$, to precipitate these cations as $CaCO_3$ and $Mg(OH)_2$ followed by filtration may be necessary if the total hardness is in excess of 2500 ppm. If the hardness is less than 2500 ppm, then ion exchange may be a better choice for removing these cations.

The lime softening process has the advantage that no anions are added during the softening process that would need to be removed in further demineralization processes. The molecular equations for the addition of lime as a softening agent are as follows:

$$Ca(HCO_3)_2 + Ca(OH)_2 = 2CaCO_3 + 2H_2O$$
$$Mg(HCO_3)_2 + 2Ca(OH_2) = 2CaCO_3 + Mg(OH)_2 + H_2O$$

The disadvantage of this process is that several of the radionuclides that may be present as contaminants are also divalent cations and will also be removed, at least in part. The precipitate formed may then be sufficiently contaminated that it would be classified as low-level waste and further treatment would then be necessary to immobilize the waste for disposal.

5.2.3 Selection of the Water Treatment System

The ion-exchange process has been selected as the generic process for treating the mine water for the following reasons:

- The process has been used extensively in large scale installations for removal of radionuclides.
- Decontamination factors for various specific radionuclides have been determined under a wide variety of operating conditions.
- It is anticipated that the water treatment plant may be utilized on infrequent occasions. Because ion exchangers are operated on an intermittent basis (a service run followed by bed regeneration), the process equipment can be readily put into a standby condition.
- The decontamination factors for radionuclides (in the absence of competing non-radioactive ionic contaminants) are on the order of 10^3 to 10^4 for a single process unit.[2,3]
- Ion exchange units can be easily arranged in series to provide multiple stages of decontamination if required.

Appendix C discusses the principle of ion-exchange-process removal of radionuclide contaminants from water.

5.3 DESCRIPTION OF THE MINE WATER TREATMENT SYSTEM

The water-treatment system for removing radionuclides from mine water is shown schematically in Figure 5-1. This system is sufficient to demonstrate that a water-treatment facility could be designed to reduce radionuclide concentration to satisfy the NRC requirements and the EPA requirements for potable water. There is no attempt to determine the relative cost effectiveness of this system relative to other systems that could also be constructed for the same purpose. The basic elements of the system are a divided sampling and holding pond, an optional pretreatment process to remove hardness (if necessary), a cation exchanger, and an anion exchanger. This system would have a DF of 10^3. In addition, there is a line connecting the discharge line to the holding ponds so that the discharged water can be re-treated if the radionuclide concentrations were to exceed the required NRC and EPA standards.

5.3.1 Sampling and Holding Ponds

The water would be discharged from the mine into a divided sampling and holding pond with sufficient capacity to hold 4-6 daily water volumes. For 5 daily volumes, a pond of approximately 80 acre-feet for a shale repository and approximately 6 acre-feet for granite and basalt would be required. The pond sections would be filled, sampled, and discharged alternately. As discussed in Section 5.1, it is anticipated that the mine water would not generally be contaminated with radionuclides and, after sampling, the water could then be discharged offsite without radwaste treatment. If the water were found to be contaminated, it would then be routed to the treatment plant.

5.3.2 Pretreatment

If the mine water should contain Ca and Mg in sufficient quantities to require frequent regeneration or replacement of the cation exchange bed, the water would be treated in the pretreatment section. This section would consist of a lime addition and filtration unit which would serve to remove gross quantities of Ca and Mg. Some decontamination of radionuclides would also occur during this process; however, no decontamination factor is assumed for this step because, at some possible repository sites, the process would not be necessary. If sufficient quantities of radionuclides are removed during this process, the precipitate may have to be treated as LLW.

5.3.3 Cation Exchanger

The cation exchanger would consist of several parallel units arranged such that a unit could be regenerated while the other units were in service. This feature would permit continuous operation of the treatment facility. The size and the choice of ion exchange medium for the cation-exchange bed will be dependent on the total flow rate and the concentration of Ca and Mg cations present in the water. Because the water discharged from the treatment system must be essentially devoid of cations, the hydrogen form of cation exchange resins is recommended for all of the cation-exchange units in the system. The H^+ ion is the least preferred cation in the series of cations, and, therefore, all other cations will exchange with the H^+ form of exchange medium. In this manner, the maximum cation exchange can occur.

The capacity of the resin for exchanging cations is usually expressed as kilograins of $CaCO_3$ per cubic foot, kgr/ft^3*. Monovalent and trivalent cations can be readily converted to the $CaCO_3$ base by means of their respective equivalent weights. A hydrogen form of cation exchanger can generally exchange 10 to 20 kgr/ft^3 before regeneration is necessary.[2] The capacity range is a function of secondary variables such as the pH and temperature of the water. When the $CaCO_3$ concentration is known, the capacity of the ion-exchange beds can be readily calculated.

The size of the cation-exchange units is dependent on the water flow rate. Generally, to reduce breakdown and attrition of the cation-exchange bed, flow rates of 15 gal/min ft^3 have been used successfully.[2]

5.3.4 Anion Exchanger

The basic design parameters affecting the selection of anion exchange media and the size of the units are similar to those for the cation exchangers. It is recommended that the strong form of the anion exchange medium be used in order to reduce ion leakage to the lowest practical level. Strong-base anion exchangers have capacities on the order of 10 to 20 kgr/ft^3 expressed in terms of $CaCO_3$.[2] When the total flow and the concentration of $CaCO_3$ are known, the size and capacity of the anion-exchange bed can be calculated.

---

*For $CaCO_3$, kgr/ft^3 can be obtained by dividing the ppm by 17.1

5.4 ESTIMATE OF CAPITAL AND OPERATING COSTS

The cost for the water-treatment systems can be estimated by using References 4 through 9 with the following assumptions:

1. The reference system can be constructed using elements similar to those given in References 4-9.
2. The costs given in the references for an ion-exchange system are for typical units represented by the primary cation-and anion-exchange units in the reference plant.
3. The costs will not be strictly additive, as each unit is not a complete "stand alone" unit but will share facilities such as cold chemical treatment and regenerating systems.
4. The ion-exchange units will be preceded by a pretreatment step. The cost of a lime-addition pretreatment system is assumed to be the same as a rapid sand-filtration system.
5. An approximate capital cost for the facility can be estimated by totaling the capital costs for each of the units and employing the factors in Table 5-1 for those units that are not completely "stand alone" components.
6. The cost data in Table 5-2 are given in 1973 dollars. An escalation factor of about 40 percent must be applied to bring the costs up to 1978 dollars.
7. The operating cost of the plant can be determined by using a similar rationale based on the data given in Table 5-3.

From Table 5-2, the cost of a 530,000 gallon-per-day treatment system is $590,000 (in 1973 dollars). Values for a 400,000 gpd and a 5,200,000 gpd plant obtained by the same method are $565,000 and $2,800,000, respectively (in 1973 dollars).

5. REFERENCES

1. Y/OWI/TM-36/21, "Ground Water Movement and Nuclide Migration."
2. "Ion Exchange Technology", edited by F. Z. Nachod and J. C. Shubert, Academic Press, 1956.
3. Applebaum, S. B., "Demineralization by Ion-Exchange in Water Treatment and Chemical Processing of Other Liquids," Academic Press, 1968.
4. "Monograph of the Effectiveness and Cost of Water Treatment Processes for the Removal of Specific Contaminants", Volume I, Technical Manual. David Volhert and Associates. PB-242-442, Prepared for the Environmental Protection Agency, August 1974. Figure RF-2.
5. Reference 2, Figure RF-5.
6. "Desalting Handbook for Planners", Office of Saline Water and the Bureau of Reclamation, Department of the Interior, May, 1972, p. 7-80.
7. Reference 4, p. 7-76.
8. Reference 4, p. 7-85.
9. Reference 4, p. 7-49.

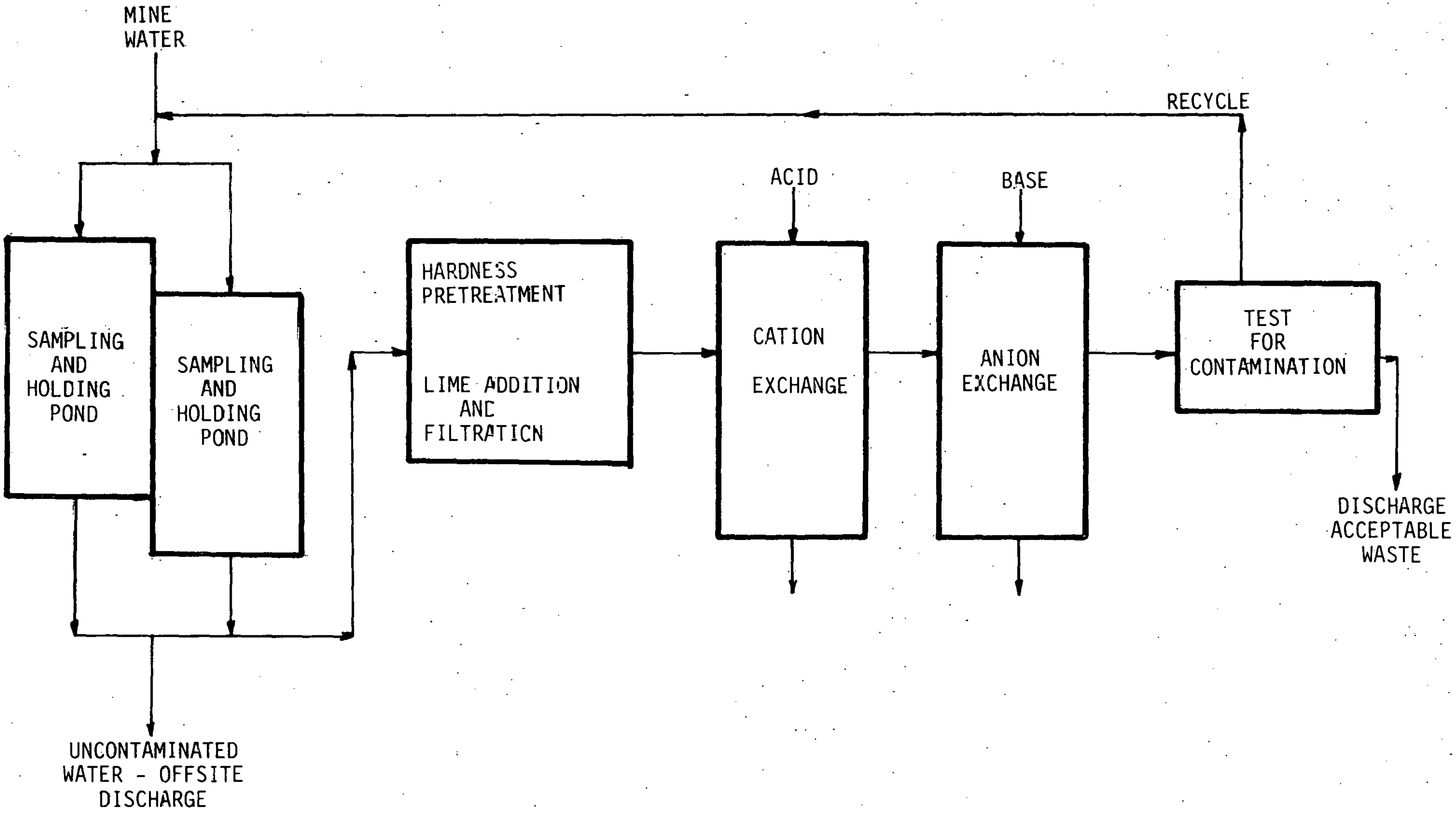

Figure 5-1 SCHEMATIC FLOW DIAGRAM
WATER TREATMENT SYSTEM - REMOVAL OF RADIONUCLIDE

TABLE 5-1

CAPITAL COST OF THE REFERENCE WATER TREATMENT PLANT

Factors for Application to EPA Capital Cost Monographs		
Reference Plant Component	Monograph Figure Description	Monograph Factor
1. Holding and Sampling Pond	No Applicable Figure	Calculate Separately
2. Pretreatment (Hardness Removal)	Construction Cost for Rapid Sand Filter - Reference 5-4	100%
3. Cation Exchange Unit	Ion Exchanger Plant Cost - Reference 5-9	80%
4. Anion Exchange Unit		60%

TABLE 5-2

BREAKDOWN OF CAPITAL COST* FOR 530,000 gpd PLANT

Reference Plant Component	Monograph Cost $1000	Factor	Capital Cost* $1000
1. Holding and Sampling Pond	---	----	Separate Est.
2. Pretreatment ($10m^3/m^2$-hr)	100	100%	100
3. Cation Exchange (1000 ppm red in TDS)	350	80	280
4. Anion Exchange	350	60	210
TOTAL (less cost of pond) ----------------------------------			$590

*In 1973 dollars.

TABLE 5-3

OPERATING COSTS FOR THE REFERENCE WATER TREATMENT PLANT

Factors for Application to EPA Operating Cost Monographs		
Reference Plant Component	Monograph Figure Description	Monograph Factor
1. Holding and Sampling Pond	No Applicable Figure	Calculate Separately
2. Pretreatment (Hardness Removal)	Operation and Maintenance for Rapid Filters	100%
3. Cation-Exchange Unit	Approximate Regenerate Cost for IX System - Reference 5-6	100%
4. Anion-Exchange Unit		80%

6.0 WASTE CONTAINER INTEGRITY

6.1 INTRODUCTION

Containment of the radioactive wastes by the canisters or drums in which they are packaged is essential to smooth repository operations. A breaching of one of these containers while it is being handled within the repository leads to operational difficulties and possibly to undesirable releases to the environment.[1] Furthermore, the baseline repository design assumed a period during which the stored canisters must remain in a readily retrievable state. Although various techniques for waste retrieval have been analyzed in other studies,[2] the preconceptual GEIS designs are based on a requirement of canister integrity throughout the retrievability period.

In addition to this retrievability requirement, analysis of potential nuclide transport from the repository after decommissioning of the facilities[3] indicates that in the case of the hard-rock (granite, shale or basalt) repositories it could be desirable to maintain a more extended period of canister integrity. This extended period of integrity allows more time for radioactive decay of some of the more soluble isotopes, thus reducing potential concentrations that could be transported to the biosphere by groundwater circulation through the repository region. This facet of the canister-integrity issue is not as important for the repository designs in salt since groundwater does not flow through these formations and flooding of a salt repository is an extremely unlikely event (Class 8 or Class 9).

Since the assurance of retrievability is the driving consideration for canister integrity in salt, the canister-integrity requirements are dictated by the length of the retrievability period. If retrievability is not required or the retrievability phase of repository operation has been terminated, canister integrity in salt is not required. However, in more detailed repository designs in the hard rocks it may be desirable to consider canister-integrity requirements for longer periods of time.

The purpose of this section is to discuss the primary sources of canister stresses (6.2), to discuss the types of canister corrosion, and to estimate the severity of corrosion as it would impact canister retrievability. The discussion on corrosion is divided into a discussion of HLW stainless steel corrosion (6.3) and the corrosion of carbon steel drums in air (6.4). ILW stainless steel canisters are not discussed, since the HLW canister corrosion rates will be more severe as a result of higher heat generation rates and temperatures. The canisters and drums analyzed in this section are described in Section 1.0, Baseline Design Considerations. Heat transfer properties were obtained from OWI/TM-36/19, Thermal Analysis Document.

6.2 SOURCES OF STRESSES

Canister failure occurs when the canister stresses exceed canister strength, either on a micro- or a macro-scale. This section discusses the sources of canister stresses, while Section 5.3 discusses the loss of canister strength by corrosion.

The important canister stresses are the waste solidification stresses and stresses caused by internal pressures. Fabrication stresses (canister closure) could cause high localized stresses. Other stresses that might be of interest include residual stresses, fabrication stresses, dead-load stresses, and external stresses.

6.2.1 Waste-Solidification Stresses

Waste-solidification stresses are primarily a concern with glass waste, as a result of the higher melting temperatures, and are the results of loading the glass-waste material into the stainless steel canisters. The two most promising methods for glass-waste solidification are in-can melting (ICM) and joule-heating with a ceramic melter (JHCM).

6.2.1.1 In-Can Melting (ICM)

The ICM process uses the canister as a crucible to melt the HLW and the glass formers[4] at temperatures of about 1,050°C (1920°F).[4] After the heaters have been turned off the melted zone is forced-gas cooled to 900°C (1650°F) until the canister is entirely filled. The filled canisters will be forced-gas quenched from 900°C (1650°F) to 400°C (750°F) and then placed in water at 90°C (175°F). Since the thermal contraction of stainless steel is greater than that of glass, the cooling of the canister results in the development of circumferential tensile stresses in the canister. A correlation has been developed in reference (5) using the properties of 304 stainless steel and commercial grade borosilicate glass.

For a cylinder diameter of 1.0 ft, the stress S_s in psi is given by:

$$S_s = 214\ \Delta T \qquad (6.1)$$

where: S_s = circumferential tensile strength, psi
ΔT = temperature decrease during cooling of glass from original high temperature at which glass is considered to be rigid, °F.

This correlation indicates that a tensile stress of 21,400 psi develops in the steel for every 100°F of cooling. Therefore, the ICM process can develop large circumferential tensile stresses in the canister and some form of stress relief (such as subcooling or shot-peening) may be desirable.

6.2.1.2 Joule-Heating with a Ceramic Melter (JHCM)

In the JHCM process, the HLW and glass formers are melted out of the can at 1150°C (2100°F), and equilibrated and homogenized before being poured into the canister.[4] The canister is at ambient air temperature when the pouring begins and is cooled with forced gas during filling to keep the maximum temperature below the devitrification temperature. The maximum canister surface temperatures will probably be about 200°C (390°).[4]

The potential for developing large canister stresses is not as great for the JHCM process as it is for the ICM process, since the canister temperatures are maintained at much lower values.

6.2.2 Internal Pressures

The internal gas pressure in the HLW glass canisters can be neglected since the glass will be prepared by high-temperature melting and all volatile gases should be removed during this operation. The only free oxides in glass are silicates, which are non-volatile. Helium will be formed by the alpha decay of actinides. However, the pressure increase due to helium buildup is not expected to be significant. Reference (6) states that a 15 psig pressure build-up is reached in a few years for PuO_2 fuels and in several hundred years for UO_2. Therefore, internal gas pressurization of glass HLW canisters is not expected to be significant with respect to the strength of the canisters.

The main concern in the calcine storage will be the pressurization of canisters due to water evaporation and due to nitrate decomposition by temperature or by radiation. The average calcine from the Idaho Chemical Processing Plant has 1 percent H_2O and 1 percent N_2O_5 immediately after calcination.[5] The pressure due to the release of these compounds in decomposition is a function of temperature and is expressed by the formula:[5]

$$P(\text{atm}) = 9.26\ T \left[\frac{(1-y)x}{y}\right] + 18.23\ T \left[\frac{(1-y)z}{y}\right] \qquad (6.2)$$

where

T = temperature of the gas space, °K

y = void fraction of canister

x = weight fraction of nitrate in calcine

z = weight fraction of water in calcine

In the above expression for P(atm), the first term is the pressure due to nitrate decomposition and the second term is due to dehydration. For calcine with 1 percent H_2O and N_2O_5, the canister will be pressured to about 50 atm at 200°C and the internal pressure could stress the canister beyond its yield strength for temperatures above 800°C. To prevent this, the content of H_2O and N_2O_5 in calcine could be lowered to 0.1 percent (weight percent) by preheating of all calcine to 800°C, or by heating of the open canister, filled with calcine (before sealing), to 800°C. With $y = 0.7$, and x and $z = 0.001$, Eq. 6.2 becomes:

$$P(\text{atm}) = 0.012T\ (^{\circ}\text{K}) \tag{6.3}$$

The internal pressure from H_2O and N_2O_5 would then be about 5.8 psig at 200°C and should not significantly overstress the canister.

6.3 CORROSION OF STAINLESS STEEL CANISTERS

The stainless steel canisters can be corroded internally by interactions between the waste form and the stainless steel, or externally by interactions between the canister and geologic medium. These methods of corrosion are discussed in this section.

Some external corrosion data was gathered in the Project Salt Vault for Schedule 40 carbon steel heater pipe in a wet-salt environment. This data are discussed in Appendix A.

6.3.1 Internal Corrosion

Internal corrosion of the canister will be chiefly a result of radiolytic gases (NO_2, O_2, halogens). For glasses, if the wall temperature is maintained below the glass melting temperature, no internal corrosive attack is expected.[3] Since the maximum wall temperature is maintained below 375°C (710°F), internal corrosion for glass canisters should be negligible. For calcines, corrosion could be observable as a result of NO_3 attack of stainless steel,[7] or as a result of Cs_2O attack of the stainless steel.[8] Corrosion by NO_3 and H_2O will probably be negligible for canister-wall temperatures between 60 and 300°C (140 and 570°F).[7] A formula for Cs_2O corrosion rate into stainless steel is:[8]

$$k(\text{cm/sec}) = (2.90 \pm 1.78) \times 10^{-3} \exp(-9500/T) \tag{6.4}$$

where T is in °K.

For the expected concentrations of Cs_2O in calcine HLW and the HLW temperatures, this is not expected to be a significant source of corrosion.

The internal corrosion of spent fuel inside the canisters will probably be negligible because no water or nitrate is present in spent fuel and the rods of spent fuel will generally be colder than the centerline of an HLW glass or calcine canister. No substantial migration of the fission products Cs or Rb is expected in spent fuel during the period of interest for canister integrity.

6.3.2 External Corrosion

The canister will be placed in a carbon steel sleeve throughout the retrievability period of the repository[9] regardless of the geologic medium of waste storage. This sleeve effectively isolates the stainless steel canister from the rock type throughout the retrievable phase of repository operations, and the corrosion of the canisters is the same as that in the air.*

Figure 6-1 shows corrosion data for steel in air.[10] Linearly extrapolating these data to a temperature of 300°C yields a corrosion rate of about 0.001 mils/year. Therefore, it is not expected that external corrosion of the stainless steel canisters will be significant for the duration of the retrievable portion of repository operation. The carbon steel sleeves will corrode at a rate of about 2 mils/yr based upon a linear extrapolation of Figure 6-1 at 300°C. However, this corrosion is not expected to be significant with respect to the structural requirements of the sleeve.

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*Stress-corrosion cracking (SCC) will not be a concern during the retrievable phase since, in general, an oxygenated aqueous solution is required for SCC. At the anticipated storage temperatures and pressures, any water reaching the canister should be a vapor. Reference 3 discusses corrosion after the retrievable phase is terminated.

6.4 CORROSION OF CARBON STEEL DRUMS

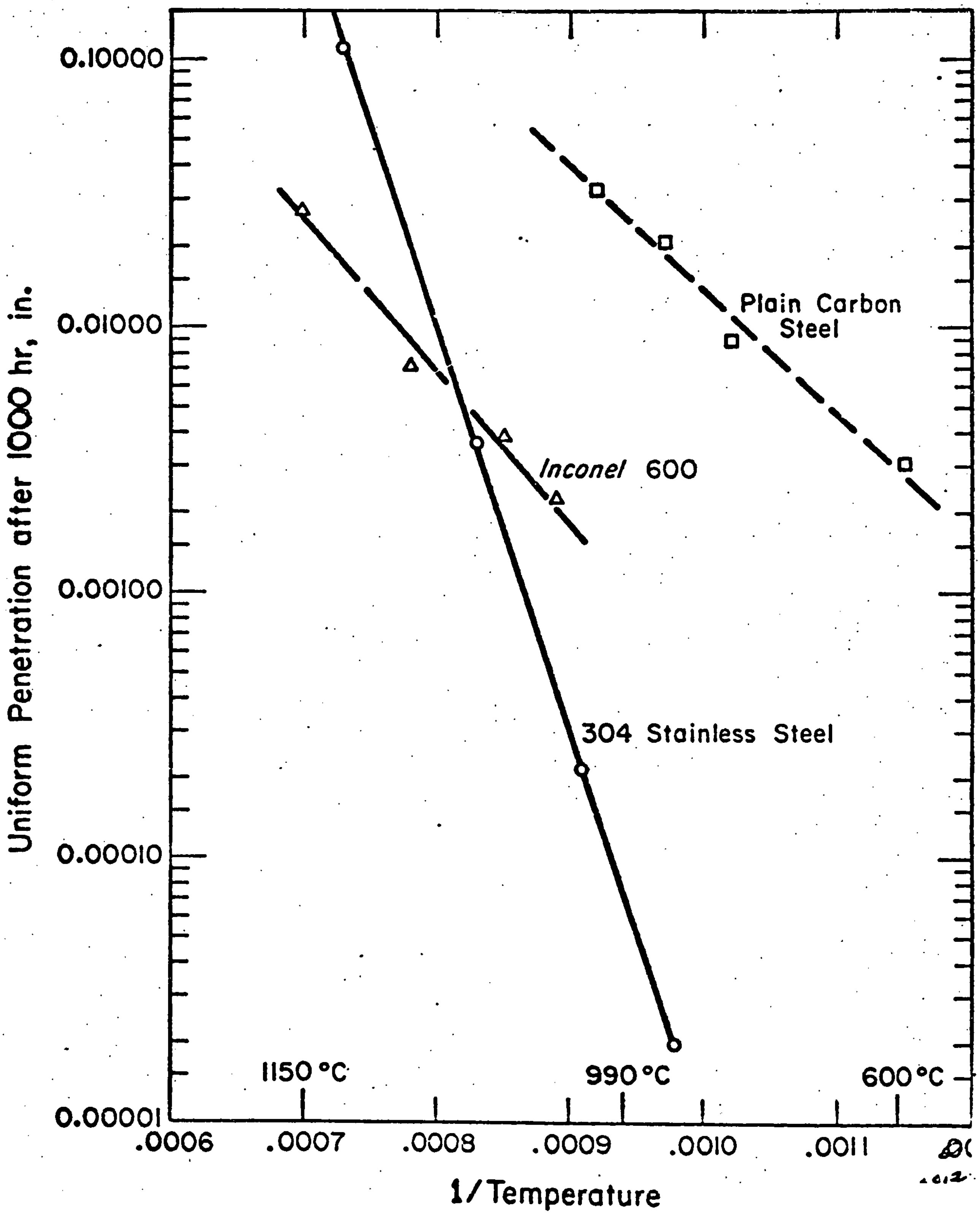
Drums containing LLW-TRU in the reference case and ILW from the alternative studies will have only about 3W of heat generation, and the maximum surface temperature of the drums will be less than 50°C. Therefore, in salt repositories, the drums should corrode similarly to carbon steel at an ocean site. Figure 6-2 shows a plot of atmospheric corrosion data at Kure Beach, N. C., for steel.[10] If the upper corrosion rate is used for carbon steel, the carbon steel drums would corrode to a maximum depth of 25 mils from the original 50 mils thickness at the end of the 5 year retrievability period. Since the carbon steel drums are not highly stressed, this corrosion rate is not expected to affect retrievability of the drums. The corrosion rates of carbon steel in granite and shale repositories are not expected to be as large, based upon a corrosion rate interpolated from Figure 6-1.

6.5 SUMMARY

Integrity of the HLW canisters is an important factor during the retrievability portion of repository operation. The possibility exists of creating large stresses in canisters containing HLW glass for the in-can melting process, but, if necessary, the stress values can be maintained to acceptable levels by stress relieving or by the use of joule-heating with a ceramic heater. Large internal pressures can be built up in calcined waste if the H_2O and N_2O_5 concentrations are not kept to 0.1 percent. Internal corrosion of stainless steel canisters by the HLW or by spent fuel is not expected to be significant. With the canister placed in carbon steel sleeves, the external corrosion will be limited to corrosion in air during the retrievable phase and should not be significant. Corrosion of carbon steel drums should have no significant effect on the retrievable phase of repository operation.

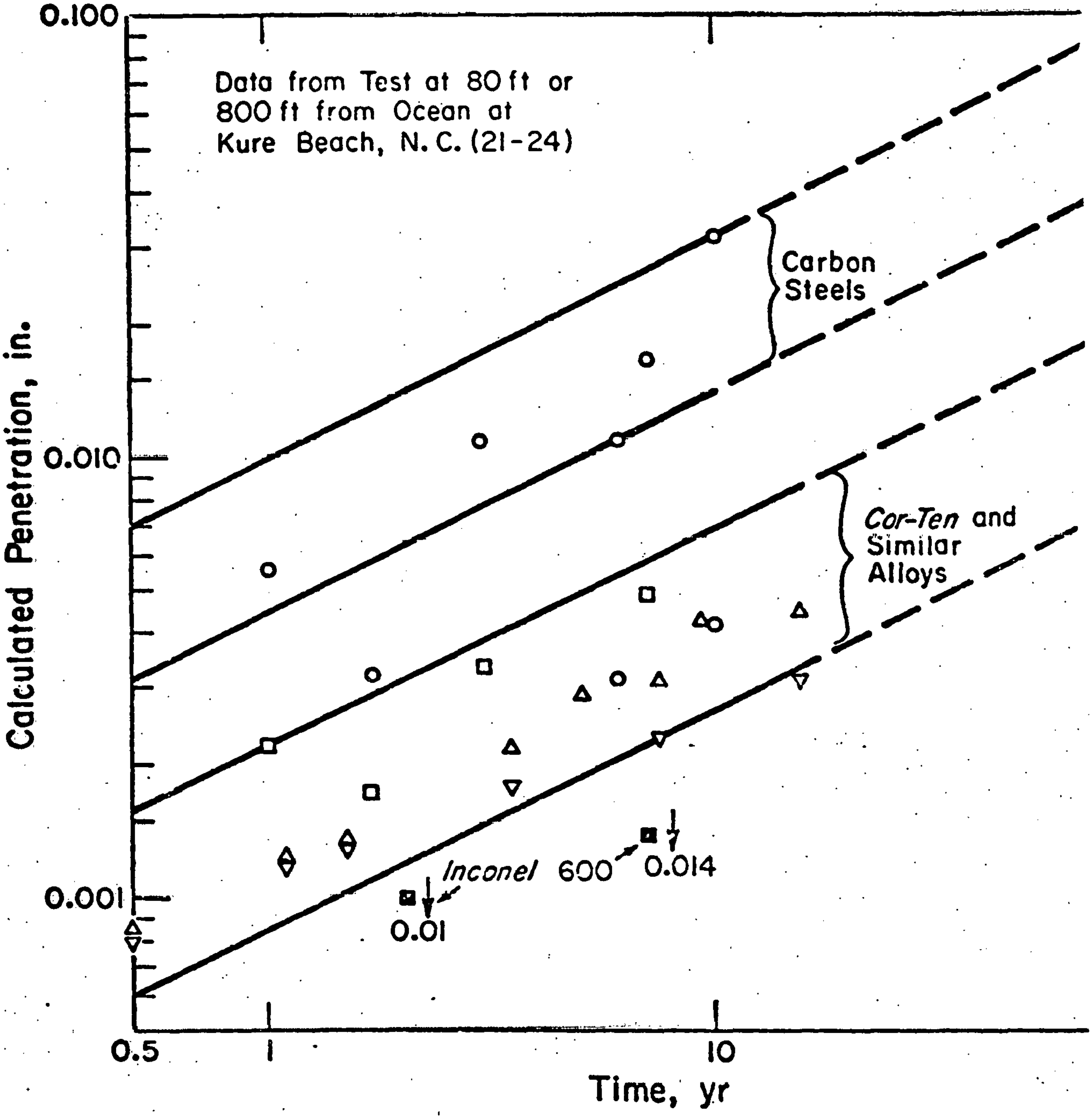
6. REFERENCES

1. OWI/TM-36/23, "Environmental Effluents Releases and Monitoring."
2. WIPP Conceptual Design Report, SAND77-0274, Sandia Laboratories, April, 1977.
3. OWI/TM-36/21, "Ground Water Movement and Nuclide Migration."
4. Mendel, J. F., et al. "Annual Report on the Characteristics of HLW Classes", BNWL-2252, June 1977, p 27.
5. Mecham, M. J., Seefelds, W. B., and Steindler, M. J.; "An Analysis of Factors Influencing the Reliability of Retrievable Storage Canisters for Containment of Solid High-Level Radioactive Waste", ANL-76-82, Argonne National Laboratory, August 1976.
6. Turcotte, R. P., "Radiation Effects in Solidified High-Level Wastes, Part 2 - Helium Behavior", BNWL-2051, Section 3.4, May 1976.
7. McElroy, J. L.; Quarterly Progress Report Research and Development Actinide Waste Fixation Program, October through December 1974, BPNL, February 1975.
8. Cohen, J. J., et al., "Determination of Performance Criteria for High Level Solidified Nuclear Waste, NUREG-0279, dated June, 1977, p. 42.
9. Y/OWI/TM-36-8, "Repository Preconceptual Design Studies: Salt."
10. Angeaman, C. L. and Rankin, W. N., "Durability of Containers for Storing Solidified Radioactive Waste," DP-MS-76-66, Savannah River Laboratory, South Carolina: Presented at: Corrosion/77 NACE Meeting, San Francisco, March 14-18, 1977.



Source: Angeaman, C. L. and Rankin, W. N., "Durability of Containers for Storing Solidified Radioactive Waste," DP-MS-76-66, Savannah River Laboratory, South Carolina: Presented at: Corrosion/77 NACE Meeting, San Francisco, March 14-18, 1977.

Figure 6-1. HIGH TEMPERATURE CORROSION OF STEEL



Source: Angeaman, C. L. and Rankin, W. N., "Durability of Containers for Storing Solidified Radioactive Waste," DP-MS-76-66, Savannah River Laboratory, South Carolina: Presented at: Corrosion/77 NACE Meeting, San Francisco, March 14-18, 1977.

Figure 6-2. ATMOSPHERIC CORROSION OF STEEL

7.0 CRITICALITY CALCULATIONS

7.1 VERIFICATION OF COMPUTATIONAL METHODS

As a matter of good practice in performing criticality calculations, the computational methods and cross-section sets used were validated according to the guidelines suggested in the proposed ANSI Standard N16.5-1974 entitled, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials." In particular, Section 4.2, paragraph 4.2.1 states, "Limits for the storage of fissile material shall be based on experimental data or the results of validated computational techniques." A validated computational technique is defined by N16.5 as, "A calculational method that has been tested, by comparison with experiment, to establish the reliability of results when the method is applied to conditions of interest."

A number of benchmark calculations were performed which followed the ANSI guidance. These involved the following analyses:

1. Calculations were performed using self-shielded cross-section data of the critical mass of plutonium-water mixtures (0.01-5 g/cc Pu) reflected by water. These calculations, performed with the ANISN code, were compared with data in ARCH-600 and in all cases a smaller critical mass was predicted. The average deviation between our results and these data was about 9 percent, with the differences at the minimum critical mass being about 3 percent.
2. These procedures were also checked against experiments for both bare PuO_2 and PuO_2 reflected by plexiglass. The experimental k_{eff} was 1.0 for all of the cases considered. These calculations were performed using the KENO Monte Carlo code with the same cross-section procedures used for the benchmark plutonium-water criticality calculations.

Cross - checking these results, where appropriate, with ANISN yielded values of k_{eff} within 2σ of those obtained with KENO. A summary of these results is presented in Table 7-1.

Based on the above comparisons, it is suggested that the SAI procedures and cross sections have been verified as described in the ANSI standard.

7.1.1 Details of Criticality Calculations

Criticality calculations were performed using the ANISN code.[1] Sixteen-group Hansen Roach cross sections were mixed in a code called BLENDR[2] which also accounts for self-shielding in the important actinides (U^{238}, U^{235}, U^{238}, Th^{232}, Pu^{238}, Pu^{239}, and Pu^{240}). The "σ_ρ" resonance correction[3] was made to compensate for the self-shielding of incident neutrons due to the nature of the material and consists of the following procedure:

1. The σ_p of a resonance nuclide in a homogeneous system is calculated according to the equation:

$$\sigma_{p_D} = \frac{\sum_{i=1}^{M} \sigma_{scatt_i} N_i}{N_D} \tag{7.1}$$

Where:

σ_{scatt} is the scattering cross section in the resonance-energy range for the ith component of the mixture,

N_i is the number density of the ith component of the mixture in atoms/barn-cm,

M is the number of components in the mixture, and

N_D is the number density of the isotope for which σ_{p_D} is being calculated.

σ_p is calculated only for the important actinides (those listed above). Infinitely dilute cross sections are used for all other nuclides.

2. The value of σ_p calculated in Eq. 7-1 is used as a parameter to pick the correct cross sections from the cross-section set. The cross sections in the cross-section set are weighted by the flux. The flux, of course, is a function of the density of materials in the system. So, when the material densities change, the cross sections change and σ_p parametrizes this change. If the calculated σ_p does not match any of the values in the cross-section set, BLENDR can be made to interpolate (log-log interpolation) between two bracketing values. When σ_p exceeds all values on the cross-section set, infinitely-dilute cross sections are used.

As an example consider the attached sheets containing σ_p's and N's for Pu from a PWR (cooled 5 years). σ_p for Pu^{239} was calculated to be 148 barns. This value falls in between the available value of 100 barns and 200 barns. Thus, the code was told to interpolate between 100 and 200 to find the cross section for $\sigma_p = 148$. The number density of Pu^{239} for this case was 7.4×10^{-3} atom/barn-cm.

7.1.2 Running ANISN

Once the cross sections were mixed and self-shielded by BLENDR, ANISN was run. The fissioning core was assumed to be spherical and surrounded by a spherical reflector. There were 45 mesh intervals from the center of the sphere to the outside edge of the reflector. Reflector materials considered were granite, shale, and sand. The PuO_2 was assumed to mix uniformly with the water in the core.

7.2 HLW (GLASS) SHIPPING CASK, 8-CANISTER CONFIGURATION

The objective was to study the criticality problems associated with the temporary storage of eight HLW-containing canisters in one shipping cask. The HLW would have 100 percent Pu and would be fixed in a glass (SiO_2) matrix.

7.2.1 Solution

The Monte Carlo method of analysis was selected in order to most accurately preserve the geometrical considerations of the problem. Specifically, the very versatile code KENO IV was used along with the KENO 16-group cross-section library, which includes a σ_p correction for the self-shielding of resonance nuclei.[3]

The system geometry was assumed to be a closely packed array of canisters, which minimizes the exterior surface area of the conglomerate. The glass content was obtained by subtracting the volume of the HLW+100% Pu (constituent masses were obtained from ORIGEN results with an effective density of these materials taken to be 10 g/cc) from the total volume of a canister and assuming that this difference is completely filled with SiO_2 having a typical glass density of 2.2 g/cc.

K_{eff} for the 8-canister assembly was found to be 0.24 with a standard deviation of .00153. An estimate of the sensitivity of k_{eff} to the SiO_2 number density was made and shows a $1.9(10)^{-3}$ fractional change in k_{eff} for a 0.01 fractional change in the SiO_2 number density.

7.2.2 Recommendations

Even though this calculation indicates that the system of 8 canisters is very subcritical, a refined calculation should be performed which includes a more accurate representation of the actual geometry and a more accurate determination of the SiO_2 and boron contents. These calculations should include the fissionable actinides above plutonium and should account for reflector effects provided by the cask itself.

These follow-up calculations should provide values of the infinite multiplication factor k_∞ and should include analyses of accident scenarios which involve flooding of the cask and/or canisters both during transport and after arrival at the repository site.

7.3 CRITICALITY CALCULATIONS FOR REFLECTED PuO_2 SLURRIES

A series of criticality calculations was performed in order to investigate the potentiality of PuO_2 from U-recycle HLW forming a critical mass after the unlikely events of selective leaching and/or selective transport and/or selective deposition. These calculations were done using the discrete-ordinates code ANISN in the manner described in the previous section. The conditions for criticality were determined for reflected slurries of PuO_2 having plutonium concentrations of 0.015, 0.03, 0.05, 0.10, 0.50, 2.0, and 5.0 grams/cc. Reflector materials considered were shale (2.4 grams/cc), granite (2.67 grams/cc), and sand (2.41 grams/cc). The calculations considered only PuO_2 and not the other nuclides present in the waste material. The ORIGEN-determined isotopic composition for the plutonium 5 years after removal is presented in Table 7-2.

The results of these calculations are presented in Tables 7-3 and 7-4 and shown in Figures 7-1 and 7-2 as critical mass (kg of Pu) and critical radius (cm) respectively. It is clear that the granite, shale, and sand have nearly the same reflector properties for the conditions of these calculations. The only significant difference seems to be slightly larger critical mass requirements for all of the shale-reflected assemblies. The shapes of the plotted results conform faithfully to those found in the literature for water-reflected assemblies. A complete computer output for a typical calculation is also included in this section (Table 7-5).

7. REFERENCES

1. Engle, Ward W., Jr., "ANISN - A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering." K-1693, ORNL (1967).
2. Simmons, G. L., BLENDR, A Program to Mix, Separate, and Collapse Cross Sections, SAI 5-310-13-74.
3. Petrie, L. M., Cross, N. F., KENO IV. An Improved Monte Carlo Criticality Program, pp. 79-80, ORNL-4938.

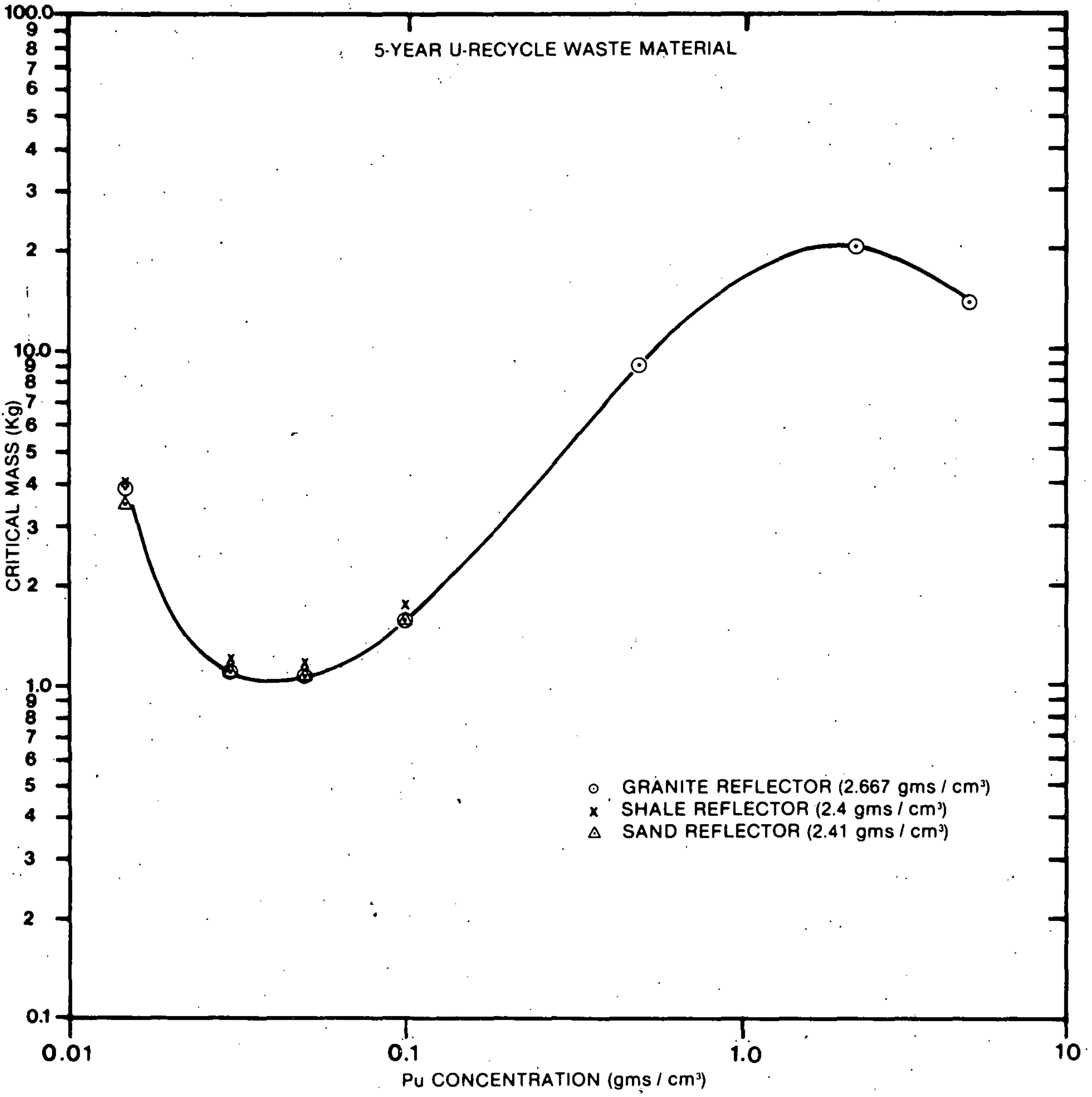


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Figure 7-1 CRITICAL MASS VS Pu CONCENTRATION

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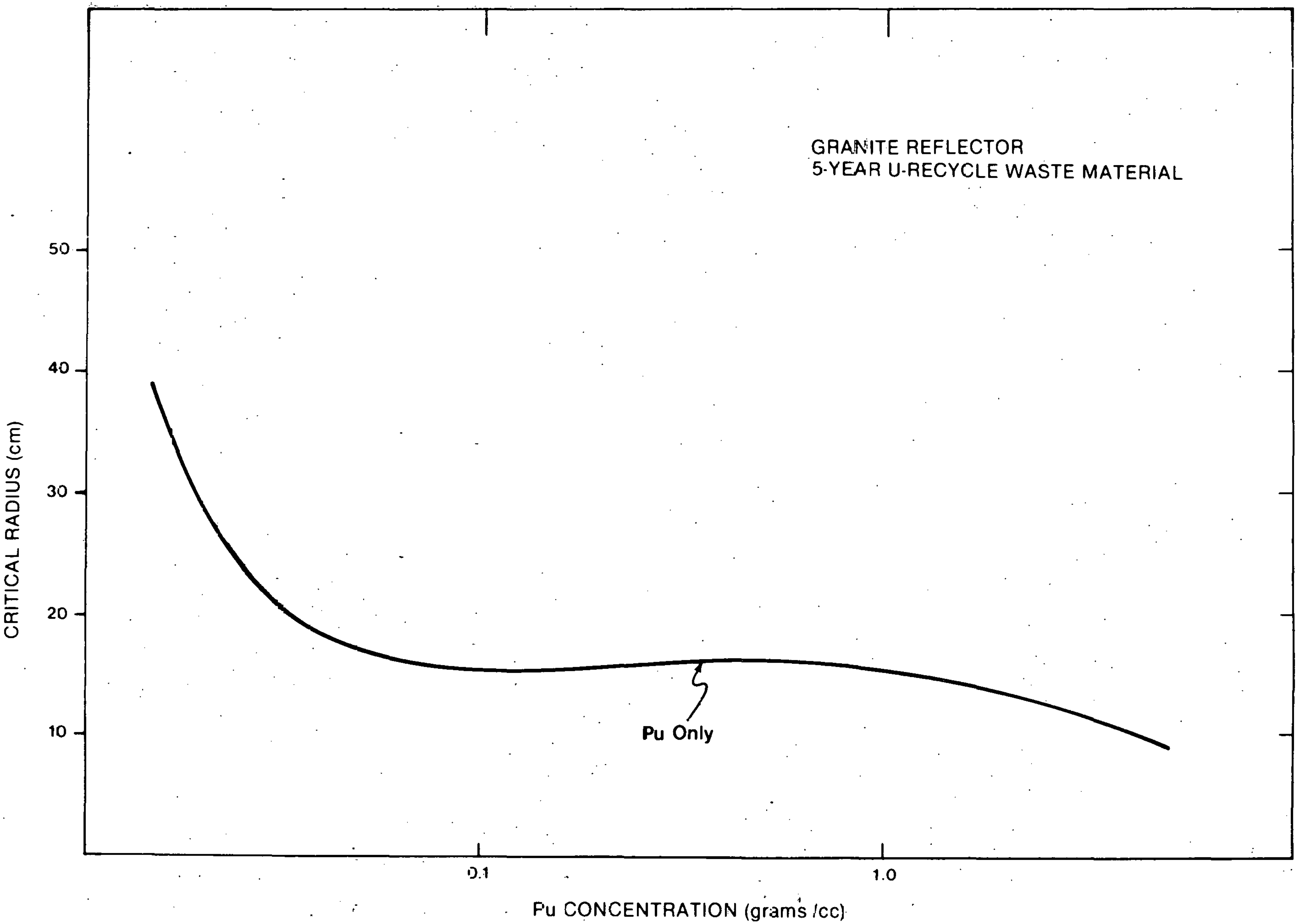

Figure 7-2. CRITICAL RADIUS VS Pu CONCENTRATION

TABLE 7-1

COMPARISON OF CALCULATED CRITICALITY PARAMETERS FOR VARIOUS GEOMETRIES OF PuO_2 EXPERIMENTAL k_{eff} = 1.000

Geometry	Bare Assemblies		Plexiglass Reflected	
	x*(cm)	Calculated k_{eff}	x*(cm)	Calculated k_{eff}
Infinite Slab	12.17 ± 0.28	1.005 ± 0.004	3.34 ± 0.10	0.992 ± 0.006
Sphere	15.81 ± 0.20	1.017 ± 0.004	11.24 ± 0.38	1.008 ± 0.006
Infinite Cylinder	11.28 ± 0.20	1.014 ± 0.004	6.69 ± 0.38	1.011 ± 0.006
Cube	26.48 ± 0.28	1.003 ± 0.004	17.36 ± 0.54	0.999 ± 0.006

*Critical Thickness of slab or cube and critical radius of the sphere and cylinder.

TABLE 7-2

ISOTOPIC COMPOSITION OF PLUTONIUM FOR 5-YEAR U-RECYCLE WASTE MATERIAL WITH ALL PLUTONIUM RETAINED

Isotope	Grams/Metric Ton	Weight Percentage
Pu-238	287	3.05
Pu-239	5530	58.84
Pu-240	2210	23.51
Pu-241	960	10.21
Pu-242	412	4.38

TABLE 7-3

CRITICAL MASS OF REFLECTED PuO_2 SLURRIES--5-YEAR U-CYCLE WASTE MATERIAL

	Critical Mass (kg)		
Plutonium Conc. (grams/cc)	Shale Reflector (2.4 grams/cc)	Granite Reflector (2.67 grams/cc)	Sand Reflector (2.41 grams/cc)
.015	3.83	3.62	3.32
.03	1.22	1.07	1.05
.05	1.20	1.05	1.04
.1	1.84	1.58	1.53
.5		8.9	
2.0		20.6	
5.0		16.9	

TABLE 7-4

CRITICAL RADIUS OF REFLECTED PU_2 SLURRIED--5-YEAR U-RECYCLE WASTE MATERIALS

	Critical Radius		
Plutonium Conc. (grams/cc)	Shale Reflector (2.9 grams/cc)	Granite Reflector (2.67 grams/cc)	Sand Reflector (2.41 grams/cc)
.015	39.4	38.6	37.5
.03	21.3	20.4	20.3
.05	17.9	17.1	17.1
.1	16.4	15.6	15.4
.5		16.2	
2.0		13.5	
5.0		9.3	

```
pU ONLY/(    GMS/CC/GRANITE REFLECTOR
                                   THE TIME ESTIMATE FOR THIS PROBLEM IS      0.00 SECONDS
   15$ ARRAY     36 ENTRIES READ
   16* ARRAY     14 ENTRIES READ
    0T

  6296 LOCATIONS WILL BE USED FOR THIS PROBLEM
```

TABLE 7-5

CRITICALITY CALCULATIONS FOR PuO_2 FROM U-RECYCLE HIGH LEVEL WASTE WITH GRANITE REFLECTOR-ANISN COMPLETE COMPUTER OUTPUT (TYPICAL CASE)

TABLE 7-5 (Cont'd)

```
ONLY/0.01GMS/CC/GRANITE REFLECTOR

ID     PROBLEM ID NO.                                                  1
ITH    0/1 = FORWARD/ADJOINT                                           0
ISCT   ORDER OF SCATTERING                                             1
ISN    ORDER OF QUADRATURE                                             8
ICE    1/2/3 =PLANE/CYLINDER/SPHERE                                    3
IBL    0/1/2/3 = VACUUM/REFLECTION/PERIODIC/WHITE                      1
IBR    0/1/2/3 = VACUUM/REFLECTION/PERIODIC/WHITE                      0
IZM    NUMBER OF ZONES                                                 2
IM     NUMBER OF INTERVALS                                            45
IEVT   0/1/2/3/4/5/6 = Q/K-EFF/ALPHA/Z/R/H                             4
IGM    NUMBER OF GROUPS                                               16
IHT    POSITION OF TOTAL CROSS SECTION                                 3
IHS    POSITION OF WITHIN GROUP SCATTERING                             4
IHM    CROSS SECTION TABLE LENGTH                                      9
MS     MIXING TABLE LENGTH                                             0
MCR    NUMBER OF MATERIALS FROM CARDS                                  0
MTP    NUMBER OF MATERIALS FROM TAPE                                   4
MT     TOTAL NUMBER OF MATERIALS                                       4
IDFM   0/1 = NONE/DENSITY FACTORS IN 21**                              0
IPVT   0/1/2 = NONE/K/ALPHA                                            0
IQM    0/1 = NONE/DISTRIBUTED SOURCE                                   0
IPM    0/1/IM = NONE / S(MM,IPP) / S(MM,IM)                            0
IPP    INTERVAL OF SHELL SOURCE                                        0
IIM    INNER ITERATION MAXIMUM                                        20
ID1    0/1/2/3 = NO/PRINT/PUNCH/BOTH                                   0
ID2    0/1/2 = NO / X-SEC TAPE / PREVIOUS CASE                         0
ID3    0/N = NO / NUMBER OF ACTIVITIES BY ZONE                         0
ID4    0/1 = NO / NUMBER OF ACTIVITIES BY INTERVAL                     0
ICM    OUTER ITERATION MAXIMUM                                        30
IDAT1 0/1/2 = NO / MINIMUM TAPE / MAXIMUM TAPE                         0
IDAT2 0/1 = NO / DIFFUSION--24$$                                       0
IFG    0/1 = NONE / FEW GROUP COLLAPSE                                 0
IFLU   0/1/2 = BOTH / LINEAR / STEP                                    0
IFN    0/1/2 = FISSION GUESS / FLUX GUESS / PREVIOUS                   0
IPRT   0/1 = PRINT CROSS SECTIONS / DO NOT                             1
IXTR   0/1 = CALCULATE P(L) / READ IN P(L)                             0

EV     EIGENVALUE GUESS                                     0.0000E+00
EVM    EIGENVALUE MODIFIER                                 -1.0000E-01
EPS    PRECISION DESIRED                                    5.0000E-04
BF     BUCKLING FACTOR                                      1.0000E-35
DY     CYLINDER OR PLANE HEIGHT                             0.0000E+00
DZ     PLANE DEPTH                                          0.0000E+00
DFM1   HEIGHT FOR VOID CORRECTION                           0.0000E+00
XNF    NORMALIZATION FACTOR                                 1.0000E+00
PV     IPVT = 1 / 2-K / ALPHA                               0.0000E+00
RYF    LAMBDA2 RELAXATION FACTOR                            5.0000E-01
XLAL   POINT CONVERGENCE = EPS IF NOT 0                     1.0000E-03
XLAH   1-LAMBDA MAXIMUM SEARCH                              5.0000E-02
EQL    EV CHANGE EPS - SEARCH                               5.0000E-02
XNPM   NEW PARAMETER MODIFIER - SEARCH                      7.5000E-01

  1748 LOCATIONS WILL BE USED TO READ CROSS SECTIONS
   13$ ARRAY      4 ENTRIES READ
    0T

 ELEMENTS FROM LIBRARY TAPE
           1        10        GRANITE AT 2.667 GMS/CC
           2        11        GRANITE AT 2.667 GMS/CC
           3       120       0.01 GMS/CC-PU ONLY-5 YRS
           4       121       0.01 GMS/CC-PU ONLY-5 YRS
    2* ARRAY     45 ENTRIES READ
    0T
```

TABLE 7-5 (Cont'd)

```
  5* ARRAY      16 ENTRIES READ
  4* ARRAY         ENTRIES READ
  6* ARRAY         ENTRIES READ
  7* ARRAY       9 ENTRIES READ
  8S ARRAY      45 ENTRIES READ
  9S ARRAY       2 ENTRIES READ
 19S ARRAY       2 ENTRIES READ
 20* ARRAY       2 ENTRIES READ
  0T
```

TABLE 7-5 (Cont'd)

```
uONLY/0.01GMS/CC/GRANITE REFLECTOR

 INT.  ZONE NUMBER     RADIUS         AREA        VOLUME     FISS DENS   DENS FACTOR
   1       1             0             0        1.55140E+02 1.00000E+00
   2       1        3.33333E+00 1.39626E+02 1.08598E+03 1.00000E+00
   3       1        6.66667E+00 5.58505E+02 2.94767E+03 1.00000E+00
   4       1        1.00000E+01 1.25664E+03 5.74019E+03 1.00000E+00
   5       1        1.33333E+01 2.23402E+03 9.46356E+03 1.00000E+00
   6       1        1.66667E+01 3.49066E+03 1.41178E+04 1.00000E+00
   7       1        2.00000E+01 5.02655E+03 2.95516E+03 1.00000E+00
   8       1        2.05714E+01 5.31788E+03 3.12398E+03 1.00000E+00
   9       1        2.11429E+01 5.61742E+03 3.29749E+03 1.00000E+00
  10       1        2.17143E+01 5.92517E+03 3.47569E+03 1.00000E+00
  11       1        2.22857E+01 6.24113E+03 3.65858E+03 1.00000E+00
  12       1        2.28571E+01 6.56529E+03 3.84616E+03 1.00000E+00
  13       1        2.34286E+01 6.89766E+03 4.03843E+03 1.00000E+00
  14       1        2.40000E+01 7.23823E+03 4.23539E+03 1.00000E+00
  15       1        2.45714E+01 7.58701E+03 4.43704E+03 1.00000E+00
  16       1        2.51429E+01 7.94400E+03 4.64338E+03 1.00000E+00
  17       1        2.57143E+01 8.30919E+03 4.85440E+03 1.00000E+00
  18       1        2.62857E+01 8.68259E+03 5.07012E+03 1.00000E+00
  19       1        2.68571E+01 9.06420E+03 5.29053E+03 1.00000E+00
  20       1        2.74286E+01 9.45401E+03 5.51562E+03 1.00000E+00
  21       2        2.80000E+01 9.85203E+03 8.70575E+03      0
  22       2        2.88571E+01 1.04645E+04 9.23859E+03      0
  23       2        2.97143E+01 1.10953E+04 9.78726E+03      0
  24       2        3.05714E+01 1.17447E+04 1.03518E+04      0
  25       2        3.14286E+01 1.24125E+04 1.09321E+04      0
  26       2        3.22857E+01 1.30988E+04 1.15282E+04      0
  27       2        3.31429E+01 1.38035E+04 1.21402E+04      0
  28       2        3.40000E+01 1.45267E+04 1.27680E+04      0
  29       2        3.48571E+01 1.52684E+04 1.34117E+04      0
  30       2        3.57143E+01 1.60285E+04 1.40711E+04      0
  31       2        3.65714E+01 1.68071E+04 1.47464E+04      0
  32       2        3.74286E+01 1.76042E+04 1.54375E+04      0
  33       2        3.82857E+01 1.84197E+04 1.61445E+04      0
  34       2        3.91429E+01 1.92537E+04 1.68672E+04      0
  35       2        4.00000E+01 2.01062E+04 5.86588E+04      0
  36       2        4.27273E+01 2.29414E+04 6.66461E+04      0
  37       2        4.54545E+01 2.59636E+04 7.51433E+04      0
  38       2        4.81818E+01 2.91727E+04 8.41503E+04      0
  39       2        5.09091E+01 3.25687E+04 9.36671E+04      0
  40       2        5.36364E+01 3.61517E+04 1.03694E+05      0
  41       2        5.63636E+01 3.99216E+04 1.14230E+05      0
  42       2        5.90909E+01 4.38784E+04 1.25277E+05      0
  43       2        6.18182E+01 4.80222E+04 1.36833E+05      0
  44       2        6.45455E+01 5.23530E+04 1.48899E+05      0
  45       2        6.72727E+01 5.68706E+04 1.61475E+05      0
  46                7.00000E+01 6.15752E+04
```

```
pU ONLY/0    MS/CC/GRANITE REFLECTOR

       FISS SPEC      VELOCITY     RT ALBEDO    LFT ALBEDO  DIFF MARKER   MAT'L/ZONE    L OF P(L)   RADIUS MOD
   1  2.04000E-01        0                                                    3              1        1.00000E-01
   2  3.44000E-01        0                                                    1              1            0
   3  1.68000E-01        0
   4  1.80000E-01        0
   5  9.00000E-02        0
   6  1.40000E-02        0
   7      0              0
   8      0              0
   9      0              0
  10      0              0
  11      0              0
  12      0              0
  13      0              0
  14      0              0
  15      0              0
  16      0              0
```

TABLE 7-5 (Cont'd)

```
uONLY/0.01GMS/CC/GRANITE REFLECTOR

          CROSS SECTION MIXING TABLE                                 ANGULAR  QUADRATURE  CONSTANTS
      MIXTURE      COMPONENT    NO. DENSITY                  COSINE (MU)      WEIGHT     REFL DIRECT    WT. X COS.
   1                                                        -1.00000E+00        0              9             0
   2                                                        -9.51190E-01  6.04938E-02          9       -5.75411E-02
   3                                                        -7.86796E-01  9.07407E-02          8       -7.13944E-02
   4                                                        -5.77350E-01  1.37037E-01          7       -7.91184E-02
   5                                                        -2.18218E-01  2.11728E-01          6       -4.62029E-02
   6                                                         2.18218E-01  2.11728E-01          5        4.62029E-02
   7                                                         5.77350E-01  1.37037E-01          4        7.91184E-02
   8                                                         7.86796E-01  9.07407E-02          3        7.13944E-02
   9                                                         9.51190E-01  6.04938E-02          2        5.75411E-02
```

TABLE 7-5 (Cont'd)

TABLE 7-5 (Cont'd)

```
ER INNER        NEUT BAL      UPSCATTER RATIO     EIGENVALUE          LAMBDA1           LAMBDA2
   0     0       0000000E+00   0.0000000E+00    0.0000000E+00     0.0000000E+00     0.0000000E+00
   1     0       0000001E+00   0.0000000E+00    0.0000000E+00     7.1007090E-01     9.4183606E+00
   2    85     ..0000000E+00   1.0000001E+00    1.0000000E-01     7.4993183E-01     1.0061506E+00
   3   159     1.0000002E+00   1.0000002E+00    1.4017550E+00     7.6355453E-01     1.0015355E+00
   4   255     1.0000001E+00   1.0000002E+00    7.0566205E+00     7.9836666E-01     1.0043933E+00
   5   355     1.0000001E+00   1.0000002E+00    7.0566205E+00     8.7159284E-01     1.0083113E+00
   6   406     9.9999976E-01   1.0000002E+00    1.4175222E+01     8.7364622E-01     1.0001524E+00
   7   517     1.0000001E+00   1.0000001E+00    2.7339360E+01     9.1010482E-01     1.0033639E+00
   8   632     1.0000002E+00   1.0000002E+00    5.2156816E+01     9.3568943E-01     1.0019249E+00
   9   755     1.0000002E+00   1.0000002E+00    7.1248439E+01     9.5052703E-01     1.0008783E+00
  10   877     9.9999971E-01   1.0000003E+00    8.8584560E+01     9.5507614E-01     1.0002177E+00
  11   999     1.0000002E+00   1.0000002E+00    1.0501278E+02     9.5742884E-01     1.0000999E+00
  12  1122     1.0000001E+00   1.0000002E+00    1.2038493E+02     9.5886978E-01     1.0000573E+00
  13  1244     1.0000001E+00   1.0000001E+00    1.3638232E+02     9.5984095E-01     1.0000366E+00
  14  1367     1.0000004E+00   1.0000003E+00    1.5160981E+02     9.6054033E-01     1.0000247E+00
  15  1491     9.9999972E-01   1.0000002E+00    1.6663854E+02     9.6105538E-01     1.0000188E+00
  16  1619     1.0000000E+00   1.0000001E+00    1.8151031E+02     9.6146083E-01     1.0000132E+00
  17  1748     1.0000001E+00   1.0000001E+00    1.9626017E+02     9.6177934E-01     1.0000105E+00
  18  1879     1.0000001E+00   1.0000002E+00    2.1091017E+02     9.6203680E-01     1.0000082E+00
  19  2012     9.9999987E-01   1.0000002E+00    2.2547984E+02     9.6224497E-01     1.0000069E+00
  20  2146     1.0000001E+00   1.0000001E+00    2.3998030E+02     9.6242303E-01     1.0000052E+00
  21  2280     9.9999998E-01   1.0000001E+00    2.5442485E+02     9.6257048E-01     1.0000046E+00
  22  2415     1.0000000E+00   1.0000001E+00    2.6882003E+02     9.6269714E-01     1.0000037E+00
  23  2550     1.0000003E+00   1.0000003E+00    2.8317290E+02     9.6280678E-01     1.0000030E+00
  24  2685     9.9999980E-01   1.0000002E+00    2.9749172E+02     9.6289502E-01     1.0000032E+00
  25  2820     1.0000001E+00   1.0000002E+00    3.1177751E+02     9.6298059E-01     1.0000020E+00
  26  2956     1.0000000E+00   1.0000001E+00    3.2603623E+02     9.6305076E-01     1.0000022E+00
  27  3092     9.9999983E-01   1.0000001E+00    3.4027092E+02     9.6311301E-01     1.0000020E+00
  28  3231     1.0000001E+00   1.0000001E+00    3.5448292E+02     9.6317102E-01     1.0000014E+00
  29  3371     1.0000005E+00   1.0000005E+00    3.6867381E+02     9.6322650E-01     1.0000009E+00
 GRP.    1 REQUIRED    5 ITERATIONS. MFD OF  6.50909E-04 OCCURRED IN INT.   30   COARSE MESH= 25
 GRP.    2 REQUIRED    6 ITERATIONS. MFD OF  3.94263E-04 OCCURRED IN INT.   29   COARSE MESH= 25
 GRP.    3 REQUIRED    6 ITERATIONS. MFD OF  4.78423E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.    4 REQUIRED    7 ITERATIONS. MFD OF  8.04723E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.    5 REQUIRED    8 ITERATIONS. MFD OF  9.68383E-04 OCCURRED IN INT.   24   COARSE MESH= 29
 GRP.    6 REQUIRED    9 ITERATIONS. MFD OF  7.53371E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.    7 REQUIRED    9 ITERATIONS. MFD OF  9.32461E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.    8 REQUIRED   10 ITERATIONS. MFD OF  6.40705E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.    9 REQUIRED   10 ITERATIONS. MFD OF  6.49035E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.   10 REQUIRED    9 ITERATIONS. MFD OF  5.82340E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.   11 REQUIRED    8 ITERATIONS. MFD OF  8.94370E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.   12 REQUIRED    9 ITERATIONS. MFD OF  5.67021E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.   13 REQUIRED    8 ITERATIONS. MFD OF  8.24248E-04 OCCURRED IN INT.   25   COARSE MESH= 28
 GRP.   14 REQUIRED    8 ITERATIONS. MFD OF  8.38358E-04 OCCURRED IN INT.   24   COARSE MESH= 28
 GRP.   15 REQUIRED   11 ITERATIONS. MFD OF  8.08525E-04 OCCURRED IN INT.   24   COARSE MESH= 28
 GRP.   16 REQUIRED   17 ITERATIONS. MFD OF  4.71984E-04 OCCURRED IN INT.   23   COARSE MESH= 29
  30  3511     1.0000000E+00   1.0000001E+00    3.8284946E+02     9.6326602E-01     1.0000018E+00
  30  3511     1.0000000E+00   1.0000001E+00    3.8284946E+02     9.6326602E-01     1.0000018E+00  FINAL MONITOR

****** OUTER ITERATION LIMIT REACHED
```

```
PU ONLY/0.01GMS/CC/GRANITE REFLECTOR

 INT. ZONE NUMBER     RADIUS    INT. MIDPOINT    AREA      VOLUME    FISSION DENS
   1        1      0.00000E+00
█****        0       9.40596E+06
█****        2       0.00000E+00
█*************       2.15486E+05
█*************       0.00000E+00
   1*********      3.27375E+02
█*************       3.47962E-10
   4        1      3.92849E+02
█*************       3.48021E+08
█****        5       0.00000E+00
█*************       3.44778E+06
█*************       0.00000E+00
   1*********      7.20224E+02
█*************       2.40232E-10
   7        1      7.85699E+02
█*************       1.79168E+08
█****        8       0.00000E+00
█*************       3.20713E+06
█*************       0.00000E+00
   1*********      8.41820E+02
█*************       1.90336E-10
  10        1      8.53045E+02
█*************       2.10727E+08
█****       11       0.00000E+00
█*************       9.63197E+06
█*************       0.00000E+00
   1*********      9.09166E+02
█*************       1.57335E-10
  13        1      9.20390E+02
█*************       2.44845E+08
█****       14       0.00000E+00
█*************       1.11708E+07
█*************       0.00000E+00
   1*********      9.76512E+02
█*************       1.19502E-10
  16        1      9.87736E+02
█*************       2.81522E+08
█****       17       0.00000E+00
█*************       1.28236E+07
█*************       0.00000E+00
   1*********      1.04386E+03
█*************       6.96993E-11
  19        1      1.05508E+03
█*************       3.20758E+08
█****       20       0.00000E+00
█*************       1.45904E+07
█*************       0.00000E+00
   2*********      1.10041E+03
█*************       0.00000E+00
  22        2      1.10084E+03
█*************       1.30632E+07
   0       23      0.00000E+00
█*************       1.52521E+07
█****        0       0.00000E+00
   2*********      1.10298E+03
█*************       0.00000E+00
  25        2      1.10341E+03
█*************       1.31243E+07
   0       26      0.00000E+00
█********   **       1.53234E+07
█****        0       0.00000E+00
   2*********      1.10555E+03
```

TABLE 7-5 (Cont'd)

TABLE 7-5 (Cont'd)

```
   28                 1.10598E+03
3*********             1.31855E+07
   0        29        0.00000E+00
3*************         1.53949E+07
3****          0       0.00090E+00
    2*********        1.10812E+03
3*************         0.00000E+00
   31        2        1.10855E+03
3*************         1.32468E+07
   0        32        0.00000E+00
3*************         1.54665E+07
3****          0       0.00000E+00
    2*********        1.11069E+03
3*************         0.00000E+00
   34        2        1.11112E+03
3*************         1.33083E+07
   0        35        0.00000E+00
3*************         1.55383E+07
3****          0       0.00000E+00
    2*********        1.11607E+03
3*************         0.00090E+00
   37        2        1.11743E+03
3*************         4.28925E+07
   0        38        0.00000E+00
3*************         1.57678E+07
3****          0       0.00000E+00
    2*********        1.12425E+03
3*************         0.00009E+00
   40        2        1.12561E+03
3*************         4.35281E+07
   0        41        0.00000E+00
3*************         1.59990E+07
3****          0       0.00000E+00
    2*********        1.13243E+03
3*************         0.09000E+00
   43        2        1.13380E+03
3*************         4.41626E+07
   0        44        0.00009E+00
3*************         1.62318E+07
3****          0       0.00090E+00
    2*********        1.14061E+03
3*************         0.00000E+00
    46                 1.14198E+03                  1.63880E+07
```

```
PU ONLY/0.01GMS/CC/GRANITE REFLECTOR

TOTAL FLUX

INT ZONE  MID PT.        GROU  1       GROU  2       GROU  3       GROU  4       GROU  5       GROU  6       GROU  7       GROU  8      G
U  9      GROU 10
  1    1    65.47     8.506E-10  1.595E-09  7.128E-10  1.029E-09  1.067E-09  7.867E-10  6.298E-10  5.834E-10  5.846E-10  4.111E-10
  2    1   196.42     8.329E-10  1.562E-09  6.979E-10  1.007E-09  1.044E-09  7.703E-10  6.167E-10  5.713E-10  5.725E-10  4.026E-10
  3    1   327.37     7.928E-10  1.487E-09  6.643E-10  9.585E-10  9.940E-10  7.332E-10  5.870E-10  5.437E-10  5.449E-10  3.832E-10
  4    1   458.32     7.310E-10  1.371E-09  6.125E-10  8.838E-10  9.165E-10  6.760E-10  5.412E-10  5.013E-10  5.024E-10  3.533E-10
  5    1   589.27     6.494E-10  1.218E-09  5.441E-10  7.851E-10  8.142E-10  6.005E-10  4.808E-10  4.454E-10  4.463E-10  3.138E-10
  6    1   720.22     5.473E-10  1.026E-09  4.586E-10  6.618E-10  6.863E-10  5.062E-10  4.053E-10  3.754E-10  3.762E-10  2.645E-10
  7    1   796.92     4.797E-10  8.997E-10  4.020E-10  5.800E-10  6.015E-10  4.437E-10  3.552E-10  3.290E-10  3.297E-10  2.319E-10
  8    1   819.37     4.571E-10  8.572E-10  3.830E-10  5.527E-10  5.731E-10  4.227E-10  3.384E-10  3.135E-10  3.141E-10  2.209E-10
  9    1   841.82     4.336E-10  8.131E-10  3.633E-10  5.242E-10  5.436E-10  4.010E-10  3.210E-10  2.974E-10  2.980E-10  2.096E-10
 10    1   864.27     4.095E-10  7.679E-10  3.431E-10  4.951E-10  5.135E-10  3.787E-10  3.032E-10  2.808E-10  2.814E-10  1.979E-10
 11    1   886.72     3.845E-10  7.211E-10  3.222E-10  4.649E-10  4.821E-10  3.556E-10  2.847E-10  2.637E-10  2.643E-10  1.859E-10
 12    1   909.17     3.587E-10  6.726E-10  3.005E-10  4.337E-10  4.498E-10  3.318E-10  2.656E-10  2.460E-10  2.465E-10  1.733E-10
 13    1   931.61     3.315E-10  6.217E-10  2.778E-10  4.008E-10  4.155E-10  3.065E-10  2.454E-10  2.273E-10  2.278E-10  1.602E-10
 14    1   954.06     3.027E-10  5.678E-10  2.537E-10  3.662E-10  3.798E-10  2.802E-10  2.243E-10  2.077E-10  2.082E-10  1.464E-10
 15    1   976.51     2.720E-10  5.099E-10  2.279E-10  3.287E-10  3.407E-10  2.512E-10  2.011E-10  1.864E-10  1.868E-10  1.313E-10
 16    1   998.96     2.381E-10  4.467E-10  1.996E-10  2.882E-10  2.990E-10  2.206E-10  1.766E-10  1.636E-10  1.639E-10  1.152E-10
 17    1  1021.41     2.014E-10  3.773E-10  1.687E-10  2.431E-10  2.518E-10  1.856E-10  1.486E-10  1.376E-10  1.379E-10  9.701E-11
 18    1  1043.86     1.594E-10  2.996E-10  1.339E-10  1.934E-10  2.009E-10  1.484E-10  1.188E-10  1.101E-10  1.103E-10  7.754E-11
 19    1  1066.31     1.151E-10  2.148E-10  9.606E-11  1.382E-10  1.428E-10  1.051E-10  8.404E-11  7.780E-11  7.793E-11  5.479E-11
 20    1  1088.75     5.908E-11  1.129E-10  5.051E-11  7.338E-11  7.671E-11  5.687E-11  4.562E-11  4.232E-11  4.245E-11  2.987E-11


 21    2  1100.41     2.265E-11  5.402E-11  2.346E-11  3.876E-11  4.646E-11  3.661E-11  2.795E-11  2.581E-11  2.561E-11  1.793E-11
 22    2  1101.26     1.985E-11  5.098E-11  2.160E-11  3.720E-11  4.647E-11  3.705E-11  2.783E-11  2.557E-11  2.525E-11  1.765E-11
 23    2  1102.12     1.755E-11  4.804E-11  1.996E-11  3.541E-11  4.587E-11  3.717E-11  2.765E-11  2.530E-11  2.490E-11  1.737E-11
 24    2  1102.98     1.563E-11  4.524E-11  1.848E-11  3.357E-11  4.494E-11  3.707E-11  2.742E-11  2.500E-11  2.453E-11  1.709E-11
 25    2  1103.84     1.398E-11  4.257E-11  1.713E-11  3.175E-11  4.378E-11  3.679E-11  2.713E-11  2.468E-11  2.415E-11  1.680E-11
 26    2  1104.69     1.255E-11  4.004E-11  1.590E-11  2.996E-11  4.249E-11  3.636E-11  2.678E-11  2.433E-11  2.375E-11  1.650E-11
 27    2  1105.55     1.131E-11  3.764E-11  1.478E-11  2.825E-11  4.110E-11  3.581E-11  2.639E-11  2.395E-11  2.334E-11  1.620E-11
 28    2  1106.41     1.020E-11  3.537E-11  1.374E-11  2.660E-11  3.965E-11  3.517E-11  2.596E-11  2.355E-11  2.292E-11  1.590E-11
 29    2  1107.26     9.220E-12  3.323E-11  1.279E-11  2.503E-11  3.815E-11  3.444E-11  2.549E-11  2.313E-11  2.249E-11  1.559E-11
 30    2  1108.12     8.342E-12  3.120E-11  1.191E-11  2.353E-11  3.663E-11  3.365E-11  2.499E-11  2.269E-11  2.205E-11  1.527E-11
 31    2  1108.98     7.554E-12  2.929E-11  1.110E-11  2.211E-11  3.509E-11  3.279E-11  2.445E-11  2.223E-11  2.160E-11  1.495E-11
 32    2  1109.84     6.847E-12  2.749E-11  1.034E-11  2.076E-11  3.356E-11  3.189E-11  2.389E-11  2.175E-11  2.113E-11  1.463E-11
 33    2  1110.69     6.209E-12  2.579E-11  9.642E-12  1.948E-11  3.205E-11  3.094E-11  2.330E-11  2.125E-11  2.065E-11  1.429E-11
 34    2  1111.55     5.634E-12  2.419E-11  8.993E-12  1.828E-11  3.055E-11  2.997E-11  2.268E-11  2.074E-11  2.016E-11  1.396E-11
 35    2  1113.34     4.649E-12  2.123E-11  7.820E-12  1.603E-11  2.755E-11  2.786E-11  2.133E-11  1.960E-11  1.910E-11  1.323E-11
 36    2  1116.07     3.418E-12  1.725E-11  6.273E-12  1.301E-11  2.327E-11  2.463E-11  1.920E-11  1.782E-11  1.744E-11  1.210E-11
 37    2  1118.80     2.520E-12  1.397E-11  5.034E-12  1.052E-11  1.943E-11  2.143E-11  1.703E-11  1.597E-11  1.573E-11  1.094E-11
 38    2  1121.52     1.850E-12  1.127E-11  4.036E-12  8.471E-12  1.606E-11  1.838E-11  1.488E-11  1.411E-11  1.398E-11  9.759E-12
 39    2  1124.25     1.374E-12  9.062E-12  3.227E-12  6.787E-12  1.313E-11  1.553E-11  1.278E-11  1.225E-11  1.223E-11  8.569E-12
 40    2  1126.98     1.016E-12  7.246E-12  2.567E-12  5.398E-12  1.060E-11  1.290E-11  1.079E-11  1.044E-11  1.050E-11  7.382E-12
 41    2  1129.71     7.494E-13  5.750E-12  2.024E-12  4.243E-12  8.411E-12  1.050E-11  8.897E-12  8.691E-12  8.791E-12  6.204E-12
 42    2  1132.43     5.510E-13  4.510E-12  1.573E-12  3.271E-12  6.508E-12  8.311E-12  7.115E-12  6.999E-12  7.119E-12  5.041E-12
 43    2  1135.16     4.018E-13  3.470E-12  1.187E-12  2.433E-12  4.822E-12  6.302E-12  5.430E-12  5.365E-12  5.481E-12  3.893E-12
 44    2  1137.89     2.872E-13  2.572E-12  8.484E-13  1.692E-12  3.312E-12  4.448E-12  3.836E-12  3.793E-12  3.887E-12  2.770E-12
 45    2  1140.61     1.914E-13  1.713E-12  4.981E-13  9.457E-13  1.796E-12  2.566E-12  2.168E-12  2.128E-12  2.186E-12  1.565E-12


INT ZONE  MID PT.        GROU 11       GROU 12       GROU 13       GROU 14       GROU 15       GROU 16
  1    1    65.47     3.758E-10  4.038E-10  3.547E-10  2.988E-10  8.672E-10  1.035E-08
  2    1   196.42     3.679E-10  3.953E-10  3.473E-10  2.925E-10  8.491E-10  1.014E-08
  3    1   327.37     3.502E-10  3.763E-10  3.306E-10  2.784E-10  8.082E-10  9.647E-09
  4    1   458.32     3.229E-10  3.470E-10  3.048E-10  2.567E-10  7.452E-10  8.895E-09
  5    1   589.27     2.869E-10  3.082E-10  2.708E-10  2.281E-10  6.620E-10  7.901E-09
  6    1   720.22     2.418E-10  2.598E-10  2.282E-10  1.922E-10  5.580E-10  6.660E-09
  7    1   796.92     2.119E-10  2.277E-10  2.000E-10  1.685E-10  4.891E-10  5.838E-09
  8    1   819.37     2.019E-10  2.169E-10  1.906E-10  1.605E-10  4.659E-10  5.561E-09
  9    1   841.82     1.916E-10  2.058E-10  1.808E-10  1.523E-10  4.421E-10  5.277E-09
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TABLE 7-5 (Cont'd)

TABLE 7-5 (Cont'd)

10	1	86[illegible]7	1.809E-10	1.943E-10	1.707E-10	1.438E-10	4.174E-10	4.981E-09
11	1	88[illegible]	1.699E-10	1.825E-10	1.604E-10	1.351E-10	3.921E-10	4.681E-09
12	1	90[illegible]	1.584E-10	1.702E-10	1.495E-10	1.259E-10	3.656E-10	4.362E-09
13	1	931.61	1.464E-10	1.574E-10	1.382E-10	1.164E-10	3.380E-10	4.037E-09
14	1	954.06	1.338E-10	1.437E-10	1.263E-10	1.063E-10	3.086E-10	3.680E-09
15	1	976.51	1.201E-10	1.290E-10	1.133E-10	9.546E-11	2.771E-10	3.314E-09
16	1	998.96	1.053E-10	1.131E-10	9.940E-11	8.371E-11	2.429E-10	2.890E-09
17	1	1021.41	8.867E-11	9.528E-11	8.370E-11	7.051E-11	2.047E-10	2.455E-09
18	1	1043.86	7.087E-11	7.614E-11	6.689E-11	5.633E-11	1.634E-10	1.930E-09
19	1	1066.31	5.007E-11	5.379E-11	4.724E-11	3.979E-11	1.155E-10	1.401E-09
20	1	1088.75	2.731E-11	2.935E-11	2.580E-11	2.173E-11	6.303E-11	7.223E-10
21	2	1100.41	1.626E-11	1.749E-11	1.551E-11	1.273E-11	3.633E-11	2.046E-10
22	2	1101.26	1.597E-11	1.718E-11	1.528E-11	1.246E-11	3.561E-11	1.895E-10
23	2	1102.12	1.569E-11	1.687E-11	1.503E-11	1.220E-11	3.490E-11	1.759E-10
24	2	1102.98	1.540E-11	1.655E-11	1.478E-11	1.195E-11	3.419E-11	1.637E-10
25	2	1103.84	1.512E-11	1.624E-11	1.452E-11	1.171E-11	3.348E-11	1.524E-10
26	2	1104.69	1.483E-11	1.592E-11	1.425E-11	1.146E-11	3.276E-11	1.420E-10
27	2	1105.55	1.454E-11	1.561E-11	1.398E-11	1.122E-11	3.205E-11	1.326E-10
28	2	1106.41	1.425E-11	1.529E-11	1.370E-11	1.098E-11	3.135E-11	1.238E-10
29	2	1107.26	1.396E-11	1.497E-11	1.342E-11	1.074E-11	3.064E-11	1.158E-10
30	2	1108.12	1.366E-11	1.465E-11	1.314E-11	1.050E-11	2.994E-11	1.083E-10
31	2	1108.98	1.337E-11	1.433E-11	1.286E-11	1.026E-11	2.924E-11	1.015E-10
32	2	1109.84	1.307E-11	1.400E-11	1.257E-11	1.002E-11	2.854E-11	9.512E-11
33	2	1110.69	1.277E-11	1.367E-11	1.228E-11	9.783E-12	2.784E-11	8.924E-11
34	2	1111.55	1.246E-11	1.334E-11	1.198E-11	9.542E-12	2.713E-11	8.379E-11
35	2	1113.34	1.181E-11	1.264E-11	1.136E-11	9.038E-12	2.567E-11	7.395E-11
36	2	1116.07	1.080E-11	1.156E-11	1.040E-11	8.267E-12	2.344E-11	6.095E-11
37	2	1118.80	9.777E-12	1.047E-11	9.424E-12	7.489E-12	2.121E-11	5.045E-11
38	2	1121.52	8.736E-12	9.365E-12	8.436E-12	6.705E-12	1.898E-11	4.183E-11
39	2	1124.25	7.686E-12	8.251E-12	7.441E-12	5.916E-12	1.674E-11	3.459E-11
40	2	1126.98	6.636E-12	7.134E-12	6.442E-12	5.125E-12	1.450E-11	2.839E-11
41	2	1129.71	5.590E-12	6.019E-12	5.442E-12	4.333E-12	1.226E-11	2.296E-11
42	2	1132.43	4.553E-12	4.909E-12	4.444E-12	3.542E-12	1.003E-11	1.809E-11
43	2	1135.16	3.523E-12	3.804E-12	3.443E-12	2.751E-12	7.783E-12	1.360E-11
44	2	1137.89	2.510E-12	2.713E-12	2.463E-12	1.969E-12	5.562E-12	9.438E-12
45	2	1140.61	1.420E-12	1.535E-12	1.398E-12	1.123E-12	3.147E-12	5.135E-12

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wMMARY FOR ZONE  1 BY GROUP INCLUDING SUM FOR ALL GROUPS IN LINE  17          ZONE VOLUME=  5.57495E+09

GRP.  FIX  SOURCE  FISS SOURCE  IN  SCATTER  SLF SCATTER  OUT SCATTER   ABSORPTION    LEAKAGE      BALANCE
  1  0.00000E+00  2.04000E-01  0.00000E+00  6.73625E-02  2.51048E-01  3.14304E-03  1.18628E-04  8.90227E-01
  2  0.00000E+00  3.44000E-01  1.49046E-01  2.10592E-01  5.77544E-01  2.02558E-04  1.44712E-04  9.20775E-01
  3  0.00000E+00  1.68000E-01  2.60163E-01  1.56453E-01  4.69467E-01  8.62014E-05  4.85708E-05  9.53842E-01
  4  0.00000E+00  1.80000E-01  5.19106E-01  3.48228E-01  7.43361E-01  1.07558E-04  3.84360E-05  9.69222E-01
  5  0.00000E+00  9.00000E-02  8.47671E-01  6.80197E-01  9.59763E-01  1.05014E-04  5.20873E-06  9.88299E-01
  6  0.00000E+00  1.40000E-02  1.02420E+00  8.32311E-01  1.04156E+00  1.06908E-04 -7.08966E-06  9.98339E-01
  7  0.00000E+00  0.00000E+00  1.02941E+00  7.74096E-01  1.02928E+00  1.28601E-04  3.14276E-06  1.00000E+00
  8  0.00000E+00  0.00000E+00  1.01847E+00  7.35164E-01  1.01814E+00  3.25109E-04  5.08247E-06  1.00000E+00
  9  0.00000E+00  0.00000E+00  1.01868E+00  7.39213E-01  1.01744E+00  1.23297E-03  7.24865E-06  1.00000E+00
 10  0.00000E+00  0.00000E+00  8.71648E-01  3.65904E-01  8.69086E-01  2.55560E-03  5.79687E-06  1.00000E+00
 11  0.00000E+00  0.00000E+00  8.30213E-01  3.01530E-01  8.26845E-01  3.36131E-03  6.03044E-06  1.00000E+00
 12  0.00000E+00  0.00000E+00  8.70843E-01  3.44066E-01  8.67581E-01  3.25462E-03  6.44652E-06  1.00000E+00
 13  0.00000E+00  0.00000E+00  8.28978E-01  2.70474E-01  7.94697E-01  3.42752E-02  4.92607E-06  1.00000E+00
 14  0.00000E+00  0.00000E+00  7.28405E-01  3.36610E-01  7.17754E-01  1.06438E-02  5.79873E-06  1.00000E+00
 15  0.00000E+00  0.00000E+00  8.95932E-01  3.33493E+00  8.04925E-01  9.09889E-02  1.46061E-05  1.00000E+00
 16  0.00000E+00  0.00000E+00  1.09096E+00  8.60165E+01  7.26795E-02  1.01794E+00  2.87138E-04  1.00002E+00
 17  0.00000E+00  1.00000E+00  1.19837E+01  9.55136E+01  1.20612E+01  1.16846E+00  6.94683E-04  9.90593E-01

GRP.  RT BDY FLUX   RT BDY J+     RT BDY J    RT LEAKAGE   LFT LEAKAGE   FISS RATE    TOTAL FLUX    DENSITY
  1  2.16637E-11  9.70025E-12  7.80203E-12  1.18628E-04  0.00000E+00  3.64413E-04  2.27847E+00  2.27847E+00
  2  4.97629E-11  1.77936E-11  9.51759E-12  1.44712E-04  0.00000E+00  6.24409E-04  4.27341E+00  4.27341E+00
  3  2.19016E-11  7.30092E-12  3.19446E-12  4.85708E-05  0.00000E+00  2.52268E-04  1.90951E+00  1.90951E+00
  4  3.53407E-11  1.03923E-11  2.52790E-12  3.84360E-05  0.00000E+00  2.94929E-04  2.75542E+00  2.75542E+00
  5  4.14334E-11  1.07950E-11  3.42573E-13  5.20873E-06  0.00000E+00  2.59913E-04  2.85743E+00  2.85743E+00
  6  3.25192E-11  8.02325E-12 -4.66281E-13 -7.08966E-06  0.00000E+00  2.38358E-04  2.10769E+00  2.10769E+00
  7  2.50703E-11  6.47549E-12  2.06696E-13  3.14276E-06  0.00000E+00  2.45828E-04  1.68737E+00  1.68737E+00
  8  2.32144E-11  6.06665E-12  3.34270E-13  5.08247E-06  0.00000E+00  3.53905E-04  1.56309E+00  1.56309E+00
  9  2.30900E-11  6.10677E-12  4.76737E-13  7.24865E-06  0.00000E+00  1.26932E-03  1.56634E+00  1.56634E+00
 10  1.61812E-11  4.30309E-12  3.81255E-13  5.79687E-06  0.00000E+00  2.30844E-03  1.10143E+00  1.10143E+00
 11  1.46957E-11  3.93397E-12  3.96617E-13  6.03044E-06  0.00000E+00  3.80643E-03  1.00673E+00  1.00673E+00
 12  1.58027E-11  4.22877E-12  4.23981E-13  6.44652E-06  0.00000E+00  2.64846E-03  1.08170E+00  1.08170E+00
 13  1.39932E-11  3.71765E-12  3.23983E-13  4.92607E-06  0.00000E+00  9.91065E-04  9.50283E-01  9.50283E-01
 14  1.15239E-11  3.12196E-12  3.81377E-13  5.79373E-06  0.00000E+00  4.84705E-03  8.00409E-01  8.00409E-01
 15  3.28577E-11  8.83023E-12  9.60630E-13  1.46061E-05  0.00000E+00  1.32528E-01  2.32328E+00  2.32328E+00
 16  1.90402E-10  5.82816E-11  1.88848E-11  2.87138E-04  0.00000E+00  1.04983E+00  2.77228E+01  2.77228E+01
 17  5.69453E-10  1.69071E-10  4.56886E-11  6.94683E-04  0.00000E+00  1.20091E+00  5.59853E+01  5.59853E+01
```

TABLE 7-5 (Cont'd)

```
wMMARY FOR ZONE  2 BY GROUP INCLUDING SUM FOR ALL GROUPS IN LINE  17              ZONE VOLUME=  6.63294E+08 .

GRP.  FIX  SOURCE  FISS SOURCE  IN  SCATTER  SLF SCATTER  OUT SCATTER   ABSORPTION    LEAKAGE       BALANCE
   1  0.00000E+00  0.00000E+00  0.00000E+00  2.44735E-04  1.20627E-04  1.01548E-05 -1.17194E-04  1.00000E+00
   2  0.00000E+00  0.00000E+00  1.00756E-04  1.25374E-03  2.48830E-04  5.97416E-07 -1.33270E-04  9.28996E-01
   3  0.00000E+00  0.00000E+00  2.13157E-04  7.76351E-04  2.64371E-04  1.10686E-07 -4.59670E-05  9.87588E-01
   4  0.00000E+00  0.00000E+00  2.88706E-04  1.96117E-03  3.26358E-04  2.05352E-07 -3.37674E-05  9.92965E-01
   5  0.00000E+00  0.00000E+00  3.39704E-04  3.62681E-03  3.36167E-04  3.36180E-07  3.24993E-06  9.99927E-01
   6  0.00000E+00  0.00000E+00  3.34870E-04  2.63339E-03  3.11318E-04  8.43518E-07  2.06115E-05  1.00314E+00
   7  0.00000E+00  0.00000E+00  3.02396E-04  2.35878E-03  2.93650E-04  4.62531E-07  7.69049E-06  1.00098E+00
   8  0.00000E+00  0.00000E+00  2.91689E-04  2.28682E-03  2.85228E-04  6.49597E-07  5.37213E-06  1.00075E+00
   9  0.00000E+00  0.00000E+00  2.85058E-04  2.23858E-03  2.79210E-04  2.10668E-06  3.51648E-06  1.00040E+00
  10  0.00000E+00  0.00000E+00  2.60104E-04  1.46709E-03  2.55819E-04  2.18107E-06  1.96612E-06  1.00026E+00
  11  0.00000E+00  0.00000E+00  2.51043E-04  1.29980E-03  2.47574E-04  2.47178E-06  1.01441E-06  9.99975E-01
  12  0.00000E+00  0.00000E+00  2.54083E-04  1.41363E-03  2.48283E-04  4.59553E-06  1.15434E-06  1.00010E+00
  13  0.00000E+00  0.00000E+00  2.43558E-04  1.22610E-03  2.33947E-04  7.44103E-06  2.04140E-06  1.00026E+00
  14  0.00000E+00  0.00000E+00  2.24538E-04  9.53073E-04  2.15058E-04  9.76547E-06 -1.69895E-07  9.99741E-01
  15  0.00000E+00  0.00000E+00  2.37506E-04  3.26926E-03  1.84829E-04  5.20917E-05  9.31663E-07  9.99270E-01
  16  0.00000E+00  0.00000E+00  2.19656E-04  1.20599E-02  1.04509E-04  4.10292E-04 -2.63081E-04  9.31977E-01
  17  0.00000E+00  0.00000E+00  3.84683E-03  3.90692E-02  3.95578E-03  5.04305E-04 -5.45901E-04  9.91321E-01

GRP.  RT BDY FLUX   RT BDY J+     RT BDY J     RT LEAKAGE   LFT LEAKAGE   FISS RATE    TOTAL FLUX     DENSITY
   1  1.30783E-13  8.74478E-14  8.74478E-14  1.43309E-06  1.13628E-04  0.00000E+00  2.93288E-03  2.93288E-03
   2  1.14193E-12  6.98228E-13  6.98228E-13  1.14426E-05  1.44712E-04  0.00000E+00  1.11321E-02  1.11321E-02
   3  2.74959E-13  1.58885E-13  1.58885E-13  2.60380E-06  4.85708E-05  0.00000E+00  4.27073E-03  4.27073E-03
   4  4.90114E-13  2.84876E-13  2.84876E-13  4.66855E-06  3.84360E-05  0.00000E+00  8.27587E-03  8.27587E-03
   5  8.87830E-13  5.16150E-13  5.16150E-13  8.45866E-06  5.20873E-06  0.00000E+00  1.31442E-02  1.31442E-02
   6  1.41038E-12  8.25109E-13  8.25109E-13  1.35219E-05 -7.08966E-06  0.00000E+00  1.29925E-02  1.29925E-02
   7  1.13885E-12  6.61048E-13  6.61048E-13  1.08333E-05  3.14276E-06  0.00000E+00  1.00958E-02  1.00958E-02
   8  1.10060E-12  6.37943E-13  6.37943E-13  1.04546E-05  5.08247E-06  0.00000E+00  9.40596E-03  9.40596E-03
   9  1.13364E-12  6.56891E-13  6.56891E-13  1.07651E-05  7.24865E-06  0.00000E+00  9.27727E-03  9.27727E-03
  10  8.17586E-13  4.73700E-13  4.73700E-13  7.76299E-06  5.79687E-06  0.00000E+00  6.47012E-03  6.47012E-03
  11  7.42362E-13  4.29879E-13  4.29879E-13  7.04485E-06  6.03044E-06  0.00000E+00  5.80702E-03  5.80702E-03
  12  8.01288E-13  4.63807E-13  4.63807E-13  7.60086E-06  6.44652E-06  0.00000E+00  6.23304E-03  6.23304E-03
  13  7.34395E-13  4.25157E-13  4.25157E-13  6.96747E-06  4.92607E-06  0.00000E+00  5.59979E-03  5.59979E-03
  14  5.93231E-13  3.43473E-13  3.43473E-13  5.62883E-06  5.79873E-06  0.00000E+00  4.47891E-03  4.47891E-03
  15  1.63920E-12  9.48119E-13  9.48119E-13  1.55378E-05  1.46061E-05  0.00000E+00  1.27303E-02  1.27303E-02
  16  2.53068E-12  1.46799E-12  1.46799E-12  2.40575E-05  2.87138E-04  0.00000E+00  4.01966E-02  4.01966E-02
  17  1.55678E-11  9.07871E-12  9.07871E-12  1.48782E-04  6.94683E-04  0.00000E+00  1.63043E-01  1.63043E-01
```

TABLE 7-5 (Cont'd)

```
wMMARY FOR SYSTEM        INCLUDING SUM OVER ALL GROUPS IN LINE  17                SYSTEM VOLUME=  6.23825E+09

 GRP.  FIX  SOURCE  FISS SOURCE  IN  SCATTER  SLF SCATTER  OUT SCATTER   ABSORPTION    LEAKAGE      BALANCE
    1  0.00000E+00  2.04000E-01  0.00000E+00  6.76072E-02  2.51169E-01  3.15320E-03  1.43309E-06  8.90201E-01
    2  0.00000E+00  3.44000E-01  1.49147E-01  2.11846E-01  5.77793E-01  2.03156E-04  1.14426E-05  9.20777E-01
    3  0.00000E+00  1.68000E-01  2.60376E-01  1.57230E-01  4.69732E-01  8.63121E-05  2.60380E-06  9.53858E-01
    4  0.00000E+00  1.80000E-01  5.19395E-01  3.50189E-01  7.43688E-01  1.07763E-04  4.66855E-06  9.69231E-01
    5  0.00000E+00  9.00000E-02  8.48010E-01  6.83824E-01  9.60099E-01  1.05351E-04  8.45866E-06  9.88303E-01
    6  0.00000E+00  1.40000E-02  1.02454E+00  8.34945E-01  1.04187E+00  1.07751E-04  1.35219E-05  9.98341E-01
    7  0.00000E+00  0.00000E+00  1.02971E+00  7.76455E-01  1.02957E+00  1.29063E-04  1.08333E-05  1.00000E+00
    8  0.00000E+00  0.00000E+00  1.01876E+00  7.37451E-01  1.01842E+00  3.25758E-04  1.04546E-05  1.00000E+00
    9  0.00000E+00  0.00000E+00  1.01896E+00  7.41451E-01  1.01771E+00  1.23508E-03  1.07651E-05  1.00000E+00
   10  0.00000E+00  0.00000E+00  8.71909E-01  3.67371E-01  8.69342E-01  2.55778E-03  7.76299E-06  1.00000E+00
   11  0.00000E+00  0.00000E+00  8.30464E-01  3.02830E-01  8.27092E-01  3.36378E-03  7.04485E-06  1.00000E+00
   12  0.00000E+00  0.00000E+00  8.71097E-01  3.45479E-01  8.67829E-01  3.25922E-03  7.60086E-06  1.00000E+00
   13  0.00000E+00  0.00000E+00  8.29221E-01  2.71700E-01  7.94931E-01  3.42826E-02  6.96747E-06  1.00000E+00
   14  0.00000E+00  0.00000E+00  7.28629E-01  3.37563E-01  7.17969E-01  1.06536E-02  5.62883E-06  1.00000E+00
   15  0.00000E+00  0.00000E+00  8.96169E-01  3.33820E+00  8.05110E-01  9.10409E-02  1.55378E-05  1.00000E+00
   16  0.00000E+00  0.00000E+00  1.09118E+00  8.60285E+01  7.27840E-02  1.01835E+00  2.40575E-05  1.00001E+00
   17  0.00000E+00  1.00000E+00  1.19876E+01  9.55527E+01  1.20651E+01  1.16896E+00  1.48782E-04  9.90593E-01

 GRP.  RT BDY FLUX   RT BDY J+    RT BDY J     RT LEAKAGE   LFT LEAKAGE   FISS RATE    TOTAL FLUX    DENSITY
    1  1.30783E-13  8.74478E-14  8.74478E-14  1.43309E-06  0.00000E+00  3.64413E-04  2.28140E+00  2.28140E+00
    2  1.14193E-12  6.98228E-13  6.98228E-13  1.14426E-05  0.00000E+00  6.24409E-04  4.28455E+00  4.28455E+00
    3  2.74959E-13  1.58885E-13  1.58885E-13  2.60380E-06  0.00000E+00  2.52268E-04  1.91378E+00  1.91378E+00
    4  4.90114E-13  2.84876E-13  2.84876E-13  4.66855E-06  0.00000E+00  2.94929E-04  2.76369E+00  2.76369E+00
    5  8.87830E-13  5.16150E-13  5.16150E-13  8.45866E-06  0.00000E+00  2.59913E-04  2.87058E+00  2.87058E+00
    6  1.41033E-12  8.25109E-13  8.25109E-13  1.35219E-05  0.00000E+00  2.38358E-04  2.12068E+00  2.12068E+00
    7  1.13885E-12  6.61048E-13  6.61048E-13  1.08333E-05  0.00000E+00  2.45828E-04  1.69746E+00  1.69746E+00
    8  1.10060E-12  6.37943E-13  6.37943E-13  1.04546E-05  0.00000E+00  3.53905E-04  1.57249E+00  1.57249E+00
    9  1.13364E-12  6.56891E-13  6.56891E-13  1.07651E-05  0.00000E+00  1.26932E-03  1.57562E+00  1.57562E+00
   10  8.17586E-13  4.73700E-13  4.73700E-13  7.76299E-06  0.00000E+00  2.30844E-03  1.10790E+00  1.10790E+00
   11  7.42362E-13  4.29879E-13  4.29879E-13  7.04485E-06  0.00000E+00  3.80643E-03  1.01253E+00  1.01253E+00
   12  8.01288E-13  4.63807E-13  4.63807E-13  7.60086E-06  0.00000E+00  2.64846E-03  1.08794E+00  1.08794E+00
   13  7.34395E-13  4.25157E-13  4.25157E-13  6.96747E-06  0.00000E+00  9.91065E-04  9.55883E-01  9.55883E-01
   14  5.93231E-13  3.43473E-13  3.43473E-13  5.62883E-06  0.00000E+00  4.84705E-03  8.04838E-01  8.04838E-01
   15  1.63920E-12  9.48119E-13  9.48119E-13  1.55378E-05  0.00000E+00  1.32528E-01  2.33601E+00  2.33601E+00
   16  2.53068E-12  1.46799E-12  1.46799E-12  2.40575E-05  0.00000E+00  1.04988E+00  2.77630E+01  2.77630E+01
   17  1.55678E-11  9.07871E-12  9.07871E-12  1.48782E-04  0.00000E+00  1.20091E+00  5.61484E+01  5.61484E+01
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TABLE 7-5 (Cont'd)

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                EFFECTIVE ONE-GROUP PARAMETERS

INPUT TRANSVERSE BUCKLING         (1/CM**2) = -4.43788E-02
OUTPUT RADIAL BUCKLING BG**2      (1/CM**2) =  1.21623E-03
RADIAL BUCKLING FOR CRITICALITY(1/CM**2) =  0.00000E+00
TOTAL BUCKLING FOR CRITICALITY (1/CM**2) =
EFFECTIVE DIFFUSION COEFFICIENT    ( CM ) =  4.67865E-01
EFFECTIVE MIGRATION AREA        ( CM**2 ) =  6.74184E+01
INFINITE MULTIPLICATION FACTOR    - - -   =  1.02733E+00

     THIS PROBLEM REQUIRED     733.00 SECONDS
```

APPENDIX A

EXAMINATION OF SPECIMENS FROM CAREY SALT MINE

The examination of specimens from the Carey Salt Mine was conducted by ORNL as part of previous salt repository investigations. The following discussion is excerpted from an ORNL intra-laboratory correspondence from J. C. Griess to A. L. Boch dated October 25, 1973:

A pillar heater pipe and several panels from the Carey Salt Mine in Kansas have been examined for the purpose of determining the type and extent of corrosion. The pillar heater pipe was a 4-in. Sch. 40 carbon steel pipe 13 ft. long and had been buried in the salt for about seven years. Initially a gap existed between the salt and the pipe. About a year after burial, water was added to a hole adjacent to the pipe, and apparently some of the water found its way into the region between the salt and the pipe. Visual examination of the pipe indicated that the water level had been about one-half the pipe depth. The lower half of the pipe was very heavily corroded, though much less corrosion was noted on the top half. The presence of water caused the salt to crystallize around the pipe and make removal difficult. When the pipe was removed, no evidence of water was found in the hole. Thus the period of time when the pipe was exposed to wet salt is not known. Similar pipes exposed under the same conditions at other locations did not show the heavy attack that was present on this pipe, and their removal from the salt was relatively easy.

The pillar pipe was cut into sections about 4 in. long and two pieces from the lower half and two from the upper half were descaled in inhibited 5% hydrochloric acid. Both pieces from the lower half were covered with an irregular, thick, red oxide, much of which was removed by gentle tapping with a hammer; most of the rest was removed by the acid. In addition to heavy general attack, many large hemispherical pits were randomly distributed on the surface of the pipe. While none of the pits penetrated the pipe wall (0.237 in. initially), several came within a few mils of doing so.

Considering the statistical distribution of pit depths, a statement that the maximum pit depth was about equal to the wall thickness is reasonably accurate.

The two pieces cut from the upper half of the pipe were uniformly covered with rust. After descaling only minor attack was found. Numerous small pits were present on the surface, but none was deeper than a few mils. Considering the duration of the exposure (7 yrs.), the attack on the upper half of the pipe was of only minor significance. Presumably the attack observed on the top of this pipe was typical of that on the other pipes which were not exposed to water.

Three sets of panels that were exposed in the same mine for about three years were also examined. Each set consisted of three panels, each 6-1/4 in. x 6-1/4 x either 1/8 or 1/16 in. One was 304 stainless steel; one was galvanized steel; and the third was carbon steel. The panels were not weighed before exposure, and only qualitative results were obtained.

One set of specimens was exposed to the mine atmosphere in a room well removed from the shaft. In this environment neither the stainless steel nor the galvanized steel showed any visible attack. The carbon steel was covered with red rust, which was easily removed in the inhibited acid solution. No localized attack was noted, and corrosion damage was inconsequential.

A second set was exposed in a similar location in the mine, except in this case the panels were buried in the salt for the three-year period. After removal of the salt, the stainless steel panel had a few rusty areas, but no significant localized attack was observed. The galvanized steel was heavily encrusted with salt, the removal of which revealed a whitish-grey material, probably zinc oxide. In a few areas some metallic zinc still remained under the white oxide. This panel was not descaled, but destructive corrosion appeared to be minimal. The zinc protected the steel, but corrosion undoubtedly would have begun after the last small amount of remaining zinc had been oxidized. The carbon steel panel was covered with a relatively heavy coating of rust, which was mostly removed in the inhibited acid. The surface of the steel was covered with small pits, which were only about 1 to 3 mils deep. Overall the attack was not great and was similar to, but less than, that observed on the upper part of the pillar pipe.

The third set of panels was exposed near the main shaft where the humidity was high. In addition, brine solutions must have actually dripped on the panels for some time since all three panels were covered with heavy adherent deposits of salt, which contained finely divided solid impurities. The 304 stainless steel had a few thinly rusted areas and one large area where the surface appeared to be slightly roughened. However, corrosion damage was insignificant. After removal of the salt deposit the galvanized panel was mostly grey with areas which were heavily rusted. No trace of metallic zinc could be found. All deposits were removed in the inhibited acid revealing a smooth surface with no significant attack. Apparently the zinc had protected the steel for most of the exposure. The carbon steel panel was covered with very heavy deposits of red oxide, most of which was detached by light tapping with a hammer. Acid treatment removed most of the rest. The panel was reduced in thickness by about 20 mils (10 mils on each side) and in addition contained numerous round-bottom pits 3 to 5 mils deep. Thus the average penetration rate was about 5 mils per year.

The results of this examination show the severity of the corrosion of carbon steel in wet salt. Based on the above information, it is not possible to arrive at an accurate corrosion-penetration rate since the time that water was in contact with the pillar pipe is not known. However, even under the most favorable conditions, the maximum penetration rate at the deepest pit was about 40 mils per year (0.237 in. in six yrs.). It should be noted that this attack occurred at mine temperature. At higher temperature higher penetration rates would be expected. How fast the corrosion rate increases with temperature cannot be predicted with any degree of confidence, since much depends on the nature of the films that form and on the changes in solution chemistry with temperature.

Galvanizing steel surfaces provides temporary protection. However, the protection lasts only as long as the zinc, and in aggressive environments the zinc is soon consumed. Since zinc affords protection to steel, one might conclude that carbon steel could be cathodically protected in the same way that many buried structures are protected. However, the electrical conductivity of the salt is so low that cathodic protection from an exterior source seems impractical.

The corrosion resistance of stainless steel in the above environments was excellent. However, at higher temperatures in the presence of water, stress-corrosion cracking would be expected. In addition, if the brine becomes slightly acid, pitting of the stainless steel would be expected even at mine temperature.

APPENDIX B

RADIATION REDUCTION COST-BENEFIT ANALYSIS BASIS

Summary

The basic approach is to calculate the total direct cost (TDC) by equating the man-rem/year reduction multiplied by a dollar per man-rem justifiable cost with the annual fixed cost. At a given interest and plant life (years), the TDC can then be calculated using the equations provided below. The method developed is consistent with that recommended by Reg. Guide 1.110.

Details

The basic formula utilized is that the Total Annual Cost (TAC) equals:

TAC = AFC + ADC + AMC

where AFC is the annual fixed cost
ADC is the annual operating cost
AMC is the annual maintenance cost

Since this calculation is performed without specific equipment in mind, ADC and AMC can be set equal to zero to provide a bounding value for the calculation.

Therefore, TAC = AFC. The AFC, per Reg. Guide 1.110, can thus be equated to the total direct cost (TDC) as follows:

$$\text{TCC} = \frac{\text{AFC}}{\text{CRF}} = \frac{\text{AFC}}{\dfrac{i(1+i)^n}{[(1+i)^n 1]}}$$

where i – interest rate
n = capitalization period.

For the case of an assumed worth of \$5000 per man-rem (i.e. a conservative estimate of justifiable cost expenditures based on health related impacts) i = 10%, n = 30 years and an indirect cost factor of 1.5, the justifiable associated direct costs for a 500 man-rem exposure reduction is:

$$\text{TDC} = \frac{(500) \times (1000)}{(1.5)\ (0.1061)} \sim \$3.1 \times 10^6;\ \text{TDC} \times 1.5 \sim \$4.7 \times 10^6 \text{ Total Capital Cost}$$

Reduction from 1.6 man-rem/year (off-site) to zero has an associated justifiable total cost of:

$$\frac{(1.6)\ (1000)}{(1.5)\ (.1061)} \sim \$10 \times 10^3;\ \text{TDC} \times 1.5 \sim \$15 \times 10^3 \text{ Total Capital Cost}$$

This means that the maximum justifiable expenditure for reducing the dose budget by 500 man-rems would be $23.5 million for the above assumptions.

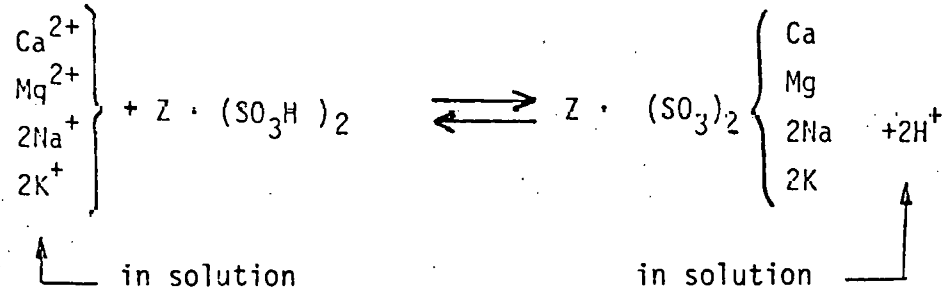
APPENDIX C

DISCUSSION OF THE ION-EXCHANGE PROCESS FOR REMOVING RADIOACTIVE CONTAMINATION FROM WATER

Salts dissolved in water dissociate into positively-charged ions (cations) and negatively-charged ions (anions). The solutions are electrically neutral, that is, the number of positive charges present balances the number of negative charges.

Ion exchangers also contain cations and anions in a condition of electroneutrality. The exchange media differ from solutions in that only one of the two ionic species is mobile. For example, a typical sulfonic acid cation exchanger has immobile ion-exchange sites consisting of the anionic radicals SO_3^-, mobile cations such as H^+ or Na^+ are attached to these sites, and these cations may be exchanged in an ion exchange reaction An anion exchanger similarly has immobile cationic sites with mobile hydroxide anions attached that can also be exchanged.

Ion exchange between a solid exchanger and ions in water takes place without structural change in the solid material. The ions in the solution diffuse rapidly into the molecular network of the exchanger, reach the ion exchange sites, and the interchange of ions occurs. The ions in the exchanger in turn diffuse out of the exchanger into the solution. Cation exchanges with hydrogen in a cation exchanger with the functional group consisting of sulfonic acid SO_3H are given in the following equation in ionic form:



$$\underbrace{\left.\begin{matrix}Ca^{2+}\\Mg^{2+}\\2Na^{+}\\2K^{+}\end{matrix}\right\}}_{\text{in solution}} + Z\cdot(SO_3H)_2 \rightleftharpoons Z\cdot(SO_3)_2\left\{\begin{matrix}Ca\\Mg\\2Na\\2K\end{matrix}\right. \underbrace{+2H^+}_{\text{in solution}}$$

in which Z represents the matrix of the exchanger.

The reaction is reversible, with the reaction proceeding to the right when the water is being treated and to the left during regeneration with acid. During the regeneration phase, the acid removes the cations from the exchanger and restores the exchanger to its hydrogen form. Similar reversible ionic transfers occur in anion exchangers. In this case, the exchanger is regenerated with a strongly basic solution such as caustic soda (NaOH).

The reaction equations indicate that a complete exchange of cations or anions could occur. Unfortunately, even in efficient ion-exchange beds, some of the ions escape into the effluent. In downflow operation the unwanted ions are progressively removed as the water percolates down through the bed. As a result, the exchanger captures more of the ions at the top than at the bottom. The purity of the exchanger at the bottom determines the purity of the effluent, because as the liquid percolates down, the ions decrease in concentration. The driving force for purification is the difference in concentration between the ions in the liquid and those in the lowest part of the exchanger bed. The bottom region of an ion-exchange medium generally can not be sufficiently well regenerated to provide for complete freedom of migration for the unwanted ions, and some ions pass through the bed as leakage.

In general, at low concentrations of ions in the water, the divalent ions are held on the exchange bed more tightly than are the monovalent, and the trivalent are held more tightly than the divalent. The ion-exchange resins exhibit relative selectivities for cations in the same valence group such that in many acid forms of ion exchangers, the major cations are exchanged in the series:

$$Ba^{2+} > Ca^{2+} > Mg^{2+} > NH_4^+ > K^+ > Na^+ > H^+$$

At low concentrations, the cation exchangers select cations such as Ca or Mg in preference to Na. During the service run, an ion-exchange bed operated in the downflow mode tends to develop a stratified form. After regeneration, the bed has a tendency to retain some of the monovalent ions in the lower region of the bed as complete regeneration requires excessive quantities of acid in the regeneration cycle. When the bed is returned to service after regeneration, the monovalent cations held in the lower region of the bed may be reexchanged by divalent cations or monovalent ions that are more tightly held on the exchange medium. These displaced cations may then be swept into the effluent stream and remain as contaminants in the process water. Anion-exchange beds exhibit similar behavior in regard to selectivity for specific anions. For this reason, ion exchangers are placed in series such that the progressively decontaminated feed streams can contact ion exchange media that are nearly free of residual unwanted ions, and the concentration gradient between the ions to be removed from the water and the exchange bed can then be maintained.