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DETERMINATION OF PERFORMANCE CRITERIA FOR HIGH-LEVEL SOLIDIFIED NUCLEAR WASTE

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ABSTRACT AND FOREWORD

The Nuclear Regulatory Commission (NRC) is responsible for the management and disposal of nuclear wastes from licensed facilities. The agency also has licensing responsibility for long-term storage and disposal of high-level radioactive wastes from commercial operations and the Department of Energy.

In discharging its responsibilities, the NRC is developing a framework of regulations, criteria, and standards. Lawrence Livermore Laboratory (LLL) provides broad technical support to the NRC for developing this regulatory framework, part of which involves the determination of performance criteria for high-level solidified nuclear waste.

Both the regulatory framework and the technical base on which it rests, have evolved in time. This document is the briefing document on performance criteria, as required by the 1977 Statement of Work between LLL and the NRC. It is being reprinted now in its original format as a UCID-series report for the sake of the historical record.

The LLL program has evolved rapidly and extensively. In addition to this expansion, a major shift in program direction was mandated during FY78 as a result of the Presidential decision (April, 1977) to defer the reprocessing of spent reactor fuel. Emphasis was shifted from solidified high-level waste disposal, to the study of spent-fuel disposal and retrieval in the environment of a deep geologic repository. For current (1979) information, the reader is referred to the most recent publications of the LLL technical support project, in particular, "A Probabilistic Safety Analysis for Solidified High-Level Nuclear Waste Management Systems: A Status Report" (UCRL 52532, NUREG/CR-0577).

PRELIMINARY DRAFT

DETERMINATION OF PERFORMANCE CRITERIA FOR HIGH-LEVEL SOLIDIFIED NUCLEAR WASTE

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1. INTRODUCTION
 - 1.1 Fuel Cycle
 - 1.2 System Analysis Approach
 - 1.3 Probabilistic Risk Assessment
2. SUMMARY
3. INTERIM STORAGE AT FUEL PROCESSING PLANT
 - 3.1 Handling Events
 - 3.2 Storage Pool Events
4. TRANSPORTATION
 - 4.1 Baseline Shipping Cask Configuration
 - 4.2 Railroad Events
 - 4.3 Truck Events
5. HANDLING AT REPOSITORY
 - 5.1 Handling Events
6. POST-SEALING REPOSITORY HISTORY
 - 6.1 Reference Repository: Geologic Setting
 - 6.2 Reference Repository: Hydrologic Setting
 - 6.3 Repository Site: Climatologic Setting
 - 6.4 Future History: Water Intrusion Events
7. RADIONUCLIDE TRANSPORT SYSTEMS AND DEMOGRAPHIC FACTORS
 - 7.1 Human Water Use Modeling
 - 7.2 Airborne Dispersion Modeling
 - 7.3 Demographic Factors

APPENDIXES

I. INTRODUCTION

1.1 Fuel Cycle

DEVELOPMENT OF REGULATIONS FOR PERFORMANCE CRITERIA FOR SOLIDIFIED HIGH LEVEL NUCLEAR WASTE IS NECESSARY TO CLOSE THE "BACK END" OF THE FUEL CYCLE

THE DEVELOPMENT OF A WASTE MANAGEMENT STRATEGY TO SAFELY MANAGE AND DISPOSE OF NUCLEAR WASTES FROM THE COMMERCIAL LIGHT WATER NUCLEAR POWER INDUSTRY IS REGARDED AS ONE OF THE PIVOTAL ISSUES AFFECTING ACCEPTANCE OF NUCLEAR POWER. THESE HIGH-LEVEL WASTES ARE GENERATED IN THE FUEL REPROCESSING PLANTS AT THE "BACK END" OF THE COMMERCIAL LIGHT WATER REACTOR NUCLEAR FUEL CYCLE.

The safe management of nuclear waste is clearly one of the pivotal issues facing the future role of power generated by the light water nuclear system. The public has apprehensions and misconceptions about the danger and difficulty involved in safely managing nuclear wastes. In addition, the federal government and the nuclear industry have failed to produce and demonstrate viable and effective waste management strategies. The quantities of and hazards associated with existing nuclear wastes (mostly military) are significant. On the other hand potential hazards attributed to radioactive waste do not differ significantly from common chemical toxin hazards, e.g., lead and chromium natural ores.

The operation of the commercial nuclear fuel cycle for the generation of electric power produces radioactive wastes. The "back end" of this fuel cycle includes all activities after removal of spent fuel rods from the reactor. From the facing figure we see that spent fuel is taken from storage, sent to a processing plant where unused uranium and plutonium produced in the reactor are recovered. High-level radioactive waste is generated at the processing plant and must be solidified, stored to allow decay of radioactive heat, and finally transported to a repository site for final

disposition.

Only the spent fuel processing portion of nuclear fuel cycle generates high level waste (HLW). The current definition of HLW is given in Appendix F, 10 CFR Part 50, as an aqueous waste from the operation of the first-cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent in a facility for reprocessing spent reactor fuels. The current regulations, Appendix F 10 CFR Part 50, require solidification of all high level wastes within 5 years of their generation and transfer to a Federal repository with 10 years.

Closing the "back end" of the commercial light water nuclear fuel cycle would provide for operation of the spent fuel processing plants; recycling of uranium and plutonium by recovery and subsequent fabrication into reactor fuel rods; solidifying and storing high level wastes; transporting these wastes to the repository site; and finally, permanently disposing of the high level wastes into deep geologic rock formations. The development of regulations to specify the performance criteria for an acceptable solidified waste form for the high level nuclear waste becomes one of the initial steps in closing the "back-end" of the fuel cycle.

SPENT FUEL

PROCESSING PLANT

PROCESSED WASTE

INTERIM STORAGE

HIGH-LEVEL
WASTE

REPOSITORY SITE

1 INTRODUCTION

1.2 System Analysis Approach

A SYSTEM ANALYSIS APPROACH IS AN EFFICIENT WAY TO DEVELOP REGULATIONS FOR SOLIDIFIED HIGH LEVEL WASTE

GIVEN THE PROCESS FLOW SHEET AND THE VARIOUS OPERATIONS ASSOCIATED WITH THE SOLIDIFICATION OF HIGH LEVEL NUCLEAR WASTES, THE REGULATOR MUST DECIDE WHAT ARE THE CRITICAL PERFORMANCE CHARACTERISTICS TO BE DEMANDED OF AN ALLOWABLE SOLID WASTE FORM. A SYSTEM ANALYSIS APPROACH PROVIDES AN EFFICIENT DECISION MECHANISM TO SOLVE THIS PROBLEM. THE SAME METHODOLOGY AND DATA BASE USED IN THE REGULATION DEVELOPMENT PHASE CAN BE EXTENDED THROUGH THE REGULATORY GUIDE PHASE AND INTO THE LICENSING AND ACCEPTANCE PHASE.

The high level waste management system has been defined as processing the solidified waste form through the interim storage operation; transportation to the repository site; handling at the repository; and finally the future history after sealing the repository. This system is shown diagrammatically in the opposite figure.

The regulator is faced with the problem of specifying the performance characteristics of a solidified waste form that will produce the minimum risk in such a waste management system.

An efficient solution of this problem is achieved through the use of a system analysis approach. Development of a computer simulation model allows use of a methodology that can rapidly compare the expectation of the overall risk to the waste form release parameters. Assembly of a data base of characteristics and release parameters of candidate waste forms allows a comparison of the risk

expectation over all reasonable performance criteria specification.

Thus, the computer simulation model may be used to determine the risk inherent in a particular solid waste form from the operation of the waste management system. We show this diagrammatically as Waste form A---N, as inputs to the waste management system in the figure opposite.

Using the methodology developed for specifying the waste form performance criteria, the regulatory guide preparation phase becomes a simple extension. However, an acceptance criteria methodology must be integrated into the waste form characteristics parameters to complete the guide phase. From both the data base and computer simulation methodology, the licensing and acceptance phase readily follows the first two steps. Thus a system analysis approach becomes an efficient tool for supplying the regulator his needs for all phases of the regulatory process.

Nuclear Fuel Cycle Operation
spent fuel reprocessing

Solidification
High Level Waste

Waste Form A

Waste Form B

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Waste Form N

WASTE MANAGEMENT SYSTEM

Interim
Storage

Transportation

Handling and
Emplacement

Post-Sealing
History

1. INTRODUCTION

1.3 Probabilistic Risk Assessment

PROBABILISTIC RISK ASSESSMENT TECHNIQUES ALLOW A QUANTITATIVE EVALUATION OF THE SUMMATION OF RISKS FROM VARIOUS OPERATIONS IN THE WASTE MANAGEMENT SYSTEM.

A SYSTEMS ANALYSIS MODEL WAS DEVELOPED WHICH DETERMINES THE RISK ASSOCIATED WITH THE OPERATION OF A SOLIDIFIED HIGH LEVEL WASTE DISPOSAL SYSTEM. FAILURE MODES, THEIR PROBABILITY OF OCCURRENCE, AND THE SEVERITY OF RESULTANT CONSEQUENCES MUST BE EVALUATED FOR ALL SYSTEM OPERATIONS. EVENT-TREE METHODOLOGY WAS USED TO DETERMINE POTENTIAL RISK MECHANISMS. THE MODEL, AS DEVELOPED TO DATE, ADDRESSES ONLY THE TECHNICAL ISSUES AND DOES NOT ATTEMPT TO ADDRESS THE ENVIRONMENTAL/SOCIETAL ISSUES ASSOCIATED WITH THE RISKS.

A systems analysis model was developed which considers releases for both routine as well as potential accident environments to which high level waste solid matrices could be exposed during interim storage, transportation, handling, emplacement in a deep geologic media, and future history after sealing of the repository.

The facing figure provides an overview of the functional operations of the waste management system.

Functional event trees were constructed for each waste management operation to identify potential release mechanisms, i.e., failure modes. Detailed operational event trees will be discussed later. Published data for failure probabilities were used whenever available. Release functions were developed to quantitatively evaluate the severity of various failure consequences expressed as an expected value of risk. Releases were normalized to Ci/MWe-yr. Individual and population radiological exposure

doses were calculated on an expected value basis, also normalized to per MWe-yr. The expected value calculations provide a means of identifying the potential "pinch points" for setting criteria. Although they provide information on integrated risks (expected value) they do not indicate the consequences of individual contributing events. The recommended values for the solid matrix performance criteria will consider the reduction of both the integrated risks and single event risks to the public to "as low as reasonably achievable" levels.

The system analysis model, as developed to date, treats only the technological issues involved in nuclear waste management and stops after defining the intermediate physical consequences from environmental modeling. Treatment of the consequences of the radiological risks within the societal, ecological or economic regime is not addressed in this portion of the study.

2. SUMMARY

SPECIFYING LIMITS ON VOLATILIZATION, PARTICULATE DISPERSION, AND DISSOLUTION CHARACTERISTICS OF SOLIDIFIED HIGH LEVEL WASTE MINIMIZES RISK FROM OPERATION OF A WASTE MANAGEMENT SYSTEM.

TO MINIMIZE RADIOLOGICAL RISK FROM THE OPERATION OF A WASTE MANAGEMENT SYSTEM, PERFORMANCE LIMITS ON VOLATILIZATION, PARTICULATE DISPERSION, AND DISSOLUTION CHARACTERISTICS OF SOLIDIFIED HIGH LEVEL WASTE MUST BE SPECIFIED. THE RESULTS SHOW CLEARLY THAT THE PRE-EMPLACEMENT ENVIRONS ARE MORE LIMITING IN ESTABLISHING THE WASTE FORM PERFORMANCE CRITERIA THAN THE POST-EMPLACEMENT ENVIRONS. ABSOLUTE VALUES OF EXPECTED RISK ARE VERY SENSITIVE TO MODELING ASSUMPTIONS. THE TRANSPORTATION AND INTERIM STORAGE OPERATIONS APPEAR TO BE MOST LIMITING IN DETERMINING THE PERFORMANCE CHARACTERISTICS REQUIRED. THE EXPECTED VALUES OF RISK DO NOT RELY UPON THE REPOSITORIES REMAINING INTACT OVER THE POTENTIALLY HAZARDOUS LIFETIME OF THE WASTE.

A summary of the expected value of risk, normalized to man-REM/MWe-yr, is given in the table opposite. We have calculated the risk associated with each waste management operation as a function of waste type. We have considered radiation doses due to external irradiation as well as internal uptake of radionuclides through various food and water pathways. The table shows only whole body dose, not critical organ doses. Thus, the contribution from the particulate dispersion component consists of both an inhalation dose (internal) as well as an external dose from radionuclide deposition on the ground. Therefore, table summarizes the contribution, to total expected risk, that is attributed directly to the controlling waste form characteristic.

Inspection of the table shows clearly that the pre-emplacment risks are much greater than the post-emplacment risks. The risk value shown is summed over the total time considered (about 10^6 years). We integrated over time to a value of 10^6 years. After 10^6 years the risk per unit time is relatively insignificant in all

cases studied. By way of comparison the peak 50 year population dose expected risk value is 8×10^{-10} man-REM/MWe-yr for the baseline repository design.

The reader should be cautioned that the absolute values given in the table are very sensitive to modeling assumptions. Indeed, variation of several orders of magnitude are possible, depending on baseline design assumptions and/or assumed input values. To be specific, we have taken as a given the initial ERDA-C.I.I repository/waste canister criteria, i.e., 3.5 Kw maximum thermal waste canister output and 12 inch O.D. canister diameter. Also, as a given emplacement at 10 years in the repository. To meet these input parameters, the calcine waste forms could not be solidified until 5 years after generation, even with dilution. No risk assessment, in this study, was made of high level liquid waste storage. The glass waste form was solidified at one-year. Referring again to the table, in the interim storage operation row, glass volatilization row we find a risk number of 1.9 in parenthesis. As a baseline design,

RADIOLOGICAL RISK (EXPECTED VALUE)
(WHOLE BODY DOSE IN man-REM/MW-y-e)

WASTE MANAGEMENT SYSTEM OPERATION		WASTE FORM MATRIX TYPE					
		Calcine			Glass		
Fuel Reprocessing Plant	Handling	Waste Form A VOL.	AIR.D.	DISS.	Waste Form B VOL.	AIR.D.	DISS.
	Filter Works	0	0	-	0	0	-
	Filter Fails	0	0	-	0	0	-
Interim Storage LCC							
	Filter works	0	0	-	1.2E-12	0	-
	Filter fails	0	0	-	2.3E-11	0	-
	Pool Drains	0	0	-	(1.9)	0	-
Transportation							
	Truck	6.1E-3	5.9E-1	1.4E-2	1.5E-1	3.2E-4	1.4E-8
	Train	4.9E-3	1.2E-1	1.5E-2	1.8E-1	8.6E-5	8.3E-9
Repository Site	Handling						
	Filter works	0	0	-	0	0	-
	Filter fails	0	0	-	0	0	-
	Surface storage	1.2E-13	1.2E-13		1.2E-13	1.2E-13	
Post Emplacement History							
		-	-	1.7E-4	-	-	1.7E-4
Sub-Totals							
	Truck	6.1E-3	5.9E-1	1.4E-2	(2.05) 1.5E-1	3.2E-4	1.7E-4
	Train	4.9E-3	1.2E-1	1.5E-2	(2.08) 1.8E-1	8.6E-5	1.7E-4
TOTALS							
	Truck	6.101E-1			(2.05) 1.505E-1		
	Train	1.399E-1			(2.08) 1.803E-1		

*Either truck or train values should be used not both
i.e., no assumptions on partial shipments in both models

we chose a spacing based on a previous study, and a sequence of operations such that waste canisters of a similar age were grouped together, to minimize handling. This baseline design gave high risk numbers, when the scenario of a massive pool integrity failure was studied. A change in spacing and different handling operation mode would reduce the expected risk to a very low number. Therefore, we have shown the total risk for glass with and without the pool drain case.

Therefore, the real "pinch point" in the waste management system appears to be the transportation system. The fine particle size associated with calcine, leads to risks from particulate dispersion.

Also, the solubility of calcine leads to higher risk numbers for dissolution than compared to glass. Severe fire conditions that lead to melting of the glass causes rapid corrosion of the canister wall. This results in higher risks associated with volatile radionuclides from glass as compared to calcine.

Finally, as will be discussed later in more detail, the calculated risk values are based on repository baseline design that does not require total containment of the waste over all of its potentially hazardous lifetime. Indeed, we have assumed immediate water intrusion into the repository immediately after sealing.

RADIOLOGICAL RISK (EXPECTED VALUE)
(WHOLE BODY DOSE IN man-REM/MW-y-e)

WASTE MANAGEMENT SYSTEM
OPERATION

WASTE FORM MATRIX TYPE

		Calcine			Glass		
Fuel Reprocessing Plant	Handling	Waste Form A			Waste Form B		
		VOL.	AIR.D.	DISS.	VOL.	AIR.D.	DISS.
	Filter Works	0	0	-	0	0	-
	Filter Fails	0	0	-	0	0	-
	Interim Storage LCC						
	Filter works	0	0	-	1.2E-12	0	-
	Filter fails	0	0	-	2.3E-11	0	-
	Pool Drains	0	0	-	(1.9)	0	-
Transportation*	Truck	6.1E-3	5.9E-1	1.4E-2	1.5E-1	3.2E-4	7.4E-8
	Train	4.9E-3	1.2E-1	1.5E-2	1.8E-1	8.6E-5	8.3E-9
Repository Site	Handling						
	Filter works	0	0	-	0	0	-
	Filter fails	0	0	-	0	0	-
	Surface storage	1.2E-13	1.2E-13		1.2E-13	1.2E-13	
Post Emplacement History		-	-	1.7E-4	-	-	1.7E-4
Sub-Totals	Truck	6.1E-3	5.9E-1	1.4E-2	(2.05) 1.5E-1	3.2E-4	1.7E-4
	Train	4.9E-3	1.2E-1	1.5E-2	(2.08) 1.8E-1	8.6E-5	1.7E-4
TOTALS							
	Truck	6.101E-1			(2.05)	1.505E-1	
	Train	1.399E-1			(2.08)	1.803E-1	

*Either truck or train values should be used not both
i.e., no assumptions on partial shipments in both models

3. INTERIM STORAGE AT FUEL REPROCESSING PLANT

3.1 Handling Events

HANDLING EVENTS CONTRIBUTE A NEGLIGIBLE AMOUNT OF RISK TO THE OVERALL WASTE MANAGEMENT SYSTEM.

THE RELEASE ASSOCIATED WITH A HANDLING CRANE STALL WAS STUDIED. SIMILARLY,
THE RELEASE ASSOCIATED WITH DROPPING A HIGH LEVEL WASTE CANISTER OUTSIDE
THE STORAGE POOL WAS INVESTIGATED.

The effect of accidental releases of high-level nuclear wastes during handling at the fuel reprocessing plant (FRP) is considered in the event tree in the opposite figure. The accidents postulated are representative of those that might occur when canisters of solidified waste are stored in a pool of water. Operations such as canister sealing, decontamination, and testing are not considered because they are conducted in-cell.

As we follow down the left side of the event tree to the melt event, we find that the conditioned

probability of canister melt is zero. Heat transfer calculations show that sufficient heat is convected away to keep the canister wall below the critical temperature.

Similarly, for the right hand side of the tree we also find the conditional probability of canister melt is zero, for similar heat transfer arguments.

In any case, if we assumed a conditional probability of occurrence equal to 1.0 and allowed a release function of 1.0×10^{-10} we would calculate a risk of $<10^{-10}$ man-REM/MWe-yr.

3. INTERIM STORAGE AT FUEL REPROCESSING PLANT

3.2 Storage Pool Events

CATASTROPHIC LOSS OF WATER COOLING DURING INTERIM STORAGE
COULD CONTRIBUTE A MAJOR RISK TO THE OVERALL WASTE MANAGE-
MENT SYSTEM.

LOSS OF COOLING WATER INVOLVING DRAINAGE OF THE INTERIM STORAGE POOL
WAS FOUND TO BE THE MAJOR RISK CONTRIBUTOR. LOSS OF ACTIVE WATER ...
COOLING CIRCULATION WILL NOT BE A MAJOR CONTRIBUTOR TO OVERALL RISK. ...
ABSOLUTE VALUES OF EXPECTED RISK ARE SENSITIVE TO MODELING ASSUMPTIONS.

During interim storage in an active
water cooled storage basin, high
level wastes are contained in
canisters, grouped in regular
arrays. Removal of heat generated
by radioactive decay is with an
active water cooling system.

Two accident scenarios were
considered; first, loss of coolant
circulation; and second,
catastrophic loss of coolant by
rupture of the basin, e.g., with a
seismic event. The canisters of
waste may fail due to thermally
induced effects. Without coolant,
the canister wall temperature rises
to the critical point where failure
occurs either due to corrosion or
mechanical failure of the canister
wall. The rapidity of failure and
therefore the conditional probability
of occurrence depends upon the
heating rate of the waste and on
the geometry of the canister array.
The event tree is shown opposite.

After catastrophic loss of cooling
water, air flow by natural
circulation around the canisters is
the prime method by which heat is
lost to the surroundings. The
modeling of this phenomena is shown
schematically in the lower figure.
These results were then combined
into a lumped parameter computer

model that represented the heat
transfer system with a small number
of nodes interconnected by a
network of transmittances. Final
thermal model results are
summarized in the table below.

Waste Age	Waste Type	
	Glass	Calcine
1 yr.	Failure	Failure
2 yr.	Failure	Failure
5 yr.	Near Failure	No Failure
10 yr.	No Failure	No Failure

Insertion of these data, with the
failure probabilities and release
fraction parameters into the event
tree model gives the final
radiological risk expected values.

As a baseline design for the pool
storage racks, we chose a spacing
used in a previous design study.
The operating mode placed canisters
of the same age in adjacent racks.
Using these assumptions a high risk
number, 1.9 man REM/MW-yr, was
calculated. A change in rack
spacings and a staggered rack
location for the operating mode,
would produce a significant decreased
in the expected risk. Indeed, by
optimal design the risk could be
reduced to a very low number.

4. TRANSPORTATION

4.1 Baseline Shipping Cask Configuration

CONCEPTUAL DESIGNS OF SHIPPING CASK CONFIGURATIONS FOR HIGH LEVEL WASTES WERE NECESSARY TO PERFORM RISK EVALUATIONS.

A CONCEPTUAL DESIGN FOR A RAILROAD SHIPPING CASK WAS REQUIRED TO EVALUATE THE RISK ASSOCIATED THE TRANSPORTATION OPERATION. THE DESIGN INCLUDED BOTH THERMAL AND DYNAMIC STRESS ANALYSES. SIMILAR WORK WAS DONE FOR A CONCEPTUAL TRUCK SHIPPING CASK.

The baseline rail shipping cask design is a LLL modification of the GE IF-300 spent fuel cask. The main features of the design are shown in the upper figure. Four 12 inch diameter high level waste canisters are contained in a stainless steel insert placed in the 37.5 inch diameter, 10 foot long cavity. The dry stainless steel insert acts to conduct heat from the canisters. Gamma shielding is accomplished through the use of a 4 inch thick depleted uranium shell. Neutron shielding is provided by a 5 inch thick water ring enclosed by a corrugated stainless steel outer sleeve. Finned cask ends act as both impact limiters and heat transfer augmentators. A stainless steel "Gray Loc" seal ring is used to seal the cask contents from release. A mechanical fan system is used to provide cooling during shipping.

The baseline truck shipping cask

design is based on the Nuclear Assurance Corporation Spent Fuel Cask, the NFS-4. The existing design was shortened to a total cavity length of 10 feet. The truck cask will contain only one 22 inch diameter waste canister. Lead shielding would be 6.65 inches thick. An ethylene glycol/water neutron shield 4.5 inches thick would be used. The end cap design is of eight inch thick steel to which are attached an asbesto sheet and 12 inch thick balsa wood impact limiters. This is shown in the lower figure.

It was necessary to verify that our baseline cask design would meet current regulations, i.e., type B packaging specifications 10CFR71 and 49 CFR 173.398. A series of computer calculations confirmed that our baseline design would survive the accident conditions spelled out in Appendix B. Both impact and thermal calculations were done.

BASELINE DESIGN RAIL SHIPPING CASK

BASELINE DESIGN TRUCK SHIPPING CASK

4. TRANSPORTATION

4.2 Railroad Events

RAILROAD TRANSPORT IS AN IMPORTANT ELEMENT IN THE RISK ASSOCIATED WITH A WASTE MANAGEMENT SYSTEM.

RELEASES OF RADIONUCLIDES CAUSED BY ACCIDENTS INVOLVING IMPACT AND FIRES WERE CONSIDERED IN DETAIL. BOTH AIR AND WATER PATHWAYS WERE MODELED TO CALCULATE EXPECTED VALUES OF RISK. A NEW RELEASE FUNCTION METHODOLOGY WAS DEVELOPED USING THE CONCEPT OF CASK THERMAL FAILURE LOCI.

From the event tree shown opposite, we may follow the left branch, labeled impact, to evaluate risk from impact-induced radionuclide releases. Best engineering judgment, based on the results of computer simulation analysis and hand calculations suggested the "corner drop" is the most severe impact accident. A "corner drop" on the cask closure end produces a bending moment which may fail the closure bolts in tension. Thus an impact which causes cask failure would almost certainly rupture the canisters with a subsequent release of radioactivity.

Using available accident statistical data, the expected value of risk can be calculated from these radioactivity release functions. The waterpath contribution is computed using the conditional probability of ruptured cask entering waterway equal to 10^{-2} .

In the case of fire-induced releases, the cask seal is the critical component. Breaching of the cask occurs because of early seal failure in a fire environment. Note that heat transfer throughout the cask as well as canister failure modes are both time dependent processes. Canister failures occur because of corrosion of the wall by

the melted glass, or in the case of calcine, by oxidation and creep of the wall. This failure modes are Arrhenius in nature, and therefore we can establish a discrete canister failure temperature. Thus we have separated the heat transfer process from the canister failure mechanism.

This allows a new methodological approach whereby we can define, uniquely, a failure locus. From the heat transfer calculations we find the time to reach the canister failure temperature, defined by the Arrhenius data treatment. This data forms the failure locus on a temperature versus time plot. The threshold for release of radionuclides is defined as when the cask experiences a fire whose characteristics, i.e., temperature time, are above this locus, implying canister failure. For glass the canister failure temperature is $1200 \pm 50^{\circ}\text{C}$, and for calcine $1300 \pm 50^{\circ}\text{C}$.

Releases due to a puncture failure of the cask/canister are not felt to be credible because of the massive nature of the shipping casks. In effect the risk would be calculated to be zero, since the probability distribution function and the release function versus severity curves do not overlap.

4. TRANSPORTATION

4.3 Truck Events

TRANSPORTATION BY TRUCK CONSTITUTES A MAJOR COMPONENT OF WASTE MANAGEMENT SYSTEM RISK

ACCIDENTS LEADING TO RADIONUCLIDE RELEASES INVOLVING TRUCK SHIPPING CASKS IN BOTH FIRE AND IMPACT SITUATIONS WERE STUDIED. AS PREVIOUSLY, BOTH AIR AND WATER PATHWAYS WERE CALCULATED. VOLATILIZATION RELEASES WERE MODELED USING EXPERIMENTAL LABORATORY DATA,

As with the railroad transport, the risk was calculated using the event tree opposite. The fire and impact accident scenarios were modeled in the same fashion, as discussed previously in railroad events. Note that the conditional probability of impact with a train locomotive is 3.1×10^{-3} . Again the conditional probability for a ruptured cask entering a waterway was taken to be 10^{-2} .

The "corner drop" was again felt to be the limiting impact case. The truck cask uses balsa wood to absorb the impact energy. Current regulations require survival of a 30 foot drop, so that the energy absorbing systems are usually designed to "bottom out" as the result of such a drop or equivalent impact. Impacts greater than that currently specified would result in the excess being absorbed by the cask itself. This would lead to high deceleration forces in the shipping container with a higher potential for damage.

Volatilization releases of

radionuclides i.e., Cs, Te, and Ru, have been modeled using recently available laboratory data. We chose a likely post-accident geometry for the waste matrix. This assumes no effective barrier being presented by the ruptured cask and the canister lies horizontally with its upper surface exposed. Obviously the release fraction is sensitive to the assumed geometry, but our model represents an average situation. The laboratory experiments indicate Cs loss from glass is surface-controlled, whereas the Cs loss from calcine is diffusion controlled. One can fit the data to an Arrhenius type equation. Similarly, a release function for Ru from calcine can be developed. It should be noted that this methodology is also applicable to the rail transportation operation.

As previously noted, releases due to puncture failure of the cask/canister are not felt to be credible because of the massive nature of the shipping casks.

5. HANDLING AT REPOSITORY

5.1 Handling Events

THE CONSEQUENCES OF HANDLING EVENTS AT THE SURFACE FACILITIES OF A HIGH LEVEL WASTE REPOSITORY ARE NEGLIGIBLE COMPARED TO OTHER OPERATIONS IN THE WASTE MANAGEMENT SYSTEM.

THE DANGER TO CANISTER INTEGRITY DURING THE HANDLING OPERATIONS AT THE REPOSITORY IS FROM IMPACT WHEN ACCIDENTLY DROPPED. AIRPLANE CRASHES INTO SURFACE FACILITIES WERE ALSO CONSIDERED.

Following down the left hand side of the event tree to the breach event we find the conditional probability of canister rupture is zero. Data derived for actual canister drop tests was used in predicting canister integrity after a crane drop.

The right hand side of the event tree considers the releases

associated with the impact of an aircraft while the waste transportation cask is at repository site. We have assumed the cask is at the surface repository facility, prior to emplacement) for a time of week (2×10^{-2} yr.) The impact and resultant fire would produce an expected value of risk of 1.2×10^{-13} man REM/MWe-y.

6 POST SEALING REPOSITORY HISTORY

6.1 Reference Repository: Geologic Setting

THE BASELINE REPOSITORY DESIGN IS SITED IN A LAYERED ROCK FORMATION

OUR REPOSITORY MODEL DOES NOT ALLOW SIGNIFICANT RADIOACTIVITY RELEASE INITIALLY. WE HAVE CHOSEN A BASELINE DESIGN WITH THE REPOSITORY LAYER BEING AN ARGILLACEOUS FORMATION, I.E., SHALE. THIS LAYERED ROCK FORMATION ADEQUATELY MODELS GEOLOGICAL PHENOMENA THAT EFFECT CONTAINMENT OF THE NUCLEAR WASTE.

Many stable rock formations are being investigated as suitable nuclear waste repository sites within the conterminus U.S. Bedded rock formations that might be satisfactory for repository sites include argillaceous formations, crystalline rock formations, and bedded rock salt deposits. Stable rock beds are characterized by regions of low earthquake activity. Also the structural element aspects of the host rock are important. The formation should contain as few as possible of joints, fractures, and faults. the host rock should also have low porosity and permeability. Given these initial conditions there will be no significant radioactivity releases from the repository.

A layered geologic model consisting of 6 layers, with variable pro-

perties for each layer, adequately models the repository. Thus we see, in the figure opposite, the repository layer, with a adjoining barrier or aquitard layers. A lower aquifer provides the driving force for water intrusion into the repository. Finally the upper aquifer allows transport of radioactivity away from the repository into a surface human use water system. The table opposite lists values used in modeling the repository site. We assumed the thickness of both the repository and barrier layers to 200 m. The dissolution time of the solidified waste was taken as 10^4 years. Ground water intrusion represents the greatest threat to containment since it provides a mechanism to transport radioactivity into the biosphere.

AQUIFER.

→ MOTION OF GROUNDWATER

	Permeability (cm/sec)	Effective Porosity	Cross Section (m ²)
Shale Repository (Vertical)	10^{-9}	0.01	10^7
Shale Barrier Layer (Vertical)	10^{-7}	0.02	10^7
Fracture Zone Around Shaft, Tunnels	10^{-5}	0.02	Tunnel: 3×10^4 Shaft: 1900
Aquifers	10^{-4}	0.10	

6 POST SEALING REPOSITORY HISTORY

6.2 Reference Repository: Hydrologic

GROUND WATER FLOW INTO THE REPOSITORY FROM NEARBY AQUIFERS PROVIDES THE MECHANISM FOR RADIONUCLIDE MIGRATION.

GROUND WATER FLOW RATES AND AQUIFER PATH LENGTHS TO SURFACE WATER USE SYSTEMS ARE BARRIERS TO MIGRATION OF RADIOACTIVITY. HYDRAULIC DISPERSION AND SORPTION RETARDATION FACTORS REDUCE RISK BY LOWERING RADIO-NUCLIDE CONCENTRATIONS AND PROVIDING TIME DELAYS FOR RADIOACTIVE DECAY.

Ground water flow rates at the repository site will be governed by the region hydraulic gradients. These gradients are controlled by regional rainfall, surface water recharge rates, and regional topography. Selection of site porosity and permeability characteristics will govern detail flow patterns within the repository per se.

For our baseline repository site modeling studies we have chosen values for the hydrologic calculations as shown in the table opposite.

Selection of aquifers with favorable dispersion characteristics as well as high sorption retardation factors is very desirable. These factors reduce radionuclide concen-

trations and provide time delays for radioactive decay.

Note that the expectation of risk values obtained are in REM/MWe-yr for the average individual. We have corrected these values to reflect population risks in man-REM/MWe-yr through the use of population limit modeling. This point will be discussed in detail later under the demography modeling section.

Control of the path length of a human water use system is similarly effect. However since loss of administrative control could conceivably shorten path length dramatically, a primary reliance should be placed on the other factors.

	Values Used	Units
Ground Water Sorption Retardation Factor		
Radioiodine	10^2	-
Other fission products	10^4	-
Actinides and Daughters (Fresh Water)	10^4	-
Dispersion Constant	25	m
Excess Hydraulic Head in Tower Aquifer	60	m
Nuclear Waste Dissolution Time	10^4	yr
Barrier Layer Thickness	200	m
Repository Layer Thickness	200	m
Distance To Aquifer Discharge	1.6×10^4	m
Aquifer Hydraulic Gradient	5×10^{-3}	
Length of Tunnel	1500	m

6.3 Repository Site: Climatologic Setting

EVALUATION OF THE STABILITY AND RANGE OF CLIMATIC EXTREMES
IS NECESSARY FOR LONG TERM PLANNING OF SITE SUITABILITY,

LARGE SCALE PAST CLIMATE VARIATION IS THE KEY TO THE FUTURE CLIMATE INPUT FOR SITE SUITABILITY STUDIES. IMPACTS OF CLIMATIC VARIABILITY ON HYDROLOGIC FACTORS SUCH AS GROUND WATER RECHARGE RATES AND ALSO ON SURROUNDING DEMOGRAPHY WILL BE HIGHLY SIGNIFICANT,

For long term planning of suitability, it is necessary to evaluate the stability and range of climatic extremes for specific regions of the conterminous U.S. Projections of possible future climate states can be generally inferred from past climate variations over period 10^3 to 10^6 YBP (years before present). Climatic variability in the past has been so large that if the future climate exhibits similar diversity, the impact on the ground hydrology and demography will be highly significant.

Indirect evidence shows that our climate has varied on nearly all time scales. Estimates of temperature and precipitation over the time interval of interest can be made using the possible causal relationship between climate and the variation of the earth's orbit. This in turn gives a time variation of the major climate regimes which can subsequently be related to the synoptic situations typical of the various regimes.

The future projection of the time line is based on Hays et al. (1976) and Vernekar (1972) and is presently being adjusted for the present configuration of the earth's orbital variation. To add confidence to the time line prediction, the per mil enrichment of oxygen -18 (^{18}O) will be correlated to the earth's orbital variation parameters (^{18}O reflects

global ice volume and has provided a basic stratigraphy for the past million years).

The synoptic scale features of future climate regimes are provided in two ways. The first is based on past climate reconstructions using proxy data such as pollen records, tree rings, ice cores, plant and animal remains, etc. We have used Lamli's (unreferenced material) reconstruction of the 1000-500 mb thickness for a preliminary estimate of moist and dry regions of the U.S. The various climatic regimes chosen as typical of the likely range for a glacial-interglacial period in the future are: 1) maximum glacial, 2) early warm similar to "Boreal" period 7,000 to 6,000 B.C., 3) warm post-glacial similar to "Atlantic" 4,000 B.C., 4) late state of warm post-glacial similar to "Sub-Boreal" 2,000 B.C., 5) colder period similar to "Sub-Atlantic" and "Little ice age" similar to 500 B.C. and 1550-1700 A.D., and 6) the present climate.

The second approach will use the output from large three-dimensional general circulation models experiments with boundary conditions similar to those of the last glacial maximum (18,000 YBP). This will give in more detail the features of the synoptic scale variations over the U.S. under a much different climatic regime.

<u>Type</u>	<u>Representative Pattern From Past Circulate</u>	<u>Characteristic</u>	<u>Location on Time Line</u>
1	20,000 - 17,000 B.C.	Full Glacial	Minimum eccentricity
2	7,000 - 6,000 B.C.	"Boreal" early warm	Few thousand years before peak in pre- cession
3	4,000 B.C.	"Atlantic" moist post glacial- warm	Maximum in precession
4	2,000 B.C.	"Sub Boreal" dry post-glacial- warm	Maximum in precession
5	500 B.C. Also 1,500 - 1,700	"Little Ice Age"	Past maximum in pre- cession

6. POST SEALING REPOSITORY HISTORY

6.4 Future History: Water Intrusion Events

WATER INTRUSION REMAINS THE MECHANISM FOR BREACHING THE REPOSITORY AND TRANSPORTING RADIONUCLIDES INTO THE BIOSPHERE.

SEISMIC EVENTS ARE IMPORTANT FACTORS IN INCREASING POTENTIAL WATER INTRUSION RATES INTO THE REPOSITORY. DETERIORATION OF SEALS AROUND MAN MADE OPENINGS IN THE REPOSITORY HOST ROCK WILL INCREASE WATER INTRUSION RATES. LOSS OF ADMINISTRATIVE CONTROL HAS ADDITIONAL IMPLICATIONS OF WATER INTRUSION,

After completion of backfilling for a nuclear waste repository, that has reach its design storage capacity, the man-made water intrusion barriers will deteriorate both with time and subsequent seismic disturbances. The types of deterioration of concern are related to increases in aquifer flow rates, or decreases in path length, sorption retardation factors, or dispersion coefficients.

Increased intrusion flow rates are expected from deterioration in borehole and shaft integrity, as seen in the crosshatched zones in the figure opposite. Seismic events may increase the permeability

both in existing faults and in induced fracture zones around man-made boreholes and shafts. They may also produce new faults with associated fracture zones.

Water intrusion along preferential flow channels can lead to void formation by dissolution. These dissolution zones can lead to underground subsidence and formation of breccia pipes. Note that backfill subsidence can lead to similar phenomena. Avoidance of sites with strata subject to dissolution reduces the potential pathways for water intrusion caused by dissolution void and channel subsidence.

7. RADIONUCLIDE TRANSPORT SYSTEMS AND DEMOGRAPHIC FACTORS

7.1 Human Water Use Modeling

A RIVER, ESTUARY, AND OCEAN WATER PATHWAY FOR DISPERSAL OF RADIONUCLIDES WAS MODELED TO CALCULATE SUBSEQUENT DOSE TO MAN.

THE CONCENTRATION OF RADIONUCLIDES CHANGES AS THE WATER FLOWS DOWN THE RIVER TO THE OCEAN. THE FIGURE OPPOSITE ILLUSTRATES THIS PHENOMENA AND LISTS THE EXPOSURE PATHWAYS TO MAN. MODELING OF THE RADIONUCLIDE MIGRATION VIA AQUIFERS FROM THE REPOSITORY TO THE RIVER SHOWS APPRECIABLE DELAY AND REDUCTION IN CONCENTRATION PER UNIT TIME.

The concentration of radionuclides in the water system changes as it flows from the river to the ocean. The processes reducing these concentrations include dilution by mixing, sedimentation, radioactive decay, and removal of water and food products by man. The figure opposite illustrates the water model and lists the exposure

pathways to man. In this system radionuclides may be stored in the water, sediment, topsoil and in the human population.

Doses are calculated by multiplying the pathway concentration by effective food intake and by the 50-year dose conversion factor.

7. RADIONUCLIDE TRANSPORT SYSTEMS AND DEMOGRAPHIC FACTORS

7.2 Airborne Dispersion Modeling

PUFF-TYPE RELEASES WERE USED TO MODEL IMPACT TRANSPORTATION
ACCIDENTS WHERE AS PLUME-TYPE RELEASES WERE USED TO CALCULATE
ALL OTHER AIRBORNE RELEASES.

AT THE REPROCESSING PLANT OR THE REPOSITORY, IT IS POSSIBLE TO HAVE
EITHER A CONTINUOUS (PLUME) AIRBORNE RELEASE OR AN INSTANTANEOUS (PUFF)
RELEASE. A TRANSPORTATION ACCIDENT WOULD MOST LIKELY RESULT IN AN
INSTANTANEOUS (PUFF) RELEASE FOR IMPACT ACCIDENTS AND CONTINUOUS (PLUME)
FOR FIRES.

One may postulate accidents involving solidified high level waste in which a portion of the waste becomes airborne. To properly calculate downward population dose, one must estimate dispersion and deposition of such airborne debris. We assume a unit curie release at one meter above the ground, with a one meter radius and half thickness initial source. At the fuel processing plant or the waste repository, it is possible to have either a continuous (plume) release or an instantaneous (puff) release. Indeed, a viable scenario lies between these two extremes. On the other hand, a transportation impact accident would most likely result in an instantaneous release. Fires associated with accidents would have continuous releases.

To calculate population doses due to inhalation of airborne debris one must calculate surface concentration. Since a plume meanders about an average trajectory at an angular variation dependent on the atmospheric

stability while a puff is influenced only by the winds along its trajectory, the standard deviations, σ , for a plume are larger than those for a puff in similar conditions of stability and windspeed. This point is best illustrated by referring to the figures opposite. Note the more rapid falloff with distance of the relative puff-center concentration (top figure) as compared to the relative plume-centerline concentration (lower figure). The average and waste case curves refer to factors involving windspeeds and atmospheric stability.

In addition to surface concentration, surface deposition has been incorporated into the calculations. This modification involves use of a deposition velocity that represents the rate of particulate impaction with the surface due to turbulent eddies near the ground. We have also modified the equations to include plume depletion as deposition occurs.

7. RADIONUCLIDE TRANSPORT SYSTEMS AND DEMOGRAPHIC FACTORS

7.3 Demographic Factors

DEMOGRAPHIC MODELS USED IN THIS STUDY ARE BASED ON REALISTIC ASSUMPTIONS.

THE POPULATION DENSITIES FOR URBAN AREAS, TOWNS AND CITIES, WERE APPROXIMATED BY GAUSSIAN CURVES. THE RURAL POPULATION DENSITY WAS ASSUMED TO BE NEARLY UNIFORM.

We have chosen population density, i.e., the number of persons per unit of area, as the best method for expressing population in analytic form. These analytic forms are useful in estimating total population dose from accidents involving solidified high level waste.

In and around towns the population density can be approximated by Gaussian curves. General expressions for such population densities are given on the opposite page. We have further assumed that the rural population density will be uniform.

We assumed that future spent fuel reprocessing plants will be located in relatively sparsely populated regions. Such plants will likely have small towns within easy commuting distance. Thus we modeled Barnwell, S.C. as a typical site, since the Barnwell Nuclear Fuel Plant is nearby.

Transportation demography was modeled using the Dallas-Fort Worth area as typical for urban transportation accidents. The rural transportation accident demography was modeled assuming the average U.S. population density (1973) of 2.2×10^{-5} persons per square meter.

The demography for a typical future waste repository site was modeled assuming the population distribution (1973) for the Nevada Test Site.

Population dose is normally

calculated via an assumed population within a 50-mile radius of the contaminated region. This procedure would require a fairly detailed knowledge of future demography to calculate future population doses. We chose an alternative procedure which does not require detailed demographic projections. Our model is based on a future in which the total vegetable intake is limited by the amount of contaminated irrigation water used. This amount is limited by the river or lake's flow rate. This model allows definition of an effective population with these assumptions all population members are adults the average diet is derived totally from contaminated soil

From these assumptions, it follows that radionuclides derived from animal products is due to animal ingestion of contaminated water and feed.

Since aquatic food yield is limited by the characteristics of the rivers, lakes, estuaries, and ocean, substantial changes in these yields are not expected. These contributions to population dose are relatively fixed and do not depend on demographic factors.

Thus, we see that a limit may be set on population dose since irrigation rates define the total quantity of vegetable and animal products derivable from lands irrigated by the contaminated water and the dose contribution from aquatic food chains is also relatively fixed.

<u>Accident Type</u>	<u>Profile Used</u>
Reprocessing-Interim Handling and Storage	Barnwell, S.C.
Transportation-Urban	Dallas-Ft. Worth <u>urban density</u>
Transportation-Rural	Dallas-Ft. Worth <u>rural density</u> (>100 Km from city center)
Repository-Handling	Nevada Test Site

1. From the accident site to the nearest city ($0 < x < x_c$),

$$\rho = (\rho_c - \rho_s) \exp [-b_1 (x - x_c)^2] + \rho_s;$$

2. From the city to outlying rural areas ($x > x_c$),

$$\rho = (\rho_c - \rho_r) \exp [-b_2 (x - x_c)^2] + \rho_r,$$

where b_1 and b_2 are coefficients determining the shape of the Gaussian curves, and where x = distance downwind from the accident (m), x_c = distance from the accident to the city (m), ρ_s = population density at the accident site (persons/m²), ρ_c = population density in the city (persons/m²), and ρ_r = rural population density (persons/m²).

STUDY COORDINATOR

R. A. HECKMAN

APPENDICES

- 1.1.1 R. A. Heckman (LLL)
- 1.1.2 R. Van Konynenburg (LLL)
- 1.1.3 R. A. Heckman (LLL)
- 1.1.4 R. Van Konynenburg (LLL) and A. Toy, Jr. (LLL)
- 1.1.5 R. Van Konynenburg (LLL)
- 1.1.6 R. Van Konynenburg (LLL) and A. Iantuono (LLL)
- 1.3.1 R. A. Heckman (LLL), J. J. Cohen (LLL) and K. Tonnessen (LLL)
- 1.3.2 -
- 1.3.3 R. Van Konynenburg (LLL)
- 1.3.4 A. Toy, Jr. (LLL)
- 1.3.5 H. A. Tewes (LLL) and Y. C. Ng (LLL)
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- 3.1.0 T. Holdsworth (LLL) A. Iantuono (LLL) and R. Van Konynenburg (LLL)
- 3.1.2 A. Sutherland (TASC), J. L. Center (TASC) and B. S. Crawford (TASC)
- 3.2.0 A. Iantuono (LLL)
- 3.2.1 A. Iantuono (LLL) and S. Oston (TASC)
- 3.2.1.1 S. Oston (TASC) and A. Sutherland (TASC)
- 4.1.0 D. Ensminger (TASC)
- 4.1.1 A. Iantuono (LLL) and S. Oston (TASC)
- 4.1.2 A. Iantuono (LLL) and S. Oston (TASC)
- 4.1.3 A. Iantuono (LLL)
- 4.1.4 R. Van Konynenburg (LLL) and A. Iantuono (LLL)
- 4.1.5 R. Larder (LLL)
- 4.1.6 R. Van Konynenburg (LLL) and D. Ensminger (TASC)
- 4.2.0 D. Ensminger (TASC)
- 4.2.1 A. Iantuono (LLL) and S. Oston (TASC)
- 4.2.1 R. Larder (LLL), J. Leake (LLL), A. Iantuono (LLL) and D. Ensminger (TASC)
- 4.3.0 A. Iantuono (LLL) and S. Oston (TASC)
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- 5.1 T. Holdsworth (LLL) and A. Sutherland (TASC)
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- 7.2.0 K. Petersen (TASC) and D. Ensminger (TASC)
- 7.3.0 K. Petersen (TASC) and D. Ensminger (TASC)

1.1 FUEL CYCLE

1.1.1 FUEL CYCLE DETAIL

To understand how radioactive wastes are generated in nuclear power production, the nuclear fuel cycle must be discussed. This cycle comprises the operations and facilities that prepare, utilize, and reconstitute nuclear fuel.

The basic fuel for commercial power reactors is uranium. From Figure 1.1-1 we see that uranium ore is mined, milled, processed, enriched, and then fabricated into reactor fuel rods. After energy is extracted from these rods in the reactor, they are sent to interim storage. From storage, the rods may then be sent to a reprocessing plant where the unused uranium and the plutonium produced in the reactor can be recovered and recycled. After separation of the uranium and plutonium, the residual radioactive material is waste, and must be treated for safe management. High-level radioactive wastes will be shipped to a federal repository for final disposition, as required by government regulations.

The nuclear fuel cycle encompasses all of these activities. The "front end" of the cycle includes all activities up to operation of the reactor. The "back end" of the cycle includes all activities after removal of the spent fuel rods from the reactor.

"Closing the fuel cycle" would provide for operating the reprocessing plants, recycling plutonium and uranium into nuclear fuel, and

permanently storing the resulting radioactive waste. Recycling uranium and plutonium would increase man's utilization of his energy resources.

References

- U. S. Energy Research and Development Administration, "The Management and Storage of Commercial Power Reactor Wastes," ERDA 76-162 (1976).
- W.P. Bishop, J. Malero, and W. M. Hewitt, "HRC and Nuclear Waste : The Process for Decisions," U.S. Nuclear Regulatory Commission paper prepared for presentation at IAEA conference (May, 1977).
- W. P. Bishop and F. J. Miraglia, Jr., eds., "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission rept. NUREG-0116 (Supp. 1 to WASH-1248), (Oct. 1976)

FUEL CYCLE DETAIL

1.1.2 SOLIDIFIED HIGH LEVEL WASTE (SHLW) DEFINED

The Code of Federal Regulations in 10 CFR 50, Appendix F (1976) defines high-level liquid radioactive wastes as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels. "SHLW," for the purposes of this study, is defined as waste resulting from the conversion of these high-level liquid wastes to a solid form, in addition to the undissolved solids removed by centrifugation during the feed clarification step that follows the dissolving of the spent fuel elements.

1.1.3 HIGH LEVEL WASTE QUANTITIES

The quantities of high-level wastes existing and projected in the U.S. are shown in Table 1.1-1.

Reference

W. P. Bishop, J. Malero, and W. M. Hewitt, "NRC and Nuclear Waste : The Process For Decisions," U.S. Nuclear Regulatory Commission paper prepared for presentation at IAEA conference (May, 1977).

HLW QUANTITIES EXISTING & GENERATED

EXISTING: JANUARY 1977

- MILITARY (150 RRY*)
- COMMERCIAL (~ 100 RRY)

280,000 m³ (MIXED)
 2,300 m³ (LIQUID) HLW
 3,500 m³ SPENT FUEL

GENERATION RATES

- 1 RRY
- PRESENT INDUSTRY 1 YR. (64 RRY)
- 1 REACTOR FOR LIFE (30 RRY)

35 m³ SPENT FUEL
 OR
 8 m³ HLW (SOLID)
 2,200 m³ SPENT FUEL
 OR
 512 m³ HLW (SOLID)
 1,050 m³ SPENT FUEL
 OR
 450 m³ HLW (SOLID)

* RRY = REFERENCE REACTOR YEARS AS USED IN NUREG-0116

46

1.1.3 HLW QUANTITIES

1.1.4 GENERAL CHARACTERISTICS OF SHLW

The main features that all SHLW forms have in common are the presence of radioactive fission products and actinides and the concomitant emission of ionizing radiation, much of which is converted to heat energy in the body of the waste. As an example, consider the SHLW which results from the generation of one megawatt electric year of energy. The volume of this waste, depending on the form selected, would lie in the range of about $1-3 \times 10^{-3}$ cubic meters; that is, a cube between 10 and 15 centimeters on a side. The external gamma dose rate and heat output of this cube are shown in Fig. 1.1-2 for three ages of the waste.

Fig. 7-1-1-2-1
CHARACTERISTICS OF SOLIDIFIED
HIGH LEVEL WASTE

1.1.4.1 RADIONUCLIDE CONTENT

The amount and species of the radionuclides present in SHLW can be calculated from a knowledge of the history of the spent reactor fuel to be processed and the partitioning that occurs in the reprocessing plant and the waste solidification facility. Calculations of this type have been performed by Blomeke et al. (1974) at Oak Ridge National Laboratory, using the ORIGEN computer code (ORNL, 1970). Because of radioactive decay and the wide spread in the values of the half-lives of the various nuclides, the amounts and species present greatly depend on the age of the wastes. For the present study, consideration has been limited to the nuclides that contribute the major hazard potential. (See Appendix 1.3.4). Tellurium has been included because of its relatively high volatility.

Given a power level of 30 megawatts per megagram of uranium, a burnup of 33,000 megawatt-days per megagram of uranium, a flux of 2.92×10^{13} neutrons per square centimeter per second, and a thermal-to-electrical conversion efficiency of 35.4 percent, the activities of these nuclides are as shown in Table 1.1-2, expressed in curies per megawatt electrical-year of waste.

The fuel is assumed to be UO_2 . If it were mixed oxide (PuO_2 as well as UO_2), the numbers would be similar for the fission products (within a factor of 2) but larger for the actinides (about a factor of 10, depending on the species).

Table 4-1. (Continued)

	150 da	1 yr	2	5	10	10 ²	10 ³	10 ⁴	10 ⁵	10 ⁶
²¹⁰ Po	0	0	0	0	0	2.3E-8	2.2E-6	8.8E-5	7.0E-4	1.7E-4
²²⁶ Ra	0	0	0	1.7E-9	3.4E-9	3.5E-8	2.2E-6	8.8E-5	7.0E-4	1.7E-4
²²⁹ Th	0	0	0	0	1.3E-9	2.3E-8	2.2E-6	1.7E-4	3.7E-3	9.1E-3
²³⁰ Th	6.6E-7	6.6E-7	6.6E-7	6.6E-7	6.6E-7	1.0E-6	1.2E-5	1.1E-4	6.9E-4	1.7E-4
²³¹ Pa	7.7E-7	7.7E-7	7.7E-7	7.7E-7	7.7E-7	7.7E-7	8.1E-7	1.2E-6	7.5E-6	1.0E-5
²³³ U	4.6E-7	4.6E-7	4.6E-7	4.6E-7	4.6E-7	4.6E-6	4.8E-5	4.9E-4	4.0E-3	9.0E-3
²³⁷ U	1.1E-2	1.1E-2	1.1E-2	1.1E-2	1.1E-2	1.1E-2	1.2E-2	1.2E-2	1.3E-2	8.5E-3
²³⁸ U	1.1E0	2.5E0	5.2E0	1.5E1	3.2E0	1.6E0	3.8E-3	0	0	0
²³⁹ Pu	5.2E-2	5.2E-2	5.2E-2	5.2E-2	5.2E-2	5.2E-2	6.4E-2	1.3E-1	1.8E-2	7.6E-9
²⁴⁰ Pu	7.6E-2	8.1E-2	9.2E-2	1.2E-1	1.4E-1	2.7E-1	2.5E-1	9.9E-2	9.7E-6	0
²⁴¹ Pu	1.8E1	1.7E1	1.6E1	1.4E1	1.0E1	1.5E-1	9.6E-3	4.6E-3	2.4E-6	0
²⁴¹ Am	5.4E0	5.4E0	5.4E0	5.4E0	5.5E0	5.1E0	1.1E0	4.6E-3	0	0
²⁴³ Am	5.4E-1	5.4E-1	5.4E-1	5.4E-1	5.4E-1	5.4E-1	5.0E-1	2.3E-1	6.6E-5	5.2E-9
²⁴² Cm	6.7E2	1.9E2	4.5E1	4.2E-1	2.2E-1	1.5E-1	2.4E-3	0	0	0
²⁴⁴ Cm	7.8E1	7.6E1	7.3E1	6.5E1	5.4E1	1.7E0	0	0	0	0

* These values are based on ORNL (1970), Bismack (1974), and Gera (1975).

1.1.4.2 HEAT GENERATION RATE

Figure 1.1-3 shows the heat output from this cube of SHLW as a function of time out to an age of 100 years.

1.1.5 REFERENCE WASTE FORMS AND THEIR PROPERTIES

During the past few years a great deal of effort has been expended in this country and elsewhere to develop SHLW forms. This work is described in the recent ERDA document on technical alternatives for waste management (ERDA, 1976) and in the proceedings of two conferences held in Europe (OECD, 1973, and IAEA, 1976). In addition, there have been numerous papers in the technical journals describing various treatment processes and waste forms.

Although the detailed performance requirements for SHLW have not been evaluated up to this time, the development of waste forms appears to have been directed toward these general objectives:

- To reduce the volume of the liquid waste.
- To convert the waste into a form having greater chemical, thermal, radiolytic, and mechanical stability.
- To develop a process that is economically feasible, taking account of simplicity, reliability, energy requirements, hardware costs, and raw materials added in the process.

These objectives are generally felt to be best satisfied by solidification of the wastes, and this has become the policy of the U.S. Government, as delineated in 10 CFR 50, Appendix F (1976), as well as that of several European countries. Accordingly, workers in

various laboratories have developed a range of solid-waste form options that meet the above objectives to varying degrees.

The most often proposed SHLW forms range from salt cake through calcines, glasses, supercalcine, coated particles, sintered products, glass ceramics, nepheline syenite, metal matrix composites, ion exchange products, and rock melt. In general, the more complex or higher technology SHLW forms are more expensive, but offer greater stability against radionuclide release.

We have used two reference SHLW forms in this study: a calcine and a glass. These waste forms were chosen because they span a considerable range in properties thought to be important to radionuclide release, and because both have received considerable development in the U.S. and elsewhere. Each will be described, and pertinent properties will be given.

1.1.5.1 REFERENCE CALCINE

The reference calcine is patterned after the spray calcine developed at Battelle Pacific Northwest Laboratories (Bonner, et al., 1976). Its composition is that designated by Mendel as PW7 (Mendel, 1974).

The reference calcine is in the form of a fine powder, and is assumed to have been heated to 900°C in order to drive off the residual water and to convert the residual nitrates to oxides.

1.1.5.2 REFERENCE GLASS

The reference glass is a zinc borosilicate glass patterned after the reference glass developed at Battelle Pacific Northwest Laboratories and described by Mendel (1977), except that it is formed using waste of PW7 composition.

1.1.5.3 PROPERTIES OF REFERENCE WASTE FORMS

The values of the main properties used in this study are shown in Table 1.1-3.

Reference

W. P. Bishop and F. J. Miraglia, Jr., eds., "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission rept. NUREG-0116 (Supp. 1 to WASH-1248), (Oct. 1976).

TABLE A.1.1.3.1 SUMMARY OF WASTE PROPERTIES

<u>Property</u>	<u>Reference Calcine</u>	<u>Reference Glass</u>
Specific Mass (kg waste/metric ton uranium)	66 (undiluted) 78 (diluted)	207
Density (kg/m ³)	1240	3300
Thermal Conductivity (Watts/ metre-Kelvin)	<u>T(°C)</u> 25 .094 200 .12 400 .14 800 .20	.85 1.0 1.2 3.7
Heat Capacity (Joules/ kg Kelvin)	<u>T(°C)</u> 25 600 400 660 800 710	630 780 780
Melting or Softening temp. (°C)	1400	900 (easy flow)
Size Distribution	Powder, 50 wt.% below 10 microns	Large block with some cracks
Solubility	All cesium and strontium immediately soluble, slow leaching of other nuclides	Very slow leaching of all radionuclides
Volatility	All cesium, tellurium, and ruthenium lost above about 1200°C.	All cesium lost above about 1200°C.

1.1.6 SHLW MANAGEMENT SEQUENCE

When spent fuel is first discharged from a reactor, its activity is very high, and it is generally stored under water for a period of time to allow decay of the shorter-lived nuclides. This reduces the decay heat output and shielding requirement to levels such that shipment to the reprocessing plant is practicable. A time of about 90 - 150 days is needed for this. Once the spent fuel is received at the reprocessing plant, it may be stored somewhat longer in order to insure a sufficiently low level of iodine 131 (half-life 8 days) and a sufficiently low external dose rate so that the plant shielding is adequate.

One of the first steps in determining pertinent input to the model is to decide what operations will be performed on the solid waste and what schedule will be followed for performing these operations. In establishing the sequence of operations, we followed the general outline given in NUREG 0116 (Bishop and Miraglia, 1976). Choice of intermediate steps in the sequence, which is shown in Figure 1.1-4, was based on available information and engineering judgment.

For the case of calcine it is assumed that either the spent fuel is held for five years or, if reprocessed sooner, the waste is held in liquid form until the total age out of the reactor is five years. At five years of age the waste is diluted with a sufficient amount of inert material to reduce the heat output of ten years of age to 3.5 kW per canister. It is then solidified and sealed in 12" (.305

m) diameter, internally-finned canisters.

For the glass, the canisters are of the same diameter, but are filled at one year and do not have fins.

For both waste forms, the waste is held in interim storage until ten years of age before shipment to a federal repository.

This can be represented graphically as follows:

Fig. A.1.1.4.2 Spent Fuel and SHLW Storage Schedule

SHLW

MANAGEMENT SEQUENCE

1.3.1 COMPARATIVE HAZARDS

Hazards associated with deep burial of solidified nuclear waste have been considered with reference to toxic elements in naturally occurring ore deposits (Tonnessen and Cohen, 1977). This problem is put into perspective by relating the hazard of a radioactive waste repository to that of naturally occurring geologic formations. The basis for comparison derives from a consideration of safe drinking water levels. Calculations for relative toxicity of fast breeder reactor (FBR) waste and light water reactor (LWR) waste in an underground repository are compared with the relative toxicity indices obtained for ore deposits of average concentration. Results indicate that nuclear waste toxicity decreases to levels below those of naturally occurring hazardous materials (See Figure 1.3-1).

Current discussion concerning disposal of hazardous high-level nuclear wastes often centers on the possibility of deep burial of the solidified waste in geologic formations. The principal hazard from such a practice would be the consequence of escape to the biosphere by leaching of radiotoxic material into water supplies. To place these hazards in perspective with other risks to society, it is beneficial to consider the health effects from release of naturally occurring toxic materials in geologic deposits. Concern has been expressed with regard to the extreme longevity of nuclear waste products, particularly the transuranic elements (See Figure 1.3-2). Plutonium-239, for example, has a half-life of 24,000 years. Therefore, the toxicity and potential hazard from these wastes will persist for a long time. The hazard associated with certain toxic, stable elements, by comparison, has been present for eons, since the beginning of geologic time. The potential for contamination of surface water supplies

by these hazardous minerals has always been present, depending on leach rates in underground deposits, and more recently as a result of mining and refining activity. Indeed, many occurrences of heavy metal concentrations in surface waters have their origins in geologic formations within the watershed areas. This fact has been valuable in prospecting for ores.

The toxic elements and radionuclides which we have considered are those for which standards have been set to protect drinking water supplies. These elements include: arsenic, barium, cadmium, chromium, mercury, lead, selenium, silver, thorium, and uranium. In the context of potential hazard resulting from toxic materials in geologic formations or from buried nuclear waste, the primary pathway for biological exposure is by ingestion. Therefore, the hazard resulting from the concentration of toxic materials in water supplies should provide the most rational basis for comparison.

This survey of naturally occurring hazardous substances in geologic formations is not an attempt to demonstrate the validity or acceptability of any type of waste disposal. Rather, it merely puts into perspective the hazards from geologic sources. The issue of disposal of high-level nuclear wastes will be resolved through both technical and sociological input. Individual attitudes may affect the final disposal plan. This discussion of naturally occurring hazards already in geologic formations may increase awareness of the hazardous elements which the public routinely accepts because of their familiarity, ubiquitous nature, and commercial value. Assuming harm or hazard to man to be the overriding consideration in the establishment of a waste disposal methodology, then the comparisons made here could prove useful.

References

- K. A. Tonnessen and J. J. Cohen, "Survey of Naturally-Occurring Hazardous Materials in Deep Geologic Formations : A Perspective on the Relative Hazard of Deep Burial of Nuclear Wastes," Lawrence Livermore Laboratory Rept UCRL-52199, Livermore, Calif., (Jan. 1977).

FIG. 12-1

Relative toxicity of nuclear waste over time, compared with that of average mineral ores of toxic elements.

Comparative Hazards

66010-2

Summary of Hazard-Index Components From a 1000-MWe PWR
(Wastes from 1 Metric Ton of Fuel at 34,000 MW_{TH} D)

1.3.3 RADIONUCLIDE RELEASE FUNCTION METHODOLOGY

In order for radionuclides to be released from a solid waste form in a sealed canister, two simultaneous conditions are necessary: the canister must be breached, and the radionuclides must leave the body of the solid waste. In the case of transportation accidents, the transport cask must also have a simultaneous breach in order for a release to occur. In this Appendix, mechanisms for each of these processes will be listed and described. Release functions for reference designs in particular accident scenarios are discussed in the corresponding Appendices. It must be noted that there are few experimental data available at present to check these methods, and they should therefore be regarded as "best estimates". The choices of mechanisms are based upon a consideration of the environments and accident conditions which seem reasonably plausible, as well as a general knowledge of the behavior of materials.

1.3.3.1 List of Transport Cask Breaching Mechanisms

- a. Fire leading to overheating of seal or other key components and failure by creep, differential thermal expansion, oxidation or melting
- b. Impact leading to failure of closure or other key components
- c. Puncture
- d. Quenching of hot cask in water leading to rupture by thermal shock
- e. Internal pressurization leading to bursting
- f. Improper sealing and leak checking

- g. Explosive attack
- h. Combinations of the above

1.3.3.2 List of Canister Breaching Mechanisms

- a. Overheating leading to rapid corrosion by waste, external oxidation in air, creep to rupture, or melting
- b. Impact leading to rupture
- c. Puncture
- d. Undetected leak or construction defect
- e. Corrosion by water
- f. Rapid quenching leading to rupture by thermal shock
- g. Internal pressurization leading to bursting
- h. Pull-out of lifting device
- i. External pressurization leading to buckling, collapse, and breaching
- j. Combinations of the above

1.3.3.3 List of Radionuclide Release Mechanisms

- a. Dissolution (leaching and etching)
- b. Volatilization
- c. Airborne particulate dispersion
- d. Melting and liquid flow

1.3.3.4 Discussion of Cask Breaching Mechanisms

- a. Fire leading to overheating of seal or other key components, and failure by creep, differential thermal

expansion, oxidation, or melting

A SHLW transport cask will have a very high total heat capacity (mass times specific heat). This fact makes it unlikely that a fire of relatively low heat output can supply sufficient energy to cause gross melting of the cask material. The most vulnerable casks to melting are probably those incorporating lead shielding, since the melting point of lead is 327°C (621°F.) compared to 1132°C (2070°F.) for uranium, 1400°C (2550°F.) for stainless steel, and 1536°C (2797°F.) for iron. (The formation of eutectics between steel and uranium is also important, but still occurs above the melting point of lead.) Even in the case of a leadshielded cask, gross melting of the steel shell of the cask is not likely. It is more likely that certain key components which are more sensitive to temperature would be affected first. Depending on the cask design, these may include the seal, the pressure relief system, fill and drain valves, or other components. Determination of the threshold for cask breaching due to fire therefore consists in identifying these components for the cask design in question, estimating the time and temperature combinations under which they will fail, and performing heat transfer calculations to determine if and when these conditions are exceeded in various fire environments. The most likely thermal failure processes are creep, differential thermal expansion, oxidation (including combustion), corrosion, and melting. Creep, oxidation, and corrosion are Arrhenius in nature. Melting occurs abruptly as temperature increases.

In order to simplify the determination of release thresholds, we have adopted the concept of "failure temperature". This is defined as the temperature at which the component of interest will no longer insure containment if held there for a time which is characteristic of the thermal conditions in the situation of interest. It is difficult to arrive at a precise value for this time, but an estimate can be made based upon the thermal time constant and the duration of the fire which is postulated. Fortunately, the failure temperature is not very sensitive to the time chosen, because of the abrupt or Arrhenius nature of most thermal failure mechanisms. For the Arrhenius processes, this can be understood as follows: Consider a process which could lead to failure of a component. The "progress toward failure" can be characterized as the product of the rate of the process and the time. The component will fail when this product reaches a critical value. The rate, in turn, has an Arrhenius temperature dependence:

$$R = R_0 e^{-E/kT},$$

where R_0 is a constant, E is the activation energy for the process, k is Boltzmann's constant, and T is the absolute temperature. Failure is thus reached when the product of the rate and the time ($R_0 t e^{-E/kT}$) reaches a critical value, where t is the time. One can therefore trade time for temperature, but a relatively small change in temperature will compensate for a large change in time. This fact is what gives validity

to the failure temperature concept, even when the time is not well-defined.

The most accurate way to determine failure temperatures is to perform thermal tests on duplicates of the actual components of interest. Where this is not possible, the next best approach is the use of test data for components which are made of the same materials and are similar in size and design. Sustained temperature limits for gasket materials are given in Machine Design (1964). Where data are not available, estimates can be made from a general knowledge of the processes involved in similar materials.

The size of the breach produced in a fire accident can be estimated from the design of the cask and a general knowledge of the behavior of materials at high temperatures.

In using heat transfer calculations to simulate actual fires, the simplest approach is to assume that the fire completely surrounds the cask, and presents it with a uniform ambient temperature and effective emissivity. In an actual fire accident, this may not be the case. In combining the release functions derived herein with the probability distributions for observed fires, we have therefore probably been conservative, since the actual heat input to the cask would probably be lower.

b. Impact

The prediction of cask behavior during an impact accident is a difficult problem. In the past, there have been three main approaches to solving it: full scale drop tests, scale model drop tests, and numerical simulations and calculations. Full scale testing is the most direct approach, but has been limited because of the high cost of shipping casks. Model studies can be done at lower cost, but the scaling must be verified. Numerical simulations of transient dynamic response of materials, involving plastic deformation and complex geometries, can be very costly in computer time, and verification of the models used is required.

In using any of these approaches, one must select the orientation of the cask at impact, the rigidity of the impact surface, and the velocity of impact. In past work, the most common procedure has been to attempt to meet the requirements of the Type B package standards (10CFR71) which are "A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected". In an actual impact accident in which a cask-carrying vehicle strikes a large object, the front of the vehicle would strike the object first. Depending on the size of the object and the materials out of which it was constructed, it would not be completely rigid, but would yield to some degree. The vehicle structure would deform plastically, absorbing a fraction of the kinetic energy of the cask which would depend on the strength of the tie-downs. In rail accidents, energy would also be dissipated

in derailment and disturbance of the rail bed. Assuming the initial velocity was sufficiently high, the cask itself might then impact the object, possibly trapping part of the vehicle structure between itself and the object, cushioning the deceleration. Its velocity at impact would be less than the initial vehicle velocity because of the processes just mentioned. The axis of the cask at its own impact would most likely be nearly parallel to the axis of the vehicle prior to impact with the object, because that is its orientation during shipment. It is not likely to rotate significantly between the time of impact of the vehicle and the time of cask impact because of its high moment of inertia and the short time available. However, the angle between the vehicle axis and the object will have a range of values for different accidents.

In the face of these problems, we have chosen to model the impact accidents by two approaches: use of the modified HONDO code for axial (end-on) impacts, and use of the analysis given by General Electric (1973) for corner impacts, both on unyielding surfaces. The corner impact calculations were used to derive the release functions because actual impacts seem more likely to occur closer to corner first than to end-on. Indeed, it is virtually impossible to achieve an exact end-on condition even in drop tests. This is a conservative approximation, however, since we have chosen the angle for which maximum damage is expected.

The rigidity of real objects has been accounted for partially

by using only the accident probabilities for "extremely rigid" objects (Clarke et al, 1976). This is believed to be conservative nevertheless, since even objects in this category (trains, bridges, etc.) are not completely rigid in a cask impact.

The velocity of the cask at impact was assumed to be given by $V_c(\text{mph}) = V_0^2(\text{mph}) - 3000$ where V_0 is the vehicle velocity at impact. This equation was derived by assuming that a constant amount of cask kinetic energy is used in deforming the vehicle structure, independent of impact velocity. A fit was then made to the two data points available from the recent truck impact tests at Sandia-Albuquerque (Yoshimura, 1977). For the 60 mph collision, the cask impact velocity was 27 mph. For the 84 mph collision, it was 62 mph. We have used the same equation for the rail impact accidents. Obviously, the factor to which we have assigned the value 3000 depends on the vehicle and the tie-downs used. Tie-down standards are currently under discussion (proposed ANSI standard N14.2 for trucks, draft ERDA standard RDT F8-12 for trains). In view of the many uncertainties in vehicle differences and tiedown strengths, we have chosen to base our model on the available data from the Sandia tests, recognizing that improvements in this area may be needed later.

c. Puncture

The analysis of puncture accidents indicated that there was

no significant risk. This conclusion was based upon the determination that the accident probability distribution function versus accident severity achieved a value of zero prior to any credible rupturing of the shipping cask. Hence, the integral of accident probability versus accident consequence over the range of credible accidents is zero.

In the event that either the general characteristics of the cask are modified or more severe accidents than presently analysed are determined to be credible, the analysis of puncture accidents will remain secondary to impact accidents. This relatively lower risk character is established by two considerations;

- 1) those SHLW characteristics that must be restricted for impact accidents include those for puncture accidents
- 2) The impact accident is felt to be a greater threat to the shipping cask than the puncture accident, because of its steel shell and thick walls. In the case of uranium shielded casks, the high density of the material also contributes to protection from penetration. It is unlikely that a penetrating object of sufficient strength to breach the cask will be present in an accident situation. Puncture of the relatively thin water

jacket is more likely, but this results only in making heat dissipation more difficult, thus raising the internal temperatures. This is less severe than the fire accident, which is also analyzed in more detail.

d. Quenching of hot cask in water leading to rupture by thermal shock

It is conceivable that an overheated shipping cask could be rapidly immersed in water. This might happen, for example, in a fire accident which was followed by attempts to extinguish the fire. In such cases, it is of interest to know whether rupture of the cask would occur by thermal shock. We have not performed a detailed analysis of this effect, since it was beyond the scope of the effort. However, it is noted that the outer shells of the casks are generally constructed of austenitic stainless steels, and it is believed that these materials possess sufficient strength and ductility to withstand such an event. The presence of the outer skin of the water jacket would reduce the likelihood of rapid cooling of the surface of the main body of the cask, as would formation of a water vapor layer due to boiling. It should also be noted that a release of the SHLW could only occur if the canisters were also breached in addition to the cask, and this does not appear to be likely for a cool quench.

e. Internal pressurization leading to bursting

Shipping casks are generally constructed with a pressure relief valve which limits the internal pressure to below the design limit for the cask. In the case of "dry" casks, that is, those without water inside the main body, it appears that for the reference waste forms considered, there will not be any materials present which are capable of generating significant pressures unless the temperatures exceed those which would produce failure by another mode first. Bursting of a cask containing SHLW therefore does not appear to be a significant threat.

f. Improper sealing and leak checking

It is assumed that a standardized procedure involving supervisory checks and tests for leak-tightness would be adopted for the processing of SHLW shipping casks. Nevertheless, it is conceivable that as a result of human error, a cask might not be properly sealed. We have not analyzed the probability of such an occurrence. However, it should be noted that even in the event that this should occur, there would be no release of SHLW unless some other event occurred which breached the canisters and provided a driving force for the SHLW release.

g. Explosive Attack/Adversary Action

This study has not included explosive attack.

h. Combinations of the above

This study has not included combinations of breaching mechanisms because of the limited scope of the effort. It should be noted that though such compound accidents are conceivable, they would be of lower probability than the individual accidents considered.

1.3.3.5 CANISTER BREACHING MECHANISMS - DISCUSSION

The relevant mechanisms and the rates at which they take place depend on the materials involved. The following discussion pertains to the reference calcine and glass waste forms in 304L stainless steel canisters.

1.3.3.5.1 MECHANISMS INDUCED BY OVERHEATING

Canisters could be overheated by internally-generated decay heat or externally applied heat from fire accidents or adjacent canisters as in a storage loss-of-coolant accident. Several mechanisms come into play at high temperatures. Their rates are strongly temperature-dependent. Failure of a canister could result from a single mechanism or a combination of them.

a. CORROSION OF THE CANISTER BY THE HEATED WASTE

Stainless steels protect themselves from corrosion by formation of an oxide layer at the surface. In order for rapid general corrosion to take place, this layer must be removed. The waste is not likely to be able to do this until it is molten. Then the oxide can dissolve readily in the melt and its concentration can be lowered at the interface by convection in the waste.

In the case of calcine, the gross melting point is quite high, near 1400°C. Since the melting point of the stainless steel is also about 1400°C, general corrosion by calcine will probably not be the limiting process. Intergranular corrosion of calcine on steel has not been thoroughly studied. However, Ma ya and Busch (1973) have reported measurements of cesium oxide corrosion of 304L stainless steels. They found kinetics which were linear in time, with a penetration rate given by

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

where temperature is expressed in Kelvins. At this rate, it would require temperature well above the melting point of the steel to penetrate a few mm in a few hours. Intergranular corrosion is therefore also not regarded as the limiting mechanism for calcine in 304L canisters.

In the case of glass waste in 304L canisters, general corrosion data have been reported by Slate and Maness (1977). A fit to these data produced the equation

$$k \text{ (mm/hr)} = 9.38 \times 10^{17} \exp (-61800/T),$$

where T is expressed in Kelvins. Applying this equation to a canister thickness of 9.5 mm, a temperature of 1306°C is required to produce failure by this mechanism alone in one hour, a temperature of 1219°C is required to produce failure in 10 hours, and a temperature of 1140°C is required for 100-hour failure.

b. EXTERNAL OXIDATION OF THE CANISTER IN AIR

The depth of oxidation of 304L stainless steel in air is taken to be

$$P(t,T) = 5.25 \times 10^9 \sqrt{t} \exp (-34,111/T),$$

where P is in mm, t is the time in hours, and T is the temperature in Kelvins. This equation is a fit to the curve given by Angerman and Rankin (1977), assuming a parabolic dependence on time. Their curve is based on data from Miller et al (1944). This equation is roughly compatible with the rate of 35

mm/yr at 1075°C reported by Slate and Maness (1977) if account is taken of the parabolic time dependence.

Applying this equation to a 9.5 mm-thick canister containing calcine, one finds that the temperature to cause failure in one hour would be above the melting point of the steel. Failure in ten hours would require a temperature of 1330°C. This indicates that oxidation would bring about failure at slightly less than the temperature at which melting would occur for times of this order.

For the glass case, oxidation would be capable of penetrating about 2 mm at the temperatures and time scales in which corrosion by the glass could cause failure. In this case, then, it appears that oxidation could contribute to failure, but would not be the dominant process.

c. CREEP TO RUPTURE

Creep begins to be measurable in 304L for stresses below the room temperature yield stress at about 500°C. At 1000°C, 304L is capable of supporting stresses of a few ksi (a few tens of MPa) for a few hours. At 1200-1300°C, its strength becomes quite low (Hickey, 1973).

In the case of calcine-containing canisters, the stresses in the canister wall will result primarily from gravity and internal pressurization. Both are expected to be quite low for the reference canister and waste. (Pressurization is discussed below.) It therefore appears that creep-rupture would not result in failure until a temperature near the melting point was reached.

For glass-containing canisters, the canister wall is highly stressed at low temperatures because of differential thermal contraction after solidification. However, when the wall is heated into the temperature range where creep becomes important, the stress is relieved by differential thermal expansion and softening of the glass. Stresses due to gravity and internal pressurization are again expected to be low.

It therefore appears that creep will not be the dominant failure mode for glass-containing canisters, and will only come into play near the melting point for calcine-containing canisters. We have not performed a detailed creep analysis for various canister orientations, but this conclusion appears to be valid at our present level of understanding.

d. MELTING

For 304L stainless steel, melting begins at about 1400°C. If the canister were heated continuously, and were able to survive the other processes discussed above, it would fail by melting at this temperature.

1.3.3.5.2 IMPACT LEADING TO RUPTURE

If a full canister is impacted, the probability of breaching, the character of the breach, and the size of the breach will depend on the velocity at impact, the rigidity of the impacted surface, the design of the canister, the canister and waste material (including its prior thermal and mechanical history), the temperature of the canister, and the location of impact.

The only experimental data of which we are aware are those of Smith and Ross (1975). In their work, all of the flat-bottomed 304L stainless steel canisters dropped from a height of 9.15 m were unbreached. About half were breached for heights equivalent to a 45 mph impact (greater than 20 meters). The reference design canister has a rounded bottom, and is expected to be even less vulnerable to impact.

For 304L the fracture is expected to be ductile for the temperatures and strain rates of interest. In this case, the breach will normally amount to only a

small crack (say, 10^{-3} times the surface area) (Munich and Poss, 1975) for impacts equivalent to up to about a 20-meter drop.

1.3.3.5.3 PUNCTURE

If a bare canister were dropped onto a relatively sharp object rather than a flat surface, it would be more likely to be breached. It is difficult to assess the likelihood of such an object being present, or to characterize its shape in the absence of more detailed knowledge of the facilities and equipment to be used. However, it seems possible to limit the height to which a canister may be lifted and/or to design the facility so that such objects are not present.

1.3.3.5.4 UNDETECTED LEAK OR CONSTRUCTION DEFECT

This type of breach would be small, say 10^{-4} times the canister area. It would require failure of the quality assurance system for this to occur. This has not been analyzed in this study.

1.3.3.5.5 CORROSION BY WATER

Corrosion of the canister by storage pool water was not analyzed in detail because it, by itself, will

not lead to exposure to the public. Occupational dose has not been considered in this study.

1.3.3.5.6 RAPID QUENCHING AND THERMAL SHOCK RUPTURE

Rupture does not seem to be very likely. As a matter of fact, rapid quenching may be a part of the glass waste processing. If the canister survives quenching during processing, it will likely survive it later. Steels are generally more ductile as the temperature increases and tend not to fracture in a brittle manner under these conditions.

1.3.3.5.7 INTERNAL PRESSURIZATION LEADING TO BURSTING

This could be caused by helium buildup from alpha-decay of actinides and (n,α) reactions on boron (if present), radiolysis and thermal decomposition of waste, transmutations to gaseous species, or volatilization of any water accidentally added to the filled, baked-out canister prior to closure welding, as well as by increase of the pressure of confined gases as a result of heating (perfect gas law). For the example waste forms considered, both of which have been heated to at least 900°C during processing, thermal decomposition will not be a problem up to temperatures where the canister would fail by corrosion or melting (Gray, 1976). For UO_2 waste,

the helium buildup is not important (Mendel, 1976). Radiolysis is not significant for the examples considered. If organic, nitrated, or hydrated materials were proposed, radiolysis and thermal decomposition could be a serious problem. The only gaseous transmutation product appears to be iodine from tellurium decay, and the quantity is too small to be important. Perfect gas law pressure increase is also insignificant for the temperature changes expected between sealing and subsequent events.

It appears that pressure buildup will be insignificant for the waste forms considered unless water is accidentally added before sealing. In this case, the canister could rupture at a temperature above 100°C, the actual value depending on material and design.

1.3.3.5.8 PULL-OUT OF LIFTING DEVICE

The top of a canister must be fitted with lifting devices for handling. If not properly designed, it is conceivable that failure of such a device could breach the canister and result in a radionuclide release. The likelihood and extent of such a release depends strongly on canister design. The reference canister is assumed to be designed to avoid this possibility.

1.3.3.5.9 EXTERNAL PRESSURIZATION LEADING TO BUCKLING, COLLAPSE AND BREACHING

When submerged in a storage pool, a canister will experience increased external pressure. For canisters of reasonable thickness and pools of reasonable depth, this problem is insignificant, because of the low stresses which result.

1.3.3.5.10 COMBINATIONS OF THE ABOVE

Combined effects have not been considered, except for the case of pressurization resulting from overheating, as discussed above. The probability of occurrence of other combined effects is expected to be lower than that for the events considered.

1.3.3.6 RADIONUCLIDE RELEASE MECHANISMS - DISCUSSION

1.3.3.6.1 DISSOLUTION

This process includes both leaching and corrosion or etching of the solid waste form by a solvent, in this case, water. Leaching involves selective diffusion of radionuclides from a depth in the matrix of the material, while corrosion or etching is related to removal of a portion of the entire matrix, including the radionuclides contained in it. The mechanisms involved are complex and not thoroughly understood. The important variables are the chemical properties of the solid matrix, the chemical properties of the solvent, and the temperature at the interface between them. The flow velocity of the solvent and the state of subdivision of the solid also influence the dissolution. For a system with fixed chemical composition, temperature, and flow velocity, one can usually fit the available data for a single radionuclide with an expression of the type

$$L = A t^{-1/2} e^{-H_1/RT} + B e^{-H_2/RT} + C ,$$

where L is the dissolution rate in grams per square meter-second, t is the time after the solvent contacts the solid in seconds, A , B , and C are constants, H_1 is the activation enthalpy associated

with leaching in joules per mole, U_2 is the activation enthalpy associated with corrosion in joules per mole, R is the gas constant, 8.314 joules per mole-kelvin, and T is the temperature at the interface in Kelvins.

The rate of release of a given radionuclide to the solvent is then given by the following equation, which assumes that the concentration of solutes in the solvent is constant:

$$S = L(t) A_L C_p,$$

where S is the rate of release in becquerels per second-cubic meter-solid, $L(t)$ is the dissolution rate at time t in grams per square meter-second, A_L is the effective area of solid exposed in square meters per cubic meter-solid, and C_p is the concentration of radioactivity in the solid for the particular nuclide, in becquerels per gram.

Unfortunately, there are insufficient data to define uniquely all the parameters in the equation for L for all the waste forms and all the nuclides of interest. In addition, the flow properties and the effective area A_L are not well defined in the situations of interest. Accordingly, some approximations must be made. The following appears to be the best procedure at our present state of understanding:

area A_L are not well defined in the situations of interest. Accordingly, some approximations must be made. The following appears to be the best procedure at our present state of understanding:

1.3.3.6.1.2 EFFECTIVE AREA

The effective area (A_L) for dissolution depends on the condition of the waste, the canister, and the leaching medium. If the canister is not breached, A_L is equal to zero. The following modes of canister breaching will be discussed separately: corrosion by water in geological storage, overheating and corrosion by waste, and impacting during an accident.

a. CORROSION OF THE CANISTER IN GEOLOGICAL STORAGE

Behavior of the canister during geological storage will depend on a complex interplay of chemical, thermal, and radiolytic processes. At this stage of the study, we have assumed that the canister fails immediately after the repository is backfilled and sealed. This is admittedly a conservative assumption. In this case, A_L for the calcine would be equal to about 10^6 square meters per cubic meter of solid.

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For glass, A_L would be equal to $\frac{20 (r+h)}{r}$, where r and h are the radius and length, respectively, in metres. This is simply equal to ten times the outer surface area of the glass block, the additional factor of ten accounting for fractures.

b. OVERHEATING AND CORROSION OF CANISTER BY WASTE, FOLLOWED BY CONTACT WITH WATER

If the canister remained in contact with water, it is not likely that the waste would corrode through it, because of the low temperature. This section will deal with the case of corrosion by overheating out of water (for example, by internal heat generation or fires) and subsequent contact with water (for example, by attempts to cool an overheated canister or to put out a fire).

If the temperature were high enough for the canister to be breached in this way, the glass waste form would be molten and would flow out until the surface area increased to the point that sufficient cooling occurred to solidify the waste. The calcine would melt at about the same temperature that canister failure occurred, so it would

likely slump down into a pile.

For the calcine, we use the starting value 10^6 square meters per cubic meter solid, and for glass we use $\frac{20(r+h)}{rh}$. The calcine value decreases substantially if melting occurs.

c. IMPACTING THE CANISTER IN AN ACCIDENT,
FOLLOWED BY CONTACT WITH WATER

For impacts resulting from glass-containing canister velocities less than 18 meters per second, we use $A_L=0$, because the canister will probably not be breached (Smith and Ross, 1975). For impacts at velocities between 18 and 36 meters per second, breaching occurs in the form of small cracks near the point of impact. Higher velocities for a bare canister do not seem probable. In the case of a calcine canister, breaching is more probable. For calcine, we use $A_L = 10^3$ square meters per cubic meter solid for still water and 10^4 square meters per cubic meter solid for flowing water.

For a glass canister breached by impact, the effective area results from a combination of the size of the crack and the increased

surface area of the glass resulting from the fracture. The area of the crack is approximately 10^3 of the canister surface area, and the glass surface area increases approximately by a factor of 2 at an impact velocity of 8 meters per second and by a factor of 10 at 19 meters per second (Smith and Ross, 1975). For a glass-containing canister that is impact-breached, we use $A_L = \frac{0.2 (r+h)}{rh}$ for still water and $\frac{2 (r+h)}{rh}$ for flowing water.

1.3.3.6.2 VOLATILIZATION

This process involves evaporation of chemical species from the waste at elevated temperatures. In order for this to occur, the nuclide of interest normally must first diffuse to the surface and then acquire sufficient energy to break its bonds and evaporate. A particular nuclide may evaporate as the element or as one of its oxides, depending on the partial pressures of oxygen and water in the surrounding gas phase, normally air in this context.

The rate of release will depend strongly on the temperature and the condition of the

canister. If the canister is not breached, obviously no release will occur. The most likely modes of breaching the canister are the same as those described above. These modes can be divided into two groups: those which open up only a small crack in the canister, and those which expose a large fraction of the canister surface area.

The mechanism of volatilization release through a crack depends on whether the combined vapor pressures of the species in the canister are in excess of atmospheric pressure or not. If the total vapor pressure is below atmospheric pressure, the radionuclides can only escape by gaseous diffusion through the crack, and a state approaching thermodynamic equilibrium will exist inside the can. (Strictly speaking, since calcines and glasses are amorphous, non-equilibrium materials, the concept of equilibrium vapor pressure has no meaning for them in the solid state.) If the total is above atmospheric pressure, the gases will undergo viscous flow driven by the pressure difference. This will still be "near-equilibrium", but at a higher pressure. Calculation of the release by these

mechanisms and of the temperature at which the total vapor pressure exceeds atmospheric pressure requires a knowledge of the vapor pressures as a function of the temperature for the chemical species present.

For crystalline materials in thermodynamic equilibrium, the equilibrium vapor pressures would depend on the partial pressures of oxygen and water in the ambient gas phase and on the (chemical) activity coefficients of the species of interest in the solid. These activity coefficients in turn would depend on the chemical composition of the solid. Unfortunately, because of the non-equilibrium nature of the waste solids, this approach is not very fruitful.

For the case of a large fraction of the canister area being exposed and rapid movement of the ambient air, the behavior is closer to free evaporation of material from the surface, without back condensation. In this case, the process appears to be limited by diffusion of the species of interest within the solid in the case of calcine, and by a surface release mechanism for the glass, which is molten at the

temperatures of interest (Gray, 1976).

In view of the lack of complete data, the best current approach is to make an empirical fit to existing data and to apply this fit to both the above cases. This is done in the Appendices devoted to particular scenarios.

1.3.3.6.3 AIRBORNE PARTICULATE DISPERSION

This refers to spreading by air currents of particulate matter containing radionuclides. The respirable fraction of the particulates (less than 10- μ diam) is particularly important. As in the other processes, canister breaching is a prerequisite for radionuclide release.

For glass, particulate dispersion will be quite small except in the case of an impact that fractures the glass. In general, it requires a high concentration of stress to break glass into fine particles, so it can be anticipated that the fines produced will be located near the point of impact. In order for release to occur, the canister must rupture at a location adjacent to the

region containing the fines. Since stress concentration is responsible for both, this is likely. The driving force for dispersion would then have to come from the mechanical energy of the impact or the motion of ambient air. It is difficult to estimate these effects in the absence of a detailed canister design and a more complete understanding of the phenomena involved. The test approach appears to be a judgment based on the data of Smith and Ford (1971).

1.3.3.6.4 MELTING AND LIQUID FLOW

The calofine melts at about 2400°C. Such a calofine temperature is difficult to maintain in an accident configuration, and volatilization would probably dominate in this case, anyway.

The glass softens at 680°C and flows well at about 600°C. If it melted in an accident, it could spread on the floor or ground until the larger surface area cooled it to a solid. The amount of spreading would be minimal. Melting as a release mechanism appears to be negligible.

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1.3.4 BIOLOGICALLY SIGNIFICANT RADIONUCLIDES IN HIGH-LEVEL WASTE

The relative composition of high-level wastes varies greatly as a function of time because of the different decay rates of the constituent radionuclides. Although the waste includes an abundant variety of radioactive species, at any given time only a relative few are potentially significant as biological hazards.

1.3.4.1 SOURCE TERM

A list of the major radioactive fission products and actinides that have intermediate or long half-lives is given in Table _____. Source terms are listed for these nuclides as a function of time after fuel irradiation. Input to the table has been derived from two sources: for post-irradiation times up to 100 years, data from the Siting of Fuel Reprocessing Plants (1970) have been used. For longer times, data have been taken from Gera (1970).

The following assumptions pertain to the data in Table _____.

Fuel has been irradiated for 3.3×10^4 MWe/Mg of uranium at a thermal efficiency of 35.4%.

The source of ^{129}I is decay of ^{129}Te , $^{129\text{m}}\text{Te}$, and 0.1% of the ^{129}I present in the waste when it is dissolved. The actual quantity of ^{129}I present in high-level waste will be dependent

on solidification time (assumed to be 150 da). If waste is not solidified until one year after irradiation, the amount present will be about one-half that shown in Table ____.

1.3.4.2 BIOLOGICALLY SIGNIFICANT RADIONUCLIDES: AIRBORNE

Tables ____ and ____ list the most biologically significant radioactive species evaluated by means of toxicity indices for airborne radioactivity. The toxicity index for each nuclide represents the volume of air that would have to be mixed with that quantity of the nuclide formed during the production of one megawatt electric year of power in order for the radionuclide concentration to be no greater than the maximum permissible concentration in air (MPC_a). Table ____ gives toxicity indices calculated for the mixture of radionuclides existing one year after fuel irradiation; only those nuclides that have toxicity indices in the two orders of magnitude below that of the most toxic nuclide (which is ⁹⁰Sr for the fission products and ²⁴⁴Cm for the actinides) are included in the list. Table ____ contains the same information as Table ____ calculated for the mixture of radionuclides existing 10 years after fuel irradiation. Dose conversion factors have been calculated for all nuclides shown in Tables ____ and ____, results are given in Appendix _____. Note that biologically significant airborne nuclides have not been identified for post-irradiation times greater than 10 years. Since proposed regulations require that all high-level waste be placed in an underground repository no more than 10 years after it has been

generated, there is no credible mechanism whereby such waste can become airborne at times greater than 10 years after fuel irradiation.

It is interesting that all of these biologically significant airborne radioactive species have half-lives less than 10 000 years.

1.3.4.3 BIOLOGICALLY SIGNIFICANT RADIONUCLIDES: WATERBORNE

Tables ____ through ____ list the most biologically significant radioactive species, which were evaluated using toxicity indices for waterborne radioactivity. These indices are calculated in the same way as the airborne radioactivity indices except that the appropriate MPC_w values have been taken from 10 CFR 20.

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Table A-1. Source terms of biologically significant nuclides
(all units are Ci/MWe-yr).

Nuclide	150 da	1 yr	10 yr	10 ² yr	10 ³ yr	10 ⁴ yr
⁸⁹ Sr	3.0×10^3	1.7×10^2	0	0	0	0
⁹⁰ Sr	2.4×10^3	2.4×10^3	1.9×10^3	2.0×10^2	4.7×10^{-8}	0
⁹⁰ Y	2.4×10^3	2.4×10^3	1.9×10^3	2.0×10^2	4.7×10^{-8}	0
⁹¹ Y	5.0×10^3	3.9×10^2	0	0	0	
⁹³ Zr	5.9×10^{-2}	5.9×10^{-2}	5.9×10^{-2}	5.9×10^{-2}	5.9×10^{-2}	5.9×10^{-2}
^{93m} Nb	1.2×10^{-3}	2.9×10^{-3}	2.4×10^{-2}	5.9×10^{-2}	5.9×10^{-2}	5.9×10^{-2}
⁹⁵ Zr	8.6×10^3	8.7×10^2	0	0	0	0
⁹⁵ Nb	1.6×10^4	1.9×10^3	5.9×10^{-7}	0	0	0
⁹⁹ Tc	4.4×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	4.4×10^{-1}	4.4×10^{-1}
¹⁰⁶ Ru	1.3×10^4	8.5×10^3	1.7×10^1	0	0	0
¹⁰⁶ Rh	1.3×10^4	8.5×10^3	1.7×10^1	0	0	0
¹²⁵ Sb	2.5×10^2	2.2×10^2	2.2×10^1	0	0	0
¹²⁶ Sn	1.7×10^{-2}	1.7×10^{-2}	1.7×10^{-2}	1.7×10^{-2}	1.7×10^{-2}	1.7×10^{-2}
¹²⁹ I	2.4×10^{-6}	2.4×10^{-6}	2.4×10^{-6}	2.4×10^{-6}	2.4×10^{-6}	2.4×10^{-6}
¹³⁴ Cs	6.7×10^3	5.5×10^3	2.6×10^2	0	0	0
¹³⁷ Cs	3.3×10^3	3.3×10^3	2.7×10^3	3.3×10^2	3.1×10^{-7}	0
¹⁴⁴ Ce	2.4×10^4	1.4×10^4	4.7×10^0	0	0	0
¹⁴⁷ Pm	3.1×10^3	2.7×10^3	2.5×10^2	1.1×10^{-8}	0	0
¹⁵⁴ Eu	2.1×10^2	2.1×10^2	1.4×10^2	2.9×10^0	0	0
²¹⁰ Pb	0	0	0	2.3×10^{-8}	2.2×10^{-6}	8.8×10^{-5}
²¹⁰ Po	0	0	0	2.3×10^{-8}	2.2×10^{-6}	8.8×10^{-5}
²²⁶ Ra	0	0	3.4×10^{-9}	3.5×10^{-8}	2.2×10^{-6}	8.8×10^{-5}
²²⁹ Th	0	0	1.3×10^{-9}	2.3×10^{-8}	2.2×10^{-6}	1.7×10^{-4}
²³⁰ Th	6.6×10^{-7}	6.6×10^{-7}	6.6×10^{-7}	1.0×10^{-6}	1.2×10^{-5}	1.1×10^{-4}

A-1-1-1
Table A-1 (Contd.)

Nuclide	150 da	1 yr	10 yr	10 ² yr	10 ³ yr	10 ⁴ yr
²³¹ Pa	7.7×10^{-7}	7.7×10^{-7}	7.7×10^{-7}	7.7×10^{-7}	8.1×10^{-7}	1.2×10^{-6}
²³³ U	4.6×10^{-7}	4.6×10^{-7}	4.6×10^{-7}	4.6×10^{-6}	4.8×10^{-5}	4.9×10^{-4}
²³⁷ Np	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	1.1×10^{-2}	1.2×10^{-2}	1.2×10^{-2}
²³⁸ Pu	1.1×10^0	2.5×10^0	3.2×10^0	1.6×10^0	3.8×10^{-3}	0
²³⁹ Pu	5.2×10^{-2}	5.2×10^{-2}	5.2×10^{-2}	5.2×10^{-2}	6.4×10^{-2}	1.3×10^{-1}
²⁴⁰ Pu	7.6×10^{-2}	8.1×10^{-2}	1.4×10^{-1}	2.7×10^{-1}	2.5×10^{-1}	9.9×10^{-2}
²⁴¹ Pu	1.8×10^1	1.7×10^1	1.0×10^1	1.5×10^{-1}	9.8×10^{-3}	4.6×10^{-3}
²⁴¹ Am	5.4×10^0	5.4×10^0	5.5×10^0	5.1×10^0	1.1×10^0	4.6×10^{-3}
²⁴³ Am	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.4×10^{-1}	5.0×10^{-1}	2.3×10^{-1}
²⁴² Cm	4.7×10^2	1.9×10^2	2.2×10^{-1}	1.5×10^{-1}	2.4×10^{-3}	0
²⁴⁴ Cm	7.8×10^1	7.6×10^1	5.4×10^1	1.7×10^0	0	0

Table A-2^a Inventory of biologically significant nuclides if released to air at 1 year after irradiation of 32 MWe·yr/Mg.

Fission products				
Nuclide	T 1/2	MPC _a ($\frac{\text{Ci}}{\text{m}^3}$)	Q ($\frac{\text{Ci}}{\text{MWe}\cdot\text{yr}}$)	Toxicity index ($\frac{\text{m}^3}{\text{MWe}\cdot\text{yr}}$)
⁸⁹ Sr	53 da	3×10^{-10}	1.7×10^2	6×10^{11}
⁹⁰ Sr	28 yr	3×10^{-11}	2.4×10^3	8×10^{13}
⁹⁰ Y	64 hr	3×10^{-9}	2.4×10^3	8×10^{11}
⁹¹ Y	59 da	1×10^{-9}	3.9×10^2	4×10^{11}
⁹⁵ Zr	66 da	1×10^{-9}	8.7×10^2	9×10^{11}
⁹⁵ Nb	35 da	3×10^{-9}	1.9×10^3	6×10^{11}
¹⁰⁶ Ru	1.0 yr	2×10^{-10}	8.5×10^3	4×10^{13}
¹⁰⁶ Rh ^a	30 sec	1×10^{-9}	8.5×10^3	9×10^{12}
¹²⁵ Sb	2.7 yr	9×10^{-10}	2.2×10^2	2×10^{11}
¹³⁴ Cs	2.0 yr	4×10^{-10}	5.5×10^3	1×10^{13}
¹³⁷ Cs	30 yr	5×10^{-10}	3.3×10^3	7×10^{12}
¹⁴⁴ Ce	280 da	2×10^{-10}	1.4×10^4	7×10^{13}
¹⁴⁷ Pm	2.6 yr	3×10^{-9}	2.7×10^3	9×10^{11}
¹⁵⁴ Eu	16 yr	2×10^{-10}	2.1×10^2	1×10^{12}
Fission product total			5.1×10^4	2×10^{14}
Actinides				
²³⁸ Pu	86 yr	7×10^{-14}	2.5	4×10^{13}
²⁴⁰ Pu	6.6×10^3 yr	7×10^{-14}	0.081	1×10^{12}
²⁴¹ Pu	13 yr	3×10^{-12}	17.	6×10^{12}
²⁴¹ Am	460 yr	2×10^{-12}	5.4	3×10^{13}
²⁴³ Am	8×10^3 yr	2×10^{-13}	0.54	3×10^{12}
²⁴² Cm	160 da	4×10^{-12}	190	5×10^{13}
²⁴⁴ Cm	18 yr	3×10^{-13}	76.	2×10^{14}
Actinide totals			2.9×10^2	3×10^{14}
Grand totals			5.1×10^4	5×10^{14}

^aIn equilibrium with ¹⁰⁶Ru, MPC₁₇ is an estimate.

A-3
 Table A-3. Inventory of biologically significant nuclides if released
 to air at 10 years after irradiation of 32 MWe·yr/Mg.

<u>Fission products</u>				
Nuclide	T _{1/2}	MPC _a $\left(\frac{\text{Ci}}{\text{m}^3}\right)$	Q $\left(\frac{\text{Ci}}{\text{MWe}\cdot\text{yr}}\right)$	Toxicity index $\left(\frac{\text{m}^3}{\text{MWe}\cdot\text{yr}}\right)$
⁹⁰ Sr	28 yr	3×10^{-11}	1.9×10^3	6×10^{13}
⁹⁰ Y	64 hr	3×10^{-9}	1.9×10^3	6×10^{11}
¹³⁴ Cs	2.0 yr	4×10^{-10}	2.6×10^2	7×10^{11}
¹³⁷ Cs	30 yr	5×10^{-10}	2.7×10^3	5×10^{12}
Fission product totals			6.8×10^3	7×10^{13}
<u>Actinides</u>				
²³⁸ Pu	86 yr	7×10^{-14}	3.2	5×10^{13}
²⁴⁰ Pu	6.6×10^3 yr	6×10^{-14}	0.14	2×10^{12}
²⁴¹ Am	460 yr	2×10^{-13}	5.5	3×10^{13}
²⁴³ Am	8×10^3 yr	2×10^{-13}	0.54	3×10^{12}
²⁴⁴ Cm	18 yr	3×10^{-13}	54	2×10^{14}
Actinide totals			63	3×10^{14}
Grand totals			6.8×10^3	4×10^{14}

Table 4- Inventory of biologically significant nuclides if released to water at 1 year after irradiation at 32 MWe/Mg.

<u>Fission products</u>				
Nuclide	T 1/2	MPC _w ($\frac{Ci}{m^3}$)	Q ($\frac{Ci}{MWe \cdot yr}$)	Toxicity index ($\frac{m^3}{MWe \cdot yr}$)
⁸⁹ Sr	53 da	3×10^{-6}	1.7×10^2	6×10^7
⁹⁰ Sr	28 yr	3×10^{-7}	2.4×10^3	8×10^9
⁹⁰ Y	64 hr	2×10^{-5}	2.4×10^3	1×10^8
⁹¹ Y	59 da	3×10^{-5}	3.9×10^2	1×10^7
⁹⁵ Zr	66 da	6×10^{-5}	8.7×10^2	1×10^7
⁹⁵ Nb	35 da	1×10^{-4}	1.9×10^3	2×10^7
¹⁰⁶ Ru	1.0 yr	1×10^{-5}	8.5×10^3	9×10^8
¹⁰⁶ Rh ^a	30 sec	1×10^{-4}	8.5×10^3	9×10^7
¹³⁴ Cs	2.0 yr	9×10^{-6}	5.5×10^3	6×10^8
¹³⁷ Cs	30 yr	2×10^{-5}	3.3×10^3	2×10^8
¹⁴⁴ Ce	280 da	1×10^{-5}	1.4×10^4	1×10^9
¹⁴⁷ Pm	2.6 yr	2×10^{-4}	2.7×10^3	1×10^7
¹⁵⁴ Eu	16 yr	2×10^{-5}	2.1×10^2	1×10^7
Fission product totals			5.1×10^4	1×10^{10}
<u>Actinides</u>				
²³⁸ Pu	86 yr	5×10^{-6}	2.5	5×10^5
²³⁹ Pu	2.4×10^4 yr	5×10^{-6}	0.052	1×10^4
²⁴⁰ Pu	6.6×10^3 yr	5×10^{-6}	0.081	2×10^4
²⁴¹ Pu	13 yr	2×10^{-4}	17.0	9×10^4
²⁴¹ Am	460 yr	4×10^{-6}	5.4	1×10^6
²⁴³ Am	8×10^3 yr	4×10^{-6}	0.54	1×10^5
²⁴² Cm	160 da	2×10^{-5}	190	9×10^6
²⁴⁴ Cm	18 yr	7×10^{-6}	76	1×10^7
Actinide totals			2.9×10^2	2×10^7
Grand totals			5.1×10^4	1×10^{10}

^aIn equilibrium with ¹⁰⁶Ru, MPC_a is an estimate.

Table A-3. Inventory of biologically significant nuclides if released to water at 10 years after irradiation at 32 MWe-yr/Mg.

<u>Fission products</u>				
Nuclide	T 1/2	MPC _w $\left(\frac{\text{Ci}}{\text{m}^3}\right)$	Q $\left(\frac{\text{Ci}}{\text{MWe-yr}}\right)$	Toxicity index $\left(\frac{\text{m}^3}{\text{MWe-yr}}\right)$
⁹⁰ Sr	28 yr	3×10^{-7}	1.9×10^3	6×10^9
⁹⁰ Y	64 hr	2×10^{-5}	1.9×10^3	9×10^7
¹³⁴ Cs	2.0 yr	9×10^{-6}	2.6×10^2	3×10^7
¹³⁷ Cs	30 yr	2×10^{-5}	2.7×10^3	1×10^8
Fission product total			6.8×10^3	6×10^9
<u>Actinides</u>				
²³⁸ Pu	86 yr	5×10^{-6}	3.2	6×10^5
²⁴¹ Am	460 yr	4×10^{-6}	5.5	1×10^6
²⁴³ Am	8×10^3 yr	4×10^{-6}	0.54	1×10^5
²⁴⁴ Cm	18 yr	7×10^{-6}	54	8×10^6
Actinide totals			63	1×10^7
Grand totals			6.9×10^3	6×10^9

Table A-6. Inventory of biologically significant nuclides if released to water at 100 years after irradiation of 32 MWe·yr/Mg.

<u>Fission products</u>				
Nuclide	T 1/2	MFC _w ($\frac{\text{Ci}}{\text{m}^3}$)	Q ($\frac{\text{Ci}}{\text{MWe}\cdot\text{yr}}$)	Toxicity index ($\frac{\text{m}^3}{\text{MWe}\cdot\text{yr}}$)
⁹⁰ Sr	28 yr	3×10^{-7}	2.0×10^2	7×10^8
⁹⁰ Y	64 hr	2×10^{-5}	2.0×10^2	1×10^7
¹³⁷ Cs	30 yr	2×10^{-5}	3.3×10^2	2×10^7
Fission product totals			7.3×10^2	7×10^8
<u>Actinides</u>				
²³⁸ Pu	86 yr	5×10^{-6}	1.6	3×10^5
²⁴¹ Am	460 yr	4×10^{-6}	5.1	1×10^6
²⁴³ Am	8×10^3 yr	4×10^{-6}	0.54	1×10^5
²⁴⁴ Cm	18 yr	7×10^{-6}	1.7	2×10^5
Actinide totals			8.9	2×10^6
Grand totals			7.4×10^2	7×10^8

Table 1-7a. Inventory of biologically significant nuclides if released to water at 1000 years after irradiation of 32 MWe-yr/Mg.

Fission products				
Nuclide	T 1/2	MPC _w $\left(\frac{Ci}{m^3}\right)$	Q $\left(\frac{Ci}{MWe \cdot yr}\right)$	Toxicity index $\left(\frac{n^3}{MWe \cdot yr}\right)$
⁹³ Zr	1.5×10^6 yr	8×10^{-4}	5.9×10^{-2}	7×10^1
^{93m} Nb	14 yr	4×10^{-4}	5.9×10^{-2}	1×10^2
⁹⁹ Tc	2.1×10^5 yr	3×10^{-4}	4.5×10^{-1}	2×10^3
¹²⁶ Sn ^a	1×10^5 yr	2×10^{-5}	1.7×10^{-2}	9×10^2
¹²⁹ I	1.7×10^7 yr	6×10^{-8}	2.4×10^{-6}	4×10^1
Fission product totals			5.9×10^{-1}	3×10^5
Actinides				
²³⁷ Np	2.1×10^6 yr	3×10^{-6}	1.2×10^{-2}	4×10^3
²³⁹ Pu	2.4×10^4 yr	5×10^{-6}	6.4×10^{-2}	1×10^4
²⁴⁰ Pu	6.6×10^3 yr	5×10^{-6}	2.5×10^{-1}	5×10^4
²⁴¹ Am	4.6×10^2 yr	4×10^{-6}	1.1×10^0	3×10^5
²⁴³ Am	8.0×10^3 yr	4×10^{-6}	5.0×10^{-1}	1×10^5
Actinide totals			1.9	5×10^5
Grand totals			2.5	5×10^5

^aMPC assumed to be equal to the smallest of all other isotopes Sn. Present in secular equilibrium with ¹²⁹Sn are ^{129m}Sb and ¹²⁹Sb. If release pathway does not partition Sb and Sn, considerations should be given to these additional isotopes.

Table A-8. Inventory of biologically significant nuclides if released to water at 10^4 years after irradiation of 32 MWe-yr/Mg.

<u>Fission products</u>				
Nuclide	$T_{1/2}$	$MPC_w \left(\frac{Ci}{m^3} \right)$	$Q \left(\frac{Ci}{MWe-yr} \right)$	Toxicity index $\left(\frac{m^3}{MWe-yr} \right)$
^{93}Zr	1.5×10^6 yr	8×10^{-4}	5.9×10^{-2}	7×10^1
^{93m}Nb	14 yr	4×10^{-4}	5.9×10^{-2}	1×10^2
^{99}Tc	2.1×10^5 yr	3×10^{-4}	4.4×10^{-1}	2×10^3
$^{126}Sn^{a,b}$	1×10^5 yr	2×10^{-5}	1.6×10^{-2}	9×10^2
^{129}I	1.7×10^7 yr	6×10^{-8}	2.4×10^{-6}	4×10^1
Fission product totals			5.7×10^{-1}	3×10^3
<u>Actinides (plus other alpha-emitters)^c</u>				
^{210}Pb	2.0×10^1 yr	1×10^{-7}	5.8×10^{-5}	9×10^2
^{210}Po	3.8×10^{-1} yr	7×10^{-7}	8.8×10^{-5}	1×10^2
^{226}Ra	1.6×10^3 yr	3×10^{-3}	8.3×10^{-5}	3×10^3
^{237}Np	2.1×10^6 yr	3×10^{-6}	1.2×10^{-2}	4×10^3
^{239}Pu	2.4×10^4 yr	5×10^{-6}	1.7×10^{-1}	3×10^4
^{240}Pu	6.6×10^3 yr	5×10^{-6}	9.9×10^{-2}	2×10^4
^{241}Am	4.6×10^2 yr	4×10^{-6}	4.6×10^{-3}	6×10^2
^{243}Am	8.0×10^3 yr	4×10^{-6}	2.3×10^{-1}	6×10^4
Actinide totals			5.2×10^{-1}	1×10^5
Grand totals			1.1×10^0	1×10^5

^aMPC is an estimate.

^bExisting in equilibrium with ^{126}Sn are ^{126m}Sb and ^{126}Sb .

^cBesides ^{210}Pb , ^{210}Po , and ^{226}Ra , other isotopes of the $4n+1$ and $4n+2$ decay series are present in the source in secular equilibrium (almost) with $1.7 \times 10^{-4} Ci/MWe-yr$ of ^{229}Th and $1.0 \times 10^{-4} Ci/MWe-yr$ of ^{230}Th .

Table A-9. Inventory of biologically significant nuclides if released to water at 10^5 years after irradiation of 32 MWe·yr/Mg.

Fission products				
Nuclide	T 1/2	MPC _w $\left(\frac{\text{Ci}}{\text{m}^3}\right)$	Q $\left(\frac{\text{Ci}}{\text{MWe}\cdot\text{yr}}\right)$	Toxicity index $\left(\frac{\text{m}^3}{\text{MWe}\cdot\text{yr}}\right)$
^{93}Zr	1.5×10^6 yr	8×10^{-4}	5.7×10^{-2}	7×10^1
$^{93\text{m}}\text{Nb}$	1.4×10^1 yr	4×10^{-4}	5.7×10^{-2}	1×10^2
^{99}Tc	2.1×10^5 yr	3×10^{-4}	3.3×10^{-1}	1×10^3
$^{126}\text{Sn}^{a,b}$	1×10^5 yr	2×10^{-5}	9.1×10^{-3}	5×10^2
^{129}I	1.7×10^7 yr	6×10^{-8}	2.4×10^{-6}	4×10^1
Fission product totals			4.5×10^{-1}	2×10^3
Actinides (plus other alpha-emitters) ^c				
^{210}Pb	2.0×10^1 yr	1×10^{-7}	7.0×10^{-4}	7×10^3
^{210}Po	3.8×10^1 yr	7×10^{-7}	7.0×10^{-4}	1×10^3
^{226}Ra	1.6×10^3 yr	3×10^{-8}	7.0×10^{-4}	2×10^4
$^{229}\text{Th}^a$	7.3×10^3 yr	7×10^{-6}	3.7×10^{-3}	5×10^2
$^{230}\text{Th}^c$	8.0×10^4 yr	2×10^{-6}	6.9×10^{-4}	3×10^2
^{233}U	1.6×10^5 yr	3×10^{-5}	4.0×10^{-3}	1×10^2
^{237}Np	2.1×10^6 yr	3×10^{-6}	1.1×10^{-2}	4×10^3
^{239}Pu	2.4×10^4 yr	5×10^{-6}	1.8×10^{-2}	4×10^3
Actinide totals			3.9×10^{-2}	4×10^4
Grand totals			4.9×10^{-1}	4×10^4

^aThe MPCs are estimates — lowest for all other isotopes of that element.

^bExisting in equilibrium with ^{126}Sn are $^{126\text{m}}\text{Sb}$ and ^{126}Sb .

^cBesides ^{210}Pb , ^{210}Po , and ^{226}Ra , other isotopes of the 4n+1 and 4n+2 decay series are present in the source in secular equilibrium with ^{229}Th and ^{230}Th .

Table A-10 Inventory of biologically significant nuclides if released to water at 10^6 years after irradiation of 32 MWe·yr/Mg.

Fission products				
Nuclide	T 1/2	MPC _w $\left(\frac{\text{Ci}}{\text{m}^3}\right)$	Q $\left(\frac{\text{Ci}}{\text{MWe}\cdot\text{yr}}\right)$	Toxicity Index $\left(\frac{\text{m}^3}{\text{MWe}\cdot\text{yr}}\right)$
^{93}Zr	1.5×10^6 yr	8×10^{-4}	3.9×10^{-2}	5×10^1
$^{93\text{m}}\text{Nb}$	1.4×10^1 yr	4×10^{-4}	3.9×10^{-2}	1×10^2
^{129}I	1.5×10^7 yr	6×10^{-8}	2.3×10^{-6}	4×10^1
Fission product totals			7.8×10^{-2}	2×10^2
Actinides (plus other alpha-emitters) ^a				
^{210}Pb	2.0×10^1 yr	1×10^{-7}	1.7×10^{-4}	2×10^3
^{210}Po	3.8×10^{-1} yr	7×10^{-7}	1.7×10^{-4}	2×10^2
^{226}Ra	1.6×10^3 yr	3×10^{-8}	1.7×10^{-4}	6×10^3
^{229}Th ^a	7.3×10^3 yr	7×10^{-6}	9.1×10^{-3}	1×10^3
^{230}Th	8.0×10^4 yr	2×10^{-6}	1.7×10^{-4}	9×10^1
^{231}Pa	3.3×10^4 yr	9×10^{-7}	1.0×10^{-5}	1×10^1
^{233}U	1.6×10^5 yr	3×10^{-5}	9.0×10^{-3}	3×10^2
^{237}Np	2.1×10^6 yr	3×10^{-6}	8.5×10^{-3}	3×10^3
Actinide totals			2.7×10^{-2}	1×10^4
Grand totals			1.1×10^{-1}	1×10^4

^aBesides ^{210}Pb , ^{210}Po , and ^{226}Ra , other isotopes of the $4n + 1$, $4n + 2$, and $4n + 3$ decay series are present in the source in secular equilibrium with ^{229}Th , ^{230}Th , and ^{231}Pa . MPC is an estimate - the lowest of all other isotopes of Th.

1.3.5 DOSE CONVERSION FACTORS

1.3.5.1 AIRBORNE RADIOACTIVITY

Only those radioactive species having relatively short half-lives ($T_{1/2} < 10\,000$ yr) were assumed to contribute significantly to radiation exposures resulting from airborne transport. Doses to members of the exposed population were considered to result from three processes:

- inhalation of radioactivity from the passing cloud,
- exposure to the gamma radiation from radioactivity deposited from the cloud onto the surface of the ground,
- ingestion of radioactivity that has entered into food chains after having been deposited on the ground from the cloud (only the forage-cow-milk-person pathway was considered).

In all cases, a short-term airborne release ("puff") rather than a long-term seep ("plume") of radioactivity is assumed. The following discussions present some features of the dose conversion factors.

1.3.5.2 INHALATION DOSE CONVERSION FACTORS

The dose delivered over a 50-year period after radioisotope intake was determined for the critical organs of a "standard

man" (Recommendations of the International Commission on Radiological Protection, 1959) for each of the nuclides of interest (see Table ____) under the assumption of a short-term exposure to airborne radioactivity.

Uptake of the radioactivity was assumed to occur over a one-day period by inhalation; the total amount of air inhaled during this period is taken as 20 m^3 (Recommendations of the ICRP, 1959). The dose conversion factors given in the tables of this appendix were calculated with the Oak Ridge INREM code (Turner et al., 1968; Killough and McKay, 1976). Basically, this code uses the single-exponential model used by the International Commission to calculate "maximum permissible concentrations in air and water" of radioactivity, where, in the latter reference, chronic exposure is assumed.

Since the method of calculating the critical-organ dose commitment resulting from exposure to a given amount of radioactivity ingested over a finite period of time is outlined in detail by Turner et al. (1968) and by Killough and McKay (1976), this aspect of calculating the dose conversion factors used for inhalation will not be further amplified here. However, the method for converting the critical-organ dose conversion factor to the "whole-body equivalent" dose conversion factor warrants further discussion.

This latter conversion is based on the implicit equivalence of doses to various human organs as exemplified by the maximum

permissible dose equivalents for occupational workers recommended by the International Commission (Recommendations of the ICRP, 1959) (See Table ____). It can be seen that, for instance, a dose of 30 rem/yr is permissible to the skin, as opposed to 5 rem/yr to the whole body. Thus, one might infer that a dose of 1 rem to the skin incurs a risk to an exposed individual only one-sixth as great as that if a 1-rem dose had been delivered to the whole body. Consequently, the "whole-body equivalent" of a 1-rem dose to the skin would be 0.17 rem. This procedure has been followed to calculate the whole-body equivalent dose conversion factors in Table _____. Again, the critical-organ dose conversion factors shown in Table _____ have been calculated with the Oak Ridge INREM code (Turner et al., 1968; Killough and McKay, 1976).

1.3.5.3 DEPOSITION DOSE CONVERSION FACTOR: EXTERNAL DOSE

Fifty-year integrated gamma-ray doses to the whole body resulting from the uniform deposition of the nuclides (see Table _____) have been calculated, with the assumption that the "whole body" is being exposed at a point three feet above an infinite plane source of the radioactive species. The method of calculation has been discussed in detail (Higgins, 1963) and includes a correction for the air attenuation of the gamma radiation.

As noted above, only gamma energy released in the first 50 years after deposition of the radioactivity is considered in

the dose conversion factors shown in Table ____; for the shorter half-life nuclides, such as ^{89}Sr , essentially all this dose is delivered in the first year after deposition. For the very long-lived species, such as ^{243}Am , the dose rate is essentially constant over the period of interest. The gamma-ray energies used in these calculations for the nuclides of interest were obtained from Lederer et al. (1967).

It should be noted that these dose conversion factors have not been corrected for "weathering". The process of weathering will slowly move the radioactivity into the soil, thus increasing the gamma-ray attenuation and decreasing the dose delivered to a point three feet above the ground surface. Clearly, this process will not be important for the shorter-lived nuclides such as ^{89}Sr , but for the longer-lived nuclides (especially those having relatively low-energy gamma rays) the dose conversion factors in Table ____ could be high by a factor of 2 or 3. However, no data exist whereby a reasonable correction for this effect can be estimated and justified.

1.3.5.4 DEPOSITION DOSE CONVERSION FACTOR: INTERNAL DOSE

Dose conversion factors have been calculated that give the internal dose (to a critical organ) delivered over a 50-year period following the deposition of a number of radioactive species whose concentrations are known for a given area. For the purpose of this approach, it has been assumed that the

predominant pathway for the ingestion of radioactivity is through the forage-cow-milk-person food chain. Since the method used has been discussed in some detail by Ng and Thompson (1966) and more recently by Ng et al. (1976), this report will only indicate briefly the approach employed in the estimation of the dose conversion factors.

The dose to a given organ by means of the forage-cow-milk pathway is calculated as the product of the total activity ingested through milk per unit deposition and the 50-year dose per unit of activity ingested. The total activity ingested through milk (in Ci) is given by,

$$\int_0^{50 \text{ yr}} I \cdot dt = \frac{Jr(UAF)F(0)f_m^*}{\lambda_p}$$

where J = rate of milk consumption (liters/da), r = retention factor on forage, (UAF) = utilized area factor (i.e., the effective area of pasture grazed daily by the cow [m^2 /da]), $F(0)$ = deposition at time zero (Ci/m^2), f_m^* = transfer coefficient to milk (i.e., the fraction of the nuclide ingested daily by the cow that is secreted in milk per liter [da/liter]), and λ_p = effective rate of removal from forage (da^{-1}).

The dose per unit deposition through milk of Table ____ are those calculated by Ng et al. (1976) for the adult. Assumptions are $J = 1$, $r = 0.5$, and $(UAF) = 45$. The half-residence time

for nuclide particles on forage is assumed to be 14 days, which leads to the following expression for λ_p :

$$\lambda_p = \frac{\ln 2}{T_r} + \frac{\ln 2}{14},$$

where T_r is the half-life for radioactive decay, expressed in days. The transfer coefficient to milk f_m^* and the 50-year dose to the critical organs and whole body per unit of activity ingested $D'_{50 \text{ yr}}$ (rem/Ci) are listed in Ng et al. (1976).

Internal dose conversion factors were calculated both for the critical organ and for the whole body. Also, the whole-body equivalent dose conversion factors were calculated by the method discussed in the section titled Inhalation Dose Conversion Factors above, and are shown in Table ____.

1.3.5.5 WATERBORNE RADIOACTIVITY

Only the relatively long-lived radioactive species were assumed to contribute significantly to radiation exposures resulting from ground water transport. However, for the sake of completeness, a few of the shorter-lived species are included in this compilation. Critical-organ doses to members of the population who drink contaminated water can be calculated with dose conversion factors based on the values of the MPC_w , given in Appendix B, Table 2, of the Nuclear Regulatory Commission Standards for Protection Against Radiation (1960, 1976). These dose conversion factors (both for critical organs and the whole

body), given in Table ____ are calculated under the assumption of a long-term exposure (integrated over 50 yr) to water containing the radionuclides listed in the first column. The appropriate values of MPC_w are also listed in Table ____ for the radioactivities of interest (both for the critical organs and the whole body), as are the annual doses that would be delivered to the critical organs by exposure to the indicated concentrations of radioactivity. (These doses were inferred by using the MPC_w values found in Recommendations of the ICRP [1959].) The critical organs were also identified through use of these references. Once again, the whole-body equivalent dose conversion factors were determined using the approach outlined under Inhalation Dose Conversion Factors above.

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Table B-1. Dose conversion factors^a for biologically significant nuclides in air: inhalation.

Nuclide	Critical-organ dose conversion factors (rem·m ³ /Ci·yr)	Critical organ	Whole-body equivalent dose conversion factor (rem·m ³ /Ci·yr)	Whole-body dose conversion factor (rem·m ³ /Ci·yr)
<u>Fission products</u>				
⁸⁹ Sr	3.0×10^9	Bone	5.0×10^8	8.4×10^7
⁹⁰ Sr	3.3×10^{11}	Bone	5.5×10^{10}	1.6×10^9
⁹¹ Y	2.4×10^9	Bone	4.1×10^8	6.5×10^7
⁹⁵ Zr	1.4×10^9	Lung	4.7×10^8	1.2×10^8
⁹⁵ Nb	4.7×10^8	Lung	1.6×10^8	3.3×10^7
¹⁰⁶ Ru	8.4×10^9	Lung	2.3×10^9	2.1×10^7
¹²⁵ Sb	1.8×10^9	Lung	6.0×10^8	2.7×10^7
^{127m} Ie	1.2×10^9	Lung	4.0×10^8	1.2×10^7
^{129m} Ie	1.4×10^9	Lung	4.8×10^8	3.2×10^7
¹³⁴ Cs	4.0×10^9	Lung	1.3×10^9	4.1×10^8
¹³⁷ Cs	3.3×10^9	Lung	5.5×10^8	2.4×10^8
¹⁴⁴ Ce	8.8×10^9	Bone	1.5×10^9	4.7×10^8
¹⁴⁷ Pm	1.4×10^9	Bone	2.3×10^8	5.0×10^7
¹⁵⁴ Eu	2.2×10^{10}	Bone	5.7×10^9	1.4×10^9
<u>Actinides</u>				
²³⁸ Pu	4.2×10^{13}	Bone	6.9×10^{12}	1.0×10^{12}
²⁴⁰ Pu	4.8×10^{13}	Bone	8.0×10^{12}	1.2×10^{12}
²⁴¹ Pu	9.0×10^{11}	Bone	1.5×10^{11}	1.9×10^{10}
²⁴¹ Am	1.5×10^{13}	Bone	2.5×10^{12}	9.9×10^{11}
²⁴³ Am	1.5×10^{13}	Bone	2.5×10^{12}	9.7×10^{11}
²⁴² Cm	3.8×10^{11}	Liver	1.3×10^{11}	2.5×10^{10}
²⁴⁴ Cm	9.2×10^{12}	Bone	1.5×10^{12}	5.4×10^{11}

^aFor dose delivered over a 50-yr period following a brief period of inhalation of radioactivity.

Table 1-2. Maximum permissible dose equivalents recommended by the International Commission on Radiological Protection for occupational workers.

Organ	Maximum permissible dose equivalent in one year (rem)
Red bone marrow, whole body, and gonads	5
Skin, thyroid, and bone ^a	30
All other single organs	15

^aNote that the National Council on Radiation Protection and Measurement currently specifies an annual maximum permissible dose of 15 rem for these organs as well (NCRPM, 1971).

Table B-3. Dose conversion factors^a for radioactivity released to the atmosphere and deposited on the ground: external exposure.

Nuclide	Whole-body dose conversion factor (rem·m ² /Ci)
<u>Fission product</u>	
⁸⁹ Sr	3.0
⁹⁰ Sr	—
⁹¹ Y	1.3×10^2
⁹⁵ Zr	2.9×10^4
⁹⁵ Nb	1.8×10^4
¹⁰⁵ Ru	6.9×10^4
¹²⁵ Sb	2.4×10^5
^{127m} Ie	7.2×10^3
^{129m} Ie	6.8×10^3
¹³⁴ Cs	8.3×10^5
¹³⁷ Cs	3.1×10^6
¹⁴⁴ Ce	9.6×10^3
¹⁴⁷ Pm	—
¹⁵⁴ Eu	1.0×10^6
<u>Actinides</u>	
²³⁸ Pu	2.9×10^4
²⁴⁰ Pu	8.6×10^4
²⁴¹ Pu	2.5×10^5
²⁴¹ Am	5.1×10^5
²⁴³ Am	3.1×10^6
²⁴² Cm	3.0×10^4
²⁴⁴ Cm	3.5×10^4

^aFor dose delivered over a 50-yr period following a single deposition of radioactivity on the ground.

Table 1. Dose^a conversion factors for radioactivity released to the atmosphere and deposited on the ground: internal exposure by means of the forage-cow-milk-person pathway.

Nuclide	Critical-organ dose conversion factor (rem·m ² /Ci)	Critical organ ^b	Whole-body equivalent dose conversion factor (rem·m ² /Ci)	Whole-body dose conversion factor (rem·m ² /Ci)
<u>Fission product</u>				
⁸⁹ Sr	8.1×10^3	Bone	1.3×10^3	1.1×10^3
⁹⁰ Sr	1.9×10^5	Bone	3.3×10^4	1.5×10^4
⁹¹ Y	8.3×10^2	G.I. ^c	2.8×10^2	2.7
⁹⁵ Zr	3.2×10^3	G.I.	1.1×10^3	2.1×10^1
⁹⁵ Nb	9.5×10^2	G.I.	3.2×10^2	7.3
¹⁰⁶ Ru	7.0×10^1	G.I.	2.3×10^1	1.4
¹²⁵ Sb	3.4×10^2	G.I.	1.1×10^2	4.9
^{127m} Te	3.4×10^3	G.I.	1.1×10^3	5.0×10^1
^{129m} Te	6.9×10^3	G.I.	2.3×10^3	8.9×10^1
¹³⁴ Cs	2.0×10^5	Whole body	2.0×10^5	2.0×10^5
¹³⁷ Cs	1.5×10^5	Whole body	1.5×10^5	1.5×10^5
¹⁴⁴ Ce	2.2×10^3	G.I.	7.2×10^2	3.4
¹⁴⁷ Pm	1.0×10^2	G.I.	3.5×10^1	5.3×10^{-1}
¹⁵⁴ Eu	5.9×10^2	G.I.	2.0×10^2	1.7×10^{-1}
<u>Actinides</u>				
²³⁸ Pu	2.3×10^1	Bone	3.3	6.7×10^{-1}
²⁴⁰ Pu	2.6×10^1	Bone	4.3	7.4×10^{-1}
²⁴¹ Pu	5.6×10^{-1}	Bone	9.4×10^{-2}	1.4×10^{-2}
²⁴¹ Am	1.8×10^5	Bone	3.0×10^4	5.1×10^3
²⁴³ Am	1.7×10^5	Bone	2.9×10^4	5.0×10^3
²⁴² Cm	3.6×10^3	Bone	6.0×10^2	1.1×10^2
²⁴⁴ Cm	9.0×10^4	Bone	1.5×10^4	2.7×10^3

^a Dose delivered over a 50-yr period following a single deposition of radioactivity on forage.

^b Inferred from Recommendations of the ICRP (1959).

^c Gastrointestinal tract.

Table B-5. Dose conversion factors for biologically significant nuclides in water.

Nuclide	MPC _w ^a (Ci/m ³)	Critical organ ^b	Annual dose ^{b,c} (rem)	Critical-organ dose conversion factor ^d (rem·m ³ /Ci)	Whole-body equivalent dose conversion factor ^d (rem·m ³ /Ci)	Whole-body dose conversion factor ^d MPC _w ^e (Ci/m ³)	Whole-body dose conversion factor ^d (rem·m ³ /Ci)
Fission products							
⁹⁰ Sr ^f	3 × 10 ⁻⁷	Bone	3	3 × 10 ⁸	5 × 10 ⁷	1 × 10 ⁻⁶	1 × 10 ⁷
⁹³ Zr	8 × 10 ⁻⁴	G.I. ^g	1.5	9 × 10 ⁴	3 × 10 ⁴	6 × 10 ⁻¹	4 × 10
⁹⁹ Tc	3 × 10 ⁻⁴	G.I.	1.5	3 × 10 ⁵	8 × 10 ⁴	1 × 10 ⁻²	3 × 10 ³
¹²⁶ Sn	2 × 10 ⁻⁵	G.I.	1.5	4 × 10 ⁶	1 × 10 ⁶	2 × 10 ⁻³	1 × 10 ⁴
¹²⁹ I	6 × 10 ⁻⁸	Thyroid	0.5	4 × 10 ⁸	4 × 10 ⁸	8 × 10 ⁻⁵	3 × 10 ⁶
¹³⁷ Cs ^f	2 × 10 ⁻⁵	Whole body	0.5	7 × 10 ⁵	7 × 10 ⁵	2 × 10 ⁻⁵	7 × 10 ⁵
Actinides							
²¹⁰ Pb ^h	1 × 10 ⁻⁷	Kidney	1.5	7 × 10 ⁸	3 × 10 ⁸	1 × 10 ⁻⁷	3 × 10 ⁸
²¹⁰ Pb ^h	7 × 10 ⁻⁷	Spleen	1.5	1 × 10 ⁸	4 × 10 ⁷	8 × 10 ⁻⁶	3 × 10 ⁶
²²⁶ Ra ^h	3 × 10 ⁻⁸	Bone	3	5 × 10 ⁹	8 × 10 ⁸	6 × 10 ⁻⁸	4 × 10 ⁸
²³⁷ Np	3 × 10 ⁻⁶	Bone	3	5 × 10 ⁷	8 × 10 ⁶	1 × 10 ⁻⁵	3 × 10 ⁶
²³⁸ Pu ^f	5 × 10 ⁻⁶	Bone	3	3 × 10 ⁷	5 × 10 ⁶	4 × 10 ⁻⁵	6 × 10 ⁵
²³⁹ Pu	5 × 10 ⁻⁶	Bone	3	3 × 10 ⁷	5 × 10 ⁶	3 × 10 ⁻⁵	8 × 10 ⁵
²⁴⁰ Pu	5 × 10 ⁻⁶	Bone	3	3 × 10 ⁷	5 × 10 ⁶	3 × 10 ⁻⁵	8 × 10 ⁵
²⁴¹ Am	4 × 10 ⁻⁶	Kidney	1.5	2 × 10 ⁷	6 × 10 ⁶	1 × 10 ⁻⁵	3 × 10 ⁶
²⁴³ Am	4 × 10 ⁻⁶	Bone	3	4 × 10 ⁷	6 × 10 ⁶	1 × 10 ⁻⁵	3 × 10 ⁶
²⁴⁴ Cm ^f	7 × 10 ⁻⁶	Bone	3	1 × 10 ⁷	2 × 10 ⁶	2 × 10 ⁻⁵	7 × 10 ⁵

^aFrom NCRP (1960, 1976).

^bInferred from Recommendations of the ICRP (1959).

^cSteady state exposure at MPC_w.

^dFor integral dose from 50 yr of exposure.

^eFor annual dose of 0.5 rem; inferred from Recommendations of the ICRP (1959).

^fAssuming an initial concentration equal to MPC_w and allowing for radioactive decay.

^gGastro-intestinal tract.

^hIn equilibrium with parent ²³⁰Th.

I. FUEL REPROCESSING PLANT

A. Handling	Value	Dimension	Inclusion of Human Factors
1. Rate of crane stall	1.5×10^{-6}	Events/Hour	Yes
2. Rate of crane drop	3.0×10^{-6}	Events/Hour	Yes
3. Total crane time per canister	3.3×10^{-1}	Hour/ Canister	Yes
4. Conditional probability of canister melt per crane stall	0.0	---	No
5. Conditional probability of canister rupture per crane drop	0.0	---	No
6. Glass release function for melted canister per MWe-yr	A.D. 0.0 V. * 1.0	---	NA
7. Glass release function for ruptured canister per MWe-yr	A.D. 1.0×10^{-2} V. 0.0	---	NA
8. Calcine release function for melted canister per MWe-yr	A.D. 0.0 V. ** 1.0	---	NA
9. Calcine release function for ruptured canister per MWe-yr	A.D. 5×10^{-1} V. 0.0	---	NA
10. Fraction of released volatiles reaching filter	1×10^{-2}	---	NA

* For volatiles only Cs is released

** For volatiles only Cs, Te, Ru are released

I. A. Handling (Cont.)	Value	Dimension	Inclusion of Human Factors
11. Fraction of released airborne dispersable particles reaching filter	0.0	---	NA
12. Rate of filter failure	1.9×10^{-8}	Failures/Week	No
13. Time for clean-up after release	1.0	Week	Yes
14. Transmission ratio for failed filter	1.0	---	Yes
15. Transmission ratio for non-failed filter	1.0×10^{-11}	---	No

I. FUEL REPROCESSING PLANT (Continued)

B. Interim Storage, Loss of Cooling Circulation	Value	Dimension	Inclusion of Human Factors
1. Rate of loss of cooling circulation	1×10^{-1}	Events/Year	No
2. Conditional probability of loss of make-up, hence boiloff	1×10^{-6}	---	Yes
3. Probability of canister melt - calcine	0.0	---	No
4. Probability of canister melt - glass	0.5 1-2 yrs old 0 > 2 yrs old	---	No
5. Calcine release fraction per MWe·Yr.	1.0	---	NA
6. Glass release fraction per MWe·Yr.	1.0	---	NA
7. Fraction of released volatiles reaching filters	1.0×10^{-2}	---	NA
8. Fraction of released airborne dispersable particles reaching filter	0.0	---	NA
9. Rate of filter failure	1.9×10^{-8}	Failures/Week	No
10. Time for clean-up after release	1.0	Week	Yes
11. Transmission ratio for failed filter	1.0	---	Yes
12. Transmission ratio for non-failed filter	1.0×10^{-11}		No

1. FUEL REPROCESSING PLANT (Continued)

C. Interim Storage, Drainage of Storage Pool	Value	Dimension	Inclusion of Human Factors
1. Rate of exceeding critical seismic level	1.0×10^{-5}	Events/Year	Yes
2. Ratio of pool rupture from seismic input	1.0×10^{-1}	Ruptures/Event	No
3. Ratio of building rupture from seismic input	1.0	Ruptures/Event	Yes
4. Rate of aircraft rupture of pool and building	1.0×10^{-12}	Events/Year	Yes
5. Conditional probability of emergency cooling not available	1.0×10^{-1}	---	Yes
6. Probability of canister melt calcine	0.0	---	No
7. Probability of canister melt glass	0.5 < 5.5 0.0 > 5.5	---	No
8. Calcine release function	0.0 V 0.0 A.D 1.0U 0.0AD	---	NA
9. Glass release function	0.0AD	---	NA
10. Fraction of released volatiles leaving building	1.0	---	NA
11. Period of calcine storage	5.0 → 10.0	Years	NA
12. Period of glass storage	1.0 → 10.0	Years	NA

II. TRANSPORTATION

A. Truck

	Value	Dimension	Inclusion of Human Factors
1. Accident rate	2.5×10^{-6}	Events/Mile	Yes
2. Distance traveled	1.5×10^{-3}	Miles	NA
3. Conditional probability of impact with rigid object *	5.1×10^{-2}	---	Yes
4. Conditional probability of impact with a train locomotive *	3.1×10^{-3}	---	Yes
5. Conditional probability of impact with puncture object *	2.0×10^{-1}	---	Yes
6. Conditional probability of energy source for airborne dispersion *	1.0×10^{-1}	---	NA
7. Conditional probability of fire *	1.0×10^{-2}	---	Yes
8. Integral of rigid object PDF with calcine release function	6.07×10^{-3}		No
9. Integral of rigid object PDF with glass release function	3.33×10^{-6}		No
10. Integral of train PDF with calcine release function	3.76×10^{-3}		No
11. Integral of train PDF with glass release function	2.04×10^{-6}		No
12. Integral of puncture PDF with release function	0.0		No

* Conditional probabilities are split 50% urban and 50% rural

11. A. Truck (Continued)

Value

Dimension

Inclusion
of Human
Factors13. Integral of fire PDF failure locus with
calcine release function 3.67×10^{-5}

No

14. Integral of fire PDF failure locus with
glass release function 8.85×10^{-4}

No

15. Conditional probability of ruptured cask
entering waterway 1.0×10^{-2}

NA

→ 16. Integral of cask rupture PDF with calcine
release function 3.40×10^{-2} → 17. Integral of cask rupture PDF with glass
release function 3.40×10^{-6}

II. TRANSPORTATION (Continued)

B. Train

	Value	Dimension	Inclusion of Human Factors
1. Accident rate	1.5×10^{-6}	Events/Mile	Yes
2. Distance traveled	1.5×10^3	Miles	NA
3. Conditional probability of impact with rigid object *	5.1×10^{-2}	---	Yes
4. Conditional probability of impact with puncture object *	2.77×10^{-1}	---	Yes
5. Conditional probability of energy source for airborne dispersion	1.0×10^{-1}		NA
6. Conditional probability of fire *	3.69×10^{-2}	---	Yes
7. Integral of rigid body PDF with calcine release function	Urban 3.3×10^{-4} Rural 3.6×10^{-2}	---	No
8. Integral of rigid body PDF with glass release function	Urban 1.67×10^{-7} Rural 2.49×10^{-5}	---	No
9. Integral of puncture PDF with release functions	0.0	---	No
10. Integral of fire PDF with calcine release function	2.16×10^{-5}	---	No
11. Integral of fire PDF with glass release function	8.06×10^{-4}	---	No
12. Conditional probability of ruptured cask entering waterway	1.0×10^{-2}	---	NA

* Conditional probabilities are split 50% Urban and 50% Rural

11. B. Brain (Continued)

Value

Dimension

Inclusion
of Human
Factors

13. Integral of cask rupture PDF with
calcine release function

8.42×10^{-2}

No

14. Integral of cask rupture PDF with
glass release function

8.42×10^{-6}

No

III. REPOSITORY SITE

A. Handling	Value	Dimension	Inclusion of Human Factors
1. Rate of crane drop	3×10^{-6}	Events/Hour	Yes
2. Time of operation per canister	1.67×10^{-1}	Hours/ Canister	Yes
3. Conditional probability of canister rupture	0.0	---	No
4. Calcine release function for ruptured canister	1.0	---	NA
5. Glass release function for ruptured canister	1.0		NA
6. Fraction of released volatiles reaching filter	1×10^{-2}	---	NA
7. Fraction of released airborne dispersable particles reaching filter	0.0	---	NA
8. Rate of filter failure	1.9×10^{-8}	Failures/ Week	No
9. Time for clean-up after release	1.0	Week	Yes
10. Transmission ratio for failed filter	1.0	---	Yes
11. Transmission ratio for non-failed filter	1.0×10^{-11}	---	No

III. REPOSITORY SITE (Continued)

B. Surface Storage

	Value	Dimension	Inclusion of Human Factor
1. Rate of aircraft crashes	1.27×10^{-10}	Events/ Year $\cdot$ Mile ²	Yes
2. Effective cask cross section area	3.22×10^{-5}	Miles ² /Cask	NA
3. Mean cask surface storage time	2×10^{-2}	Years	Yes
4. Calcine release function	1.0		Yes
5. Glass release function	1.0		Yes

Whole Body [Q]° x [DCF]**		1 Year Old Waste Volatiles Airborne Dispersables		5 Year Old Waste Volatiles Airborne Dispersables		10 Year Old Waste Volatiles Airborne Dispersables	
$\frac{\text{rem m}^2}{\text{Ci}}$	Internal	1.6×10^9	1.6×10^9	7.3×10^8	7.6×10^8	4.6×10^8	4.9×10^8
X m/yr.	External	1.5×10^{10}	1.6×10^{10}	1.0×10^{10}	1.1×10^{10}	8.6×10^9	8.7×10^9
Deposition (must be multiplied by 3.16×10^6)	Total	1.7×10^{10}	1.8×10^{10}	1.1×10^{10}	1.2×10^{10}	9.1×10^9	9.2×10^9
		5.4×10^{15}	5.4×10^{15}	3.5×10^{15}	3.9×10^{15}	2.9×10^{15}	2.9×10^{15}
$\frac{\text{rem m}^3}{\text{Ci Yr}}$	Internal	3.2×10^{12}	6.9×10^{13}	1.3×10^{12}	6.2×10^{13}	7.5×10^{11}	4.3×10^{13}
Inhalation	External	0	0	0	0	0	0
	Total	3.2×10^{12}	6.9×10^{13}	1.3×10^{12}	6.2×10^{13}	7.5×10^{11}	4.3×10^{13}
TOTAL DEPOSITION		3.2×10^{12}	6.9×10^{13}	1.3×10^{12}	6.3×10^{13}	7.6×10^{11}	4.3×10^{13}
PLUS INHALATION		5.4×10^{15}	5.4×10^{15}	3.5×10^{15}	3.9×10^{15}	2.9×10^{15}	2.9×10^{15}

*[Q] is the matrix of radionuclide inventory, i.e., Ci/MWe-Yr

$$= \frac{\text{REM M}^3}{\text{MWe Yr Yr}}$$

**[DCF] is the matrix of dose conversion factors, i.e., REM x m³/Ci Yr

Air Pathway Demography Factor	Location	Puff or Plume	Volatiles or Airborne Dispersables	Value of Factor*
Fuel Reprocessing Plant	Barnwell	Plume	Volatile	9.8×10^{-11}
Transportation	Urban	Puff	Airborne Dispersables	3.1×10^{-9}
		Plume	Volatiles	1.7×10^{-9}
	Rural	Puff	Airborne Dispersables	2.1×10^{-10}
		Plume	Volatiles	2.43×10^{-6}
Repository Site	NTS	Plume	Volatiles and Airborne Dispersables	5.9×10^{-13}

* Integrated from 10^2 to 10^6 meters.

3.1 INTERIM STORAGE AT FUEL REPROCESSING PLANT

3.1.1 REFERENCE CANISTER

A sketch of the reference canister design is shown in Figure 3.1-1.

We have arbitrarily assumed that the heat generation per canister must not exceed 3.5 kW at the time of emplacement in the federal repository. This time has been taken to be 10 years, as discussed in the appendix entitled "SHLN Management Sequence". Some dilution of the calcine waste form is necessary to meet this requirement.

For the calcine waste management sequence selected, the canister must be internally finned to maintain the waste centerline temperature below the bakeout value during normal operating conditions. Although this may be a conservative limitation, it is considered to be "good engineering practice".

3.1.2 REFERENCE INTERIM STORAGE FACILITY DESIGN

It is assumed that the filled and sealed stainless steel canisters are stored in a concrete basin filled with water and are dependent on the building structure and mechanical cooling system to provide shielding, physical protection and cooling. Other general assumptions pertaining to interim storage of the solidified waste are as follows:

- 4 Large numbers of canisters are grouped in regular arrays in multiple storage cells, each cell having nominal capacity for

about one year inventory from the reprocessing plant.

- The basins are filled with demineralized water, which cools the waste and provides radiation shielding.
- Heat is removed by the constant circulation of the pool water in direct contact with the canisters. It is transferred to a second air cooling loop and rejected to the atmosphere from standard open cooling towers.
- Water purity is maintained at less than 10 ppm chloride ions.
- All operations in interim storage take place within a building that is sealed under normal operating conditions.
- A transfer cask is used to protect the canister during most movement out of water. The cask is capable of protecting the canister from damage due to drops. The only time the canister is handled separately is during insertion into and removal from the pool.
- Equipment within the interim storage water basin area is arranged so that the transfer cask cannot be moved over the basins, which precludes dropping the cask into the basins.
- The function air filtration system passes 10^{-11} of all volatiles released from canister accidents within the plant. This is made possible by stacking HEPA filters.

- The probability of filtration-system failure is 10^{-6} per year. This is identical to the HEPA failure rate given in the U.S. NRC study of reactor safety (1975). The probability that filter failure will allow release of volatile materials from an accident within the sealed building is based on the probability that the filter will fail within one week after the accident ($1/52 \times 10^{-6}$). One week is required to clean up failed canisters.
- If the filter fails within one week of an accident, 1 percent of the volatiles released within the building will escape into the atmosphere. The rest will condense inside the building.
- Nominal probability of crane drop is 3×10^{-6} per hour of operation (from U.S. NRC reactor safety study [1975]).
- Nominal probability of crane stall is 1.5×10^{-6} per hour of operation (Smith, 1976).
- Nominal time spent handling each canister in air outside the cask is 20 minutes. This includes time for insertion into the pool, retrieval, and insertion into a transportation cask.

EVENT TREE

Figure 3.1-2 shows the event tree used to represent major accidental release modes during HLSM handling at the water storage site. Consistent with the assumptions stated above, crane stall and crane

drop when the canister is not over water are the two major accident modes.

RELEASE FRACTIONS FOR HANDLING EVENTS

Possible canister failure mechanisms are:

- Impact onto a hard surface resulting from a crane drop
- Overheating of a canister by internally generated decay heat

For the case of a crane drop, the maximum equivalent height of drop is not likely to be greater than 4 meters, since crane lift height for most cell handling operations is not expected to exceed the canister length by more than a meter - admittedly the water basin is deeper than this, but the water would substantially reduce the impact velocity of a canister which was dropped into the pool, and thus we believe that no release would occur if a canister fell into the pool.

Smith and Ross (1975) performed experiments which involved corner dropping of flat-bottomed canisters, which is a more severe condition than dropping of canisters with hemispherical bottoms (as the reference canister has). These tests showed no breaching of canisters dropped from heights of 9.15 meters (30 feet) and about half of them breaching at heights in excess of 20.7 meters (68 feet). In view of this, we believe the release fraction would be zero for all accidents involving canister drop at the interim storage facility.

For the case of a crane stall, heat transfer calculations for a canister containing 1 year old glass and suspended in air gives a steady state wall temperature of about 900°F (482°C). For the canister containing 5 year old diluted calcine, this temperature is 428°F (220°C).

For a canister filled with glass waste, corrosion by the waste is predicted to be the limiting process of interest, and the failure temperature is estimated at $1200 \pm 50^\circ\text{C}$ ($2190 \pm 90^\circ\text{F}$) for a time of a few hours, combining the mechanisms discussed in the Appendix on Release Function Methodology.

For a canister filled with calcine waste, the earliest corresponding failure temperature is estimated at $1300 \pm 50^\circ\text{C}$ ($2370 \pm 90^\circ\text{F}$), since it is likely that creep and/or corrosion would cause failure somewhat below the melting point.

Since the canister temperatures are well below these failure temperatures, the release fraction is believed to be zero for the crane stall case also.

CALCULATION OF RELEASE

The assumptions and probabilities stated above make possible the computation of expected values of waste released. In these calculations, we have assumed that 100 percent of the volatiles available (specifically Cs, Ru, and Te) are released into the building that houses the pool, even though the actual release fractions are

believed to be zero, as stated above. This very conservative assumption is made in order to demonstrate that the expected release from handling accidents at the interim storage site is insignificant relative to expected releases from other portions of the waste management sequence even if the highest possible release fractions from the canister are used.

CRANE STALL

Progressing down the left side of the event tree in Figure 3.1-2, the probability of a crane-stall accident per canister of solidified waste is

$$(1.5 \times 10^{-6} \frac{\text{stalls}}{\text{hr of operation}}) \left(\frac{1 \text{ hr operation}}{3 \text{ can}} \right) = 5 \times 10^{-7} \frac{\text{stalls}}{\text{can}},$$

Each canister contains 100 megawatt electric*years of solid waste. Therefore, there are 5×10^{-9} stalls per megawatt electric*year. Two possibilities ensue: If the filtration system works for one week after the accident, 10^{-11} of the amount released will enter the atmosphere. Assumint 100 percent of the volatiles available are released, the expected amount released per megawatt electric*year is 5×10^{-20} times the activity in one canister (due to 100 MWe*yr of waste).* If the filtration system fails in the week following the accident, 10^{-2} of the available material will be released. The probability of filter failure is

$$(10^{-6} \frac{1}{\text{yr}}) \left(\frac{1 \text{ yr}}{52 \text{ wk}} \right) = 1.92 \times 10^{-8} \frac{1}{\text{wk}}.$$

*Strickly speaking, this must be multiplied by one minus the probability of filter failure. To a very good approximation, the number is unity.

The expected amount released due to crane-stall accidents followed by filter failure (1-yr-old waste only), then, is

$$(5 \times 10^{-9}) (10^{-2}) (1.92 \times 10^{-8}) = 9.6 \times 10^{-19} \frac{1}{\text{MWe} \cdot \text{yr}}$$

times the activity in one canister. Note that this is approximately 20 times the expected release from crane-stall accidents where the filter does not fail.

The net expected release for a crane-stall accident is the sum of the above two numbers (i.e., 10^{-18} times the activity of the contents of one nominal canister). This is small relative to releases expected for storage and transportation accidents. Differences in population distribution used in dose calculations do not alter this conclusion.

CRANE DROP

A similar sequence of calculations yields the expected release for a crane-drop accident. The probability of a crane drop is twice that of a stall; all other numbers in the calculations are the same. Consequently, the expected release due to a crane-drop accident is the same as that for a crane-stall accident: 10^{-18} times the activity of the contents of one nominal canister.

Handling accidents at the interim storage site do not significantly contribute to the expected dose from high-level nuclear waste disposal.

FILLED CANISTER CHARACTERISTICS

	<u>GLASS</u>	<u>CALCINE</u>
APPROXIMATE WEIGHT OF WASTE (LBS.)	660	248
FUEL EQUIVALENT (MTU/CAN)	2.5	3.1
ENERGY EQUIVALENT ($\frac{\text{MW(CI) BTU}}{\text{CAN}}$)	81	100
VOLUME (m^3)	0.2	0.2

FIGURE 3.61.1

REFERENCE WASTE CRITERIA

Fig. 5-2. Event tree for accidental release during handling at water storage site.

3.20 LOSS OF COOLANT EVENTS

A seismic event or other catastrophic accident could cause rupture of the basin and loss of the coolant water. As a consequence, without the prime heat removal medium, failures of the canisters could well occur due to corrosion or melt-through of the canister wall. The rapidity of failure depends upon the heating rate of the wastes and on the geometry of the canister array.

Modeling of the thermal response of single and multiple waste canisters was performed by ARHCO (Jansen et al, 1974). Of particular interest here were their analyses of transient temperature following an instantaneous loss of cooling water from a Retrievable Surface Storage Facility (RSSF). This facility was assumed to consist of 506 waste canisters, each producing 5 kW of heat and packed into a rectangular concrete basin. The ARHCO approach was to use a lumped parameter model that represented the system with a small number of nodes interconnected by a network of transmittances. We have developed a similar but expanded capability that differs from ARHCO's work in the following ways:

- The HEATING5 code (Turner et al, 1976) is utilized to solve the finite difference heat flow equations. Although a 1-D lumped parameter

approach was used for initial results, a full 2-D treatment of heat transfer through the waste-canister array is possible.

- Interim storage results were obtained over a variety of waste forms and ages (heat generation rates).
- Heat transfer between the waste canisters and naturally circulating air was modeled to reflect more accurately the actual air flow rates and convective transfer coefficients.

The interim storage configuration examined was very nearly that treated by ARHCO:

- A 36 1/2 ft X 18 1/2 ft X 36 ft high basin with an open top
- 1 ft diameter - 10 ft long canisters at 18-inch-center-to-center spacing
- 506 upright canisters in two layers, one directly above the other with a 1 ft spacing between
- Initial canister temperatures of 127°F and 120°F for the air and concrete basin, respectively
- Instantaneous loss of cooling water to initiate the temperature excursion
- Cooling of the canisters by the mechanisms of (1) natural convection

to air, (2) radiation to space, (3) radiation to the concrete basin walls, and (4) conduction through the concrete walls.

The waste canister region was modelled as a single block with homogenized SHLW and stainless steel properties and a volume of 12680 ft³. Instead of the actual rectangular solid shape, a cylinder of equivalent top area and height was used for computational convenience to represent the waste region. This approximation allowed the advantage of a 2D geometry with axial symmetry with little loss of accuracy. The cylinder dimensions were:

Radius: 13.86 ft

Height: 21 ft

Surrounding the waste block and separated from it by an air gap was an annular ring of concrete. Its thickness was 4 ft, but the thermal resistance input was equivalent to an 8 ft thickness (8 ft was found by ARHCO to be essentially an infinite thickness). The height of the concrete ring was 36 ft.

Located on top of the waste block was imposed a strictly artificial mass of air whose purpose was to act as a heat exchange medium. By means of a simulated boundary condition between the air and the waste block, natural convective heat flow between canister surfaces and air was represented. The heat transfer process simulation is described in more detail below. Figure 3.2-1 is a sketch of the geometry input to HEATING5 with boundary conditions shown:

4.1.2 ACCIDENTS CONSIDERED

Before conducting analyses to determine the structural integrity of the shipping cask and canisters when subjected to various hypothesized accident scenarios, it was necessary to validate the baseline designs selected and to establish their normal performance characteristics. In addition, we verified whether the shipping casks selected conform to the Code of Federal Regulations for the shipment of radioactive material, i.e., the type B packaging accident test criteria of 100CFR71 and 49CFR 173.398.

Once we determined that the cask satisfied the above requirements, we analysed the response of the cask to a series of more severe accidents. Although these accidents extend well beyond the conditions specified in the regulations and have extremely low probability of occurrence, it was necessary to include them to determine the expected risk and releases from all transportation events, as well as to establish the thresholds for the release functions.

A summary of all the accidents examined is given in Table 4.1-1.

4.1.3 ASSURED FAILURE MECHANISM FOR CASK IN FIRE

In order to estimate radionuclide release in an accident, a model must be constructed for the failure process. We have assumed the following sequence of events for fire accidents:

1. The fire surrounds the cask and burns continuously at constant flame temperature.
2. The neutron shielding water jacket on the cask loses its water by boiling off through a pressure relief valve or by puncture of its containment.
3. The cask and canister temperatures increase, with temperatures at the ends rising more rapidly for the rail cask because of the higher surface-to-mass ratio there. For the truck cask, the ends are insulated by balsa wood and asbestos, so the middle heats up more rapidly.
4. Canister(s) reach failure temperature. Canister(s) fail by processes which depend on the waste form.
5. ^{134}Cs , ^{137}Cs , and ^{135}Ru escape from canister(s), depending on the waste form. Since the cask is hotter than the canister(s), it is assumed that it no longer presents an effective barrier to volatilization or particulates.
6. SKM remains at high temperature for a time of the order of the thermal time constant (about four hours for the rail cask, and two hours for the truck cask). Nucleides continue to escape by volatilization and particulate dispersion.
7. Cask cools. Volatilization stops. Particulate dispersion continues unless waste has melted into monolithic mass.

Specific assumptions for the truck cask thermal analysis are:

Very rapid disappearance of the balsa wood impact limiters

Very rapid loss of the neutron shielding water plus
disintegration of the outer neutron shield tank wall

Specific assumptions for the rail cask thermal analysis are:

Corrugated sleeve is presumed to remain in place during a
moderate fire (below 2000°F).

Severe fires (above 2000°F), on the other hand, are assumed
to completely remove the corrugated sleeve.

4.1.4 ASSUMED FAILURE MECHANISM FOR CASK BY IMPACT

The corner drop is postulated to be the most severe condition of the many possible accidental drops. A corner drop is defined as a drop in which the center of gravity is directly above the corner which impacts the unyielding surface first. In terms of radioactive release, a corner drop on the end of the container containing the bolted closure is the most severe. A corner drop on the closure end produces a bending moment which tends to fail the bolts in tension. Impact in any orientation is limited by the energy absorbing system. In the case of the IF-300 cask, impact energy is absorbed by lateral steel fins. The truck cask uses balsa wood to absorb the impact energy. Any type of energy absorbing system is limited by its maximum

deflection. Because the regulations require survival of a 30 ft drop, the energy absorbing systems are usually designed to "bottom out" as the result of such a drop. Increasing the drop height beyond 30 ft will result in the excess energy being absorbed by the cask itself. The excess energy results in much higher accelerations in the shipping container with higher potential for damage.

The canisters may act as a secondary barrier against the release of radioactive material but in this analysis it was assumed that any accident which could fail the closure lid would produce enough acceleration to cause a release of radioactive material from the canisters. Once the closure lid is removed there is no restraint against the canisters coming out of the shipping container, and subsequent motion of the container after the drop is likely to cause failure of the cans.

4.1.5 RELEASE FUNCTION $F_{ijq}(x)$ FOR TRUCK AND TRAIN ACCIDENTS

Table 4.1-2 describes the analytic release function forms $F_{ijq}(x)$ used in calculating the transportation accident probabilities P_{ijq} of Section 3.]1-1.

The indices i , j and q in $F_{ijq}(x)$ and P_{ijq} are defined as follows:

<u>i</u>	<u>Accident Type</u>
1	Truck Impact on Extremely Rigid Fixed Object
2	Impact of Train on Truck

- 3 Truck Fire
- 4 Train Impact on Extremely Rigid Fixed Object (Urban Area)
- 5 Train Impact on Extremely Rigid Fixed Object (Rural Area)
- 6 Train Fire

j Type of Release

- 1 Air Dispersion
- 2 Air Volatilization
- 3 Water Dissolution

g Solid Form of HLW

- 1 Reference Calcine
- 2 Reference Glass

TABLE 4.1-1

Thermal Accident and- Condition Investigated	Type	
	Rail	Truck
Normal Steady State Characteristics for both 10 year glass and 10 year calcine waste	X	X
Now - Fire Accidents 10 year glass and Calcine		
(a) Loss of neutron shielding water due to a faulty valve or puncture of the corrugated sleeve.	X	-
(b) Loss of mechanical cooling	X	
(c) Both (a) and (b) occur simultaneously.	X	
Fires (10 year glass only)		
• Moderate temperature fire Such as the 1475°F 30 minute fire type B	X	-
• 3 hour cool down after type B fire	X	
• Severe Migu temperature fires in excess of 1890°F that may induce waste canister damage after a long enough duration		X
• 1890°F	1 hr fire	-
• 2100°F	18 hr fire	8 hr fire
• 2400°F	12 hr fire	6 hr fire
• 2700°F	6 hr fire	4 hr fire
Impact Accident Conditions Investigated *		
Flat Bottom End Drop from Equivalent height for 30 MPH Impact(30 ft)	X	-
Equivalent height for 60 MPH Impact(>120 ft)	X	-
Equivalent height for 80 MPH Impact(>214 ft)	X	-
Corner drop with center of gravity directly above the corner which impacts the unyielding surface.	X	X
*All impact drops are on to an unyielding surface.

FIG. 1 BASELINE RAIL SHIPPING CASK

FIG. 1 NFS-4 SPENT FUEL SHIPPING CASK

TABLE C-1
TABLE C-1

ANALYTIC FORMS USED TO MODEL NEW TRANSPORTATION ACCIDENT RELEASE FUNCTIONS*

I	J	K	x_{f1}	Accident	Range of x (Severity)	Release Function $F_{1,ij}(x)$	$MF_{1,ij}$
8, 13	1	1	180	Train/Track Extr. Rigid Impact (Dispersion) (Calcine)	$0 \leq x \leq 67$ (mph)	0	
9, 12	1	1	120		$67 \leq x \leq 75$ $75 \leq x \leq x_f$	$0.0625 (x-67)$ 0.5	1.0×10^9
8, 13	1	2	180	Train/Track Extr. Rigid Impact (Dispersion) (Glass)	$0 \leq x \leq 67$ (mph)	0	
9, 12	1	2	120		$67 \leq x \leq x_f$	$7.692 \times 10^{-6} (x-67) + 1.0 \times 10^{-4}$	1.0×10^6
10	1,2	1,2		a) All Train Fire Accidents	All	0	0
14	1	2		b) Train Fire (Dispersion) (Glass)			
14	1	1	2,000	Train Fire (Dispersion) (Calcine)	$0 \leq x \leq 1730$ (10^3 °F-min) $1730 \leq x_f$	0 1	1.0×10^{-2}
14	2	1	2,000	Train Fire (Volatilization) (Calcine)	$0 \leq x \leq 1730$ (10^3 °F-min) $1730 \leq x_f$	0 1	1.0×10^0
14	2	2	2,000	Train Fire (Volatilization) (Glass)	$0 \leq x \leq 575$ (10^3 °F-min) $575 \leq x \leq 1,875$ $1,875 \leq x \leq x_f$	0 $(6.1312673 \times 10^{-7})x^2$ $-(7.3048128 \times 10^{-4})x$ $+(2.1729055 \times 10^{-1})$ 1	1.0×10^9

* See Section 2.1 for explanation of indices I, J, and K.

¹ x_f = Severity threshold beyond which the probability density function is not equal to 0. Established through best engineering judgment.

The radionuclide releases from transportation accidents fall into two categories: fire-induced and impact-induced.

4.1.5.1 FIRE - INDUCED

As mentioned earlier, in a transportation accident, both the cask and canister must be breached in order for radionuclides to be released.

In the case of fire-induced releases, as discussed under "Results" below, the cask seal is breached rather early in the fire. Consequently, breach of the canister determines the threshold for release.

Both the heat transfer through the cask and the canister failure mechanisms are time-dependent processes. However, because of the Arrhenius nature of the canister failure mechanisms, as discussed earlier, it is possible to establish a discrete canister failure temperature. This allows one to separate the two parts of the analysis, and to consider the time dependence of the heat transfer process by itself. We have adopted a "failure locus" approach, in which the time-to-canister-failure for fires of several temperatures are calculated. These combinations of T and t then form a failure locus on a T - t plot. Whenever a cask experiences a fire whose characteristics are above and to the right of this locus, canister failure is deemed to have occurred, and this constitutes the threshold for release

of radionuclides.

The canister failure temperature has been taken to be $1200 \pm 50^\circ\text{C}$ for the glass-containing case and $1300 \pm 50^\circ\text{C}$ for the calcine-containing case. The glass case is based primarily on failure due to waste-on-canister corrosion, with small contributions from external oxidation and creep-rupture. The calcine case is based primarily on external oxidation, with contributions from the other two mechanisms.

Given an accident that exceeds the threshold failure locus, the magnitude of the release depends primarily on the release mode, the properties of the waste form, the mechanical configuration assumed for the cask, canister(s) and waste, and environmental factors at the time of the accident. Since many of these factors are not well-defined, the magnitude of release is subject to uncertainties which are probably as large as a few orders of magnitude in some cases. Estimation of uncertainties is very difficult without a more precise knowledge of the likely range of these factors than is presently available.

4.1.5.1.1 FIRE - VOLATILIZATION

The fire-volatilization release functions are based primarily on the work of Gray at Battelle Northwest (Gray, 1976). He measured volatilization from small samples of glass and calcine as a function of temperature.

Because of the high values of the threshold temperatures and of temperatures where volatilization is significant, it is assumed that the cask no longer presents an effective barrier to volatilization in this region. The canister is assumed to be lying on its side, and the surface of the waste is exposed on the upper side (see sketch). These assumptions are somewhat arbitrary. A different choice could raise or lower the calculated release by a few orders of magnitude, but this is believed to be an "average" situation.

The exposed surface is taken to be 15 x 300 cm. It is necessary to scale Gray's small sample results to this situation. The approach taken is to do this at 1200°C, and then to derive an Arrhenius expression that fits Gray's data over a range of temperatures.

Gray's glass sample was 2 grams in mass, with a

surface area of 1.98 cm^2 and a thickness of about 3.2 mm. In a time of 4 hours, the Cs loss was about 20 mg/cm^2 . He states that a "thick" sample would be about 1 cm thick, and would have twice the loss. His experiments indicate that the process is surface-controlled.

The inventory of Cs in a fresh waste canister can be determined approximately by scaling up Gray's Cs inventory

$$(360.0 \text{ mg}) \times \frac{0.2 \text{ m}^3 \times 3300 \text{ Kg/m}^3}{.002 \text{ Kg}} \times \frac{155 \text{ Kg/MTU}}{207 \text{ Kg/MTU}} = 8.77 \times 10^6 \text{ mg}$$

The fraction lost is then obtained by assuming that the surface loss rate is the same for the large and small samples:

$$\frac{2 \times 20 \text{ mg/cm}^2 \times .15 \text{ cm} \times 300 \text{ cm}}{8.77 \times 10^6 \text{ mg}} = \underline{\underline{0.02}}$$

A fit to Gray's data for Cs loss in dry air gives an Arrhenius factor $\exp(-29,200/T)$. Matching up at 1200°C gives an equation:

$$\text{Release Fraction} = 8.13 \times 10^6 \exp(-29,200/T). \quad (\text{Cs in glass})$$

For calcine the calculation is similar, except that the process is diffusion-controlled rather than surface-controlled. At 1200°C , Gray's sample lost

about 18 mg/cm^2 in four hours out of an inventory of 142 mg of Cs. The amount lost for a "thick" sample would be twice as much, but the inventory would also be twice as much. The fraction for a "thick" sample would be:

$$\frac{2 \times 18 \text{ mg/cm}^2 \times 2 \text{ cm}^2}{2 \times 142 \text{ mg}} = .253$$

For the canister, the effective thickness would be about 25 cm. Assuming a diffusion-controlled process for which a 1 cm thick sample is a "thick" sample, the canister release fraction would be given by:

$$0.253 \times \frac{1 \text{ cm}}{25 \text{ cm}} \approx \underline{\underline{.01}}$$

In the same way as before, the equation is derived:

$$\text{Release Fraction} = 1.74 \exp (-7500/T) \text{ (Cs in calcine)}$$

In the same manner, the Ru release from calcine is given by:

$$\text{Release Fraction} = 2.36 \times 10^6 \exp (-27,900/T) \\ \text{(Ru in calcine)}$$

4.1.5.1.2 FIRE - PARTICULATE DISPERSION

For glass, this release function is taken to be zero,

Because the glass would be molten at the failure temperature. When it cooled, it would solidify as a large mass.

For calcine, a step function release fraction of 0.01 of all nuclides is assumed after the canister failure. This is based on the following logic: About 0.5 of the calcine is of respirable size to begin with. When heated to the canister failure temperature, it will begin to get sticky and coalesce. At still higher temperatures, it will melt and no longer be of respirable size. The cask will offer some barrier to movement of particulates until it melts completely. All things considered, 0.01 seems like a reasonable estimate.

4.1.5.1.3 FIRE - DISSOLUTION

The situation modeled here is failure of the cask and canister by fire, followed by immersion in water for one day. In the case of glass, the release is taken to be a step function value of 10^{-7} of all nuclides at the failure threshold. This is based on the leaching behavior of the glass, as discussed earlier, using an area for the failed cask seal of $3 \times 10^{-3} \text{ m}^2$, and a water temperature of 288K.

For calcine, a step function value of 0.05 is used

for the Sr and Cs at the threshold. This is an estimate, taking account of protection offered by the cask. The other nuclides are ignored, because of the much greater solubility of Sr and Cs, as well as their higher hazard potential for 10-year old waste.

4.1.5.2 IMPACT - INDUCED

The thresholds for impact-induced releases are calculated from the corner-drop model and corrected for vehicle crush using the equation discussed earlier. We have used a threshold of 67 miles per hour for the rail cask and ____ miles per hour for the truck case. There was no correction for the fact that real objects are not completely rigid.

4.1.5.2.1 IMPACT - PARTICULATE DISPERSION

For glass, the step-ramp function used reflects the fact that some glass has been crushed to respirable size before the impact threshold is exceeded. At higher velocities, the amount of respirable glass increases.

For calcine, the ramp-step function used reflects the fact that half the calcine is of respirable size initially. As the velocity rises above the threshold value, more of it is released, until the maximum of 0.5 is reached.

4.1.5.2.2 IMPACT - DISSOLUTION

In the case of glass, the ramp function reflects the increase in surface area due to fracturing, which increases the total dissolution during the one-day immersion. The function applies to all nuclides because the glass is assumed to dissolve by etching.

For calcine, a step function is used for Cs and Sr to reflect the high solubility of these species.

4.1.6 RISK ASSESSMENT - RISK CALCULATIONS (P_{ijq})

For all transportation accidents, the probability of release is

$$P_{ijq} = \prod_{m=1}^{M_i} a_{imq} \int_{x=0}^{x=\infty} P_i(x) F_{ijq}(x) dx$$

where:

m = index of the conditional probability a_{imq}

($M = 1, \dots, M_i$)

a_{imq} = the m^{th} conditional probability of the occurrence of an accident of type i involving SHLW of solid type q

$p_i(x)$ = The probability density function (PDF) for transportation accident severity for SHLM of solid type q undergoing accident of type i with respect to accident severity

$F_{ijq}(x)$ = The release function with respect to accident severity for SHLM of solid type q undergoing release type j given a transportation accident of type i

The probability density functions (Center et al, 1976) used in the prior analyses consisted of discrete functions obtained via graphical differentiation of the cumulative distribution functions (CDF's) of accident frequency vs. severity from Clarke et al (1976). In order to permit maximum flexibility of use of PDF's in subsequent analyses, it is desirable to express them as analytic functions. This has been accomplished for each truck and train accident type of interest by the following procedure:

Values of the CDF for each accident type were tabulated by sampling points from all graphically-depicted CDF's, or by using CDF tables directly when available. Certain assumptions were made to make an appropriate division of frequencies among urban and rural train impact accidents (see Table 4.1-3).

Graphical methods were then used to fit Weibull-type CDF's (form: $F(x)=1-\exp [-(x/\alpha_1)^{\alpha_2}]$; $\alpha_1, \alpha_2 > 0$) to the tabulated CDF values obtained for each accident type. Weibull-type functions were chosen because of their ability to fit a wide

variety of distributions and because of their use in the Sandia analysis (Clarke et al, 1976). The procedure used for obtaining estimates of α_1 and α_2 is outlined below.

The PDF was then computed as the analytic derivative with respect to x of the CDF:

$$\begin{aligned} \text{(form: } p(x) &= (\alpha_2/\alpha_1) e^{-(x/\alpha_1)^2} (x/\alpha_1)^{-1}; \\ &\alpha_1, \alpha_2 > 0) \end{aligned}$$

The calibration results are summarized in Table 4.1-3. In some cases, a trade off had to be made between the desire to fit the function to all points equally well and the desire to use only one analytic PDF function throughout the entire severity range. In those cases where one analytic function did not fit the data well over the entire severity range, calibration was done using a single function fit to the tail end of the curve (i.e., for large severities). The rationale for choosing this severity range for calibration was to provide the best PDF fit where the value of the corresponding release function curve was appreciably different from 0 and to minimize the chance of errors in evaluating the area under the PDF curve out to large severities.

In addition to the substitution of new continuous PDF curves in the probability calculations, the following changes were made to the assumptions outlined in pp 2-16 to 2-19 of Center et al (1976). (All other assumptions remain the same. Sources of these new data are stated in parentheses.)

Truck

Only one canister of SHLW will be transported in a truck cask

Train

The overall probability of a rail car accident per rail car-mile rises to 1.5×10^{-6} Clarke et al (1976), Vol. IV, pg. 16).

Only four canisters of SHLW will be transported per rail cask.

The probability of an "extremely rigid" impact accident, given that an accident has occurred, is the same for both urban and rural trains ($1/2 \times 0.051 = 0.0255$ for each) (follows from previous assumption on equal frequency of accidents in urban and rural areas -- Center et al (1976), p. 2-16.

4.1.6.1 EVALUATION OF PARAMETERS FOR CONTINUOUS PDF'S

The equation for the CDF

$$F(x) = 1 - e^{-(x/\alpha_1)^{\alpha_2}}$$

can be transformed into the following format to facilitate graphical estimation of α_1 and α_2 :

$$\ln [-\ln(1-F(x))] = \alpha_2 \ln x - \alpha_2 \ln \alpha_1$$

Let $g(x) = \ln [-\ln(1-F(x))]$. A plot of $g(x)$ vs $\ln x$ using tabular CDF data is then made. If all points are approximately collinear, then fit a line visually through them and evaluate its slope M and intercept B . It follows, that:

$$\alpha_2^0 = M;$$

and

$$\alpha_1^0 = e^{-B/M},$$

where α_1^0 and α_2^0 represent first-cut visual estimates of α_1 and α_2 . If all points cannot be fit well with one straight line, two alternate methods can be employed:

In the first approach, if the data points on the graph

consist of two or more subsets of collinear points (which is usually the case), one can fit a separate straight line to each subset of points by the method outlined above and then use the above equations to estimate α_1 and α_2 for each line. This will result in a piecewise-continuous set of CDF's that provide a good fit to the data over the entire severity range. The PDF functions obtained by differentiating these CDF's, however, will not be continuous at the break points. Thus, this approach is undesirable if a single, continuous PDF is desired for use in subsequent analyses, which is the case here.

To remedy this problem one can, as a second approach, choose a single CDF (and, hence, a single PDF) by fitting one straight line to collinear points in the severity range over which the best fit to data is desired (the "range of calibration" -- Table 4.1-3). This method will provide a function with good fit to data within the range of calibration but possibly at variance with the data outside this range. This method was chosen over the former one, using each accident type's higher severity levels as the range of calibration. This range was chosen because the normalized release functions always have values near zero for low accident severities, but values near unity for high severities. Hence, in any analysis requiring multiplication of the PDF and release function-curve

for an accident type, accuracy of the PDF at low severity levels is of far less importance than accuracy at higher levels, especially for calculations involving integration out to arbitrarily large severity values.

Center, J. L., Crawford, B. S., Ross, B. and Sutherland, A. A., Analysis of Nuclear Waste Management", The Analytic Sciences Corporation, TR-772-2-1, October 1976.

Clarke, R. K., Foley, J. T., Hartman, W. F., and Larson D. W., "Severities of Transportation Accidents", Sandia Laboratories, SLA-74-0001, July 1976.

TABLE A-1

SUMMARY OF ESTIMATED PARAMETER VALUES FOR CONTINUOUS TRANSPORTATION
ACCIDENT PROBABILITY DENSITY FUNCTIONS

ACCIDENT TYPE	SOURCE IN SRA-71-0011 (7/76 VERSION)	ADDITIONAL SOURCE FIGURE	ESTIMATED PARAMETER VALUES		RANGE OF CALIBRATION
			n_1	n_2	
Truck					
Impact on extremely rigid fixed object	Vol. III, Fig. 4 (theoretical curves)		45.36	3.36	50-70 (mph)
Impact of train on truck	Vol. IV, Table II		10.53	0.8245	5-80 (mph)
Fire	Vol. III, Integrate product of Fig. 5 CDF and PDF corresponding to dotted line in Fig. 7	Used numerical integration	19.1	1.05	10K-150K (OP/min)
Procedures	Vol. III, Fig. 15	Convert Fig. 15 to corresponding CDF	0.797	0.281	120-300 (sec ⁻¹)
Train					
Impact on extremely rigid fixed object (urban areas)	Vol. IV, Table III	Half of all accidents in each 5 mph group under 40 mph occur in urban areas;	14.37	1.25	7.5-40.5 (mph)
Impact on extremely rigid fixed object (rural areas)	Vol. IV, Table III	the other half occur in rural areas. All accidents at speeds exceeding 40 mph occur in rural areas	27.21	0.998	25-80.5 (mph)
Fire	Vol. IV, Fig. 10		45.2	0.52	90K-200K (OP/min)
Procedures	Vol. IV, Fig. 28 (Assumption B)	Convert Fig. 28 to corresponding CDF	0.0266	0.171	63-172 (sec ⁻¹)

Notes:

$$F(x) = 1 - e^{-(x/n_1)^{n_2}}; \text{ (cumulative distribution function)}$$

$$f(x) = \frac{dF(x)}{dx} = \frac{n_2}{n_1} e^{-(x/n_1)^{n_2}} (x/n_1)^{n_2-1}; \text{ (probability density function)}$$

$$n_1 \cdot n_2 = n$$

3.2.0.1 HEAT TRANSFER MECHANISMS AND BOUNDARY CONDITIONS

After removal of cooling water, air flow by natural circulation around the canisters is a prime method by which heat is lost to the surroundings. The flow patterns are quite complex, with cooler air moving rapidly down the sides of the basin and air circulating more slowly back up while being heated by the waste canisters. Figure 3.2-2 is a sketch of the natural convection process indicating air entering the bottom of the basin at an initial ambient temperature $T_0 = 120^\circ\text{F}$ and leaving the top of the canister array at temperature T_A after exchanging heat isothermally with canisters at temperature T_C . Using the notation:

q = heat exchange rate between canisters and air
(BTU/hr)

W_A = mass flow of air (lbs/hr)

C_A = heat capacity of air (BTU/lb- $^\circ\text{F}$)

A_f = flow area between canisters (ft^2)

h = heat transfer coefficient between air and
vertical
cylinders (BTU/hr- ft^2 - $^\circ\text{F}$)

A_C = surface area of canisters (ft^2),

a heat balance on the air gives

$$q = C_A M_A (T_A - T_0) = h A_c (T_c - T_A) \quad (3.2.0.1)$$

- Heat transfer across the gap between the air and waste block in Figure 3.2-1 (boundary condition B1) across an area πR^2 simulated the right hand side of Equation (3.2.0.1) using a heat transfer coefficient

$$h' = h A_c / \pi R^2 \quad (3.2.0.2)$$

- Heat transfer from air to ambient in Figure 3.2-1 (boundary condition B4), also across an area πR^2 , simulates the left hand side of Equation (3.2.0.1) with a heat transfer coefficient

$$h'' = \frac{C_A M_A}{\pi R^2} \quad (3.2.0.3)$$

- The unknown in Equation (3.2.0.3) is the natural circulation flow rate, W_A . This quantity was estimated by setting the h determined from natural convection heat transfer calculations with the h determined from forced convection (turbulent flow) calculations; the latter depends on W_A and, thus, a solution for the mass flow can be obtained.

- Both h' and h'' , the convective heat transfer coefficients for naturally circulating air, are temperature-dependent.

- Boundary condition B5 from the waste block into the gap represents radiation from can tops to space at 75°F. A view factor of 0.5 and an emissivity of 0.65 have been assumed.
- Boundary condition B2 accounts for both radiation and convective losses from the waste block across a gap to the concrete basins. Heat transfer from canister tops, bottoms, and sides are included here.
- Boundary condition B3 appears to be a convective loss term at the outer side of the concrete wall to a 120°F ambient. However, in reality, the heat transfer coefficient has been selected to match the thermal resistance of 4 additional feet of concrete. Therefore the concrete temperatures are those of the inner 4 ft of an 8 ft thick basin.

3.2.0.2 MATERIAL PROPERTIES

Table 3.2.0.1 gives the material properties, all assumed temperature-independent, used in the interim storage thermal calculations. Note that:

- Air and waste thermal conductivities were inputted as very high values to give essentially uniform spatial temperatures. Hence, derived temperatures are actually spatial averages and radial and axial gradients are neglected.

• Glass and calcine densities are averaged over the waste block and reduced because of the large air volume.

• Glass and calcine heat capacities include a contribution from the stainless steel canisters.

TABLE 3.2.0.1

<u>Material Name</u>	<u>Conductivity (BTU/HR-FT-°F)</u>	<u>MATERIAL PROPERTIES</u>	
		<u>Density (LB/FT³)</u>	<u>Specific Heat (BTU/LB-°F)</u>
Glass	High	83.8	0.170
Calcine	High	43.5	0.156
Air	High	0.06	0.237
Concrete	0.792	131.0	0.210

Cases Run

Loss-of-coolant accidents were simulated by HEATING5 for times up to 40 hours after start of the thermal transient. In addition, a steady-state run was performed for each transient depicting the temperature distributions at a very long time after the accident. The various cases differed in the type and age of waste; these were:

- 1 yr. glass
- 2 yr. glass
- 5 yr. calcine
- 10 yr. glass

Figures 3.2-3 through 3.2-6 show the temperature-time profiles and the long time steady-state temperatures for the four cases. Illustrated are the average canister temperatures and the average air temperature.

- For purposes of evaluating the onset of canister failure it is desirable as well to have an estimate of the peak canister temperature rather than the average. An exact computation of the radiation transfer from canister to canister in a large array is difficult. However, a rough determination of the T from peak central canister to average canister temperature was carried out using an approximation that modeled the canisters as concentric radiation shielding rings. Details are not given, but the results are:

<u>Age of Waste (yrs)</u>	<u>T(°F)</u>
1	11
2	17
5	36
10	72

- As can be seen, the T correction is small.
- The difference between calcine and glass temperatures (at the same age, i.e., same heat generation rate) lies only in the transient temperature profile. The calcine is of lower density and its temperature will, therefore, approach state-state more rapidly. The steady-state or long term temperatures will be

3.2.0.2 EFFECT OF LOSS-OF-COOLANT ACCIDENT

Because of the longer times involved in loss of coolant events, lower failure temperatures are used. Assuming canister failure occurs at a canister temperature of 1150°C (2102°F) for glass (due to corrosion of the stainless steel) and 1250°C (2282°F) with calcine (due to creep and oxidation), the following table gives the canister survival/failure results for the cases examined, assuming a steady-state temperature distribution is attained.

TABLE 3.2.0.2

CANISTER FAILURE FOR INTERIM STORAGE LOSS-OF-COOLANT ACCIDENT

<u>Waste Age</u>	<u>Waste Type</u>	
	<u>Glass</u>	<u>Calcine</u>
1 yr.	Failure	-
2 yr.	Failure	-
5 yr.	Near Failure*	No Failure
10 yr.	No Failure	No Failure

- * Corrosion rate effects must be accounted for more exactly than a single failure temperature criterion allows, in order to ascertain actual failures.

3.2.1 Expected Releases

The probability of having a severe earthquake at the interim storage facility away from tectonic plate boundaries is 10^{-5} per year (Schneider and Platt, 1974)

Assume there is only a 0.1 probability of loss of coolant resulting from a severe earthquake.

Given a major earthquake, it is assumed that there is no restoration of cooling water.

Meteorological and demographic assumptions are given in appendices to Sec. 7. Dose conversion factors are presented in Appendix 1.3.5.

Events causing pool drainage also rupture the air seal on the structure that normally prevents release to the atmosphere.

Tornados and hurricanes are assumed to be no danger to the pool, which is below grade and strong enough to withstand impacts of missiles from these causes.

The heat transfer calculations for the reference design indicate that failure of about half the canisters would occur for the glass waste, and no failures would occur for the calcine waste. Because of the prolonged period at elevated temperature, it is assumed that all the canisters that rupture would volatilize their entire inventory of Cs, Te, and Ru. The release functions

were therefore taken as 0.5 for Cs, Te, and Pu for the glass waste, and zero in the calcine case.

3.20-2

Figure 1 Geometry for Interim Storage Problem

FIG. 2D.7. NATURAL CONVECTION IN INTERIOR STORAGE CANISTER ARRAY

FIG. 3.2.3 INTERIM STORAGE LOSS-OF-COOLANT TEMPERATURE PROFILES

FIG. 1.2.0.4 INTERIM STORAGE LOSS-OF-COOLANT TEMPERATURE PROFILES

Fig. 3.5.5. 1. TERM STORAGE LOSS OF - COOLANT TEMPERATURE PROFILES.

FIG. 2.1.1: INTERIM STORAGE LOSS OF COOLANT TEMPERATURE PROFILES

3.2.1.1 RISK CALCULATIONS (P_{ijqs})

For all non-transportation pre-emplacement accidents,

$$P_{ijq} = \prod_{m=1}^{M_i} a_{imq}$$

where =

m = index of the conditional probability a_{imq} ($m = 1, \dots, M_i$)

a_{imq} = the m^{th} conditional probability of the occurrence of an accident of type i involving HLW of solid type q

$p_i(x)$ = the probability density function for transportation accidents of type i with respect to accident severity

$F_{ijq}(x)$ = the release function with respect to accident severity for HLW of solid type q undergoing release type j given a transportation accident of type i

The indices i , j , and q are defined as follows:

i Accident Type

- 1 Handling at Reprocessing Plant, Crane Stall,
Filtration Fails
- 2 Handling at Reprocessing Plant, Crane Stall,
Filtration Works
- 3 Handling at Reprocessing Plant, Crane Drop,

Filtration Fails

- 4 Handling at Reprocessing Plant, Crane Drop,
Filtration Works
- 5 Interim Water Basin Storage, Loss of Cooling Circulation,
Filtration Fails
- 6 Interim Water Basin Storage, Loss of Cooling Circulation,
Filtration Works
- 7 Interim Water Basin Storage, Drainage of Storage Pool

j Type of Release

- 1 Air Dispersion
- 2 Air Volatilization

q Solid Form of HLW

- 1 Reference Calcine
- 2 Reference Glass

COMBINATIONS FOR WHICH THE EXPECTED RELEASE IS ZERO

When this occurs, it is due to either a zero element appearing in the conditional probability tree (a_{imq}) or a release function that is equal to zero or which assumes a zero value until extremely large severity values for a particular transportation accident.

COMBINATIONS OF i and q THAT HAVE ZERO EXPECTED AIRBORNE RELEASE

<u>i</u>	1	2	3	4
<u>q</u>	2	2	2	2

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4.2. Encapsulation

Shipping casks designed specifically for the shipment of solidified high-level waste (SHLW) have never been built. In fact, a recent survey conducted of the available literature indicated that only two conceptual cask studies (Ferreira et al, 1970, and Pereira and Bickner, 1972) have been published to date. Several scale models for SHLW shipment have recently been built and exhibited under an ERDA funded research contract, but information on full scale designs have not been made public.

Since SHLW cask designs are, however, available to reproduce the casks currently available for spent fuel, we decided to modify currently licensed spent fuel shipping casks to arrive at our reference design.

A recent ERDA study (ERDA, 1976) states that most shipments of SHLW are expected to be made by rail. Truck shipments are technically feasible, but less efficient and may well be economically less attractive because of the smaller payload to cask ratio. However, since the present study deals only with evaluation of the potential risk to the public, and does not include an economic optimization, we have analyzed both a truck and a rail cask as baseline designs.

Selection was made from licensed spent fuel casks which had reached the fabricated state at the time the reference design was needed. A review of spent fuel technology indicates the following casks had been built:

For Rail Shipment

IF-300

NEL-CA-70

For Truck Shipment

NFS-4 (NAC-1)

TN 2/9

NLI 1/2

For shipment of SHLW at 10 years by rail, either one of the above casks could easily satisfy the thermal requirements of the waste. However, initially we had planned to investigate shipment of early age waste (approximately one year) and on that basis selected the uranium shielded, IF-300 cask. Furthermore, we felt that the internal cavity size and thermal characteristics of the IF-300, in general, could better accommodate the transport of 12-inch diameter waste cans. In the future, we do plan to examine alternate cask configurations, but these are not expected to change the risk assessment appreciably.

The truck cask was selected somewhat more arbitrarily since all three casks are lead shielded and similar in design. Selection of the NFS-4 cask was made on the basis of cavity size; its 13.5-inch diameter opening was considered a good match for transporting one 12-inch diameter canister.

4.1.0 REFERENCE DESIGN - RAIL SHIPPING CASK

The baseline rail shipping cask is an LLL modification of the GE IF-300 spent fuel cask described by General Electric (1973). The salient features of this cask, depicted in Figure 4.1-1, are the following:

A 37.5-inch diameter, 10 ft. long cavity containing four 12-inch

diameter SHW canisters

A dry stainless steel insert surrounding the canisters and acting as a heat transfer medium

The use of a depleted uranium gamma-ray shield 4 inches thick

The use of a 5-inch thick water neutron shield on the outside of the cask, enclosed by a corrugated stainless steel outer sleeve

Finned ends to act as impact limiters and also to augment heat transfer to the ambient air

A stainless steel GrayLoc seal ring to prevent release of contents (GrayLoc is a trademark of the Gray Tool Co., Houston, Texas)

Mechanical cooling (fans) during transportation

4.1.1 REFERENCE DESIGN - TRUCK SHIPPING CASK

As mentioned above, the truck shipping cask studied was heavily based on a Nuclear Fuel Services spent fuel cask, the NFS-4 (NAC-1). The important characteristics of the reference design are:

120-inch long by 13.5-inch diameter internal cavity

Lead gamma-ray shielding 6.65 inches thick

Ethylene glycol/water neutron shield 4.5 inches thick

A single 12-inch diameter fuel or waste canister contained therein

8-inch thick steel end caps to which are attached 12-inch balsa wood impact limiters and an asbestos shield

This cask was rated for a maximum internal decay heat load of 12.4 kW. A sketch of the NFS-4 cask is given in Figure 4.1-2. It was adopted from Nuclear Fuel Services (1972). The only major modification made to the NFS-4 design was to reduce the cavity length to 10 feet to coincide with present SHLW canister designs.

4.2.0 THERMAL ANALYSIS - RAIL CASK

The HEATING5 computer code was the chief computational tool for performing thermal analysis on the rail shipping cask. The code is described in detail by Turner (1976). Basically, HEATING5 is a generalized heat conduction code designed to solve steady-state and/or transient heat conduction problems in one-, two-, or three-dimensional Cartesian or cylindrical coordinates or one-dimensional spherical coordinates. The thermal conductivity, density, and specific heat may be both spatially and temperature-dependent. The thermal conductivity may be anisotropic. Materials may undergo a change of phase. Heat generation rates may be dependent on time, temperature and position, and boundary temperatures may be time-dependent. The boundary conditions, which may be surface-to-boundary or surface-to-surface, may be fixed temperatures or any combination of prescribed heat flux, forced convection, natural convection, and radiation. The boundary condition parameters may be time- and/or temperature-dependent. The mesh spacing can be variable along each axis. The code is designed to allow a maximum of 100 regions, 50 materials, and 50 boundary conditions. The maximum number of lattice points can be easily adjusted to fit the problem and the computer storage requirements.

The baseline waste form with which the current work was carried out is 10-yr.-old borosilicate glass solidified waste rated at 3.5 kW output. With the exception of the temperature distribution in the waste itself, the results are applicable to suitably diluted calcine, i.e., dilution to the 3.5 kW level. Wastes with a shorter cooling time were examined as part of preliminary studies. These early

studies, however, were intended primarily to test the adequacy of various cask/canister concepts for normal, non-accident operation. Design conservatism dictated the use of 10-yr.-old waste material in conjunction with the cask described previously.

4.2.0.1 MODEL GEOMETRY

As can be seen in Figure 4.1-1, the actual rail shipping cask is a complex, three-dimensional configuration. It was necessary to simplify the geometrical description of the cask and canister so that HEATING5 could perform thermal analyses with a minimum of computer time. It was decided that the most convenient way of proceeding was to encode the cask geometry with two separate views in cylindrical coordinates:

An R-Z representation, i.e., treating the cask as though symmetry about the axis (centerline) existed and looking at temperature distributions axially and radially from a side view.

An R- θ representation, i.e., treating the cask as though it were infinitely long, thereby ignoring end effects, and examining the radial and angular temperature variations from a top view.

It was felt that the R-Z geometry gives a good overall picture of cask temperature profiles, although it misses a detailed picture of the distributions around the individual canisters.

The R- θ geometry is a better description of the canister temperatures, especially away from the cask ends and near the mid-plane. A combination of the two representations gives a complete thermal analysis.

Figure 4.2-1 shows an end view, roughly to scale, of the cask interior. The HEATING5 code is restricted to define only region boundaries that are parallel to coordinate axes; in the present case, a cylindrical geometry with the origin on the cask centerline was deemed best. For an R-Z view, regions must be made up of lines of constant radius (circles centered on the axis) and lines parallel to the axis. It is seen that the waste canisters themselves do not obey this restriction; they are cylinders, but are not centered on the axis.

The solution was to convert the canister region into "equivalent" cylinders able to be handled by the code. Thus, that portion of Figure 4.2-1 labelled "insert" and "waste" (the cavity) was mapped into three new regions:

1. An inner cylinder consisting of insert material (stainless steel)
2. An annulus of waste material surrounded by thin annuli representing canisters and air gaps
3. An annulus of insert material

Requirements on the dimensions of the waste material annulus were the following:

1. The volume of the equivalent annulus had to equal the total volume of the four waste canisters.
2. The first moment of the mass distribution of the waste annulus had to equal that of the actual waste cylinders.

These constraints enabled the inner and outer radii of the waste region to be obtained. Figure 4.2-2 shows the R-Z representation used by HEATING5, with dimensions. Note that outside the cavity, the R-Z geometry does not require approximations to acquire cylindrical symmetry. Figure 4.2-2 depicts one quadrant of the entire cask cross-section.

To utilize R- θ coordinates, the waste canisters in Figure 4.1-1 must be described in terms of lines of constant radius (circles centered about the axis) and constant angle lines emanating from the axis in a "spoke-like" fashion. The intersections of these two types of possible region boundaries form wedges. Hence, the mapping converted the actual waste cylinders into "equivalent" wedges with three dimensions to be determined: inner radius, outer radius, and wedge angle. The unknowns were obtained from the following sufficient conditions:

1. The wedge areas were forced to equal the waste cylinder areas.

2. The maximum distance between wedge boundary and the waste cylinder boundary was minimized.

Figure 4.2-3 gives the R- θ geometry encoded into HEATING5. The smallest repeated unit, 45°, is shown; this symmetry cell includes half of one of the four waste canisters.

Principally, the R-Z representation was employed to model cask/canister thermal response, whereas the R- θ geometry modelled cases where waste temperatures in the interior of canisters were of interest.

4.2.0.2 BOUNDARY CONDITIONS

Boundary condition numbers (B1, B2, etc.) are indicated on Figure 4.2-2.

B1 - side of cask, includes radiation and forced convection to 130°F ambient from a corrugated sleeve. A 1 BTU/hr-in² incident time-invariant solar flux is assumed. Convection includes an increased area due to corrugations. The radiation view factor includes provisions for radiation to cask support structures as well as to ambient air. During a loss-of-mechanical cooling accident, the forced convection heat transfer coefficient is replaced by a natural convection coefficient.

B2 through B5 - these represent boundaries with fins. Heat

transfer coefficients for radiation and natural convection are adjusted for fin efficiencies and effective areas. Fin dimensions and numbers vary with position; hence B2 through B5 have different heat transfer coefficients. Solar heating and forced convection are not heat loss mechanisms on these boundaries. The differences among these boundaries are:

- B2 - 6" fins
- B3 - 6" + 9 1/2" fins
- B4 - 9 1/2" fins
- B5 - bare stainless

B6 and B7 - these are internal boundary conditions which allow for radiation (in addition to conduction) across the air gaps. The emissivity of the canister is assumed to be 0.6; that of the insert is taken to be 0.3. During an accident involving loss of neutron shielding water, radiation is also allowed across the neutron shield annulus, following a B7 internal surface-to-surface boundary condition.

During fires, the external boundary conditions change. The only heat transfer mechanism allowed is radiation from the flame at the flame temperature (1475°F to 2700°F) with an effective flame emissivity of 0.8. Forced and natural convection and solar heating are eliminated. In the case of moderate temperature fires (below 2000°F), the corrugated outer sleeve is presumed to remain intact except for loss of the water; the fire acts on the sleeve and heat is transferred

across the air filling the annulus formerly occupied by the water. In the case of severe fires (above 2000°F) the assumption is that the sleeve is destroyed and an external boundary is placed on the outer gamma shield liner, thus by-passing completely the neutron shield annulus.

4.2.9.3 RESULTS

Table 4.2-1 summarizes the results of the HEATING5 normal, steady-state computer runs, the non-fire accidents, and the Type B 1475°F, 30-minute fire. Results are also included for a 3 hour cooldown period following the Type B fire. The last column, post-fire steady-state, gives the temperatures at critical cask locations after infinite time cooldown; this condition is equivalent to a steady-state with both loss-of-shielding water and loss-of-mechanical cooling together.

No potential damage modes are apparent for any of the conditions reported in Table 4.2-1.

Figures 4.2-4 through 4.2-6 give temperature-time histories in the R-Z geometry for rail casks under severe fire conditions for regions that are most likely to form release pathways for radioactive materials. These are as follows:

Canister end-point ($R = 13.73"$, $Z = 60"$)

Canister mid-point ($R = 13.73"$, $Z = 0"$)

Seal ($R = 12.75"$, $Z = 60"$)

It can be seen that, for the cases examined, critical temperatures are exceeded first at the canister end-point. Table 4.2-2 summarizes the failure times for the various fires and waste forms.

TABLE 4.2-2
CANISTER FAILURE TIMES* FOR RAIL CASK IN FIRES

Fire Temperature °F	Waste Form	
	<u>Glass</u>	<u>Calcine</u>
2100	No Failure	No Failure
2400	5.4	1.7
2700	3.0	1.3

*Failure times are given in hours

Figure 4.2-6 shows the GrayLoc seal temperatures as a function of time for each of the three severe fire conditions, plus the 1 hr. 1890°F fire. Assuming a 1000°F failure point for this seal, failure occurs in the range of 0.5 to 0.7 hours for the severe conditions. Seal failure does not take place within 1 hour during the 1890°F fire; a rough extrapolation would yield a failure time in the vicinity of 1.25 to 1.5 hours at that temperature.

A least squares fit to the canister failure locus (Temperature, T, versus time, t) has been performed. The best fit to a

functional form $t = a/(T-b)$, where a and b are constants, is:

FOR GLASS

$$a = 2025$$

$$b = 202^{\circ}\text{F}$$

FOR CALCINE

$$a = 1668$$

$$b = 2261$$

where t has units of hours and T has units of $^{\circ}\text{F}$.

4.2.0.4 CONCLUSIONS

For the baseline design rail cask, corrosion failures of the canister can take place with glass waste, assuming fire temperatures in excess of 2100°F .

For calcine waste, canister failure by the oxidation, creep, and corrosion mechanisms occurs with fire temperature above 2100°F .

The corrosion failure locus for glass waste can be correlated very well by an expression of the form failure time $\times$ (fire temperature - 2025) = 2025, with failure time in hours and fire temperature in $^{\circ}\text{F}$ and is superior to time-temperature product to correlate failure results. The corresponding expression for calcine waste is failure time $\times$ (fire temperature - 2261) = 1668.

The baseline rail cask design is an adequate one for normal steady-state operation (with mechanical cooling needed) for the following reasons:

1. Calcine centerline temperature does not exceed the calcine bakeout temperature.
2. Canister maximum temperature is well below that required for rapid corrosion.
3. Stainless steel outer tank wall temperatures are quite moderate, approximately 170°F.

Failure of the GrayLoc seal takes place rapidly under severe fire temperatures, but not under 1475°F, 30 minute duration conditions.

4.2.1 IMPACT ANALYSIS - RAIL CASK

A free body diagram of the modified G.E., 1F-300 container is shown in Figure 4.2-7. The loads shown are the forces in kilopounds as the result of an acceleration of one g (32.2 ft/sec.^2). The weight of the modified cask is assumed to be 99.3 kips, while the lid weighs 7.4 kips and the contents weigh 25.45 kips.

Given that the loads are twice the weight of the cask times the acceleration plus one times the weight of the contents gives a reaction of

$$(99.3 - 25.45) \times 2 + 25.45 = 173.15 \text{ kips}$$

The component of this parallel to the end of the cask is 60.6 kips and the component perpendicular to the lid is 162.2 kips. The components of the lid weight are

$$7.4 \times 2 \sin 20.484 = 5.16 \text{ kips parallel to the end of the container}$$

and $7.4 \times 2 \cos 20.484 = 13.86 \text{ kips perpendicular to the end of the container}$

This last component can be added to the component from the slug, giving a total axial load of 37.66 kips. This means that the load on the cask must be 124.54 kips axial and 53.36 kips lateral.

The moment on the cask lid is;

$$M = 162.2 \times 24.75 - 60.6 \times 6 + 3.61 \times 5.18 = 3670 \text{ G}$$

and the axial load is;

$$P = 124.54 \text{ G} - 639 \text{ kips}$$

where G is the maximum acceleration during the drop and the 639 kips represent the sum of the preload on the bolts and the internal pressure. The impact energy will be less for this cask than the original

$$\frac{99,300}{140,000} = 70.9\%$$

but following the procedures used by General Electric (1973) to qualify the cask, the accelerations experienced while the impact absorber is acting will be the same for the modified cask as the original design. The maximum acceleration was predicted to be 57 g's until the impact absorber bottomed out.

After the impact absorber bottoms out, the closure lid itself must absorb the impact. By assuming that the stainless steel lid material is at yield, the peak acceleration can be estimated by equating the excess impact energy to the work done in deforming the lid. Because the cask is still falling on its corner the work done in deforming the lid is a function of the area of the corner which is crushed. See Figure 4.2-2 for a sketch of how the corner may crush when impacting on a rigid surface. The acceleration (a) can be related

to the crushed area (A) of the lid and the yield stress of the lid (σ). Conversely, given the acceleration which will fail the lid bolts, the crushed area can be calculated:

$$A = \frac{a \text{ Wt.}}{\sigma} = \frac{92 \cdot 99.3}{40} = 228 \text{ in}^2$$

This can be compared to 1924 in², the total area of the lid. The deformation can be calculated by trial and error from the following equations

$$A = r^2 \left(\frac{\pi}{360} - \cos \gamma / 2 \sin \gamma / 2 \right)$$

and

$$\delta = r \sin \theta \cos \theta (1 - \cos \gamma / 2)$$

If $r = 24.75"$, then $\theta = 49.05^\circ$

and $\delta = 2.83"$

If one assumes a linear relationship between impact force and crush deformation, the additional energy absorbed by crushing the corner of the cask itself can be calculated by:

$$\Delta E = 1/2 \cdot A \cdot \delta$$

and the energy required to bottom out the impact absorber is:

$$E = m g h$$

so the fractional increase in energy required to completely fail the closure bolts is;

$$\frac{\Delta E}{E} = \frac{\sigma A \delta}{2 m g h} = \frac{40 \cdot 228 \cdot 2.88}{2 \cdot 99.3 \cdot 521} = 25.4\%$$

Since the energy is a function of the velocity squared, the increase in the drop velocity is approximately 12%.

4.2.1.1 RESULTS

Specifically, the question was raised, that if the container is designed such that no release occurs for a 30-ft drop height, for what drop height will significant release occur? This information is needed for subsequent use with the probability estimates on accidental drops so as to predict the expected amount of radioactivity release associated with the transportation of this material.

In the case of the IF-300 cask, the energy absorbing system limits the accelerations to 57 g's. According to our analysis, an acceleration of 92 g's will produce sufficient moment to cause tensile failure of the closure lid bolts when resulting from a corner drop on the lid end of the container. Any additional velocity which remains after the energy absorber has bottomed out produces accelerations equivalent to direct impact of the shipping container. Simple calculations show

that a 12% increase in the impact velocity is all that is required to increase the accelerations from 57 g's to 92 g's. Because the container for these high level wastes is lighter than the G.E. IF-300 container, it will sustain a somewhat higher impact velocity of approximately 36 mph before the impact limiter bottoms out. The 12% increase in impact velocity is the same as a 25% increase in drop height. The implication of these simple calculations is that a 36 mph impact will produce no radioactive release while a 40 mph drop will produce very significant release. A very significant release is implied by the complete failure of the bolts holding the closure lid in place. The primary reason that this release is so sensitive to the impact velocity is that a rigid impact surface is assumed. Any realistic impact surface will yield when a 50 ton container is dropped on it. Yielding will reduce the loads significantly and it will probably reduce the loads as a result of the impact limiter bottoming out in larger proportion than the reduction of the normal loads.

In addition to the corner drop condition, several end drop conditions were examined using an asymmetrical finite element code called HONDO V. This code is a modified version of the HONDO code (Key, 1974), written by John Hallquist of LLL. The principal change is allowance for relative motion (slipping or separation) at interfaces. This analysis was used basically to verify the structural integrity of the reference design and to see if vertical drops up to 80 mph equivalent height induce sufficient damage to the cask to cause a release. This study

was conducted for the rail cask only, since it was soon learned that the corner drop condition imposes the more severe loading condition and would be used to determine the thresholds on the release function curves.

The results of the flat end-drop indicate several points worth noting.

30 MPH (30.0 ft Drop) Impact

With no fins, high compressive stresses (approx. 10^5 psi) were calculated in the area of impact.

A more detailed model showed these peak stresses occur away from critical weld areas.

The conclusion is that with no fins the cask would sustain considerable yielding, but probably no breach to the outside would occur, since the stainless steel is very ductile.

With fins, the peak stresses were low (approx 30.0 to 40.0 ksi). The constant fin load was just enough to keep the cask from bottoming out against the rigid surface

60.0 MPH (120.0 ft Drop)

With fins, high compressive stresses were calculated (approximately 10^5 psi) in the area of impact.

The tensile stresses on the end away from impact were moderate (approx. 10^4 psi).

For the 60.0 MPH impact, the cask would sustain considerable yielding, but would probably not open up.

80.0 MPH (214.0 ft Drop) Impact

With fins, high compressive peak stresses (approx. 10^5 psi) were calculated in the area of impact.

High tensile stresses were calculated on the end away from the impact.

It is probable that the cask would not open up due to the high compressive stress on the impact end.

The high tensile load on the other end could cause the cask to open up (A detailed model is being generated to examine this).

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FIGURE 2.
SHIPPING CASK
END VIEW

FIGURE 3 CROSS-SECTION OF MODIFIED GE IF-300 PHL SHIPPING CASK FOR R-Z THERMAL ANALYSIS

FIG. 9. HEATING CODE R-0 GEOMETRY

TABLE I

SUMMARY OF BASELINE RAIL CASK CHARACTERISTICS
FOR NON-FIRE ACCIDENTS AND TYPE B FIRE

	<u>NORMAL</u> <u>STEADY-</u> <u>STATE</u>	<u>LOSS OF</u> <u>SHIELDING</u> <u>WATER</u>	<u>LOSS OF</u> <u>MECHANICAL</u> <u>COOLING</u>	<u>1475°F,</u> <u>30 MIN.</u> <u>FIRE</u> <u>TRANSIENT</u>	<u>3 HR.</u> <u>COOLDOWN</u> <u>TRANSIENT</u>	<u>POST-FIRE</u> <u>STEADY-</u> <u>STATE</u>
CALCINE } WASTE $\frac{1}{2}$	1340	1510	1380	1340	1340	1520
GLASS }	708	883	751	708	709	904
CANISTER MIDPOINT	619	803	661	619	623	824
CAVITY SURFACE	447	571	495	448	464	590
CORRUGATED SLEEVE	172	165	257	1190	203	224
SEAL	199	271	243	280	334	281
CASK TOP AT AXIS	218	256	240	660	328	261

ALL TEMPERATURES IN °F
DRY INSERT CASK DESIGN
10 YR. OLD WASTE

FIG. 5 RAIL CISTIC CANISTER END TEMPERATURE PROFILES FOR SEVERE FUMES

FIG. 4. RAIL CAR CANISTER MID-POINT TEMPERATURE PROFILES FOR SCENARIO 1

(Fig. 7) RAIL CASE SEAL TEMPERATURE PROFILES FOR FIRES

FIGURE V32. CORNER DROP LOADS

$$\epsilon = S / \cos \theta$$

$$d = \epsilon / \tan \theta$$

$$= S / \tan \theta \cos \theta$$

and

$$d = r(1 - \cos^{1/2} \theta)$$

then

$$\gamma = \cos^{-1}(1 - S / r \cos \theta)$$

and

$$A = \pi r^2 \gamma / 360$$

$$= r^2 \cos^{1/2} \theta \sin^{1/2} \theta$$

SECTION A-A

FIGURE 2
IDEALIZED CORNER IMPACT GEOMETRY

4.3.0 THERMAL ANALYSIS - TRUCK CASK MODEL GEOMETRY

Figure 4.3-1 is a cross section of the NFS-4, a simplified representation in R-Z cylindrical coordinates used as input to the HEATING5 code. Dimensions are in inches. Boundary condition numbers are also indicated. The spray calcine waste is enclosed by a 3/8 inch stainless steel canister, with a 3/8 inch air gap between the canister and the stainless steel inner shell (5/16 inch). A lead gamma ray shield, 6 5/8 inches thick, is enclosed by two stainless steel shields, the outer one being 1 1/4 inches thick. The last two regions include the 4 1/2 inch water neutron shield and a 0.165 inch tank wall.

The cask end consists of 8 inches of stainless steel covered with a 1/8 inch asbestos layer. The balsa impact limiter is not included, as explained below.

4.3.0.1 BOUNDARY CONDITIONS

For normal, steady-state operation, heat travels radially and is removed at the cask wall by radiation and natural convection to 130°F ambient air (boundary condition B4). The cask end enclosed by balsa wood, a very poor conductor, is taken to be insulated (boundary conditions B1, B2, and B3). The internal boundary condition E5 allows for radiation and conduction across the canister-inner shell air gap. In the normal state, solar heating at the rate of 1 BTU/hr-in² is uniformly distributed around the cylinder.

For fire conditions, it is assumed that all external boundaries are subjected to a flame at temperatures from 2100°F to 2700°F with a flame emissivity of 0.8. Furthermore, it is assumed that with temperatures in this extreme range, the cask wall will lose its mechanical integrity. Hence, during a fire the entire neutron shield region "goes away" and the fire environment impinges on the stainless steel outer shell (boundary condition B5).

4.3.0.2 RESULTS

Normal steady state temperatures are found as circled numbers in the cask diagram, Figure 4.3-1.

Figures 4.3-2 through 4.3-4 are temperature-time histories during the above fires (2100°F, 2400°F, and 2700°F) at three locations within the cask/canister system.

Figure 4.3-2 - canister mid-point ($Z=0$, $R=6.00$ inches)

Figure 4.3-3 - canister end ($Z=60$ inches, $R=6.00$ inches)

Figure 4.3-4 - waste midpoint ($Z=0$, $R=0$ inches)

For the three fires, the zero-time temperature was obtained from the normal steady-state run with the neutron water shield in place and with solar heating. The steady-state temperature distribution served as the same initial condition for each of the fire transient code runs.

In Figure 4.3-2 through 4.3-4 two failure criteria are specified as horizontal lines.

2200°F (1200°C), which corresponds to rapid corrosion of the canister within a time of several hours, if the waste form were a borosilicate glass

2370°F (1300°C), which corresponds to oxidation, creep, and corrosion of the stainless steel canister, the failure modes with calcine waste

Note that canister temperatures are virtually independent of waste form, i.e., calcine and glass at 3.5 kW give nearly identical canister temperatures; only waste centerline temperatures would be substantially different.

Figure 4.3-2 depicts temperatures as a function of time at the canister midpoint. For the truck cask,

canister failure occurs first at the midpoint rather than at the ends. Table 4.3-1 below summarizes the times at which canister failure first occurs.

TABLE 4.3-1

CANISTER FAILURE TIMES* FOR TRUCK CASK IN FIRES

<u>Fire Temperature</u> <u>°F</u>	<u>Waste Form</u>	
	<u>Glass</u>	<u>Calcine</u>
2100°F	No Failure	No Failure
2400°F	1.7	3.0
2700°F	1.3	1.5

*Failure times are given in hours

Figure 4.3-3 shows canister end temperatures. These temperatures lag the midpoint temperatures, although the same failures eventually do occur.

Figure 4.3-4 gives waste (calcine) centerline, midpoint temperatures versus time. For the 2400°F and 2700°F fires, the calcine bakeout temperature is exceeded, at times ranging from 2.6 to 3.2 hours.

A least squares fit to the canister failure (Figure 4.3-2) gives a failure locus in the temperature range $2100 < T < 2700^\circ\text{F}$ of the following form:
 $t = a/(T-b)$ where a and b are constants

For glass: $a = 735$

$b = 2025$

For calcine: $a = 900$

$b = 2100$

Failure of the shipping cask TFE O-ring seal would take place very quickly, within a few minutes of the start of the fire. It is expected that this result would hold also for fire temperatures down to the hydrocarbon minimum, 1400°F.

4.3.0.3 CONCLUSIONS

For the baseline design truck cask, corrosion failures of the canister can take place with glass waste, assuming fire temperatures in excess of 2100°F

For calcine waste, canister failure by the oxidation, creep, and corrosion mechanisms occurs with fire temperature above 2100°F

The corrosion failure locus can be correlated reasonably well by an expression in the form failure time $\times$ (fire temperature - 2025) = 735, with failure time in hours and fire temperature

in °F.

The baseline truck cask design is an adequate one for normal steady state operation, with no mechanical cooling needed, for the following reasons:

1. Calcine centerline temperature does not exceed the calcine bake-out temperature.
2. Canister maximum temperature is well below that required for rapid corrosion.
3. The lead shielding remains in the solid state.
4. Stainless steel outer tank wall temperatures are moderate, approximately 270°F.

Calcine bake-out temperatures are exceeded in the SHLW by fires with temperatures above 2400°F. The oxidation, creep and corrosion failure locus can be expressed in the form failure time $\times$ (fire temperature - 2100) = 900.

References

Nuclear Fuel Services, Inc., "Safety Analysis Report for Nuclear Fuel Services, Inc., Spent Fuel Shipping Cask Model NFS-4," September 29, 1972.

FIG. 2 CROSS-SECTION OF MODIFIED NSF-4 SHIPPING CASK (TRUCK) USED FOR THERMAL ANALYSIS

Fig. 3

CANISTER MAPPING TEMPERATURE
PROFILES - TRUCK CASE IN FIRE ASSE.

Fig. 6 CANISTER AND TEMPERATURE PROFILES
TRUCK CASK IN FIRE ACCIDENTS

Fig. 5 WASTE CENTERLINE TEMPERATURE PROFILES
TRUCK CRISK IN FIRE ACCIDENTS

4.3.1 IMPACT ANALYSIS - TRUCK CASK

Although considerable effort was directed to flat-bottom end-drop analysis of the rail cask, mainly for background purposes, it was decided to restrict the truck analysis to the corner drop condition, since this was seen to be more severe.

5.1 HANDLING EVENTS

This appendix describes calculations establishing the effect of accidental releases of high-level waste during handling at the federal repository. It is shown that the expected release from this portion of the waste-disposal operation is relatively unimportant compared to that from interim storage and transportation.

5.1.1 ASSUMPTIONS AND SOURCES OF DATA

The following assumptions were used in analyzing this portion of the waste management sequence:

The head of the repository shaft is contained within a sealed building. Under normal conditions, any contamination escaping into the air in this building is prevented by a filtration system from entering the atmosphere. The filtration-system parameters (probability of failure, efficiency) are identical to those used in analyzing the interim-storage portion of the waste management operation.

Transportation vehicles enter the sealed storage area for unloading. The canisters are not removed from their transportation casks until they are inside the building and the air seal is reestablished.

The only time the canisters are "bare" is during transfer between casks and while they are being lowered into the repository.

At this point in the disposal sequence, bare canisters can dissipate all heat generated into air. Therefore, the only danger to canister integrity during handling is from impact when accidentally dropped.

The nominal time the canister is outside the casks is 10 minutes per canister, which includes movement between transportation cask and transfer cask and between transfer cask and final location.

The probability of a drop due to crane failure is the same as in the analysis of handling at interim storage: 3×10^{-6} per hour of operation.

The principal danger to the canisters while waiting to be lowered into the repository is impact from an aircraft. The casks will protect against all other dangers, such as earthquake, tornado, etc. Meteors are an extraordinary occurrence of insignificant probability. The same probabilities

of aircraft-crash density used before apply.

The transportation cask remains on the surface at the repository for a nominal period of one week before it is unloaded. This is the period of exposure to an aircraft impact.

The water path for released wastes was not analyzed since precautions can be taken to avoid a direct path into surface-water bodies serving large populations.

5.1.2 EVENT TREE

Figure 5.1-1 shows the event tree analyzed to determine the consequences of accidental release during handling at the repository.

Two accident modes are present: crane drop due to failure and impact by an aircraft.

5.1.3 CALCULATION OF RELEASE PER MEGAWATT ELECTRIC * YEAR

The expected release per megawatt electric * year for each path in the event tree shown in Figure 5.1-1 can be calculated when it is conservatively assumed that the release fractions are unity and that the canister will always breach when dropped or impacted; the expected release is insignificant compared to other parts of the waste-management process. (In fact, as discussed in previous appendices, the canister is expected to be able to withstand drops through heights anticipated to be possible.

5.1.4 CRANE DROP

Using the crane-drop probability ($3 \times 10^{-6}/\text{hr}$) and the bare-canister handling time (10 min), the probability of a canister drop per megawatt electric * year is

$$(3 \times 10^{-6} \frac{\text{accidents}}{\text{hr}}) (\frac{1 \text{ hr}}{6 \text{ can}}) (\frac{1 \text{ can}}{100 \text{ MWe}\cdot\text{yr}}) = \frac{5 \times 10^{-9} \text{ accidents}}{\text{MWe}\cdot\text{yr}}$$

This is the same accident probability as that computed for crane stall in Section 5.2. The effect of filter failure or correct operation also is the same, so, the results from that section apply. The expected release due to a drop caused by crane failure accident is 10^{-18} liter per megawatt electric*year times the activity contents (100 MWe*yr) of one canister, or 10^{-16} megawatt electric*year waste per megawatt electric*year.

5.1.4 IMPACT BY AN AIRCRAFT

The expected release from an aircraft accident is computed using the accident density (1.27×10^{-10} crashes/yr*mi²) given in Section 5.3 and a train-transportation cask cross section to aircraft crashes of 100 square yards per cask = 3.22×10^{-5} square miles per cask:

$$(1.27 \times 10^{-10} \frac{\text{crashes}}{\text{yr}\cdot\text{mi}^2}) (3.22 \times 10^{-5} \frac{\text{mi}^2}{\text{cask}}) = 4.09 \times 10^{-5} \frac{\text{crashes}}{\text{cask}\cdot\text{yr}}$$

Since the cask is on site for one week, the expected release for aircraft crash on a train cask is (for release fractions of unity)

$$\frac{4.09 \times 10^{-15}}{52} = 7.09 \times 10^{-17} \frac{\text{mole-yr waste}}{\text{mole-yr}}$$

Fig. 5-6. Event tree for handling at the geologic isolation site.

5.1-1

7.1.0³⁻¹ ~~3-1~~ Migration through Surface Water

Figure ~~3-1~~⁷⁻¹ diagrams the model for the surface-water system. In this system nuclides may be stored in the water, sediment, topsoil and in the human population. The mass balance for the water is shown in Fig. ~~3-2~~⁷⁻². Figure ~~3-3~~⁷⁻³ and ~~3-4~~⁷⁻⁴ show the mass balance for the topsoil and the people. Figure 3.1-9 shows the mass balance for the sediment, subsoil, etc.

These mass balances correspond to first-order ordinary differential equations for nuclide concentrations in each component of the system. Appendix B lists these equations.

Figure ~~3-1~~⁷⁻¹ Model for Surface-Water System

Figure ~~6-2~~ Mass Balance for Water

7-2

Figure ~~6-3~~ Mass Balance for Topsoil

7-3

P-24957

Figure ~~7-3~~ ⁷⁻⁴ Mass Balance for People

Figure ~~7-4~~ ⁷⁻⁵ Mass Balance for Sediment, Subsoil, etc.

The notation used in Figs. ~~2-4~~⁷⁻² through ~~2-5~~⁷⁻⁵ is as follows:

A_s = the area of the sediment and subsoil (m^2)

d = the average diffusion depth of the sediment and subsoil (m)

F_I = the irrigation flow rate (m^3/yr) from the water system

F_O = the flow rate (m^3/yr) of water out of the system

N_p = the number of people using the water system

I_{wj} = the effective water intake rate (m^3/yr) for each individual in the system

F_R = the rainfall flow rate (m^3/yr)

C_{Tj} = the concentration in the topsoil (Ci/m^3)

C_{Sj} = the concentration in subsoil and sediment (Ci/m^3)

C_{pj} = the concentration in the people ($Ci/person$)

I_{Tj} = the effective topsoil intake rate (m^3/yr)

λ_{pj} = the biological wash-out coefficient (yr^{-1})

$$= \frac{0.693}{\text{biological half life}} + \frac{1}{\text{average life span}}$$

7.1.1 ~~2-4~~ Dose Calculations for Water Release

The dose calculations used in Ref. 20 were modified to be compatible with explicit modeling of radionuclide buildup in the topsoil. The equations used are presented below.

Nuclides released in a water system and deposited in the topsoil find their way to the human population through the food chain and through external radiation. The yearly radiation dose for each organ is the sum of the dose from the internally accumulated store of the nuclide and the dose from the dissolved nuclide which passes through the body with ingested water:

$$\text{dose}(i) = \sum_j \text{DI}(i,j) \text{IA}(i,j) + \sum_j \text{DE}(i,j) \text{Cw}_j$$

(7-1)
~~Equation~~

where

$\text{dose}(i)$ = the total yearly dose to organ i (Rem/yr)

$\text{DI}(i,j)$ = the dose conversion factor ((Rem/hr)/Ci) for ingestion of nuclide j .

$\text{IA}(j)$ = The internal accumulation of nuclide j (Ci).

$\text{DE}(j,j)$ = the dose conversion factor ((Rem/hr)/(Ci/m³)) for external exposure.

Cw_j = concentration in water (Ci/m³)

ET = exposure time (hr/yr).

The internal accumulation is computed by solving the differential equation giving the balance between intake and loss. Losses are computed using a coefficient which lumps together radioactive decay and departure of the nuclide from the organism. This coefficient is called the biological half-life.

$$\frac{d}{dt} IA(i,j) = - \frac{0.693}{BHL(i,j)} IA(i,j) + RI(j)$$

(7.1-2)
(2.1-5)

where

BHL(i,j) = biological halflife (yr) of nuclide
j in organ i

RI(j) = intake rate (Ci/yr) of nuclide j.

The intake rate is the sum of the intakes coming through the food chain from the water and the topsoil:

$$RI(j) = 1000 IW(j) Cw(j) + IT(j) C_T(j)$$

7.1-3
(2.1-6)

where

IW(.) = effective intake rate from water
(l/yr)

IT(.) = effective intake rate from topsoil
(m²/yr)

C_T(j) = concentration in topsoil (Ci/m³)

The effective intake rates from water and topsoil are expressed in the model as the sums of the intakes from various types of food. The routes modeled are fish, other seafood, various vegetables and animal products, and drinking water. For each type of food, there is a concentration factor relating the nuclide concentration in the food to the nuclide concentration in the water. In the case of vegetables and animal products, these concentration factors may be expressed in terms of irrigation and feeding rates. Transfer coefficients will then relate nuclide concentrations in topsoil and drinking and irrigation water to concentrations in food. The equations describing these relationships are listed in ~~Appendix~~.

~~APPENDIX~~ 7.1.1 CONCENTRATION EQUATIONS FOR WATER, SOIL, AND PEOPLE

The concentration of nuclide j in the water, given the mass balance shown in Fig. ~~7-1~~⁷⁻³, is given by the first-order ordinary differential equation

$$\begin{aligned} \frac{d}{dt} C_{Wj} = & - \left[\lambda_j + \frac{2D A(n)}{Lz V_w} + \frac{D A_s}{d V_w} + \frac{F_I + F_O}{V_w} + N_P \frac{I_{Wj}}{V_w} \right] C_{Wj} \\ & + \left[\frac{v}{K_j} \frac{A(n)}{V_w} + \frac{2D A(n)}{K_j Lz V_w} \right] C_{j(n)} + \left[\frac{F_I + F_R}{K_j V_w} \right] C_{Tj} \\ & + \left[\frac{D A_s}{K_j V_w d} \right] C_{Sj} + \left[\frac{N_P \lambda_{Bj}}{V_w} \right] C_{Pj} \end{aligned} \quad \begin{matrix} (7.1-4) \\ (22) \end{matrix}$$

where

V_w = the volume of the water system (m^3)

and the meaning of other notation is as given in ~~Section 7.1.0~~⁷⁻³ ~~7-5~~. Figures ~~7-2~~⁷⁻³ and ~~7-3~~⁷⁻⁴ show the mass balance for the topsoil and the people. Figure ~~7-4~~⁷⁻⁵ shows the mass balance for the sediment, subsoil, etc. The concentration equations are:

For topsoil

$$\frac{d}{dt} C_{Tj} = - \left[\lambda_j + \frac{F_R + F_I}{K_j V_T} + K_P \frac{I_{Tj}}{V_T} \right] C_{Tj} + \frac{F_I}{V_T} C_{Wj} \quad (7.1-5)$$

where

V_T = the volume of topsoil (m^3)

For people

$$\frac{d}{dt} C_{Pj} = - \left[\lambda_j + B_j \right] C_{Pj} + I_{Tj} C_{Tj} + I_{Wj} C_{Wj} \quad (7.1-6)$$

For sediment, subsoil, etc.

$$\frac{d}{dt} C_{Sj} = - \left[\lambda_j + \frac{D A_S}{K_j V_{Sd}} \right] C_{Sj} + \left[D \frac{A_S}{V_{Sd}} \right] C_{Wj} \quad (7.1-7)$$

where

V_S = the volume of sediment and subsoil (m^3)

7.1.2

EQUATIONS FOR NUCLIDE INTAKES

The effective rate at which a person takes in nuclide j from water is

$$IW(j) = DWI(j) + AFI(j) + VI(j) + DAPI(j)$$

(7.1-8)

where

$DWI(j)$ = drinking water intake (l/yr)

$AFI(j)$ = aquatic food intake (l/yr)

$VI(j)$ = vegetable intake (l/yr)

$DAPI(j)$ = domestic animal product intake (l/yr)

The aquatic food intake is

$$AFI(j) = BF(j) FC + BC(j) CC + BM(j) MC$$

(7.1-9)

where

$BF(j)$, $BC(j)$, $BM(j)$ = Biological conversion factors $\left(\frac{C_i/K_d}{C_i/C_0}\right)$ for fish, crustacea, and mollusc

FC , CC , MC = fish, crustacea, and mollusc consumption (Kg/yr)

The vegetable intake is

$$VI(j) = \sum_{k=1}^n CV(j,k) VC(k)$$

(7.1-10)

where

$CV(j,k)$ = the concentration factor ($C_i/Kg/C_i/l$)
for nuclide j in vegetable k

VC_k = yearly consumption (Kg/yr) of
vegetable k

$$CV(j,k) = IR(k) \frac{T(k)}{V(k)}$$

(7.1-11)
~~(7.1-11)~~

where

$IR(k)$ = the irrigation rate (l/m^2 -day)
during the growing season for
vegetable k

$T(k)$ accounts for translocation from
plant surface to edible portion
(day)

$V(k)$ = yield (Kg/m^2)

The domestic animal product intake is

$$DAPI(j) = \sum_{L=1}^5 CA(j,L) AC(L)$$

(7.1-12)
~~(7.1-12)~~

where

$CA(j,L)$ = the concentration factor ($C_i/unit/C_i/l$)
for nuclide j in animal product L

$AC(L)$ = yearly consumption of animal product L
(unit/yr)

The concentration factors are

$$CA(j,L) = S(j,L) [CF(j,L) Q(L) + QW(L)]$$

(7.1-13)
~~(7.1-13)~~

where

$S(j,L)$ = transfer coefficient (Ci/unit/Ci/day)

$CF(j,L)$ = concentration factor for feed
(Ci/Kg/Ci/L)

$QW(L)$ = water consumption (L/day)

$Q(L)$ = feed consumption (Kg/day)

The intake from the topsoil is

$$IT(j) = B(j) \left[\sum_{k=1}^9 VC(k) + \sum_{L=1}^5 S(j,L)Q(L)AC(L) \right] / D_T \quad (7.1-14)$$

where

D_T = the density of the topsoil (Kg/m³)

and

$B(j)$ = plant concentration factor (Ci/Kg/Ci/Kg)

~~7.1.3~~ 7.1.3

CONCENTRATION EQUATIONS

The concentration in the water changes as it flows from the river to the ocean. The processes reducing concentration include dilution by mixing, sedimentation, radioactive decay, and removal of water and food products by man. Figure 1 illustrates the water model and lists the exposure pathways to man.

7-6
Figure 3 Water and Food Pathways Model

7.13.1 TIME DEPENDENT EQUATIONS FOR THE ESTUARY AND OCEAN PATHWAY

The full time-dependent concentration equation for nuclide j in the estuary is given by

$$\begin{aligned} \frac{dCE_j}{dt} = & - \left[\lambda_j + \frac{1}{T} + \frac{D_{EASE}}{dV_E} + \frac{N_p IWE_j}{V_E} \right] CE_j \\ & + \left[\frac{F_O}{V_E} \right] CW_j + \left[\frac{N_p \lambda_{Bj}}{V_E} \right] CPE_j + \left[\frac{D_{EASE}}{dV_E} \right] CSE_j \end{aligned} \quad \begin{array}{l} (7.1-15) \\ (\text{---}) \end{array}$$

for people

$$\frac{d}{dt} CPE_j = - \left[\lambda_j + \lambda_{Bj} \right] CPE_j + IWE_j CE_j \quad \begin{array}{l} (7.1-16) \\ (\text{---}) \end{array}$$

for sediment, subsoil, etc.

$$\frac{d}{dt} CSE_j = - \left[\lambda_j + \frac{D_{EASE}}{K_j V_{SE} d} \right] CSE_j + \left[\frac{D_{EASE}}{V_{SE} d} \right] CE_j \quad \begin{array}{l} (7.1-17) \\ (\text{---}) \end{array}$$

These equations follow closely the notation and the analysis *assumes* ~~earlier~~ ~~equations 6.11 and 6.12~~. There are three differences.

- T is the flushing time for the estuary where F/T = the total quantity of fresh water in the estuary at high tide.

- The water is assumed too salty for drinking or irrigation. Here IWE_j represents the intake of aquatic food only.
- Diffusion to the sediment and subsoil must be altered to take account of physical, chemical, and biological properties specific to the estuarine region.

The time-dependent concentration equation for nuclide j in the ocean is given by

$$\frac{dCO_j}{dt} = - \left[\lambda_j + \frac{1}{T_o} \right] CO_j + \left[\frac{V_E}{TV_o} \right] CE_j$$

(7.1-18)
(7.1-18)

where

V_o = effective ocean volume.

V_E = estuary volume at high tide.

T_o = flushing time, or residence time for coastal waters.

Plume concentration equations for the ocean have been given in Ref. 2 and Ref. 3. Using these or utilizing site-specific data a mean concentration for the near-shore plume may be calculated, CPL_j . The need for this calculation depends

heavily on the variation of fish concentration in the coastal ocean. Typically fish concentration falls off rapidly as the distance from shore increases, so that plume dynamics become important.

Typical values for the parameters in the concentration equations appear in ~~Appendix~~ 7.1.6

7.1.3.2

STEADY-STATE EQUATIONS FOR THE RIVER, ESTUARY, AND OCEAN PATHWAYS

For the river,

$$CW_j [T_o + F_I] = F_N C_j(N) + \left[\frac{F_I + F_R}{K_j} \right] CT_j \quad (7.1-19)$$

where $F_N C_j(N)$ represents the inflow from the aquifer. It is assumed that λ_j is small and that the sediment comes quickly to equilibrium. Similar simplification can not be made for the top-soil since it is not assumed to reach equilibrium. The top-soil concentration is calculated from

$$CT_j = CW_j \left[\frac{F_I}{V_T} \right] \left[\frac{1 - e^{-\lambda_T t_b}}{\lambda_T} \right]$$

where for now $t_b = 50$ years, and

$$\lambda_T = \left[\lambda_j + \left(\frac{F_I + F_R}{E_j V_T} \right) \right]$$

For sorption constants equal to 1 or 10^{-2} equilibrium may be safely assumed. However, for $K_j = 10^6$ a more explicit modeling of the soil retention is required to avoid results that are much too conservative.

For the estuary,

$$CE_j \left[\frac{V_E}{F} \right] = F_o CW_j \quad (7.1-20)$$

and for the ocean,

$$CO_j \left[\frac{V_o}{T_o} + \lambda_j V_o \right] = \left[\frac{V_F}{T} \right] CE_j \quad (7.1-21)$$

(7.1-21)

Since the residence time for the ocean is much longer than for the estuary, radioactive decay is included in the ocean equation.

7.1.4
4.

AVERAGE MAN AND MAXIMUM MAN DOSE CALCULATIONS

Doses are calculated by multiplying pathway concentration by effective food intake and by the 50-year dose conversion factor. A discussion of the 50-year dose conversion factor is given in ~~appendix 7.1.7~~. The effective intake equations, ~~Equation 7.1.8~~, are modified as follows.

The average or maximum man's diet of aquatic food is broken down by pathway.

$F(i,j)$ = the fraction of food product i (fish, crustacean or mollusk) consumed in pathway j (river, estuary, plume, or ocean).

$$\sum_j F(i,j) = 1 \quad (7.1-22)$$

One way to generate $F(i,j)$ is to assume that the amount of aquatic food i caught in pathway j divided by the total catch of i for all pathways equals $F(i,j)$. More conservative assumptions are possible and will be investigated.

For the estuary and ocean, salt-water bioaccumulation factors are used.

Parameters necessary for the above calculations as well as average and maximum man usage factors are tabulated in ~~Appendix 7.1.6~~. 7.1.6

7.1.5
as

POPULATION DOSE CALCULATION

Population dose is normally calculated via an assumed population within a 50 mile radius of the contaminated region. For calculations far into the future this is somewhat more difficult. An alternative procedure is used which eliminates the need for population statistics.

Total vegetable intake is limited by the amount of contaminated irrigation water used. This amount is in turn limited by the river or lake's flow rate. An effective population is calculated with the following assumptions.

- All members of the population are adults
- The population eats an average diet grown entirely on contaminated soil.

Hence,

$$P = FM / \sum_{K=1}^9 (VC(K)/A(K))$$

(7.1-23)
(~~22~~)

and

$$A(K) = Y(K)/IR(K) G(K)$$

(7.1-24)
(~~22~~)

where

- P = the effective population
- VC(K) = average yearly intake of vegetable K (kg/yr)
- A(K) = number of kg of vegetable K produced per liter of irrigated water per year
- IR(K) = irrigation rate (l/m^2 day)
- Y(K) = Yield (kg/m^2)
- G(K) = growing period (day/yr)
- FM = irrigation flow rate for food eaten directly by man. (l/yr)

Intake of radionuclides from animal products is due to animal ingestion of contaminated water and/or feed. The contribution to man's dose from animal products other than milk is generally negligible when compared to man's dose from drinking water or milk. For the milk pathway about 94% of the contamination is due to cows eating feed rather than cows drinking water. Thus it is fairly safe to base the population dose from animal product intake on an irrigation rate, FA, for animal feed. Since FA plus FM must equal the total irrigation rate, a limit is set on the population dose.

For fish consumption, the data in ^{7.1.8} ~~Appendix C~~ on fish yields from estuaries, rivers, lakes and the ocean are used. Since the productivity of aquatic food products is limited, substantial changes in these yields are not expected. Conservative estimates where warranted have been taken.

For the drinking water, shoreline, and swimming pathways the data from Ref. 4 has been used:

- Average river drinking water usage was 618×10^3 people.
- Average lake drinking water usage was 2×10^6 people.
- Swimming and boating usage was 0.14 hours/acre-day.

7.1.6

PHYSICAL AND DYNAMIC DATA FOR THE
RIVER, ESTUARY, AND OCEAN

7.1.6.1 RIVER FLOW, RESIDENCE TIMES AND VOLUMES

$$T = 2.5 \text{ days}$$

$$V_E = 2.6 \times 10^9 \text{ M}^3$$

$$F_O = 3.8 \times 10^3 \frac{\text{M}^3}{\text{sec}}$$

$$T_O = 122 \text{ days}$$

$$V_O = 6. \times 10^{12} \text{ M}^3$$

7.1.6.2 PLUME DATA

Ocean Velocity	U	15 $\frac{\text{CM}}{\text{sec}}$
----------------	---	-----------------------------------

Plume Depth	d	20 M
-------------	---	------

Plume Area	A_{pl}	$1.25 \times 10^{10} \text{ M}^2$
------------	----------	-----------------------------------

Mean Plume Concentration	CPL_j	$\left[\frac{V_E}{16NTxF_O} \right] CE_j$
--------------------------	---------	--

For site specific calculations, Ref. 2

$$CPL_j(X,Y) = \left[\frac{CE_j VE}{T} \right] \frac{\exp\left(\frac{-UY^2}{4DX}\right) \exp\left(\frac{-YX}{L}\right)}{2d \sqrt{1-UXD}} \quad (7.1-25)$$

D = lateral dispersion coefficient

X = distance parallel to current

Y = distance normal to current

7.1.7

DOSE FACTORS FOR CHRONIC INGESTION

The appropriate equation for the calculation of 50-year dose factors for chronic ingestion of radionuclides is,

$$D50 = 18.7 \left\{ \frac{f_w e^{-50\lambda}}{m} \frac{50\lambda + e^{-50\lambda} - 1}{\lambda^2} \right\}$$

(7.1-26)

where

D50 = 50-year dose factor $\frac{\text{mrem}}{\text{pCi/yr}}$

f_w = the fraction of the ingested nuclide reaching the organ of interest

m = mass of the organ in grams

e = effective energy of the nuclide in the organ under consideration (MeV/dis) (rem/rad)

λ = $(0.693/\text{BHL} + 0.0001/\text{AHL})$ 365 days/year

BHL = biological half-life for the nuclide in the organ (days)

AHL = atomic half-life for the nuclide (days)

Our procedure has been to calculate dose factors using those already tabulated and multiply by a dose accumulation factor, Ref. 5.

The dose factors tabulated represent the year's accumulated dose from a single intake of radionuclides on

the first day of the year. The equation used is:

$$D = 18.7 \left\{ \frac{f_{\text{max}}}{E} \frac{1-e^{-k}}{k} \right\} \frac{\text{area}}{\text{PCI}}$$

(7.1-27)

The dose accumulation factor is,

$$\text{DAF} = \frac{50}{1-e^{-k}} + \frac{1-e^{-50k}}{(1-e^{-k})(1-e^{-k})}$$

(7.1-28)

For the range of k considered the two methods yield virtually identical answers. By substituting $(-)$ for $(1-e^{-k})$ in equation (7.1-28), the two calculations are identical.

~~ANALYTIC SCIENCES CORPORATION~~

7.1.8

STANDARD USAGE FACTORS FOR
AVERAGE AND MAXIMUM MAN

Except where noted average and maximum diets were based on Ref. 6. These are

7.1.2.1 ~~42~~AVERAGE DIET

Milk	111 li/yr	
Fish	6.9 kg/yr	
Seafood	1.0 kg/yr	0.5 kg/yr mollusk
Water	370 li/yr	0.5 kg/yr crustacean
eggs	487 eggs/yr	
Berries	5 kg/yr	
Melons	7 kg/yr	
Orchard	43 kg/yr	
Grain (wheat)	50 kg/yr	
other	4 kg/yr	
Beef	32 kg/yr	
Poultry	17 kg/yr	
Pork	39 kg/yr	
Leafy	5 kg/yr	
O.A.G.	5 kg/yr	
Roots	28 kg/yr	
Potatoes	44 kg/yr	
Shoreline	8.3 hr	
Swimming and boating	5 hr, Ref. 5	

7. 1.8.2
~~7. 1.8.2~~MAXIMUM DIET

Milk	310	l/yr	
Fish	21	kg/yr	
Seafood	5	kg/yr	{ 2.5 kg/yr mollusk
Water	730	l/yr	{ 2.5 kg/yr crustacean
Eggs	730	eggs/yr	

Berries	10	kg/yr
Melons	14	kg/Yr
Orchard	90	kg/yr

Grain (wheat)	113	kg/yr
Other	12	kg/yr

Beef	45	kg/yr
Pork	45	kg/yr
Poultry	20	kg/yr

Leafy	32	kg/yr
O.A.G.	32	kg/yr

Roots	88	kg/yr
Potatoes	131	kg/yr

Shoreline	12	hr/yr
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Swimming and boating	200	hr, Ref. 8
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7.1.8.3

AQUATIC FOOD HARVEST

	River	Estuary and Near-Shore	Ocean	Lake
Fish	1.5×10^5	1.2×10^7	$2. \times 10^8$	7.5×10^6
Mollusk	_____	4.5×10^6	_____	_____
Crustacean	1.0×10^4	1.7×10^6	_____	_____

Weights all expressed in edible lb.

Fish productivity in estuary = $4.6 \times 10^{-2} \frac{\text{lb}}{\text{m}^2}$ Fish productivity in the ocean = $1.3 \times 10^{-3} \frac{\text{lb}}{\text{m}^2}$

7. / REFERENCES

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7.2.0

~~7.2~~ METEOROLOGY

In the event of an accidental release of solid radioactive waste at a fuel reprocessing plant (FRP), along a transportation route, or at a FWR, a portion of the waste might become airborne and thereby affect populations downwind. Therefore, it is necessary to estimate the dispersion and deposition of such debris. For our calculations, we assume a unit-curie release at one meter above the ground. The initial source is assumed to have a one meter radius and half-thickness. At the FRP or the FWR, it is possible to have either a continuous (plume) release, an instantaneous (puff) release, or perhaps something in between these two extremes. A transportation accident would most likely result in an instantaneous release.

The Gaussian diffusion equations for these two types of releases at the surface are given below. If the plume is sector-averaged over 22.5°,

$$X/Q = \left(\frac{2}{\pi}\right)^{3/2} \frac{4}{\sigma_z \bar{u}} \exp\left(-\frac{1}{2} \frac{h^2}{\sigma_z^2}\right), \quad (7.2-1)$$

~~(4-5)~~

where X = surface air concentration on plume centerline (Ci/m^3), Q = source term (Ci/sec), $\bar{u}$ = mean surface wind speed (m/sec), σ_z = vertical standard deviation of wind (m), x = distance downwind (m), and h = height of plume center above ground (m).

For a puff,

$$X/Q = \frac{1}{2 \pi^{3/2} \sigma_{x_i} \sigma_{y_i} \sigma_{z_i}} \exp\left(-\frac{1}{2} \frac{h^2}{\sigma_{z_i}^2}\right), \quad (7.2-2)$$

~~(4-5)~~

where Q is in curies and the subscripts signify instantaneous values of the standard deviation in the x (downwind), y (crosswind), and z (vertical) directions. Since a plume meanders about an average trajectory at an angular variation dependent on the atmospheric stability while a puff is influenced only by the winds along its trajectory, the σ values for a plume are larger than those for a puff in similar conditions of stability and windspeed.

In addition to surface concentration, surface deposition has been incorporated into the calculations. This modification of the diffusion equations involves the use of a deposition velocity that represents the rate of particulate impaction with the surface due to turbulent eddies near the ground. A deposition velocity of 0.01 meters per second has been assumed, and the above equations have been modified to include plume depletion as deposition occurs.

For the purpose of an accident analysis, we have considered both an average and a "worst" case. The average was made by assuming a Pasquill-Gifford "D" stability with a five-meter-per-second wind speed. To determine the worst case, we have estimated the upper envelope of concentration for both plumes and puffs in the following stability categories and windspeeds:

Stability category	Windspeed, m/sec
A	1
B	3
C	4
D	5
E	2
F	1

From 0.1 to 100 kilometers from the source, average- and worst-case analytical equations for average, unit-source continuous and instantaneous surface releases (plume and puff, respectively) are given in Table 7.2-1. Graphs of

7.2-1
 table 7.2-1. Analytical expressions for estimating relative centerline concentrations^a from a solid nuclear waste accident; χ/Q is relative concentration (yr/m^3), x is distance downwind (m), plume or puff height is 1 m, and deposition velocity is 3×10^5 m/yr.

Type of estimate	Plume	Puff
Average-case χ/Q		
(D stability, $\bar{u} = 5$ m/sec)	$5.6 \times 10^{-8} x^{-1.7}$	$3.8 \times 10^{-4} x^{-3.7}$
Worst-case χ/Q		
(upper envelope)	$1.3 \times 10^{-6} x^{-2}$	$4.1 \times 10^{-3} x^{-3.9}$

^aEquivalent expressions for the relative surface deposition rate (in units of m^{-2}) may be obtained by multiplying the coefficients of the expressions in the table by 3×10^5 m/yr, the deposition velocity.

(7.2-1) (7.2-2)
 these functions are shown in Figs. 7.2-1 and 7.2-2. Similar equations for the deposition rate (in m^{-2}) can be obtained by multiplying Eqs. (D-1) and (D-2) by 0.01, the deposition velocity in meters per second.

Concentrations obtained from the worst-case expression for a plume have been compared with NRC guidance for nuclear power stations (*Preparation of Environmental Reports*, 1976, p. 7-1). At a distance of 100 meters from an accident, the expression for a plume (Table 7.2-1) yields a χ/Q about a factor of 5 greater than the value obtained from *Preparation of Environmental Reports*. The concentrations from both methods are the same at six kilometers. At 100 kilometers, the Table 7.2-1 expression gives a concentration about a factor of 3 less than that indicated by *Preparation of Environmental Reports*.

7.2-1
 Fig. 7.2-1. Graph of analytical expressions for a plane. Release height is one meter; sector averaging was used.

7.2-2
 Fig. 7.2-2. Graph of analytical expressions for a puff. Release height is one meter.

7.2.1

~~7.2~~ AREA INTEGRALS OF THE PRODUCT OF CONCENTRATION, AREA, AND POPULATION DENSITY

If we obtain the integral of $\chi/Q \cdot A \cdot \rho$ (where A is the area under consideration), we need then only multiply by the appropriate dose conversion factors to obtain the total population dose (in man-rem) out to 100 kilometers over the swath of the plume or puff. Since the three quantities χ/Q , A, and ρ are interrelated with respect to distance or, equivalently, time, one must integrate their product over either distance or time to obtain population dose.

Let this integral be denoted as $\sum$.

For a plume, Eq. ~~(7.2-1)~~^{7.2-1} incorporates sector-averaging over a 22.5° sector.

The integral is,

$$\sum = \int_{10^2}^{10^5} \frac{\chi}{Q} \rho \, dA,$$

(7.2-3)
~~(7.2-3)~~

where $dA = \frac{\pi}{8} \cdot dx$, x in meters. Substitution of the appropriate χ/Q expression for Table 4-2-1 and the population-density equation from the following subsection yields an integral equation with distance downwind as the sole variable. Numerical integration out to 10^5 meters results in the plume values shown in Table 4-2-2.

For the puff, the approach is similar, except that the integral is different (one cannot sector-average a puff):

$$\Sigma = \int \left(\frac{\chi}{Q} A \right) \rho \, dt.$$

7.2-4
(2-1)

Since $dt = \frac{1}{u} dx$ (u represents wind speed), the equation becomes:

$$\Sigma = \frac{1}{u} \int_{10^2}^{10^5} \left(\frac{\chi}{Q} A \right) \rho \, dx.$$

(7.2-5)
(2-2)

We have considered an expanding puff whose radius at a given x is equal to two standard deviations ($\sigma_x = \sigma_y$) and derived the average concentration χ/Q over

7.2-2
Table 4-2-2. Integrated values of $\left(\frac{\chi}{Q} \cdot \rho \cdot A \right)$ for plumes and puffs in various accident scenarios. Units are $\text{mg} \cdot \text{yr} / \text{m}^3$.

Accident scenario	Plume		Puff	
	Average value	Worst value	Average value	Worst value
FRP	4.9×10^{-11}	1.1×10^{-10}	9.6×10^{-11}	(Same as average value in all scenarios)
Transportation				
Rural	4.5×10^{-11}	7.6×10^{-11}	3.7×10^{-11}	
City	1.6×10^{-9}	3.4×10^{-9}	2.9×10^{-9}	
FWR	3.2×10^{-13}	5.7×10^{-13}	6.7×10^{-13}	

4-2-2-1

A at each distance x. Since area and concentration both use the same Gs at distance x, it is convenient to evaluate their product for all xs. Hence, for the average value (D stability, u = 5 m/sec),

$$\frac{\bar{X}}{Q} A = 4.3 \times 10^{-6.65},$$

7.2-6
(1000)

where $\bar{X}/Q$ is the average concentration of radioactivity (in units of sec/m^3) at a distance x, and over an area A of $\pi(2\sigma)^2$. To within a factor of 2, this expression for a puff also gives the concentration for the upper envelope of all stability categories. Numerical integration of the equation that results from substituting into Eq. (7.2-5) results in the puff values given in Table 7.2-2. The products of the values in this table and the appropriate dose conversion factors ~~in Table 7.2-3~~ yield estimates of population doses in man-rem.

~~REFERENCES~~ 7.2 References

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7.3.0 ~~ENVIRONMENTAL~~ DEMOGRAPHY

In an estimate of the total population living within 10 miles of an accident involving solid waste, population is based on 1980 census data.

population density in the rural area is assumed to be constant and equal to the rural population density, p_r .

It is reasonable to assume that the rural population density, p_r , will be nearly uniform. In either urban or rural areas, the population density can be approximated by the function given by the first given by a $\exp(-ax)$.

General expression for the population density are:

1. From the accident site to the nearest city ($0 \leq x \leq x_0$),

$$p = (p_c - p_r) \exp[-b_1(x - x_0)^2] + p_r;$$

(7.3-1)
(2-3)

2. From the city to adjoining rural areas ($x \geq x_0$),

$$p = (p_c - p_r) \exp[-b_2(x - x_0)^2] + p_r,$$

(7.3-2)
(2-3)

where b_1 and b_2 are coefficients determining the shape of the function curve p , and where x is distance measured from the accident site, x_0 is distance from the accident to the city (mi), p_c = population density at the accident site (person/mi²), p_r = population density in the city (person/mi²), and p_r = rural population density (person/mi²).

7.3.1

7.3.1 Fuel Reprocessing Plant Demography

It is reasonable to assume that future FRPs will be located in relatively sparsely populated regions. Such plants will likely have small towns (population less than 10,000) within easy commuting distance.

Since it represents the state of the art in reprocessing plants, the Barnwell Nuclear Fuel Plant (BNFP) (see Environmental Statement...Barnwell, 1974), located on the eastern edge of the Savannah River Plant (SRP) in South Carolina, was selected as a typical site for defining the constants, b_1 and b_2 . ~~For (7.3-1) and (7.3-2)~~. The BNFP has a special advantage of having the sparsely populated SRP to the west. Therefore, only the population to the east was considered. The BNFP environmental statement was used in determining the regional demography. The constants substituted into Eqs. (7.3-1) and (7.3-2) (p

[illegible]

The χ^2 difference between the data and the prediction

1. When is the date of the test

$$\rho = 0.5 \times 10^{-5} \text{ g/cm}^3 = 0.5 \times 10^{-5} \times (10^3 \text{ kg/m}^3) = 5 \times 10^{-6} \text{ kg/m}^3$$

2. Beyond 12 VDC voltage,

$$\rho = 7.6 \cdot 10^{-5} \text{ g./cm.}^3; [-0.00000018 - 121^2] = 2.2 \cdot 10^{-5}$$

11/15/73

Journal of Management Education 36(7)>

7.3.2

Figure 1 The effect of the concentration of the solution on the adsorption capacity of the polymer.

Urban Site - The population distribution in the Dallas-Fort Worth area was modeled as a typical distribution for transportation corridors with high radioactive waste in urban areas. This particular corridor was selected because:

- o The rail lines and truck routes through Dallas-Fort Worth could be used to transport solid nuclear waste to a sparsely populated FWZ that may be located in the southwestern United States.
- o Central Texas has a high occurrence of tornadoes. Although it is not likely, a large tornado could pick up and capture a solid-waste transportation cask.
- o Dallas-Fort Worth was the tenth largest metropolitan area in the U.S. in 1973 (Statistical Abstract, 1975).

For the purposes of this study, we assume an accident in the center of the metropolitan area (area of maximum population density). Also, we assume that the population density falls off in a Gaussian fashion to the rural area. The 1973 population density of the Dallas-Fort Worth metropolitan area (World

area, 1.4 x 10⁶ m², is approximately 400 km². The total city limit area is 11.7 x 10⁶ square kilometers. Within the city limits of Dallas, the 1970 population density was 1200 persons per square kilometer, while the area was 48 square kilometers (Geographical Abstracts, 1975). If we assume that the city limits form a circle, it would have a radius of 19 kilometers; the radius of the metropolitan area under the same assumption would be 83 kilometers. Using these assumptions, one can determine the Gaussian equation for the population density ρ in units of persons per square meter as:

$$\rho = 1.26 \times 10^{-3} \exp(-1 \times 10^{-9} x^2),$$

(7.3-5)
(7.3-5)

where x is the radial distance in meters from the center of the city. Figure 7-1 shows a plot of Eq. (7-5) with ρ in persons per square kilometer. The coefficient of 1.26×10^{-3} was chosen so that the curve between 0 and 1.9×10^4 meters (radius of the city limits) gives equal areas of the city above and below the average population density of 1.23×10^{-3} persons per square meter.

Rural Site — The demography of a rural transportation accident can be assumed uniform over distance. Then, $\rho = \rho_r$ at all distances, where ρ_r is the 1973 U.S. average population density of 2.2×10^{-5} persons per square meter.

7.3.3

Federal Waste Repository Demography

The population distribution (Selected Census...NTS, 1973) around the Nevada Test Site (NTS) has been used in this study as typical of that around a FWR site. The city of Las Vegas dominates the population density within a few hundred kilometers of NTS. Therefore, since the wind blows toward Las Vegas less than 5% of the time, its population has been omitted in order to obtain an "expected value" population dose. Hence, the population was

averaged over 16-kilometer bands, excluding the population of Las Vegas. With the NIF population data fitted to a Gaussian curve by Eqs. (D-3) and (D-4), the equations for population density, in units of persons per square meter are:

1. Within 45 kilometers,

$$\rho = 3.8 \times 10^{-7} \exp[-4 \times 10^{-3}(0.001x - 45)^2],$$

2. Beyond 45 kilometers,

$$\rho = 3.8 \times 10^{-7} \exp[-4 \times 10^{-3}(0.001x - 45)^2] + 2.0 \times 10^{-7}.$$

Note that the population of the FBR has been assumed virtually zero. It is expected that only a few guards would be on duty during normal circumstances. Figure D-5 shows a graph of these equations with ρ in persons per square kilometer.

7.3.4 AREA INTEGRALS OF THE PRODUCT OF CONCENTRATION, AREA, AND POPULATION DENSITY

If we obtain the integral of $\chi/Q \cdot A \cdot \rho$ (where A is the area under consideration), we need then only multiply by the appropriate dose conversion factors to obtain the total population dose (in man-rem) out to 100 kilometers over the swath of the plume or puff. Since the three quantities χ/Q , A , and ρ are interrelated with respect to distance or, equivalently, time, one must integrate their product over either distance or time to obtain population dose. Let this integral be denoted as Σ .

For a plume, Eq. (D-1) incorporates sector-averaging over a 22.5° sector. The integral is,

$$\Sigma = \int_{10^2}^{10^5} \frac{\chi}{Q} \rho \, dA, \quad (7.3 - 6)$$

7.3.-1
Fig. 7.3-1. Population density vs distance from a localized accident at a fuel processing plant with a peak of 100 persons/km².

7.3-2

Fig. 7.3-2. Population density vs distance from a hypothetical transportation accident at the center of an urban area.

7.3-3
Fig. 7.3-3. Population density vs distance from a hypothetical accident at a federal nuclear waste repository.