SAFETY ANALYSIS OF A LIQUID RADIOACTIVE WASTE EVAPORATOR

A Thesis

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TABLE OF CONTENTS

	Page
ACKNOWLEDGEMENTS	ii
LIST OF TABLES	iv
LIST OF FIGURES	vi
ABSTRACT	vii
INTRODUCTION	1
CHAPTER 1: Site Description	3
CHAPTER 2: Design Features to Mitigate Potential Accidents	9
CHAPTER 3: Radiological Source Terms	23
CHAPTER 4: Evaporator Shielding	30
CHAPTER 5: Atmospheric Dispersion	63
CHAPTER 6: Conclusions and Recommendations	83
REFERENCES	85
APPENDIX A. Buildup Factor	87
APPENDIX B. Hand Calculation	88
APPENDIX C. The QAD Computer Code	89
VITA	99

LIST OF TABLES

Table		Page
3-1	Non-precipitable Radionuclides	26
3-2	Results of Multichannel Analysis of Precipitable Radionuclides	27
3-3	Abnormal Incident, Isotopes and Concentrations (μCi/ml)	28
3-4	MHA Concentrations	29
4-1	Correlation of Sludge Height to Dose Rate	46
4-2	Correlation of Dose Rates to the Volume of Processed Liquid Waste	48
4-3	Correlation of Dose Rate to Amount of ^{137}Cs	50
4-4	Volume of Waste Necessary to Produce a Dose Rate of 50 mR/hr	52
4-5	Hand Calculation Parameters	54
4-6	Data Points for Hand Calculation and QAD Code	57
4-7	Other Parameters for the Hand Calculation	59
4-8	Model Description for the QAD Code	61
4-9	Atomic Parameters for QAD Model	62
5-1	Comparison of CHI Values for Tritium to Concentration Guide Limits at Indicated Distances from the RLWTF	74
5-2	Abnormal Incident Values of CHI (Ci/m^3) for Specified Isotopes at Indicated Distances from the RLWTF	75
5-3	Concentration Guides (CG) for Specified Isotopes	76
5-4	Abnormal Incident Ratio of CHI Values to Concentration Guides at Indicated Distances from RLWTF	77

LIST OF TABLES (CONTINUED)

Table		Page
5-5	Whole Body Doses from Inhalation and Submersion MHA Conditions.	78
5-6	MHA, Total Dose from Inhalation and Submersion for Specified Isotopes at the Site Boundary (5000 m)	79
5-7	Dose Commitment Factors in Units of mrem in 50 Years Per Curie Inhaled in One Year or Less	80
5-8	Values for the Breathing Rate (BR) and Exposure Time (t) for Receptors at the Indicated Distances from RLWTF	81

LIST OF FIGURES

Figure		Page
2-1	Process Air System.	19
2-2	Simplified Schematic of the RLWTF Process Liquid System	20
2-3	RLWTF Ground Floor.	21
2-4	RLWTF Second Floor.	22
4-1	Dimensions of RLWTF Drum Evaporator	45
4-2	Dose Rate as a Function of Sludge Height. .	47
4-3	Dose Rate as a Function of the Volume of Liquid Waste Processed Assuming 0.08 μCi/ml ^{137}Cs in the Waste.	49
4-4	Relationship of Dose Rate to the Number of Curies of ^{139}Cs in an Evaporator.	51
4-5	Gallons of Feed Required to Give a Dose Rate of 50 mR/hr as a Function of the Concentration of ^{137}Cs in an Evaporator . .	53
4-6	Model for Hand Calculation Sludge Levels in Drum Evaporator	55
4-7	Model for QAD Computer Code Sludge Levels in Drum Evaporator.	56
4-8	Comparison of Hand Calculation to QAD Computer Code Calculation	58
4-9	Geometry for QAD Calculation.	60
5-1	RLWTF Exhaust Plume Analysis.	82

ABSTRACT

Included herein is a preliminary safety analysis for the Radioactive Liquid Waste Treatment Facility (RLWTF) which will be built later this year (1981) at Argonne National Laboratory's site near Idaho Falls, Idaho.

This new facility is being built to process aqueous low level waste generated by other Argonne facilities at the same site. This report describes the site location and analyzes the safety features attendant to the facility housing and process system. It also includes a computer study on the adequacy of the waste evaporator shielding, as well as calculations to determine the effect of atmospheric dispersion of radionuclides vented from the facility.

Results of the atmospheric dispersion calculation indicate that release of radioisotopes to the atmosphere should fall within the applicable limits--during both normal operation and accident conditions. Results from the evaporator computer study are used to delineate the shielding limitations.

INTRODUCTION

It is the author's intention, in writing this thesis, to provide information which will be helpful in the eventual completion of a safety evaluation report for the new low level radioactive waste treatment facility being built at the site of Argonne National Laboratory in Idaho Falls, Idaho. Such a report will be required by the U. S. Department of Energy, for which Argonne is a primary contractor. Although the author has not attempted to write a complete safety analysis report, he has attempted to address some of the topics which will need to be included in the final report. These include a description of the site and a discussion of the safety features attendant to the facility housing and process system. Also included is an assessment of shielding adequacy, and an evaluation of atmospheric dispersion of radionuclides from the facility.

Argonne currently has a low level waste treatment process at their Idaho Falls site. The age and location of the current facility were primary factors in the decision to build a new one. The existing process is twenty years old and is located adjacent to laboratories, administrative offices, and other heavy traffic areas.

Current standards would require siting which inherently provides greater separation from areas housing large numbers of personnel. In addition, the present facility is characterized by a high susceptibility to total shut-down on the failure of a single component. The new facility is designed to circumvent this problem. This matter is discussed in chapter two of this paper.

Chapter one is devoted to a description of the site, and chapter three explains how the source terms for shielding and atmospheric dispersion calculations are derived. The results of shielding calculations are presented in chapter four. The calculations include both a computer study and a hand calculation. An evaluation of airborne radiation emitted by the new facility is described in chapter five. The latter examines the consequences of normal releases, but also discusses accidental emissions.

CHAPTER 1

Site Description

The proposed Radioactive Liquid Waste Treatment Facility (hereinafter referred to as RLWTF) to be located at Argonne National Laboratory's site (ANL-W) near Idaho Falls, Idaho, is a low level liquid waste processing system. The object of this facility is to produce a sludge containing dissolved and particulate radionuclides by removing water from liquid waste. The system is designed to detain the accumulated sludge in six disposable evaporators which can be removed (once the residual radiation reaches a pre-determined level) and replaced. The processing capacity of the system has been determined at 60,000 gallons of liquid waste annually with provisions being made for the future installation of six additional evaporators, doubling the processing capacity to 120,000 gallons per year. By limiting the handling and transferring of radioactivity contaminated waste, this addition will effectively reduce the potential for exposure to personnel, thereby providing a major safety advantage.

Safety features also extend to the building which will house the RLWTF. A two-story, reinforced concrete-structured frame with concrete filler block walls will enclose the control room, the laboratory, and the personnel support areas. The roofing will consist of a structural steel framing system with metal roof deck.

Processing at the building site will begin when tank trucks deliver the radioactive liquid waste to a truck lock on the north side of the building. The liquid will then be off-loaded by pneumatic conveyance to the facility's holding tanks. From the holding tanks the liquid is pumped to the six evaporators, each of which will have a nominal capacity of 30 gallons. To prevent overfill, these evaporators will be fitted with automatic fill valves. Six stacked trays inside the evaporators will act as heat transfer surfaces to vaporize the liquid as it cascades down over the trays. Air drawn from outside the building and heated to 250°F will provide the heat for vaporization.

Hot, low-humidity air will enter the evaporators through a four-inch duct and will circulate over the trays, vaporizing the water and leaving a residue of suspended and dissolved radionuclides. The hot air and water vapor exit the evaporators through a ten-inch exhaust duct cocentric with the intake duct. This exhaust duct routes the moist air (relative humidity 40%) through a 5.5-inch-thick metal knitted moisture separator and then through two banks of absolute High Efficiency Particulate (HEPA) filters. After particles are removed by these filters, the effluent is vented to the atmosphere via a 40-foot high stack.

Eight alternative design concepts were considered and are summarized in the Preliminary Proposal.[1] The decision to design the facility on the disposable evaporator concept was based primarily on the simplicity of the design and on the probability of its minimizing the radiation exposure to personnel. Most of the exposure would be from handling and transferring of the radioactive waste.

The proposal to locate the RLWTF within the confines of ANL-W results in the nearest adjacent structure being the Hot Fuel Examination Facility-North. The Argonne site is located within the southeastern sector of the Idaho National Engineering Laboratory (INEL), which is situated 22 miles west of the city of Idaho Falls, Idaho, and occupies 894 square miles. Encompassed within this area are portions of Bingham, Butte, Clark, and Jefferson counties.

Topographically, the INEL is located along the northwestern edge of the upper Snake River in southeastern Idaho.[2] Mountains as high as 10,000 and 11,000 feet above sea level entirely surround the plain. Bordering the rolling, sometimes broken, Snake River Plain to the north are the Lost River, the Lemhi, and the Beaverhead-Centennial Mountain Ranges. To the south lie two buttes of volcanic origin.

Adjacent to the foothills of the Lemhi and the Lost River Ranges, the INEL rests upon a mixture of lacustrive and alluiral sediments and volcanic rocks. Across previously investigated sites, the depth of the lava rock at ANL-W varies greatly. Load bearing values vary from 3000 to 6000 psf.[3] To include both dead and live loads, the RLWTF has been designed with a foundation bearing value of 3000 psf.

The meteorological description of the INEL site consists of a belt of prevailing westerly winds which channel themselves between the mountains into the valley to produce a generally SW wind. Highest winds are encountered when the prevailing westerlies become strong at the gradient level (approximately 5000 feet above the surface) and are channeled into the confined regions between the mountains.[4]

Another important wind is the down valley wind. This occurs at night when the air along the slopes of the mountains cools at a faster rate than the air of the same elevation over the valley and becomes denser--sinking, consequently, towards the valley. Upon reaching the valley, this dense air flows toward lower levels across the area. This wind can be a NE wind from the main valley, a NW wind from the Birch Creek Valley, or a combination of the two.

At the ANL-W, the average annual wind speed is 7.5 mph at 200 feet above the desert floor. Reference 4 reports the following key parameters for the INEL weather station:

average annual rainfall - 8.5 inches;
average annual snowfall - 28.5 inches;
average maximum temperature in winter - 27°F;
average minimum temperature in winter - 3°F;
average maximum temperature in summer - 87°F;
average minimum temperature in summer - 50°F;
average annual relative humidity - 50%;
minimum relative humidity - 27% (July); and
maximum relative humidity - 79% (Dec.).

The INEL is classified within Seismic Risk Zone Three of the Uniform Building Code. Seismological data records the first earthquake in the Snake River Plain as occurring in 1884. Since, there have been 29 earthquakes in the region which have measured Modified Mercalli Intensities (MMI) of V or greater. Though the region has never suffered a destructive quake (MMIX or higher), the intensities of these 30 earthquakes were sufficient to be noticed by nearly everyone in the area. The strongest recorded earthquake occurred in August, 1959. Its epicenter was estimated to be near the northwest corner of Yellowstone Park, approximately 100 miles northeast of the INEL. Though it caused no structural damage at the INEL, this quake measured 7.1 on the Richter Scale.

Hydrology at the INEL site can be described in terms of very little water entering by surface inflow and none leaving by surface outflow except for minor runoff. Three streams enter the INEL from the northwest and move out of the valley across alluvial fans into sinks in the northwestern and northern portions of the INEL site. Due to seepage, evaporation, and heavy irrigation, the streams are usually dry before reaching the INEL.[5] A natural underground reservoir of water, the Snake River Aquifer is 200 to 800 feet below the soil surface and maintains a lateral flow of 1.3 billion gallons per day. No effluents from the RLWTF will enter or affect the reservoir.

Any possibility of flooding from the Big Lost River has been largely mitigated through the provision of an alternate path for the water to a holding basin at the south end of the INEL. Thus, the possibility of flooding is small, although a local cloudburst might induce heavy flash runoff for a very short time.

CHAPTER 2

Design Features to Mitigate Potential Accidents

For the purpose of discussing the designed safety features, it is convenient to break down the process into three major categories, these include: the facility housing, the process liquid system, and the process air system. In general, the process liquid system is responsible for delivering waste feed to the evaporators, while the process air system provides heated dry air to the evaporators to serve as the evaporative medium. Each of these categories are discussed herein.

Design of the facility housing is an extremely important aspect of the overall safety system. The housing is a two story structure. It is divided into controlled and uncontrolled areas. The uncontrolled areas will provide a region of comparative safety from which safe shutdown of the process may be expected, in the event of an accident involving the release of radiation within the facility housing itself. Separate heating ventilation and air conditioning (HVAC) systems will serve the two regions to insure that an airborne release of radioactive material will not prevent safe shutdown.

Controlled areas include the first and second floor process areas encompassing the evaporators, holding tanks, and process air system. All other regions within the facility are designated as uncontrolled. In general, the controlled areas will be those regions where radioactive material is likely to be present while uncontrolled areas, including the control room and health physics office, will be free of any radioactive substances (see Figures 2-1, 2-2, 2-3, and 2-4).

Two air locks--one on each floor will separate controlled from adjacent uncontrolled areas. The controlled side of the building will be under a negative pressure. This pressure differential will be maintained by an air damper regulating intake to the process area. In the event the pressure differential cannot be maintained this damper will close automatically. Simultaneously, an alarm will sound in the control room. Control room operators will have access to a differential pressure gauge which will display the pressure differences between the two areas.

The process liquid system is conveniently subdivided into three smaller categories--the holding tanks, the circulating system, and the overflow system.

Four holding tanks, with a capacity of 1504 gallons each, will act as resevoirs for contaminated waste water and allow for controlled delivery of the water to the evaporators. As an additional utility, these tanks will aid in cleaning the valves and piping of radioactive particulates. Fresh water can be pumped into the tanks and circulated through the entire system to remove entrained radionuclides. Each tank will be designed for an internal working pressure of 25 psig and 1 psig, external pressure. The design of the stainless steel tanks includes seismic analysis as defined by the Uniform Building Code for Seismic Zone 3.

Waste water enters the holding tanks via pneumatic transferance from tank trucks. This operation will involve some hands-on work. As a safety precaution control valves and holding tank liquid level indicators will be located so that the off-loading operation can be handled conveniently and safely by one man. In route to the holding tanks the waste water goes through a coarse basket-type filter to remove the larger particulates. The removal and replacement of these filters will have to be done manually. In order to minimize exposure time these filters will be mounted with quick disconnects so they can be easily bagged and discarded in accordance with procedures approved by the Radiological Engineering Department of ANL-W. Any one tank can be isolated from the system for repair and maintenance. If a problem develops with a tank during operation, the valve and piping arrangement will permit isolation of the problem tank. The system can then be operated off of the remaining tanks.

The liquid waste circulating system consists of four pumps and associated piping which circulates waste water continuously through the holding tanks and supplies make-up water to the evaporators. It is considered necessary to have a circulating loop attached to each holding tank for the purpose of continuously agitating the contents. Agitation will prevent settling of suspended particulates and thereby prevent buildup of high radiation levels at the bottom of the tanks. The same pumps which provide holding tank circulation will deliver waste water to the evaporators. The four holding tanks are connected in parallel to the six evaporators.

There is sufficient redundacy evident in this system to allow the process to continue even in the event that parts of the system are inoperable. For example, a circulating pump malfunction will not cripple the entire system since the process could continue to operate off the remaining pumps. Further redundacy is inherent in the parallel connections between holding tanks and evaporators. These connections make it possible to shut off flow to a single evaporator without interrupting flow to the remaining units. Closing the valves between the evaporator and the waste water feed line will effectively isolate a single evaporator from the rest of the circulating system as shown in Figure 2-2.

Safety advantages inherent in circulating system redundacy are further augmented by related process instruments and controls. It will be possible to operate essential valves and pump motors by remote manual controls located in the control room. The control room is separated from the process area by an air lock. Thus, it will be possible to isolate a holding tank or evaporator from a distant area of safety. This will be a distinct safety advantage should a leak or spill occur which would preclude the possibility of an operator entering the process area.

In addition to the four circulating pumps associated with the holding tanks, a second set of six circulating pumps are included to serve the evaporators. Their function in to aid in heat transfer by providing a continuous circulation of water over the heat transfer trays which are the major functional components of the evaporators.

A laboratory water line will be available to fill the evaporators with fresh water should, for any reason, liquid waste not be available. The idea is that the evaporators should have an ample supply of water at all times to prevent freeze-up of the circulating pumps which could, ostensibly, be caused by the high concentration of brines likely to accumulate in the evaporators. Should such a "freeze-up occur," no adverse affects are likely to be included except for a lowered evaporator efficiency due to diminished heat transfer.

The volume of waste entering the evaporators is automatically regulated. This is accomplished by a level controller which operates an automatic fill valve. Each evaporator is equipped independently with a separate fill valve and level controller. The advantage of this arrangement is to relieve the operators of the burden of having to control the fill level of each evaporator separately; this insures essentially automatic operation of the process. Only startup and shutdown will have to be done manually.

The possibility of a malfunctioning automatic fill valve presents the prospect of overfilling an evaporator and causing a spill of contaminated waste water. This prospect has been taken into account in the design and is compensated for by the liquid waste overflow system. This system consists of six overflow lines, a catch tank, and two transfer pumps. Overflow lines provide gravity drains to automatically shut excess water to the catch tank which has a nominal capacity of 100 gallons. Water is automatically removed from the catch tank by either of two redundant transfer pumps and returned to the holding tanks. Because it is a batch type process their will always be sufficient capacity in the holding tanks to accomodate the overflow. Liquid level monitors in the catch tanks will automatically notify control room operators of an evaporator overfill. Remedial action to isolate the affected evaporator can be enacted directly from the control room without risking exposure to personnel. The essential valves can be closed by remote control allowing the rest of the system to continue operating.

A process air system is provided to supply low humidity heated air to act as the evaporative medium. This air is drawn from outside the building. Design criteria call for 400 CFM of air heated to 250°F to be delivered to each of the six evaporators. The safety features of the process air system are associated with the following: evaporator redundancy, HEPA filtration, the exhaust fans, and the elevated stack.

Inlet air will pass through three banks of filters placed in series. The first is a roughing filter with an efficiency of about 35%. The second is a high capacity, high efficiency filter rated at 90%. The third is a bank of HEPA filters.

In a power outage, automatic dampers will close preventing exhaust air from escaping through the stack.

A resulting pressure build-up in the process air system might force air out through the air intake. In such an incident, the bank of HEPA filters at the intake will protect the environment.

Exhaust air passes through a pair of HEPA filters before being dispersed through the stack. These filters will remove radioactive particulates which may inadvertantly escape the evaporators. The only radionuclide to be routinely vented to the environment is tritium in the form of HTO. Calculations show that the tritium levels at the site boundary and Mud Lake will be well within the limitations of ERDA Appendix 0524 (see Chapter 5).

Redundancy in the process air system is a direct result of the multiple evaporator design approach. Each evaporator will have its own hot air intake and exhaust lines. Strategically placed butterfly valves will permit any single evaporator to be closed off for repair or maintenance while the rest of the system continues to operate.

This redundancy in the process air system will result in the following safety advantages:

- It will permit parts of the system to be shutdown for repair and maintenance while the rest of the system continues to operate.

- It will distribute the air stream through several smaller as opposed to a single large conduit thus reducing the relative magnitude of a contaminated vapor release.

Starting and stopping air flow through the process air system is accomplished simply by switching on or off the exhaust fans. This can be done from the control room so that it is difficult to conceive of an accident occurring in the process area that would prevent system shutdown. The control room is separated from the process area by an air lock and the control room air in pressurized relative to the process area--thus preventing an airborne release from entering the control room.

Process air is heated by steam coils. There are two sets of coils, the first of which will operate continuously and at full capacity. The second will be automatically regulated to keep the final temperature at 250°F. These coils are not expected to present a fire hazard since no substances having a flash point below 250°F are to be used at the facility.

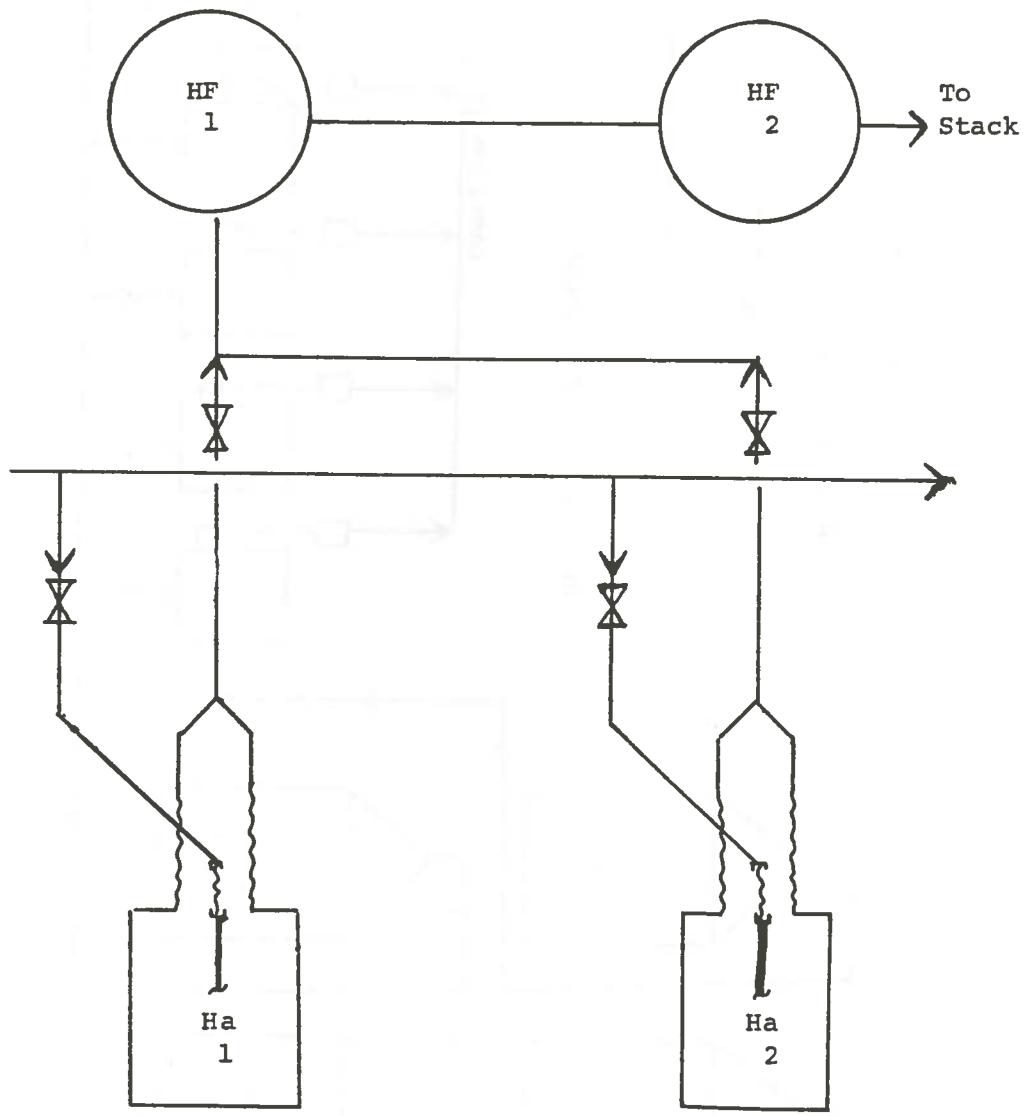
The exhaust from the process air system will afford additional redundancy. A pair of evaporator exhausts, connected in parallel, is routed through a pair of HEPA filters connected in series (see Figure 2-1). The effect of this design is to reduce the magnitude of a potential accident. A HEPA filter fire, for example, could be expected to produce a smaller release than would be the case if effluent from all six evaporators was routed through a single set of filters. Such a fire, however, is not expected since dry dust accumulation is not likely. Exhaust air relative humidity is expected to be about 40%.

The pressure differential necessary to draw air through the evaporators is provided by two exhaust fans--each sized to deliver the design air flow rate. Since failure of the lead fan could cause an air flow reversal, an interlock is provided as a safety device to automatically start the backup. Evaporator effluent is eventually released to the atmosphere via a forty foot high stack. The stack will be equipped with a radiation monitor which will interlock with the exhaust fans. Should the stack monitor detect an excessive radiation release, the exhaust fans will automatically shut off and air control dampers before and after each fan will automatically close.

The facility will also include a number of fire, smoke, and radiation detectors. Specifics on these are not available and radiation monitoring systems will comply with the ANL-W safety manual.

Figure 2-1

Process Air System



HA: Hot Air Evaporator
HF: HEPA Filter

Figure 2-2

Simplified Schematic of the RLWTF Process Liquid System

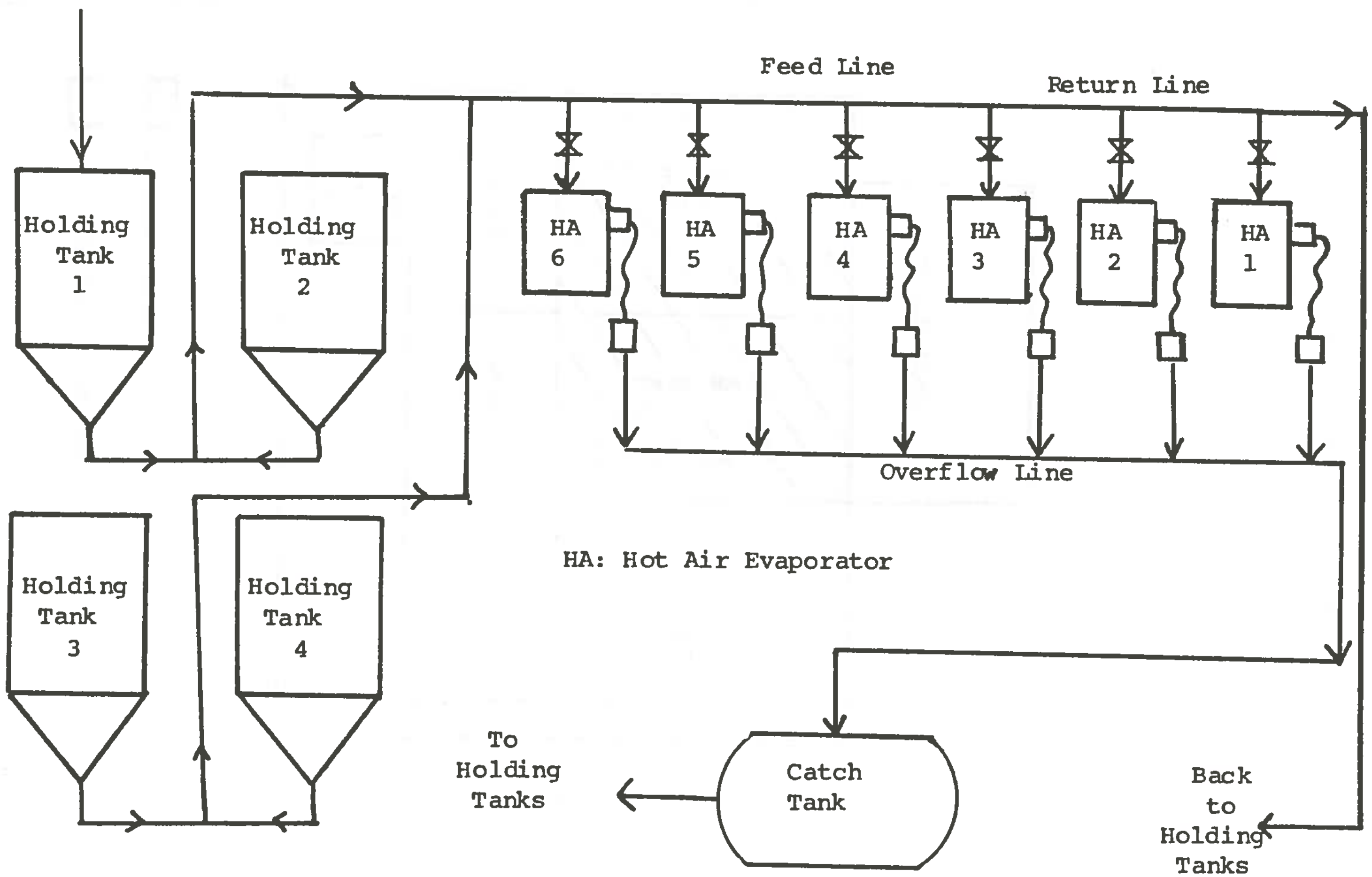


Figure 2-3

RLWTF Ground Floor

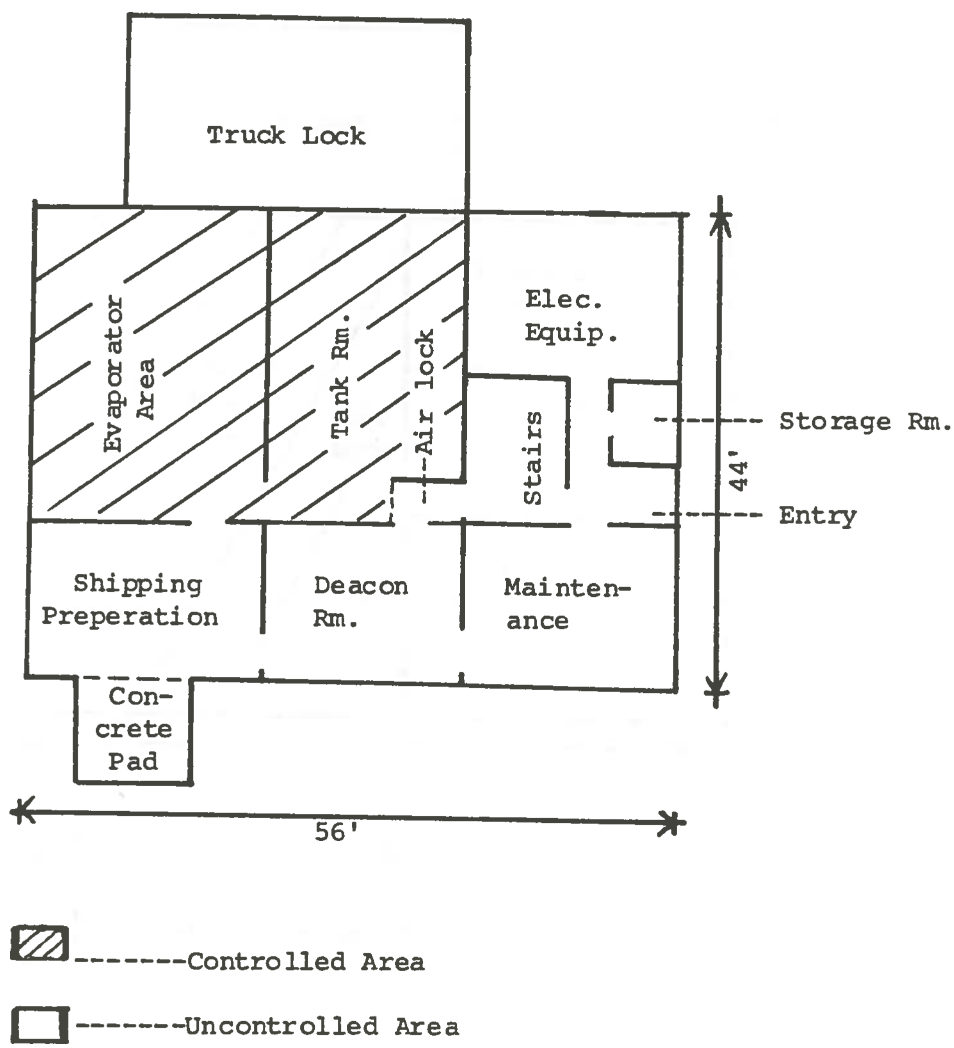
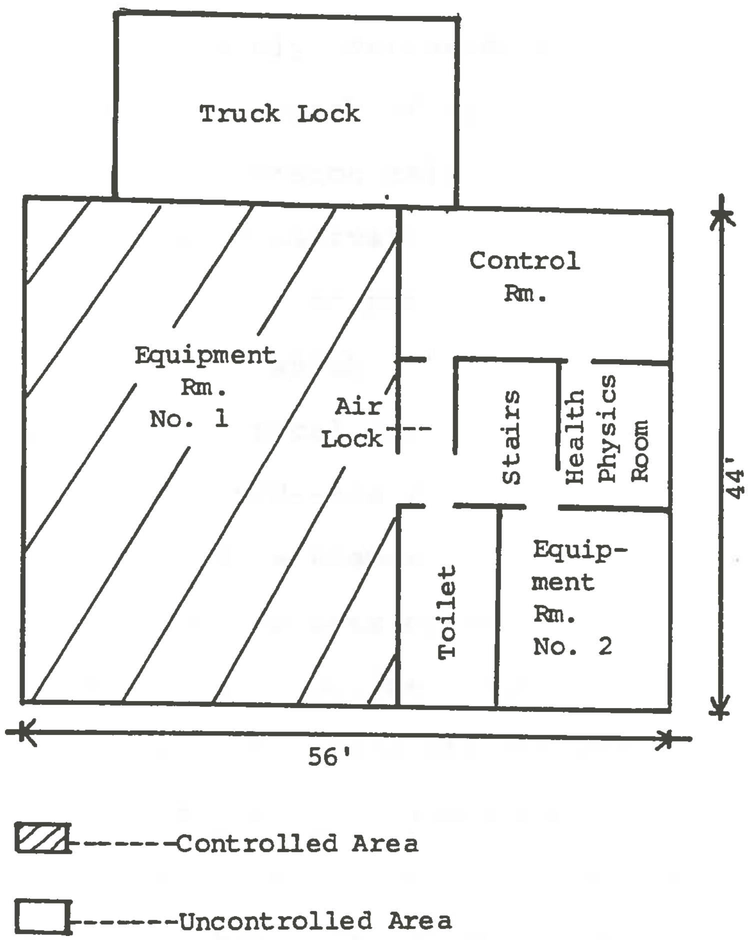


Figure 2-4

RLWTF Second Floor



CHAPTER 3

Radiological Source Terms

The purpose of this chapter is to provide quantitative information on the source terms likely to be encountered in the RLWTF. The author considers it prudent to moderately overestimate the concentration of radionuclides in a batch of waste feed so that the shielding and dispersion calculations in Chapters 4 and 5 will result in conservative (i.e., higher than actual) values. The idea is to produce shielding and atmospheric dispersion figures which will reflect limiting rather than average or typical values. Two sets of analytical data are represented--one from ANL-W in Table 3-1 and the other from LATA (Los Alamos Technical Associates) in Table 3-2.[6] The two sets of empirical data are combined to produce the radionuclide "cocktail" in Table 3-3 to be used in Chapter 5 to calculate concentrations of radionuclides in air due to an abnormal incident. Table 3-4 is based on the same information and provides a "cocktail" for the maximum hypothetical accident (MHA).

Table 3-1 is compiled from measured data provided by ANL-W. It represents waste water feed analysis for the Laboratory and Office Building (L & O) Evaporator (the predecessor of RLWTF). The figures shown represent all batches processed during 1978 and 1979.

These are considered to be the worst case years in that relatively high radionuclide concentrations were encountered during this period. This data, however, does not include precipitable material which would have fallen out of solution before analysis. Data on this material (sludge) is presented in Table 3-2.

Data in Table 3-2 is from 34,000 grams of sludge deposited in the South Concentrator of the L & O Evaporator. It represents precipitable gamma activity deposited from 160,000 gallons of liquid waste. The total of 34.5 μCi/g of sludge represents a gamma concentration of 0.00193 μCi/ml of water. Since the average non-precipitable gamma concentration is 0.0768 μCi/ml a total concentration of 0.08 μCi/ml is considered to be indicative of the waste water feed. For the purpose of shielding calculations this figure is assumed to represent only ^{137}Cs.

To arrive at a radionuclide inventory for the abnormal incident the average tritium concentration of 6.796×10^{-5} μCi/ml is used. Since the breakdown of alpha emitters is not known, the average alpha concentration of 3.956×10^{-6} μCi/ml is used. The breakdown of gamma emitters is assumed to be that given in Table 3-2. The relative amounts of these nuclides are used in Table 3-3 to calculate absolute concentrations. The percentages are multiplied by the total gamma concentration of 0.08 μCi/ml to arrive at specific concentrations for the individual radionuclides.

The MHA is assumed to consist of an accident whereby the total content of all 6 evaporators is dispersed, first throughout the facility housing, and then to the atmosphere at 85 m^3/sec. Volume of the facility housing is about 1400 m^3. The total number of curies of gamma emitters is assumed to be 2.51 per evaporator or 15.06 curies for all 6 evaporators. Relative amounts of gamma emitters are the same as in Table 3-2. The total number of curies for each isotope is divided by the volume of air in the facility to arrive at the "MHA Concentrations" in Table 3-4. Concentrations for ^{3}H and ^{239}Pu are assumed to increase proportionately with the gamma emitters relative to the concentrations listed in Table 3-3.

Table 3-1

Non-precipitable Radionuclides

L & O Evaporator Feed

(All batches, 1978 and 1979)

	Average	Maximum	Minimum
Tritium (μCi/ml)	6.796×10^{-5}	8.80×10^{-4}	4.50×10^{-6}
Alpha (μCi/ml)	3.956×10^{-6}	6.60×10^{-5}	1.30×10^{-7}
Beta (μCi/ml)	1.15×10^{-2}	4.30×10^{-1}	8.20×10^{-3}
Gamma (μCi/ml)	7.677×10^{-2}	8.30×10^{-1}	4.90×10^{-3}
Solids (μg/100 ml)	0.037	0.078	0.020

Table 3-2

Results of Multichannel Analysis of Precipitable Radionuclides

Isotope	Measured ($\mu Ci/g$)	% Abundance
^{54}Mn	0.056	0.162
^{60}Co	0.051	0.147
^{134}Cs	1.31	3.80
^{137}Cs	31.9	92.46
^{134}Ce	1.02	2.96
^{155}Eu	0.19	0.55
Total	34.5	100.00

(Date are from Reference 6).

Table 3-3

Abnormal Incident, Isotopes and Concentrations (μCi/ml)

Isotope	Percent	Multiply by	Concentration
^{3}H	100	6.796×10^{-5}	6.796×10^{-5}
^{54}Mn	0.162	0.08	1.296×10^{-4}
^{60}Co	0.147	0.08	1.176×10^{-3}
^{134}Cs	3.80	0.08	3.04×10^{-3}
^{137}Cs	92.46	0.08	7.397×10^{-2}
^{144}Ce	2.96	0.08	2.368×10^{-2}
^{155}Eu	0.55	0.08	4.40×10^{-4}
^{239}Pu	100	3.956×10^{-6}	3.956×10^{-6}

Table 3-4

MHA Concentrations

Isotope	Total Curies	MHA Concentrations (Ci/m^3)
^{3}H	1.28×10^{-2}	9.14×10^{-6}
^{54}Mn	2.44×10^{-2}	1.74×10^{-5}
^{60}Co	2.21×10^{-2}	1.58×10^{-5}
^{134}Cs	5.72×10^{-1}	4.09×10^{-4}
^{137}Cs	1.39×10^{1}	9.95×10^{-3}
^{144}Ce	4.46×10^{-1}	3.18×10^{-4}
^{155}Eu	8.28×10^{-2}	5.92×10^{-5}
^{239}Pu	7.45×10^{-4}	5.32×10^{-7}

CHAPTER 4

Evaporator Shielding

In accordance with the design criteria for the RLWTF, 60,000 gallons of liquid waste per year is to be processed by the 6 evaporators.[1] The liquid portion of the waste will be vaporized to steam and vented to the atmosphere via a 40 foot high stack. The remaining particulate matter stays in the evaporators in the form of a sludge. When the radiation level outside of an evaporator reaches a predetermined level the entire evaporator will be removed and replaced. Thus radiation levels will increase as the sludge level increases and the level of the sludge will depend on how much waste is processed and how much sludge is in the waste. In addition the radiation levels depend on the concentration of radioactive particles in the sludge itself (i.e., the number of μCi per cm^3 of sludge).

The purpose of this chapter is to evaluate the adequacy of shielding in the RLWTF. Since the major radiation sources will be concentrated in the evaporators, the shielding of these vessels is most important from the standpoint of radiation safety. Design standards call for a limiting dose rate of 50 millirem per hour on the outer surface of the shield.

Each evaporator is shielded by 6 inches of high density concrete (see Figure 4-1). Results from the QAD computer code indicate that a dose rate of 50 mR/hr will be achieved when the sludge level reaches 12.3 inches (Figure 4-2). This computation assumes that all radiation is from cesium-137 and that the concentration of this radionuclide is 50 μCi per cubic centimeter of sludge. Based on a cesium concentration of 0.08 μCi per ml of liquid waste (as discussed in Chapter 3), the sludge level of 12.3 inches corresponds to a liquid waste volume of 8,295 gallons (see Table 4-2 and Figure 4-3) and to a total ^{137}Cs content of 2.51 curies (see Table 4-3 and Figure 4-4). In other words, 8,295 gallons of liquid waste could be processed before the radiation level reaches the limiting dose rate of 50 mR/hr. It is noted that this is short of the 10,000 gallons per year for each evaporator required by the design criteria. This means that when the incoming waste has a concentration of gamma emitters averaging 0.08 μCi/ml the evaporators would have to be replaced before they had seen a full year of service. The average of 0.08 μCi/ml is however, a worst case estimate, and it is likely that the average gamma emitter concentration will be less than this. Assuming that this turns out to be the case, Table 4-4 and Figure 4-5 predict the dose rates which would result from lower gamma emitter concentrations. As indicated in Figure 4-5 a concentration of 0.066 µCi/ml will allow the desired amount of 10,000 gallons to be processed before the dose rate reaches 50 mR/hr. Further reductions in the feed water concentration increase evaporator capacity quite dramatically. For example, at a concentration of 0.02 µCi/ml 33,154 gallons could be processed and at 0.01 µCi/ml 66,309 gallons could be evaporated before dose rates outside the shield reach 50 mR/hr. Thus the conclusion is that as long as the concentration of gamma emitters is less than 0.066 µCi/ml it should be possible to process 10,000 gallons in each of the six evaporators thereby fulfilling the design requirements. Higher concentrations will exceed the 50 mR/hr limit and necessitate more frequent replacement.

It is important to point out that the conclusions stated in the preceding paragraphs do not take into account scatter and streaming through shield penetrations such as the hot air intake and exit and the liquid waste overflow lines. Another important assumption which requires some scrutiny is the value of 50 µCi/cm^3 used as the estimate of the concentration of gamma emitters in the sludge. As of this writing a measured value for this parameter is not available, however an analysis performed by LATA on a similar type of sludge showed 34.5 µCi/cm^3. Sludge from the RLWTF should be somewhat higher than this although exactly how much higher is not known. The author believes the estimate of 50 $\mu Ci/cm^3$ to be a reasonable approximation to what will be found in the RLWTF.

Two sets of calculations have been done in order to produce the data presented herein. An initial hand calculation was made in order to produce a first approximation. More refined values were attained by using the digital computer code QAD.[8] Some aspects of the models used with the 2 calculations are the same. Thus in each case a succession of sludge levels has been postulated within the 30 gallon evaporator drum. Each level of sludge fills a specified volume beginning at the bottom of the drum as depicted in Figures 4-6 and 4-7. As Table 4-5 indicates, the drum is considered to be systematically filled with sludge and dose rates calculated every one-fifth of the way up. For example, when sludge fills the bottom one-fifth of the drum the top of the sludge layer will be at a height of 5.6 inches and the dose rate is calculated at a receptor point outside the shield and 2.8 inches from the base of the drum. Similarly when the sludge occupies two-fifths of the available volume of 30 gallons, assuming a uniform distribution, the top of the sludge would be at a height of 11.2 inches above the base of the drum. The receptor point would be 5.6 inches above the base and outside the shield. The receptor point is always placed at one-half the total height of the sludge since this is the point at which the maximum dose rate will be recorded.

Major differences between the two models include the nature and distribution of the source term. With the hand calculation the source is assumed to be uniformly distributed throughout the specified volume. In contrast, the QAD code breaks down the source into a number of point isotropic sources. These sources are distributed as shown in Figure 4-6. A cylindrical void space is left in the middle to correspond to the 4 (ID) inch air intake line.

Other similarities between the computer code and the hand calculation include some of the parameters used in determining the final dose rate. Thus, in both cases a buildup factor of 12.48 was used. Calculation of the buildup factor is described more completely in Appendix A. The major assumption made was that buildup from the sludge would be equivalent to the buildup from one inch of high density concrete. Therefore the number of mean free paths was based on seven rather than six inches of concrete. This assumption was made only for the purpose of calculating a buildup factor and was not utilized either in the hand calculation or in the computer program for calculation of uncollided fluxes. The buildup factor was calculated from the Berger form with a C value of 1.2344 and a D value of 0.0730.[17] Shielding was assumed to consist of heavy aggregate concrete with a density of 4.5 grams/cm^3 and a linear attenuation factor (μ) equal to 0.3375 cm^{-1}.

Other values common to both computations are the average energy (E), the dose rate to energy flux conversion factor, and the source density. The latter was set at 50 μCi/cm^3 as discussed previously. For the sake of simplifying the calculations and because there is not an abundance of data available on the relative amounts of the various gamma emitters in the waste, all were assumed to be ^{137}Cs with an average gamma energy of 0.662 MeV per disintegration. The dose rate to energy flux conversion factor was set at one millirem per hour for every 570 MeV/cm^2-s for both computations.

Despite these similarities between the two sets of calculations differences appear in the results as denoted by Figure 4-8 and Table 4-6. Whereas the computer calculation provides a straight line through the origin, the hand calculation results in a rather peculiar curve which overestimates the computer dose rates at low sludge levels and underestimates them at high levels. Thus at a sludge height of 5.6 inches the QAD code gives a value of 22.69 mR/hr while the hand calculation shows the higher value of 44.82 mR/hr. At 28 inches the computer calculation provides a value of 113.06 mR/hr while the hand calculation gives 80.71 mR/hr. The shape of the hand calculation curve in Figure 4-8 is due to the type of model used and the equation which was employed.[15] This equation is shown below,

$$\phi_E = \frac{BQ_r E\phi_o}{2\pi\mu_s} L_o(\phi_o, b)[e^{-b}\sigma\overline{G}(\sigma,b) - e^{-b'}\sigma\overline{G}(\sigma,b')] \qquad (4\text{-}1)$$

where

ϕ_E = energy flux

B = buildup factor

Q_V = source strength of a volume source

E = energy of the primary gamma

μ_s = absorption coefficient of the source

b = number of mean free paths

and b', σ, L_o (ϕ_o,b), $\overline{G}$ (σ,b), $\overline{G}$ (σ,b') all depend on the geometry of the cylinder as indicated in Appendix B.

In following the methodology outlined earlier for the use of this equation it was mentioned that different sludge levels in the evaporator are postulated and dose rates are then calculated corresponding to these levels. However it is implicitly assumed when using the above equation that the source is uniformly distributed over the specified volume. Thus it is not possible to specify a volume of 30 gallons and then distribute the source overy say, the bottom one-fifth. This problem has been circumvented by assuming a succession of cylinder heights corresponding to the various sludge levels. A cylinder height of 5.6 inches, for example, corresponds to a volume fraction of 1/5 and to a sludge level of 5.6 inches. When the 30 gallon drum is completely full the cylinder height is, of course, the total cylinder height of 28 inches and the volume fraction is 5/5. A similar rationale is used to calculate the dose rates at 2/5, 3/5, and 4/5 levels.

It turns out that by changing the cylinder height to correspond to the various sludge levels in equation (4-1) the only parameter which is compelled to change is σ, $\overline{G}$ (σ, b) and $\overline{G}$ (σ, b'). All other quantities remain constant. What this means is that increases in the dose rate come about only because of changes in geometry--in other words, only because of changes in cylinder height. In going from 1/5 to 5/5 of the total cylinder height of 28 inches the relative change in σ becomes smaller and smaller. For example, in going from 1/5 to 2/5 of the total height, σ increases by 87%, from 0.2729 to 0.5105 (see Table 4-7). On the other hand, in going from 2/5 to 3/5 increases from 0.5105 to 0.6987 for an increase of only 36%. Subsequent increases are by even smaller fractions causing relatively low values for ϕ_E and therefore relatively low dose rates at the higher sludge levels. More information on this equation is provided in Appendix B.

In contrast to the hand calculation, the QAD code does not require the source to be uniformly distributed throughout the cylinder. Using a set of weighting factors it is possible to distribute source points in any convenient manner throughout the cylinder. For the purpose of approximating the distribution likely to result from the RLWTF evaporator, the source points are distributed in a manner similar to the example shown in Figure 4-7. A gap is left in the middle to correspond to the air inlet pipe which has a diameter of 4 inches. Because the QAD code allows for a more accurate model of the actual problem the results from it are likely to be closer to the true values. Therefore the computer code values are the basis for the conclusions drawn in the first paragraphs of this chapter. The values from the hand calculations were used only as a first approximation and as a check on the order of magnitude of the dose equivalent rates.

The appropriate QAD code geometry corresponding to the RLWTF evaporators is represented in Figure 4-9. The coordinate axis origin was placed at the top, 20.32 centimeters or 8 inches away from the interior 30 gallon drum. It was assumed that the intervening area was completely shielded--in other words, the penetrations by the hot air intake and exit lines were not accounted for. This however, should not alter results along the lateral surfaces of the shield.

In order to completely describe the model the QAD code requires that for each of the 6 regions specified in Figure 4-9 there be assigned a composition number correlating to a particular composition. As Table 4-7 indicates there are a total of 3 types of compositions consisting of the source, the shield, and a void space between the source and the shield. The void space is considered to be a vacuum. This is reasonable since in practice it will contain a low atomic number thermal insulation which will have a minimal contribution to shielding. The source is represented as 500 isotropic point sources distributed through Region I, the 30 gallon drum. As Table 4-8 indicates the drum is considered to be filled with water with the source points distributed within. The source points are distributed to reflect sludge heights ranging from 1/5 to 5/5 of the total drum height of 71.12 centimeters or 28 inches.

The shield is composed of ferro-phosphorus concrete with a density of 4.5 grams/cm^3. Atom densities for the major constituent elements are listed in Table 4-8. This type of concrete differs from ordinary siliceous concrete by its relatively high density and disproportionately high concentrations of phosphorus and iron.

A complete listing for input to the QAD code is provided in Appendix C. For gamma ray calculations, QAD uses the point-kernel ray-tracing technique. In this method, the point kernel representing the transfer of energy by the uncollided flux along a line-of-sight path is combined with an appropriate buildup factor to account for the contribution of scattered photons. Although a buildup factor is provided by the code the variety of materials is limited. Since the code will not provide a buildup factor for concrete, a hand calculated value of 12.48 was substituted with a distributed source, the point kernel is integrated over the source volume for each source energy considered. Expressed as an equation, the gamma ray dose rate at any point due to an isotropic source emitting S photons of energy E per second per unit volume is,

$$D(\vec{r}) = K \int_{r} \frac{S(\vec{r})B(\mu|\vec{r} - \vec{r}'|,E)\exp(-\mu|\vec{r} - \vec{r}'|)dr}{4\pi|\vec{r} - \vec{r}'|^2} \qquad (4\text{-}2)$$

where

r = point at which the gamma dose rate is to be calculated,

r' = location of source in volume V,

V = volume of source region,

μ = total attenuation coefficient at energy E,

$|\vec{r}-\vec{r}'|$ = distance between source point and the point at which the gamma intensity is to be calculated,

$B(\mu|\vec{r}-\vec{r}|),E$ = dose buildup factor, and

K = conversion factor (flux-to-dose rate).

The results of five QAD runs comprise the base data for the conclusions drawn in this chapter. As indicated by Table 4-1 and Figure 4-2 the dose rate can be related to the sludge height in the evaporators by a simple linear relationship. This is expressed as,

$$DR = H \times 4.052 \tag{4-3}$$

where

DR = dose rate in millirem/hr,

H = sludge height in inches (assuming a source density in the sludge of 50 $\mu Ci/cm^3$ of sludge).

Therefore a dose rate of 50 mR/hr corresponds to a sludge height of,

$$H = 50/4.052 = 12.3 \text{ inches}$$

Inspection of Table 4-3 and Figure 4-4 leads to a similar relationship between the dose rate outside the shield and the total curie content within,

$$DR = C \times 19.905 \tag{4-4}$$

where

DR = dose rate in millirem/hr

C = the total number of curies (assuming a concentration of 50 $\mu Ci/cm^3$ of sludge).

Utilizing equation (4-4) provides the conclusion that for a dose rate of 50 mR/hr the total number of curies would be,

$$C = 50/19.905 = 2.51 \text{ curies.}$$

A relationship derived from Table 4-2 and Figure 4-3 indicates the relationship between the dose rate and the number of gallons of processed feed water to be,

$$DR = G \times 6.028 \times 10^{-3} \qquad (4\text{-}5)$$

where

DR = dose rate in millirem per hour

G = the number of gallons of waste water processed (assuming a concentration of 0.08 μCi/ml of ^{137}Cs).

Using equation (4-5) the number of gallons of liquid waste averaging 0.08 μCi/ml which would produce a dose rate of 50 mR/hr is given by,

$$G = 50/6.028 \times 10^{-3} = 8{,}295 \text{ gallons.}$$

The graph in Figure 4-5 relates the volume of processed feed water required to produce a dose rate of 50 mR/hr outside an evaporator at various ^{137}Cs

concentrations. This graph can be reduced to the following equation,

$$G = 663.092/CON \tag{4-6}$$

where

G = the number of gallons of feed water required to produce a dose rate of 50 mR/hr.

CON = the average concentration in units of μCi/ml of ^{137}Cs in the incoming waste.

Since design criteria calls for 10,000 gallons of waste per evaporator per year to be processed before the dose rate reaches 50 mR/hr, using equation (4-6) gives,

$$CON = 663.092/10{,}000 = 0.066\ \mu Ci/ml.$$

This means that in order to comply with the 10,000 gallon per year criterion the average concentration of gamma emitters in the feed should not exceed 0.066 μCi/ml. Should this average be exceeded, the effect will be to reduce the operating lifetime of the evaporators and cause more frequent replacement.

To summarize these results: The dose rate outside an evaporator shield will reach 50 mR/hr when there is 2.51 curies of ^{137}Cs in an evaporator. Assuming a concentration of ^{137}Cs in the sludge of 50 $\mu Ci/cm^3$ the quantity of 2.51 curies corresponds to a sludge height of 12.3 inches. If a worst case estimate of 0.08 μCi/ml is made for the average concentration of gamma emitters in the feed, the total volume of processed waste corresponding to 2.51 curies and thus to 50 mR/hr is 8,295 gallons. This is below the design criteria of 10,000 gallons per evaporator per year. In order to meet this criterion the average concentration of gamma emitters in the incoming waste should not exceed 0.066 μCi/ml.

Figure 4-1

Dimensions of RLWTF Drum Evaporator

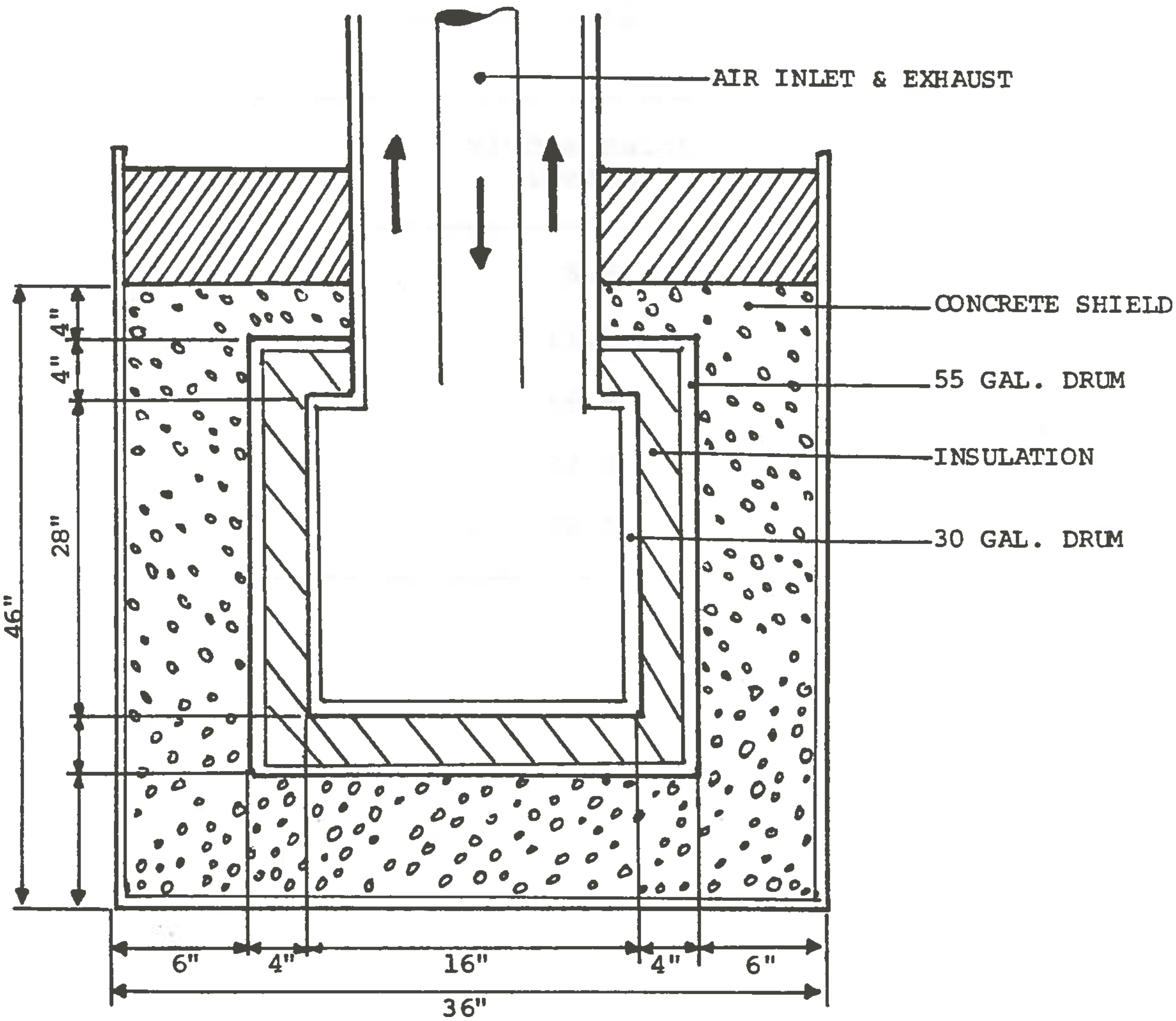


Table 4-1

Correlation of Sludge Height to Dose Rate

Sludge Fraction	Sludge Height (inches)	QAD Dose Rate (mR/hr)
1/5	5.6	22.69
2/5	11.2	45.19
3/5	16.8	67.88
4/5	22.4	90.37
5/5	28.0	113.06

Figure 4-2

Dose Rate as a Function of Sludge Height

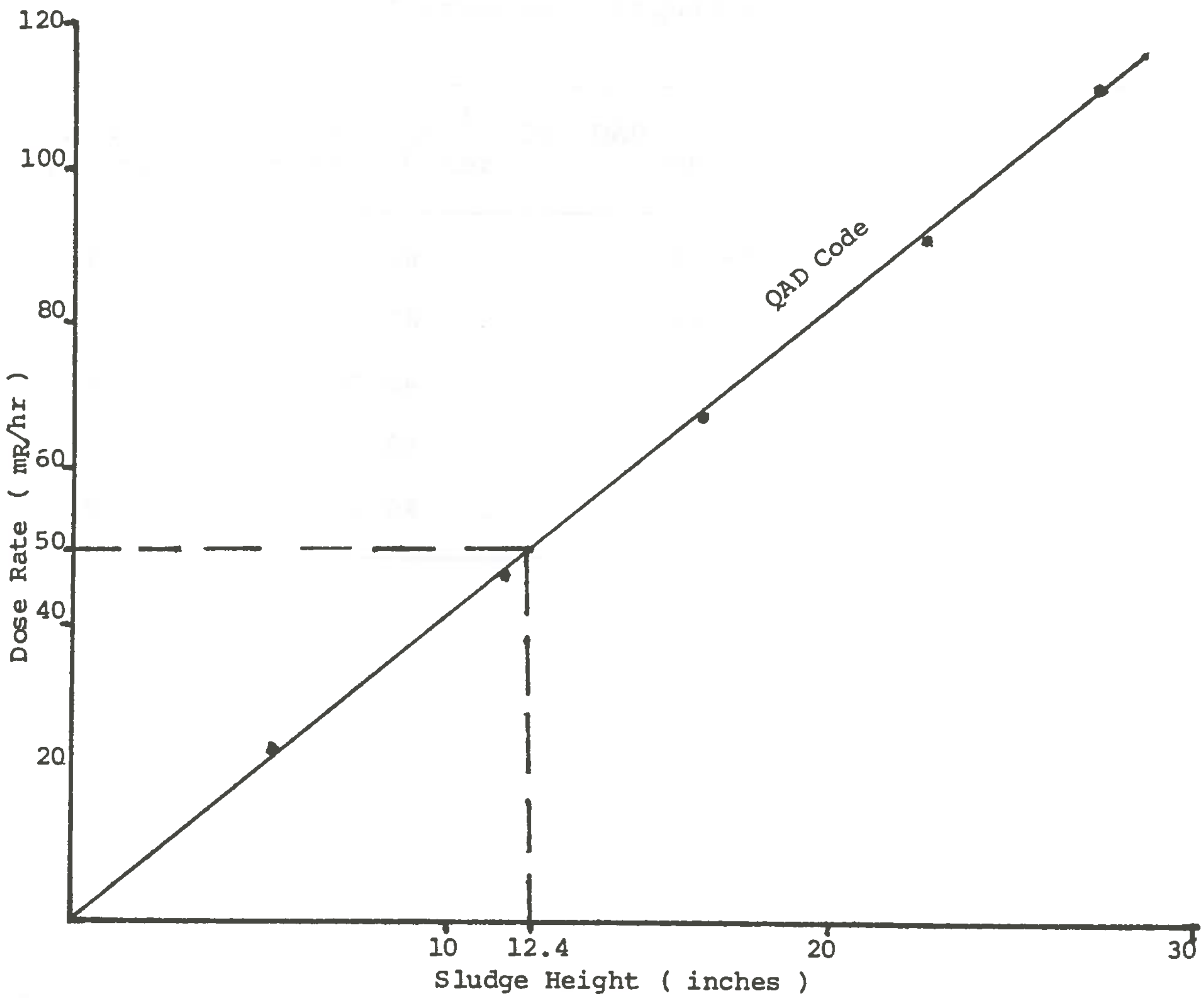


Table 4-2

Correlation of Dose Rates to the Volume of Processed Liquid Waste

Sludge Fraction	(μCi/ml) of ^{137}Cs in Feed Water	QAD Dose Rate (mR/hr)	Gallons of Liquid Waste
1/5	0.08	22.69	3,765
2/5	0.08	45.19	7,496
3/5	0.08	67.88	11,261
4/5	0.08	90.37	14,992
5/5	0.08	113.06	18,757

Figure 4-3

Dose Rate as a Function of the Volume of Liquid Waste Processed Assuming 0.08 μCi/ml ^{137}Cs in the Waste

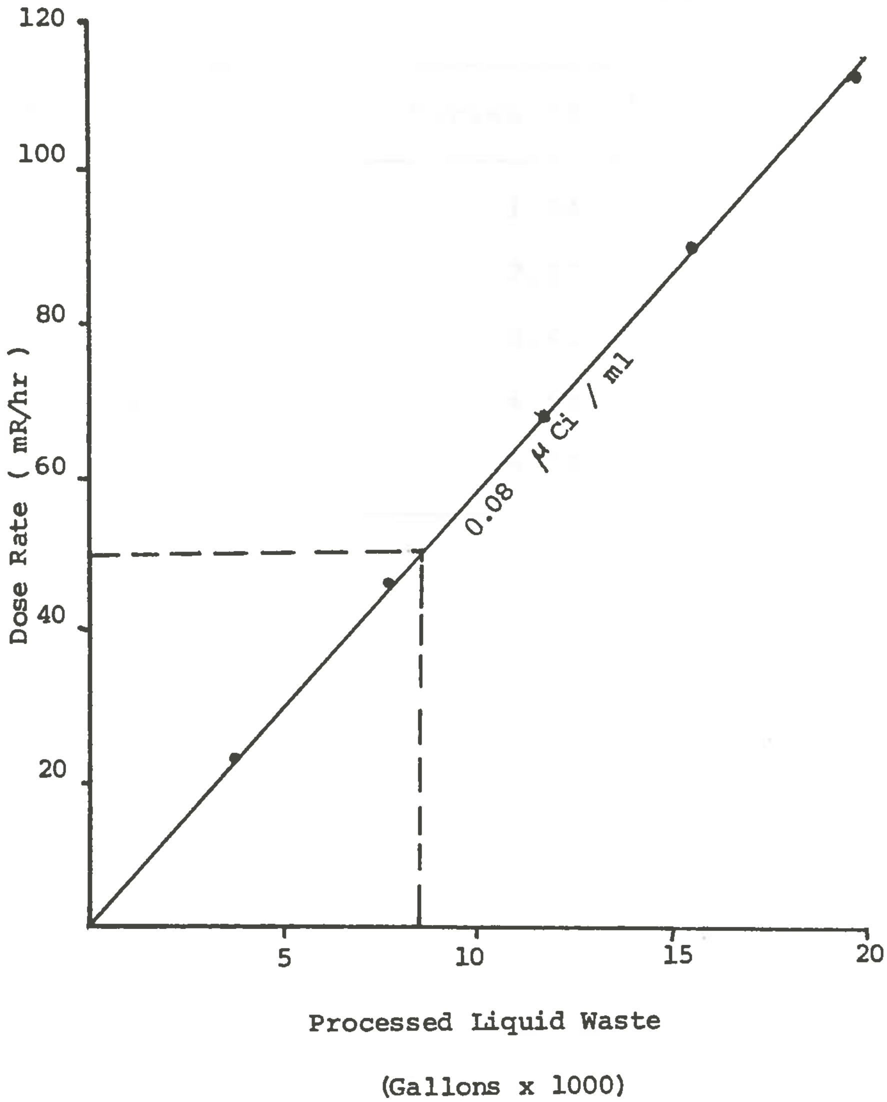


Table 4-3

Correlation of Dose Rate to Amount of ^{137}Cs

Sludge Level	Curies of ^{137}Cs	Dose Rate (QAD)
1/5	1.14	22.69
2/5	2.27	45.19
3/5	3.41	67.88
4/5	4.54	90.37
5/5	5.68	113.06

Figure 4-4

Relationship of Dose Rate to the Number of Curies of ^{137}Cs in an Evaporator

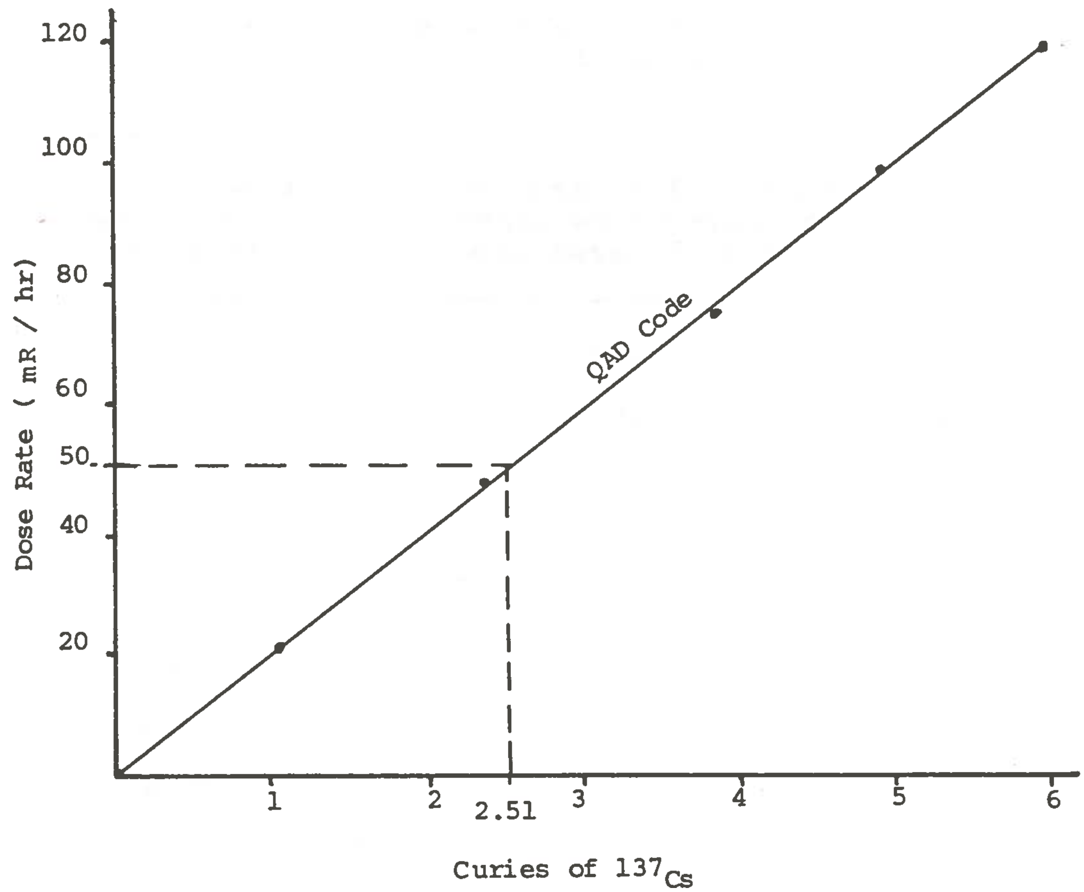


Table 4-4

Volume of Waste Necessary to Produce a Dose Rate of 50 mR/hr

Feed Water Cesium Concentration (μCi/ml)	Gallons of Feed Water Required to Produce a Dose Rate of 50 mR/hr	Total Curies
0.01	66,309	2.51
0.02	33,154	2.51
0.03	22,103	2.51
0.04	16,577	2.51
0.05	13,262	2.51
0.06	11,051	2.51
0.07	9,472	2.51
0.08	8,289	2.51

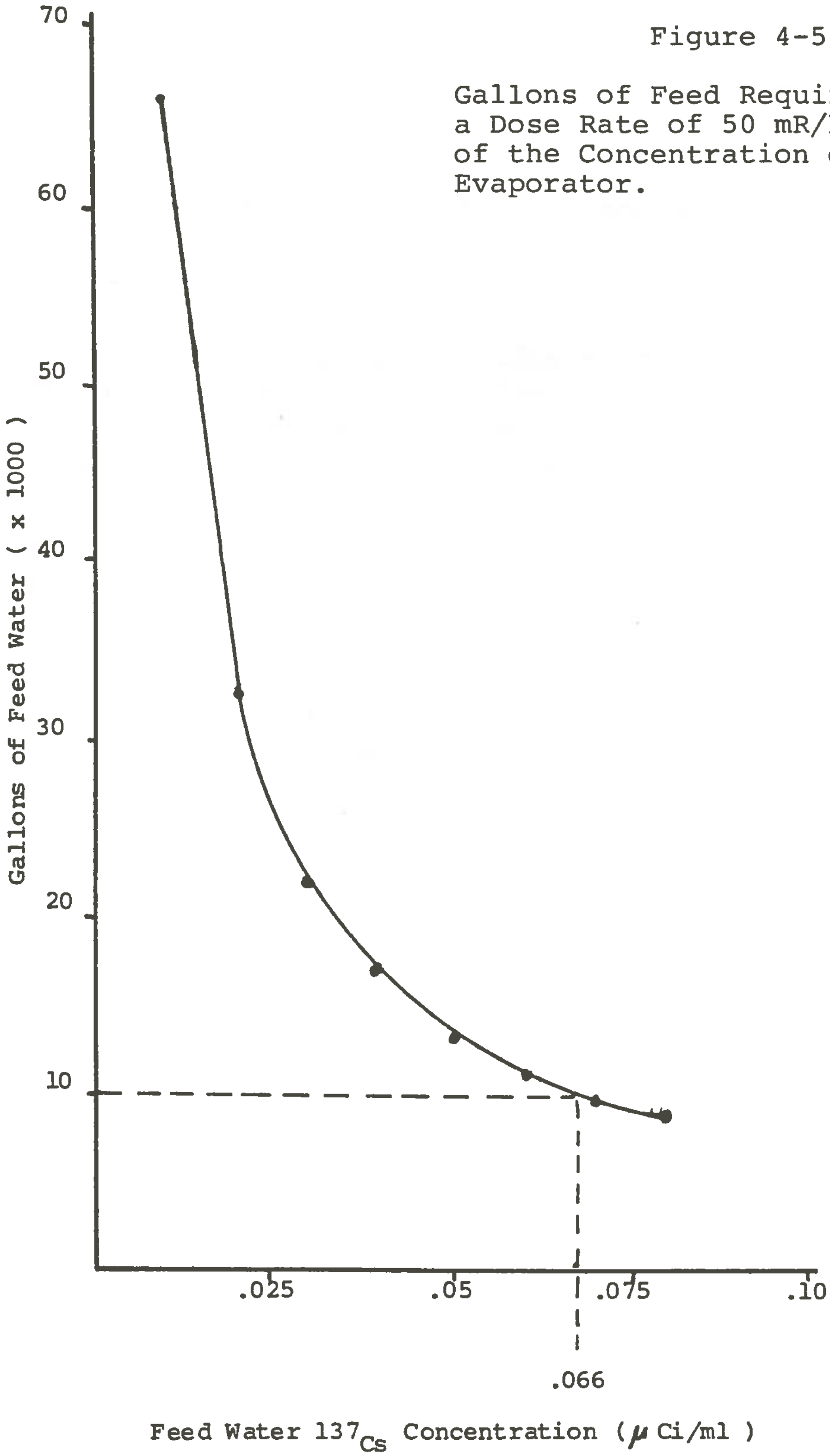


Figure 4-5

Gallons of Feed Required to Give a Dose Rate of 50 mR/hr as a Function of the Concentration of ^{137}Cs in an Evaporator.

Table 4-5

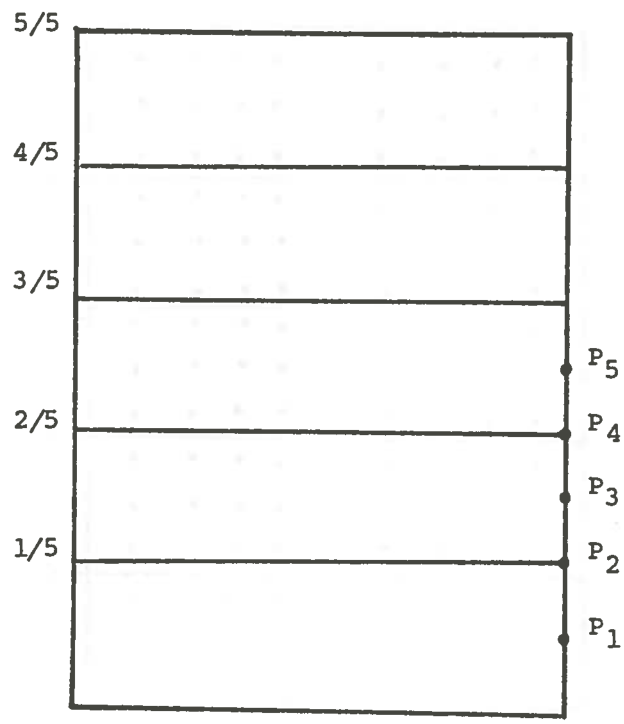
Hand Calculation Parameters

Sludge Fraction	Sludge Volume (gallons)	Sludge Height (inches)	Receptor Point	Receptor Height (inches)
1/5	6	5.6	P_1	2.8
2/5	12	11.2	P_2	5.6
3/5	18	16.8	P_3	8.4
4/5	24	22.4	P_4	12.0
5/5	30	28.0	P_5	14.0

Figure 4-6

Model for Hand Calculation Sludge Levels in Drum Evaporator

The source is considered to be uniformly distributed.



Sludge Levels in Drum Evaporator

Figure 4-7

Model for QAD Computer Code Sludge
Levels in Drum Evaporator

The source is broken down into point sources and distributed to leave a gap 4 inches in diameter in the center of the drum.

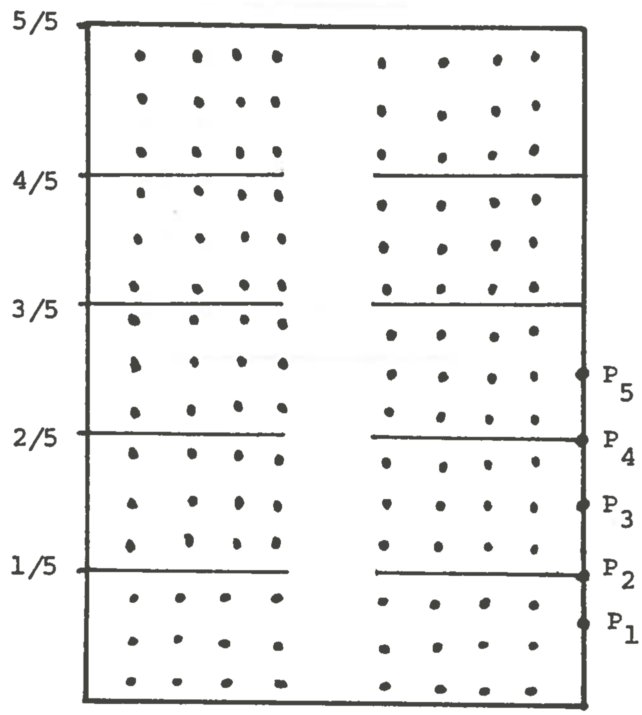


Table 4-6

Data Points for Hand Calculation
and QAD Code

Sludge Fraction	Sludge Height (inches)	Dose Rate (Hand Calculation)	Dose Rate (QAD Code)
1/5	5.6	44.82	22.69
2/5	11.2	68.21	45.19
3/5	16.8	77.54	67.88
4/5	22.4	79.00	90.37
5/5	28.0	80.71	113.06

Figure 4-8

Dose Rate vs. Sludge Height
Comparison of Hand Calculation
to QAD Computer Code Calculation

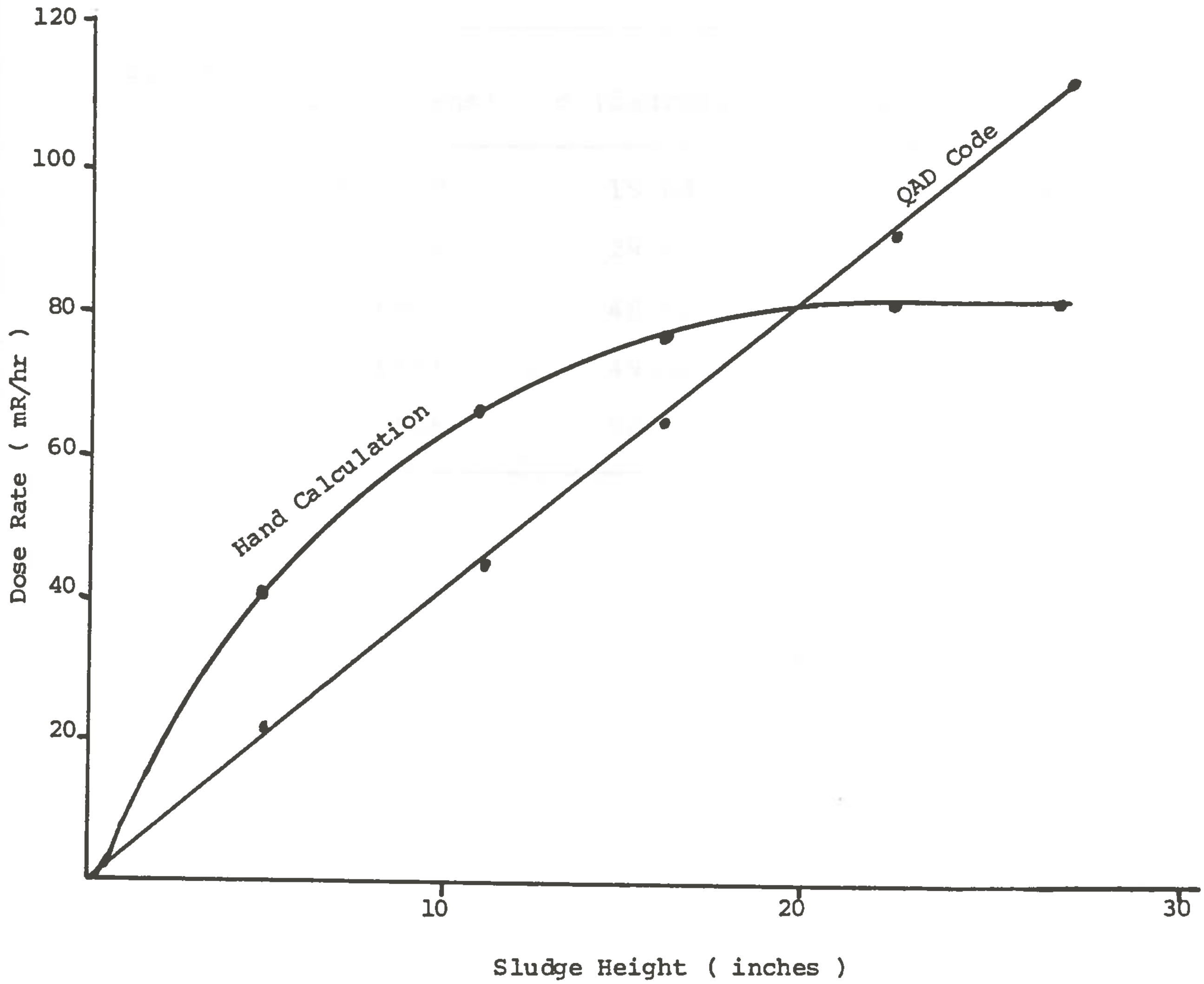


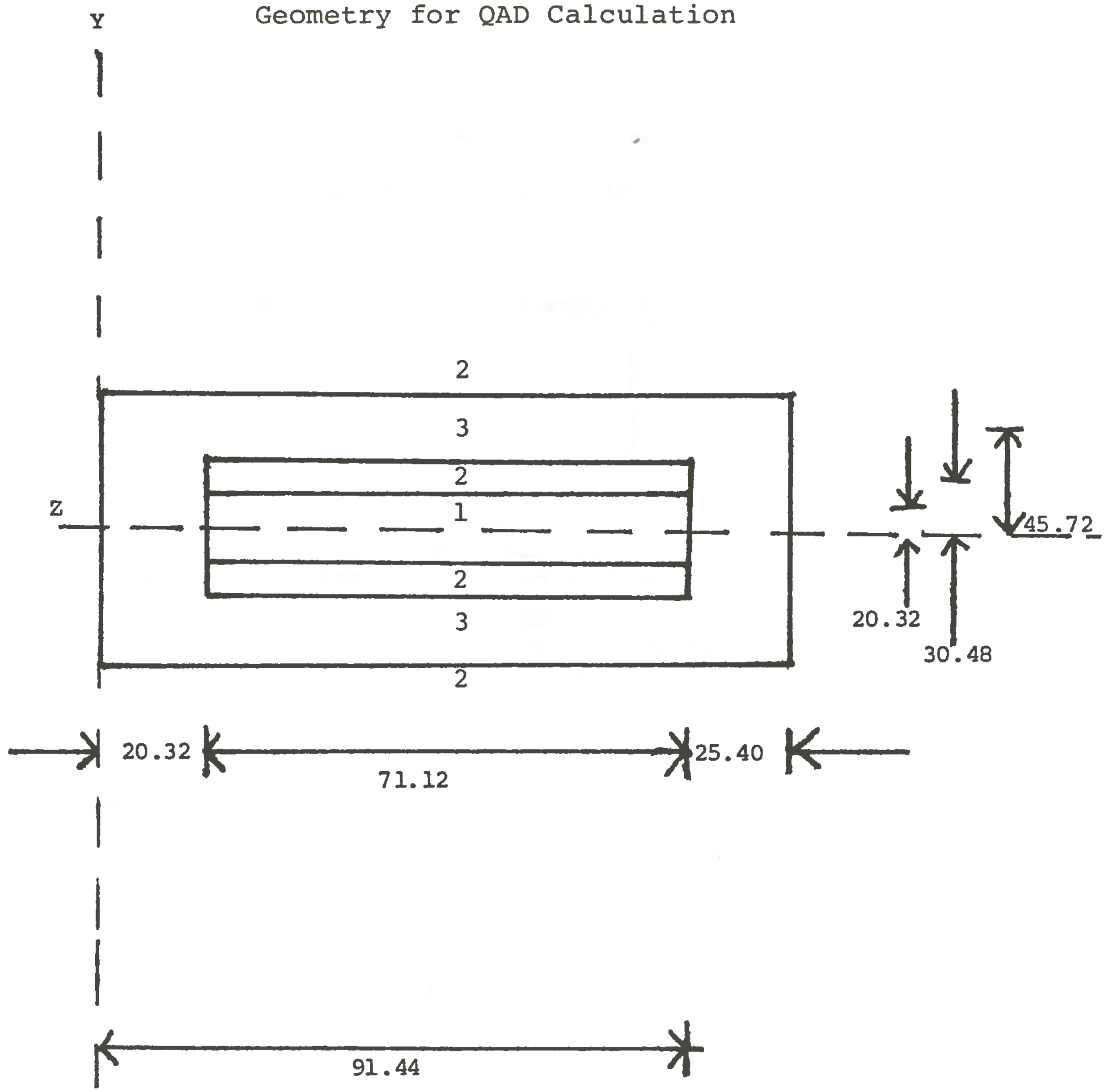
Table 4-7

Other Parameters for the Hand Calculation

Sludge Fraction	σ (radians)	σ (degrees)	$\bar{G}$ (σ, b)	$\bar{G}$ (σ, b')
1/5	0.2729	15.64	0.95	0.88
2/5	0.5105	29.25	0.77	0.62
3/5	0.6987	40.03	0.64	0.53
4/5	0.8591	49.22	0.53	0.43
5/5	0.9505	54.46	0.49	0.41

Figure 4-9

Geometry for QAD Calculation



All dimensions in centimeters.

Table 4-8

Model Description for the QAD Code

Region Number	Composition Number
1	1
2	2
3	3
4	2
5	2
6	2

Composition Number	Material
1	source - water
2	void - vacuum
3	shield - concrete

Table 4-9

Atomic Parameters for QAD Model

Composition	Atom density of materials					
	H	O	Al	Si	P	Fe
Source	0.111	0.889	--	--	--	--
Void	--	--	--	--	--	--
Shield	0.024	0.490	0.019	0.160	0.927	2.881

Material	Symbol	Atomic Number
Hydrogen	H	1
Oxygen	O	8
Aluminum	Al	13
Silicon	Si	14
Phosphorus	P	15
Iron	Fe	26

CHAPTER 5

Atmospheric Dispersion

The RLWTF is designed to vent gaseous radioactive effluent to the environment via a 40 foot high stack. The purpose of this section is to evaluate potential radiation exposure to individuals which may result from this venting. Calculations have been made to predict exposure rates and effluent concentrations from normal dispersion and from accidental releases. Predicted values are compared to commonly used standards.

Under normal conditions only tritium, in the form of HTO, will be emitted from the stack. Other isotopes either remain in the evaporators in the form of a sludge or they are filtered out by a pair of HEPA filters. Tritium is a pure beta emitter, it is not concentrated by biological species, and it passes moderately quickly through the human body. Its MPC (maximum permissible concentration) values are therefore among the highest of any radionuclide.

ANL-W has adopted maximum permissible concentration standards from ERDA Manual Chapter 0524 as criteria for establishing release limits of various radionuclides to the air.[10] This document refers to the MPC values as concentration guides (CG). The CG value for tritium in an uncontrolled area is 2×10^{-7} Ci per m^3 of air. Table 5-1 gives the calculated concentrations (CHI) which are expected to result from normal stack release from the RLWTF. Three CHI values are given at three different distances from the stack--100, 5000, and 35,000 meters. Note how the values of CHI become smaller with increasing distance because the plume is increasingly diluted by the atmosphere. As Table 5-1 indicates, the calculated values of CHI range from 6.65×10^{-2}% (at 100 meters) to 8.47×10^{-6}% (at 35,000 meters) of the concentration guides. Thus, it is concluded that radiation exposure during normal operations will be well within the prescribed limits.

To determine the potential hazard resulting from the failure of the stack monitor, an abnormal incident has been postulated. In this incident, all of the isotopes assumed to be present in a typical batch of waste feed (see Chapter 3) are vented to the atmosphere via the stack. Table 5-2 presents the expected concentration of these isotopes in air at the indicated distances from the stack. Concentration guides for these isotopes are presented in Table 5-3.

In cases where a mixture of radionuclides is present Amnex A to EARDA Appendix 0524 suggests that the following rule be applied:

"If radionuclides A, B, and C are present in concentrations C_A, C_B, and C_C, and if the applicable CG's are CG_A, CG_B, and CG_C respectively, then the concentrations should be limited so that the following relationship exists:

$$\frac{C_A}{CG_A} + \frac{C_B}{CG_B} + \frac{C_C}{CG_C} \leq 1."$$

When this rule from 9524 is applied to the isotopes listed in Table 5-4. As this table shows, isotope concentrations vary from 23.4% of the Guides at 100 meters to 2.91×10^{-3}% at 35,000 meters. At the site boundary (5000 meters), the isotope concentration is only 0.900% of the appropriate Guides.

Strictly speaking the concentration Guides are not applicable to the abnormal incident described herein. But what Table 5-4 serves to show is that with the low level waste the RLWTF is ordinarily expected to handle, failure of the stack monitor and filtration system will not necessarily lead to excessive amounts of radionuclides being vented to the atomosphere and will, therefore, be well below limits for an accidental release.

To determine the overall radiation hazard from the RLWTF, a maximum hypothetical accident (MHA) has been postulated. In this accident, the entire inventory of radionuclides is assumed to be dispersed, first throughout the facility, and then to the atmosphere via a ground-level duct at the rate of 50 cubic feet per minute (CFM). Initial concentrations, for the radionuclides considered, are given in Table 3-4.

The total whole body doses (from both inhalation and submersion in a semi-infinite cloud) are summarized in Table 5-5. These doses range from 47.2% of the 10CFR100 limit at 100 meters from the facility to 0.10% at 35,000 meters.[11] At the site boundary the total dose is 201 mrem or 0.80% of the limit. Table 5-6 shows the contribution that individual nuclides make to the total of 201 mrem at the site boundary. The major contributors are ^{239}Pu, ^{137}Cs, and ^{134}Cs. In the actual facility, these nuclides will be bound to non-volatile compounds so that their dispersion through the atmosphere is highly unlikely. Nevertheless, when this fact is ignored the resultant doses are still well within the 10CFR100 limit.

The submersion and inhalation doses in Table 5-5 were calculated independently. Inhalation doses were calculated from the following equation,

$$D(\text{mrem}) = X \cdot t \cdot BR \cdot DCF \qquad (5\text{-}1)$$

where,

X = the concentration of a particular isotope in curies per cubic meter,

t = the time period during which nuclides are inhaled,

BR = the breathing rate in cubic meters per second, and

DCF = the dose commitment factor in mrem per Curie inhaled.

The report NUREG-0172 gives values for DCF;[12] the units of DCF are mrem in 50 years per picocurie inhaled in one year or less. Values of DCF for isotopes of interest are listed in Table 5-7.

For the receptor at the site boundary Regulatory Guides 1.3 and 3.3 state that the dose commitment is calculated for an individual who remains at this location for two hours after the initial release of activity from the facility.[13,14] At a wind speed of 1 meter per second it will take 5000 seconds or 1.39 hours for the plume to reach the site boundary. Thus the receptor time t, at 5000 meters, is 0.61 hours. A similar argument applied to the receptor at 100 meters yields an exposure time of 1.97 hours. The receptor at the low population zone, Mud Lake, must remain in place for the duration of the accident, 9.72 hours. In accordance with Regulatory Guides 1.3 and 3.33, BR is taken to be 3.47×10^{-4} m^3 per second for the receptors at 100 and 5000 meters. For the receptor at 35,000 meters, BR is taken to be 1.75×10^{-4} m^3 per second. These figures are summarized in Table 5-8.

Submersion doses for beta or gamma radiation for a person standing on the ground, from an effectively semi-infinite cloud of radionuclides surrounding him, may be obtained from,

$$D = C \cdot \bar{E} \cdot X \cdot t \qquad (5\text{-}2)$$

where,

D = the dose in rem,

C = a constant; $C = 0.30$ for beta and gamma radiation,

$\bar{E}$ = the mean effective energy per disintegration in MeV,

X = the concentration in Curies per cubic meter of air (CHI), and

t = the exposure time from Table 5-8.

CHI in equations 1 and 2 was calculated from,

$$X = Q \cdot X/Q \qquad (5\text{-}3)$$

where,

Q = the release rate in Curies per second, and

X/Q = the relative dispersion in seconds per cubic meter.

For the release of tritium from the stack, Q was based on 8.80×10^{-4} µCi/ml of tritium in the waste feed. This is the maximum tritium concentration from Table 3-1. This tritium is diluted by air before it leaves the stack. Based on a relative humidity of 42.5% and a temperature of 118°F, stack air will contain 73.4 g/m^3 of water. These factors combined with the exhaust velocity of 4800 CFM goves a release rate of 6.46×10^{-7} Ci/sec. For the abnormal incident, these same parameters apply, except that concentrations for specific isotopes in water are taken from Table 3-3.

For the MGA, concentrations of individual nuclides in air are multiplied by an exhaust velocity of 50 CFM to give values of Q.

The relative dispersion, X/Q, for a ground level release is calculated from,

$$X/Q = \frac{1}{\pi \mu \sigma_y \sigma_z} \tag{5-4}$$

where,

μ = the wind speed in meters per second,

σ_y = the horizontal standard deviation of the plume in meters, and

σ_z = the vertical standard of the plume in meters.

The values of σ_y and σ_z are determined by the stability of the atmosphere and the distance from the source to the receptor of interest. Moderately stable (Pasquill Type F) conditions were assumed to persist for the time intervals of interest. A value of 150 meters was used for σ_y and 35 meters for σ_z.[14] The wind speed is assumed to be one meter per second and to continue unchanged in direction and magnitude. When these parameters are used equation 5-4 gives a X/Q of 6×10^{-5} seconds per cubic meter at the site boundary.

For the receptor at Mud Lake the plume was assumed to meander and spread uniformly over a 22.5 degree sector. The relevant equation is,

$$X/Q = \frac{2.032}{\mu \; X \; \sigma_z} \tag{5-5}$$

where,

μ = 1 meter per second,

X = the distance to the receptor, 35,000 meters, and

σ_z = is taken to be 69 meters.[14]

When these values are used, X/Q for Mud Lake is 8.5×10^{-7} sec/m^3.

For an elevated release from a height of h meters, the ground level X/Q values are multiplied by:

$$e^{-\frac{h^2}{2\sigma_z^{2}}}. \tag{5-6}$$

While ground level releases are assumed for the MHA, the normal dispersion and abnormal incident assume the activity issues from a 12.2 meter high stack. Under Type F conditions, the values for Equation 6 are 0.94 for the receptor at the site boundary and 0.98 for Mud Lake.

The receptor at 100 meters is a special case since this receptor encounters downwash due to obstruction of the wind flow by the adjacent facility HFEF/N. This is shown in Figure 5-1. Downwash is defined as the aerodynamic flow on the lee side of a stack or building caused by relatively high wind speeds which bring about a downward movement of effluent along the lee surface of the obstruction to the general air flow. To estimate X/Q due to this downwash as empirical equation derived from wind tunnel tests was used. That equation is,

$$X/Q = \frac{K}{\mu \cdot D^2} \qquad (5\text{-}7)$$

where,

K = is a coefficient, generally taken to be "2" for conservative results,

μ = the wind speed, considered to be one meter per second, and

D^2 = the projected area of the building obstructing the air flow (HFEF/N), taken to be 930 m^2.

These values give a X/Q value at 100 meters of 2.06×10^{-3} seconds per cubic meter. Facilities Systems Engineering

(FSEC) has done a study on the phenomena of downwash as it applies to the RLWTF. They estimate that this phenomena will occur between 5% and 10% of the time during the winter and as much as 30% of the time during the summer.

Within the downwash area (in the immediate area of the facility) will be a volume of stagnant air which will hamper diffusion somewhat. FSEC suggests that this will not prove to be a problem since bouyancy of the heated plume will eventually free the exhaust from the cavity and shearing action will eventually integrate the plume into the prevailing air flow. It is noted however, that rules of thumb typically dictate the stack height be 2.5 times the height of the nearest adjacent structure.[7] The stack height for the RLWTF is only 40 feet while the adjacent building HFEF/N is 100 feet high. The author concludes that dispersion might be more efficient if the RLWTF were not built immediately adjacent to HFEF/N.

To summarize, four basic conclusions have been drawn in this chapter. First, routine, continuous release of tritium from the RLWTF will be well within the prescribed limits. At the site boundary, tritium concentrations will be on the order of 2.63×10^{-3}% of the Concentration Guides. Second, failure of the stack monitor and filtration system, will not necessarily lead to a release in excess of the Concentration Guides for routine releases, and will certainly be within the limits for an accidental release. Third, release of the entire radionuclide inventory of the RLWTF in a MHA will be well within the 10CFR100 standard of 25 rem. At the site boundary, the dose would be on the order of 0.8% and at Mud Lake, 0.1% of the 10CFR100 limit. And fourth, it is concluded that the efficiency of dispersion in the immediate vicinity of RLWTF could probably be improved if the facility were not built immediately adjacent to HFEF/N.

Table 5-1

Comparison of CHI Values for Tritium to Concentration Guide Limits at Indicated Distances from the RLWTF.

Distance	CHI (Ci/m^3)	CG (Ci/m^3)	% CHI/CG
100 m	1.33×10^{-10}	2×10^{-7}	6.65×10^{-2}
5000 m	5.26×10^{-12}	2×10^{-7}	2.63×10^{-3}
35,000 m	1.69×10^{-14}	2×10^{-7}	8.47×10^{-6}

Table 5-2

Abnormal Incident Values of CHI (Ci/m^3) for Specified Isotopes at Indicated Distances from the RLWTF.

Isotope	Concentration of 100 m	5000 m	35,000 m
^{3}H	1.40×10^{-13}	5.52×10^{-15}	1.78×10^{-17}
^{54}Mn	2.67×10^{-13}	1.06×10^{-14}	3.40×10^{-17}
^{60}Co	2.42×10^{-13}	9.56×10^{-15}	3.08×10^{-17}
^{134}Cs	6.26×10^{-12}	2.48×10^{-13}	7.98×10^{-17}
^{137}Cs	1.52×10^{-10}	6.02×10^{-12}	1.94×10^{-14}
^{144}Ce	4.67×10^{-12}	1.93×10^{-13}	6.20×10^{-11}
^{155}Eu	4.06×10^{-13}	3.59×10^{-14}	1.15×10^{-16}
^{239}Pu	8.15×10^{-15}	3.22×10^{-16}	1.04×10^{-18}

Table 5-3

Concentration Guides (CG) for Specified Isotopes

Isotope	CG (Ci/m^3)
^{3}H	2×10^{-7}
^{54}Mn	1×10^{-8}
^{60}Co	1×10^{-8}
^{134}Cs	1×10^{-9}
^{137}Cs	2×10^{-9}
^{144}Ce	3×10^{-10}
^{155}Eu	3×10^{-4}
^{239}Pu	6×10^{-14}

(From ERDA Manual Chapter 0524)

Table 5-4

Abnormal Incident Ratio of CHI Values to Concentration Guides at Indicated Distances From RLWTF.

Isotope	100 m	5000 m	35,000 m
^{3}H	7×10^{-7}	2.76×10^{-8}	8.91×10^{-11}
^{54}Mn	2.67×10^{-5}	1.06×10^{-6}	3.40×10^{-9}
^{60}Co	2.42×10^{-5}	9.56×10^{-7}	3.08×10^{-9}
^{134}Cs	6.26×10^{-3}	2.48×10^{-6}	7.98×10^{-10}
^{137}Cs	7.60×10^{-2}	3.02×10^{-3}	9.71×10^{-6}
^{144}Ce	1.56×10^{-2}	6.27×10^{-4}	2.07×10^{-6}
^{155}Eu	1.35×10^{-4}	1.20×10^{-5}	3.83×10^{-8}
^{239}Pu	1.36×10	5.37×10^{-3}	1.73×10^{-5}
Total	0.234	9.00×10^{-3}	2.91×10^{-5}
% of Limits:	23.4	0.900	2.91×10^{-3}

Table 5-5

Whole Body Doses from Inhalation and Submersion MHA Conditions

Distance	Dose (mrem)	% of Limit*
100 m	11,800	47.2
5000 m	201	0.80
35,000 m	25.6	0.10

* 25 rem as in 10CFR100

Table 5-6

MHA, Total Dose from Inhalation and Submersion for Specified Isotopes at the Site Boundary (5000 m)

Isotope	Dose (mrem)
^{3}H	0.029
^{54}Mn	0.159
^{60}Co	0.155
^{134}Cs	10.705
^{137}Cs	26.870
^{144}Ce	0.879
^{155}Eu	0.206
^{239}Pu	162.000
Total	201.003
% of Limit: (25 rem)	0.80

Table 5-7

Dose Commitment Factors in Units of mrem in 50 Years Per Curie Inhaled in One Year or Less

Isotope	DCF
^{3}H	1.58×10^{5}
^{54}Mn	7.87×10^{5}
^{60}Co	1.85×10^{6}
^{134}Cs	5.99×10^{6}
^{137}Cs	4.05×10^{4}
^{144}Ce	5.80×10^{4}
^{155}Eu	2.40×10^{5}
^{239}Pu	7.73×10^{10}

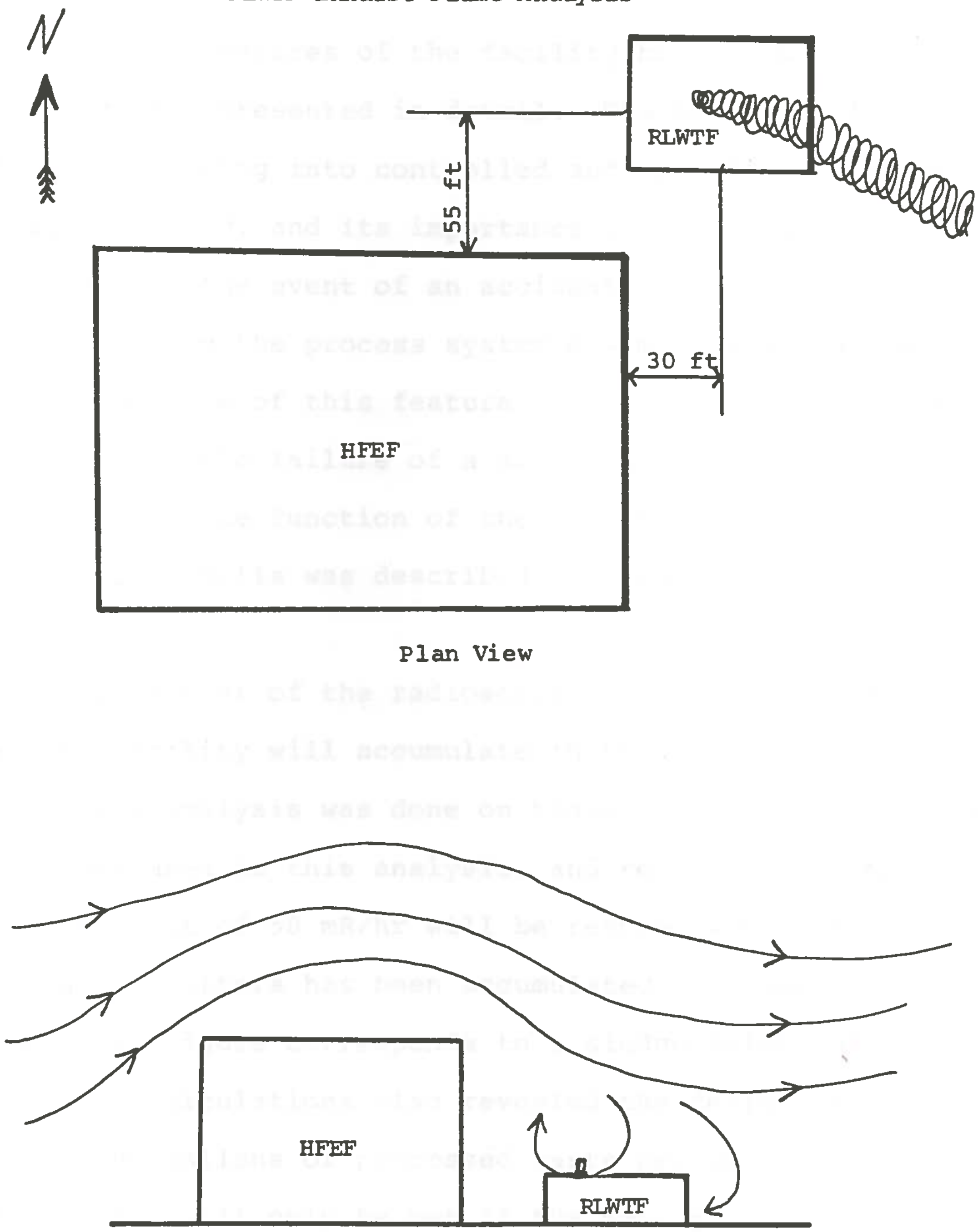
Table 5-8

Values for the Breathing Rate (BR) and Exposure Time (t) for Receptors at the Indicated Distances from RLWTF.

Distance	BR (m^3/s)	t (hr)
100 m	3.47×10^{-4}	1.97
5000 m	3.47×10^{-4}	0.61
35,000 m	1.75×10^{-4}	9.72

Figure 5-1

RLWTF Exhaust Plume Analysis



Cavity Due To South-
Westerly Winds

CHAPTER 6

Conclusions and Recommendations

Safety features of the facility housing and process systems were presented in detail. The division of the facility housing into controlled and uncontrolled areas was emphasized, and its importance in insuring safe shutdown in the event of an accident was noted. Redundancy in the process system design was discussed. The importance of this feature in preventing total system shutdown on the failure of a single component was explained. The function of the overflow system in preventing spills was described as a major back-up safety system.

Since most of the radioactive materials processed in the facility will accumulate in the evaporators, a shielding analysis was done on these vessels. A computer code was used in this analysis, and results indicate the design limit of 50 mR/hr will be reached when 2.51 curies of gamma emitters has been accumulated. It was noted that this figure corresponds to a sludge height of 12.3 inches. Calculations also revealed the design criteria of 10,000 gallons of processed waste per year for each evaporator will only be met if the average concentration of gamma emitters in the feed does not exceed 0.066 μCi/ml.

A concise site and facility description was presented in the manner usually adhered to in safety reports of this kind. Also, an explanation of the derivation of source terms used in shielding and atmospheric dispersion calculations was given.

Because the facility is designed to continuously vent radioactive effluents to the environment, calculations were made to predict the magnitude of such releases and comparisons were made to the applicable safety standards. It was concluded that routine releases of tritium, in the form of HTO, will be well within the prescribed limits.

In addition, it was concluded that a maximum hypothetical accident would result in a dose on the order of 0.8% of the 10CFR100 limit of 25 rem for a receptor at the site boundary. Finally, it was noted that dispersion of radioactive effluents from the RLWTF is somewhat impaired by the nearby building, HFEF/N. It was suggested that dispersion would probably be improved if the RLWTF was not situated immediately adjacent to the larger structure.

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10. "Standards for Radiation Protection", ERDA Manual, Chapter 0524, Washington, DC (1975).

11. "Reactor Site Criteria, Title 10, Code of Federal Regulations, Part 100, AEC, Washington, DC (1961).

12. G. R. Hoenes and J. K. Soldat, "Age Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake", Battelle Pacific Northwest Laboratories Report NUREG-0172, Richland, WA (November 1977).

13. U. S. Nuclear Regulatory Guide 3.33, "Assumptions Used for Evaluating the Potential Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant", Washington, DC (April 1977).

14. U. S. Nuclear Regulatory Commission Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors", Revision 2, Washington, DC (June 1974).

15. J. C. Courtney, "A Handbook of Radiation Shielding Data", ANS/SD-76/14 (July 1976), p. 2-16.

16. "QAD Computer Code", Ref. 8, p. 116.

17. N. M. Schaeffer, ed., "Reactor Shielding for Nuclear Engineers", U. S. Atomic Energy Commission Report TID-25951, Oak Ridge, TN (1973), p. 628.

APPENDIX A

Buildup Factor

A buildup factor (B) of 12.48 was used both in the hand calculation and in the QAD computer code. This buildup factor was calculated from the Berger equation,

$$B = 1 + C \cdot \mu \cdot t \cdot \exp(D \cdot \mu \cdot t),$$

where

$C = 1.2344$,

$D = 0.0730$, and

$\mu = 0.3375\ cm^{-1}$ for high density concrete.

This equation and the values for C and D were interpolated from Schaeffer.[17] It is known that in addition to buildup in the shield, the sludge and other evaporator internals will tend to increase, somewhat, the magnitude of the buildup factor--exactly how much is difficult to estimate. In order to insure conservative results, the buildup factor was calculated based on a shield thickness of 7 inches instead of 6 inches of high density concrete. Thus, contributions from evaporator internals increase the buildup factor from 10.23 to 12.48.

APPENDIX B

Hand Calculation

The equation used for the hand calculation (equation 1, Chapter 4) is from A Handbook of Radiation Shielding Data.[14] Values for parameters not given in Chapter 4, are presented below. These parameters were used in equation (1) for calculation of the energy flux outside of an evaporator shield. Values for b and b' are based on attenuation coefficients of 0.3375 cm^{-1} and 0.07 cm^{-1} respectively. Shield thickness was taken to be six inches. Other parameters pertinent to this equation are listed in Table 4-7 and below as:

$B = 12.48$, buildup factor;

$Q_V = 1.85 \times 10^6$ (γ/cm^3-sec), source strength;

$E = 0.662$ MeV, average γ energy;

$\mu_s = 0.07\ cm^{-1}$, attenuation coefficient for source;

$b = 5.14$;

$b' = 7.48$;

$\phi_o = 0.461$; and

$L_o(\phi_o, b) = 0.81$.

APPENDIX C

The QAD Computer Code

The QAD-P5A(360) computer code was used to evaluate the evaporator shielding. References for this code, the code itself, and a sample problem are available from the Radiation Shielding Information Center at Oak Ridge.[16] These items may be obtained by writing:

Codes Coordinator
Radiation Shielding Information Center
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, TN 37830

The input to QAD-P5A(360) consists of the following:

Card A: FORMAT (18A4)

72 Columns of Hollerith Information

Card B: FORMAT 8 (I5,I4)

Column	Variable	
1-5	LSO	Number of increments along the R(cyl.), X(cart.), or ρ(spher.) axis which specify the division of source volume. ($1 \leq LSO \leq 100$)
6-9	MSO	Number of increments along the Z(cyl.), Z(cart.), or θ(spher.) axis which specify the divisions of source volume. ($1 \leq MSO \leq 100$)
10-14	NSO	Number of increments along the δ(cyl.), Y(cart.), or δ(spher.) axis which specify the divisions of source volume. ($1 \leq NSO \leq 100$)

Column	Variable	
15-18	MAT	Total number of elements used in problem. $1 \leq MAT \leq 20$. (If $MAT \leq 0$, Cards M through W not needed; cards of previous case are used; see list of options below).
19-23	NCOMP	Number of compositions. $1 \leq NCOMP \leq 40$. (If $NCOMP \leq 0$, cards 0 through W not needed; Cards of previous case are used; see list of options below).
24-27	NREG	Number of regions. $1 \leq NREG \leq 50$. (If $NREG \leq 0$, use regions of previous case; see list of options below).
28-32	NRGY	Number of gamma-ray energy groups. $1 \leq NRGY < 30$.
33-36	NBOUND	Total number of boundaries. $1 \leq NBOUND \leq 50$. (If $NBOUND \leq 0$, use boundaries of previous case; see list of options below).
37-41	NSOPT	Geometry of source: Set NSOPT = 0 for Cylinderical NSOPT = 1 for Cartesian NSOPT = 2 for Spherical (See list of options below for description of additional input options that may be used here).
42-45	NZSO	The most probable source region, i.e., the region most likely to contain the source points.
46-50	ISRC	Type of source: ISRC = 0; source of the previous case is used. ISRC = 1; cosine-distributed source is used. ISRC = 2; source is computed using the weighting volume input along each coordinate axis.

Column	Variable	
51-54	INEUT	If INEUT > 0, neutron calculation will be made; if INEUT $\leq$ 0, no neutron calculation will be made.
55-59	NGPF	First source point for which ray geometry printout is desired.
60-63	NGPL	Last source point for which ray geometry printout is desired.
64-68	NGPI	The incremental step size used to select additional source points for ray geometry printout.
69-72	NGINT	Number of additional gamma calculations to be made.

NOTE: The ray geometry printout will give the distance traveled through each region from the specified source point to the detector. The first source point for which this is done is NGPF. The next source point is selected by adding NGPI to NGPF. This is continued until the last source point, NGPL, is reached. (The ray geometry from the coordinate origin is always printed out.) This is useful for finding errors in the geometry input.

Card C: FORMAT (8E 9.4)

Column	Variable	
1-9	ASO	The total source strength is fissions/sec, captures/sec, or decays/sec.
10-18 12-27	XISO(1,1) XISO(2,1)	Constants for cosine source distribution (if ISRC = 1) as a function of the R(cyl.), X(cart.), or δ(spher.) coordinate.
28-36 37-45	XISO(1,2) XISO(2,2)	Constants for cosine source distribution (if ISRC = 1) as a function of the δ(cyl.), Y(cart.), or δ(spher.) coordinate.
46-54 55-63	XISO(1,3) XISO(2,3)	Constants for cosine source distribution (if ISRC = 1) as a function of the Z(cyl.), Z(cart.), or θ(spher.).

NOTE: Using this cosine-distributed source OPTION (ISRC = 1), the code calculates the source strength at a point (A, B, C) as

Source Strength (A,B,C) = ASO cos [XISO(1,1) (A-XISO(2,1))] x cos [XISO(1,2)(B-XISO(2,2))] cos [XISO(3,1)(C-XISO(3,2))]

where A, B, and C are the coordinates of the point in the appropriate coordinates system. If all values of XISO are input as zero, a flat or uniformly distributed source will be obtained when using this source option.

Card(s) D: FORMAT (8E9.4)

Column	Variable	
	RSO	Coordinates of source volume for R(cyl.), X(cart.), or ρ(spher.) (LSO + 1 values).

Card(s) E:

	ZSO	Coordinates of source volume for Z(cyl.), Z(cart.), or θ(spher.) (MSO + 1 values).

Card(s) F:

	PHISO	Coordinates of source volume for ϕ(cyl.), Y(cart.), or ϕ(spher.) (NSO + .1 values).

NOTE: The QAD code will select a point midway between the coordinates specified as the midpoint of the source volume.

Card(s) G: (If ISRC = 2) FORMAT (8E9.4)

Column	Variable	
	FL	Weighting factors for source strength at each point specified on Card D.

Card(s) H: (If ISRC = 2) FORMAT (8E9.4)

	FM	Weighting factor for source strength at each point specified on Card E.

Card(s) I: (If ISRC = 2) FORMAT (8E9.4)

Column	Variable	
	FN	Weighting factor for source strength at each point specified on Card F.

Card(s) J: FORMAT (I5,I4)

Column	Variable	
1-5	IV	Number of boundaries used in describing this region; if region is an outer region, a minus sign must precede the number ($1 \leq IV \leq 6$).
6-9	NCOMPZN	Composition number of this region.

Card(s) K: FORMAT (6(I5,I4))

Column	Variable	
1-5	±LBD	± boundary number; a + or - sign indicates whether this boundary is + or - with respect to the region being described.
6-9	NTRYZN	Probable region of entry if this boundary is crossed.
10-14	±LBD	± boundary number.
15-18	NTRYZN	Probable region of entry.
19-23	±LBD	± boundary number.
24-27	NTRYZN	Probable region of entry.
28-32	±LBD	± boundary number.
33-36	NTRYZN	Probable region of entry.
37-41	±LBD	± boundary number.
42-45	NTRYZN	Probable region of entry.
46-50	±LBD	± boundary number.
51-54	NTRYZN	Probable region of entry.

Card(s) M: FORMAT (8(I5,I4))

Column	Variable	
1-5	NBLD	Buildup factor selection: 1 = H_2O dose buildup factor, 2 = Al dose buildup factor, 3 = Fe dose buildup factor, 4 = Pb dose buildup factor, 5 = H_2O energy absorption buildup factor, 6 = Al energy absorption buildup factor, 7 = Fe energy absorption buildup factor, 8 = Pb energy absorption buildup factor.
6-9	MATZ	Atomic number of first element used.
10-14	MATZ	Atomic number of second element used.
	•	
	•	
	•	
	MATZ	Atomic number of last element used.

Card(s) N: FORMAT (8E9.4)

1-9	COMP	Density of first element in this composition.
10-18	COMP	Density of second element in this composition.
	•	
	•	
	•	
	COMP	Density of last element in this composition.

NOTE: A similar list of the densities (in g/cm^3) must be included for each composition. The density for each of the elements specified on Card M must be listed even though it may be zero for the particular composition. The order of these cards must correspond to the composition numbers. The list of densities for a particular composition must begin on a new Card N.

Card(s) O: FORMAT (8E9.4)

Column	Variable	
	EBAR	Mean gamma energy for each group (MeV).

Card(s) P: FORMAT (8E9.4)

Column	Variable	
	GAMEN	The gamma-ray source energy spectrum, i.e., MeV/fission, MeV/capture, or MeV/decay in each gamma-ray energy group. (The units here depend upon the units on Card C for ASO).

Card(s) Q: FORMAT (8E9.4)

Column	Variable	
	CONV	Gamma flux-to-dose conversion factor for each gamma-ray energy group. For example, (rads/hr)/(MeV/cm³ sec).

Card(s) R: FORMAT (8E9.4)

Column	Variable	
	FEABSG	Gamma flux-to-heat conversion factor in Fe for each gamma-ray energy group. For example, (W/g)/(MeV/cm² sec).

Card(s) S: FORMAT (2A4)

Column	Variable	
1-8	WIDTHT	Total width of gamma-ray energy range (Example, 0.1-4.0).

Card(s) T: FORMAT (6(3A4))

Column	Variable	
1-12	WIDTHG	Width of first gamma group (Example, 4.0-3.0).
13-24	WIDTHG	Width of second gamma group (Example, 3.0-2.0).
	⋮	
	WIDTHG	Width of last gamma group (Example, 0.5-0.1).

Card U: FORMAT (18A4)

Column	Variable	
1-12	UNITG	Units of gamma flux (MeV/cm^2 sec).
13-24	UNITG	
25-36	UNITG	Units for gamma dose according to
37-48	UNITG	Card Q (rads/sec).
49-60	UNITG	Units for gamma heating in Fe
61-72	UNITG	according to Card R (W/g).

Card(s) V: FORMAT (8E9.4) (Use only if NGINT > 0 on Card B.)

1-9	WTG	Conversion factor for first gamma-ray energy group.
10-18	WTG	Conversion factor for second gamma-ray energy group.
	•	
	•	
	•	
	WTG	Conversion factor for last gamma-ray energy group.

NOTE: These are conversion factors to convert from gamma energy flux to any other units one may wish to obtain. There must be NGINT sets of these factors.

Card W: FORMAT (6(3A4)) (If NGINT > 0)

	UNITGI	The units for each of these additional quantities according to the conversion factors of card V; these units will be used to title the output.
		Format is the same as for Card U.

Card(s) X: FORMAT (3E9.4, 2(I5,I4))

1-9	RRC	The R, X, or ρ detector coordinate (cm).
10-18	ZRC	The Z, Z, or θ detector coordinate (cm or radians).

Column	Variable	
19-27	PHIRC	The ϕ, Y, or ϕ detector coordinate (cm or radians).
28-32	NRCOPT	Receiver Geometry Option: 0 = Cylindrical 1 = Cartesian 2 = Spherical
33-36	NGPF	Same meaning as on Card B; if these
37-41	NGPF	values are greater than zero, they
42-45	NGPF	override the values on Card B.

NOTE: One Card X must be used for each detector; any number of detectors may be used.

For the first case in any run, a full set of data cards (A through Y) is required. For succeeding cases stacked behind the first case some of the data cards may be omitted if various options are employed. Descriptions of these options are as follows:

Option 1. If ISRC = 0 on Card B, Cards C and I are not required. The source of the previous case is used.

Option 2. If on Card B, NSOPT = 10 (cylindrical), = 11 (cartesian), or = 12 (spherical), read Card P (gamma spectrum) next. This option is useful with Options 1 and 3 for specifying capture gamma or decay gamma sources which occur in the core. In this case Card B would be followed by Card P and the detector Cards X. Option 2 may be used with Option 3 when the capture gamma source is to be specified in a region other than the core. In this case Card P follows Card I. The detector Cards X then follow Card P.

Option 3. If NREG $\leq$ 0 on Card B, Cards J through W are not required. Cards of the previous case are used.

Option 4. If NBOUND $\leq$ 0 on Card B, Cards L through W are not required. Cards of the previous case are used.

Option 5. If MAT $\leq$ 0 on Card B, Cards M through W are not required. Cards of the previous case are used.

Option 6. If NCOMP $\leq$ 0 on Card B, Cards 0 through W are not required. Cards of the previous case are used.

VITA

Greg Dischler was born in Eunice, Louisiana on 23 March 1954. He graduated from Eunice High School in 1972 and immediately proceeded to the Baton Rouge campus of LSU where he received a B.S. degree in December of 1976. After working for two years, for Olin Chemical and Dresser Industries, he returned to LSU where he is presently a candidate for the degree of Master of Science in Nuclear Engineering. Greg is the son of Representative and Mrs. Louis Dischler of Eunice.