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CENTER FOR NUCLEAR WASTE REGULATORY ANALYSES

TRIP REPORT

SUBJECT: EPA Briefing on Performance Assessment and DOE/NRC technical exchange on Performance Assessment. (Account No. 20-3702-065)

DATE AND PLACE: EPA Briefing - November 27, 1990
DOE/NRC Technical Exchange - November 28-29, 1990, Albuquerque, NM.

AUTHORS: B. Sagar, P. Nair

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AUTHORS: B. Sagar, P. Nair

PERSONS PRESENT: Participants represented DOE, NRC, Nevada, Affected Indian Tribes, ACNW, and NWTRB. Attendance list for the 27th is attached.

BACKGROUND AND PURPOSE OF TRIP:

A copy of the agenda is attached. An EPA briefing of their current work on performance assessment was held on November 27, 1990. DOE presented the results of some of their recent work on Performance Assessment Calculational Exercises (PACE) on November 28th and 29th.

SUMMARY OF PERTINENT POINTS:

- As per the standard rule for DOE/NRC technical exchanges, no handouts were provided. Neither were any official commitments made. However, it was stated that the DOE expects to provide a written report on Performance Assessment Calculational Exercises (PACE) in early 1991. It was also stated that this was the first technical exchange on performance assessment but that such technical exchanges will be held periodically from now on.

- DOE asserted that performance assessment was an integral part of their site characterization program. It was indicated that a second iteration of PACE has now started.

- DOE is currently performing task prioritization work based on performance assessment. This prioritization will result in their first undertaking those tests that are important to determining site suitability.

- The exchange of technical information between participants was quite free and useful.

SUMMARY OF ACTIVITIES:

November 27, 1990

Ms. Priscilla Bunton from EPA briefed meeting participants on the recent performance assessment work done by their contractor - Arthur D. Little, Inc. This work is actually a repeat of their 1985 work with a few new assumptions and is generic. Numerous simplifying assumptions were made to incorporate scenarios in their analyses. Nobody from Arthur D. Little, Inc. was present to provide greater detail of their work, consequently, a number of questions from the audience were unanswered.

A copy of the EPA's work is attached.

November 28-29, 1990.

In his opening remarks, Russ Dyer of DOE indicated that more technical DOE/NRC exchange meetings on specific topics of performance assessment will be held in the future. He cautioned that the present meeting will only partially cover the performance assessment work done by the DOE over the past two years.

He then proceeded to give DOE's response to NRC's Site Characterization Analysis (SCA) comment #1. This NRC comment has many parts to it, but in the main suggests that the role of performance assessment in integrating the work outlined in DOE's SCP is unclear. Subparts of this comment include the role of performance allocation and how it will be updated; site characterization activities related to scenario definition; methods of developing a CCDF; and the role of iterative performance assessment in delineating the usefulness of site data and its use in updating future data needs.

Russ Dyer in his prepared response affirmed that iterative performance assessment will be used to evaluate data as these come in and determine future data needs. However, in this meeting, it was not possible to discuss in detail all of the DOE response. In any case, this meeting was not the forum to reach a formal resolution of the SCA comments. In the later part of his presentation, Russ discussed DOE's Test and Evaluation Plan which outlines methods to evaluate various tests described in the SCP. The Test and Evaluation Plan is a DOE controlled document. Because the NRC does not receive DOE controlled documents any more, it is not available with the NRC.

The DOE presenters used the rest of the meeting time in presenting the results of the PACE exercises. The PACE exercises were undertaken with the objective of determining capabilities of various DOE contractors to conduct performance calculations. Although different computer codes were compared, benchmarking of codes was not an objective. The PACE were executed in three parts: (1) analysis of a 'nominal case'; (2) analysis of some disturbed cases; and (3) sensitivity analyses. Various national laboratories and other contractors were involved in the PACE work. While some of the work was repeated (using different tools) by various participants, some other work was done by one group and provided as input to other groups. It was made clear that although as much of Yucca Mountain data were used in PACE as possible, that the results are not to be interpreted as an assessment of that site.

Ms. Holly Dockery (SNL) introduced the nominal case which consisted of up to 11 hydrogeologic units (all PACE participants did not use the same number of units in their modeling effort) with a uniform 0.01 mm recharge rate. Generally equivalent continuum models were used with no geometric inclusion of faults and fractures. The models varied from one- to three-dimensional and were generally steady state. Only liquid pathways were considered in the nominal case.

Dr. Mick Apted (PNL) presented the work on source term. For the source term, all waste packages failed at 300 years, but liquid water did not contact waste form in a package until its temperature was lowered to 96 degree Celsius. Thermal analysis was used to determine the 96 degree isotherm to determine when contribution to source term from a particular waste package started. Only four nuclides (Tc-99, I-129, Cs-135, and Np-237) were considered. For the source term a flow rate of 0.5 mm/year (compared to 0.01 mm/year for far-field transport) was used. For calculating the source term, a fractional alteration rate of 10^{-3} for UO_2 was used. Dr. Rawley Barnard (SNL) presented the main results of the nominal case. Calculations were in a deterministic mode. Results were presented as plume locations at 10,000 years rather than as CCDFs at the accessible boundary. From one-dimensional codes, this amounted to finding the penetration depth of the plume. For calculating mass transport, the fluid flow was assumed to be at steady state. Cesium-135 with the highest retardation coefficient (610) did not show any movement. The non-retarding Iodine-129 showed the most movement, however, because of the low recharge rate (0.01 mm/year), the transport was dominated by diffusion. In two- and three-dimensional modeling, the plumes were, therefore, symmetric around the repository horizon.

Dr. Bill O'Connell (LLNL) presented his results on the sensitivity analyses of the source term. He formulated an analytic relationship to find the sensitivity of the source term to air gap dimensions and package failure rates. However, this work was not directly related to the nominal case discussed above.

Ms. Maurine McGraw (PNL) presented her analysis on sensitivity of far field models to assumptions on boundary conditions. She indicated that recharge at the Yucca Mountain may occur from outcrops at the side of mountain. She compared the moisture distribution calculations based on different assumptions on boundary conditions, i.e., fixed pressure versus zero flow versus flow boundaries. Her work indicated that with the assumptions in the model, the boundary effects penetrated a small distance from the boundary.

Ms. K. Birdsell (LANL) discussed her three-dimensional transient simulation of the plume. Although the stratigraphy used by her was different from the nominal case, she compared her results to those obtained from the nominal case. The three-dimensional simulation was found to lead to greater spread of the plume as compared to lower dimensional models. Ms. Birdsell presented her results in a video form.

Dr. Paul Kaplan presented his probabilistic analysis of the ground water travel time. He explained the use of the maximum entropy principal in selecting parameter probability distributions. In view of its flexibility regarding shape (it is a four parameter distribution), he seems to favor the beta distribution. His results showed that given the present data and his assumptions there is 11% probability of violating the 1,000 year travel time criterion. This conclusion, however, is based on the assumption that there is a continuous fracture pathway. He proposes this analysis to define his data needs for site characterization.

Mr. Jack Gauthier (Spectra) discussed his analysis which was designed to check the correctness of the 0.01 mm/yr recharge at Yucca Mountain. Taking saturation data from a few bore holes, he conducted his analysis to see what recharge rate will produce the observed saturation profile in the steady state. He used the one-dimensional TOSPAC code for the analysis. A rate of 0.01 mm/yr seemed to approximate the observed saturation profile. Of course, in addition to many other assumptions, his analysis also suffers from the assumption that the observed profiles are steady state.

Dr. Tom Buscheck (LLNL) presented results of his two-dimensional transient analysis taking into consideration discrete fractures. His analysis indicated that the equivalent continuum assumption underestimates the effect of fractures in calculating advances of saturation fronts.

Dr. George Barr (SNL) discussed the fault tree approach of defining scenarios. Assignment of probabilities to the scenarios was not discussed. Three scenarios were discussed in some detail. G. Valentine (LANL) discussed the basaltic volcanism scenario. He postulated a sill penetrating the repository horizon. The thermal effects in terms of air convection were discussed, again with the help of a video. Dave Gallegos (SNL) discussed the climate change scenario. This scenario was reduced to an elevated recharge scenario with or without water table rise. The results were presented in terms of ground water travel time. The travel time was calculated to be shorter for higher recharges. A. Macintyre (LLNL) talked about the human intrusion scenario. She indicated that the work has progressed to the point of defining a scenario, but the scenario has not been analyzed yet. The scenario consisted primarily of loss of drilling fluid in one of the drifts.

Dr. Russ Dyer (DOE) concluded the workshop by thanking participants. He emphasized that other work related to performance assessment is ongoing in DOE. The NRC and the State of Nevada had no concluding remarks.

IMPRESSIONS/CONCLUSIONS:

The informal technical exchange format is excellent for learning from each other. Most participants seemed to speak freely and the exchange seemed to occur in an unconstrained manner. The information provided is helpful in learning the status of DOE's use of performance assessment as well as any advances in the technology.

RECOMMENDATIONS

The technical exchange format should be pursued to talk about some specific topics. These should include method for developing alternative conceptual models and their inclusion in PA; Definition of scenarios; and effects of simplifying PA models.

SIGNATURE:

P. Nair
Manager - Engineered Barrier System

REFERENCES:

1. Attendance sheet.

2. Agenda - PPA Briefing on Performance Assessment, November 27, 1990.

3. Agenda - DOE/NRC Technical Exchange on Performance Assessment, November 28-29, 1990.

4. High-Level and Transuranic Radioactive Waste Background Information Document, dated June 29, 1990.

CONCURRENCE SIGNATURES AND DATE:

B. Sagar — 12/27/90 Date
Manager - Performance Assessment

Allen R. Whiting — 12/27/90 Date
for Director-Systems Engineering and Integration

1/27/90

ATTENDANCE

EPA BRIEFING ON PERFORMANCE ASSESSMENT

NAME	ORG	TELEPh.
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Ron Ballard	NRC	301-492-3462
Seth M Coplan	NRC	301-492-0410
NORMAN EISENBERG	NRC	301-492-0324
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11/27/90

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EPA BRIEFING ON PERFORMANCE ASSESSMENT

November 27, 1990
1:00 - 5:00 PM

Albuquerque Marriott Hotel

Albuquerque, NM

PURPOSE: To provide an opportunity for EPA to present its approach to performance assessment and a forum for exchange of performance assessment assumptions and methodologies with DOE and NRC.

Agenda Topic	Discussion Leader
o EPA performance assessment results in support of re-promulgation of 40 CFR 191	F. Dunton/EPA
Open discussion	all
Adjourn	

November 14, 1990

AGENDA

DOE-NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT

November 28-29, 1990
Wednesday 8:00 - 5:00 PM
Thursday 8:00 - 5:00 PM

Albuquerque Marriott Hotel

Albuquerque, NM

PURPOSE: To discuss performance assessment calculation exercises (PACE) conducted during fiscal year 1990 and DOE's planned responses to SCA Comment 1.

SCOPE: This technical exchange will focus on performance assessment calculation exercises (PACE), including descriptions of the nominal case, sensitivity studies, related efforts, and disturbed configurations. Future work in this area will also be presented. Discussions of the Test and Evaluation Plan and performance assessment organization will address DOE's planned response to SCA Comment 1.

Wednesday, November 28, 1990

Agenda Topic	Discussion Leader
Opening Remarks	DOE, NRC, State
o Introduction	DOE
o Discussion of SCA Comment 1	DOE
- Summary of DOE's response * Test and Evaluation Plan * Test and Evaluation Planning Basis * Performance assessment organization	
Discussion	all

DOE-NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT
(continued)

Agenda Topic	Discussion Leader
Performance Assessment Calculation Exercises for Fiscal Year 1990	
o Nominal case	DOE
- Introduction and desc ption	
- Radionuclide source term	
- Comparison of COVE 2A and HYDROCOIN	
- Summary and comparison of results	
Discussion	all
o Sensitivity studies	DOE
- Source term sensitivity studies	
- Boundary condition sensitivity studies	
- Transport calculations in three dimensions	
Discussion	all
Adjourn	

Thursday, November 29, 1990

Opening remarks, comments from Day 1	DOE, NRC, State
Performance Assessment Calculation Exercises for Fiscal Year 1990 (continued)	
o Related efforts	DOE
- Theory behind the probabilistic approach to PA	
- Comparison of hydrology data with "inverse" calculations	
- Non-equilibrium fracture flow	
Discussion	all
o Disturbed Configurations	DOE
- Update on event trees and scenario development	
- Basaltic volcanism PDM and preliminary calculations	
- Climate change/elevated flux cases	
- Human intrusion PDM and preliminary calculations	
Discussion	all

DOE-NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT
(continued)

Agenda Topic	Discussion Leader
Performance Assessment Calculation Exercises for Fiscal Year 1990 (continued)	
o Future Work	DOE
- Future structure of PACE - Probabilistic analyses - Relation to COVE and INTRAVAL - Relation to site suitability and issue resolution	
Discussion	all
Concluding discussion and final remarks	DOE, NRC, State
Adjourn	

• AGENDA •

DOE/NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT

NOVEMBER 28 - 29, 1990

WEDNESDAY, 11/28

8:00 AM

- OPENING REMARKS (DOE / NRC / STATE)
- INTRODUCTION (Russ R. DYER / DOE - YMSCPO)
- DISCUSSION OF SCA COMMENT #1 AND DOE RESPONSE (R. DYER / DOE - YMSCPO)
- TEST AND EVALUATION PLAN (R. DYER / DOE - YMSCPO)

Yucca Mountain Site Characterization Program Office

10:30 AM - BREAK -

10:45 PERFORMANCE ASSESSMENT CALCULATIONAL EXERCISES FOR FISCAL YEAR 1990 (PACE-90)

Performance Assessment Calculational Exercise

- DESCRIPTION OF NOMINAL CASE (Holly Dr H. DOCKERY / SNL)
- RADIONUCLIDE SOURCE TERM (M. APTED / PNL)

12:00 - LUNCH -

• AGENDA •

DOE/NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT

NOVEMBER 28 - 29, 1990

WEDNESDAY, 11 / 28

1:00 PM PACE-90 NOMINAL CASE (CONTINUED)

- COMPARISON TO COVE 2A AND HYDROCOIN — Polly (P. HOPKINS / SNL)
- SUMMARY AND COMPARISON OF RESULTS FOR PACE-90 NOMINAL CASE (R. BARNARD / SNL)

2:45 - BREAK -

3:00 SENSITIVITY STUDIES

- SOURCE TERM SENSITIVITY STUDIES (W. O'CONNELL / LLNL)
- BOUNDARY CONDITION SENSITIVITY STUDIES — Maurice (M. MCGRAW / PNL)
- TRANSPORT CALCULATIONS IN THREE DIMENSIONS (K. BIRDSELL / LANL)

5:00 - ADJOURN -

• AGENDA •

DOE/NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT

NOVEMBER 28 - 29, 1990

THURSDAY, 11 / 29 8:00 AM

- OPENING REMARKS, COMMENTS FROM DAY 1 (DOE / NRC / STATE)

PACE 90: RELATED EFFORT

- INTRODUCTION (R. DYER / DOE - YMPSCO)
- THEORY BEHIND PROBABILISTIC APPROACH TO PA (P. KAPLAN / SNL)
- COMPARISON OF HYDROLOGY DATA WITH "INVERSE" CALCULATIONS (J. GAUTHIER / SPECTRA)
- NON-EQUILIBRIUM FRACTURE FLOW (T. BUSCHECK / LLNL)

10:30 -BREAK-

10:45 PACE - 90: DISTURBED CONFIGURATIONS

- UPDATE ON EVENT TREES AND SCENARIO DEVELOPMENT (G. BARR / SNL)
- BASALTIC VOLCANISM (G. VALENTINE / LANL)

12:00 - LUNCH -

• AGENDA •

DOE/NRC TECHNICAL EXCHANGE ON PERFORMANCE ASSESSMENT

NOVEMBER 28 - 29, 1990

THURSDAY, 11 / 29

1:00 PM PACE - 90: DISTURBED CONFIGURATIONS (CONTINUED)

- CLIMATE CHANGE / ELEVATED FLUX CHANGE (D. GALLEGOS / SNL)

- HUMAN INTRUSION CASE (A. MACINTYRE / LLNL)

2:30 PM - BREAK -

2:45 FUTURE WORK (R. DYER / DOE - YMSCPO)

4:00 CONCLUDING DISCUSSION AND FINAL REMARKS (DOE / NRC / STATE)

- ADJOURN -

Arthur D Little

High Level and Transuranic Radioactive Wastes

Background Information Document

June 29, 1990

Submitted to Environmental Protection Agency Office of Radiation Programs

Arthur D. Little, Inc.
Acorn Park
Cambridge, Massachusetts
02140-2390

Reference 64582

List of Tables

List of Figures

8.5.1 Population Risks

8.5.1.1 System Models. The waste disposal systems considered in this risk assessment are based on current DOE plans to develop mined geological repositories for disposal of high-level radioactive wastes. Such repositories consist of underground mines or excavations with working levels between 300 and 1000 meters below the surface. The repositories being considered are in rock formations of bedded salt, salt domes, basalt, and tuff. Recent focus has shifted to evaluating tuff as a host rock for disposal of high level radioactive waste.

The radioactive wastes themselves will consist of either spent fuel from nuclear power reactors or solidified reprocessing wastes in a relatively durable form, such as borosilicate glass. These wastes will contain a wide variety of radioactive elements, ranging from highly active fission products with relatively short lives, to long-lived elements such as transuranic radionuclides created through neutron capture by uranium atoms. For disposal in a mined geologic repository, the wastes would be packaged in canisters and placed in holes in the walls or floors of mined rooms in the repository. After emplacement of the wastes, the repository would be backfilled to enhance its mechanical stability and to retard the movement of fluids. The repository's various connections (e.g., shafts and boreholes) to the surface would be severed and sealed. The intent in selecting a repository is to provide a highly stable geologic environment in which ground water contact with the radioactive waste is inhibited or greatly restrained. However, despite the care that will be exercised during the development of a repository, possible future disruptions could lead to the release of radioactive wastes.

The purpose of the risk assessments carried out by the Agency is to identify the most important mechanisms that could lead to releases of radioactive waste to the accessible environment and to estimate the likelihood and consequences of the releases over long time periods in the future. While this analysis varies from one geologic environment to another, the structure of the analysis can be represented as shown in Figure 8.5-1. The components of the system to the right of the vertical dotted line represent the "accessible environment." The components on the left side of the diagram represent the release and transport mechanisms from the repository to the accessible environment.

In order for radioactive wastes to reach the accessible environment, radioactive material must be released from the waste form itself, which might be a borosilicate glass, unprocessed spent fuel, or some similar type of structure. Having left the waste form, such radionuclides must escape from the waste canister and then enter the backfilled openings of the repository. Radionuclides may travel from the repository to the accessible environment in two general ways: 1) direct pathways to the land surface, such as might occur if future generations penetrated the repository during an exploratory drilling program and accidentally contacted the

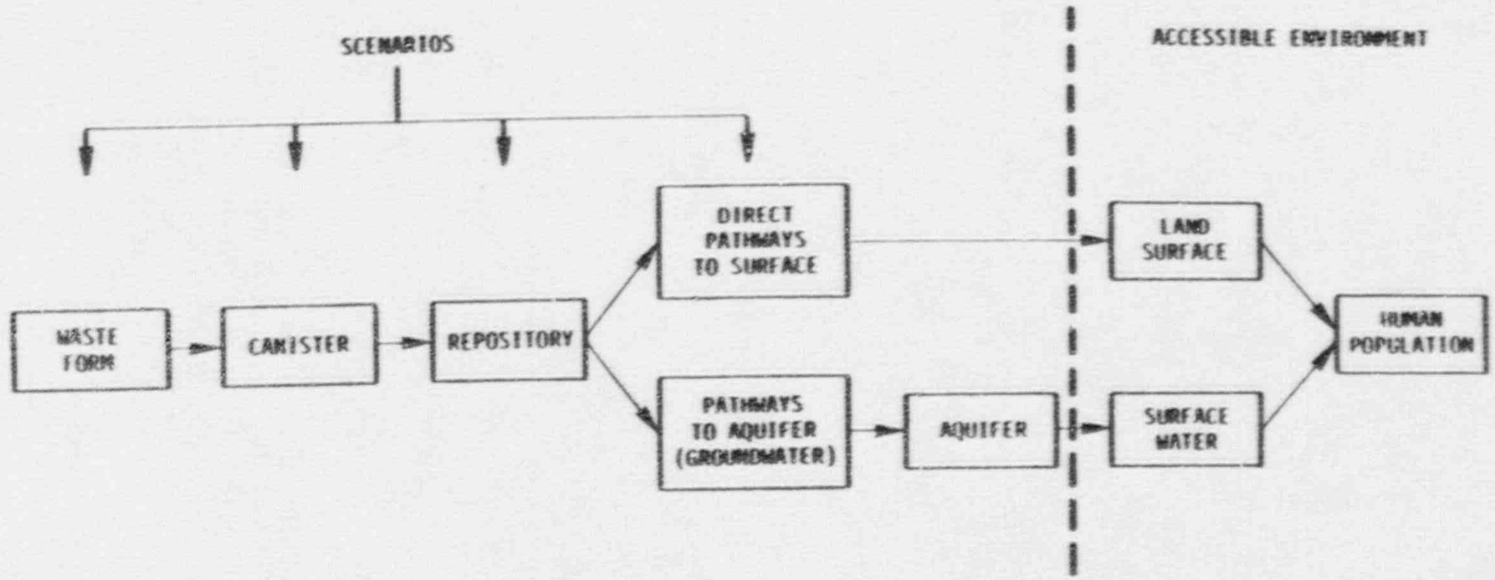


Figure 8.5-1: Components included in the risk assessment of radioactive waste releases

wastes, and 2) migration in slowly moving ground water to an aquifer and then to the surface.

The movement of radionuclides from the waste form, through the canister, through the repository, and ultimately to the accessible environment depends on a number of possible future scenarios that might alter the conditions of the underground environment. Such processes are called "release mechanisms" or "release scena)s." They may affect any of the four components indicated in Figure 8.5-1. The risk analyses reported here consider a number of potential release mechanisms. The results of the calculations for individual release mechanisms are then combined into integrated representations of the risk from a hypothetical repository.

8.5.1.2 Site Parameters. Models were originally assembled for four of the five sites tentatively identified by the DOE, in accordance with the NWPA, for nomination as potential sites for the first repository. These four sites are: 1) the bedded salt deposits in the Palo Duro basin, 2) the bedded salt deposits in the Paradox Formation in Utah, 3) the basalt flows on the Hanford reservation in Washington, and 4) the unsaturated volcanic tuff formations at Yucca Mountain in Nevada. In addition, the Agency assembled several models of repositories in granitic formations. Granite repositories are no longer seriously considered for disposal of high level radioactive waste, therefore they are not considered in this risk assessment.

A number of parameters are used to describe the geometry and hydrologic conditions assumed for each of the model repository sites. These are listed in Table 8.5-1 and can best be understood in conjunction with the generic cross section shown in Figure 8.5-2, with certain exceptions. The conceptual framework of the lithology for most sites is that the repository is located between two aquifers called, respectively, the "upper aquifer" and the "lower aquifer." The tuff model, however, assumes unsaturated conditions at the repository level with an underlying aquifer only.

To simulate conditions present at a real site, the upper and lower aquifer do not generally represent single hydrostratographic units, but rather they represent "synthetic aquifers" whose properties are defined to approximate the combined properties of a number of transmissive units above and below the repository horizon. For example, if a number of such transmissive units are present above the repository at a particular site and if the application of a generic model described here is intended to represent conditions similar to those at the site, then one can calculate the combined volumetric flows in the upper units and define appropriate hydrologic parameters so that the synthetic aquifer represents the same total flow. Similarly, by varying one or more additional parameters, it is possible to simulate the effective fluid velocity in any one of the actual units. This will be illustrated in subsequent sections when specific lithologies are discussed.

Assumpt.

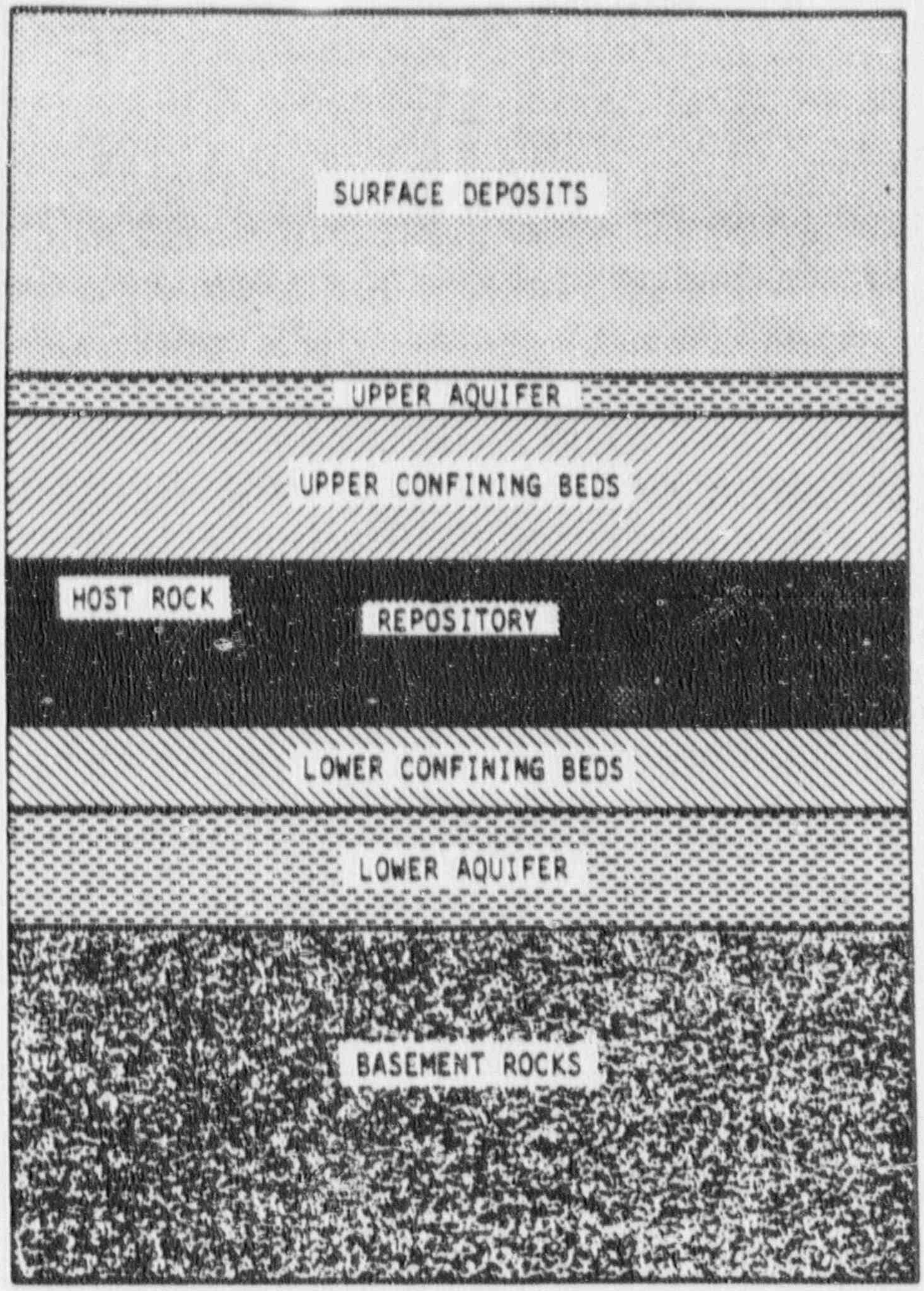
Table 8.5-1: Site parameters considered in risk assessment

Parameter	Basalt	Palo Duro Bedded Salt	Paradox Bedded Salt	Tuff
Distance from repository to overlying aquifer, meters	22	1105[a]	666	150[a]
Hydraulic conductivity of the host rock between the repository and the aquifer, after thermal effects (centimeters/sec)	10^{7}	0	0	4×10^{9}
Porosity of rock between the repository and aquifer	10^{4}	n/a	n/a	0.1
Natural hydraulic gradient between the repository and aquifer	0.025	0.26[a]	0	1.0[a]
Thickness of aquifer, meters	30	300	20	100
Hydraulic conductivity of aquifer (centimeters/sec)	10^{3}	5×10^{6}	2×10^{5}	2.38×10^{4}
Porosity of aquifer	0.01	0.05	0.2	0.005
Horizontal gradient in aquifer	0.0003	0.005	0.02	0.00034
Horizontal distance along the aquifer to the accessible environment, meters	2000	2000	2000	5000

[a]For these sites, ground water flow is assumed to be downward to the lower aquifer, and all aquifer characteristics are those of the underlying aquifer.

Source: EPA85

Figure 8.5-2: Cross section of the rock formation at the generic repository site



The general way in which the REPRISK model constrains the risk analyses is that the upper aquifer is considered to be the pathway of ground water release of any radionuclides that leave the repository. (An upward gradient is considered to exist along any pathway that might be present or develop between the lower aquifer and upper aquifer.) This is why greater emphasis is placed on the properties of the upper aquifer in Table 8.5-1. At potential repository sites, however, the hydrogeologic environment may be different from that assumed in the generic model. For example, there may be no significant aquifer below the repository (as in a number of crystalline rock sites), or above the repository (as in the case of a repository in the unsaturated zone), or there may be a prevailing gradient that is downward from the upper aquifer, in which case the lower aquifer would appear to be the more likely release pathway. The REPRISK model for tuff assumes downward flow from the repository, in the unsaturated zone, to the uppermost aquifer. A vertical gradient of one (1) is assumed, where hydraulic conductivity in the unsaturated zone is equal to the flux rate. Between the repository and the saturated zone, natural variations in hydrologic properties are simplified to provide REPRISK with a single set of "vertical leg" parameters for each conceptual release model. Potential releases to the accessible environment are modeled through the uppermost aquifer, located 150 to 200 meters below the repository.

With respect to the specific hydrologic parameters listed in Table 8.5-1, hydraulic conductivity is used in conjunction with Darcy's Law to estimate volumetric flow rates through various components, such as pathways from the repository to the upper aquifer or along the upper aquifer itself. For further elaboration on the mathematical equations referred to, one may consult EPA80 and the references cited there. Only Darcian flow has been treated in the analyses, and work by DOE at specific sites tends to confirm that the flow regimes are such that this approach is adequate. The porosity is used to convert volumetric flow rates into average effective fluid velocities in the direction of movement. In particular, the volumetric flow rate is divided by porosity to obtain an effective fluid velocity. It is important to make this distinction when considering the time of arrival of contaminated ground water at the discharge point to the accessible environment. Hydraulic conductivities, specified as part of the description of a conceptual site, may be modified in the course of the modeling as a result of other phenomena assumed to be present, as described later in the section on release mechanisms. Ref.

Geochemical characteristics constitute another important set of site parameters. These include the ways in which the movement of any radioactivity released from the waste would be slowed or "retarded" by geochemical interactions with the surrounding rocks, and the expected limits that the geochemical conditions would place on the solubility of many of the radioactive elements in the waste. Table 8.5-2 lists the retardation factors and solubility limits used in each of the repository models for the elements of primary interest. The retardation factors describe the relative speed that the respective radionuclides travel in ground water compared with the speed of the ground water itself (i.e., an isotope with a retardation factor

Table 8.5-2: Geochemical parameters used in risk assessment

Parameter	Basalt	Palo Duro Bedded Salt	Paradox Bedded Salt	Tuff
Retardation Factors[a]				
carbon	1	1	1	1
strontium	200	10	10	21000
zirconium	5000	1000	1000	2600
technetium	5	5	5	1
tin	1000	100	100	530
iodine	1	1	1	1
cesium	1000	10	10	41000
uranium	50	20	20	58
neptunium	100	50	50	58
plutonium	500	200	200	740
americium	500	1000	1000	24000
Solubility Limits[a] (parts per million)				
carbon	none	none	none	none
strontium	0.6	none	none	none
zirconium	0.0001	0.0001	0.0001	0.0001
technetium	0.001	0.001	none	none
tin	0.0001	0.001	0.001	0.0001
iodine	none	none	none	none
cesium	none	none	none	none
uranium	0.001	0.01	0.01	none
neptunium	0.0001	0.001	0.001	0.1
plutonium	0.00001	0.001	0.001	0.001
americium	0.001	0.1	0.01	0.001

[a]For use with all isotopes of radionuclides listed.

Source: EPA85

of 10 moves one tenth (1/10) as fast as the ground water). The methods by which REPRISK utilizes retardation values has been described previously (Sm82), which assumes that dispersion can be neglected in the calculations. This same assumption has been used extensively in the literature. Ref.

8.5.1.3 Repository Parameters. Certain assumptions need to be made pertaining to the geometry and physical characteristics of the repository. The DOE has frequently modified its conceptual designs for repositories. An examination of the Agency's risk analysis models indicates that they are not highly sensitive to these engineering assumptions. Current engineering plans for the tuff repository have updated the generic repository parameters utilized for modeling purposes in the 1985 BID. These parameters are summarized in Table 8.5-3. Predicted waste volumes now require the utilization of 5.6×10^6 square meters (m^2) for the long term disposal of an estimated equivalent of 70,000 metric tons of uranium (MTU) waste.

The currently envisioned tuff repository will be constructed in unsaturated volcanic tuff. The water table is approximately 150 to 200 meters below the repository at its nearest point. Seasonal ground water fluctuations in this region are negligible. Current designs for tuff repository are shown in Figures 8.5-3 and 8.5-4.

The repository design assumes free drainage in the surrounding rock and the ability to dissipate the anticipated thermal loading from the decay of spent fuel. A controlled area consisting of 100 square kilometers surrounding the repository will provide a distance of approximately five kilometers between the repository edge and the accessible environment in a down-gradient direction. Aquifer flow directions in the tuff model (below the proposed repository) are from the northwest to the southeast. For modeling purposes, the cross sectional area of the ground water flow path is defined by the length of the repository (perpendicular to the flow path), multiplied by the thickness of the uppermost aquifer:

3,500 meters x 95 m - 3.3 E5 m^2.

The mined volume of the repository, as well as the porosity of the backfill, must be considered in calculating the amount of radionuclides that might dissolve in the ground water that would gradually seep into the repository after its closure. Because such dissolution might be limited by solubility factors, this water volume is significant to some models. For example, it can be used to estimate the amount of dissolved radionuclides that might be withdrawn by an exploratory well that penetrates the repository at some point in the future. ?

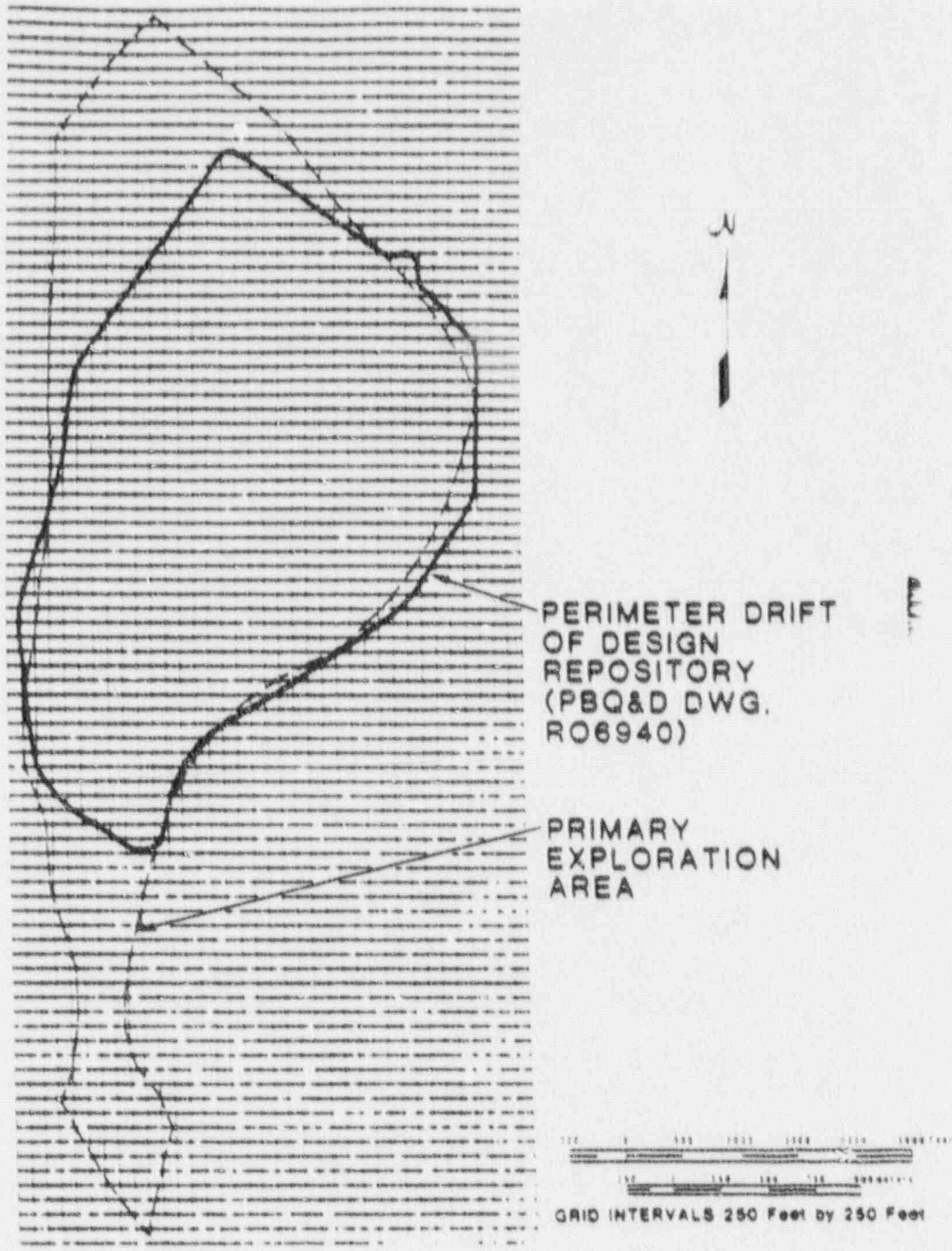
The dissolution of wastes into water percolating down into the tuff repository is assumed only to take place in the actual quantity of water that is in the "footprint" of one of the waste canisters. This is a relatively small fraction of the water percolating downward through the repository and provides a very small volume to Assm

Table 8.5-3: Repository parameters considered in risk assessment

Parameter	Value
Dimensions of repository:	
Length:	3,500 meters
Width:	2,000 meters
Height:	4 meters
Total mined-out volume:	10^7 meters^3
Average porosity of backfilled repository	0.2
Time to maximum backfill compaction due to plastic flow (salt only)	200 years
Number of canisters of high-level waste:	35,000
Canisters per drift:	20
Length of waste drift:	500 meters
Canister spacing:	26 meters

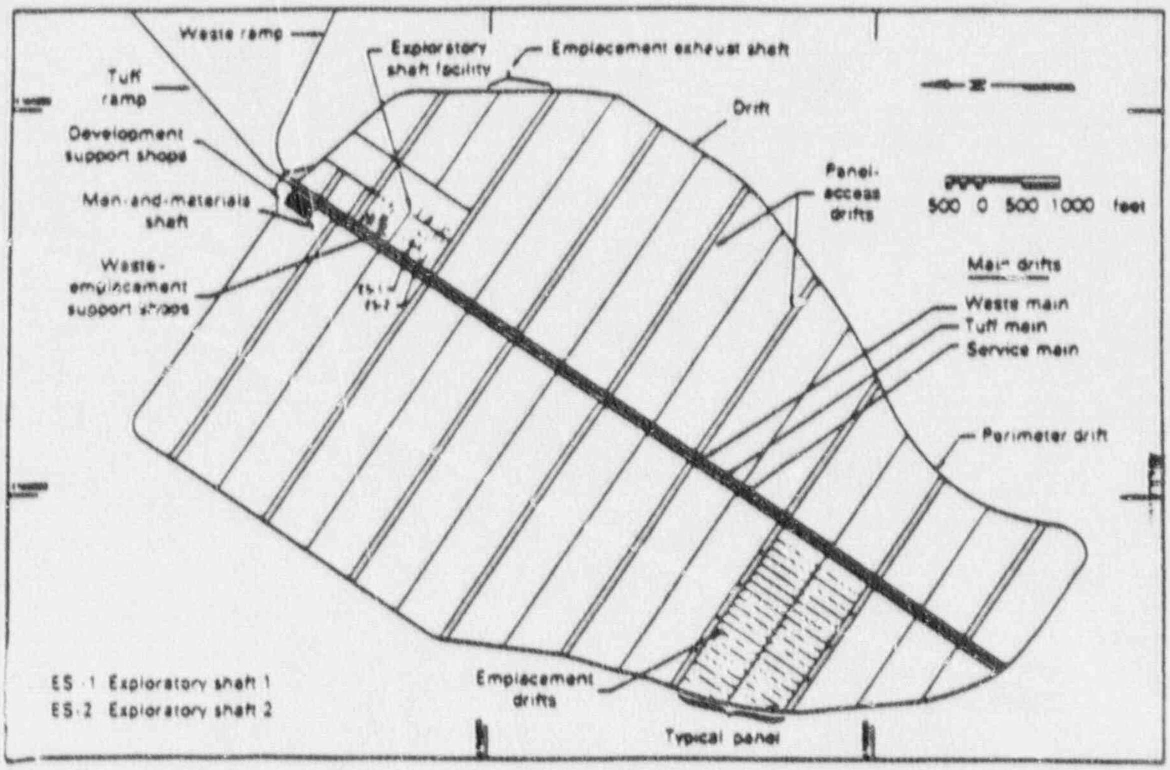
Source: EPA85

Figure 8.5-3: Underground Repository Layout



SOURCE: GE CALMA PRODUCT NO. 0114

Figure 8.5-4: Underground Repository Layout for Vertical Waste Emplacement



(Source: DOE/RW-0199)

which solubility limits must be applied. Leaching (i.e., leach rate limited dissolution), as provided for in REPRISK, is independent of the amount of water available. It is assumed that a given unit of water spends only a relatively short time in proximity to a waste container, and during this time period REPRISK determines whether the corresponding dissolved concentration results from a leach rate limited process or a solubility limit. In most cases, this will turn out to be a solubility limit, due to the small quantity of water available. After the water runs down the side of the canister and re-enters the geologic system, it is assumed to be well mixed with water that has moved vertically in parallel, but that has not directly contacted a waste container. This is reasonable because the slow flow rates should allow for substantial dispersion in the horizontal direction. assump

The case of a salt repository is different, because it would be expected to gradually seal after closure as result of salt creep. In this case, the time to closure and the amount of moisture present at that time are important for the risk analysis.

Waste emplacement methods are a critical component of the waste repository design. Vertical emplacement of waste was the preferred method, but options have expanded to include horizontal emplacement of wastes. This risk assessment assumes vertical emplacement. Either strategy must consider the requirement that the 18 proposed panels will dissipate the impact on thermal conditions from spent fuel which peaks within 500 years after closure. The maximum thermal loading is estimated to be 57 KW/acre, based on the potential heat generated by the spent fuel waste components and the heat dissipation properties of tuff.

Vertical emplacement involves the drilling of vertical boreholes into the drift floor and emplacement of one container of waste in each vertical borehole. Vertical emplacement holes are drilled at a distance between individual containers of 26 m (85 ft.). A borehole must be 7.6 m deep and 76 cm in diameter.

In the tuff repository, a metal casing and support plate will be inserted into the hole. A waste container is inserted into the casing and capped by a metal plug. Crushed tuff is packed over the metal shielding and the borehole is subsequently closed with a metal cover. An air gap of approximately 5 cm width will surround each waste package, thereby enhancing long term container performance by breaking hydraulic continuity between the rock and the waste package.

Design parameters for horizontal emplacement are similar to the vertical emplacement design, with some exceptions. Emplacement drifts would be excavated at greater distances from each other than that considered for vertical emplacement. The waste panels for horizontal emplacement are required to be approximately twice the size of vertical emplacement panels. Unlike the single waste container per vertical borehole, horizontal emplacement establishes the maximum load to be 18 containers of DHLW and 14 containers of spent fuel.

8.5.1.4 Waste Form Parameters. The high level radioactive waste package is assumed to consist of two main components: the waste form and the waste canister. The waste form consists of an initial inventory of radionuclides contained in a physical matrix, which may be unreprocessed spent fuel, borosilicate glass or some other alternative.

Spent fuel is an enriched uranium oxide matrix containing transuranic nuclides, fission products and activation products from commercial nuclear power reactors. The form of these wastes will include components such as:

1. Intact assemblies,
2. Metallic components including space grids or tubing,
3. Contaminated zirconium alloy or stainless steel cladding, and
4. Canisters and consolidated fuel rods.

Additional waste forms, including "failed fuel" and non-fuel hardware are expected. "Failed fuel" is structurally damaged fuel rods contained by a protective canister to reduce the release of particulates prior to emplacement. Non-fuel hardware includes the metal fittings and structural components of the intact assemblies. Defense high level waste (DHLW), generated by fuel reprocessing or specific defense-related activities at DOE sites, will be stored at the repository. Heat generation due to radioactive decay of DHLW is expected to be relatively insignificant. A staggered emplacement of DHLW with spent commercial fuel waste is intended to help reduce the projected thermal gradient.

Table 8.5-4 shows the assumed initial inventory of radionuclides. A risk assessment based on the radionuclides shown provides an adequate representation of the risks associated with a repository.

The canister is a protective container assumed to inhibit the leaching or the dissolution of the waste form and the consequent transport of radionuclides toward the accessible environment. In these risk assessments the performance of the canister has been conservatively approximated by a single lifetime. Up until this time is reached, no radionuclides are assumed to be released from undisturbed waste packages. Once the lifetime is reached, the canister is neglected in the subsequent analysis, thus ignoring any benefits the remaining portions of the canister might offer.

Currently six types of metal canisters are under review for their individual mechanical, physical and microstructural properties. These six container designs are either austenitic alloys, copper or copper based alloys. Currently accepted reference waste package material is AISI 304L stainless steel. Each container is designed to have a diameter of 66 cm (2.2 ft.) with a length for the spent fuel or DHLW ranging from 3.1 to 4.7 m (9-15 ft.). Containers will weigh between 2.7

Table 8.5-4: Breakdown of the Inventory by Nuclide

Nuclide	Inventory (Curies)	Half Life (Years)
Am-241	1.7E8	458
Am-243	1.7E6	7650
C-14	2.8E4	5730
Cs-135	2.2E4	3E6
Cs-137	8.6E9	30
I-129	3.8E3	1.6E7
Np-237	3.3E4	2.14E6
Pu-238	2.2E8	89
Pu-239	3.3E7	24400
Pu-240	4.9E7	6260
Pu-242	1.7E5	3.8E5
Sr-90	6.0E9	29
Tc-99	1.4E6	2.1E5
Sn-126	5.6E4	1.0E5
Zr-93	1.9E5	9.5E5

Artificial Inventory

Nuclide	Inventory (Curies)	Half Life (Years)
Am-241	8.0E9	458
Np-237	1.7E8	2.14E6

and 6.4 metric tons when fully loaded with waste and will average 3.3 KW/container in decay heat.

8.5.1.5 Release Mechanisms. The release mechanisms through which radioactive wastes may escape the tuff repository include normal ground water flow, faulting, volcanism, and inadvertent intrusion by exploratory drilling. These release mechanisms may lead to the direct transport of radionuclides to the land surface or to the atmosphere, or they may lead to the ground water transport of waste away from the repository.

All scenarios involving ground water release are modeled in REPRISK using a Darcian flow system. REPRISK release pathways normally involve a vertical and a horizontal leg. In the case of the tuff repository, the vertical leg is from the repository down to the aquifer. The horizontal leg is the distance from the edge of the repository to the accessible environment.

The five values needed to predict Darcian flow for each leg are distance, hydraulic conductivity, porosity, gradient, and cross-sectional area. The first four are used to find travel time by the expression:

$$T = (d*\eta)/(i*K)$$

where:

- T is the resulting fluid travel time in years (years)
- d is the length of the leg (meters)
- η is the effective porosity
- i is the gradient
- K is the hydraulic conductivity (meters/years)

Volumetric flow is found by:

$$V=K*i*A$$

where:

- V is the volumetric flow (cubic meters/year)
- i is the gradient, and
- A is the cross section of the pathway

A conservative modeling approach requires considering the largest defensible volumetric flow and the smallest defensible travel time. Therefore, the largest realistic gradient and hydraulic conductivity and the smallest realistic distance and porosity are generally used. As a conservative assumption, flow in all legs for the tuff repository has been modeled as fracture flow, which reduces travel time. assump

The purpose of this section is to review each of the release mechanisms and summarize the conceptual models that have been used in the risk assessment. Additional equations used to implement the conceptual models are discussed in the risk assessment that supported the proposed rule (Sm82). Further background on the release mechanisms can be found in other technical support documents for the proposed rule (ADL79). The release mechanisms considered by REPRISK for all modeled repository hosts, together with the parameters describing them, are shown in Table 8.5-5.

8.5.1.5.1 Normal Ground Water Flow. Except for the case of a repository in salt, normal ground water flow refers to the movement of water through the repository horizon according to the natural hydrologic conditions, perturbed to some degree by the presence of the repository. During the construction and operation of the repository, water in the surrounding rock would be expected to gradually drain so that the rock will enter an unsaturated condition near the openings. After the end of the operational period and sealing of the repository, water would be expected to gradually seep back into pores and fractures in the rock and establish a flow regime connected to the regional ground water system. In the case of tuff, the repository would be located in a rock mass that would be unsaturated at the start.

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?

The resulting flow patterns may be different from those prior to the excavation of the repository. For example, the heat generated by the waste may modify the hydraulic conductivity of the surrounding rock and may also change the properties of water, making it less dense and less viscous. The lower density can lead to a buoyancy effect that may cause an increased vertical hydraulic gradient. The decreased viscosity may enable the water to flow more easily through the rock and hence allow for potential increases in flow rates.

Tuff note

In the case of a repository in tuff, normal ground water flow refers to the downward percolation of water through the unsaturated rock toward the water table. This downward movement is not expected to be influenced greatly by the presence of the repository, because the limiting factor is essentially the amount of water available. Based on calculations of unsaturated vertical flow velocity, undisturbed ground water flow does not result in a release to the accessible environment over 10,000 years, therefore, all modeled vertical legs for the tuff repository result from failure events such as faulting or inadvertent borehole intrusion.

Salt

In the case of a salt repository, the salt formations are assumed to be effectively impermeable to ground water movements. Therefore, "normal" ground water flow has been used to characterize a number of processes, such as leakage along repository shafts, which are grouped under this category because they are treated in the same part of the REPRISK model. These mechanisms were reviewed in the risk assessment supporting the proposed rule and were generally not expected to cause releases within 10,000 years after disposal (Sm82).

Table 8.5-5: Release mechanism parameters considered in risk assessments

Parameter	Basalt	Bedded Salt	Tuff
Normal Ground Water Flow			
Fraction of the repository with which ground water can communicate	1.0	-	1.0
Hydraulic conductivity of flow path (centimeters/sec)	10^{7}	0.0	$4x10^{9}$
Porosity of flow path	0.0001	-	0.1
Cross-sectional area of flow path (square meters)	$8x10^{6}$	-	$6x10^{6}$
Probability of occurrence	1.0	-	1.0
Fault Movement			
Fraction of the repository with which ground water can communicate	1.0	$5x10^{2}$	1.0
Hydraulic conductivity of flow path (centimeters/sec)	10^{2}	10^{4}	$4x10^{9}$
Porosity of flow path	0.1	0.1	$3x10^{4}$
Cross-sectional area of flow path (square meters)	4000	4000	30000
Frequency of occurrence	$3x10^{5}$	10^{6}	$1x10^{5}$

Table 8.5-5: Release mechanism parameters considered in risk assessments (continued)

Parameter	Basalt	Bedded Salt	Tuff
Breccia Pipe Formation			
Fraction of the repository with which ground water can communicate	-	1.6×10^{2}	-
Hydraulic conductivity of flow path (centimeters/sec)	-	10^{2}	-
Porosity of flow path	-	0.2	-
Cross-sectional Area of flow path (square meters)	-	30000	-
* Frequency of occurrence (per year, after 1000 years)	0	10^{8}	0
Drilling (does not hit a canister)			
Fraction of the repository with which ground water can communicate	1.0	6×10^{5}	1.0
Volume of ground water brought to the surface	200	1.14	$.314M^{3}$
* Frequency of occurrence (per year after 100-year control period ends)	10^{3}	2×10^{2}	1.8×10^{3}

Table 8.5-5: Release mechanism parameters considered in risk assessments (continued)

Parameter	Basalt	Bedded Salt	Tuff
Drilling (hits a canister)			
Fraction of one canister brought to the surface	0.15	0.15	0.053
Frequency of occurrence (per year	10^{5}	$2x10^{5}$	$6.1x10^{6}$
Volcanoes			
Fraction of the repository inventory dispersed	$4x10^{4}$	$4x10^{4}$	$4x10^{3}$
Frequency of occurrence (per year)	$6x10^{10}$	10^{10}	$1x10^{8}$

Source: EPA85

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8.5.1.5.2 Fault Movement. The faulting release mechanism covers both the case of a new fault occurring at a repository site (and intersecting the repository itself) as well as the case of the reactivation of an old, apparently stable fault. Fault reactivation is assumed to lead to an intersection with the repository, which is treated in the model as a vertical planar structure with increased hydraulic conductivity, greater than that of the original rock.

For the risk assessment of a tuff repository, faulting is modeled in the same conceptual framework as drilling scenarios. A hypothetical fault is assumed to create a pathway from the repository level to the lower aquifer, in which there is accelerated percolation and no retardation. This may be an overly conservative hypothesis. It is incorporated to establish bounding values for demonstrating compliance with release limits. In fact, the opposite may be true, where the "wicking effect" of unsaturated tuff may draw water away from the fault. In the unsaturated zone, volumetric flow in the faulting event is limited to the unsaturated flow rate in the rock matrix. That is, the flow is not Darcian (in which case it would be controlled by hydraulic conductivity and gradient), but rather it is limited by water availability. However, since the REPRISK code only contains models of Darcian flow, artificial Darcian parameters have been created for this particular application in order that the flow rates and velocities calculated by REPRISK turn out to be those that result from the actual unsaturated flow situation.

note

In addition to creating a flow pathway, the model for faulting assumes that this can be a relatively violent and disruptive event, destroying the integrity of waste packages within a certain distance of the fault. The result can be an earlier onset of leaching (if this had not already begun by the time of fault movement). As in the case of normal ground water flow, the only releases by faulting are assumed to be via a pathway connecting the repository with an aquifer, thereby enabling ground water transport of the waste. Faulting has been treated as a random stochastic process for purposes of the Agency's risk assessments. This is not to say that faulting is a random process, but only that faulting at a real repository site should be able to be bounded by modeling it as a physical process that occurs randomly. The likelihood of new or reactivated faults is estimated on the basis of geometric arguments and simple probability concepts (ADL79).

8.5.1.5.3 Inadvertent Intrusion by Exploratory Drilling. Future exploratory drilling at a repository site cannot be ruled out, even though steps will be taken to signal to future generations that dangerous materials are buried there. The Agency has considered a wide range of potential purposes for drilling in different geologic media and has estimated drilling rates that are intended to be upper bounds on the future likelihood of drilling at a repository site. In estimating these values, no credit has been taken for the communication to future generations of the presence of the repository, except that for 100 years after disposal, it is assumed that such communications would be completely effective.

The purposes for future drilling may vary from one kind of geologic setting to another. For example, for salt deposits in sedimentary basins, the dominant drilling is expected to be in search of oil and gas, whereas in a tuffaceous terrain, the dominant purpose might be exploration for water or minerals (ADL79). The drilling release mechanism contains a number of components. First, it is possible that the drill hole would actually intersect a waste canister, resulting in waste materials brought directly to the surface with the drilling mud. In this case, such materials would likely go unnoticed or unrecognized, at least for some time, and hence they could be distributed on the land surface. Even if a drill hole does not intersect a waste package, the drilling fluid could carry dissolved radionuclides to the surface, assuming that the waste packages have been attacked by ground water and waste materials have begun to be leached out. This is especially the case if the drillers recognize that they had passed through a relatively porous zone (the backfilled repository) and pumped water from that level in order to see if it might be a suitable source of water. This mechanism would also lead to the release of radionuclides directly to the land surface.

note inconsistent with p8-1

A third method by which future drilling could lead to the release of radionuclides from a repository is that the decommissioned borehole, after being filled and perhaps sealed, might still represent a relatively permeable pathway from the repository to adjacent water-bearing zones. This could facilitate the flow of ground water through the repository and the release of radionuclides to an aquifer.

Two drill hole scenarios for the tuff repository are based on an overall drilling rate of three holes per square kilometer, the maximum rate for non-sedimentary formations specified in the draft Standard. The first drill hole scenario considers the possibility of direct contact between the drill and a canister, resulting in the transfer of contaminated material to the surface. The repository area is eight square kilometers, leading to a total of 24 holes over the 10,000 year dose commitment. However, the probability that a given drill hole actually intersects a canister is approximately 0.001 and the average or expected fraction of a canister that would be removed to the surface by such a drill hole is 15 percent (ADL79). The basis for these previous estimates is consistent with the current concepts for the tuff repository. These estimates are based on vertical emplacement holes, which is still the reference DOE design, even though horizontal emplacement is being considered.

The second tuff drill hole scenario considers the possibility of both surface releases and ground water releases without directly contacting a waste canister (near miss scenario). Surface releases involve the bringing to the surface of any and all contaminated water in the pore spaces of the rock at or below the repository level in the immediate "footprint" of the drill hole. In particular, assuming a conservative value of 0.1 square meters for such a footprint, and assuming total mixing of dissolved radionuclides immediately below the level of the waste packages, the fraction of the dissolved repository inventory that will be brought to the surface by one such drill hole will be 0.1 square meters divided by the repository area of 8×10^6 square meters, or a fraction of 1.25×10^{-8}.

Ground water releases from a future borehole are modeled by assuming that such a borehole provides a preferential pathway for percolation of water and that it has been filled with material that does not provide any substantial sorption or retardation capacity. This preferential flow pathway essentially "drains" an area corresponding to a circle with radius five meters, and any radionuclides dissolved in ground water percolating downward within that circle are assumed to be transported from the repository level to the lower aquifer with minimal time delay and no retardation. This is still a very small fraction of the repository inventory, and retardation in the horizontal aquifer leg is still available as a barrier.

8.5.1.5.4 Volcanism. Volcanism also has the potential to release waste from a repository, either by transporting it directly to the surface or by translocating it in an underground volcanic structure, such as a sill or a dike, which may also encounter an aquifer. Earlier calculations indicated that this latter mode of transport is overwhelmingly dominated by the fault release mechanism, in terms of likelihood of occurrence, and that it is roughly similar in terms of consequences (ADL79). Therefore, it has not been included in these assessments, because its contribution is negligible. The release of radionuclides to the surface, however, by way of magma, ash, or gases passing vertically upward through the repository has the potential for significant distribution of radioactivity to the accessible environment, even though its likelihood is small.

For the tuff risk assessment, volcanism is modeled using the basic parameters developed in Volume D (ADL79), except that the probabilities are taken from published USGS results for the tuff repository. The volcanism model is for basaltic events only; the probability of rhyolitic volcanism is sufficiently small that it is outside the area of regulatory concern.

The Agency does not believe that volcanism is likely at any of the sites that are being investigated, but only that it may be the dominant low probability, high consequence event and hence should be included in the calculations to give an adequate perspective on the spectrum of risks. In the case of the release of radionuclides by volcanism, the model has components to deal with waste releases to both the air and land surface.

8.5.1.6 Risk Assessment Results. Using the approach summarized in the previous sections, the Agency estimated the long term risks to future populations from disposal of high-level radioactive wastes in several different types of mined geologic repositories, with recent focus on the tuff repository. The Agency's REPRISK program was used to estimate the long-term population risks over 10,000 years from four models that the Agency believes are representative of most of the sites being considered by DOE as candidates for the first repository.

Figures 8.5-5 and 8.5-6 summarize the results of these various assessments of long-term population risks. Both figures display the same information; however, Figure 8.5-5 shows the population risks on a linear scale, while Figure 8.5-6 introduces the logarithmic scale that will be used for many of the results in this chapter. Figure 8.5-6 also highlights the risk level of 1000 fatal cancers over 10,000 years that has been used as a basis for the containment requirements (Section 191.13) in the final 40 CFR Part 191. Finally, Figure 8.5-6 indicates (where applicable) the corresponding risk estimates developed in the technical basis for the proposed rule; this perspective demonstrates how the analytical revisions made in response to the SAB review have resulted in substantially lower population risk estimates for the various geologic media considered. As Figures 8.5-5 and 8.5-6 show, there is a considerable variation between the risks projected for different types of geologic media; however, all of these reference case estimates are well below the level of protection sought by the containment requirements.

Table 8.5-6 provides more detail on these assessments by displaying the risks from four important release mechanisms for several of the geologic media, with the dominant radionuclides for each release mechanism also displayed.

For the crystalline rocks below the water table (i.e., basalt), where some amount of normal ground water flow is expected, this release mechanism dominates the long-term risks. The radionuclides that flow most readily with the ground water, such as carbon-14 and iodine-129, cause most of this long-term impact for the basalt model. In media where such normal ground water flow is not expected, different release mechanisms dominate. For the bedded salt models, risks from inadvertent human intrusions that bring contaminated water to the surface are the most important contributors to the long-term risks. For the tuff model, fault movement is an important contributor because it can initiate increased ground water flow. Inadvertent intrusion is less likely to bring contamination to the surface than it is for the other models, because the repository horizon would not be saturated with ground water.

Figures 8.5-7 and 8.5-8 display the long-term population risks in another way. For the four models of potential first repository sites, these figures show the complementary cumulative distribution functions of the population risks over 10,000 years, as discussed previously. The figures also indicate corresponding consequences in terms of the ratios of the projected radionuclide releases to the release limits specified in Section 191.13 of the final rule. Finally, the figures depict the probability and consequence limits that must not be exceeded in order to comply with section 191.13.

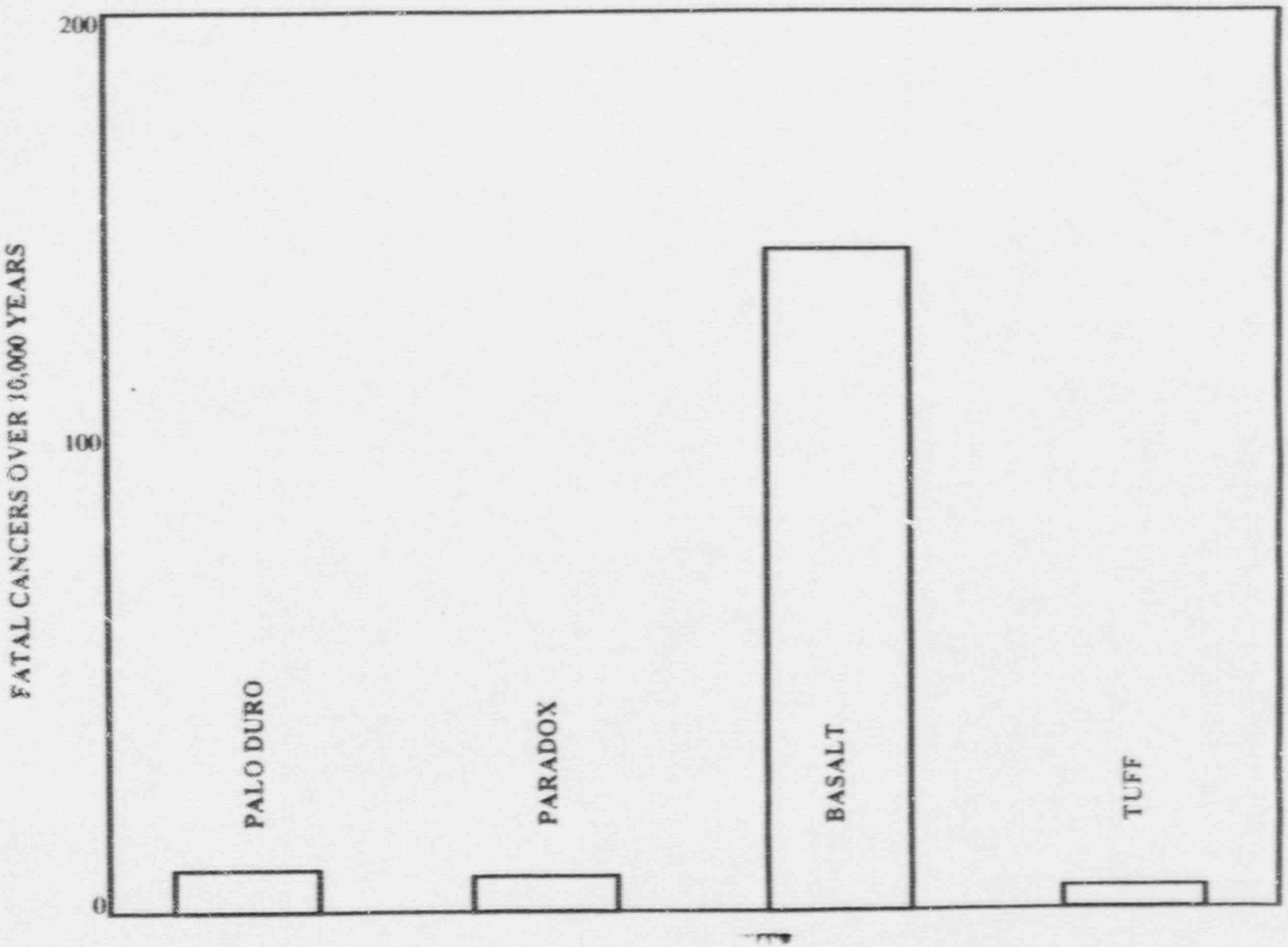

Figure 8.5-5: Population risks from disposal in geologic repositories (reference cases)

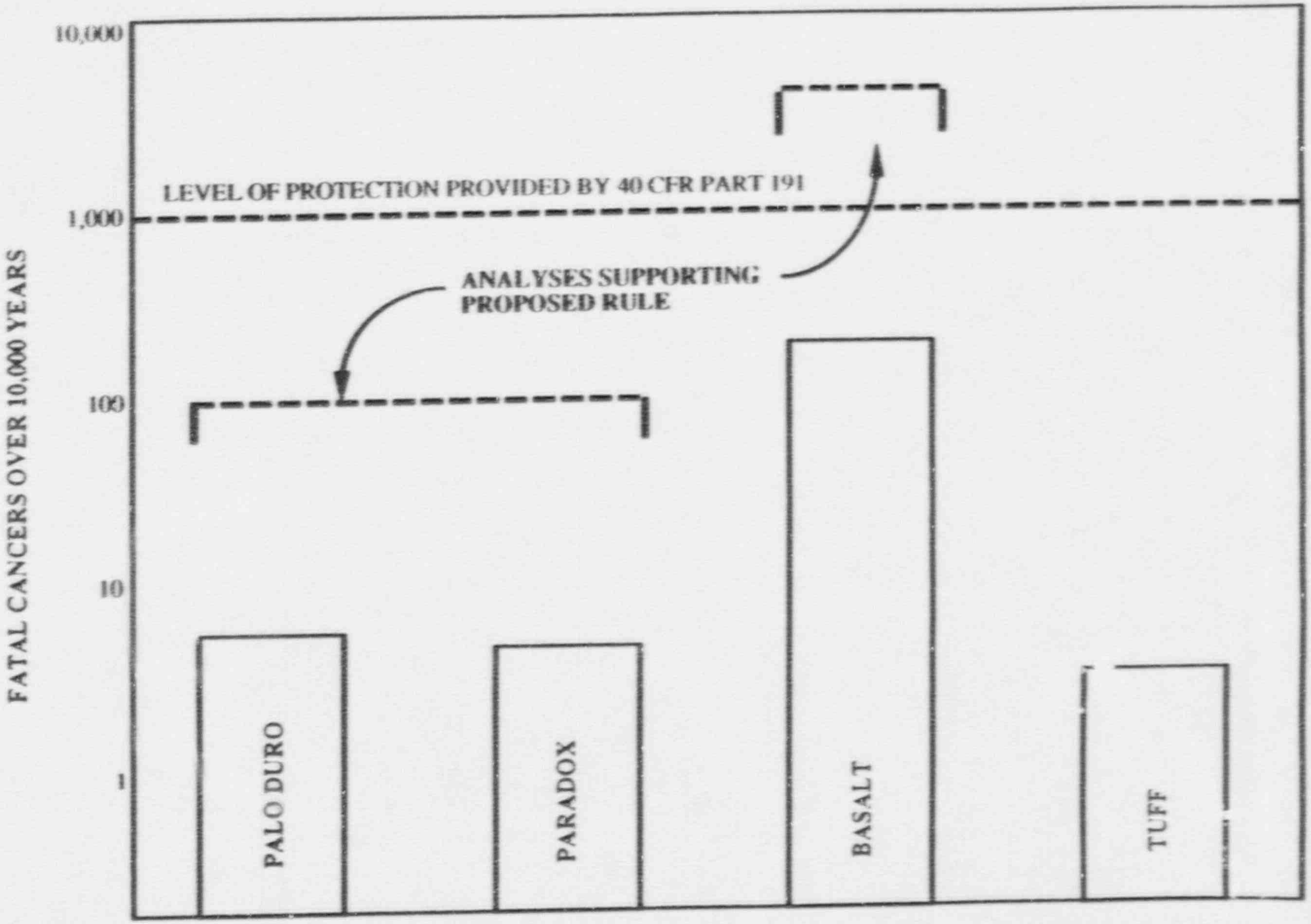

Figure 8.5-6: Population risks from disposal in geologic repositories (logarithmic scale, reference cases)

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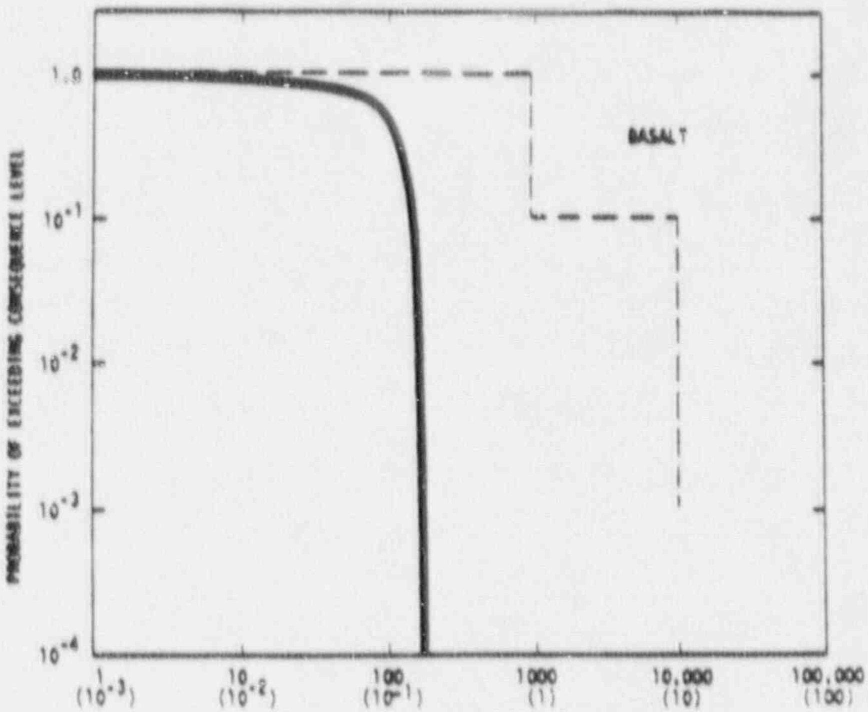
Table 8.5-6: Fatal cancers over 10,000 years by release mechanism and radionuclide

Release Mechanism	Nuclide	Basalt	Palo Duro Bedded Salt	Paradox Bedded Salt	Tuff
Normal Ground Water Flow	C-14	[illegible]	-	-	-
	Tc-99	0.56	-	-	-
	I-129	-	-	-	-
	Np-237	-	-	-	-
	Pu-239	-	-	-	-
	Pu-240	-	-	-	-
	Total	97.0	-	-	-
Drilling (misses canister)	C-14	0.29	0.41	0.41	-
	Tc-99	-	-	0.27	-
	I-29	0.12	0.17	0.17	-
	Pu-240	-	-	-	-
	Am-241	1.61	2.20	0.29	-
	Am-243	0.22	0.27	0.03	-
	Total	2.30	3.16	1.29	-
Drilling (hits canister)	Pu-238	0.02	0.03	0.03	-
	Pu-239	0.58	1.13	1.13	0.15
	Pu-240	0.59	1.16	1.16	0.15
	Am-241	0.51	1.01	1.01	0.13
	Am-243	0.03	0.05	0.05	0.01
	Total	1.73	3.41	3.41	0.44
Fault Movement	C-14	17.1	-	0.11	1.66
	Tc-99	0.1	-	-	0.59
	I-129	7.2	-	0.05	0.75
	Np-237	-	-	-	-
	Pu-239	-	-	-	-
	Pu-240	-	-	-	-
	Total	24.4	-	0.16	3.00

Figure 8.5-7: Complementary cumulative distribution functions of the population risks for disposal in basalt and tuff



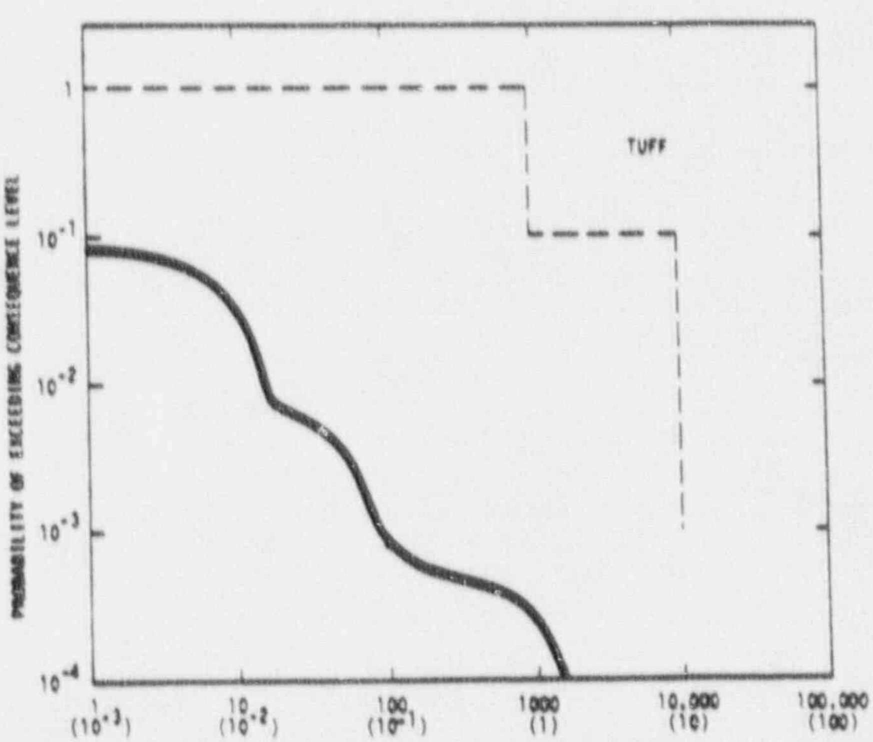
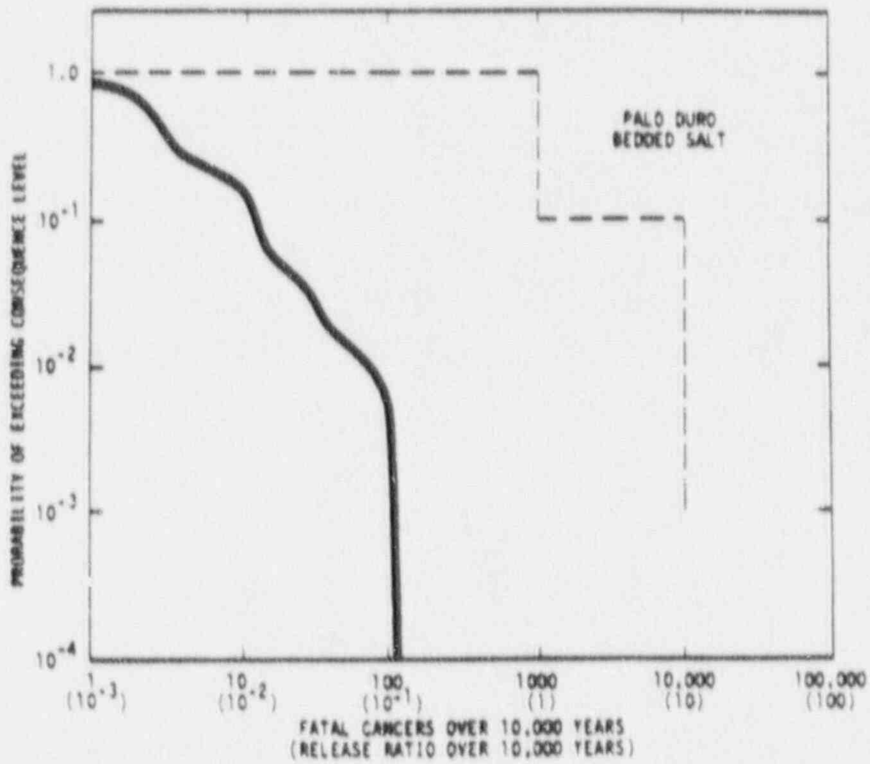


Figure 8.5-8: Complementary cumulative distribution functions of the population risks for disposal in bedded salt



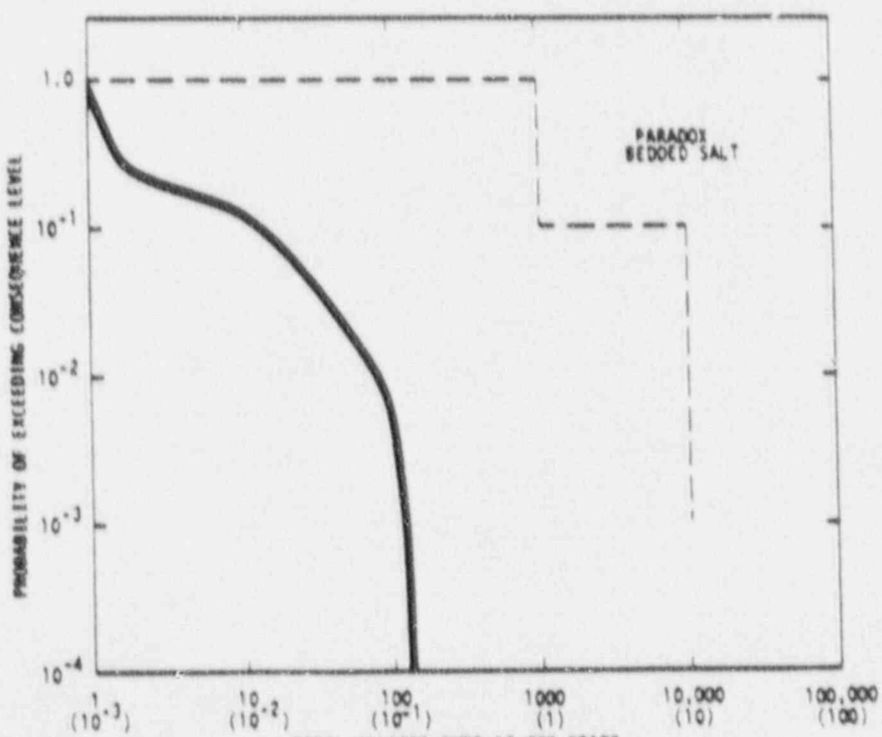


Figure 8.5-9: CCDF for Base Case

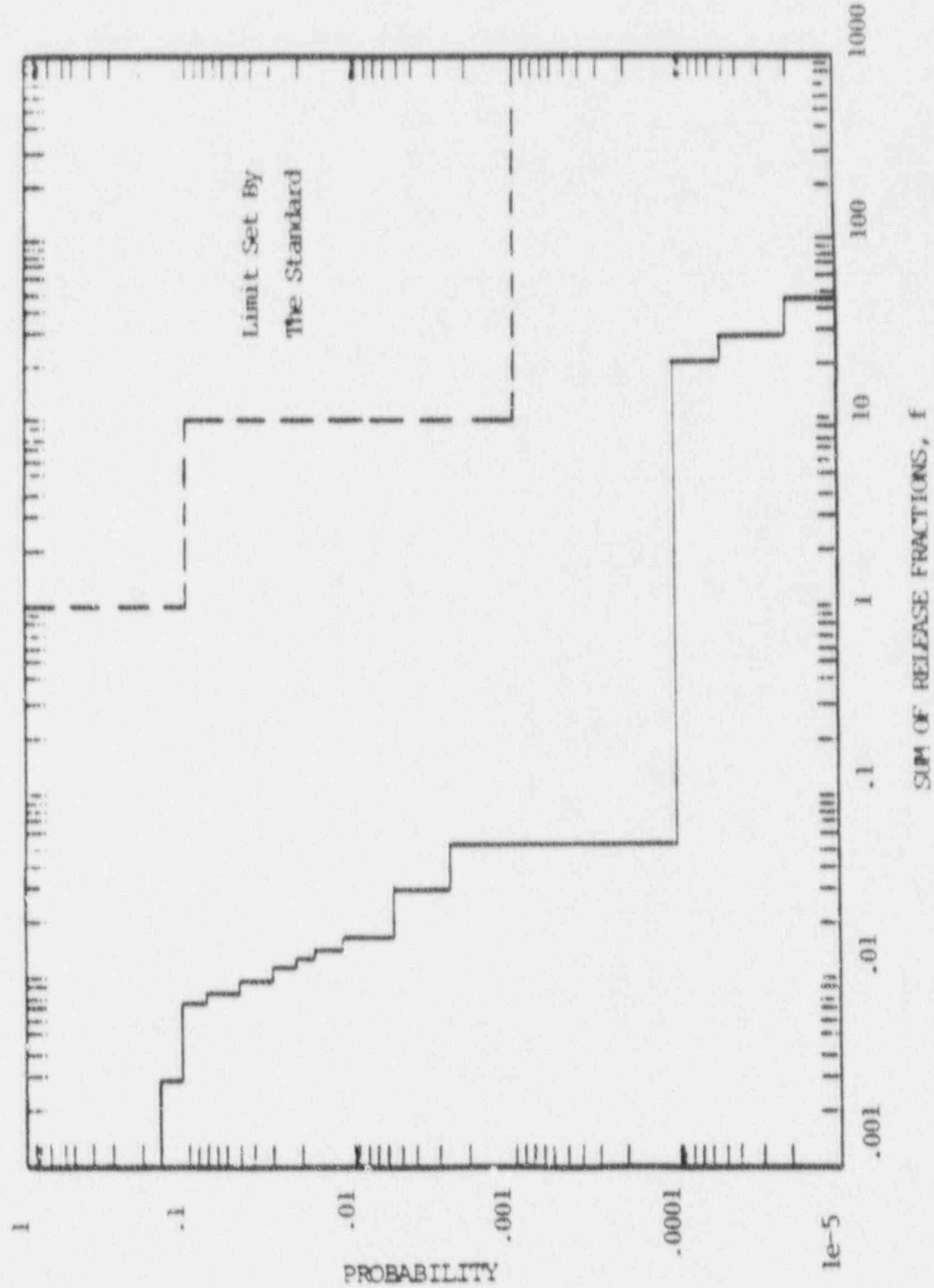
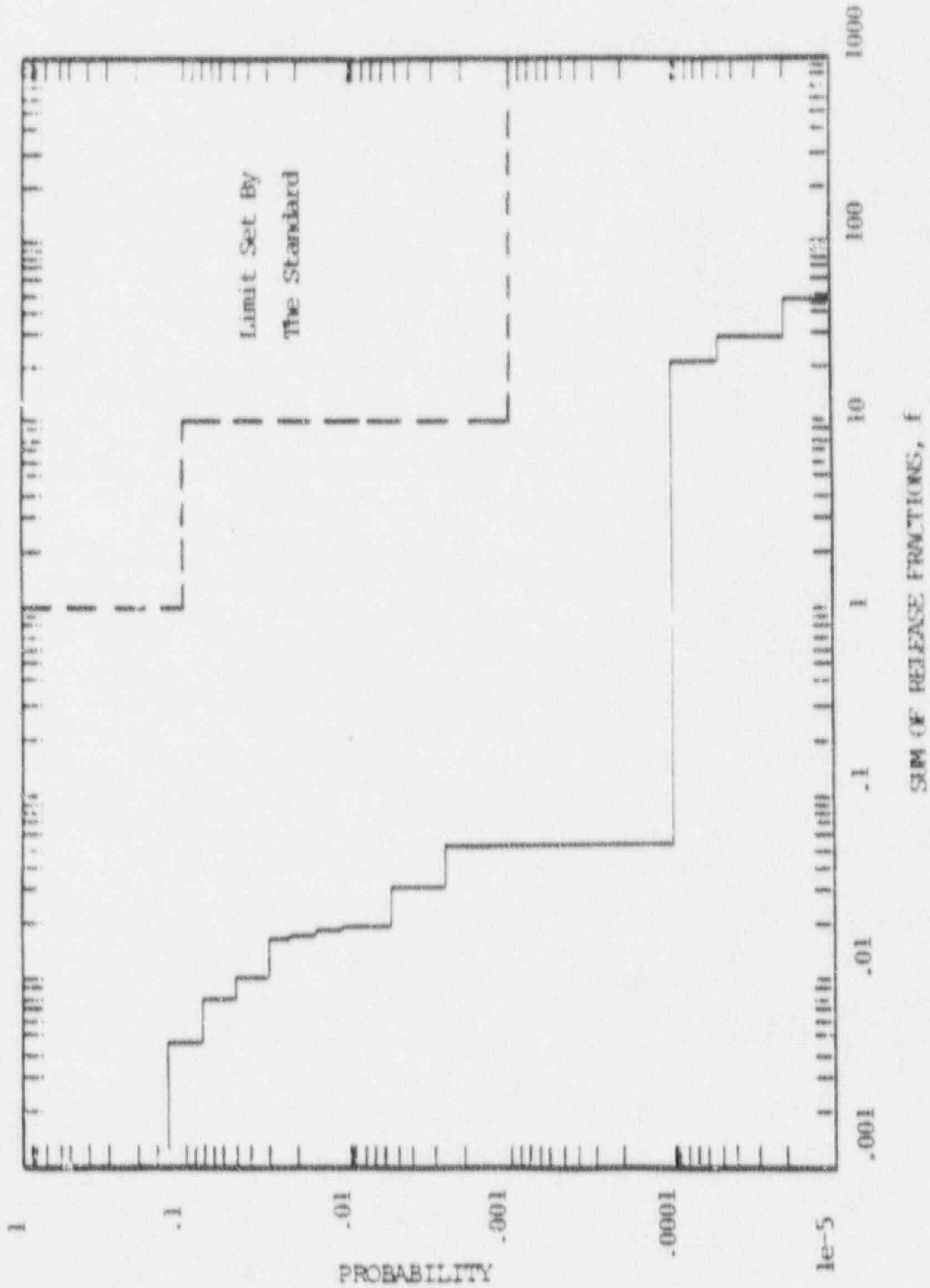


Figure 8.5-10: CCDF for No Retardation in Lower Aquifer



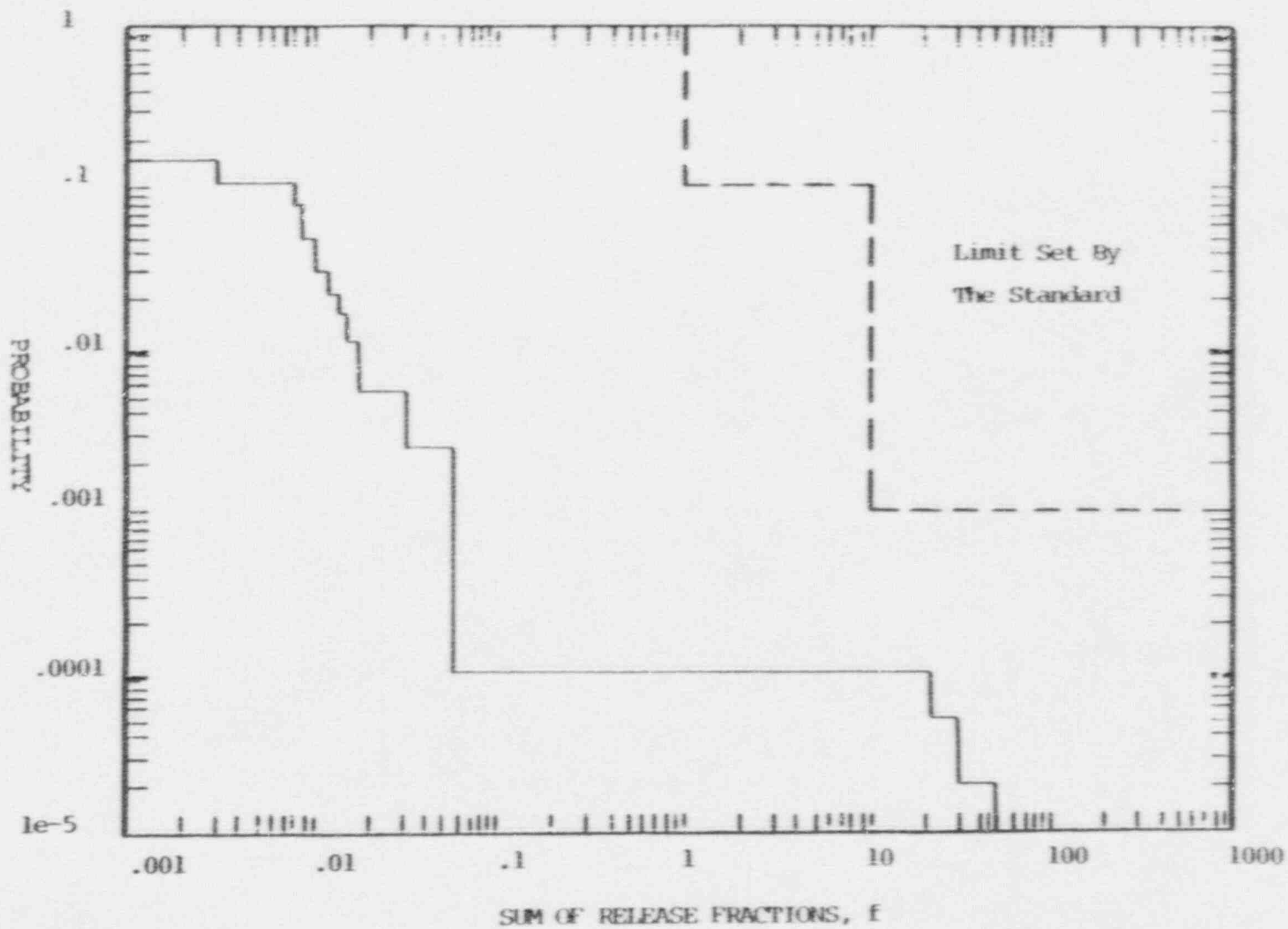


Figure 8.5-11: CCDF for Solubility Increase by a Factor of Ten

Figure 8.5-12: CCDF for Greater Volume of Water for Waste Dissolution

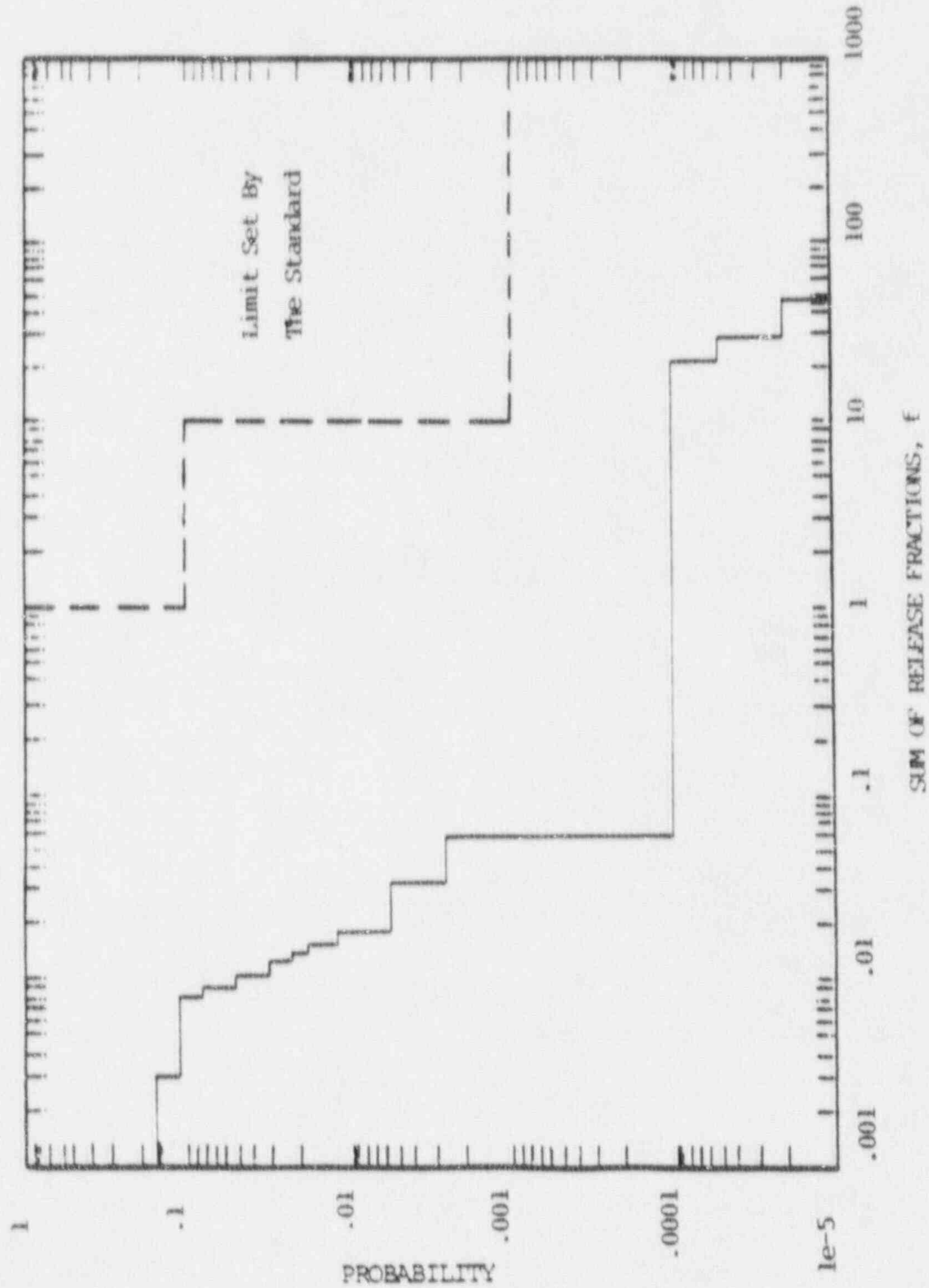


Figure 8.5-13: CCDF for WISP Retardation Values Used in the Aquifer

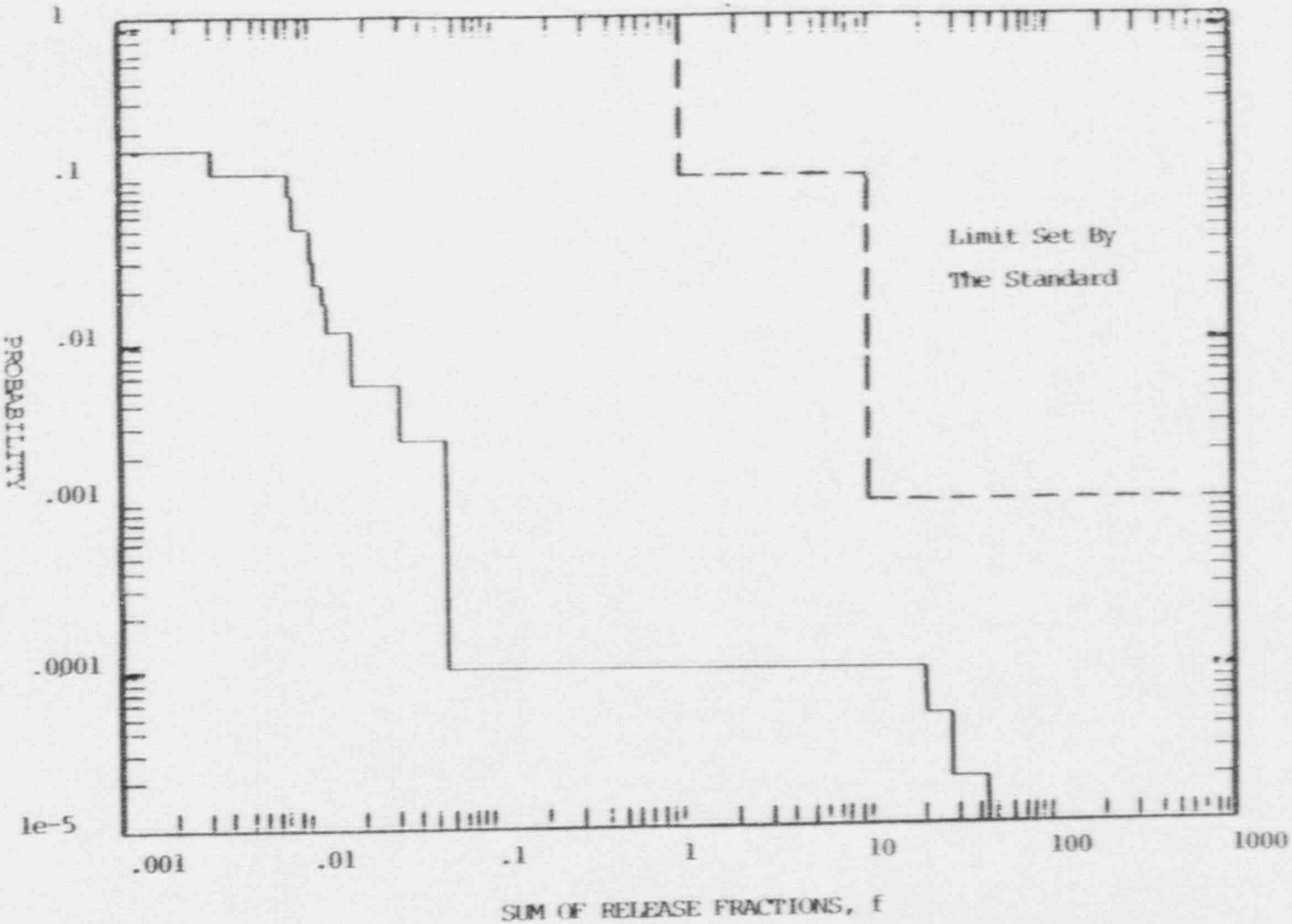
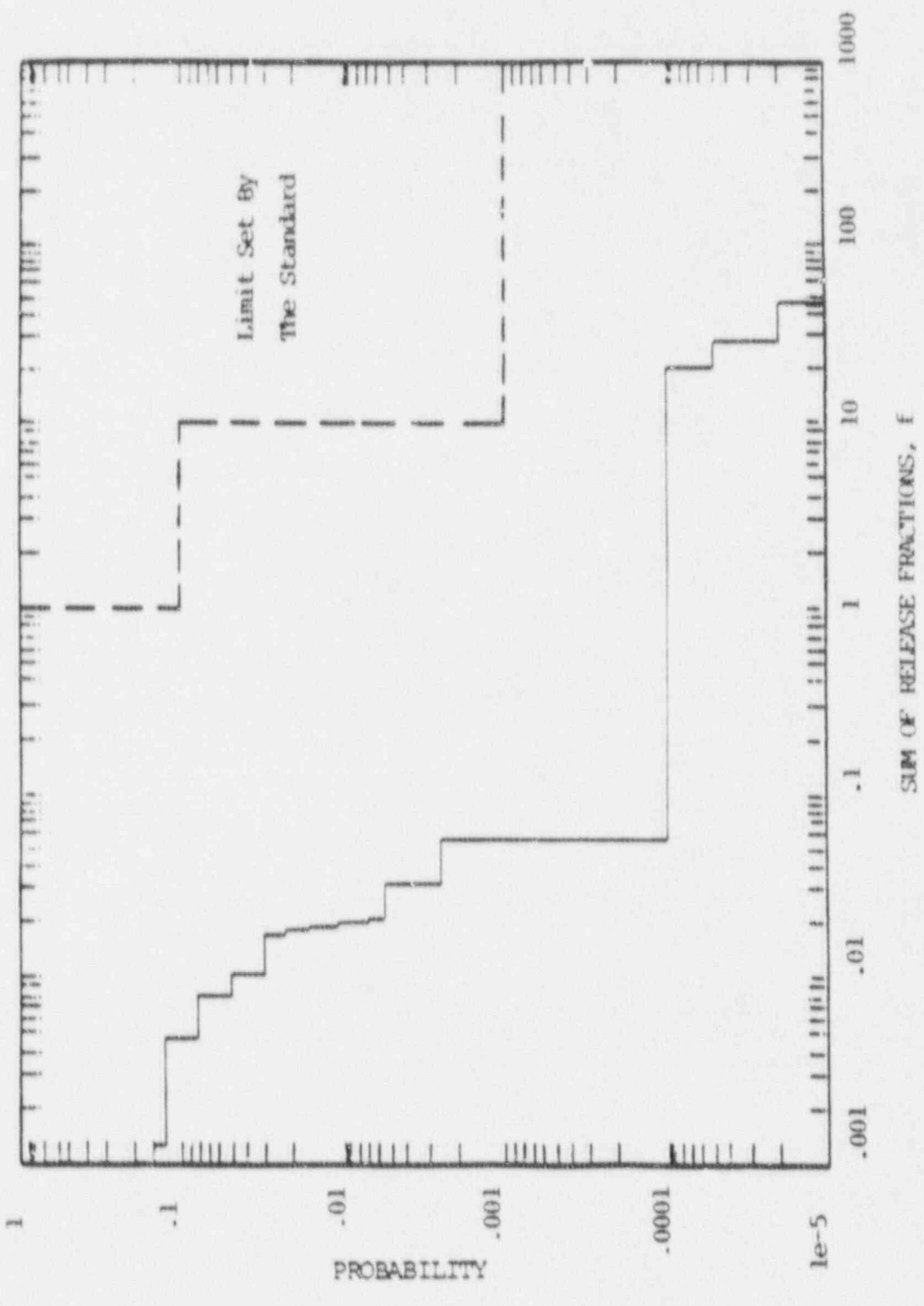


Figure 8.5-14: CCDF for Aquifer Porosity Reduced by a Factor of Ten



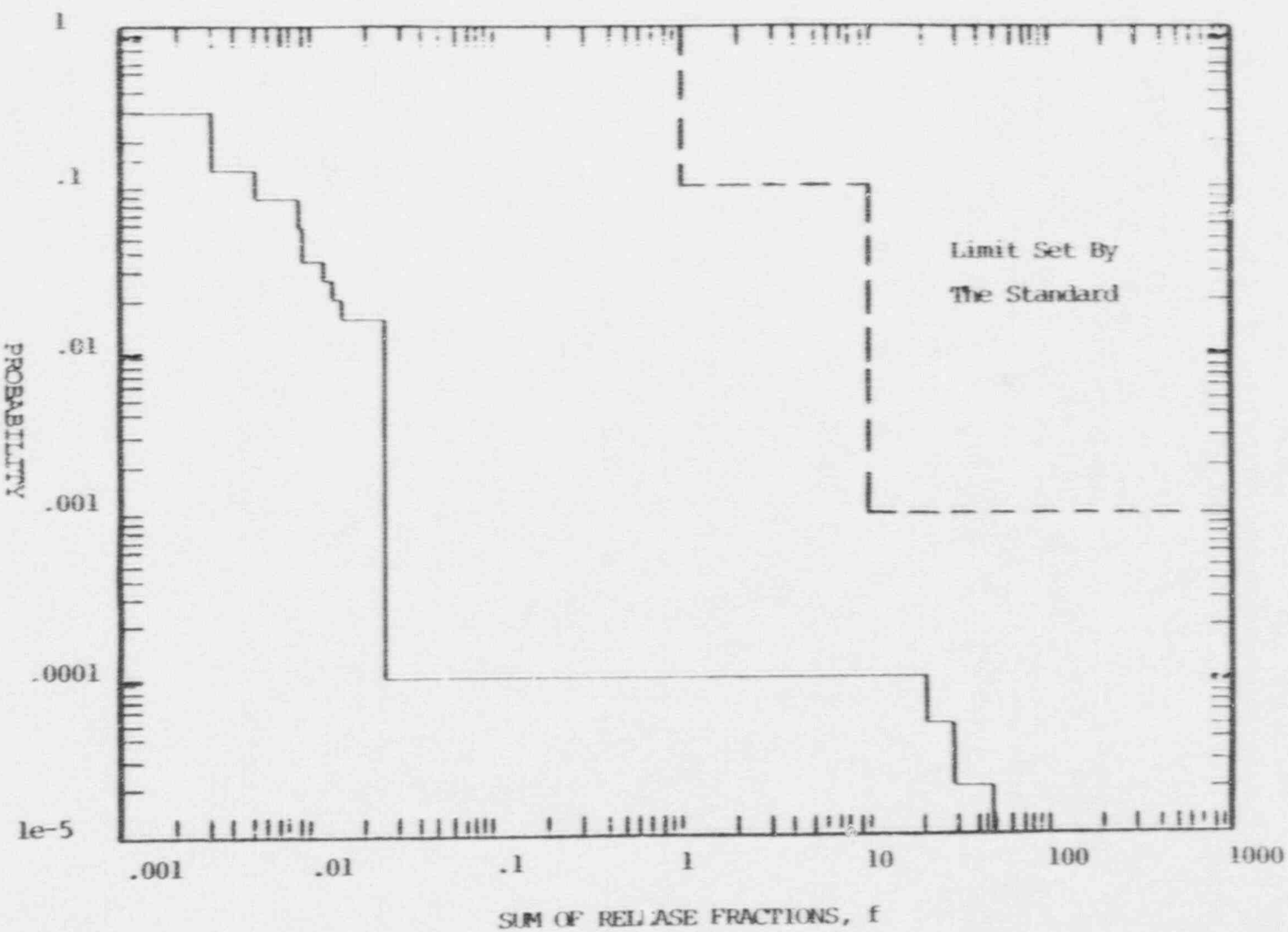


Figure 8.5-15: CCDF for Horizontal Emplacement of Waste Canisters

Figure 8.5-16: CCDF for Larger Borehole Diameter

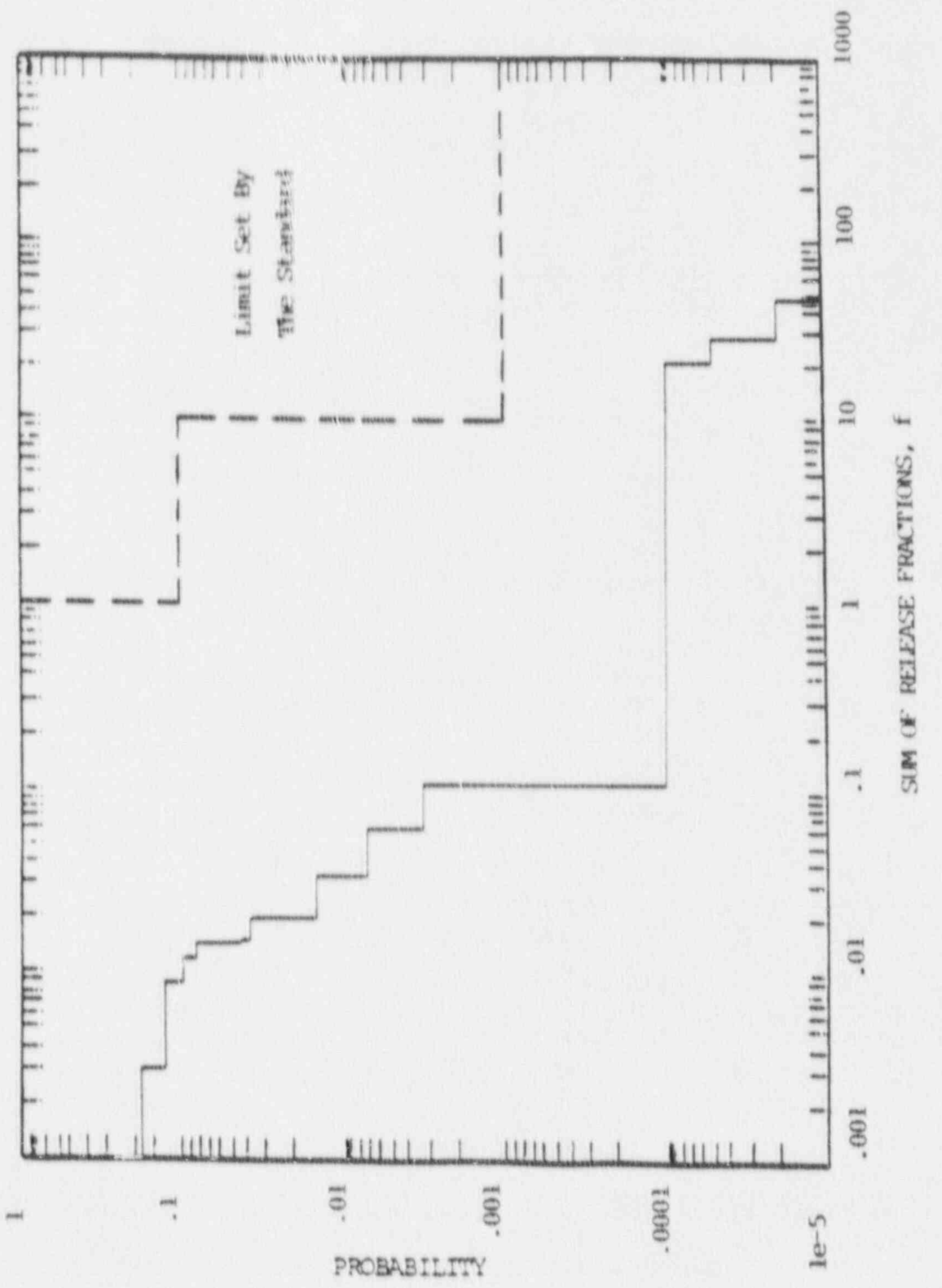


Figure 8.5-17: CCDF for Modified Volcano Probability

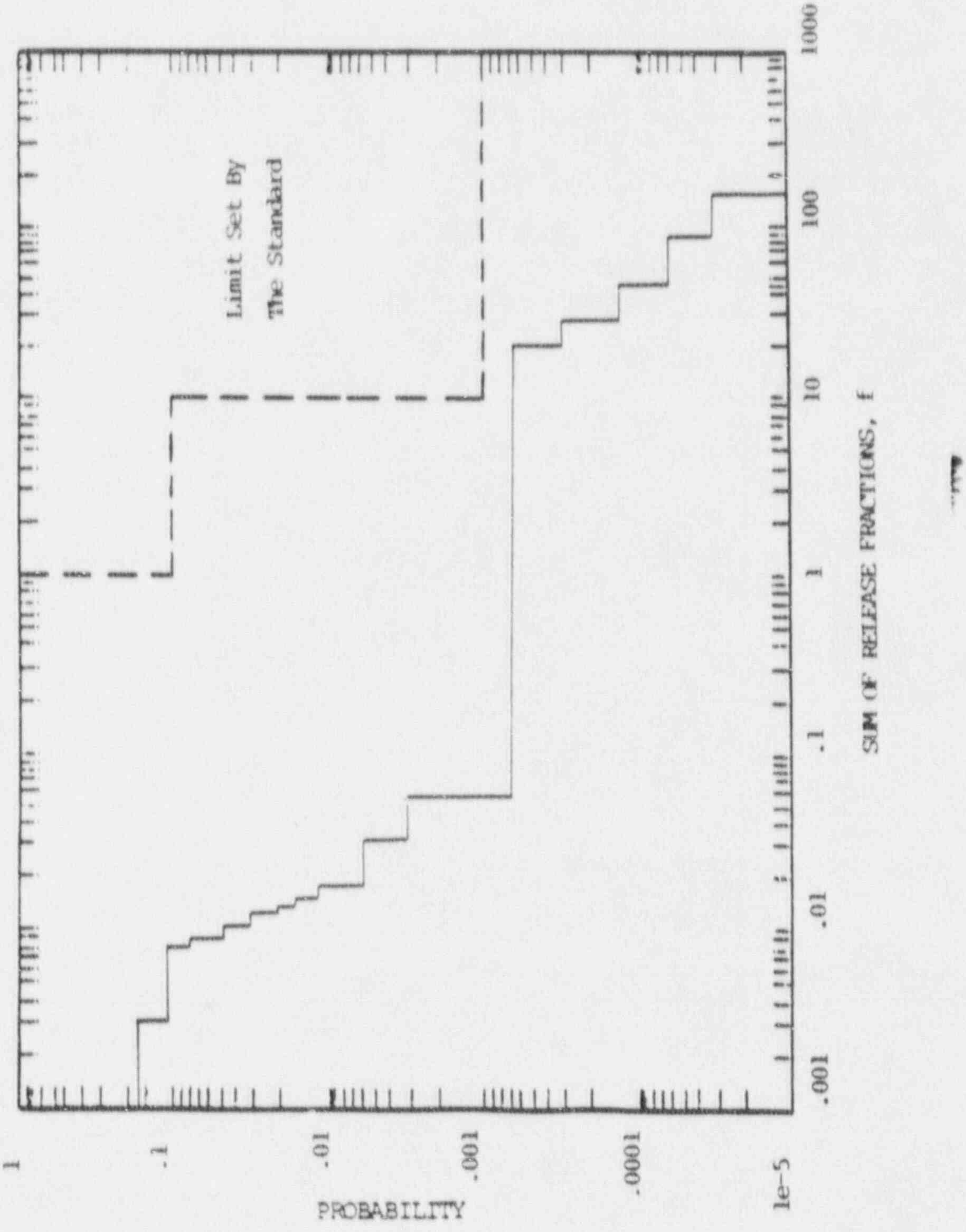


Table 8.5-7: Summary of Events Included in CCDF Base Case

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	7.2×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	2.4×10^{-3}	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.47×10^{-2}
	100 - 500	4×10^{-3}	1.44×10^{-2}
	500 - 1000	5×10^{-3}	1.35×10^{-2}
	1000 - 2000	1×10^{-2}	1.22×10^{-2}
	2000 - 5000	3×10^{-2}	8.64×10^{-3}
	5000 - 10000	4.9×10^{-2}	3.07×10^{-3}
Volcanism	0 - 100	1×10^{-6}	4.2×10^{2}
	100 - 500	4×10^{-6}	1.5×10^{2}
	500 - 1000	5×10^{-6}	8.3×10^{1}
	1000 - 2000	1×10^{-5}	4.5×10^{1}
	2000 - 5000	3×10^{-5}	2.7×10^{1}
	5000 - 10000	5×10^{-5}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-8: Summary of Events Included in CCDF
No Retardation in Lower Aquifer

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	8.3×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	2.4×10^{-3}	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.93×10^{-2}
	100 - 500	4×10^{-3}	1.89×10^{-2}
	500 - 1000	5×10^{-3}	1.80×10^{-2}
	1000 - 2000	1×10^{-2}	1.64×10^{-2}
	2000 - 5000	3×10^{-2}	1.24×10^{-2}
	5000 - 10000	4.9×10^{-2}	4.61×10^{-3}
Volcanism	0 - 100	1×10^{-6}	4.2×10^{2}
	100 - 500	4×10^{-6}	1.5×10^{2}
	500 - 1000	5×10^{-6}	8.3×10^{1}
	1000 - 2000	1×10^{-5}	4.5×10^{1}
	2000 - 5000	3×10^{-5}	2.7×10^{1}
	5000 - 10000	5×10^{-5}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-9: Summary of Events Included in CCDF
Solubility Increased by a Factor of Ten

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	8.3×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	2.4×10^{-3}	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.93×10^{-2}
	100 - 500	4×10^{-3}	1.89×10^{-2}
	500 - 1000	5×10^{-3}	1.80×10^{-2}
	1000 - 2000	1×10^{-2}	1.64×10^{-2}
	2000 - 5000	3×10^{-2}	1.24×10^{-2}
	5000 - 10000	4.9×10^{-2}	4.61×10^{-3}
Volcanism	0 - 100	1×10^{-6}	4.2×10^{2}
	100 - 500	4×10^{-6}	1.5×10^{2}
	500 - 1000	5×10^{-6}	8.3×10^{1}
	1000 - 2000	1×10^{-5}	4.5×10^{1}
	2000 - 5000	3×10^{-5}	2.7×10^{1}
	5000 - 10000	5×10^{-5}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-10: Summary of Events Included in CCDF
Greater Volume of Water for Dissolution

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	7.4×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	2.4 x [illegible]	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.47×10^{-2}
	100 - 500	4×10^{-3}	1.44×10^{-2}
	500 - 1000	5×10^{-3}	1.35×10^{-2}
	1000 - 2000	1×10^{-2}	1.22×10^{-2}
	2000 - 5000	3×10^{-2}	8.64×10^{-3}
	5000 - 10000	4.9×10^{-2}	3.07×10^{-3}
Volcanism	0 - 100	1×10^{-6}	4.2×10^{2}
	100 - 500	4×10^{-6}	1.5×10^{2}
	500 - 1000	5×10^{-6}	8.3×10^{1}
	1000 - 2000	1×10^{-5}	4.5×10^{1}
	2000 - 5000	3×10^{-5}	2.7×10^{1}
	5000 - 10000	5×10^{-5}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-11: Summary of Events Included in CCDF
WISP Retardation Values Used in Aquifer

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	7.0×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	2.4×10^{-3}	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.27×10^{-2}
	100 - 500	4×10^{-3}	1.24×10^{-2}
	500 - 1000	5×10^{-3}	1.18×10^{-2}
	1000 - 2000	1×10^{-2}	1.08×10^{-2}
	2000 - 5000	3×10^{-2}	8.23×10^{-3}
	5000 - 10000	4.9×10^{-2}	3.04×10^{-3}
Volcanism	0 - 100	1×10^{-6}	4.2×10^{2}
	100 - 500	4×10^{-6}	1.5×10^{2}
	500 - 1000	5×10^{-6}	8.3×10^{1}
	1000 - 2000	1×10^{-5}	4.5×10^{1}
	2000 - 5000	3×10^{-5}	2.7×10^{1}
	5000 - 10000	5×10^{-5}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-12: Summary of Events Included in CCDF Aquifer Porosity Reduced by a Factor of Ten

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	9.1×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	$2.4 \times 10^{[illegible]}$	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.99×10^{-2}
	100 - 500	4×10^{-3}	1.94×10^{-2}
	500 - 1000	5×10^{-3}	1.84×10^{-2}
	1000 - 2000	1×10^{-2}	1.67×10^{-2}
	2000 - 5000	3×10^{-2}	1.26×10^{-2}
	5000 - 10000	4.9×10^{-2}	4.61×10^{-3}
Volcanism	0 - 100	1×10^{-6}	4.2×10^{2}
	100 - 500	4×10^{-6}	1.5×10^{2}
	500 - 1000	5×10^{-6}	8.3×10^{1}
	1000 - 2000	1×10^{-5}	4.5×10^{1}
	2000 - 5000	3×10^{-5}	2.7×10^{1}
	5000 - 10000	5×10^{-5}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-13: Summary of Events Included in CCDF
Drill Hit Modified: Horizontal Emplacement

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	7.2×10^{-7}
Drill/Hit	0 - 100	0	.025
	100 - 500	1.6×10^{-2}	.009
	500 - 1000	2.0×10^{-2}	.005
	1000 - 2000	3.9×10^{-2}	.003
	2000 - 5000	1.1×10^{-1}	.002
	5000 - 10000	1.8×10^{-1}	.001
Faulting	0 - 100	1×10^{-3}	1.47×10^{-2}
	100 - 500	4×10^{-3}	1.44×10^{-2}
	500 - 1000	5×10^{-3}	1.35×10^{-2}
	1000 - 2000	1×10^{-2}	1.22×10^{-2}
	2000 - 5000	3×10^{-2}	8.64×10^{-3}
	5000 - 10000	4.9×10^{-2}	3.07×10^{-3}
Volcanism	0 - 100	6.6×10^{-6}	4.2×10^{2}
	100 - 500	2.6×10^{-5}	1.5×10^{2}
	500 - 1000	3.3×10^{-5}	8.3×10^{1}
	1000 - 2000	6.6×10^{-5}	4.5×10^{1}
	2000 - 5000	2.0×10^{-4}	2.7×10^{1}
	5000 - 10000	3.3×10^{-4}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 8.5-14: Summary of Events Included In CCDF
Larger Borehole Diameter

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	1.1×10^{-6}
Drill/Hit	0 - 100	0	.295
	100 - 500	3.0×10^{-3}	.102
	500 - 1000	3.8×10^{-3}	.058
	1000 - 2000	7.6×10^{-3}	.032
	2000 - 5000	2.3×10^{-2}	.019
	5000 - 10000	3.8×10^{-2}	.014
Faulting	0 - 100	1×10^{-3}	1.47×10^{-2}
	100 - 500	4×10^{-3}	1.44×10^{-2}
	500 - 1000	5×10^{-3}	1.35×10^{-2}
	1000 - 2000	1×10^{-2}	1.22×10^{-2}
	2000 - 5000	3×10^{-2}	8.64×10^{-3}
	5000 - 10000	4.9×10^{-2}	3.07×10^{-3}
Volcanism	0 - 100	6.6×10^{-6}	4.2×10^{2}
	100 - 500	2.6×10^{-5}	1.5×10^{2}
	500 - 1000	3.3×10^{-5}	8.3×10^{1}
	1000 - 2000	6.6×10^{-5}	4.5×10^{1}
	2000 - 5000	2.0×10^{-4}	2.7×10^{1}
	5000 - 10000	3.3×10^{-4}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

Table 3.5-15: Summary of Events Included in CCDF
Volcano Modified: Highest Sandia Probability

Event	Time Period (Years)	Probability, P	Consequence, C
Drill/No Hit (18 Holes)	100 - 10,000	1.0	7.2×10^{-7}
Drill/Hit	0 - 100	0	.159
	100 - 500	2.4×10^{-3}	.055
	500 - 1000	3.0×10^{-3}	.031
	1000 - 2000	6.1×10^{-3}	.017
	2000 - 5000	1.8×10^{-2}	.010
	5000 - 10000	3.0×10^{-2}	.008
Faulting	0 - 100	1×10^{-3}	1.47×10^{-2}
	100 - 500	4×10^{-3}	1.44×10^{-2}
	500 - 1000	5×10^{-3}	1.35×10^{-2}
	1000 - 2000	1×10^{-2}	1.22×10^{-2}
	2000 - 5000	3×10^{-2}	8.64×10^{-3}
	5000 - 10000	4.9×10^{-2}	3.07×10^{-3}
Volcanism	0 - 100	6.6×10^{-6}	4.2×10^{2}
	100 - 500	2.6×10^{-5}	1.5×10^{2}
	500 - 1000	3.3×10^{-5}	8.3×10^{1}
	1000 - 2000	6.6×10^{-5}	4.5×10^{1}
	2000 - 5000	2.0×10^{-4}	2.7×10^{1}
	5000 - 10000	3.3×10^{-4}	2.0×10^{1}
Normal Groundwater Flow	0 - 10000	1.0	0

With the recent focus of effort on examining potential risks from the tuff repository, a more detailed assessment of releases is presented here. Figures 8.5-9 through 8.5-17, and the corresponding Tables 8.5-7 through 8.5-15, contain a detailed risk analysis of potential release events and a sensitivity analysis based on variations in select parameters. (The calculations supporting these data exclude consideration of gaseous carbon-14 releases. The issue of carbon-14 release remains controversial and unresolved.)

Figures 8.5-9 through 8.5-17 display the probability of event occurrence in tuff plotted against the sum of release fractions in a log-log format. The various cases are numbered one through nine. Case one represents the "base case" and its results are given in Table 8.5-7 and Figure 8.5-9. The dashed line in each of these "PC" diagrams depicts the probability and consequence limits set by the Standard. The eight alternative cases examine variations in parameter values such as:

- No retardation in the lower aquifer
- Nuclide solubilities raised by a factor of ten
- Increase in water volume
- Porosity reductions, etc.

Additional sensitivity analyses are treated in greater detail in Section 8.5.1.7.

A general conclusion drawn from these diagrams is that the tuff repository appears to be able to meet the performance requirements of the Standard. The single factor that contributes most significantly to this favorable behavior is the small amount of water available in the unsaturated tuff system for direct contact with the waste package as it moves downward through the repository horizon. This small quantity of water limits the amount of dissolved radionuclides to their corresponding solubility limits.

8.5.1.7 Uncertainties in the Risk Assessment. The results of the risk assessments discussed in Section 8.5.1.6 encompass many uncertainties, which are due to a number of factors such as the following:

- The long time frame over which predictions are needed.

- The simplified nature of the models in comparison with the real physical situation.

- The generic nature of the modeling, i.e., no detailed site-specific data for all but the tuff repository.

The purpose of the risk assessment is to make rough approximations of the capabilities of geologic disposal of radioactive waste. Therefore, despite these uncertainties, the Agency believes that the estimates generated herein provide an adequate technical basis for the associated regulations.

In order to lend perspective to the uncertainties in these calculations, the Agency has proceeded as follows: First, in estimating parameters or in choosing models to represent various processes, an attempt has been made to consistently overestimate factors that contribute to risks from the repository. This is the same philosophy that was adopted in risk assessments for the proposed rule, although the degree of over-estimation has been reduced in response to many of the recommendations of the Agency's Science Advisory Board. The parameters required to model the tuff repository were largely derived from the range of values found in published literature. Again, a conservative approach was taken in the selection of many parameters, but sufficient site specific work has been done by previous studies to provide a high degree of certainty to some parameters.

Second, extensive use has been made of sensitivity analyses in order to understand how much the results of the assessment change with variations in certain model components or parameters. For parameters that are particularly important in determining the final risk results, special attention has been devoted to choosing appropriate values. For example, published values for lower aquifer hydraulic conductivity in the tuff model range from 700 to 0.007 meters per year. A moderately conservative value of 75 meters per year was chosen as a realistic approximation for the specific hydraulic characteristics of the geologic unit in question.

Third, in cases where it has been difficult to model the characteristics of a site or a process on a generic basis, several choices of parameters have been made to understand the range of potential risk results.

It is important to distinguish between the type of uncertainty included in the generic analysis reported here and the uncertainties that would remain with real sites when they are characterized and modeled in connection with the decision on where to put a repository. Many of the uncertainties included here might better be characterized as variabilities. At a real site there might be a wide variation in the property in question. The attempt in these risk assessments is to include such variations, which correspond then to an uncertainty in the final results as to how well they characterize the performance of the repository. A real site will include the additional uncertainties associated with data collection, site complexity, and difference of opinion about a specific site's characteristics. Such uncertainties are not within the scope of the work reported here.

Alternate cases (see Section 8.5.1.6) were used to model ground water flow for several release scenarios, including faulting and borehole near miss, for the tuff repository. The two values identified as having the greatest uncertainty are solubility and retardation. Previously used values for these parameters vary over orders of magnitude. An alternate case for the fracture flow porosity was also used because of a possible non-conservative assumption.

An area of relatively high uncertainty is the effect of retardation on nuclide migration. Two alternate cases were used to examine uncertainties in retardation. The first alternate case uses the values from the WISP report (NRC, 83) for retardation in the horizontal leg. It is also possible that the rock surface characteristics of the fracture flow path cause little or no retardation. The second alternate retardation scenario sets all retardations to 1, thereby neglecting its effect.

Two alternate cases were also modeled for solubility. The first case assumed that each canister was exposed to 5 liters of water per year, resulting in a yearly dissolution volume (of water) of 175 cubic meters. The second case raised the solubility limit for each nuclide by a factor of ten. Neither alternate case had an effect on ground water releases, because all nuclides with low enough retardation to travel the entire horizontal flow path in 10,000 years were leach rate limited. The alternate solubility cases increased the borehole near miss surface release by increasing waste concentration in the unsaturated region.

The remainder of this section assess several of the uncertainties in this risk assessment by examining the sensitivity of the results to many of the assumptions used in the various repository models.

Sensitivity to Waste Package Parameters. The analyses summarized in this section investigated how the population risks vary with different assumptions about waste package lifetime and waste form release rate. Waste package lifetimes of zero and 1000 years were considered for population risks, and lifetimes of 300 to 1000 years were considered for the individual exposures. The waste form release rate was varied from one part in 1000 (10^{-3}) per year to one part in 1,000,000 ((10^{-6}) per year.

Figure 8.5-18 displays the results of these analyses for the three different media (basalt, bedded salt, and tuff) being considered for the first repository. As this figure shows, variations over the range of waste package lifetimes considered have relatively little effect on the population risks--except for the bedded salt models, where a very short canister lifetime coupled with a rapid release rate, allows some of the short-lived fission products to be brought to the land surface by inadvertent intrusion. On the other hand, variations in waste form release rate can cause the projected population risks within 10,000 years to vary by up to three orders of magnitude, where releases through ground water flow pathways dominate the analysis.

Figure 8.5-19 shows how the potential individual exposures for normal ground water flow from the basalt model vary for four different assumptions about waste form release rate and two different waste package lifetimes. Increases in package

Figure 8.5-18: The effect of canister life and waste form leach rate on population risks for three potentially suitable repository media

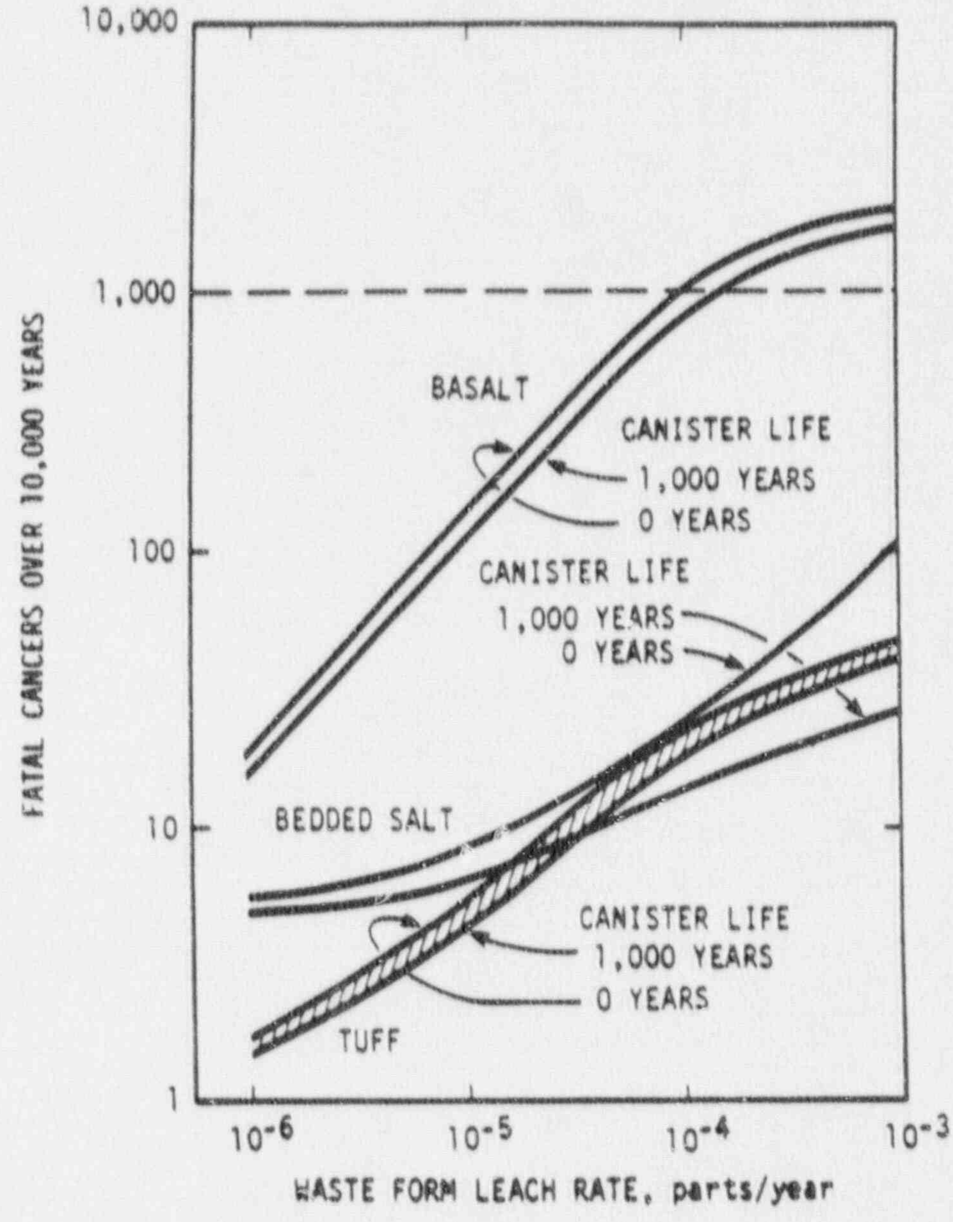
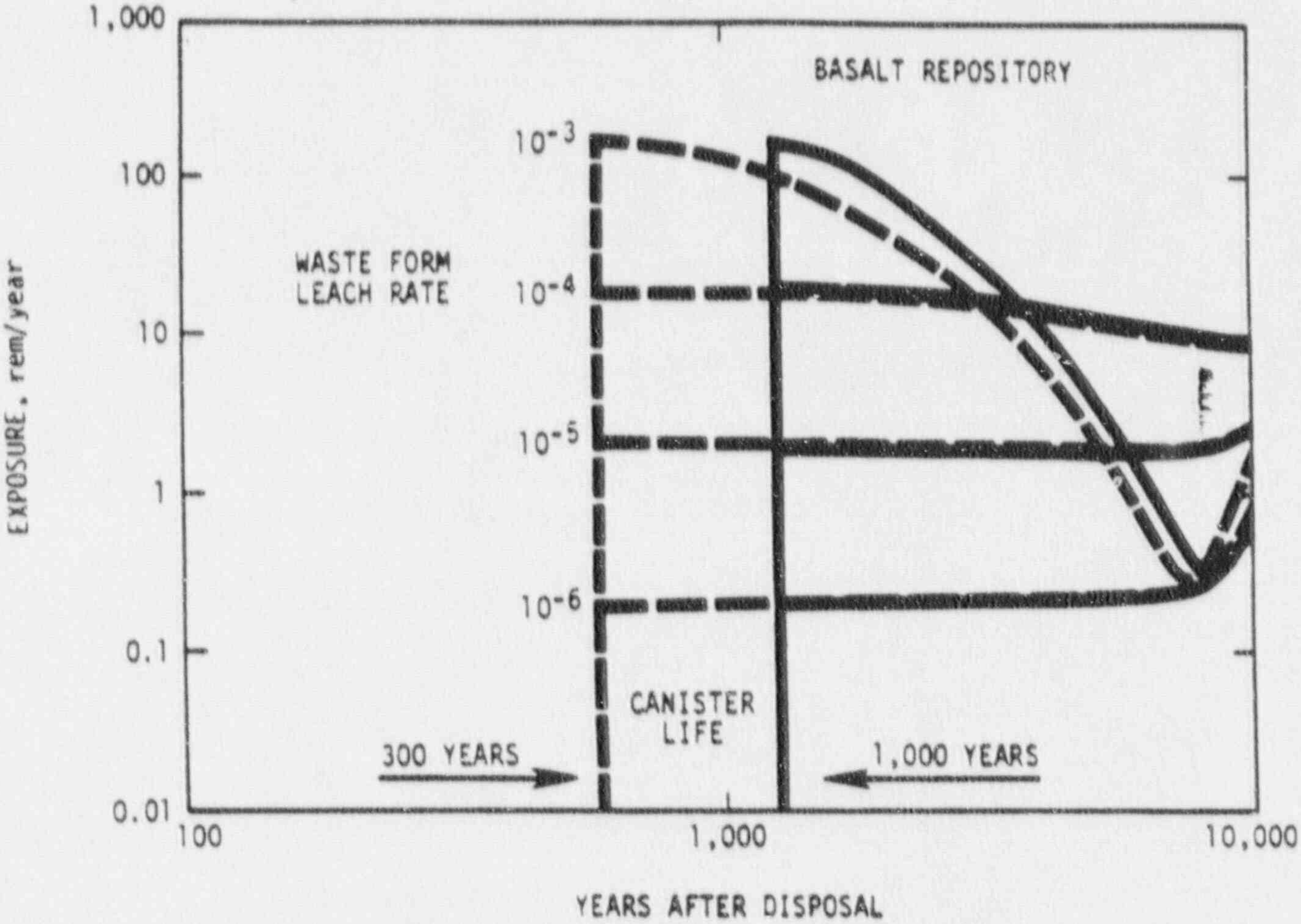


Figure 8.5-19: Effect of canister life and waste form leach rate on radiation exposures from drinking ground water at 2 kilometers from repository



lifetime primarily act to delay the time at which exposures begin to appear. Increases in waste form release rate cause higher initial exposures, which then may decrease with time for the larger release rates as the peak concentration passes the 2-km point of interest.

Sensitivity to Geochemical Parameters. The analyses described in this section indicate how the population risks over 10,000 years for the basalt and bedded salt models vary with different assumptions about the geochemical parameters. For one set of analyses, the geochemical retardation factors were varied to consider the two alternative sets indicated in Table 8.5-15, which are primarily taken from a NAS report (NAS83). For another set of analyses, the solubility limits used for those elements were varied up and down by two orders of magnitude from those used in the base cases. Figure 8.5-20 shows the results of these variations. The variations had significant effects on the results in only a few cases. Increasing the solubility limits substantially increased the risks for the bedded salt model. Lowering the solubility limits or increasing geochemical retardation generally did not substantially reduce the risks projected for the reference cases.

Sensitivity to Americium Solubility. One important change from the analyses supporting the proposed rule concerned the solubility limit assumed for the americium radioisotopes. Because of uncertainty regarding this parameter, the Agency chose a very conservative upper bound of 50 parts per million (ppm) for its earlier analyses. This was one of the parameters that the SAB suggested be reexamined, and more recent information supports the much lower choices described in Section 8.5.1.2 (which vary for the different geologic media). The analyses described in this section indicate the importance of this change. The base case models for basalt and bedded salt were reanalyzed; the only change was to return the americium solubility limits to the 50-ppm value used in the earlier analyses. Figure 8.5-21 displays the effects on the projected population risks over 10,000 years. As can be seen by comparing these results with the comparison of the earlier and later technical analyses included in Figure 8.5-6, the changes in this one parameter account for a significant part of the differences in the two sets of analyses, particularly for the bedded salt models.

Sensitivity to Ground Water Travel Time. The effect of longer ground water travel times on population risks was examined by setting the boundary to the accessible environmental at 5 and 10 km in addition to the value of 2 km that has been used in most of these analyses. The tuff model assumed a distance of five kilometer to the accessible environment for all calculations supporting the results of Section 8.5.1.6. Since this will only affect the results for cases where a significant portion of the risk comes from release modes involving ground water flow, this section only considers the basalt model. Figure 8.5-22 demonstrates how the risk from this model changes due to the longer distance and travel time. note

Table 8.5-15: Alternative geochemical parameters considered in risk assessment

Parameter	Basalt	Bedded Salt	Tuff[b]
Lower Retardation Factors[a]			
carbon	1	1	1
strontium	50	1	1
zirconium	500	300	1
technetium	1	1	1
tin	100	10	1
iodine	1	1	1
cesium	100	1	1
uranium	20	10	1
neptunium	10	10	1
plutonium	100	10	1
americium	60	300	1
Higher Retardation Factors[a]			
carbon	1	1	1
strontium	2000	100	200
zirconium	5000	1000	5000
technetium	100	20	5
tin	5000	1000	1000
iodine	50	1	1
cesium	10000	2000	500
uranium	1000	60	200
neptunium	500	300	100
plutonium	5000	10000	200
americium	10000	3000	1000

[a]For use with all isotopes of radionuclides listed.
[b]Assumes fracture flow

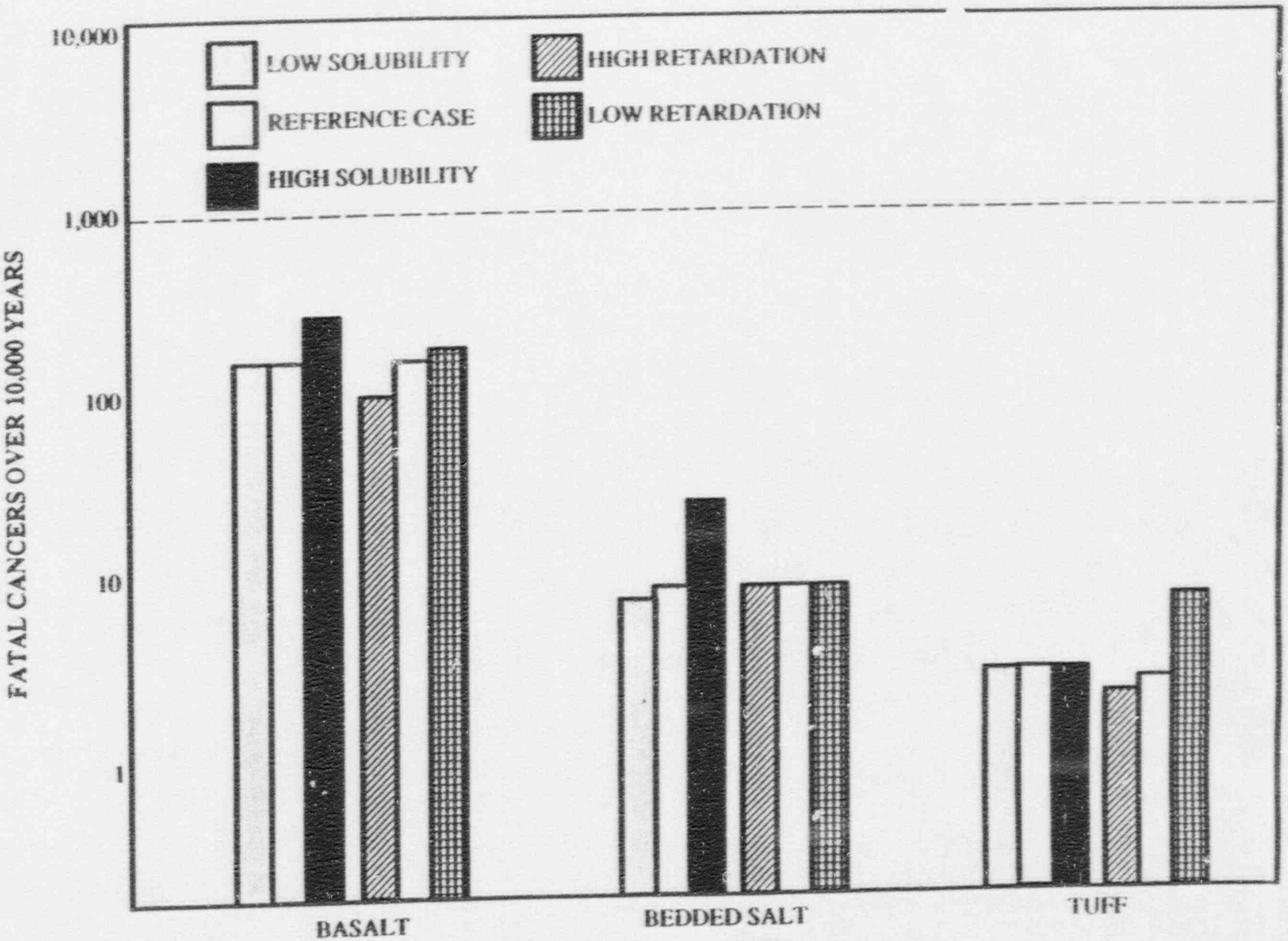


Figure 8.5-20: Effect of geochemical parameters on population risks for different geologic media

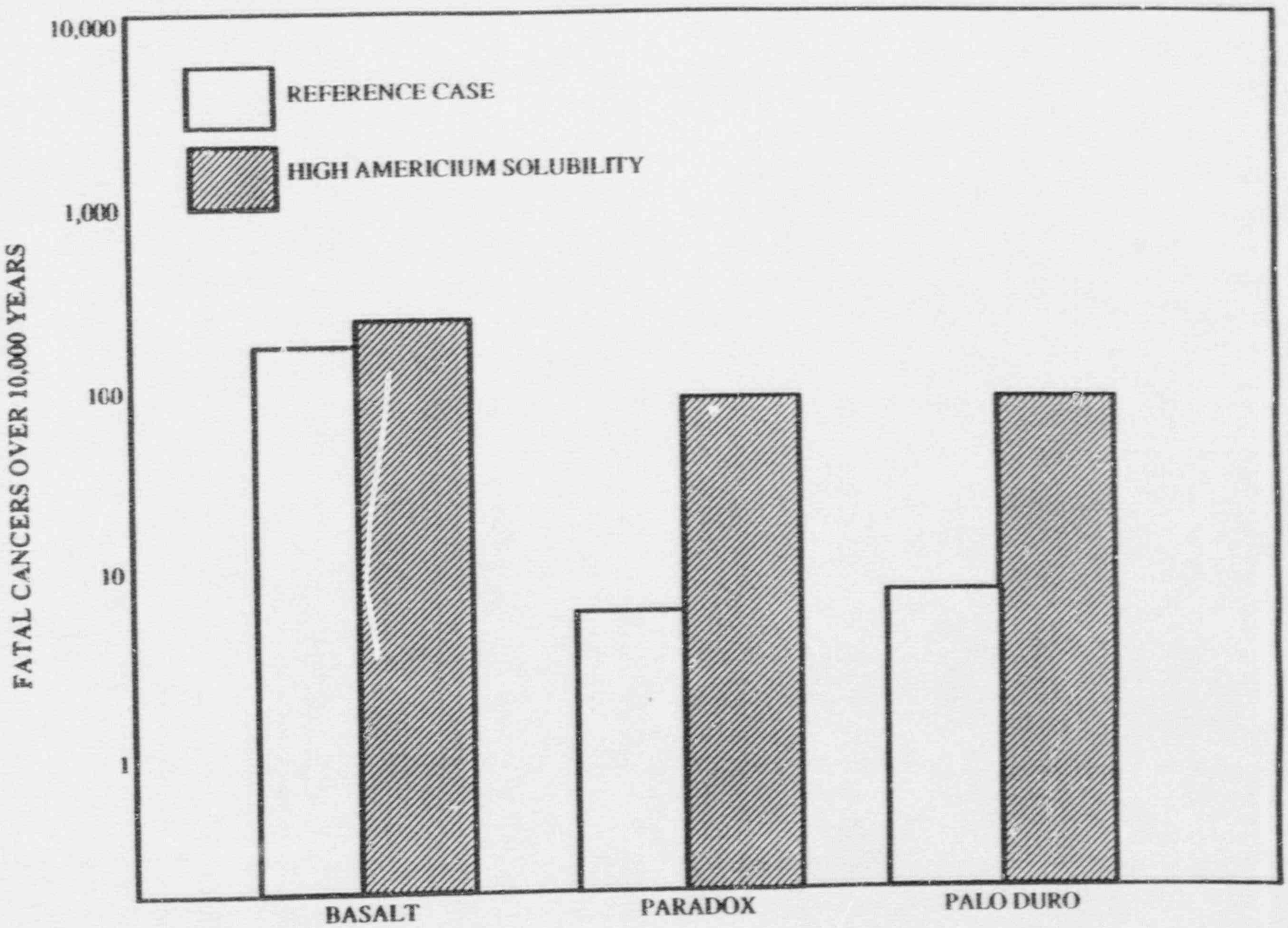


Figure 8.5-21: Effect of solubility limits on population risks for americium in different geologic media

The negligible effect for basalt is due to the relationship of the aquifer flow velocity and the retardation factors of the various radionuclides. The ground water velocity in the upper aquifer of the basalt model is about 9.4 meters per year, which corresponds to a travel time of 210 years to 2 km, 530 years to 5 km, and 1060 years to 10 km. Retardation factors are 1 for iodine-129 and carbon-14, 5 for technetium-99, and at least 50 for all other radionuclides. With these values, all three of the fast moving radionuclides reach the accessible environment throughout most of the 10,000-year period for any of the three distances considered, while none of the other radionuclides can reach even the shortest distance in 10,000 years. Therefore, changing the distance through the range of 2 to 10 km has very little effect for the basalt model.

Sensitivity to Event Probabilities. Most of the event probabilities used in these analyses were intended to be conservative values that probably overestimate the frequently of the various disruptive events. Therefore, the sensitivity of the population risk estimates to lower event probabilities was evaluated. For these analyses, the fault movement frequency was decreased by an order of magnitude or more for each of the media considered (except for basalt).

In addition, the inadvertent human intrusion frequency was decreased by a factor of two, and no intrusions were assumed to occur sooner than 500 years after disposal. Figure 8.5-23 displays the results of these changes for the models in basalt, bedded salt, and tuff. The variations in the population risks in response to these changes are relatively modest, indicating the risk estimates are not highly sensitive to the values used for the frequency of disruptive events.

Sensitivity of Bedded Salt Models to Host Rock Permeability. The models used so far for repositories in bedded salt assume that the salt formation itself is essentially impermeable unless it is disturbed. Therefore, all releases occur either as a result of disruptive events or because of ground water leakage through shaft or borehole seals (and none of the seal leakage pathways results in predicted releases to the accessible environment within 10,000 years after disposal).

In the analyses described in this section, the assumption of impermeability was relaxed in recognition of the possibility that impurities and irregularities in the salt formation could lead to normal ground water flow through the salt. It was assumed that 5 percent of the repository inventory and volume was subjected to ground water flow through the salt, with this flow zone having a hydraulic conductivity of 10^{-9} cm/s and a porosity of 0.05. The gradients and the remainder of the parameters used were the same as for the two bedded salt base case models.

No results are displayed for these analyses, however, because these changes caused no additional releases for either of the bedded salt models. Therefore, the finding that there are no projected releases due to undisturbed performance of bedded salt

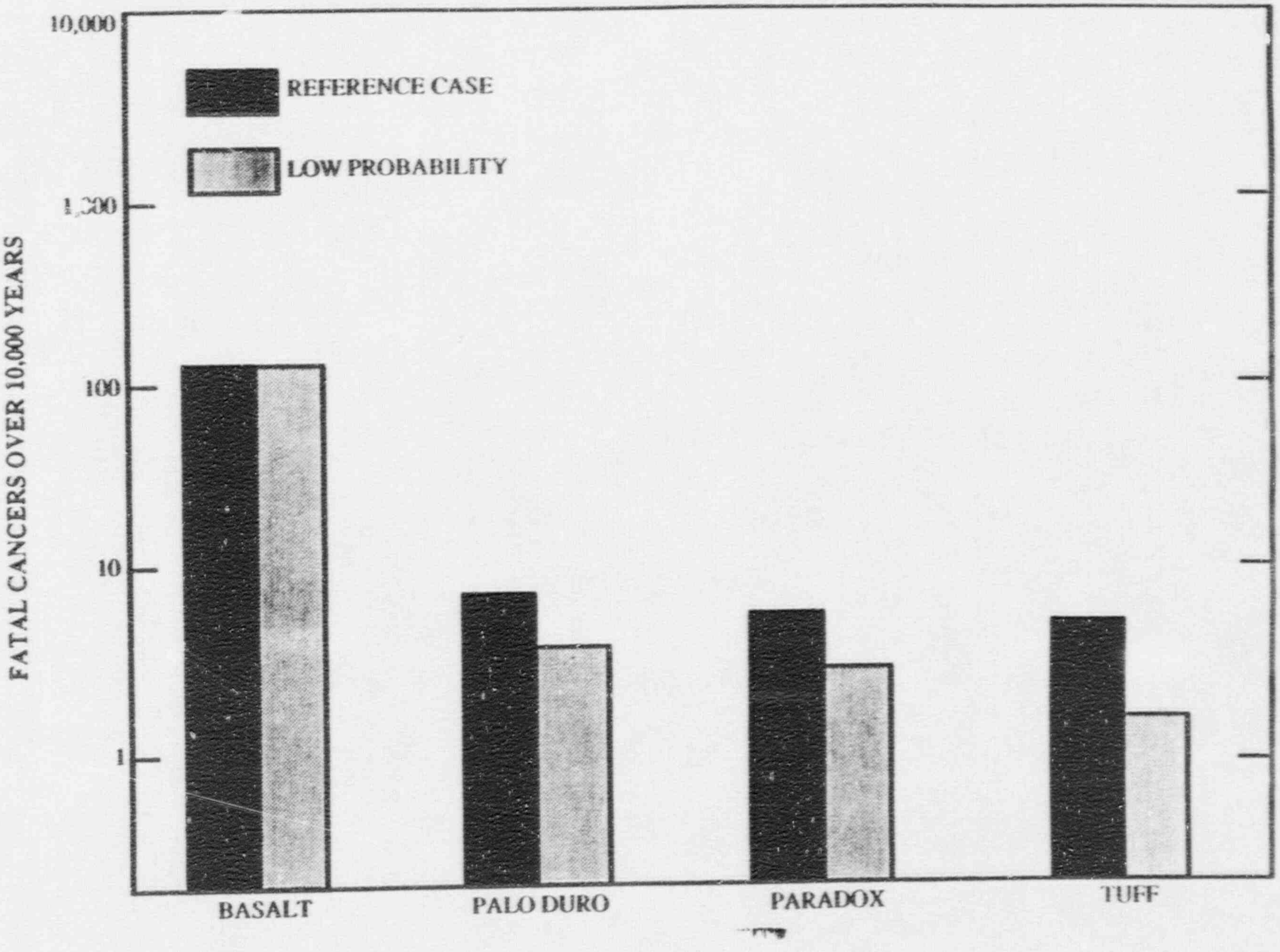


Figure 8.5-23: Sensitivity of population risks to event probabilities

repositories appears to hold, even if there are small permeable zones of this magnitude through the salt formation.

Summary of Sensitivity Analyses. The sensitivity analyses performed in support of the final rule provide several perspectives concerning the relative importance of the various parameters to the projection of population and individual risks, as well as to the achievability of the disposal standards. First, it must be noted that the great majority of the various combination of assumptions regarding site and engineered barrier characteristics result in population risk projections indicating compliance with the final containment requirements. Second, it is apparent that certain parameters are more important than others for demonstrating such compliance.

With regard to the engineered barriers, the waste form release rate always appears more significant to the predictions of population risks than the canister lifetime, and the waste form release rate appears to be as significant as many of the geologic characteristics. Thus, a good waste form may be able to overcome uncertainties about the characteristics of a site. On the other hand, the canister lifetime is particularly significant for keeping risks to individuals (using ground water near a repository) to acceptable levels, since even a very good waste form cannot keep the radionuclide concentrations in ground water near the repository small enough to avoid significant exposures to a nearby individual.

Among the geological characteristics, the analyses performed in support of the proposed rule indicated that geochemical retardation and limits of the solubility of many of the elements in the waste were very important to long-term risks, compared to cases where no retardation or solubility limits are assumed (Sm82). However, the analyses performed for the final rule suggest that the risks generally are not unusually sensitive to these factors *within a range of reasonable assumptions* such as those provided by the NAS WISP report. Thus, assuming the existence of some minimal retardation and modest limits on solubility, determining the exact contribution of these rather uncertain processes may not be of primary importance relative to compliance with the disposal standards. Relatively unimportant parameters with the ranges of sensitivity considered were the frequency of occurrence of disruptive events and travel distance to the accessible environment (although travel time and distance are particularly important in terms of when significant risks to individuals, using ground water at a particular location, may occur).

These analyses reinforce the Agency's belief that the final disposal standards can be successfully implemented. There appear to be a wide range of potential geologic media and repository designs that can reasonably achieve the desired level of protection. Furthermore, those parameters that appear to be of particular importance to the long-term release projections (e.g., waste form release rate and physical characteristics of the host rock) are those which should be relatively well known after site characterization and design and testing of the engineered portions of the planned disposal systems.

References

ADL 79 Arthur D. Little, Inc., Technical Support of Standards for High-Level Radioactive Waste Management Vol. A-D and Addendum. U.S. Environmental Protection Agency, EPA 520/4-79-007, Washington, D.C., 1979.

Sm82 Smith C.B., Egan D.J., Williams W.A., Gruhlke J.M., Hung C.Y., and Serini B., Population Risks from Disposal of High-Level Radioactive Wastes in Geologic Repositories. U.S. Environmental Protection Agency, EPA 520/3-80-006, Washington, D.C., 1982.

EPA85 U.S. Environmental Protection Agency, Risk Assessment of Disposal of High-Level Radioactive Wastes in Geologic Repositories, EPA 520/1-85-028, Washington, D.C., 1985.

8.6.1 Population Risks

8.6.1.1 System Model. The waste disposal system considered for this risk assessment is based on DOE plans to operate a mined geologic rerpository in bedded salt for the permanent disposal of defense-related transuranic (TRU) wastes. The risk assessment presented here describes the choice of radionuclide release scenarios and physical and chemical parameters used to determine, in a preliminary way, the potential compliance of the proposed salt-hosted TRU waste geologic repository with the Standard. These calculations differ from many of the previous EPA calculations in that almost all of those previous calculations were for a high level waste repository, rather than a transuranic waste repository. This is important not only from the standpoint of radionuclide inventory, but also because the mode of emplacement and the geometry of the repository are quite different. Furthermore, there are relatively few canisters for TRU wastes, and the waste form is such that a restrictive leach rate does not appear to be a limiting factor on levels of release.

To promote waste isolation, the planned repository will be situated at a depth of approximately 650 meters below the land surface in a thick salt bed with minor anhydrite interbeds. The use of bedded salt as a waste repository host was first suggested in the early 1900's. The physical and chemical properties of salt lend themselves to this purpose better than most natural geologic materials. Under "saturated" conditions, the hydraulic conductivity of salt is extremely low, averaging several orders of magnitude less than that of a dense clay. A hydraulic conductivity of this magnitude generally translates to low fluid flow velocities. Due to its plastic behavior, salt bodies "flow" under gravitational or compressive forces. Salt flow is accompanied by a natural annealing process, such that fractures, faults and voids self-seal over short periods of time. Salt is also an excellent thermal conductor. Combined, these properties provide desireable characteristics for minimizing potential impacts to the environment from deep disposal of transuranic wastes.

Unlike the proposed tuff repository, for the disposal of high-level radioactive wastes, the TRU repository will undergo rapid physical and chemical changes before attaining a final state of consolidation and waste encapsulation. A partial list of these natural processes includes creep closure, brine infiltration, gas generation, waste compaction, and chemical and biological degradation of wastes. Creep closure of the salt repoository is a constant, ongoing process, even prior to waste emplacement. DOE projections of complete repository closure range from 60 to 100 years following waste emplacement. Compaction of wastes and backfill material will proceed until a sufficient density is achieved to prevent further compaction, however, brine volume, gas pressure, and the behavior of the disturbed rock zone surrounding underground excavations will impact the rate of closure. DOE estimates that the final porosity of the post-closure repository will range from 0.15 to 0.21 (SAND89-0462). Ref.

Brine from the matrix and fractures of the host salt formation is expected to flow into the proposed repository. The precise mechanics of brine influx and the future rate and volume of flow are uncertain. DOE conservative estimates for brine influx are on the order of 1.3 cubic meters per year per panel. Brine inflow is driven by the differential between the pre-excavation pore pressure in the salt and atmospheric pressure in the repository. Once the repository is sealed, brine will continue to flow into excavated panels until the waste-generated gas pressure is equivalent to the fluid pore pressure in the salt. Significant questions remain concerning the behavior of gas and brine flow in the disturbed salt surrounding excavations and the rates and nature of gas generation due to the chemical and biological breakdown of wastes.

Current DOE efforts are focussing on repository backfill materials that may be used to speed consolidation, reduce gas generation, reduce brine inflow, and limit the mobility of radionuclides once waste containers have failed. Surface monuments for warning future generations of the presence of the underground facility are planned, however, this risk assessment modeling assumes that passive institutional surface controls are effective for only the first 100 years following facility closure.

The risk assessment described here was conducted by the Agency to achieve two purposes: 1) to identify the most important potential radionuclide escape pathways from the TRU repository, and 2) to estimate the probability of radionuclide release by various mechanisms to the accessible environment and the attendant population health consequences due to those releases at select time periods in the future.

Radionuclide escape from the repository represents a threat to future generations only if a viable means exists by which material from the repository can reach the accessible environment. At its nearest point, the accessible environment is located approximately 3 kilometers from the repository. A variety of radionuclide escape "scenarios" have been examined in the past by both DOE and others. These scenarios generally fall into one of two categories: natural occurences or human intrusion. Natural system scenarios include escape pathways such as flow through the bulk rock, flow through degraded shaft seals, and flow through breccia pipes and faults. Human intrusion into the repository at some time in the future may result from drilling for either energy resources, mineral resources, or water. Natural occurence release mechanisms were reviewed in the risk assessment supporting the proposed rule and were generally not found to cause releases to the accessible environment over the next 10,000 years, therefore this risk assessment considers only human intrusion (drilling) scenarios, from a quantitative standpoint, for the proposed TRU waste repository.

Radionuclide escape to the accessible environment through a drilling event will likely occur by one or both of the following transport pathways: 1) direct borehole pathway to the surface, or 2) slow ground water flow through a bore hole to an overlying aquifer and subsequently to the surface. Each of these release mechanisms is described in greater detail in Section 8.6.1.5, Release Mechanisms. Sensitivity analyses are presented to demonstrate the potential consequences to populations from release mechanisms operating under a variety of physical and chemical parameters.

8.6.1.2 Site Parameters. The following section describes the physical setting of the proposed TRU repository and select parameter values utilized in the risk assessment modeling of radionuclide escape from the repository. Further, this section will describe the REPRISK model structure for both natural and human intrusion release scenarios.

The proposed repository is situated in a thick Permian basin evaporite sequence, consisting of bedded halite, anhydrite, shales, and carbonates. The repository host unit is an impure halite bed ranging in thickness from 530 meters to 625 meters. Thin interbeds of anhydrite are located throughout the halite unit, notably a continuous one meter thick unit located one meter below the repository. Overlying the host halite unit is a sequence of evaporites, mudstone and dolomite beds. The dolomite layers are important to this risk assessment, in that they form the first continuous aquifers above the repository. The repository is underlain by thick evaporites to a depth of approximately 1200 meters below the surface. Pressurized brine pockets occur at depth below portions of the proposed repository location.

Two general models of radionuclide escape from the TRU repository have been examined: natural flow pathways and human intrusion pathways. The probability of occurrence of different natural flow pathways varies markedly. For example, there is a probability of one (1) that brines will flow under natural or induced gradients from the repository toward the accessible environment. The probability of this occurrence is high, however, the corresponding consequence is so small that it is of minimal concern from the standpoint of exceeding the Standard. Calculations of natural flow through the bulk rock suggest that the sum total of radionuclide fractions released over 10,000 years is negligible, therefore this scenario is not considered further in a quantitative sense. Alternatively, natural flow pathways such as faults and breccia pipes may have significant potential consequence from the release standpoint, however, the probability of such an event occurring in the salt bed repository is negligible. Human intrusion pathways consist of boreholes drilled from the surface and solution mining impacts. The probability of impacting the facility through solution mining was determined to be inconsequential, therefore it is not considered further. Consequently, human intrusion into the salt bed repository is the only release mechanism considered for modeling in this risk assessment.

The first laterally-continuous aquifer overlying the repository is the likely pathway of contaminant transport to the accessible environment for all bore hole scenarios except the "direct hit", which releases contaminants to the land surface. The previously-described dolomite aquifer is characterized by a combination of matrix and fracture flow hydraulic behavior. Hydraulic conductivity over the flow path to the accessible environment ranges over several orders of magnitude, from 10^{-4} m/s to 10^{-7} m/s. For the risk assessment modeling, a conservative value was used to reflect the dominant role of fracture controlled ground water flow from the proposed repository site to the accessible environment. The minimum transport distance (in the dolomite aquifer) from the edge of the repository "footprint" to the accessible environment is approximately 3 kilometers.

An upward natural gradient is assumed to exist for any borehole pathway that might be present. Exploration boreholes are assumed to be at least partially sealed with a concrete-like material following completion of drilling activities. The long term hydraulic character of a borehole seal is assumed to reflect an anticipated degradation with time. The risk assessment presented here assumes the borehole to have a hydraulic conductivity of 10^{-5} m/s (sandy gravel) with a porosity of .25 and a cross sectional area of .05 m^2.

8.6.1.3 Repository Parameters. Certain engineering specifications for the TRU repository are assumed here. These dimensional and structural parameters (from SAND89-0462) are shown in Figures 8.6-1 and 8.6-2 and Table 8.6-1. The repository will be situated at a depth of 655 meters below the ground surface. The underground workings consist of an experimental area and a storage area, separated by a shaft and access tunnel central area. The waste storage area is equidimensional in plan view, measuring 700 meters to a side. Wastes will be stored in a series of eight panels, each consisting of seven rectangular rooms. The waste storage rooms measure 4 meters high by 10 meters wide by 90 meters in length. Four shafts will service the underground facility, including a waste handling shaft (measuring 5 meters by 5 meters), and exhaust and intake shafts. Ref.

Waste isolation systems in the facility will incorporate a series of panel seals and seals between the waste storage area and the access drifts. Panel seals will be located at the junction of main north-south access drifts and the individual panel access routes. Seals will be constructed of "block-form" preconsolidated pre-crushed salt, with an expected permeability of 10^{-20} m^2 and a porosity of 0.05. Rigid temporary composite seals may be used for short term brine flow control measures and pressure attenuation between the waste storage area and the access drifts. Access drifts will be filled with crushed salt backfill.

A proposed shaft seal system for the TRU waste repository is illustrated in Figures 8.6-3 and 8.6-4. A two component seal system is devised for each shaft. The seal system consists of an upper and lower seal, separated by loosely consolidated backfill. The lower seal measures 200 meters vertically and is designed for long

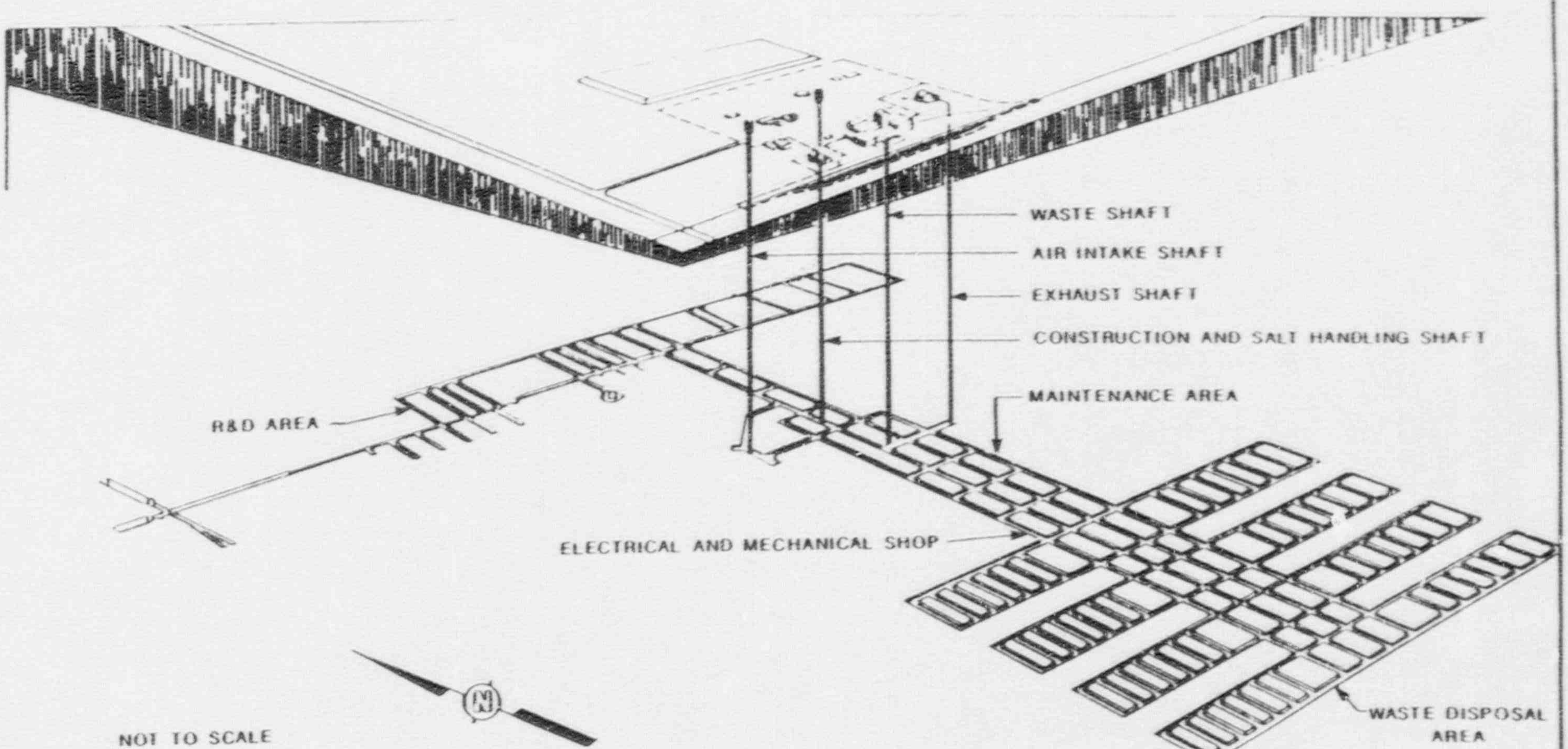


Figure 8.6-1: Schematic of the Salt Repository

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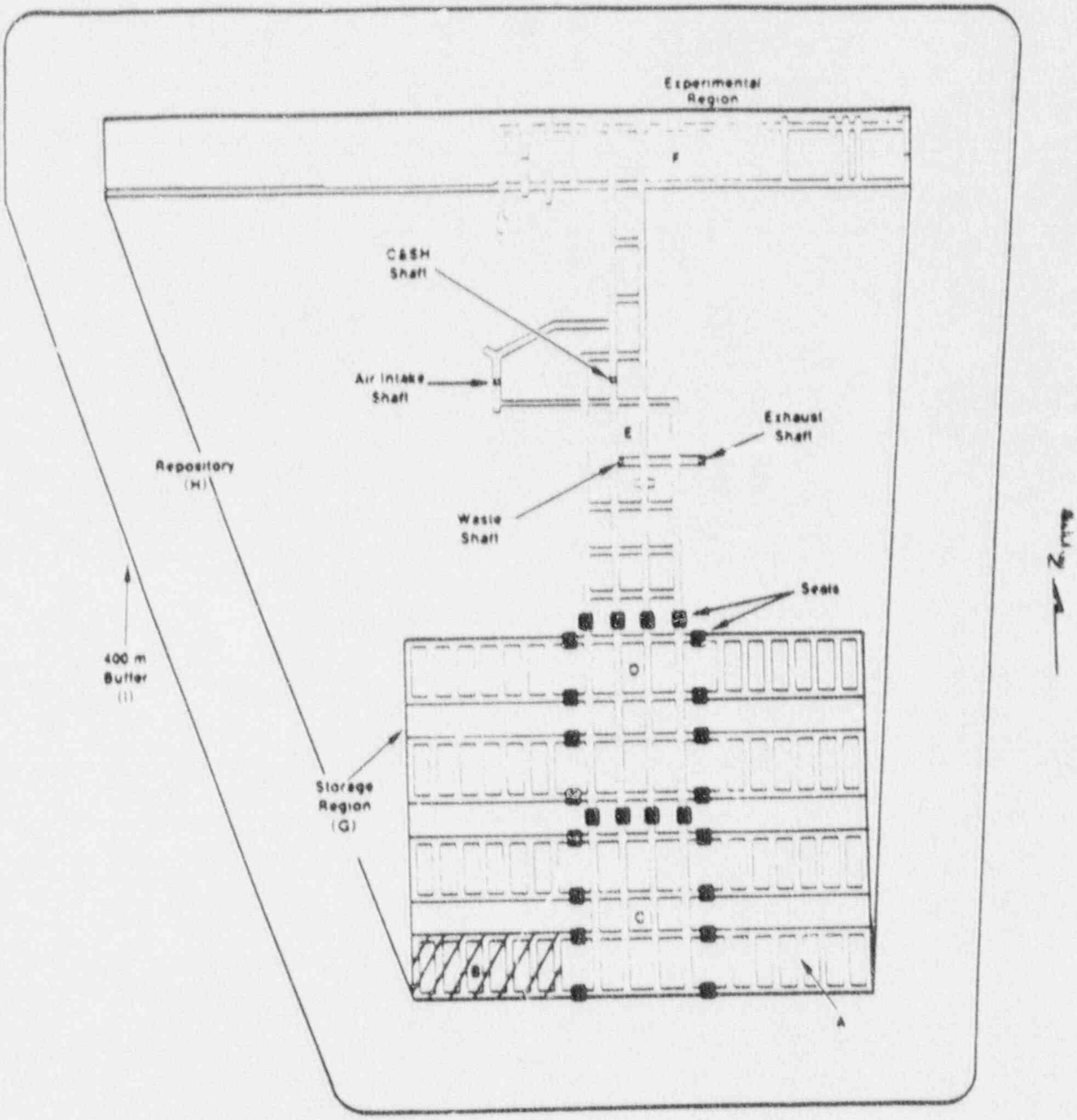
Figure 8.6-2: Salt Repository Layout



- 8 waste panels
- 7 rooms per panel
- Room dimensions = 4m(h) x 10m(w) x 92m(l)
- 4 shafts
- Total repository excavated volume = 5.8 x $10^5 m^3$
- Shaft seals composed of reconsolidated crushed salt and salt backfill
- Estimated inventory: 385,000 steel drums
  19,500 "boxes"

Arthur D Little

Table 8.6-1: Repository Parameters

Region	Areas: Excavated (10^4 m^2)	Areas: Enclosed (10^4 m^2)	Volume: Excavated (10^4 m^3)	Volume: Enclosed (10^4 m^3)
Room	0.092	0.092	0.36	0.36
One panel	1.2	2.8	4.6	11.0
Southern equivalent panel	0.84	3.5	3.3	14.0
Northern equivalent panel	0.87	3.6	3.4	14.0
Access drifts	2.2	28.0	7.8	100.0
Experimental area	2.2	30.0	7.2	110.0
Total storage area	11.0	49.0	43.0	190.0
Total repository	15.0	170.0	58.0	690.0
Buffer zone (only)	0.0	270.0	0.0	
Land-withdrawal zone	0.0	3700.0	0.0	
Four shafts (only) to base of Rustler Fm.	0.009	0.009	3.5	3.5
DRZ in storage region	0.0		0.0	57.0

No. of waste panels	8
No. of rooms per panel	7
Room height	4m
Room width	10m
Room length	92m

Source: Lappin, et al., 1989

Figure 8.6-3: Design of Lower Shaft Seal System

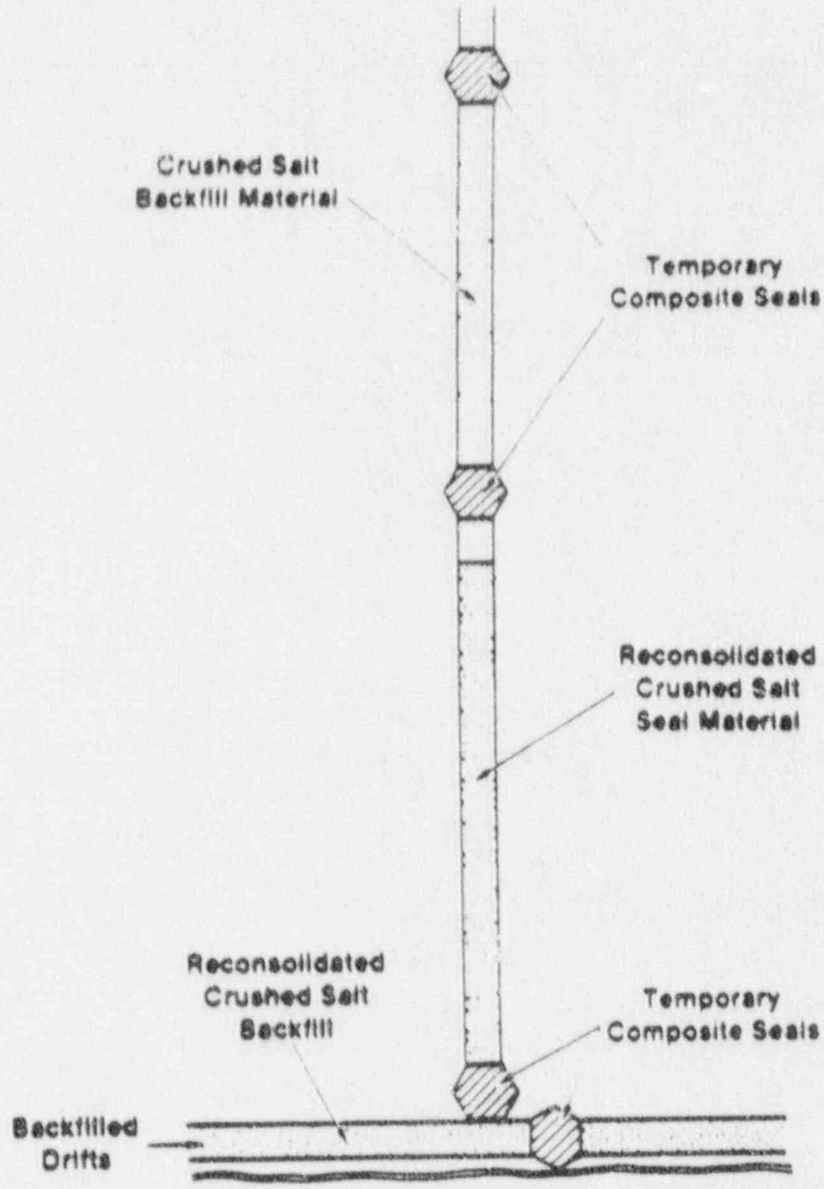
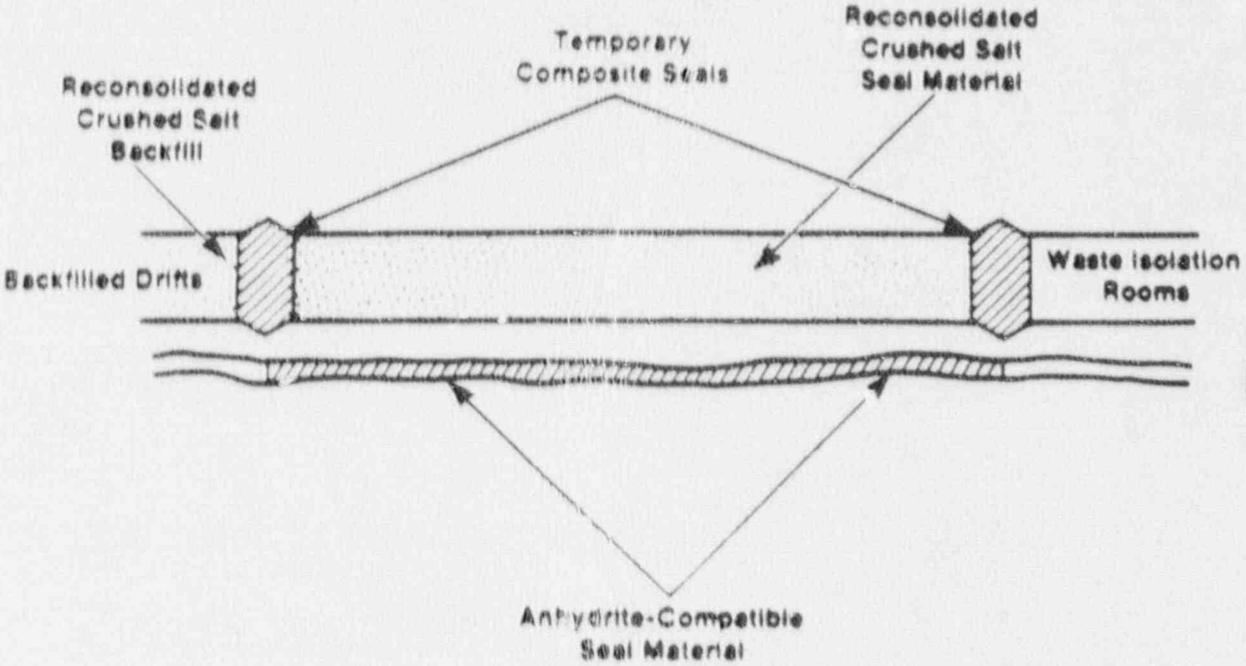


Figure 8.6-4: Design of Panel Seals



term performance and isolation of TRU wastes. The upper seal is designed largely to prevent water inflow to the shafts from water-bearing units overlying the repository.

8.6.1.4 Waste Form Parameters. The TRU wastes consist of alpha-emitting radionuclides generated through plutonium reprocessing, fabrication, research, and development activities at defense-program DOE sites. The wastes include a variety of contaminated materials such as laboratory trash and solidified waste water treatment sludge. TRU wastes are classified as either CH (contact handled) or RH (remote handled), depending upon the radiation dose rate at the surface of each package. CH TRU waste is categorized as that having a surface dose rate of less than 200 millirem per hour (mrem/hr); conversely, RH TRU waste is that which measures greater than 200 millirem per hour. Approximately 97 percent of the TRU waste designated for disposal at the proposed facility consists of CH materials. The principal CH waste radionuclides, in terms of projected Curie content, include plutonium-238,-239,-240, and -241, americium-241, cesium-244, and uranium-233. The principal RH waste radionuclides include uranium-235 and plutonium-239 (see Table 8.6-2 for waste inventory).

In addition to TRU wastes, the facility will store a variety of comingled, potentially hazardous chemical constituents generated through defense-related activities. The comingled wastes, refered to as "mixed wastes", share similar physical and radiological characteristics with TRU wastes which do not contain additional chemical constituents. Lead, in the form of glove box parts and lead-lined aprons, is a major chemical constituent in the mixed wastes. Additional mixed wastes include metal contaminants, such as barium, cadmium, and chromium, and organic solvents, such as toluene and methylene chloride. Additional detail concerning the waste forms can be found in (DOE-EIS-0026).

All CH TRU waste will be containerized in sealed 55 gallon steel drums and boxes. Current DOE plans for waste disposal methods include a three-tier packing configuration of CH waste drums and horizontal storage of RH TRU wastes in specially designed canisters. Horizontal boreholes in the walls of waste panels will be used for the permanent disposal of RH TRU waste canisters to effectively dissipate the expected minor thermal loading.

8.6.1.5 Release Mechanisms. Two potential radionuclide release mechanisms are described in this section; normal ground water flow and inadvertent intrusion by exploratory drilling. On the basis of previous screenings of potential release scenarios from the proposed salt repository, especially in connection with the probability thresholds described in the draft 40 CFR 191 Standard, only the case of human intrusion by means of future boreholes has been retained for quantitative evaluation of potential releases. This scenario falls into the category of "reasonably foreseeable" events, and thus must be compared against the release limits specified in Table 1 of the Standard. The normal ground water flow

Table 8.6-2: TRU Waste Inventory

Initial CH and RH Waste Inventory

CH Waste Radionuclide	t1/2 (yr)	Curies
Th-232	1.41×10^{10}	2.74×10^{-1}
U-233	1.59×10^{5}	7.72×10^{3}
U-235	7.04×10^{8}	3.70×10^{-1}
U-238	4.47×10^{9}	1.47
Np-237	2.14×10^{6}	8.02
Pu-238	8.77×10^{1}	3.90×10^{6}
Pu-239	2.41×10^{4}	4.25×10^{5}
Pu-240	6.54×10^{3}	1.05×10^{5}
Pu-241	1.44×10^{1}	4.08×10^{6}
Pu-242	3.76×10^{5}	1.80×10^{1}
Am-241	4.32×10^{2}	6.37×10^{5}
Cm-244	1.81×10^{1}	1.27×10^{4}
Cf-252	2.64	2.03×10^{4}

RH Waste Radionuclide	t1/2 (yr)	Weight Fraction
Sr-90	28.5	.0009
Ru-106	268 days	0
Sb-125	2.7	0
Cs-137	30	.0011
Ce-144	285 days	.0003
Eu-155	4.7	0
U-235	7.04×10^{8}	.75
Pu-239	2.41×10^{4}	.227
Pu-240	6.54×10^{3}	.0206
Pu-241	1.44×10^{1}	.0021

Total RH Curies = 5.17×10^{5}

Source: Lappin et. al., 1989.

scenario is described here due to the high probability of occurrence, with the understanding that the consequence is negligible.

The conceptual framework for normal ground water flow releases is described in detail in SAND89-0462 and is summarized below. Following completion of waste emplacement, all panels, shafts and nearby boreholes are sealed and fractures in the thin anhydrite bed underlying the repository are sealed below excavations. Initially the repository is free of brine, but brine influx begins immediately following closure. Gas pressure slowly increases from the microbial decay of organic wastes until it approaches a sufficient level to inhibit further brine influx. DOE assumes that gas generation continues for at least 2000 years. Gas pressure diminishes as microbial decay of organic material ceases. Brine inflow resumes, resulting in saturation of all wastes. Brines containing released radionuclides begin to migrate from the repository under a pressure differential, or hydraulic gradient, between the repository and the overlying aquifer. Potential releases to the accessible environment result from: 1) flow through the bulk rock overlying the repository, 2) flow through the fractured anhydrite underlying the repository, and/or 3) flow through the fractured anhydrite, to the shafts, and subsequently to the overlying aquifer system.

On the basis of asssumed physical and chemical parameters, DOE calculated that the least retarded nuclides reach the overlying aquifer, by way of the shafts, in 2.8 million years and by way of the bulk rock in 400,000 years. Under degraded shaft seal parameters, DOE calculated that the least retarded nuclides reach the overlying aquifer in 25,000 years.

These calculations point out the likelihood that no releases of radionuclides to the accessible environment will occur from the proposed salt repository under expected or degraded conditions by way of normal ground water flow within 10,000 years, thereby satisfying the Standard under this scenario. No quantitative evaluation of normal ground water flow is presented here.

The conceptual framework for the modeling of ground water releases by way of inadvertent exploratory drilling is as follows. On the basis of reference drilling rates in this kind of geologic environment, it is expected that approximately 15 boreholes would be drilled through the repository or the disturbed rock zone over the next 10,000 years. These boreholes are not modeled individually. Rather, it is assumed that they provide sufficient long-term interconnection between pressurized brine below the repository and the repository level. Therefore, it is assumed that for an indefinitely long period, the repository storage rooms and the underlying fractured anhydrite are pressurized to equilibrium with the underlying pressurized brines, and that they remain connected to this source of replacement water for any water that may be taken out by future drilling events into the repository.

The boreholes that penetrate to the pressurized brine are not considered as direct release pathways, because it is assumed that any water moving up through them would be dominated by uncontaminated brine, rather than the relatively small contribution expected to be made at the repository level. Furthermore, holes that intersect pressurized brine would be expected to be plugged or at least filled with drilling mud, so that they do not continue to flow upwards towards the aquifer or the surface in any quantity significant from the standpoint of radionuclide releases.

Given this general framework, the future drilling event considered is a sequence of three boreholes at reference points in time: 100 years, 1000 years, and 5000 years. This event is characterized as reasonably foreseeable. (Variations in the releases have been calculated based on changes in the time of drilling and in the particular characteristics of the repository.) A borehole intersecting the repository is assumed to bring three drum equivalents of waste directly to the surface. The corresponding inventory will vary with time only as a result of radioactive decay. The borehole is further assumed to be plugged, but the plug is characterized as unconsolidated aggregate, such as might result from the leaching and degradation of a cement plug or from sections that had simply been filled with sand or drilling mud. The corresponding hydraulic conductivity is assumed to be constant at 10^{-3} cm per second.

Pressure at the repository level is assumed to equilibrate with pressure in the underlying pressurized brine reserviors, and this translates into a vertical gradient of 0.76 in any borehole connecting the pressurized repository with the overlying aquifer. The borehole has an area of roughly 0.05 square meters and therefore carries a flow of approximately 11.6 cubic meters per year. When this flow reaches the aquifer, the nuclides are essentially deposited in the flow system and move according to the water velocity and retardation factors applicable to this aquifer.

When a borehole intersects the repository, the groundwater through the borehole can be thought of as having two nearby contributing sources: the waste storage room itself and the underlying fractured anhydrite. It is envisioned that the contributions of each of these two zones to the flow within the borehole will be proportional to the transmissivities (hydraulic conductivity times height). This has some effect on the releases during the base case scenario, but the effect will be much more pronounced under alternative cases which may provide for much lower hydraulic conductivity (and hence transmissivity) of the waste storage room.

Dissolution of waste in the waste storage rooms is assumed to be controlled only by solubility limits. However, where two or more isotopes of the same element are present, the solubility limits of each have been reduced in proportion to their contribution to the total mass available of the element. This can have a significant effect in reducing the mobile inventory of certain nuclides.

8.6.1.6 Risk Assessment Results. The result of the base case calculations using the methods and parameters in the previous section is that the total sum of release fractions, corresponding to the parameter required to be evaluated by the quantitative part of the 40 CFR 191 Standard, is:

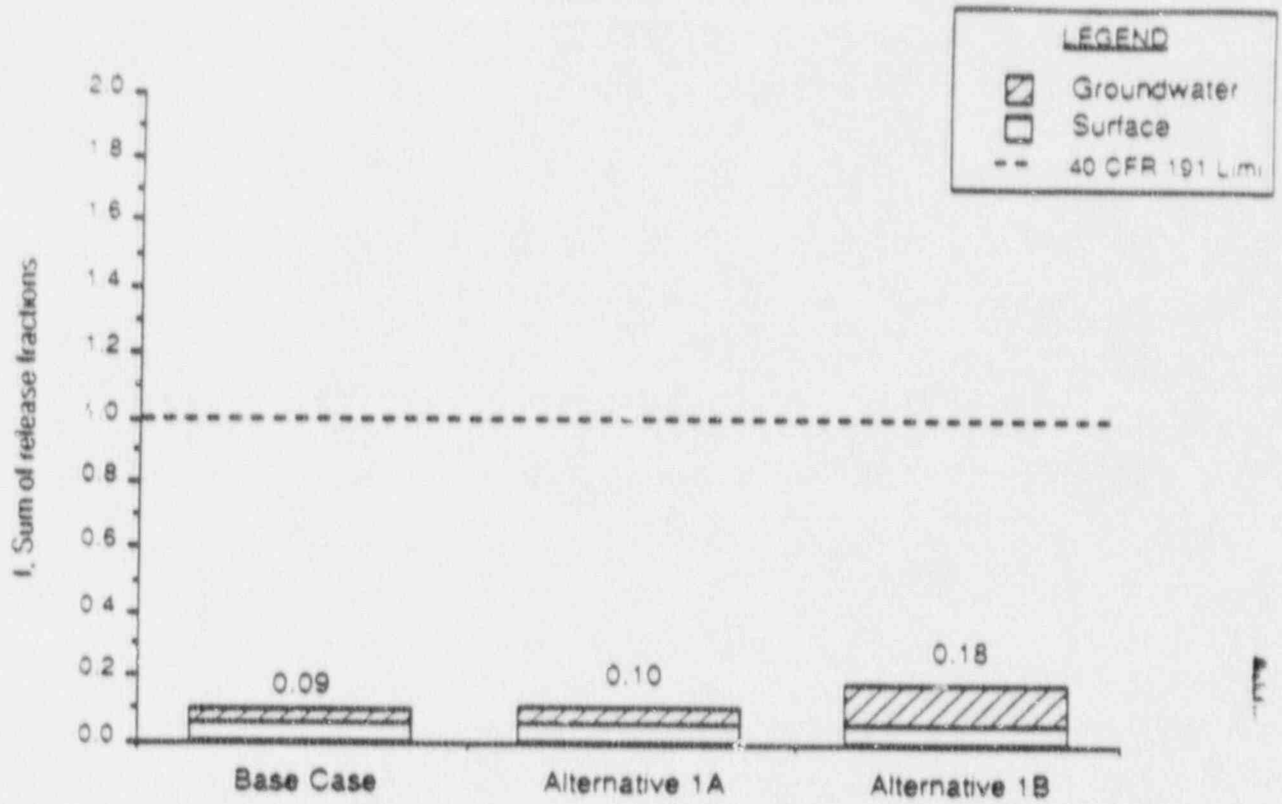
$$f = 0.09$$

The predicted releases, according to these models, lead to a total "sum of fractions" equal to 0.09, where the Standard requires that this number be less than 1. That is, the expected releases are approximately 9 percent per of the total allowable release. This implies that the base case assumptions lead to compliance with the Standard.

8.6.1.7 Uncertainties in the Risk Assessment. A number of alternative assumptions have been used for additional calculations. These alternative assumptions correspond to changes in the distance to the accessible environment, the time of future drilling events, travel time within the aquifer, aquifer distribution and retardation values, the hydraulic conductivity of plugged boreholes, the availability of water to flow through future boreholes, pressure levels in the future at the repository level, and radionuclide solubility. It should recognized that there are uncertainties associated with all of these parameters. A comparison between the base case and various alternative values in these parameters is shown graphically in Figures 8.6-5 through 8.6-12. Even with alternative values of the parameters, compliance with the Standard is indicated in all these sensitivity calculations.

While this does not imply absolute certainty that the repository can meet the Standard, especially in light of the larger uncertainties that remain involving many site characteristics, engineered systems performance characteristics, and future site evolution and human behavior, it does provide a strong indication that it is likely that the DOE will ultimately be able to demonstrate convincingly that the TRU waste salt repository will comply with the 40 CFR 191 Standard.

Figure 8.6-5: Variation of estimated performance with alternative assumptions about distance to accessible environment



Notes:

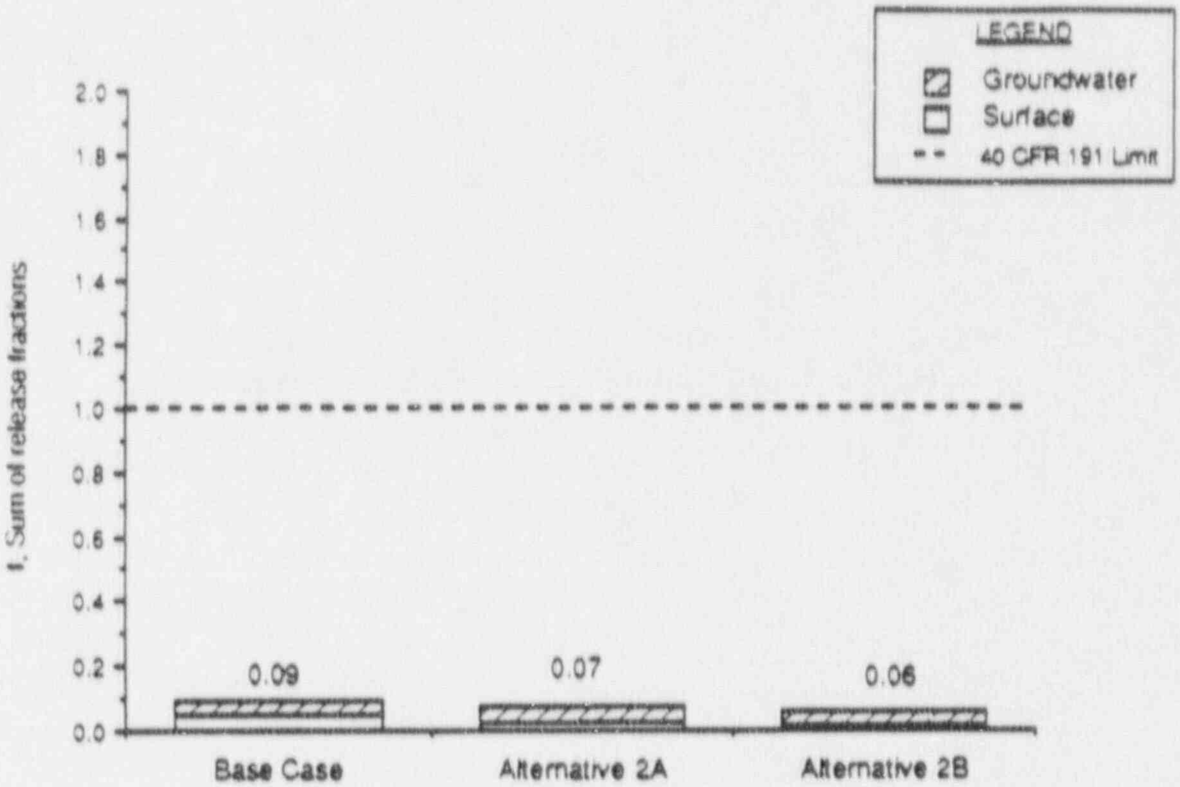
1. The results reported in this and the accompanying figures are based on simplified conceptual models of the proposed salt repository and its environment. It is recognized that there may be large uncertainties in many of the parameters used in these models, as well as uncertainties in the nature of the underlying physical and chemical processes addressed by the models. Accordingly, the models have generally been developed using a conservative approach, i.e., one that would tend to overestimate the releases. The alternatives to the base case model assumptions are intended to indicate the degree of sensitivity of the model results to changes in certain input assumptions.

2. The base case assumes a ground water travel distance of 5 kilometers in the upper aquifer from the location of any future contaminated borehole to the accessible environment.

3. Alternative 1A assumes a ground water travel distance of 3 kilometers in the upper aquifer from the location of any future contaminated borehole to the accessible environment.

4. Alternative 1B assumes a ground water travel distance of 1 kilometer in the upper aquifer from the location of any future contaminated borehole to the accessible environment.

Figure 8.6-6: Variation of estimated performance with alternative assumptions about time of drilling events



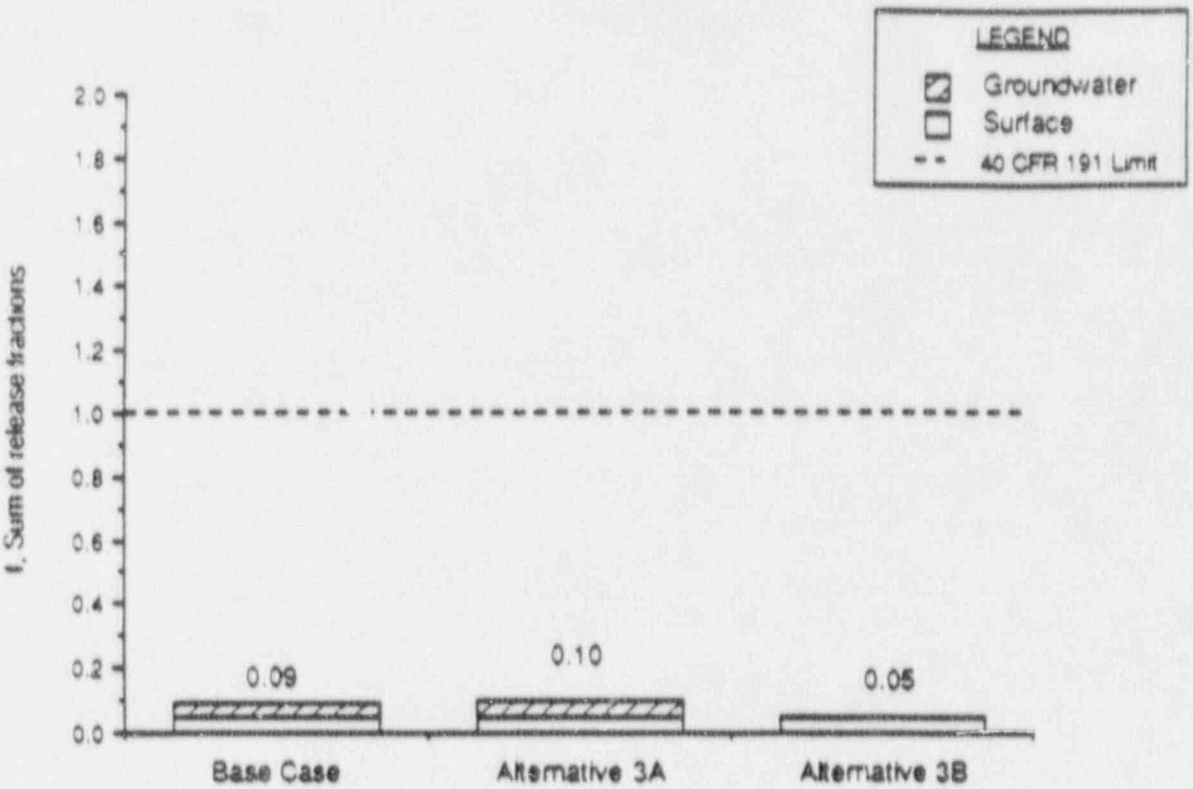
Notes:

1. The base case assumes that there are three future boreholes that intersect waste storage rooms, and that these occur at 100, 1,000, and 5,000 years after repository closure.

2. Alternative 2A assumes that there are three future boreholes that intersect waste storage rooms, and that these occur at 300, 1,000, and 5,000 years after repository closure.

3. Alternative 2B assumes that there are three future boreholes that intersect waste storage rooms, and that these occur at 500, 3,000, and 6,000 years after repository closure.

Figure 8.6-7: Variation of estimated performance with alternative assumptions about aquifer travel time



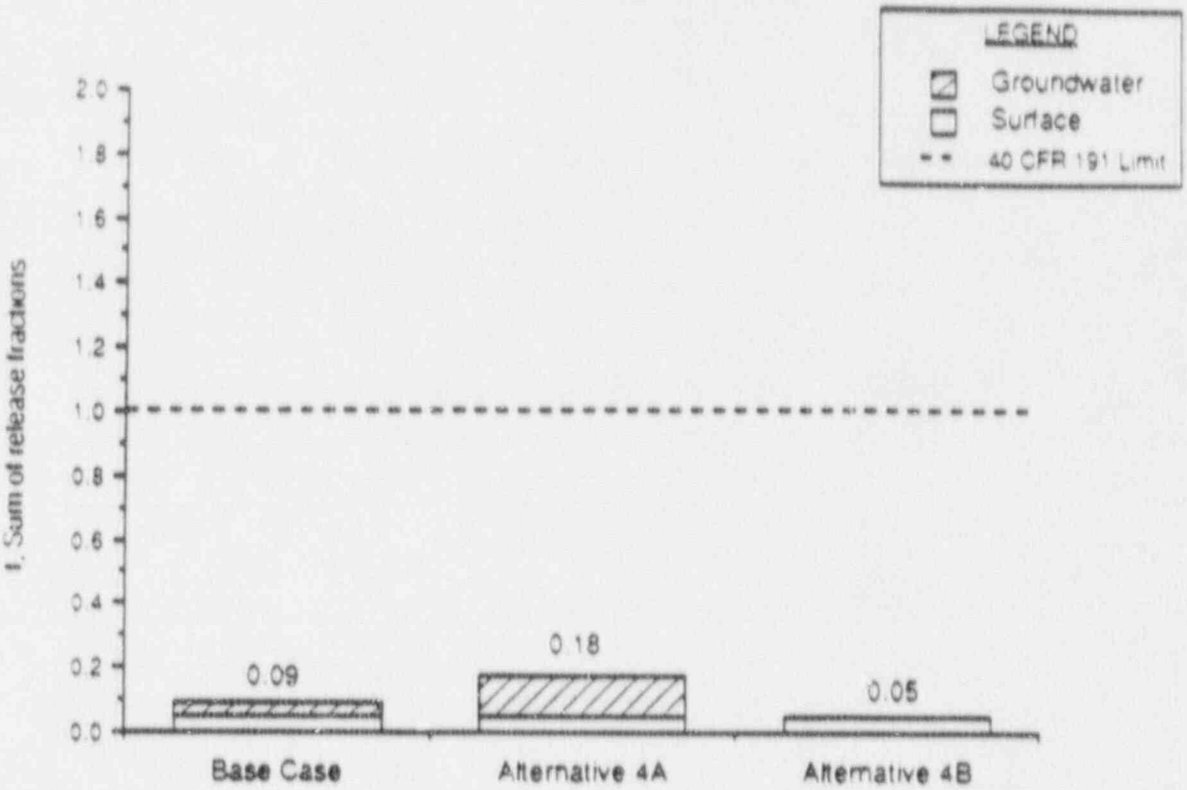
Notes:

1. The base case assumes that the ground water travel time within the upper aquifer from the location of any contaminated borehole to the accessible environment (located 5 kilometers downstream) is 187 years.

2. Alternative 3A assumes that the ground water travel time within the upper aquifer from the location of any contaminated borehole to the accessible environment (located 5 kilometers downstream) is 125 years.

3. Alternative 3B assumes that the ground water travel time within the upper aquifer from the location of any contaminated borehole to the accessible environment (located 5 kilometers downstream) is 12,000 years.

Figure 8.6-8: Variation of estimated performance with alternative assumptions about aquifer distribution coefficients



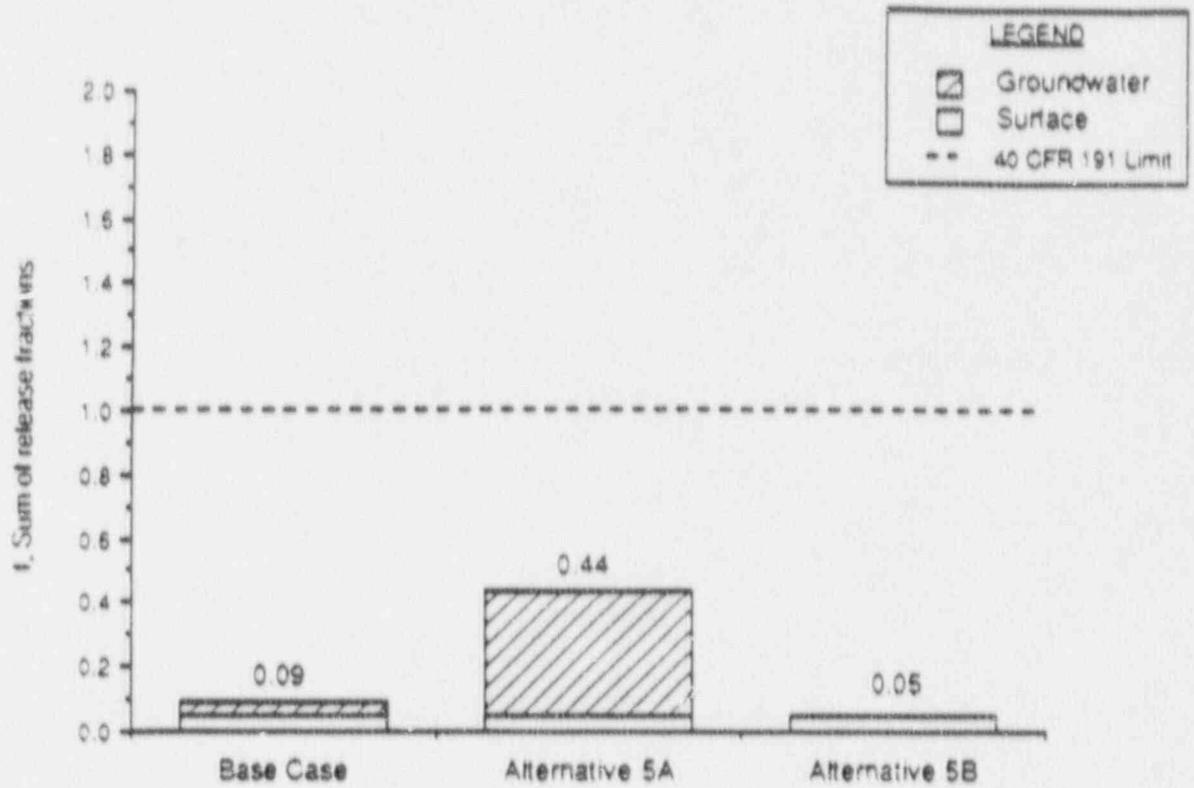
Notes:

1. The base case is based on distribution coefficients from DOE's Case IIA, as reported in Lappin *et al.* (1989). Retardation values have been calculated from these distribution coefficients based on the equation provided in that reference for matrix flow.

2. Alternative 4A is based on zero values for all distribution coefficients. This implies that there is no sorption, and thus that the retardation values are all equal to one.

3. Alternative 4B is based on distribution coefficients from DOE's Case I, as reported in Lappin *et al.* (1989). Retardation values have been calculated from these distribution coefficients based on the equation provided in that reference for matrix flow.

Figure 8.6-9: Variation of estimated performance with alternative assumptions about borehole hydraulic conductivity



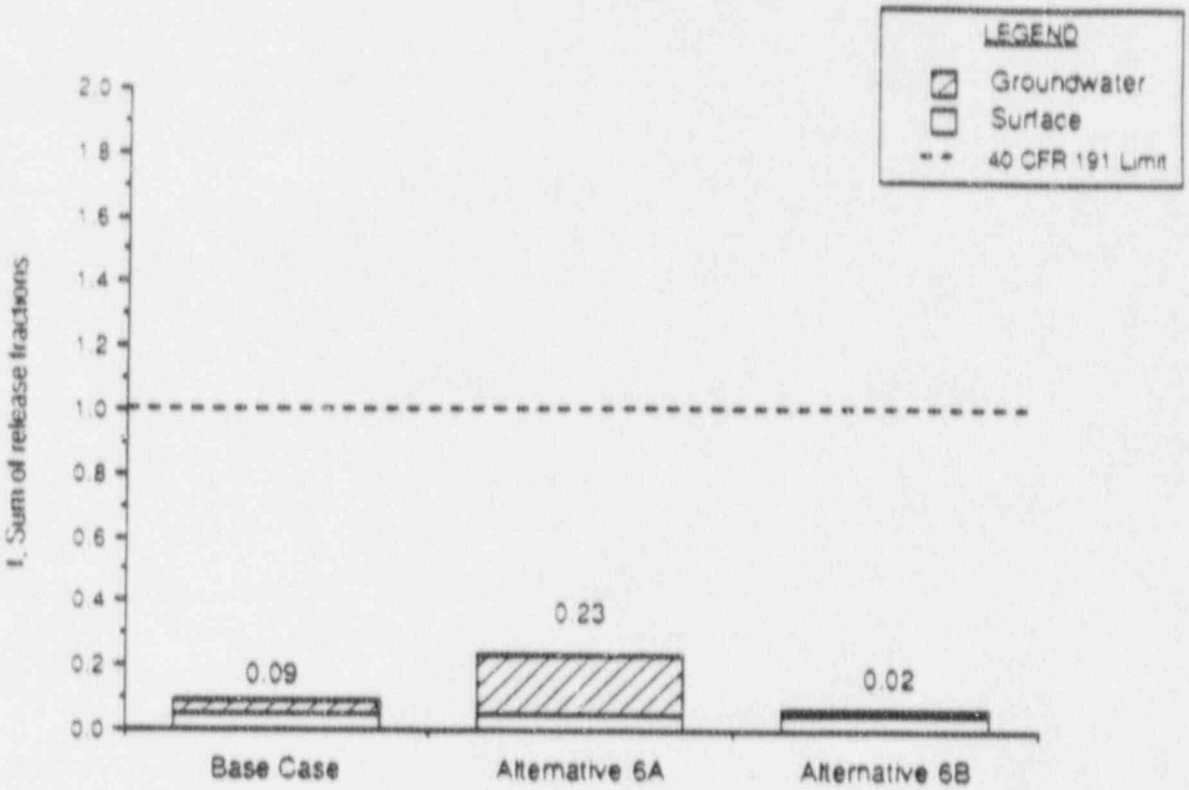
Notes:

1. The base case assumes that the hydraulic conductivity of the residual material filling the borehole is 10^{-5} cm per second.

2. Alternative 5A assumes that the hydraulic conductivity of the residual material filling the borehole is 10^{-3} cm per second.

3. Alternative 5B assumes that the hydraulic conductivity of the residual material filling the borehole is 10^{-6} cm per second.

Figure 8.6-10: Variation of estimated performance with alternative assumptions about water source to future borehole



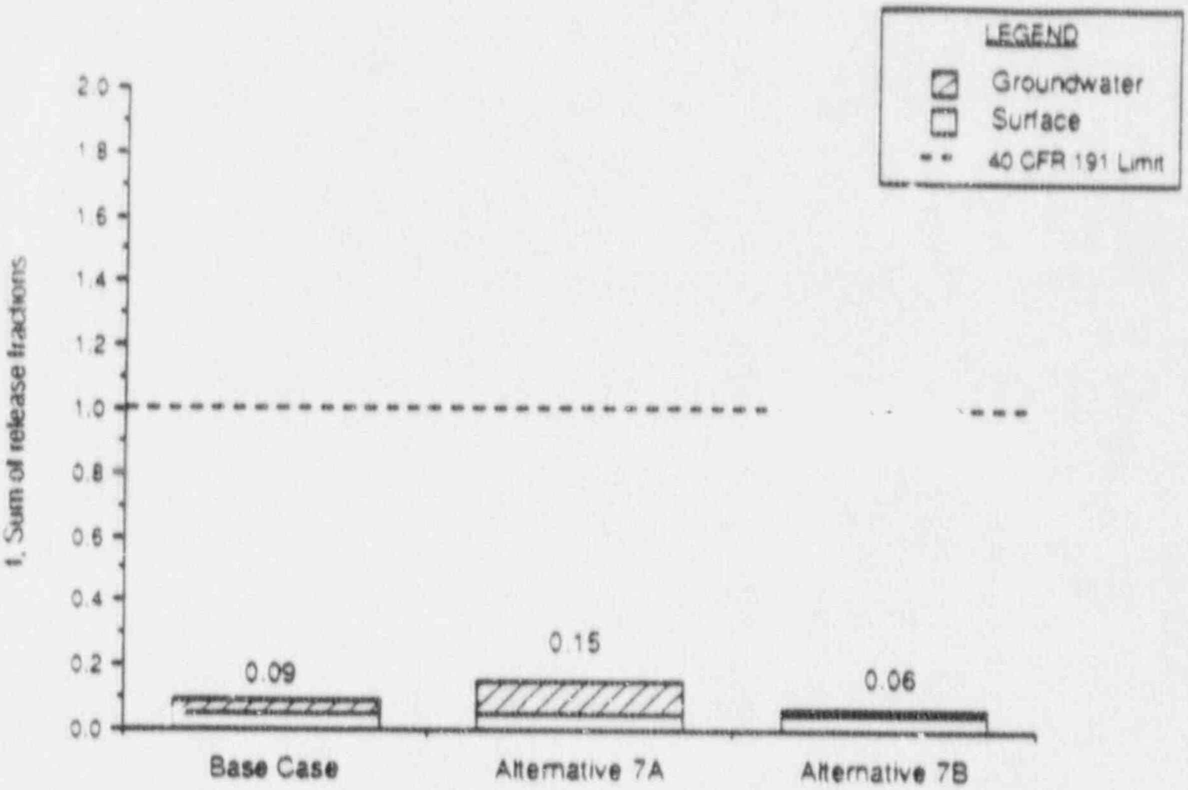
Notes:

1. The base case assumes that water that enters a future borehole intersecting a waste storage room comes partially from the waste storage room itself (at full chemical saturation of radionuclides) and partially from the underlying fractured anhydrite (at one tenth full saturation). The relative contribution from the two sources is proportional to their transmissivity, which changes in time because of the compaction and reconsolidation of the repository room.

2. Alternative 6A assumes that all water entering any future borehole comes from the waste storage room itself at full chemical saturation.

3. Alternative 6B assumes that all water entering any future borehole comes from the underlying anhydrite bed and is at 10 percent of its chemical saturation limit.

Figure 8.6-11: Variation of estimated performance with alternative assumptions about pressure at repository level



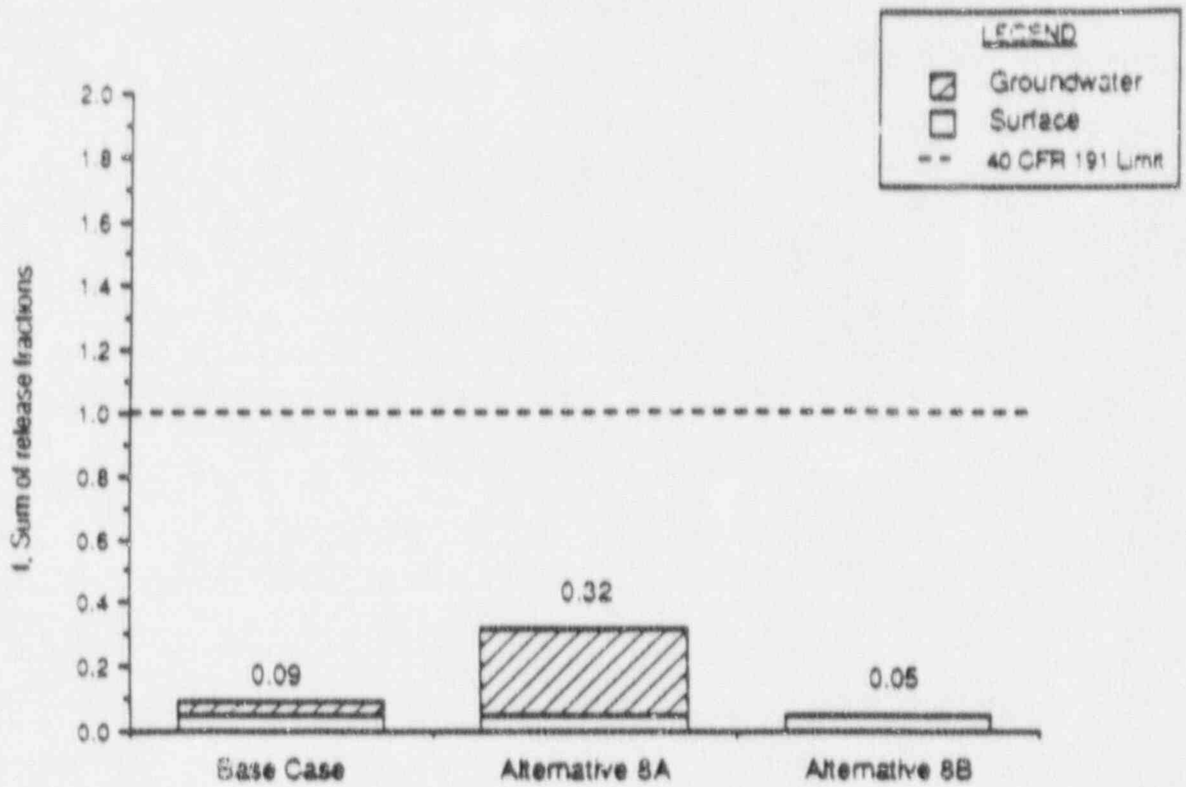
Notes:

1. The base case assumes that the repository is in hydrostatic equilibrium with high pressure brines in the underlying pressurized brines.

2. Alternative 7A assumes that brine at the repository level is at lithostatic pressure.

3. Alternative 7B assumes that brine at the repository level is at one-half lithostatic pressure.

Figure 8.6-13: Variation of estimated performance with alternative assumptions about radionuclide solubility



Notes:

1. The base case assumes that each relevant element is soluble to a concentration of 10^{4} molar, and that this solubility is distributed over the various isotopes present in proportion to their mass fraction.

2. Alternative 8A assumes that each radionuclide is present at a concentration of 10^{4} molar, with no additional limitation imposed by the presence of multiple isotopes of the same element.

3. Alternative 8B assumes concentration limits one order of magnitude lower than those assumed in the base case.

DOE/EIS-0026 Draft Supplement Environmental Impact Statement: Waste Isolation Pilot Plant, 1989 Department of Energy.

SAND 89-0462 Lappin, A.R., *et al.*, 1989; Systems Analysis, Long Term Radionuclide Transport, and Dose Assessments, Waste Isolation Pilot Plant (WIPP), Southeastern New Mexico; Sandia National Laboratories.