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PRE-SEALING RISK ANALYSIS

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Prepared by:

Daniel A. Ensminger
Michael E. Hough
Steven G. Oston

Approved by:

Thomas J. Kabele
John W. Bartlett

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THE ANALYTIC SCIENCES CORPORATION
Six Jacob Way
Reading, Massachusetts 01867

ABSTRACT

This report describes studies of accidents involving high-level radioactive waste before sealing the waste into a repository. The report summarizes work done in this area during Fiscal Year 1978 and supplements previous work. Models of accident probability, severity, and consequences are refined and extended.

TABLE OF CONTENTS

	<u>Page No.</u>
ABSTRACT	ii
List of Figures	v
List of Tables	vi
1. INTRODUCTION	1-1
1.1 Background	1-1
1.2 Organization of the Report	1-2
2. ACCIDENT MODELS	2-1
2.1 Thermal Analysis of Interim Storage Accidents	2-1
2.1.1 Description of Problem and Background	2-2
2.1.2 Waste Forms	2-5
2.1.3 Geometry	2-5
2.1.4 Heat Transfer Mechanisms and Boundary Conditions	2-8
2.1.5 Material Properties	2-12
2.1.6 The Baseline Case	2-13
2.1.7 Results of Baseline Case Analysis	2-14
2.1.8 Sensitivity Studies	2-21
2.1.9 Conclusions and Recommendations	2-24
2.2 Human Factors	
2.2.1 The Scope of a Human Factors Analysis	2-26
2.2.2 Identification of Error-Likely Situations	2-28
2.2.3 Impact of Human Factors Accidents Upon Risk	2-31
2.2.4 The Available Human Factors Data	2-34
2.2.5 Conclusions	2-36
3. CONSEQUENCE MODELS	3-1
3.1 Air Dispersion Models	3-1
3.2 Aggregation of Doses to Organs	3-3
3.2.1 Introduction	3-3
3.2.2 Equivalent Risk Model	3-3
3.2.3 Application of the Equivalent Risk Model	3-6
4. CALCULATION OF DOSE TO THE "MAXIMUM INDIVIDUAL"	4-1
4.1 Definition of the "Maximum Individual" and Other Assumptions	4-1
4.2 Components of the Maximum-Individual Dose Calculation	4-3
5. SUMMARY	5-1

TABLE OF CONTENTS (Cont.)

	<u>Page No.</u>
APPENDIX A CONVECTIVE HEAT TRANSFER FROM THE CANISTER WALL	A-1
APPENDIX B RADIATION VIEW FACTORS	B-1
APPENDIX C SOURCES OF HUMAN FACTORS INFORMATION	C-1
REFERENCES	R-1

LIST OF FIGURES

<u>Figure No.</u>		<u>Page No.</u>
2.1-1	Geometry and Principal Heat Transport Mechanisms for the 2D HEATING5 LOCA Model	2-6
2.1-2	Node Structure for 2D HEATING5 Code Loss-of- Coolant Accident Model	2-8
2.1-3	Axial Temperature Profiles for 5-yr Calcine Waste	2-16
2.1-4	Axial Temperature Profiles for 1-yr Glass Waste	2-16
2.1-5	Axial Temperature Profiles for 2-yr Glass Waste	2-17
2.1-6	Axial Temperature Profiles for 10-yr Glass Waste	2-17
2.1-7	Axial Temperature Profiles for 5-yr Multibarrier Waste	2-18
3.2-1	The Potential Hazard of HLW from the Plutonium- Stowaway Fuel Cycle Measured as Population Dose	3-7
3.2-2	The Potential Hazard of Spent Fuel Measured as Population Dose	3-7
3.2-3	The Potential Hazard of HLW from the Plutonium- Throwaway Fuel Cycle Measured as Population Dose	3-8

LIST OF TABLES

<u>Table No.</u>		<u>Page No.</u>
2.1-1	Waste Forms and Ages for Baseline Cases	2-5
2.1-2	Thermal Conductivities	2-13
2.1-3	Maximum Waste and Canister Temperatures for Baseline Cases	2-15
2.1-4	Comparison of HEATING5 2D Maximum Canister Temperatures with Previous 1D and TRUMP Results	2-19
2.1-5	Heat Loss Mechanisms and Rates for Glass Waste	2-21
2.1-6	Maximum Temperatures for Two Canister Spacings	2-22
2.1-7	Effect of Side Radiation on Maximum Canister Wall Temperature	2-23
3.2-1	Cancer Risk Factors	3-4
3.2-2	Risk Coefficients for Latent Cancer Fatalities	3-6

1. INTRODUCTION

1.1 BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) is charged with assuring that nuclear activities are conducted in a manner consistent with public health and safety. NRC has engaged the Lawrence Livermore Laboratory (LLL) to provide broad technical support in order to place the regulation of nuclear waste management activities on a sound, rational basis. TASC is under contract to LLL to help develop information which can be used by NRC as the basis for development of regulations, criteria, and standards for nuclear waste management.

Risks from disposal of nuclear waste may be divided into two categories:

- Risks which arise prior to sealing of a final repository
- Long-term risks from geologic isolation or other final disposal methods

In previous work (Ref. 1) a comprehensive framework for analysis of the pre-sealing phase of the waste management system was established. It was shown that the dominant contributions to pre-sealing risk are provided by accidents during transportation and interim storage of the waste.

This report presents some additional analyses of risks during the pre-sealing phase completed subsequent to the work reported in Reference 1. These analyses complement and extend

the studies in Reference 1. They are directed at three principal purposes:

- More detailed examination of processes which the previous analysis had highlighted as significant contributions to total risk
- Consideration of additional accident scenarios
- Development of more effective methods of presenting the information generated by the modeling effort.

All the analyses in this report deal with management of high-level waste (HLW) from the "plutonium-stowaway" fuel cycle, in which power is generated from low-enrichment uranium fuel and both uranium and plutonium are removed from the spent fuel in a reprocessing plant. Due to a shift in program emphasis during the study toward direct disposal of spent fuel, resources were not available to assemble the results of these studies into a calculation of total pre-sealing risk like that of Reference 1.

1.2 ORGANIZATION OF THE REPORT

Chapter 2 of the report describes studies of the probabilities and mechanisms of pre-emplacement accidents. Section 2.1 reports a thermal analysis of interim storage accidents, while Section 2.2 discusses the possible contribution of human errors to different accident scenarios.

The consequences of various accidents are addressed in Chapter 3. Section 3.1 deals with modeling the dispersion of radionuclides released to the atmosphere. Section 3.2 presents a method of aggregating the doses of radioactivity

received by several organs in the human body so as to reflect the total risk of health effects.

Chapter 4 extends the summary dose calculations of Reference 1 by presenting a method of calculating doses to a "maximum", or worst-situated, individual. A brief summary of the entire report is presented in Chapter 5.

2.

ACCIDENT MODELS

2.1 THERMAL ANALYSIS OF INTERIM STORAGE ACCIDENTS

This section describes heat transfer calculations for estimating the thermal response of high-level solid waste (HLSW) canisters in an interim storage basin after a loss-of-coolant accident (LOCA). The principal modes of heat transfer from canister walls considered were convection to air and radiation to space. Calcine, borosilicate glass, and supercalcine multi-barrier waste forms with ages ranging from 1 year to 10 years after irradiation were considered.

The primary objectives of this analysis were:

- To extend previous one-dimensional interim storage LOCA thermal analyses (Ref. 1, Appendix B) by the use of a two-dimensional model
- To estimate the effects of canister spacing on waste and canister temperatures.

The HEATING5 computer code was used to estimate the maximum waste centerline and canister wall temperatures for steady-state conditions after the LOCA. Results were obtained as a function of:

- Waste form and age
- Canister spacing
- Boundary conditions.

The primary objective of extending the previous one dimensional thermal analysis with a two-dimensional one was accomplished. The maximum canister wall temperatures computed were found not to differ significantly from previous results. However, in accomplishing the second objective, it was observed that the accuracy of the thermal analysis could be greatly improved by incorporating a more sophisticated air flow model into the analysis. Thus our results should be viewed as a development of methodology rather than a rigorous thermal analysis of the problem.

2.1.1 Description of Problem and Background

For the purposes of this study, it was assumed that during interim storage at a surface storage facility, high-level wastes are contained in stainless steel canisters. Large numbers of canisters are grouped in a square array and are located in a concrete basin filled with water to remove heat generated by the waste itself. A seismic event or other catastrophic accident could cause rupture of the basin and loss of the coolant water. As a consequence, failure of the canisters could well occur due to corrosion or melting of the canister wall; both are thermally induced effects. The rapidity of failure depends upon the heating rate of the wastes and on the geometry of the canister array.

In a previous TASC study (Ref. 1, Appendix B), the transient temperature response of a canister array following an instantaneous loss of cooling water was analyzed. The facility modeled closely paralleled that suggested in Reference 2. It was assumed to consist of 506 10-ft long waste canisters set into a rectangular concrete basin. The top of the basin was assumed to be open so that heat could be completely removed from the vicinity of the canister array. The canisters were

vertical in two layers, one directly above the other, with a 1-ft gap between. The canisters were taken to be 1 ft in diameter on a 1.5-ft center-to-center spacing.

The one-dimensional study used a lumped-parameter model in which the canister array was represented by a single node in a one-dimensional geometry. Radiation to space from the canister ends and convection to naturally circulating air were the principal heat transfer mechanisms treated. Computations were performed with the HEATING5 computer code (Ref. 3) to solve the finite difference heat flow equations.

The results obtained provided information on:

- Canister temperature-time histories for various waste ages
- Sensitivity of steady-state canister temperatures to canister spacing
- Sensitivity of canister failure probabilities to the distribution of waste ages in the storage facility.

A recommendation made at the conclusion of the one dimensional study was to improve the accuracy of the results by means of a two-dimensional analysis of heat transfer through the waste canister array.

More recently, Lawrence Livermore Laboratory (LLL) (Ref. 4) performed preliminary analyses with the TRUMP code on the interim storage LOCA. The study examined the steady-state axial temperature gradients in a canister. It obtained higher temperatures than were produced by the lumped waste model and concluded that there was a potential for more serious accident consequences. The model used in the LLL study features the following heat transfer paths:

- Axial conduction in the waste
- Radiation from canister top to space
- Convection from canister sides to air
- Radiation from canister bottom to basin.

The waste and canister regions were together represented by a single node at each axial position; hence, radial conduction effects in the canister were ignored. Material properties were assumed to be constant with temperature.

The purpose of the work reported here was to extend the prior studies by considering:

- A two dimension steady-state treatment of the LOCA problem, including axial conduction
- Waste and canister as separate components including radial conduction
- Radiation heat transfer from canister sides
- Temperature-dependent material properties
- Various waste forms and ages at the time of the LOCA

Primary emphasis in this work was placed on developing a methodology for addressing the above elements. While this was accomplished, it was discovered that a much more rigorous approach must be used to produce an accurate, comprehensive thermal analysis of this LOCA problem. Certain elements of the work reported here would provide an important contribution to such an analysis.

2.1.2 Waste Forms

Calcine, borosilicate glass, and supercalcine multi-barrier waste forms were included in the study, with ages ranging from 1 to 10 years after irradiation. The waste was taken to be diluted to the extent that all 10-yr wastes produced 3.5 kW of heat per canister. The heat generation rates per canister at other ages for the same dilution were:

- 5-yr: 7.4 kW
- 2-yr: 18.1 kW
- 1-yr: 32.0 kW.

The matrix of wastes and ages for baseline cases addressed in this study are represented in Table 2.1-1.

TABLE 2.1-1
WASTE FORMS AND AGES FOR BASELINE CASES

WASTE FORM	WASTE AGES (yr)			
Calcine	1	2	5	
Glass	1	2	5	10
Multibarrier	1	5		

2.1.3 Geometry

Figure 2.1-1 depicts the assumed geometry for LOCA calculations. The two-dimensional HEATING5 model consisted of the following material regions:

Figure 2.1-1 Geometry and Principal Heat Transport Mechanisms for the 2D HEATING5 LOCA Model

- A waste canister with appropriate boundary conditions
- A concrete basin underlying the canister
- An earth layer beneath the concrete slab.

No attempt was made here to model the concrete sides of the basin -- i.e., the canister array was taken to be infinitely large and without edge effects. The air gap between vertical canisters was not modeled separately; the stacked canisters were assumed to act as one continuous medium. Heat transfer to the flowing air was treated in terms of a boundary condition, as discussed in Section 2.1.4.

Figure 2.1-2 depicts the arrangement of nodes that constitute the HEATING5 model.* The waste canister was taken to be 20 ft in length to represent the two-layer stack of canisters, each 10 ft long.† Nodes in the waste region were spaced vertically every 0.5 ft. In the radial direction the waste region nodes were spaced every 0.25 ft.

The stainless steel canister was similarly divided axially into 40 zones. Further radial subdivisions in the canister were not deemed necessary since the canister is only 3/8 in. thick.

Below the canister region were located a 1-ft thick concrete slab (the basin) and 8 ft of earth. Node spacing in the earth layer was at 2-ft intervals in the axial (z) direction.

*This diagram represents R-z (cylindrical) coordinates with the canister centerline coinciding with the z axis.

†The term "canister" in the subsequent text will refer to the 20-ft long canister region although strictly a single canister is but 10 ft long.

Figure 2.1-2 Node Structure for 2D HEATING5 Code Loss-of-Coolant Accident Model

2.1.4 Heat Transfer Mechanisms and Boundary Conditions

The principal heat transfer mechanisms taking place during a LOCA are indicated schematically in Fig. 2.1-1. These mechanisms may be divided into two categories: conduction within solid material regions -- i.e., the waste canister and the underlying basin and earth -- and boundary conditions on the waste canister that include both convection and radiation. For steady-state conditions, conduction is simply a function of the thermal conductivity of the materials and the geometry. The convection and radiation boundary conditions are more involved and are discussed in some detail below.

Convection to Air from Canister Sides - Heat was assumed to be transferred from the canister wall to the surrounding air by convection. A rigorous approach to this thermal analysis would employ a free convection heat transfer model using appropriate mass, momentum, and energy balances on the air stream and waste canisters. However, such an approach was beyond the scope and resources of this task. Therefore, a simplified and admittedly non-rigorous approximation was used to model the convective heat transfer. As will be seen, this approximation, while not strictly rigorous, satisfies the limited objectives of this analysis.

The approach chosen to obtain a convective heat transfer coefficient for the transfer of thermal energy from the waste canisters to the air stream was a modification of the method employed by LLL in related studies (Refs. 4 and 5). A value of air flow rate was arbitrarily but carefully chosen. Previous results from similar analysis (Refs. 2, 4, and 5) guided this choice of air flow rate. This fixed value of air flow was then used in conjunction with a forced convection heat transfer coefficient to model the heat transfer to the air in the vertical channels between the waste canisters.

For this analysis, the physical and thermal properties of the air stream were evaluated at the film temperature, roughly equal to the arithmetic mean of the canister wall temperature and the bulk air temperature. Based upon these properties (taken from Ref. 6), values of the convection heat transfer coefficient, h_c , were computed as a function of the film temperature. A linear fit to the h_c versus temperature over the temperature range of interest for the chosen air mass flow rate yielded the following expression for the temperature dependence of the convection coefficient:

$$h_c = 0.475 + 1.05 \times 10^{-4} T_f \quad (2.1-1)$$

where T_f is the film temperature in °F and h_c has units of Btu/hr-ft²-°F. As a matter of perspective, this expression for the heat transfer coefficient yields values which are typical of free convection heat transfer coefficient for gases (Ref. 7).

The convective heat transfer coefficient was used by HEATING5 to determine the heat flux, q_c , for each axial zone along the length of the canister according to

$$q_c = h_c(T_w - T_a) \quad (2.1-2)$$

where

T_w = canister wall temperature (°F)

T_a = air (bulk) temperature (°F)

Both T_w and T_a depend upon the axial position, z .

The bulk air temperature as a function of z is not known except at the bottom of the canister array where it has been assumed to be 120°F (Refs. 1 and 4). Appendix A outlines a modification made to the HEATING5 computer code to handle the type of situation where heat exchange is accomplished by the flow of a coolant. The new subprogram determines the air temperature and an effective value of h_c by solving the air heat balance equation numerically.

Radiation From Canister Sides --Two separate 2D radiation problems were considered here:

- Radiation from canister sides to space
- Inter-canister radiation.

The information needed for HEATING5 to compute radiation-to-space heat fluxes was the following:

- The view factors from various zones along the canister length to space. Determination of these view factors is discussed in Appendix B
- The emissivity of the canister, assumed to be $\epsilon = 0.45$ (Ref. 2)
- The equivalent blackbody temperature of space, assumed to be 75°F (Ref. 2).

The inter-canister radiation problem was not treated. Such calculations might have proved whether or not multiple reflections would lead to a net transfer of heat by radiation in an upward direction, thus reducing the axial canister temperature gradients. However, with a geometry consisting of a square array of finite length circular cylinders, tracing the multiple reflection paths to determine gray body shape factors was judged to be too complicated a mathematical exercise to pursue. Neglect of this effect is conservative; that is, it leads to canister maximum temperatures that may be somewhat too high.

Convection to Air from Canister Bottom - The heat transfer mode for convection from air to the canister bottom remains the same as in the previous study. The natural convection coefficient for flat plates (Ref. 8):

$$h = 0.1375 (\Delta T)^{1/3} \quad (2.1-3)$$

was employed, with ΔT expressed in °F. Convection to air from the canister was determined to be an unimportant mechanism (see Section 2.1.8).

Radiation to Basin from Canister Bottom - The following assumptions, identical to those in the previous work, were made:

- View factor = 1.0
- Canister bottom emissivity = 0.45
- Concrete emissivity = 1.0.

Radiation to Space from Canister Top - The radiation heat losses from the canister top were computed using:

- A view factor of 1.0. In Reference 1 a view factor of 0.5 was used for canister tops to space. An additional radiation interchange term to the sides of the concrete basin was included also with a view factor of 0.5. No provision in this study was made for the concrete basin sides above the level of the canister tops; hence all radiation was considered to be directed to space. Refer to Section 2.1.8 for further discussion of this point
- An emissivity of canister tops equal to 0.45
- A space equivalent black body temperature of 75°F.

2.1.5 Material Properties

Table 2.1-2 gives the thermal conductivities for all materials used in these calculations, plus the sources of the data. The HEATING5 code interpolates linearly between tabulated entries. At temperatures outside the range indicated in the tabulation, conductivities are taken to be constant at the lower or upper limit values. Since the calculations were all performed for steady-state conditions, density and specific heat information were not required for solid materials. The properties of air used in calculating the convective heat

TABLE 2.1-2
THERMAL CONDUCTIVITIES

MATERIAL	THERMAL CONDUCTIVITY (Btu/hr-ft-°F)	REFERENCE
Calcine HLSW	32°F 0.097	9
	1652°F 0.226	9
Glass HLSW	77°F 0.408	9
	1112°F 0.010	9
	1346°F 1.44	9
Multibarrier HLSW	5.76	9
Stainless Steel	9.8	9
Concrete	0.25	4
Earth	0.85	4

transfer coefficient are standard (Ref. 6) and are not repeated here.

2.1.6 The Baseline Case

A baseline case was established whose geometry and heat transfer parameters had reasonable nominal values and that could be used to compare the effects of different waste forms and waste ages on equilibrium temperatures. The following key parameters characterize the baseline case:

- Air mass flow rate of 500 lb/hr. This value was chosen to match that recommended by LLL in Ref. 4; it is discussed further in Section 2.1.7
- 1.5-ft canister center-to-center spacing (Ref. 2).

The waste forms and ages treated were presented in Table 2.1-1.

2.1.7 Results of Baseline Case Analysis

The HEATING5 code provided maps of the steady-state temperatures in the waste canister and basin materials at each of the 188 nodes indicated in Fig. 2.1-2, and the air temperatures associated with each of the 40 nodes along the canister wall. For purposes of judging the canister failure temperatures, however, just two locations were of vital interest. These locations were where the maximum temperatures occurred anywhere along the length of the canister:

- In the waste material (at the canister center-line)
- At the stainless steel canister inner wall surface.

For a potential canister corrosion failure the canister wall temperature is the more important.

Table 2.1-3 presents the baseline case maximum waste-centerline and canister wall temperatures. For reference, the canister temperature failure criteria based upon internal corrosion (Refs.9 and 10) for the three waste forms are:

- Calcine: $2370 \pm 90^{\circ}\text{F}$
- Glass: $2190 \pm 90^{\circ}\text{F}$
- Multibarrier: $2370 \pm 180^{\circ}\text{F}$.

The following observations are of interest:

- Radial temperature gradients in the calcine are large owing to the low thermal conductivity of that waste form. Calcine centerline temperatures are near the melting point (2550°F) for 5-yr waste and greatly exceed it for

TABLE 2.1-3
MAXIMUM AND CANISTER TEMPERATURES
FOR BASELINE CASES

WASTE		MAX. WASTE TEMPERATURE (°F)	MAX. CANISTER TEMPERATURE (°F)
MATERIAL	AGE (yr)		
Calcine	1	****	****
	2	****	****
	5	2415	1520
Glass	1	****	****
	2	****	****
	5	1660	1520
	10	990	860
Multibarrier	1	****	****
	5	1550	1510

Air Flow: 500 lb/hr
Canister Spacing: 1.5 ft

***Canister wall temperatures beyond the ~2600°F stainless steel melting point.

younger wastes. Use of internal fins would obviate this problem, as discussed in References 9 and 10

- Maximum canister wall temperatures exceed the failure criteria for all three waste forms aged 1-yr and 2-yr
- Maximum canister wall temperatures do not depend significantly upon the waste form. In Table 2.1-3 temperature differences among the three waste forms at the same age reflect numerical errors in the code finite difference solutions. Thus, for a given age, the wide differences in the thermal conductivity between calcine and multibarrier will not affect canister failures by corrosion.

A demonstration of the axial position dependence of temperatures in the waste, canister, and the air predicted by HEATING5 is given in Figs. 2.1-3 through 2.1-7 for 5-yr calcine;

Figure 2.1-3 Axial Temperature Profiles for 5-yr Calcine Waste

Figure 2.1-4 Axial Temperature Profiles for 1-yr Glass Waste

Figure 2.1-5 Axial Temperature Profiles for 2-yr Glass Waste

Figure 2.1-6 Axial Temperature Profiles for 10-yr Glass Waste

Figure 2.1-7 Axial Temperature Profiles for 5-yr Multibarrier Waste

1-yr, 2-yr, and 10-yr glass; and 5-yr multibarrier. Features of these plots are:

- Sharp temperature gradients at both ends of the canister (due to radiation and convection at the ends), with relatively broad maxima
- Maximum wall and waste centerline temperatures occurring toward the bottom of the canister, a result of side radiation heat losses
- Air temperature rising monotonically and then leveling off near the top of the canisters where the air and canister-wall temperatures approach the same value.

Table 2.1-4 compares maximum canister wall temperatures for 1-yr to 10-yr glass wastes computed by the two-dimensional HEATING5 model with results obtained from previous models. These results are assumed independent of waste form, as indicated in the discussion of Table 2.1-3. Referring to Table 2.1-4 for each waste age:

- Column 2 repeats the results presented in Reference 1 for the lumped-parameter 1D HEATING5 model. The 1D temperatures computed were uniform over the canister region because of the nature of the simple model. They represent average canister wall temperatures that cannot be easily related to 2D model results because of the non-linear processes involved.
- Column 3 repeats the baseline glass results from Table 2.1-3 of this report for a constant value of the air mass flow rate of 500 lb/hr.

TABLE 2.1-4

COMPARISON OF HEATING5 2D MAXIMUM CANISTER TEMPERATURES WITH PREVIOUS 1D AND TRUMP RESULTS

WASTE AGE (yr)	HEATING5 MODELS		
	1D (Ref. 1)	2D CONSTANT MASS FLOW RATE	TRUMP (Ref. 4)
1	****	****	****
2	****	****	****
5	1720	1520	2200
10	970	860	1100

Temperatures in °F

***Canister wall temperatures beyond the ~2600°F stainless steel melting point.

- Column 4 presents the findings of the preliminary TRUMP studies. These temperatures are radially averaged over the waste canister region for calcine and, therefore, are not directly comparable to the canister wall temperatures of the three previous columns.

The principal conclusion derived from Table 2.1-4 is that the 2D HEATING5 results compare well with the previous 1D results. Both 1D and 2D HEATING5 temperatures are considerably lower than the corresponding TRUMP temperatures. Three reasons can be advanced for these differences:

- TRUMP-computed temperatures for the waste canister are radially averaged temperatures, which are higher than the canister wall temperatures
- TRUMP did not account for heat loss by side radiation
- The TRUMP convective heat transfer coefficient h_c was a constant (at 0.46 Btu/hr-ft²-°F). The HEATING5 h_c was temperature-dependent and slightly higher than the TRUMP value (see Eq. 2.1-1).

Table 2.1-5 compares the relative importance of the various radiation and convection heat loss mechanisms from the waste canister for glass waste aged 1, 2, 5, and 10 years. Heat losses from the canister sides dominate end effects under all circumstances. Except for the 1-yr waste, side convection is the chief heat transfer mode. Hence, canister wall temperatures in the threshold failure regime, wastes aged 2 to 5 years, will be most sensitive to model parameters that affect convection rather than those that affect radiation.

TABLE 2.1-5
HEAT LOSS MECHANISMS AND RATES FOR GLASS WASTE
(AIR FLOW OF 500 lb/hr)

HEAT LOSS MECHANISM		LOSS RATES FOR GLASS HLSW			
PROCESS	LOCATION	1 YR	2 YR	5 YR	10 YR
Convection	Side	52,800	38,300	19,400	9,900
Radiation	Side	56,500	21,600	5,200	1,300
Radiation	Top	2,700	900	400	100
Convection	Bottom	2,200	1,300	200	200
Radiation	Bottom	100	100	0	0
TOTAL HEAT LOSS		114,300	62,200	25,200	11,500

Heat losses are expressed in units of
Btu/hr per 10 ft of canister length

2.1.8 Sensitivity Studies

Canister Spacing Sensitivity - One option for reducing waste and canister temperatures in a LOCA is to increase the inter-canister spacing. In the current work, temperatures were compared at the nominal 1.5-ft spacing and at a greater, 3.0-ft spacing. The results are shown in Table 2.1-6 for maximum canister wall and waste centerline temperatures using 1, 2, 5, and 10-yr glass.

There does appear to be a moderate temperature decrease due to doubling of the spacing. The main reason for the reduction in canister maximum temperature at a 3-ft spacing is the increased view factor for radiation to space from the canister sides, particularly near the canister mid-plane where maximum temperatures generally occur.

TABLE 2.1-6
MAXIMUM TEMPERATURES FOR TWO CANISTER SPACINGS

SPACING (ft)	LOCATION	TEMPERATURE FOR GLASS HLSW			
		1 YR	2 YR	5 YR	10 YR
1.5	Canister	****	****	1520	860
	Waste Centerline	****	****	1660	990
3.0	Canister	****	2230	1260	730
	Waste Centerline	****	2575	1400	865

Temperatures in °F

****Canister wall temperatures beyond the ~2600°F stainless steel melting point.

In this sensitivity study the air flow rate was maintained at 500 lb/hr. Physically this is not a realistic assumption; a change in flow rate is intuitively expected for larger unit cells because of the greater volume of air present. It is a consequence of the constant flow rate model that heat losses due to convection actually were decreased by going to a larger canister spacing. With a model that allowed flow rate to increase with spacing, temperature differences between the 1.5-ft and 3.0-ft cases would have been even greater.

In Reference 1 it was concluded that canister temperature sensitivity to spacing was low. For instance, with 2-yr glass waste the previous 1D results showed a decrease in canister temperature of 490°F between 1.5-ft and 3.0-ft spacing. The present 2D results indicate a 600°F canister wall temperature difference. Again, the inclusion of side radiation effects is the cause.

Sensitivity to Boundary Conditions - A sensitivity study was performed to determine whether the assumption of an end radiation view factor to space of 1.0 compared to the view factor of 0.5 used in the previous 1D work would affect maximum canister temperatures. In addition, the assumption at the canister top of no convection to air in the current calculations was tested against an upper bound end convection heat transfer coefficient equal to twice the value given by Eq. 2.1-3. In neither case did maximum canister wall or waste centerline temperatures change. These studies were carried out with 5-yr calcine; over a variety of top boundary conditions the maximum canister wall temperatures only varied from 1520°F to 1527°F. Similar results would be expected for younger wastes because canister top heat losses are comparably small.

Also evaluated was the sensitivity of maximum canister temperatures to side radiation. Steady-state calculations were performed for 2, 5, and 10-yr glass with no side radiation allowed. The results are compared in Table 2.1-7 to the base-line cases with side radiation.

TABLE 2.1-7
EFFECT OF SIDE RADIATION ON MAXIMUM
CANISTER WALL TEMPERATURE

GLASS HLSW AGE (yr)	MAXIMUM CANISTER TEMPERATURES (°F)	
	WITH SIDE RADIATION	WITHOUT SIDE RADIATION
2	****	****
5	1520	1750
10	860	940

****Canister wall temperatures beyond the ~2600 °F stainless steel melting point.

Where canister failures will clearly occur -- i.e., for 2-yr waste -- neglect of side radiation gives a significant temperature increase. For 5-yr waste the effect of side radiation is considerably less, and the computed 230°F temperature difference does not take the canister wall over the failure threshold of 2200°F. Thus, any uncertainties in the side radiation view factors are more important for the younger waste, but in any event they are unlikely to significantly affect failure conclusions.

2.1.9 Conclusions and Recommendations

The two-dimensional thermal analysis reported here was conducted primarily to evaluate methodology. The analysis permits some general conclusions concerning the usefulness of these methods. However, these conclusions should be evaluated in the light that they are drawn from a simplified, parametric analysis. Their use therefore should be restricted to definition and scoping of potential future research and analysis topics. Our conclusions and observations are summarized below.

- The current 2D HEATING5 maximum canister wall temperatures are in the same range as those of the previous 1D calculations
- Convection from the canister sides to circulating air is the predominant heat transfer mechanism for all but very young wastes
- The thermal conductivity of the waste form in the axial conduction model does not significantly affect maximum canister wall temperatures
- Sharp temperature gradients occur at both ends of waste canisters for all three waste forms studied. Temperature maxima tend to be broad and to be located near or below the interface between the two layers of canisters

- Uncertainties in canister side radiation view factors are not likely to alter canister failure probabilities substantially, although they will affect canister temperatures, for young wastes especially
- Canister and waste temperatures are somewhat sensitive to changes in canister spacing
- Details of canister top boundary conditions do not significantly influence maximum waste or canister temperatures.

The following additional studies are recommended to extend the work reported in this section:

- A much more detailed set of air flow calculations involving mass, momentum, and energy balances should be carried out to characterize the convective heat transfer process more completely
- The current two-dimensional studies were performed for an open storage basin where heated air and thermal radiation leave the storage area. The calculations should be redone for a basin with a pool that would recirculate cooling air and reflect radiation, thus increasing canister temperatures.
- A catastrophic accident during interim storage less severe than a LOCA could cause a loss of coolant circulation and also compromise the capability of the facility to contain airborne radioactive particles without necessarily destroying the integrity of the basin itself. A study should be undertaken to determine the transient thermal response of the basin during the boil-off period. Times-to-failure would be longer than for a LOCA.

2.2 HUMAN FACTORS

Human factors accidents affecting the nuclear waste management cycle from waste solidification through emplacement in the final burial site were considered. The major steps of this cycle include events at the reprocessing plant, during transportation between the reprocessing plant and repository, and at the repository.

2.2.1 The Scope of a Human Factors Analysis

Human reliability engineering came into existence for the purpose of "optimizing" system design (e.g., increasing productivity, etc.) despite the presence of an error-prone human who must operate the system. Sandia Laboratories pioneered in these efforts using an "interactive" (i.e., observe human errors and modify the system to reduce these errors) design technique called "man-machine systems analysis" (Ref. 10), which consists of:

- Identifying all the goals and functions of a system in all phases of operation
- Identifying all relevant human actions in each phase with particular attention to situations which are "error-likely" in the judgment of the analyst
- Estimating the effects of "error-likely situations"
- Modifying the system to eliminate "error-likely situations".

For the high-level solidified waste (HLSW) management cycle, the "system" consists of all operations in the waste management sequence from the time that the waste is solidified through sealing in an underground repository. The phases of the cycle to which particular emphasis is given include:

- Activities at the fuel reprocessing plant after solidification
- Transportation by truck and train to the repository
- Activities at the repository up to emplacement.

The goal was to define the "risk" associated with human factors accidents affecting these operations.

Accidents resulting in release of radioactive material to the biosphere include handling and interim storage accidents at the reprocessing plant, train and truck impact and fire accidents, and handling accidents at the repository. In order to identify the most "error-likely situations" it is first necessary to consider the basic types of human errors that are possible:

- Errors to omission (i.e., failure to perform a task)
- Errors of commission (i.e., given that a task is performed, failure to do it correctly)
- Sequential errors (i.e., failure to do the task in the proper order)
- Extraneous acts (i.e., performance of a task that shouldn't have been done)
- Time errors (i.e., violation of deadlines).

While it is possible to find information (Ref. 11) giving order-of-magnitude estimates of errors in these five categories, the most meaningful analysis of "error-likely situations" in the nuclear waste management cycle consists of direct observation of and interview with the people doing the work. Although

this is not within the scope of the analysis, a list of the most significant human factors was compiled (see Section 2.2.2).

In order to estimate the effects of human factors accidents, the expected value of dose must be computed. Expected dose is proportional to the product of:

- Probability of occurrence of a human factors error which would, if uncorrected, lead to an accident
- Given that the error occurs, the probability that it goes uncorrected
- The fraction of nuclear waste released to the biosphere given the occurrence of the uncorrected error.

Although an accident may have a low probability of occurrence, the associated severity may cause a substantial contribution to the risk. Sections 2.2.3 and 2.2.4 address the difficulties which were encountered in obtaining numerical estimates of these factors. Although the ultimate goal is modifying the cycle to eliminate the possibility of accidental release, this will not be addressed by this analysis.

2.2.2 Identification of Error-Likely Situations

Handling Accidents at the Fuel Reprocessing Plant -

Human factors can cause crane drop accidents in the reprocessing plant. The probability of occurrence of 1.5×10^{-6} crane drops per hour used in Reference 1 is totally due to human factors. However, since the release fraction for handling accidents is zero, human factors handling accidents do not affect dose.

Interim Storage Accidents - In the normal course of maintenance of the storage pool, errors of omission could result in drainage valves being left open. This could not lead to pool drainage, however, since these storage pools are drained from above rather than below.* Accidental activation of pool drainage pumps is not considered to be a plausible human factors accident because of safety interlocks, alarms, and the large time constants required for drainage to occur. A large earthquake leading to destruction of the entire building remains the most significant drainage accident. Faulty maintenance and inadequate design of the pool support equipment (e.g., pumps), both of which are errors of omission, can lead to a mechanical breakdown of the pool circulation system. Based on nuclear reactor operating experience (Ref. 12), it is possible that the support equipment cannot withstand a "small" earthquake of insufficient intensity to rupture the pool. Should this lead to a mechanical failure of the cooling circulation system with no possibility of repair, boiloff can occur, resulting in a melt-down of the waste canisters and release of volatiles and airborne particulates. In order for this accident to produce a significant change in expected dose, the air filtration system must fail. Although this scenario was considered in Reference 1, the impact of human factors causing failure of the air filtration system was not addressed.

Failure of the Air Filtration System - Ganged HEPA filters provide an excellent barrier against the escape of airborne particulates released from accidents occurring inside the reprocessing plant. Although the likelihood of improper installation or damage as a result of installation (e.g., thumbholes) is relatively high, failure to detect a leak

*Personal communication with G. Kniazewycz, TERA Corporation, Berkeley, Calif., 20 December 1977.

during post-installation checkout is very unlikely.* A flooding accident affecting the HEPA filters is also significant since the HEPA filters become completely inoperable. Although nuclear reactor operating experience (Ref. 13) indicates that "common-mode" accidents involving air filtration systems have occurred, the likelihood of these accidents cannot be assessed without a specific plant configuration. Charcoal filters (for volatiles) and silica-gel filters (for volatiles and airborne particulates) were also considered, but meaningful data discussing the relevance of human factors could not be found.

Transportation by Truck and Train - Although spent fuel transportation casks were not visibly damaged in truck and train impact testing at Sandia Laboratories in 1977, human factors can compromise the integrity of the cask through improper closure. For example, if most or all of the bolts which seal the cask are either not tightened sufficiently or left out altogether, the cask lid can open as the result of an impact accident. Further analysis is needed to determine whether the canisters leave the cask during the instant at which impact occurs. If this occurs, airborne particulate dispersion of waste could result.

Activities at the Repository - Human factors at the repository enter via crane drop accidents while moving the waste canister between its transportation and transfer casks. Human factors were already included in the probability of occurrence of 1.5×10^{-6} drops per hour of operation used in Reference 1. As in the case of handling accidents at the reprocessing plant, this accident has an associated release fraction of zero, with no contribution to risk. Further con-

*Personal communications with air filtration system specialists. See Section 2.2.3 for specifics.

sideration of human factors initiating these accidents is not necessary.

Summary - Error-likely situations involving human factors can cause failure of air filtration systems in re-processing plant accidents and aggravate releases to the biosphere in transportation impact accidents.

2.2.3 Impact of Human Factors Accidents Upon Risk

In order to quantify the risk, data for probability of occurrence of human error, probability of lack of detection and correction of the error (given that it has occurred), and fraction of release of waste as a result of the accident must be obtained. Information such as probability of occurrence of a human factors accident is equivalent to the product of the probabilities of an error and lack of detection or correction given the error. Probability of an error can be obtained easily from the literature (e.g., Refs. 10 and 11) when considering general errors of omission, commission, and the like. A "generic" human factors analysis, however, is not meaningful since error rates will depend on presence of checklists, redundancy of personnel performing a task, etc. Although a detailed "task analysis" is often necessary, the approach taken for this analysis was to construct probabilities of occurrence based on actual accidents involving radioactive materials, where data was available. In the worst case, the release fraction was assumed to be one (i.e., total release). Detailed analysis of the release fraction is necessary to refine this conservative estimate. The primary accidents considered, based upon conclusions from the previous section, involved failure of air filtration systems and transportation.

Air Filtration System - All air filtration system specialists contacted concur that failure of HEPA filters, due

to their fragility, is almost exclusively caused by improper installation despite strict installation standards. Based on operating experience, more than 1 out of every 100 HEPA's fail due to improper installation.* Although charcoal bed filters for trapping volatiles are less fragile, again the potential for failure comes from improper installation. However, improper installation is usually detected when in-place filter tests are conducted at periodic (e.g., every 18 months intervals).† Based on operating experience and given that the HEPA system is tested once after installation, the chances of overlooking a leak are less than 1 in 1000.* These numbers were of little value as stated since they do not address the problem of how often these failures actually occur.

Air filtration system accidents at nuclear power plants between 1969 and the present (Ref. 13) indicate that one maintenance error and two operator errors occurred in which improper installation was followed by faulty monitoring of the particulate filtration system. Another cause of HEPA failure was a common-mode accident affecting the HEPA system. Although a large number of human common-mode accidents are possible in the reprocessing plant, the most significant accident involves flooding the filter with water. This is confirmed by accident reports (Ref. 13) collected over the past decade from nuclear power generating plants in which one installation error and three operator errors elsewhere in the system resulting in HEPA flooding. When flooded, the HEPA filter disintegrates* and the transmission ratio of the filter is unity. If an accident resulting in release of airborne radioactive particulates were to occur at the same time as

*Personal communication with J. Jacox, Nuclear Consulting Services, Inc., Columbus, Ohio, 14 December 1977.

†Personal communication with C. Burchsted, ORNL, 15 December 1977.

such a failure, and if the particulates were to reach the filter, a significant fraction might escape. The radiological consequences of such an accident would depend on meteorological conditions and the demography of the surrounding region.

The preliminary findings indicate that human errors in HEPA air filtration systems, compounded with other system failures, can pose a potentially large risk. Data on charcoal and silica-gel failures was even harder to find, probably because these filters are less susceptible to damage. Further investigation is necessary.

Transportation by Truck and Train - Accidental release of radioactive material caused by impact and fire associated with vehicular accidents was considered in detail in Reference 1. Since human factors are obviously involved in the causes of the transportation accidents, no further consideration of these is necessary since the accident statistics used in that report already take them into consideration. Instead, attention was focused on human errors which could aggravate releases by compromising the ability of the shipping casks to withstand the accident. In the worst possible case, omission of all of the bolts which seal the lid of the shipping cask coupled with faulty inspection prior to shipping could enable the canisters to escape from their protective environment in an impact accident. Due to strict monitoring procedures prior to shipping, improper closure is unlikely.* Quantitative

*Personal communications with A. Carson, G.E. San Jose, Calif., and E. E. Boyland, G.E. Operations Manager regarding IF-300 shipping casks. Mr. Boyland reports that between 1972 and 1977, there were 415 truck shipments of spent fuel elements. None were involved in any accidents.

likelihood estimates from cask shippers* and manufacturers were impossible to find, making evaluation of this scenario impossible.

Human factors specialists at Sandia Laboratories, Albuquerque, N.M., performed a study to determine the transportation risks of all radioactive material based on actual accidents which occurred throughout the U.S. Over the past year, Sandia has been funded by NRC to assess the environmental impacts of human errors in all radioactive material transportation, including train and truck, through urban areas. Probabilities of occurrence of human error in storage, handling, and packaging operations were constructed from accident reports submitted to D.O.T. (1971-1977) and NRC (1975 only). Since none of the incidents reported (Ref. 14) involved spent fuel or HLSW shipping casks, a direct quantitative assessment of transportation risk could not be made. Conservative estimates of risk could be made using other available data not relevant to this specific type of packaging, but these estimates would not be meaningful.

2.2.4 The Available Human Factors Data

Direct evaluation of the probability of occurrence of a human factors accident and the release fraction given the accident as they apply to the nuclear waste management cycle was not possible. After numerous personal interviews with specialists in government, private industry, and academia, it was clear that any available relevant information on human factors consisted of incident reports pertaining to the nuclear power industry. Some of the more useful sources of data were:

*Personal communication with E. Rutenkroger, Tri-State Co., Joplin, MO, 15 November 1977. Mr. Rutenkroger reports that even if any human factors accidents occurred, no record was kept in order to keep the paperwork within limits.

Accidents in the Nuclear Power Industry - Oak Ridge National Laboratory computer files have a wealth of human factors accident reports on the operation of nuclear power generators over the time period 1969 through the present. Specific incidents relating to circulation failure at spend fuel pools, air filtration systems (HEPA and charcoal filters), waste handling, and transportation were obtained. While these incidents themselves could not be used to construct meaningful probabilities, they did provide useful insight into the type of human factors accidents which can occur.

Transportation of Radioactive Material - Human factors specialists at Sandia Laboratories have assessed the environmental impact of human factors in the transportation of radioactive material through urban areas (Ref. 14). Their data base consists of incident reports prepared for governmental agencies, such as the Department of Transportation Office of Hazardous Materials and NRC, as well as reports prepared by dedicated shippers of radioactive materials, such as Tri-State Co. (Jopkin, MO). These reports, together with a survey conducted by Battelle (Ref. 15) to determine the total number of shipments of radioactive material throughout the U.S. on a per-year basis, can be used for constructing probabilities of occurrence of human errors in handling, packaging, and storage of all radioactive materials by isotope and package type. Due to the absence of accident data where spent fuel and HLSW were involved, the probabilities calculated by Sandia are too general to be useful in specific scenarios such as failure to tighten bolts on a shipping cask. The numbers provided by Sandia are perhaps the only pertinent human reliability data in transportation available in the U.S.

A report entitled "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants,"

published in 1972 (Ref. 16), briefly addresses human errors in transportation. An estimate is given (without derivation) that 1 in 100,000 Type B packages (including spent fuel casks) is improperly closed, but no corresponding estimates of release fraction are given. This error rate is the same as the provided in the Sandia report (Ref. 14).

Activities at an Actual Reprocessing Plant - A complete task analysis was performed at the Savannah River Reprocessing Plant, Aiken, S.C., under the supervision of W.S. Durant. This plant reprocesses liquid waste and is involved with the caustic storage of fission products. Mr. Durant cannot share any numerical results concerning the frequency of human error in these operations without permission of the Department of Energy.

A more complete listing of sources used in addition to those presented above and in the reference list can be found in the bibliography in Appendix C.

2.2.5 Conclusions

Human factors accidents can result in higher expected dose to the population in the following cases:

- Errors in sealing transportation casks coupled with an impact accident
- Errors in design and maintenance of the air filtration system and storage pool circulation system resulting in a loss of cooling circulation accident and pool boiloff.

In spite of numerous personal conversations with specialists in government and private industry, numerical data such as

probability of occurrence or release fraction given the occurrence which are necessary to calculate impact upon expected dose were impossible to find. Further work at a level of effort far exceeding that programmed for this investigation is necessary to assess the impact of human factors in a quantitative manner.

3. CONSEQUENCE MODELS

3.1 AIR DISPERSION MODELS

Expected dose calculations for pre-sealing accidents involving high-level solidified waste are described in Reference 1. Two meteorology models were considered: the plume (continuous-source) model and the puff (instantaneous-source) model. The plume model was assumed when simulating the spread of airborne radionuclides from interim storage accidents at the reprocessing plant and from transportation fire accidents, since these accidents all result in prolonged releases of radioactivity. The puff model was chosen to best simulate the spread of airborne particulates from transportation impact accidents, since such releases are more correctly modeled as instantaneous than continuous in nature.

Recent communications between Lawrence Livermore Laboratory and TASC have called into question the validity of the computational procedure of Reference 9 for determining population-integrated concentration measures using the puff model. This technique had been employed in a draft version of Reference 1. The method used there involved integrating over all downwind distances the product of the puff's average concentration at each downwind distance and the population within the total ground area covered by the puff when it was centered at that downwind distance. This resulted in overlapping areas and hence multiple counting of the population in computing the population-integrated concentration measures. However, there was also an error of approximately eight orders

of magnitude (on the low side) in the normalized centerline concentration formulas shown in Table D-1 of Reference 9. On the basis of a document recently issued by LLL (Ref. 17), it appears that these two errors approximately cancel each other.

To summarize the corrections, the following changes should be made to Table D-1 of Reference 9:

- Change puff average-case χ/Q value from $3.8 \times 10^{-4} x^{-3.7}$ to $1.2 \times 10^{-4} x^{-3.7} \text{ m}^{-3}$
- Change puff worst-case χ/Q value from $4.1 \times 10^{-3} x^{-3.9}$ to $1.3 \times 10^{-5} x^{-3.9} \text{ m}^{-3}$.

The modification that eliminates "double" counting of population involves averaging the puff over an $\ell(x) \times \ell(x)$ "equivalent square" that is equal in area to the cloud at ground level at a distance x meters downwind from the source. The meteorology-demography integral then becomes

$$\int_{x=x_0}^{x_f} \frac{\chi(x)}{Q} \rho(x) t(x) dA(x) \quad (\text{man-sec/m}^3) \quad (3.1-1)$$

where

$\frac{\chi(x)}{Q}$ is the puff concentration per unit curie emitted averaged over an $\ell(x)$ by $\ell(x)$ area (m^{-3}).

$\rho(x)$ is the population density averaged over the width of the cloud at a distance x meters downwind of the source (persons/m^2).

$t(x)$ is the time of puff passage at distance x (sec). Note that $t(x) = \ell(x)/u$, where $\ell(x)$ is the length of the puff at ground level and u is the wind speed in m/sec .

$dA(x)$ is the differential area in m^2 and is equal to $\ell(x) dx$. This resolves the "double"-counting problem because populations within successive differential areas are now mutually exclusive of one another.

Two other minor changes were made:

- The rural population density of 2.2×10^{-5} persons/m² was added to the tail of the Dallas-Fort Worth Gaussian population density for urban accidents (Eq. D-7, Ref. 9)
- The first coefficient in Eq. D-9, Ref. 9 was changed from 3.8×10^{-7} to 1.8×10^{-7} .

3.2 AGGREGATION OF DOSES TO ORGANS

3.2.1 Introduction

In order to simplify output and to make results more meaningful, a method of computing net whole-body-equivalent doses has been developed. These doses are calculated by summing the doses to each individual organ weighted by the cancer risk. The method was suggested by Dr. Yook Ng of Lawrence Livermore Laboratory (Ref. 18) and is based on Reference 19. In this discussion the formulas and the weighting factors to be used are explicitly described. The method may be applied to both pre- and post-emplacement releases.

3.2.2 Equivalent Risk Model

The following equation is used to calculate net equivalent dose factors:

$$DF50(r,p,0) = \sum_{u=1}^8 DF50(r,p,u)R(u) \quad (3.2-1)$$

where:

DF50(r,p,u) is the fifty-year dose factor
for chronic exposure to organ
u from nuclide r via pathway p

DF50(r,p,0) is the net fifty-year dose equiva-
lent factor for chronic exposure
from nuclide r via pathway p

R(u) is the cancer risk factor for
organ u

The four pathways which could be considered are in-
gestion, inhalation, immersion, and external exposure. The
organs considered and the corresponding cancer risk factors
are given in Table 3.2-1.

TABLE 3.2-1
CANCER RISK FACTORS

u	ORGAN	R(u)
1	Whole Body	0.451
2	GI-LLI	0.158
3	Thyroid	0.034
4	Bone	0.034
5	Liver	0.014
6	Lung	0.223
7	Kidney	0.086
8	Skin	0.0

The risk factors are based on cancer risk coeffi-
cients given in Reference 20 and reproduced in Table 3.2-2.
The total risk to an adult is the sum over all cancers and
amounts to 5.83×10^{-6} deaths/yr-rem.* The risk factors in

*The cancer risk to an adult for leukemia was weighted by
5/6 to reflect its shorter plateau period.

Table 3.2-1 were calculated by normalizing the risk coefficients to an adult given in Table 3.2-2 by the total risk.

A number of difficulties arose because risk coefficients are not available for all the organs for which TASC computes doses. Furthermore, several cancers for which risk coefficients are available, notably leukemia, do not have computed dose coefficients.* These problems were addressed in the following way: risk coefficients for the thyroid and liver based upon data presented in Reference 20 were found given in References 11 and 21, respectively. The values given were 0.2×10^{-6} deaths/yr-rem for the liver. Since the cancer risk ascribed to the liver is included in Table 3.2-2 in the risk to the gastrointestinal tract, lower large intestine (GI-LLI), the value given for the GI-LLI was reduced to 0.92×10^{-6} deaths/yr-rem. A cancer risk for the kidney of 0.5×10^{-6} deaths/yr-rem was based upon an estimate given in Reference 19 and the cancer risk for the skin was assumed to be negligible based upon data from Reference 11.

The risks in Table 3.2-2 ascribed to leukemia and breast cancer were then aggregated into the classification "all other". This represents the residual cancer risk to the whole body and amounts to 2.63×10^{-6} deaths/yr-rem. Since dose factors were not available for these cancers they were multiplied by the whole-body dose factor in the calculation of net dose equivalent. This is a conservative assumption for most dose calculations since for the ingestion and inhalation pathways there is little exposure to the breast.

*It should be mentioned that a study of health effects would include cancer incidence and genetic effects as well as cancer fatalities.

TABLE 3.2-2
RISK COEFFICIENTS FOR LATENT CANCER FATALITIES

TYPE OF CANCER	AGE AT TIME OF IRRADIATION	LATENT PERIOD (years)	LATENT PERIOD (years)	RISK COEFFICIENT $\times 10^6$ (Deaths/yr-rem)
Leukemia	In utero	0	10	25
	0-9.5	2	25 ^(a)	2
Lung	10+	15	30 ^(a)	1.3
Gastrointestinal tract, including stomach	10+	15	30 ^(a)	1.0
Breast	10+	15	30 ^(a)	1.5 ^(b)
Bone	10+	15	30 ^(a)	0.2
All other ^(c)	In utero	0	10	25 ^(d)
	0-9.5	15	30 ^(a)	1 ^(d)
	10+	15	30 ^(a)	1 ^(e)

(a) Remaining life expectancy stated as an alternative plateau period.

(b) Includes males and an assumed 50% cure rate.

(c) Includes thyroid and skin.

(d) "All other" denotes all cancers except leukemia.

(e) "All other" denotes all cancers except those specified in table.

Source: Ref. 20

The result of this procedure is to provide a reasonable and convenient measure of the hazard to an individual or to a population. This provides a useful supplement to the calculations of doses to the whole body and individual organs.

3.2.3 Application of the Equivalent Risk Model

Figures 3.2-1 through 3.2-3 show examples of the potential hazard of waste from three different fuel cycles in terms of the whole-body-equivalent population dose per MWe-yr of waste entering a reference surface water system. The three fuel cycles considered are the plutonium-stowaway cycle which is described elsewhere in this report, the once-through cycle in which the waste is spent fuel, and the plutonium throwaway cycle in which uranium is recycled and plutonium is disposed of as part of the waste. The radionuclide inventories were calculated using the ORIGEN code and then used to determine

Figure 3.2-1 The Potential Hazard of HLW from the Plutonium-Stowaway Fuel Cycle Measured as Population Dose

Figure 3.2-2 The Potential Hazard of Spent Fuel Measured as Population Dose

Figure 3.2-3 The Potential Hazard of HLW from the Plutonium-Throwaway Fuel Cycle Measured as Population Dose

the potential hazard as described in Appendices G and H of Reference 1. All nuclides which contribute significantly to the potential hazard were included in the calculation. It should be noted that the above three graphs represent simplified examples of the application of the risk model. Comparisons between the graphs may be meaningful; absolute values shown on the graphs may not be representative of realistic situations.

4. CALCULATION OF DOSE TO THE "MAXIMUM INDIVIDUAL"

This chapter outlines the procedures used for computing radiological dose to the "maximum" individual from pre-emplacment accidents involving high-level solidified waste from the Pu-storage fuel cycle. The risk measure adopted here is that of expected dose to a worst-situated individual. This is, therefore, a measure of risk to the worst-situated individual averaged over all possible scenarios including both accident occurrence and non-occurrence.

An alternate approach could have been to compute per-accident doses to individuals exposed to radiation from "maximum-severity" accidents. This would have required a determination of how severe an accident would be appropriate in each case. Also, the use of a per-accident risk measure for each pre-emplacment accident would render comparison of risk among accident types and waste forms meaningless, since the probability of accident occurrence would not be considered.

Note that a substantial portion of this chapter's mathematical analysis is based on the work reported in Reference 1. This reference should be consulted as supplementary material.

4.1 DEFINITION OF THE "MAXIMUM INDIVIDUAL" AND OTHER ASSUMPTIONS

The maximum individual is assumed to behave as follows for each of the pre-emplacment accidents:

- He initially stands 100 m downwind of the accident site for one hour. This distance corresponds to the inner limit of calibration of the Gaussian diffusion air transport model and is considered a reasonable distance from which a spectator might view the accident. During this time he is exposed to direct gamma radiation from the canisters in certain accidents, as described in Section 4.2, and inhales contaminated air from all accidents.
- After an hour, he moves far enough away to eliminate gamma radiation from the accident site as a dominant factor. He continues to inhale contaminated air at this distance for a total of one day* and experiences a 50-yr dose commitment from this radiation. (Since inhalation makes only a small contribution compared to ingestion dose, this distance need not be more precisely defined.)
- He drinks contaminated milk obtained from cows grazing on contaminated grass at a distance 100 m downwind of the accident for a continuous period of 50 yr following the accident, and receives a dose from this radiation. He also consumes a "maximum" diet of water, meat, vegetables, and seafood that have been contaminated via waterborne radioactivity for a continuous period of 50 years following the accident, and receives a dose from this radiation.

In addition to those cited above, the following assumptions and those listed in Chapter 2 of Reference 1 apply to the calculation of maximum-individual dose from pre-employment accidents:

- No consideration is given to 50-year chronic external exposure to gamma radiation from the ground when computing dose to the maximum individual, since a single

*For accidents having instantaneous releases, such as truck and train impact accidents, the period of inhalation is the time of passage of the contaminated air mass.

individual is assumed to be able to evacuate the contaminated area after his initial exposure.

- Dose due to direct gamma radiation from the passing cloud is considered negligible compared to exposure via other pathways.

4.2 COMPONENTS OF THE MAXIMUM-INDIVIDUAL DOSE CALCULATION

Changes made to the Pre-emplacment Risk Calculations of Reference 1 - Expected dose to the maximum individual is computed in a manner identical to that depicted in the pre-emplacment risk flow chart (Fig. 2.1-1) of Reference 1.

There are several important changes within the components of this flow chart, however:

- The event tree is enlarged to include the scenario of loss of gamma shielding in severe truck and rail fire accidents (see discussion to follow in this section).
- The meteorology and demography models are altered as follows:

The average air concentration factors in the meteorology model are evaluated at 100 m directly downwind of the accident, instead of over an area

The only demography is that of the single individual

- The dose models are changed as follows:

The deposition dose due to chronic 50-year exposure to gamma radiation on the ground is removed, due to the assumption that the individual is free to move out of the contaminated area

The dose due to gamma radiation from a transportation cask that has lost its gamma shielding in a fire accident is added to the calculations.

Gamma Radiation from the Accident Site - In severe transportation fire accidents, loss of the cask's gamma shielding can occur, thus allowing gamma radiation to escape from the canisters even if the canisters remain unbreached and in the cask. (In all impact accidents, the casks's gamma shield is assumed to continue to protect against exposure of individuals to gamma radiation from the canisters.) Gamma radiation from the wreck, while unimportant when computing population dose from pre-emplacment accidents, can be significant when computing doses to the maximum individual at a distance of 100 m.

The important assumptions implicit in these calculations are:

- The point-source formula in air is valid for computing gamma dose at 100 m from a canister of waste, given a loss of cask gamma shielding. Attenuation of this radiation by the canister's stainless steel is assumed to cancel the additional contribution from ground-reflected radiation not accounted for by the point source formula.
- The fraction of canister gamma radiation emitted from the cask in the direction of the receptor is modeled as a ramp-step function of fire temperature for the truck and rail cask:

For the truck cask, the lead gamma shield melts at 327°C and the surrounding stainless steel softens at 1400°C (Ref. 1). The fraction of gamma radiation emitted from the cask, F_y , is assumed to rise linearly from 0 at the former temperature to 1 at the latter, and then to remain at 1 for all temperatures greater than 1400°C.

For the rail cask, the depleted-uranium gamma shield melts at 1100°C and the surrounding stainless steel softens at 1400°C (Ref. 1). The release fraction, F_y , again is assumed to rise linearly from 0 to 1 between these temperatures and then to level off at 1.

- The fire temperature probability density function (pdf) of Appendix A, Reference 1, is used.

The following equations are used for computing expected gamma-radiation dose to the maximum individual from exposure to unshielded canisters in transportation fire accidents:

$$D(R) = \phi_1 \phi_2 D'(R) T \quad (4.2-1)$$

$$\phi_1 = P_1 P_2 P_3 P_4 / N_C N \quad (4.2-2)$$

$$\phi_2 = N_C \phi_y N \quad (4.2-3)$$

$$\phi_y = \int_0^{\infty} p(T) F_y(T) dT \quad (4.2-4)$$

where:

$D(R)$ = expected gamma-radiation dose to an individual from unshielded canisters in transportation fire accidents located R meters away. (Units are rem per MWe-yr of waste produced.)

ϕ_1 = probability of a truck or train urban or rural fire accident per MWe-yr of waste produced.

ϕ_2 = expected amount (MWe-yr) of waste emitting gamma radiation due to a loss of cask gamma shielding, given a truck or train fire accident.

ϕ_y = expected gamma-radiation transmission factor given exposure to cask gamma shielding loss (dimensionless).

For the rail cask, the depleted-uranium gamma shield melts at 1100°C and the surrounding stainless steel softens at 1400°C (Ref. 1). The release fraction, F_Y , again is assumed to rise linearly from 0 to 1 between these temperatures and then to level off at 1.

- The fire temperature probability density function (pdf) of Appendix A, Reference 1, is used.

The following equations are used for computing expected gamma-radiation dose to the maximum individual from exposure to unshielded canisters in transportation fire accidents:

$$D(R) = \phi_1 \phi_2 D'(R) T \quad (4.2-1)$$

$$\phi_1 = P_1 P_2 P_3 P_4 / N_C N \quad (4.2-2)$$

$$\phi_2 = N_C \phi_Y N \quad (4.2-3)$$

$$\phi_Y = \int_0^{\infty} p(T) F_Y(T) dt \quad (4.2-4)$$

where:

$D(R)$ = expected gamma-radiation dose to an individual from unshielded canisters in transportation fire accidents located R meters away. (Units are rem per MWe-yr of waste produced.)

ϕ_1 = probability of a truck or train urban or rural fire accident per MWe-yr of waste produced.

ϕ_2 = expected amount (MWe-yr) of waste emitting gamma radiation due to a loss of cask gamma shielding, given a truck or train fire accident.

ϕ_Y = expected gamma-radiation transmission factor given exposure to cask gamma shielding loss (dimensionless).

$D'(R)$ = the unattenuated gamma-radiation dose rate to the maximum individual from bare canisters located at a distance R meters away from the maximum individual (rem/hr per MWe-yr of unshielded waste).

T = the maximum individual's exposure time (hr).

P_1 = probability per mile of a train or truck accident of any kind.

P_2 = probability of a fire accident, given an accident.

P_3 = probability that the fire accident occurs in an urban (or rural) location.

P_4 = number of miles traveled per truck or rail car shipment.

N_C = number of canisters carried per truck or rail car (also the number of canisters assumed to be exposed to shielding loss per accident).

N = number of MWe-yr of waste per canister.

$p(T)$ = probability density function of fire temperature ($1/^\circ F$).

$F_Y(T)$ = fraction of canister gamma radiation emitted from cask due to loss of shielding at a given fire temperature T (dimensionless).

The value of $D'(R)$ is taken to be 0.14 rem/hr-MWe-yr and the value of T to be 1 hr. $F_Y(T)$ is defined according to the assumptions listed earlier in this section, while values for P_1 , P_2 , P_3 , P_4 , N_C , N , and $p(T)$ may be found in Reference 1. The resulting expected gamma-radiation doses are approximately 10^{-6} rem/MWe-yr for truck fire accidents and 10^{-7} rem/MWe-yr for train fire accidents at a distance of 100 m. These gamma-dose figures are preliminary and should be used with caution.

5.

SUMMARY

The analyses described in this report have extended previous work (Ref. 1) in a number of directions. For each topic addressed, the approach used and the effects on the results of that work are summarized here.

The pool drainage accident was modeled in two dimensions, in order to improve on the previous one-dimensional lumped parameter approach. The temperatures computed by this method did not differ appreciably from those calculated by the one-dimensional approach, thus confirming this aspect of the previous analysis.

Human factors considerations were found to have the potential to affect previous risk estimates in two areas:

- Errors in sealing transportation casks coupled with an impact accident
- Errors in design and maintenance of air filtration and water circulation systems at an interim storage facility.

Insufficient data were available for quantitative assessment of the contribution of these possibilities to expected values of doses.

Revision of the puff model analysis outlined in Reference 9 resulted in a computationally consistent algorithm for evaluating population doses from instantaneous releases to the air. This revised technique was employed in Reference 1 for computing expected doses from train and truck impact accidents.

Use of the net equivalent dose concept of Section 3.2 would change the specific numerical values presented in Reference 1, but would not affect any of the qualitative conclusions.

Finally, the calculation of expected values of doses to a maximum individual extends the work of Reference 1.

These analyses will make possible more complete and more accurate risk calculations and a more comprehensive consideration of relevant factors in any future study of pre-sealing hazards from high-level radioactive wastes.

APPENDIX A
CONVECTIVE HEAT TRANSFER FROM THE CANISTER WALL

A.1 DESCRIPTION OF PROBLEM

The primary mode of heat transfer from HLSW canisters in an interim storage loss-of-coolant accident is convection to the surrounding air. The scenario for air motion adopted in previous studies first by ARHCO (Ref. 2) and later by TASC (Ref. 1) consists of the following elements:

- Air moves rapidly downward along the sides of the interim storage basin
- At the bottom of the canister array near the concrete basin floor, air is at a temperature of 120°F
- Air is heated by the canister walls and rises due to buoyancy effects; the velocity of air determines the magnitude of the convective heat transfer coefficient between the air and the canister wall
- The heated air forms a thermal plume that leaves the storage basin area, i.e., it does not recirculate.

For the current two-dimensional problem, heat transfer along the length of the canister is of interest. To describe the convective process adequately requires knowledge of the canister wall temperatures, the air temperatures, and the appropriate heat transfer coefficients as a function of axial position. The HEATING5 computer code cannot explicitly handle the type of heat transfer problem that involves mass transfer (i.e., air flow) into and out of the boundaries of the problem space. Code material regions are fixed in space; their motion is not

allowed. However, by treating convective heat transfer simply as a boundary condition to the HLSW canisters without having to represent the air space between canisters as a separate material region, an effective method has been found to model the convective process. The purpose of this Appendix is to describe the approach used to compute the needed convection boundary condition parameters.

A.2 HEAT TRANSFER MODEL

Consider a unit cell in the canister array consisting of a waste canister surrounded by air flowing upward at a constant flow rate. The following assumptions have been invoked:

- Steady-state (time-independent) conditions prevail
- The convective heat transfer can be described as a forced convection process
- The air flow is sufficiently turbulent that complete radial mixing is promoted, i.e., there are no radial temperature gradients in the air stream.

At a small section of the canister (see Fig. A-1) at height z whose width is dz , the rate of heat transfer by convection to air, dq_c , is

$$dq_c = 2\pi r h_c (T_w(z) - T_a(z)) dz \text{ (Btu/hr)} \quad (\text{A-1})$$

where

r = outer radius of the canister (ft)

h_c = forced convection heat transfer coefficient
(Btu/hr-ft²-°F)

Figure A-1 Sketch of Convective Heat Transfer Process

T_w = canister wall temperatures ($^{\circ}\text{F}$)

T_a = air temperature ($^{\circ}\text{F}$)

The air, in flowing across this section, will experience a temperature rise given by

$$dq_c = \dot{m} C dT_a \quad (\text{Btu/hr}) \quad (\text{A-2})$$

with $\dot{m}$ = mass flow rate of the air per canister (lb/hr)

C = heat capacity of the air (Btu/lb- $^{\circ}\text{F}$)

Equating these expressions then yields the air heat balance equation

$$\frac{dT_a}{dz} = \frac{2\pi r h_c}{\dot{m} C} (T_w(z) - T_a(z)) \quad (A-3)$$

Associated with the differential equation Eq. A-3 is the air temperature at the bottom of the basin

$$T_a(0) = 120^\circ\text{F} \quad (A-4)$$

A.3 TREATMENT OF CONVECTION IN HEATING5

To treat the convection process as a boundary condition on the canister it is necessary, at each zone along the canister length, to input the following parameters:

- A heat transfer coefficient, h_c
- The temperature of the heat sink.

Given this information, the code provides the canister surface temperature and the area over which the heat transfer takes place to complete the heat transfer calculation.

For simplicity, it was decided to instruct HEATING5 to transfer heat by convection to a heat sink at a constant temperature, T_o , instead of at T_a , and to adjust the heat transfer coefficient accordingly (T_o was arbitrarily selected to be 75°F). Computation of an "effective" heat transfer coefficient was accomplished by a subroutine written specifically for this problem. The major steps in the subroutine are described in the following paragraphs.

At each axial zone the heat transfer process by convection is given by Eq. A-1, which can be rewritten using a heat transfer area, A, as

$$q_c = Ah_c(T_w - T_a) \quad (A-5)$$

To base the heat transfer on a sink temperature, T_o , in lieu of the actual air temperature, T_a , requires the use of an adjusted convection coefficient, h' , found from

$$h' = h_c \frac{T_w - T_a}{T_w - T_o} \quad (A-6)$$

The special subroutine determined the air temperature, T_a , in Eq. A-6 by solving Eq. A-3 using an iterative process. The canister was partitioned axially into N zones of equal length Δz . As indicated in Fig. 2.1-2, 40 zones were employed. The position of the k^{th} zone is

$$z_k = k\Delta z; k = 1, 2, \dots, N \quad (A-7)$$

The derivative of the left-hand side of Eq. A-1 was approximated by a backward difference of the form

$$\frac{dT_a}{dz}(z_{k+1}) = \frac{T_a(z_{k+1}) - T_a(z_k)}{\Delta z} \quad (A-8)$$

Using Eq. A-8 in Eq. A-3 gives

$$T_a(z_{k+1}) = \frac{2\pi r h_c \Delta z T_w(z_{k+1}) + \dot{m} C T_a(z_k)}{2\pi r h_c \Delta z + \dot{m} C} \quad (A-9)$$

which expresses the air temperature at zone $k+1$ in terms of the air temperature of zone k and the wall temperature at zone $k+1$. Substitution of this result into Eq. A-6 then provides the effective coefficient, h' , at zone $k+1$. Note that h_c is not necessarily a constant but may be input as a function of air temperature (see Eq. 2.1-1).

APPENDIX B

RADIATION VIEW FACTORS

B.1 DESCRIPTION OF PROBLEM

An important heat transfer mechanism from HLSW canisters in a loss-of-coolant accident is radiation to space from the hot canister walls. In the prior one-dimensional studies and in the current two-dimensional work, heat loss by radiation from the canister ends is easily computed. The view factors (i.e., the fraction of radiation emitted to space) were essentially unity. Radiation from the canister sides, on the other hand, is much more complicated to determine. The geometry consists of a square array of circular cylinders. The view factors depend on z , the coordinate in the axial direction -- the closer to the canister top, the larger the view factor to space. The problem addressed in this appendix is that of determining, approximately, the radiation view factors from the sides of waste canisters to space as a function of distance along the length of the canister.

B.2 DETERMINATION OF VIEW FACTORS

A typical pathway for radiation from a canister side wall to space was illustrated in Fig. 2.1-1. If one draws a square array of cylinders it is apparent that the solid angle subtended by all other cylinders from a point on the wall of a given canister is difficult to calculate. The approach taken here was to assume that the net effect of all surrounding cylinders, due to intercepting radiation from the sides of a given

cylinder, could be represented by enclosing the latter by a concentric cylinder of larger radius.

The simplified geometry is shown in Fig. B-1. If this representation is valid, the problem is reduced to computing the view factor from a region on the inner cylinder, indicated as Region 2, to space, indicated as a planar Region 7. Radiation that intercepts the sides of the outer cylinder -- i.e., Regions 4, 5, and 6 -- and the concrete bottom of the pool -- i.e., Region 10 -- is ignored. Selection of an appropriate radius, R , for the outer cylinder is important and will be discussed later; the radius of the inner cylinder, r , and the cylinder length, L , are the actual physical dimensions of the waste canisters. The regions, such as Region 2, for which the view factors are to be determined have been chosen to be horizontal rings so as to coincide with acceptable regions for the HEATING5 code in an R-z geometry (that is, regions whose boundaries are parallel to curves of constant R and z).

Following the nomenclature from Holman (Ref. 6) we define $F_{a,b,c-x,y,z}$ as the view factor, or shape factor, from a surface consisting of subsurfaces a , b , and c to another surface consisting of subsurfaces x , y , and z .

Consider again Fig. B-1; it is desired to calculate the view factor F_{2-7} , which can be accomplished by use of reciprocity and conservation of energy relationships.

From Surface 1 we have

$$F_{1-7} + F_{1-4} + F_{1-8} = 1 \quad (B-1)$$

By symmetry

$$F_{1-7} = F_{1-8} \quad (B-2)$$

REGIONS 1, 2 AND 3 ARE ON INNER VERTICAL CYLINDER
 REGIONS 4, 5 AND 6 ARE ON OUTER VERTICAL CYLINDER
 REGIONS 7, 8, 9 AND 10 ARE HORIZONTAL PLANES

Figure B-1 Geometry for Determining Radiation View Factors to Space from Regions on the Inner of a Pair of Concentric Cylinders

Therefore,

$$F_{1-7} = 1/2 \left[1 - F_{1-4} \right] \quad (B-3)$$

By a similar argument for composite surface 1, 2

$$F_{1,2-7} + F_{1,2-4,5} + F_{1,2-9} = 1 \quad (B-4)$$

and

$$F_{1,2-7} = F_{1,2-9} \quad (B-5)$$

Therefore,

$$F_{1,2-7} = 1/2 \left[1 - F_{1,2-4,5} \right] \quad (B-6)$$

The total radiation arriving at surface 7 from the combined area $A_{1,2}$ is the sum of the radiations from surfaces 1 and 2. Thus,

$$A_{1,2} F_{1,2-7} = A_1 F_{1-7} + A_2 F_{2-7} \quad (B-7)$$

Next, Eqs. B-3 and B-6 were substituted into Eq. B-7, using

$$A_{1,2} = A_1 + A_2 \quad (B-8)$$

to obtain, after some algebra,

$$F_{2-7} = 1/2 \left[1 + \frac{A_1}{A_2} F_{1-4} - \left(1 + \frac{A_1}{A_2} \right) F_{1,2-4,5} \right] \quad (B-9)$$

In the limit of thin regions where $A_2 \rightarrow 0$,

$$A_2 F_{2-7} = \frac{A_1}{2} \left[F_{1-4} - F_{1,2-4,5} \right] \quad (B-10)$$

The quantities F_{1-4} and $F_{1,2-4,5}$ appearing in Eq. B-9 are simply the view factors between two concentric, coaxial cylinders of equal length. Hottel (Ref. 22) gives that view factor relationship from the surface of the outer cylinder to the inner cylinder as

$$\begin{aligned}
F_{o-i} = & \frac{r}{R} - \frac{1}{\pi} \frac{r}{R} \cos^{-1} \left[\frac{(L/R)^2 + (r/R)^2 - 1}{(L/R)^2 - (r/R)^2 + 1} \right] \\
& + \left[\left(\frac{1}{2\pi} \left(\frac{L}{R} + \frac{R}{L} + \frac{r^2}{RL} \right)^2 - \left(\frac{2r}{L} \right)^2 \right)^{1/2} \cos^{-1} \left[\frac{(L/R)^2 + (r/R)^2 - 1}{(L/R)^2 - (r/R)^2 + 1} \frac{r}{R} \right] \right. \\
& \left. + \frac{1}{2\pi} \left(\frac{L}{R} - \frac{R}{L} + \frac{r^2}{RL} \right) \sin^{-1} \frac{r}{R} - \frac{1}{4} \left(\frac{L}{R} + \frac{R}{L} - \frac{r^2}{RL} \right) \right] \quad (B-11)
\end{aligned}$$

The corresponding relationship for the view factor from the inner to the outer cylinder is

$$F_{i-o} = \frac{R}{r} F_{o-i} \quad (B-12)$$

In terms of the present problem (see Fig. B-1) the ratio A_1/A_2 is given by

$$A_1/A_2 = z_1/(z_2 - z_1) \quad (B-13)$$

and F_{1-4} is evaluated at $L = z_1$, while $F_{1,2-4,5}$ is evaluated at $L = z_2$ using Eq. B-11 and Eq. B-12.

B.3 NUMERICAL RESULTS OF VIEW FACTOR CALCULATIONS

For calculation of radiation heat transfer in HEATING5 from canister sides the 20-ft long canisters were divided axially into 10 2-ft long zones. The radius r of the inner cylinder for each zone was 0.5 ft. The radius, R , of the outer cylinder was assumed to be 1.5 ft. This distance was selected because it represents exactly the canister center-to-center

spacing and, roughly, it should correspond to an average distance from the center of the inner cylinder to the outer walls of the surrounding cylinders (the actual range of these distances varies from 1 ft to about 3 ft). Table B-1 gives the computed view factors F_{2-7} with a 1.5-ft canister spacing for each of the 10 zones; note that distances are measured from the top of the canister.

In order to check whether the computed view factors were sensitive to the somewhat arbitrary choice of outer radius, F_{2-7} was recalculated for reasonable lower and upper bounds on R and compared to the nominal ($R=1.5$ ft) view factor.

TABLE B-1
RADIATION VIEW FACTORS FROM CANISTER SIDES

ZONE LOCATION (ft)	VIEW FACTOR F_{2-7}
0-2	0.3426
2-4	0.0735
4-6	0.0272
6-8	0.0140
8-10	0.0085
10-12	0.0057
12-14	0.0040
14-16	0.0030
16-18	0.0024
18-20	0.0019

These results are presented in Table B-2 for two typical zones located at 2 to 4 ft and 16 to 18 ft from the top of the canisters.

TABLE B-2
SENSITIVITY OF COMPUTED VIEW FACTORS TO
SELECTION OF OUTER RADIUS, R

OUTER RADIUS (ft)	F ₂₋₇ AT	F ₂₋₇ AT
	ZONE 2 TO 4 ft	ZONE 16 TO 18 ft
1.0 (lower bound)	0.058	0.0018
1.5 (nominal)	0.073	0.0024
2.0 (upper bound)	0.076	0.0025

As can be seen, the sensitivity of the view factor calculation on the assumed equivalent outer radius is not large, giving greater confidence to the results.

APPENDIX C
SOURCES OF HUMAN FACTORS INFORMATION

C.1 A LIST OF INDIVIDUALS CONTACTED

C.1.1 Human Factors Specialists

- Dr. Alan Swain, Sandia Laboratories,
Albuquerque, N.M
- Dr. Brian Finley, Sandia Laboratories,
Albuquerque, N.M.
- Dr. Richard Pew, Bolt, Beranek and Newman,
Inc., Cambridge, MA
- Dr. Thomas Sheridan, MIT, Cambridge, MA
- Dr. William Duffey, University of Massachusetts,
Amherst, MA
- Dr. Paul Ronco, Man-Machine, Inc., Reading,
MA
- Mr. Gordon Chipman, Mr. Matthew Taylor,
Mr. Joseph Murphy, Mr. William Vesely
and Mr. Michael Cullingford, all of NRC
Accident and Analysis Branch, Bethesda, MD
- Mr. Harold Jacobs, Dunlap and Associates,
Lexington, MA (private consultants)
- Dr. David Meister, University of California
San Diego, San Diego, CA.

C.1.2 Human Factors at the Reprocessing Plant

- Mr. W.S. Durant, Savannah River Reprocessing
Plant, Aiken, SC
- Mr. George Kniazewycz, TERA Corporation,
Berkeley, CA

- Mr. Thomas Shea, Materials Handling Institute, Pittsburgh, PA
- Mr. Keith Hatsell, Sheppard-Niles Crane and Hoist Company, Watkins Glen, NY.

C.1.3 Human Factors in Air Filtration Systems

- Dr. Melvin First, Harvard School of Public Health, Boston, MA
- Mr. Clifford Burchsted, ORNL, Oak Ridge, Tenn.
- Mr. Humphrey Gilbert, private consultant on HEPA filters, McLean, VA
- Mr. John Jacox, Nuclear Consulting Services, Inc., Columbus, Ohio. (HEPA and charcoal filters)
- Mr. Frederick Lackie, Nuclear Containment Systems, Inc., Columbus, Ohio. (HEPA and charcoal filters)
- Mr. Donald Rhodes, Idaho National Engineering Laboratory, Idaho Falls, Idaho. (silica-gel filters)
- Mr. Dade Moeller, Harvard School of Public Health, Boston, MA
- Mr. Ronald Bellamy and Mr. John Donohew, NRC Effluent Treatment Systems Branch, Bethesda, MD
- Mr. A. Gilman, Nuclear Metals, Inc., Concord, MA.

C.1.4 Human Factors in Transportation

- Dr. Brian Finley, Sandia Laboratories, Albuquerque, NM
- Mr. Arthur Carson and Mr. E.E. Boyland, General Electric Corporation, San Jose, CA (IF-300 spent fuel shipping casks)

THE ANALYTIC SCIENCES CORPORATION

- Mr. Earl Rutenkroger, Tri-State Company, Joplin, MO (dedicated shippers of radioactive material)
- Mr. H. Sonnenberg, DOT Hazardous Materials Division, Washington, D.C.
- Mr. Richard Kaplan, Highway Safety Research at the University of Michigan, Ann Arbor, MI
- Mr. John Treat, Institute for Public Safety at Indiana University, Bloomington, IN
- Mr. Norman Eisenberg, NRC Transportation and Products Standards Division, Bethesda, MD
- Mr. Donald Williams, Battelle Pacific Northwest Laboratories, Richland, WA
- Mr. Ralph Fullwood, Science Applications, Inc., Palo Alto, CA

C.1.5 Human Factors at the Repository

- Mr. David Pentz, Golder Associates, Kirkland, WA
- Mr. George Purifoy, Applied Science Associates, Inc., Valencia, PA (formerly of the American Institute of Research, Pittsburgh, PA)
- Mr. Donald Rogich, Mine Systems Engineering, Bureau of Mines, Washington, D.C.
- Mr. Arthur Nelson, MESA Metal and Nonmetal Mine Health and Safety, Arlington, VA
- Mr. Donald Walker, MESA Health and Safety Analysis Center, Denver, CO
- Mr. William Wiheagan, Bureau of Mines, Pittsburgh, PA
- Mr. C. Stevens, Science Applications, Inc., Palo Alto, CA

- Dr. James Altman, Synectics Company, Pittsburgh, PA (formerly of the American Institute for Research, Pittsburgh, PA).

C.1.5 General

- Mr. Joel Buchanan, Mr. R. Scott, and Mr. M.L. Winton, Nuclear Safety Information Center, ORNL, Oak Ridge, Tenn.
- Mr. Lames Leva, New Jersey Central Power and Light Company, Morristown, NJ (simulated nuclear leaks at power generating stations)
- Mr. A. Garcia, Science Applications, Inc., Palo Alto, CA
- Mr. Joel Kramer, Bureau of Standards, Bethesda, MD
- Mr. William Dickey, Idaho National Engineering Laboratory, Idaho Fall, Idaho
- Mr. Stuart Parsons, Lockheed Missiles and Space Company, Sunnyvale, CA (task analysis at reactors done for EPFL)
- Mr. W. Michael Diekman, American Nuclear Society, Lagrange Park, Illinois.

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