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PRELIMINARY PERFORMANCE ASSESSMENT

FOR A HLW REPOSITORY AT

YUCCA MOUNTAIN, NEVADA

First Draft

January 17, 1990

DRAFT

Preface

The work described in this report was a team effort that included input from the authors as well as from additional staff members.

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Because of the need to issue the report on a very tight schedule, there was not sufficient time to allow its review by all contributors prior to issuance of this First Draft. We anticipate that this larger group of participants will provide comments on this First Draft to be incorporated into a Final Draft.

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Contents

CONTENTS

		Page
0.	EXECUTIVE SUMMARY	0-1
1.	INTRODUCTION	1-1
2.	PURPOSE AND SCOPE	2-1
3.	ORGANIZATION AND STAFFING OF PHASE 1 OF THE MOU	3-1
	3.1 Administration of Phase 1 of the MOU	3-1
	3.2 Evolution of the Definition of Technical Work in the MOU	3-1
	3.3 Technical Organization of Phase 1 of the MOU	3-2
4.	SYSTEM CODE	4-1
	4.1 Introduction	4-1
	4.2 Requirements for the NRC PA Total System Code	4-1
	4.3 Survey of Existing PA Total System Codes	4-2
	4.4 Description of the NRC PA Total System Code	4-3
5.	SOURCE TERM	5-1
	5.1 Introduction	5-1
	5.2 Definition of Important Issues for Selecting Source Term Models	5-1
	5.3 Model Selection and Justification	5-4
	5.4 Source Term Inventory	5-6
	5.5 References	5-7
6.	FLOW AND TRANSPORT MODELS	6-1
	6.1 Introduction	6-1
	6.2 Definition of Issues for Selecting Performance Assessment Transport Models	6-1
	6.3 Computer Program Review and Selection	6-10
	6.4 References	6-13
7.	METHODOLOGY FOR SCENARIO DEVEOLPMENT	7-1
	7.1 Introduction	7-1
	7.2 Methodology	7-2
	7.3 Conclusions	7-12
	7.4 References	7-12

8. AUXILIARY ANALYSIS SUMMARIES.............................. 8-1

8.1 Introduction.. 8-1
8.2 Gaseous Releases of C-14............................ 8-1
8.3 Testing Statistical Convergence..................... 8-1
8.4 Analysis of Hydrologic Data......................... 8-1
8.5 Two-Dimensional (Cross Section) Flow Simulation..... 8-2

9. ANALYSIS AND RESULTS...................................... 9-1

9.1 Treatment of Scenarios.............................. 9.1-1
9.2 NEFTRAN Source Term Model........................... 9.2-1
9.3 Flow and Transport Models........................... 9.3-1
9.4 Parameters.. 9.4-1
9.5 Sensitivities and Uncertainties for Liquid Pathway Analysis.................................... 9.5-1
9.6 Total CCDF.. 9.6-1

10. CONCLUSIONS AND RECOMMENDATIONS........................... 10-1

APPENDIX A - SYSTEM CODE REVIEW.............................. A-1

A.1 System Program Summaries............................ A-1
A.2 References.. A-3

APPENDIX B - SOURCE TERM MODEL REVIEW......................... B-1

B.1 Introduction.. B-1
B.2 References.. B-6

APPENDIX C - FLOW AND TRANSPORT CODE REVIEW................... C-1

C.1 Regional Flow Program Summaries..................... C-1
C.2 Two-Phase Flow and Heat Transport Program Summaries. C-2
C.3 Geochemical Program Summaries....................... C-4
C.4 Transport Program Summaries......................... C-5
C.5 References.. C-8

APPENDIX D - GASEOUS RELEASES OF C-14......................... D-1

D.1 Introduction.. D-1
D.2 Source Term... D-1
D.3 Gaseous Transport Model............................. D-7
D.4 Conclusions and Recommendations..................... D-11
D.5 References.. D-12

APPENDIX E - TESTING STATISTICAL CONVERGENCE.................. E-1

APPENDIX F - ANALYSIS OF HYDROLOGIC DATA...................... F-1

APPENDIX G - TWO-DIMENSIONAL (CROSS SECTION) FLOW MODEL......... G-1

G.1 Introduction.. G-1
G.2 Purpose... G-1
G.3 Problem Set-up...................................... G-1
G.4 Results and Conclusions............................. G-1
G.5 References.. G-2

APPENDIX H - ANALYSIS FOR DRILLING SCENARIO..................... H-1

APPENDIX I - SYSTEM CODE STEPS.................................. I-1

APPENDIX J - DOCUMENTATION OF FILES AND PROGRAMS ON INEL CRAY XMP/24 FOR REPOSITORY PERFORMANCE CALCULATIONS..................................... J-1

J.1 Introduction.. J-1
J.2 FORTRAN Programs.................................... J-1
J.3 Batch Script Files.................................. J-2
J.4 Data Files.. J-3
J.5 Output Files.. J-3

SECTION 0

EXECUTIVE SUMMARY

5520 NAME: MOUO

EXECUTIVE SUMMARY

A Memorandum of Understanding (MOU) between the NRC Offices of NMSS and RES was signed September 1, 1988. The objective of this MOU was to expand and improve the independent NRC staff capability to conduct performance assessments. By expanding and developing the NRC staff capability to conduct such analyses, the NRC would be better able to conduct an independent technical review of the DOE licensing submittals for a HLW repository. The original plan for work to be conducted under this MOU consisted of three tasks:

1. INTRAVAL

2. Engineered Barrier System Performance

3. Acquisition and Application of Methodology for Tuff (far-field analysis)

This report concerns only Tasks 2 and 3. As planning and work proceeded for Tasks 2 and 3, Task 3 was expanded to include a total system performance assessment and the development of computational tools required to conduct such a complete analysis; in addition, the acquisition of computer codes from contractors was deferred to a later time, to allow Sandia National Laboratories to complete development of the codes and the CNWRA to be in a position to assist in the acquisition of the codes. The Task 2 and 3 MOU activities were divided into Phase 1 and Phase 2 activities. The Phase 1 activities were to be conducted by the NRC staff with minimal input from NRC contractors; the Phase 2 activities were to involve NRC contractors actively and to provide for the transfer of technology.

Purpose.

Given this organizational background, the primary focus of the Phase 1 activities for Tasks 2 and 3 of the MOU was to demonstrate the capability of the staff to conduct a total system performance assessment in an independent fashion. By demonstrating such an independent capability, the NRC staff has provided evidence of a readiness for the forthcoming review of licensing material provided by the DOE. In addition, by exercising this capability for independent review, the NRC staff has accomplished several secondary objectives, including:

1. Performing an evaluation of the adequacy of existing analytical tools, both methodologies and computational methods.

2. Obtaining valuable insights into the need for further development of methodologies and computational tools.

3. Obtaining valuable insights into the data needed from the DOE Site Characterization Program to conduct performance assessments, including the priority of these data needs.

Scope

The performance assessment is considered to be comprised of two parts:

(1) quantitative estimation of total system performance through the use of predictive models and

(2) documentation, including detailed auxiliary analyses where appropriate, to support the assumptions, data, and modeling approaches used to obtain quantitative estimates of performance.

Both of these aspects of performance assessment were addressed in the Phase 1 effort.

The total system performance measure for a high level waste repository can be expressed by a complementary cumulative distribution function (CCDF) of radionuclide releases to the accessible environment, weighted by a factor approximately proportional to radiotoxicity, integrated over an appropriate period of time (10,000 years is the current regulatory requirement). This performance measure is estimated by following the steps outlined in the information flow diagram (Figure E.1). For the Phase 1 effort, these steps were all executed, but some (for example steps 2 and 3) were only executed to a limited degree. These steps are described briefly below:

1. System Description - The repository is partitioned into its component parts for the purposes of modeling. These parts include the source term model and the flow and transport model. Computer codes are adapted or written to simulate models of these components. Ranges of parameter values are chosen to bound the expected behavior of the system models.

2. Scenario Analysis - Scenarios representing alternative futures for the system and possible future states of the environment are screened and chosen. Probabilities are estimated for chosen scenarios.

3. Consequence Analysis - The consequence in terms of cumulative release of radionuclides to the accessible environment over a specified time period (usually 10,000 or more years) is calculated for each scenario and usually numerous realizations of possible parameter values. In addition to being incorporated by way of cumulative releases into the CCDF (step 4), certain types of consequences might also be considered separately to compare to standards for maximum doses to individuals and for maximum concentration in groundwater.

4. Performance Measure Calculation (CCDF) - The consequences for each scenario, in terms of normalized cumulative releases of radionuclides to the environment over a specified period of time, are calculated and the results are displayed in a curve of consequences versus the probability that such consequences will not be exceeded. Compliance

with the performance criteria is determined by comparing the curve to two fixed points, which provide limits the curve must not exceed.

5. Sensitivity and Uncertainty Analysis - Sensitivity analysis investigates the change in performance measures caused by incremental changes in the values of input parameters and data. Uncertainty analysis attempts to quantify the uncertainty in performance estimates in terms of the major sources of uncertainty, including uncertainty in input parameters, uncertainty in modeling (both the conceptual model of the geometry and characterization of the system and the process model of what physicochemical processes occur and how they are manifested), and uncertainty about future states-of-nature. Modeling uncertainty was not quantified in Phase 1.

6. Documentation - The most effective documentation must make clear the assumptions used in the analysis, their basis, and the implications of their use explicit.

Two types of uncertainty are usually treated explicitly in the generation of the CCDF: (1) uncertainty due to future states of nature and (2) uncertainty in the values of parameters determining system performance Modeling uncertainty is usually treated in a different way. In a safety analysis for a more conventional type of system, the response of the system to any single future state of nature to be considered would be a single-valued estimate of system performance (in the parlance of the repository system, a single value of consequence). System performance would then be described by the plot of consequences versus the likelihood of the future state of nature (scenario) producing that consequence; such a curve would be the distribution function. The integral of such a curve over probability would yield a cumulative distribution function; i.e. the likelihood that the consequence would be at least of a certain magnitude. The complementary cumulative distribution function would be the curve of the likelihood that the consequence would be a certain magnitude or less. For the repository system considerable uncertainty exists concerning the values of parameters used to estimate the consequences of the repository. Traditionally the uncertainty from this source is also displayed on the CCDF, by combining the probability of a given scenario with the probability of a given set of input parameters for that scenario.

Because of the complexity of the calculation of the CCDF, the staff deemed it appropriate, but not absolutely necessary that the generation of the CCDF be performed by a computer code.

As explained above, only a rudimentary performance assessment is intended for Phase 1 of the MOU, because of limited data, resources, and time and because input from NRC contractors, which could contribute to the goals of the MOU, is not currently available. Because of the constraints on this activity the scope of the effort was limited. Some of these limitations were:

- only a preliminary analysis was intended in Phase 1
- the effort was scaled down from the original 1/89 plan for this work
- only currently available modeling tools were to be used; computer code development was to be minimized
- the analysts were to take advantage of the limited data available for the Yucca Mountain Site
- the scope of the analyses were constrained by the time and resources made available to do it
- As many components of the methodology as possible were to be executed, given the limited time and resources available; this required reducing the depth to which certain aspects were demonstrated.
- For the Phase 1 effort the EPA containment standard was to be the only performance standard considered. The EPA standards for individual protection and groundwater protection will be investigated later. Also the 10 CFR Part 60.113 subsystem requirements were not to be a subject of the Phase 1 work and perhaps not included in the Phase 2 work.
- Phase 1 was executed by NRC staff only.
- Other than existing reports, papers, and computer software packages already delivered, no contractor input was available for Phase 1, except infrequent and short personal communication.
- CNWRA involvement in Phase 1 was primarily as an observer, but would become more active as the CNWRA PA capability expands.

Work Performed

In order to perform this preliminary performance assessment and demonstrate the staff capability to conduct such work, the following types of activities were performed:

1. Computations & Support
   data input
   model setup
   code development & testing
   code execution
   output analysis

2. Auxiliary Analyses
   evaluation of assumptions
   preprocessing raw data

3. Documentation
   Draft report

By conducting the activities listed above, the NRC staff achieved the following major accomplishments:

1. The NRC staff demonstrated its capability to conduct independently performance assessments for a HLW repository; in doing so the staff gained insight into the performance of the Yucca Mountain repository and increased its insight into the capabilities and limitations of the currently available performance assessment methodology.

2. Developed a CCDF to describe performance of a Yucca Mountain HLW repository for a limited set of scenario classes, using preliminary data.

3. Modeled the liquid pathway of the undisturbed scenario class for the Yucca Mountain repository using:

   (1) the NEFTRAN computer code to simulate transport in the unsaturated zone

   (2) four vertical transport legs under the repository to account for spatial variability

   (3) an improved treatment of waste form dissolution

   (4) a nonmechanistic model of waste package failure

   This liquid pathway modeling was extended to treat pluvial conditions

4. Developed and used a total system code.

5. Developed a model and the corresponding computer code for human-intrusion by drilling.

6. Performed a preliminary statistical analysis of results (sensitivity and uncertainty) using several techniques including Latin Hypercube Sampling (LHS) and regression analysis methods.

7. Executed several auxiliary analyses:

   - potential for non-vertical flow
   - sampling requirements for CCDF generation
   - consequences of C-14 gaseous releases
   - statistical analysis of available hydrologic data for input to flow and transport models

Tenatative Results

In presenting some tentative results, the authors want to state some important caveats to be kept in mind when contemplating these results. Taking these tentative results of a preliminary analysis out of context or separating these tentative results from these caveats, may lead to the inappropriate interpretation and use of the results, for which the authors of this report cannot be responsible.

1. The results presented here have had limited peer review, has numerous simplifying assumptions, and are based on limited data; therefore, THE NUMERICAL RESULTS SHOULD NOT BE TAKEN AS REPRESENTATIVE OF THE PERFORMANCE OF A REPOSITORY AT YUCCA MOUNTAIN, NV.

2. The analysis is replete with UNCERTAINTIES regarding:

   o conceptual models
   o data
   o physiochemical models
   o models and data for predicting scenarios

3. Only a limited set of scenario classes were incorporated in the modeling, so the total CCDF presented in this report cannot truly represent total system performance.

4. The modeling of waste package failure is nonmechanistic and rudimentary; therefore, this aspect of repository performance is probably not adequately represented.

5. The flow and transport models used attempt to simulate key aspects of the performance of a repository at Yucca Mountain, but do so indirectly through modifications of transport analysis for saturated rock. A more direct representation of flow and transport in partially saturated, fractured rock is needed to assure an adequate level of confidence in the results.

Given the caveats stated above, the authors wish to remind the reader that the tentative conclusions stated below should be used only with these substantial limitations kept in mind. Based on a preliminary analysis, the staff has reached the following tentative major conclusions:

1. The areal extent of the Yucca Mountain repository appears to be an important aspect determining performance and should be included in models of performance; important aspects appear to be areal variability of:

   o waste package failure
   o depth of rock to water table
   o potential of rock units to sustain fracture flow

2. The gaseous release of C-14 could be an important issue in repository performance, but more analysis and data are needed (DOE is primarily responsible for gathering the needed data).

3. The potential for nonvertical flow at Yucca Mountain appears to be great and could have a substantive effect on the performance of a repository there. There could be perching of water along interbeds and diversion of water to shorter paths to the water table. More analyses and additional data collection by DOE are required to evaluate this significant alternative to DOE's preferred conceptual model of predominantly vertical flow.

4. For the "liquid pathway" scenario class, the most significant contributors to the consequences represented by the CCDF are isotopes of plutonium. Because plutonium behavior is poorly understood, large uncertainties exist regarding:

   o colloids
   o retrograde solubility
   o sensitivity of chemistry to oxidation state

5. For the "liquid pathway" scenario class, the important input parameters appear to be:

   o infiltration flux
   o fraction of infiltrating groundwater contacting the waste
   o uranium matrix solubility
   o saturated hydraulic conductivity for the Calico Hills Vitric unit

6. Consequence codes currently used may not be sufficiently efficient to allow analyzing many scenarios each with many input parameter vectors, so that total system performance is adequately characterized.

Preliminary Recommendations

Based on this preliminary analysis and the limitations noted, the staff has some preliminary recommendations regarding the directions for further technical work to take. These recommendations for technical improvements include improvements to (1) modeling used to estimate performance, (2) analyses used to support the estimates of performance, (3) scientific input and research needed to provide a better basis for the estimates of performance.

Recommended improvements to modeling of performance:

General

1. Add the capability for modeling additional scenario classes.

2. Test the system code using the consequence codes as subroutines, instead of generating data sets external to the system code.

3. Acquire, test, and evaluate codes developed by SNL for a repository in the unsaturated zone.

4. Explore, with the CNWRA, the adaptation of the FPPA (Fast Probabilistic Performance Assessment) methodology to generate the total system CCDF.

5. Evaluate additional codes, which could not be acquired and evaluated during this short-time effort, to determine whether existing codes can meet the NRC modeling needs or whether additional code development is needed.

Flow and Transport

1. Refine groundwater modeling (e.g., by considering higher dimensions).

2. Incorporate a model of gas-pathway transport in the calculation of the CCDF.

3. Include flow and transport through the saturated zone.

4. Directly model transport through a partially saturated, fractured rock, instead of the indirect, approximate representation used in Phase 1.

5. Explicitly model fracture/matrix coupling.

Source Term

1. Attempt to develop or use a previously developed mechanistic model of waste package failure

2. Develop a mechanistic model of contact between groundwater and the waste

3. Treat the repository as a source of radionuclides distributed in time and space, instead of as a point source

Recommended improvements to and extensions of auxiliary analyses:

1. Perform detailed geochemical analyses to investigate:

   - use of K_D's (distribution coefficients)
   - effects of spatially varying saturation on radionuclide migration
   - waste form, groundwater, tuff reactions
   - waste package degradation
   - oxidation of the spent fuel matrix
   - plutonium behavior

2. Evaluate heat effects at early time periods; estimate the thermal, hydrologic, and geochemical environment of the repository at early times.

3. Evaluate safety and reliability value added by NRC subsystem requirements beyond total system requirements of EPA (some work of this type is discussed in Section 9.4).

4. Estimate health effects from releases to evaluate adequacy of 40 CFR 191.

5. Evaluate importance of thermally and barometrically driven air flow on repository performance at Yucca Mountain.

6. Perform detailed hydrologic analysis for Yucca Mountain, to provide a better input to the transport analysis and to examine, in more detail, various alternative hypotheses regarding hydrology at Yucca Mountain.

Recommendations for additional scientific input (some of these items could be performed by either the DOE or NRC, while others are clearly the responsibility of DOE):

1. Develop and demonstrate a mathematically rigorous, scientifically robust method for scenario analysis.

2. Obtain geoscience input for modeling vulcanism.

3. Obtain geoscience and hydrologic input to modeling faulting, uplift, and subsidence at Yucca Mountain.

4. Obtain laboratory chemical analysis to determine the partitioning of radionuclides in various compartments of the spent fuel waste form.

5. Obtain field and laboratory data on phenomena important to the near-field behavior of the repository, especially the effects of heat.

6. Obtain more data on plutonium geochemistry.

7. Obtain a better understanding of waste package corrosion in the unsaturated zone.

SECTIONS 1 & 2

1. INTRODUCTION

2. Purpose and Scope

5520 NAME : M001-2

1. INTRODUCTION

This report describes the results of the performance assessment (PA) activities carried out as Tasks 2 and 3 of Phase 1 under the NMSS/RES Memorandum of Understanding (MOU) of September 1, 1988. Plans for this work are described in:

- The memorandum of December 9, 1988, which implements the MOU
- The Detailed Program Plan for Tasks 2 and 3 (January 31, 1989), which describes in greater detail the work to be performed under various subtasks, how the various subtasks relate to each other, the schedule for that work, and the individuals responsible for the work
- "Jump Starting the MOU," a memorandum, dated August 4, 1989, from Eisenberg and Randall to Ballard and Silberberg, reconfiguring the work on the MOU to fit into a three-month completion schedule
- "Scope of Phase 1 Performance Assessment Demonstration," dated September 1, 1989, from Ballard to Browning.

Task 1 of the MOU is work pursuant to the international INTRAVAL study, which is not discussed in this report. The purpose of Task 3 of the MOU activities is to perform a total system performance assessment for the proposed Yucca Mountain Repository, and by doing so, to extend the NRC capability to model repository performance pursuant to the regulatory review of the Yucca Mountain Project. Task 2, the source term modeling effort, is broken out as a separate activity, but is an essential part of the overall PA activities in Task 3; therefore, Tasks 2 and 3 were treated together except for the purposes of making work breakdown schedules and personnel assignments.

The September 1, 1988 MOU describes the three Tasks comprising the MOU activities in broad outline. The December 9, 1988 implementing memorandum describes the various subtasks, persons assigned to various subtasks, and staff time commitments. The Detailed Program Plan provided more detail about these matters and how the work is envisioned to proceed. Task 2 and 3 MOU activities, are to proceed in two phases: Phase 1, was intended to: (1) to result in a framework for PA modeling; (2) with the limited resource allocated to perform this activity, provide only a rudimentary demonstration of a PA modeling capability; (3) be accomplished with a minimum of technical input and interaction with NRC contractors, except for work already documented and products delivered to the NRC. Phase 2, is intended to: (1) be accomplished in FY 90 and beyond; (2) incorporate significant products to be delivered by NRC contractors, most notably the Tuff Performance Assessment Methodology currently under development by Sandia National Laboratories under FIN-A1266 (3) provide a more complete, accurate, sophisticated, and realistic PA modeling capability.

An interdisciplinary, integrated approach was envisioned when the initial plans for this activity were developed. Although some work was continued by some staff for a time, sustained effort by several staff on the MOU Tasks 2 and 3 did not resume until August/September 1989. During that time period the two memoranda cited above were issued to restructure the MOU Tasks 2 and 3 effort. The major features of this restructuring included:

- o Conclusion of the Phase 1 work in three months, no later than November 30, 1989.

- o Attempting to execute as many steps in the performance assessment methodology, while at the same time tailoring the activities to fit into the time and resources allowed.

- o Establishing a smaller core group of MOU participants to be responsible for the work. The involvement of other staff and continual peer review as originally envisioned in late 1988, would be deferred until after November 30, 1989, to expedite the effort.

- o The work would be divided into five parts:

  1. Scenario Analysis
  2. Flow and Transport
  3. Source Term
  4. System Code
  5. Auxiliary Analyses

The first four topical ares corresponded to four working groups or Teams. These Teams roughly correspond to the methodological steps of performance assessment shown in Figure 1-1. The members and leaders of these teams and other details of the project organization are discussed in Section 3.

2. PURPOSE AND SCOPE.

The primary purpose of Phase 1 of the MOU Tasks 2 and 3 was to demonstrate the capability of the NRC staff to conduct, independently, a performance assessment of a proposed repository. An independent assessment capability is considered to be an important aspect of the licensing review to be conducted by the NRC staff. In order to achieve these goals a limited, preliminary total system performance was conducted.

The performance assessment is considered to be comprised of two parts:

(1) quantitative estimation of total system performance through the use of predictive models and

(2) documentation, including detailed auxiliary analyses where appropriate, to support the assumptions, data, and modeling approaches used to obtain quantitative estimates of performance.

Both of these aspects of the performance assessment were addressed in the Phase 1 effort.

By accomplishing this primary goal, some worthwhile secondary goals were achieved:

- o The existing analytical tools to conduct a performance assessment (both methodologies and computer codes) were evaluated
- o Insight was obtained into the needs for the development or improvement of methodologies
- o Insight into the needs for site characterization was obtained.

The total system performance measure for a high level waste repository can be expressed by a complementary cumulative distribution function (CCDF) of radionuclide releases to the accessible environment, weighted by a factor approximately proportional to radiotoxicity, integrated over an appropriate period of time (10,000 years is the current regulatory requirement). This performance measure is madated by the EPA standard (40 CFR 191) for the containment of waste by a HLW repository. This performance measure is incorporated into the NRC's regulation (10 CFR 60), along with additional performance measure relating to (1) waste package lifetime, (2) fractional release of radionuclides from the engineered barrier system, and (3) ground water travel time. The representation of repository performance by a CCDF of weighted cumulative releases incorporates (a) consideration of the various components impeding the movement of radionuclides to the environment and (b) consideration of a range of conditions and events that could affect future performance . This performance measure is estimated by following the steps outlined in the information flow diagram (Figure 1.1). For the Phase 1 effort, these steps were all executed, but some (for example steps 2 and 3) were only executed to a limited degree and only parts of others (for example step 5) were done. These steps are described briefly below for the Phase 1 effort:

1. System Description - The repository is broken into its component parts for the purposes of modeling. These include the source term model and the flow and transport model. Computer codes are adapted or written to simulate models of these components. Ranges of parameter values are chosen to bound the expected behavior of the system models.

2. Scenario Analysis - Scenarios representing alternative futures for the system and possible future states of the environment are screened and chosen. Probabilities are estimated for chosen scenarios.

3. Consequence Analysis - Consistent with the requirements of the EPA standard, the consequence in terms of cumulative release of radionuclides to the accessible environment over a specified time period (usually 10,000 or more years) is calculated for each

scenario and usually numerous realizations of possible parameter values. In addition to being incorporated by way of cumulative releases into the CCDF (step 4), certain types of consequences might also be considered separately to compare to standards for maximum doses to individuals and for maximum concentration in groundwater (but are beyong the scope of Phase 1). For purposes of dividing up the work, the consequence analysis was conducted by the Source Term Team and the Flow and Transport Team.

4. Performance Measure Calculation (CCDF) - The consequences for each scenario, in terms of normalized cumulative releases of radionuclides to the environment over a specified period of time, are calculated and the results are displayed in a curve of consequences versus the probability that such consequences will not be exceeded. Compliance with the performance criteria is determined by comparing the curve to a compliance curve, that provides limits that the calculated the curve must not exceed.

5. Sensitivity and Uncertainty Analysis - Sensitivity analysis investigates the change in performance measures caused by incremental changes in the values of input parameters and data. Uncertainty analysis attempts to quantify the uncertainty in performance estimates in terms of the major sources of uncertainty, including uncertainty in input parameters, uncertainty in modeling (both the conceptual model of the geometry and characterization of the system and the process model of what physiochemical processes occur and how they are manifested), and uncertainty about future states-of-nature. Uncertainty in modeling was not quantified in Phase 1.

6. Documentation - A largely self explanatory effort documentation must make the assumptions used in the analysis, their basis, and the implications of their use explicit and clear.

Two types of uncertainty are usually treated explicitly in the generation of the CCDF: (1) uncertainty due to future states of nature and (2) uncertainty in the values of parameters determining system performance. In a safety analysis for a more conventional type of system, the response of the system to any single future state of nature to be considered would be a single-valued estimate of system performance (in the parlance of the repository system, a single value of consequence). System performance would then be described by the plot of consequences versus the likelihood of the future state of nature (scenario) producing that consequence; such a curve would be the distribution function. The integral of such a curve over probability would yield a cumulative distribution function; i.e. the likelihood that the consequence would be at least of a certain magnitude. The CCDF would be the curve of the likelihood that the consequence would be a certain magnitude or less. For the repository system considerable uncertainty exists concerning the values of parameters used to estimate the consequences of the repository. Traditionally the uncertainty from this source is also displayed on the CCDF by: (1)describing some or all of the parameters used to estimate consequences as distributions of values rather than point estimates, (2)choosing a value of each parameter required to describe system performance from these distributions representative of some portion of the various distributions, (3)estimating performance based on a given realization of parametric values, (4)noting the conditional parametric probability, i.e. the joint probability density for the given realization or region of parameter space (for uncorrelated parameters this would be the product of the individual parameter probabilities), (5)calculating the CCDF using the parametric probability multiplied by the probability of the scenario. This process is complicated further when consideration of different scenarios makes it is necessary: (1) to vary the consequence models for different scenarios, (2) to vary the distributions of parameters (either the range of parameters, the magnitude of the parameters, or the shape of the distribution) depending on the scenarios.

Because of the complexity of the calculation of the CCDF the authors thought it appropriate, but not absolutely necessary, that the generation of the CCDF be performed with the aid of a computer code. At a minimum such a code would need to: (1) sequence through all the scenarios to be considered, (2) choose the consequence models and parametric distributions corresponding to the scenario being analyzed, (3) sample the parameter space appropriate to the given scenario, (4) estimate consequences based on the models and parameter values for the scenario, and (5) combine the parametric and scenario probabilities and the calculated consequences to generate a CCDF.

The primary goal of Task 2 is to provide a simplified radionuclide source term in the form of a table or a computer code, to the overall system performance activities in Task 3. The goal of Task 3 is to conduct a preliminary performance assessment of the high level waste repository at Yucca Mountain, Nevada.

As explained in Section 1, only a rudimentary performance assessment is intended for Phase 1 of the MOU, because of limited resources and time and because input from NRC contractors, that could contribute to the goals of the MOU, is not currently available. Because of the constraints on this activity the scope of the effort was limited; some of these limitations were:

- only a preliminary analysis was intended in Phase 1
- the effort was scaled down from the original 1/89 plan for this work
- only currently available modeling tools were to be used; computer code development was to be minimized
- the analysts were to take advantage of the limited data available for the Yucca Mountain Site

- the scope of the analyses were constrained by the time and resources made available to do it

- As many components of the methodology as possible were to be executed, given the limited time and resources available; this required reducing the depth to which certain aspects were demonstrated.

- For the Phase 1 effort the EPA containment standard was to be the only performance standard considered. The EPA standards for individual protection and groundwater protection will be investigated later. Also the 10 CFR Part 60.113 subsystem requirements were not to be a subject of the Phase 1 work and perhaps not included in the Phase 2 work.

- Phase 1 was executed by NRC staff only.

- Other than existing reports, papers, and computer software packages already delivered, no contractor input was available for Phase 1, except infrequent and short personal communication.

- CNWRA involvement in Phase 1 was primarily as an observer, but would become more active as the CNWRA PA capability expands.

In order to perform this preliminary performance assessment and demonstrate the staff capability to conduct such work, the following types of activities were performed:

1. Computations & Support

   data input
   model setup
   code development & testing
   code execution
   output analysis

2. Auxiliary analyses

   evaluation of assumptions
   preprocessing raw data

3. Documentation

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section 3

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SECTION 3.0

ORGANIZATION AND STAFFING

3.0 ORGANIZATION AND STAFFING OF PHASE 1 OF THE MOU

NRC staff members from both NMSS and RES worked on Phase 1. In order to coordinate the efforts of the two Offices, the organizers of the MOU effort designated an administrative project manager from NMSS and two Office technical coordinators: one from NMSS and one from RES. The technical staff involved in Phase 1 came from both Offices. The assignment of technical staff to technical efforts in Phase 1 was done without regard to Office affiliation.

3.1 Administration of Phase 1 of the MOU

Brian Thomas of NMSS/HLPD was Phase 1's administrative project manager. He scheduled meetings, called meetings recommended by the technical staff, kept the notes of meetings, and kept records of outstanding technical disagreements until the technical staff resolved them. Norman Eisenberg and John Randall, respectively of NMSS/HLPD and RES/DE/WMB, were the Office technical coordinators for Phase 1. They recommended to their management which technical staff members of both Offices to assign to particular efforts in Phase 1. Richard Codell of NMSS/HLPD helped them in making the recommendations.

3.2 Evolution of the Definition of Technical Work in the MOU

Definition of work in the MOU began in the Summer of 1988, following an NRC managerial decision to end several years of HLW performance assessment technical assistance and research work at Sandia National Laboratories by the end of FY 90. The decision to terminate NRC-supported HLW work at Sandia is consistent with NRC's policy to eliminate a potential conflict of interest by terminating its HLW work at all of the National Laboratories operated by DOE, the HLW licensee.

Codell and Randall formulated a plan for the NRC waste management staffs in NMSS and RES to acquire Sandia's HLW performance assessment methodology while Sandia continued to finish development of a methodology for assessment of HLW repository in unsaturated welded tuff. Their recommended course of action became the core of the September 1, 1988 MOU. Briefly, the organizational structure recommended in the September 1, 1988 MOU consisted of three tasks: 1) INTRAVAL; 2) source term modeling (which is not part of the Sandia methodology); and 3) acquisition of Sandia's methodology, consisting mainly of computer implementations of groundwater flow and radionuclide geosphere transport models. Tasks 2 and 3, the subject of this report, were to consist of two phases, as described in Section 1 of this report.

There was no recommended staffing in the original MOU. Codell and Randall, joined by Eisenberg, prepared two memoranda in December 1988 and January 1989 on the implementation and staffing of the MOU. In preparing the two memoranda, they decided to change the topic of Task 3 to total system performance modeling to reflect that Phase 1 consisted mainly of NRC staff efforts without contractor help.

Work on the MOU began in January 1989 but the requirement that the NRC waste management staff review and comment on DOE's Yucca Mountain, NV Site Characterization Plan interrupted the MOU work. Although the review of the SCP was in one respect an interruption of the MOU, it also helped the MOU by giving the NRC waste management staff a chance to learn more about DOE's HLW performance assessment plans and plans for obtaining data to support performance assessment. By the Summer of 1989, the NRC staff had finished the SCP review and Codell, Randall, and Eisenberg prepared plans for an accelerating and compressing work on Phase 1 of the MOU so that the NRC waste management staff could finish it before 1990. On August 4, 1989, Eisenberg and Randall issued a memorandum called "Jump Starting the MOU," which recommended a plan for accelerating and compressing the MOU's Phase 1 work. Ronald Ballard and Mel Silberberg, respectively the Chiefs of NMSS/HLGP and RES/DE/WMB, reviewed the plan and modified it. The modified plan appeared in a September 1, 1989 memorandum from Ballard to Robert Browning, Director of NMSS/DHLWM. The technical organization of the accelerated and compressed MOU Phase 1 technical work follows the outline given in Ballard's memorandum.

3.3 Technical Organization of Phase 1 of the MOU

The Ballard memorandum concerned Phase 1 of Tasks 2 and 3 of the MOU and set up five technical efforts with associated staff assignments, as listed below.

System Integration: Eisenberg (technical leader) (30% of his time), Park (30%)

Source Term: Codell (technical leader) (30%), Mo (20%), Chang (30%), Park (30%), Peterson (not designated in Ballard memo)

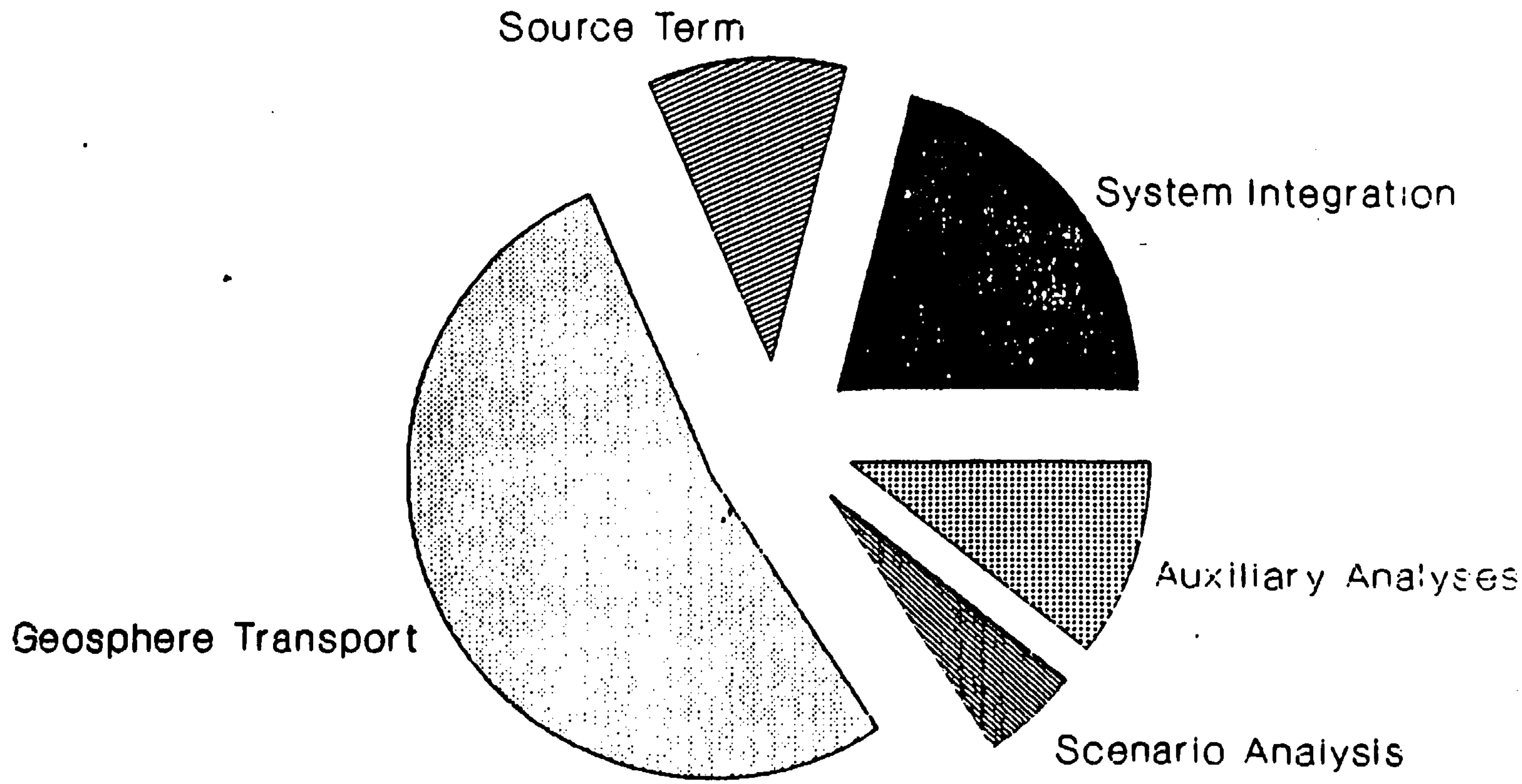
Geosphere Transport: McCartin (technical leader) (50%), Margulies (65%), Park (30%), Codell (60%), Pohle (20%), Ford (not designated in Ballard memo), Bradbury (not designated in Ballard memo), Eisenberg (10%), Fehringer (10%)

Scenario Analysis: Fehringer (technical leader) (30%), Trapp (designated, but no percentage specified, in Ballard memo), Eisenberg (20%), Pohle (10%)

Auxiliary Analyses: McCartin (two-dimensional transport) (15%), Ford (analysis of hydrogeolgic data) (not designated in Ballard memo), Bradbury (analysis of geochemical data) (50%), Codell (gas transport and sensitivity and uncertainty analysis) (not designated in Ballard memo), Margulies (volcanism) (not designated in Ballard memo)

Figure 3.1 shows roughly the relative time spent on each of the above efforts in the actual execution of Phase 1. The time spent on one effort, scenario analysis, was less than that assigned because the technical leader had to spend time on higher-priority work dealing with the EPA HLW standard. However, the time spent on all other efforts was much greater than that assigned and shows the amount of time that analysts need for an effort such as an HLW performance assessment. Such an effort is typically time- and labor-intensive. The reader should note that several of the contributors to the above efforts spent time that the Ballard memo did not assign. Furthermore, there was a general shortfall of staff resources for the Phase 1 work that was offset by condensed work being done beyond normal working hours. Plans for the Phase 2 work need to provide a better match between the work that has to be done and the staff resources needed to do it.

Relative Time Spent
on Technical Efforts in Phase I



* **The total FTE's spent far exceeded the estimates given in the 9/1/89 Ballard to Browning memo.**

Figure 3.1

Section 4.0

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MOU CHAPTER ON SYSTEM CODE

SECTION 4.0

SYSTEM CODE

4.0 SYSTEM CODE

4.1 Introduction

The system code plays a central role in processing information needed to generate a Complementary Cumulative Distribution Function (CCDF) representative of the performance of a HLW repository at Yucca Mountain, Nevada. In order to obtain the CCDF, the code treats sequentially a set of scenarios, which represent possible future states of nature at the site. Consequence modules associated with the available release pathways calculate the cumulative radionuclide release for each scenario for the 10,000 year simulation time. These modules are products of work performed on the Source Term and Flow and Transport Tasks, which are documented elsewhere in this report. Each scenario may yield numerous cumulative release values, which result from the multiple input vectors of parameters used in an iteration. Probabilities assigned to each consequence within each scenario are then combined with the likelihoods of the scenarios themselves to form the CCDF.

In accomplishing these tasks, the system code handles two types of uncertainty inherant in a CCDF. First, it treats the uncertainty in the future states of nature by looking at sets of scenarios which attempt to describe those future states. Secondly, the code handles the uncertainty related to the variability in model parameters by using multiple sets of parametric input vectors when executing the pathway consequence modules.

4.2 Requirements for the PA Total System Code

The development of the NRC system code is a continuing process, consistent with the ongoing iterative performance assessment activity. Throughout its development, this code should meet certain minimum requirements:

1. The computational modules for calculating consequences, comprised of one or more codes for the source term and transport calculations, produce output in terms of cumulative radionuclide release to the environment. The system code must be capable of receiving this data.

2. The system code must be able to treat two of the types of uncertainty incorporated in a CCDF characterizing repository performance: (1) the uncertainty in future states of nature, and (2) the uncertainty in model parameters used to estimate cumulative releases.

3. In order to treat uncertainty in future states of nature properly, the system code must be able to treat different scenarios (or more properly scenario classes) which attempt to describe those future states and obtain the corresponding data on cumulative releases of radionuclides.

4. In order to treat properly the uncertainty related to the variability of parameters used in the consequence models, the system code must be able to collect and process cumulative release data generated from multiple sets of parametric input vectors.

5. Because many scenarios are expected to allow radionuclide releases by more than one pathway (e.g. in groundwater, by gas, and/or by direct release), the system code must be able to obtain cumulative releases corresponding to the specified pathways.

[Requirements 2 through 5 mandate that the system code will be handling a four-dimensional array of cumulative release estimates, where the dimensions are: scenario number, radionuclide number, pathway number, and input parameter vector number.]

6. The system code should have built-in protection to assure the consistency of the assumptions used within a single simulation. For example, the performance time period (10,000 years for the current EPA standard) should be the same for all scenarios and pathways in any given representation of the repository to which the system code is applied. One way to assure consistency would be to have the system code call the consequence modules as subroutines of the main program. A second method would be to have the consequence results generated outside the system code and stored in a file. This file would include a sufficient record of the critical assumptions and parameters to permit a consistency check. Note: It is not clear presently whether the consequence modules can be called as subroutines by the system code and still be practicable in terms of program size and run-time. The NRC system code allows both methods of operation, but only the latter has been tested.

7. Tabular and graphical presentations of the results should be obtainable from the system code.

4.3 Survey of Existing PA Total System Codes

The staff evaluated several codes to determine their suitability (as a whole or in part) for use as a system program in the MOU demonstration. Although all the surveyed codes are not "total system codes" per se, each was reviewed in terms of how well it fit the requirements expressed in Section 4.2. The codes are described briefly in Table 4.1, while Appendix 1.1 provides a more detailed look.

Based on the results of the review, the staff decided to develop its own system code rather than to adopt an existing one. There were several reasons for this choice. First, adapting an existing program to meet the staff's needs and to be compatible with the NRC computing environment would likely be as time consuming as development of a new code. Secondly, an NRC written code could be more closely tailored to the specific requirements and needs of the project than one developed outside the NRC. Finally, the more promising system codes for potential use in this work would not be available to the staff within the timeframe set.

4.4 Description of the NRC PA Total System Code

4.4.1 Introduction

This section presents a brief description of the system code developed by the staff for this phase of the MOU. The manner of code execution (i.e. internal vs. external), the input data requirements, the type of output available, and a brief outline of the system program are all presented.

4.4.2 Internal vs. External Runs

The system code can be executed in either the "internal" or the "external" mode (Figure 4.1). This distinction refers to the time at which the output files from the consequence models are generated. In the internal mode, consequence modules are run and cumulative radionuclide releases calculated as the code is executed. This requires that the modules be incorporated as subroutines in the main program. For external runs however, the modules are separate from the system code, and as a result, the cumulative releases can be generated and placed in files at any time prior to iteration of the code.

Internal executions would appear to make sensitivity analyses easier, because simulation parameters are global. Thus changes to the input files for subsequent runs need only be made once. This decreases the opportunity for error, while offering increased convenience and quality assurance to the analyst.

Simulations in the external mode offer the opportunity to repeat earlier runs as long as the output files from the consequence modules are uniquely identifiable. In addition, external runs would appear to be more economical in terms of both computer time and money since they do not require the execution of either the LHS routine or the consequence models.

Note: As yet, the system code has been demonstrated only in the external mode.

4.4.3 Input to the System Code

The system program requires input data in the following five areas:

1) general run information (e.g. execution mode, time period, etc.)
2) the particular scenarios to be considered,
3) probabilities of those scenarios occurring,
4) EPA limits for the initial radionuclide inventory, and
5) cumulative releases due to the effects of the scenarios.

The analyst creates a file called SYS.INP, which consists of both the general run data and the scenario-specific information. This file supplies the execution mode, the simulation time period, and the amount of output desired, as well as the scenarios (total number, names, release pathways) to consider.

A scenario's probability is estimated by combining the probabilities of the processes and events making up the scenario. For the MOU demonstration, the staff modeled four scenario classes based upon two fundamental events: a pluvial period (or not) and drilling at the site (or not). Figure 4.2 shows the probabilities assigned to the events and scenarios.

The EPA limits are taken from 40 CFR Part 191 Appendix A Table 1. Given in curies released per 1000 Metric Tons of Heavy Metal (MTHM), these limits are converted in the system code to limits for the initial inventory of 70,000 MTHM assumed for the MOU demonstration. EPA ratios are calculated, using these limits, for each released radionuclide.

The cumulative releases of radionuclides are calculated by the consequence modules, which model the repository release via the pathways assigned per scenario.

4.4.4 System Code Operation

In order to obtain a Complementary Cumulative Distribution Function (CCDF) representative of repository performance, the system code treats a set of scenarios describing possible future states of nature at Yucca Mountain, and accesses the estimated cumulative releases corresponding to each scenario. The code next combines this data from the scenarios into the CCDF, and finally it prints the CCDF out in the form of a graph and/or a table. This section, along with Figures 4.3 and 4.4, provides a more detailed explanation of how the system code accomplishes these tasks.

The first step in the execution of the system code is to read into the program the simulation-specific information from the SYS.INP file. Each scenario class identified in SYS.INP will be considered in sequence using the data provided. Before this begins however, parametric input vectors for the pathway consequence modules are generated via the Latin Hypercube Sampling (LHS) routine, if the run is internal.

The effects of each scenario are then assessed in the following manner. Consequence modules for the potential release pathways specified for a scenario are executed if the simulation is run in the internal mode. Next, the cumulative releases calculated by the modules either internally or externally are read into the program and stored in temporary arrays. Each nuclide-release pair is compared to its EPA limit and a corresponding normalized EPA ratio calculated by the following formula:

$$\text{Normalized Release of Radionuclide } i = \frac{\text{Cumulative Release of Radionuclide } i}{\text{EPA Limit for Radionuclide } i}$$

These normalized releases are then placed into a four-dimensional array arranged by scenario, radionuclide, vector, and release pathway (Figure 4.5). Once the effects of all scenarios have been modeled, this array is used as a data base over which different summation routines take place. These routines create a second array of summed normalized EPA releases ordered by scenario and vector by adding up normalized releases for all radionuclides over all release pathways.

Then, for each scenario, probabilities are calculated for the consequences associated with a particular input vector. These likelihoods are based on the assumption that every vector within the scenario is equally probable. For example, given this assumption, the likelihood of occurrence of a single vector within a scenario containing 500 vectors is equal to 1/500 or .002. Following the assignment of probabilities, the consequences within each scenario class are sorted, duplicates eliminated, and the likelihoods adjusted accordingly.

The array for each scenario now contains unique, ordered consequences with associated likelihoods of occurrence. Then, in order to obtain a cumulative distribution function representative of repository performance, scenario probabilities are factored in. This is done by multiplying the probability of each consequence by the likelihood of its scenario.

The final program steps combine the results from all scenarios considered: the summed normalized releases and their probabilities are ordered and sorted, and a running sum of the probabilities is created. This outcome can be graphed as a CCDF on a log-log plot of summed normalized EPA release against cumulative probability.

4.4.5 System Code Output

Results generated by the system code can be written to two output files. In addition to the data needed to graph the total CCDF, these files can contain normalized releases broken down by scenario, vector, release pathway, and radionuclide, or various combinations of these categories.

Plotting the CCDF can be accomplished using any of the variety of graphics packages currently available. The GRAPHER software package was used to plot the graphs presented in this report.

Table 4.1 System Code Survey

1.	AREST	EBS code; partially documented by PNL; code not available
2.	SPARTAN	DOE total system code; oversimplified flow and transport; does not treat radionuclide chains; documented by SNL and DOE; code not available
3.	TOSPAC	DOE total system code; documented by SNL; code not available
4.	REPRISK	EPA total system code; considers four scenario classes; developed for saturated porous media; calculates EPA ratios and health effects; code and documentation available as of 10/89
5.	SUNS	SNL sensitivity and uncertainty analysis shell; interactive; code and limited documentation available
6.	Code Coupler	Provides linkage between different scale models in a total PA; designed for set suite of models, including NEFTRAN; LHS used to create common site description for all models; code and documentation available as of 11/89

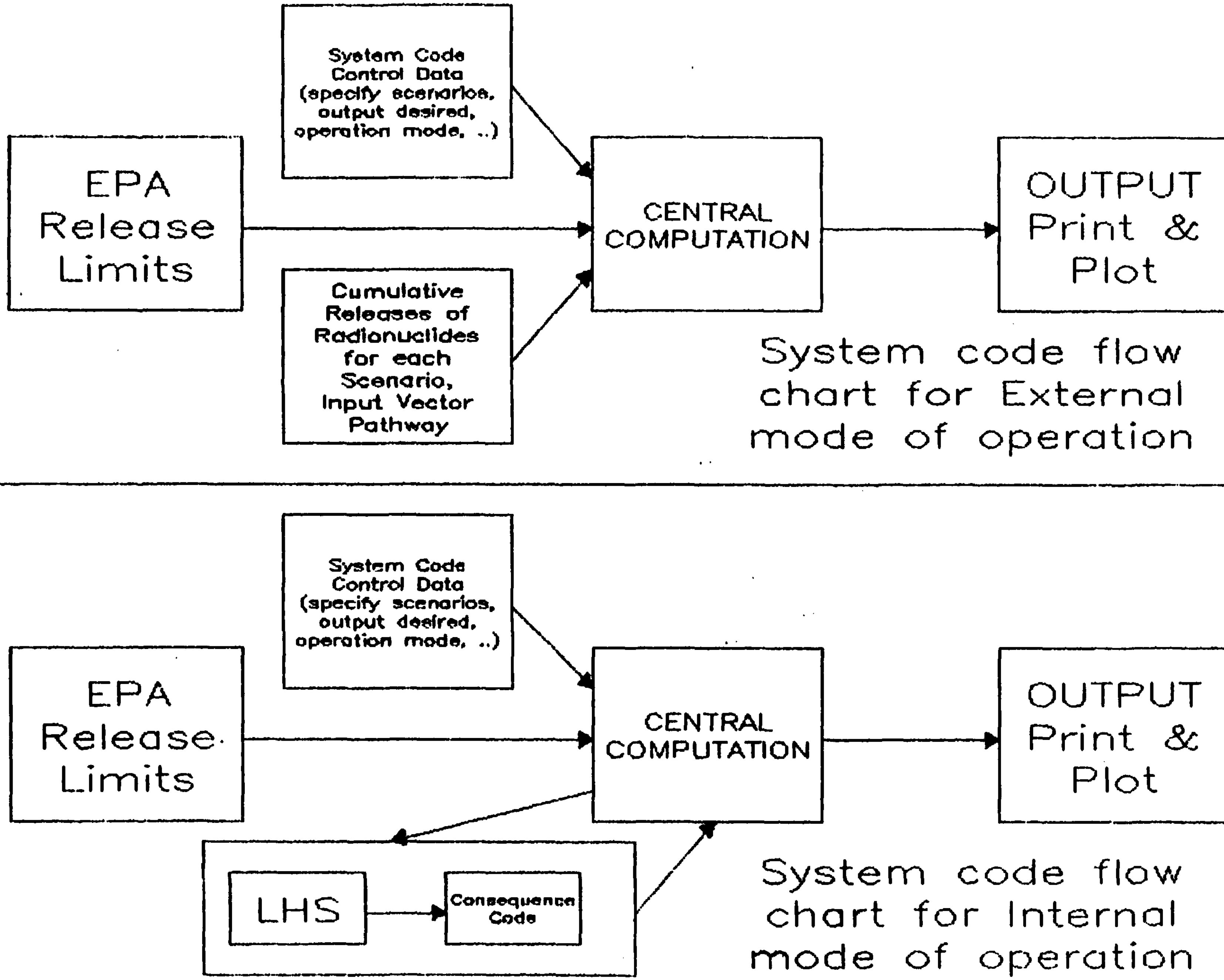

System Code Control Data (specify scenarios, output desired, operation mode, ..)
EPA Release Limits
CENTRAL COMPUTATION
OUTPUT Print & Plot
Cumulative Releases of Radionuclides for each Scenario, Input Vector Pathway
System code flow chart for External mode of operation
System Code Control Data (specify scenarios, output desired, operation mode, ..)
EPA Release Limits
CENTRAL COMPUTATION
OUTPUT Print & Plot
LHS
Consequence Code
System code flow chart for Internal mode of operation

Figure 4.1

DETERMINATION OF SCENARIO PROBABILITIES
FROM THE PROBABILITIES OF FUNDAMENTAL EVENTS

	$\bar{P}$ 0.9	P 0.1
$\bar{D}$ 2.3×10^{-7}	scenario class # 0 probability $= 2.0 \times 10^{-7}$	scenario class # 1 probability $= 2.3 \times 10^{-8}$
D ~ 1.0	scenario class # 2 probability ~ 0.9	scenario class # 3 probability ~ 0.1

$\bar{P}$ is not pluvial

P is pluvial

$\bar{D}$ is no drilling

D is drilling

scenario class # 0 is no drilling, not pluvial

scenario class # 1 is no drilling, with pluvial

scenario class # 2 is drilling, not pluvial

scenario class # 3 is drilling and pluvial

Note: Probability combinations assume that fundamental events have independent probabilities of occurence; this is not a general restriction.

Figure 4.2

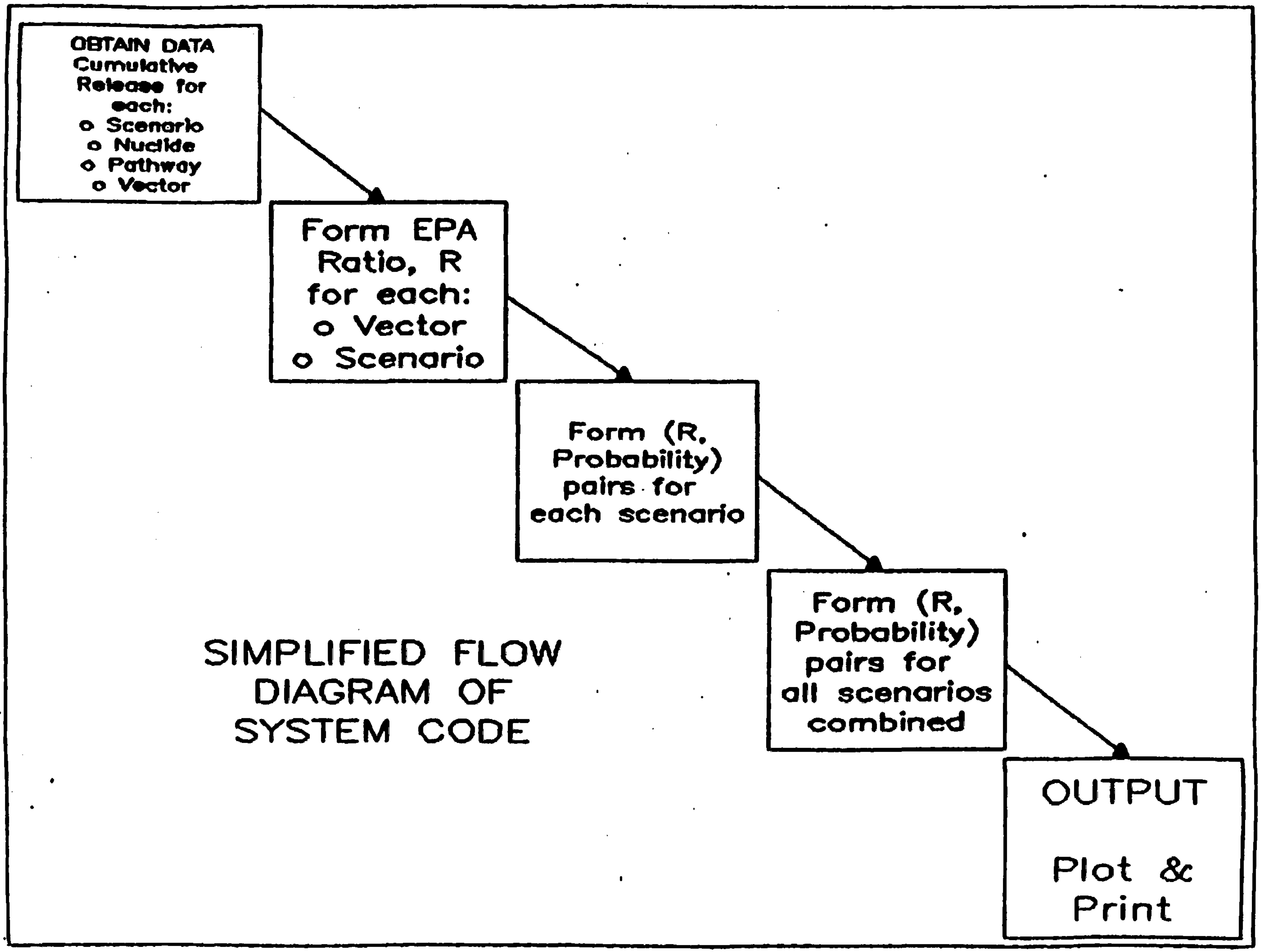

OBTAIN DATA
Cumulative Release for each:
o Scenario
o Nuclide
o Pathway
o Vector
Form EPA Ratio, R for each:
o Vector
o Scenario
Form (R, Probability) pairs for each scenario
Form (R, Probability) pairs for all scenarios combined
OUTPUT
Plot & Print
SIMPLIFIED FLOW DIAGRAM OF SYSTEM CODE

Figure 4.3

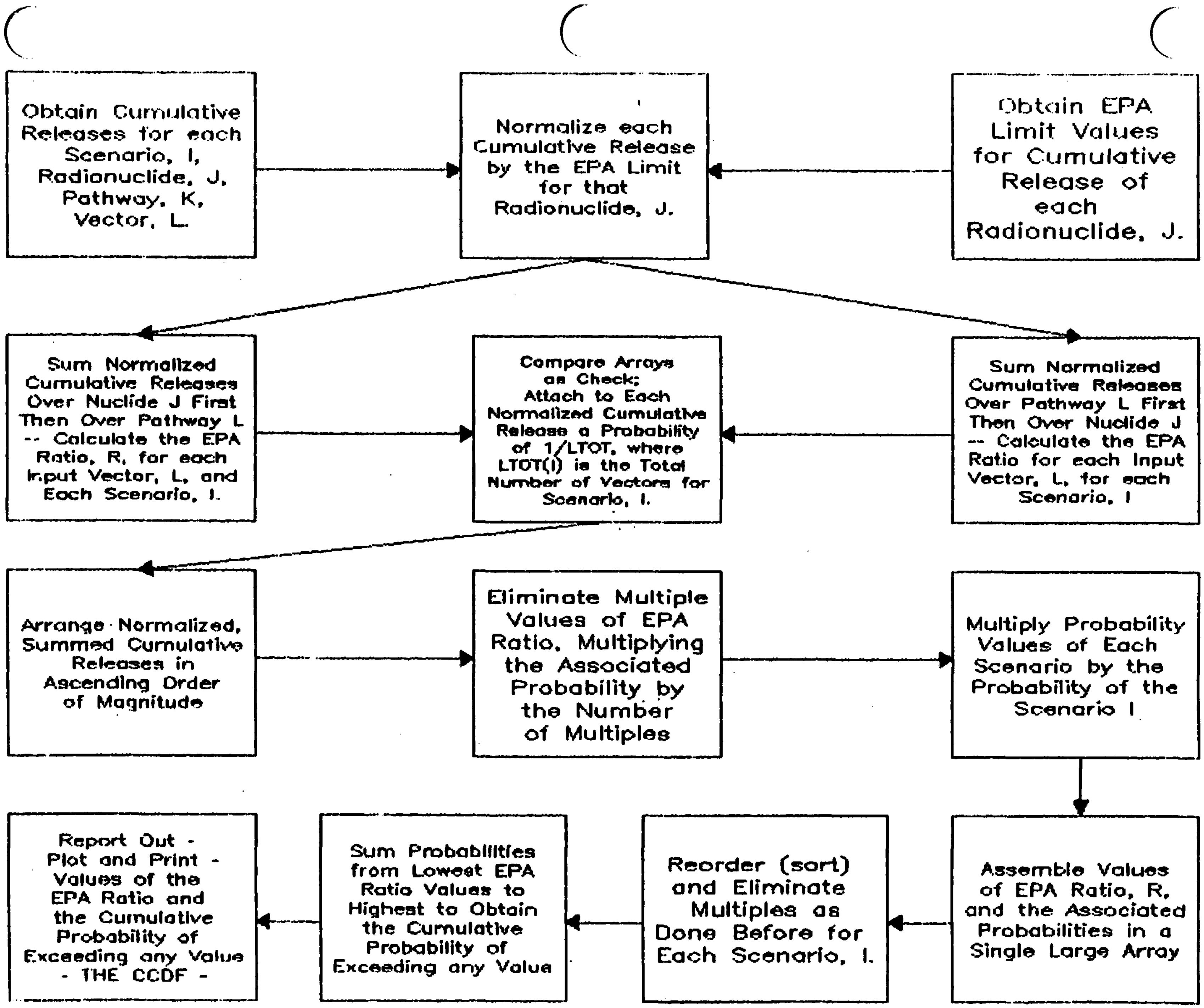
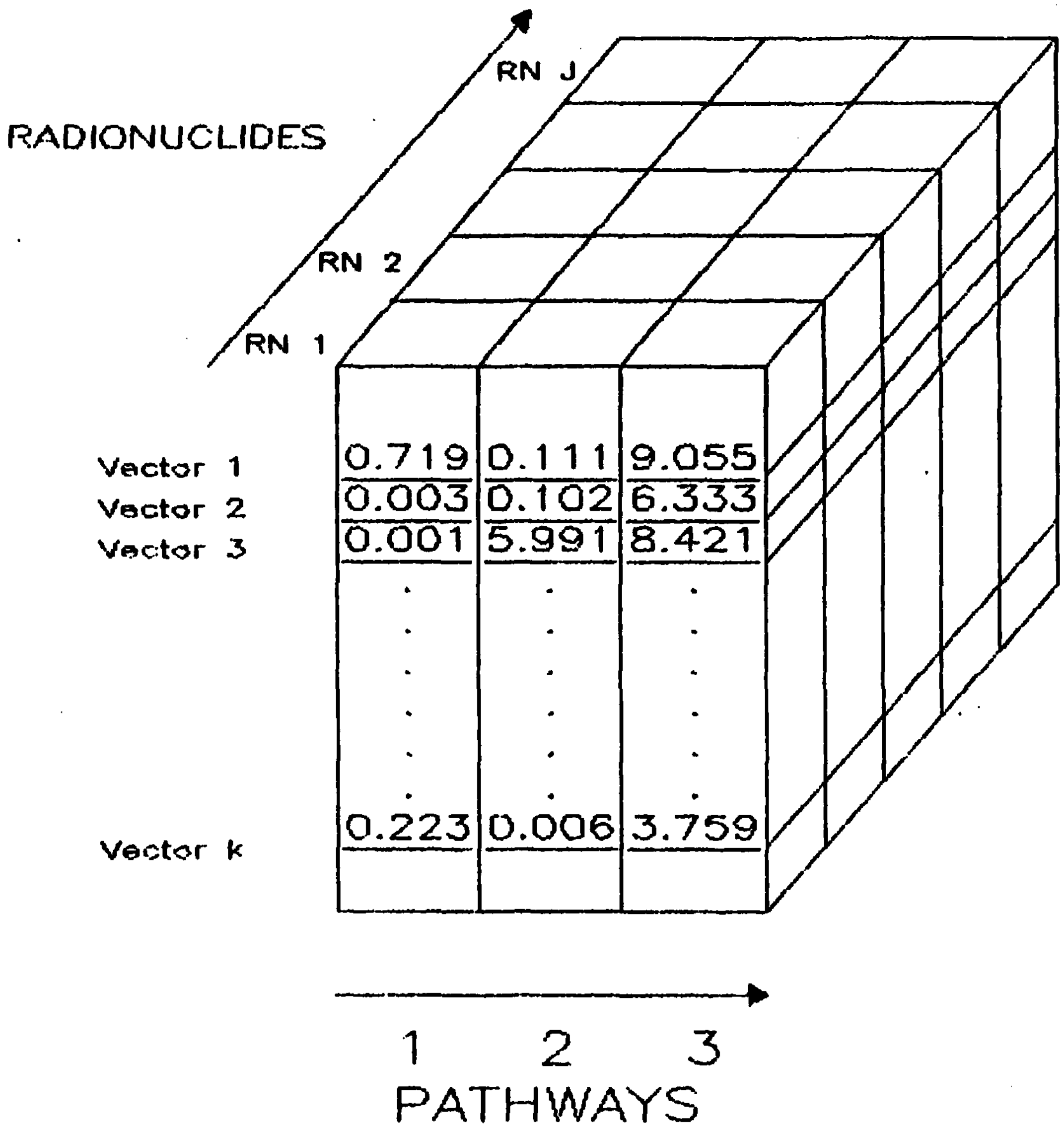

Obtain Cumulative Releases for each Scenario, I, Radionuclide, J, Pathway, K, Vector, L.
Normalize each Cumulative Release by the EPA Limit for that Radionuclide, J.
Obtain EPA Limit Values for Cumulative Release of each Radionuclide, J.
Sum Normalized Cumulative Releases Over Nuclide J First Then Over Pathway L -- Calculate the EPA Ratio, R, for each Input Vector, L, and Each Scenario, I.
Compare Arrays as Check; Attach to Each Normalized Cumulative Release a Probability of 1/LTOT, where LTOT(I) is the Total Number of Vectors for Scenario, I.
Sum Normalized Cumulative Releases Over Pathway L First Then Over Nuclide J -- Calculate the EPA Ratio for each Input Vector, L, for each Scenario, I
Arrange Normalized, Summed Cumulative Releases in Ascending Order of Magnitude
Eliminate Multiple Values of EPA Ratio, Multiplying the Associated Probability by the Number of Multiples
Multiply Probability Values of Each Scenario by the Probability of the Scenario I
Assemble Values of EPA Ratio, R, and the Associated Probabilities in a Single Large Array
Reorder (sort) and Eliminate Multiples as Done Before for Each Scenario, I.
Sum Probabilities from Lowest EPA Ratio Values to Highest to Obtain the Cumulative Probability of Exceeding any Value
Report Out - Plot and Print - Values of the EPA Ratio and the Cumulative Probability of Exceeding any Value - THE CCDF -

Figure 4.4

DATA ARRAY FOR SYSTEM CODE INPUT



Array for
Scenario # 0

Figure 4.5

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MCCARTIN

Author's Name:
codell

Document Comments:
chapter 5 on source term

5.0

5.0 Source Term

5.1 Introduction

The demonstration of the performance assessment methodology depends on developing or adopting a source term model that considers the rate of release of the radionuclides from the engineered barrier system for the Yucca Mountain repository. The Staff has reviewed several assessments of the Yucca Mountain site performed for DOE by several national laboratories. The Staff has also reviewed other source term models not developed for the Yucca Mountain case. A synopsis of our reviews is given in Appendix B. None of these models is fully satisfactory because important data on actual spent fuel under expected repository conditions are not yet available.

The staff's model draws on the features found in these assessments. In many cases, the Staff has found it necessary to make simplifying assumptions. These assumptions are believed to lean on the side of conservatism.

5.2 Review ~~DEFINITION~~ OF IMPORTANT ISSUES FOR SELECTING SOURCE TERM MODELS

The radioactive waste, consisting mainly of spent light water reactor fuel will be stored in metal canisters. A typical canister according to current DOE plans is about 4.5 meters long, 0.5 meters in diameter and have a wall thickness of 1 cm (SCP, section). Small amounts of nuclear wastes in other forms will also be stored in the repository such as vitrified defense wastes, but the present study will focus only on the spent fuel wastes. The source term model must account for the processes in the near field that determine the rate at which radionuclides are released, including corrosion and physical destruction of the waste package, oxidation of the cladding and the spent fuel, gaseous releases, contact between liquid water and the fuel, and transport of the released radionuclides beyond the confines of the engineered barrier.

5.2.1. Waste Package Lifetime

The canisters will be sealed and most probably filled with an inert gas. They must first be breached before there can be any releases of radionuclides. Several measures will be used to reduce the likelihood of canister breaching. The canisters will be made of corrosion resistant material, with each canister placed with an air gap between itself and the host rock to prevent any direct contact with pore water. The decay heat may create a dry zone for several hundreds of years after emplacement, further isolating the canisters from contact with liquid water.

Irrespective of these measures, canisters may still fail. Some of the mechanisms that might lead to failure are:

- Mechanical damage by excavation failure, earthquakes, magmatic intrusions or human intrusions.

- Corrosion by hot steam or water dripping through fractures.
- Corrosion by direct contact of canister with rock; e.g. loss of air gap because of spallation of rock or infilling by water-borne sediment.
- Corrosion by immersion because of rise in regional water table or perched water table.
- Heat pipe effect.

5.2.2 Cladding Failure

Most of the spent fuel will be protected by thin cladding, usually zirconium alloy, but in some cases stainless steel. In a small fraction of the cases, the cladding will be flawed by pinhole leaks or damaged (Van Konyenburg, 1987). The cladding is an additional layer of corrosion resistance for the fuel. It would protect the fuel from oxidation or water contact for a time. Since it is very thin (typically 0.6 mm) relative to the canister thickness, cladding has usually been ignored in performance assessment studies.

Aside from the potential corrosion protection offered by the cladding, the cladding itself is likely to contain C-14 produced by activation of impurities in the zirconium metal or picked up from the circulating water in the reactor. Cladding corrosion thus might prove to be a source term for the release of C-14 from the waste. Releases of gaseous C-14 are discussed in Appendix D.

5.2.3 Oxidation of uranium dioxide matrix

Uranium dioxide is unstable in an oxidizing environment (Grambow, 1989). Because the repository will be located in unsaturated rock, there will be oxygen available to oxidize the uranium dioxide following failure of the waste package and cladding. Prior to failure, the canisters will most likely be filled with an inert gas to prevent oxidation, although it is possible to have oxidation directly from water that might be contained in the fuel rods, particularly those which have already failed. The rate of oxidation depends among other things on temperature, so the time that the waste package fails might be important. Oxidation of the uranium dioxide is potentially important to the performance model, because uranium in higher valance states is much more soluble than in low valence states. If the fuel is immersed in water, the rate of oxidation may be the limiting rate for congruent dissolution of the fuel matrix (Doctor, 1988). In addition, oxidation of the fuel under dry or moist steam conditions can cause an increase in its volume and porosity, with the consequence that the ease at which the gaseous radionuclides such as C-14 could be released might increase.

5.2.4 Release of dissolved radionuclides from the fuel

Initially, the canisters and the spent fuel are likely to produce sufficient heat to dry out their surroundings or create a dry steam environment.

Eventually however, liquid water might come into contact with the spent fuel, allowing it to dissolve and release its inventory of radionuclides to the environment. Most of the inventory of radionuclides will be entrapped by the uranium dioxide matrix of the fuel, and will be released slowly as the matrix disintigrates. Some of the radionuclides released from the matrix might precipitate immediately because of their low solubility, thereby limiting their release (Ogard,), or may form colloids (Bonano,) Some of the more-volatile radionuclides such as C-14, cesium and iodine tend to migrate from the matrix and collect at intergranular boundaries and in the gap between the fuel and the cladding, particularly while still in the reactor. These volatile radionuclides will be released more quickly than those released by congruent dissolution.

5.2.4.1 Water contact fraction

DOE plans to emplace the canisters in the host rock in a manner that reduces the likelihood of water coming into contact with the waste. This includes vertical storage and an air gap between the canister and the rock walls. Furthermore, DOE believes that the heat generated by the waste may create a significant zone of dry rock around the canisters, isolating them until such time that the water can resaturate the rock. Water may still come into contact with the canisters by several mechanisms:

- Circulating water generated by the decay heat
- Infiltrating water flowing through fractures and dripping onto the canisters.
- Loss of the air gap caused by failure of the emplacement holes through mechanical and thermal stresses, or mineral and sediment infilling.

We characterize the water contact by a factor relating the fraction of water infiltrating the site coming into contact with the waste. The staff performed simple calculations to estimate the fraction of the waste canisters exposed to purely vertical infiltration by taking the ratio of the cross-sectional area of the canisters to the total area of land surface. This ratio was about 0.00078. In its uninterrupted state infiltrating water is likely to flow around the canisters because of the matrix suction of the unsaturated rock, so this simple figure does not capture the true nature of water contact. The analysis in the Environmental Assessment (DOE, 1986) assumed a contact fraction of 0.025, but the authors specified no basis for this choice. Other analyses have specified that all water infiltrating the site contacts the waste (Doctor, 1988)

For the present analysis, ~~We~~ we do not consider other sources of water that could come into contact with the waste. Two potentially important sources are (1) lateral inflows from areas of perched water and (2) liquid water circulation caused by heat-driven evaporation and condensation. If such a diversion was possible, some canisters might get a greater share of the overall infiltration at the expense of the remaining canisters being exposed to less water. Diversion of infiltrating water from outside the repository area might also be possible, but since the site occupies mainly an area of high ground, this is unlikely.

The significance of the issue of thermally driven water circulation is difficult to determine at this time. It may be the case that these phenomena are short-lived, and unimportant during the period of canister integrity, during which most of the water driven off would be diverted from the canisters rather than returning. Sophisticated models such as TOUGH would be required to carry these arguments further. They are beyond the scope of the Phase 1 study, but should be planned for subsequent studies.

5.2.5 Release of Gaseous Radionuclides

There are several gaseous radionuclides in spent fuel, although many of these are short-lived and of no long-term concern. The most significant radionuclides are C-14 and possibly I-129 (only at elevated temperatures). Carbon-14 would be released from the cladding, the cladding-fuel gap, and the matrix. The gaseous releases would be partitioned between the groundwater and air, depending on environmental factors such as saturation, temperature and concentration of bicarbonate ions.

None of the models reviewed in Appendix B handle the releases of C-14 in a very sophisticated way. The models either treat the C-14 as a component of the fuel released to the groundwater by congruent dissolution of the fuel matrix, or all is released instantaneously upon failure of the waste canister.

The release of C-14 from the repository is ~~a potentially serious problem~~ of interest at the Yucca Mountain repository because there is at least the possibility of a fast pathway to the accessible environment through the unsaturated fractured rock, excavations and tunnels. Two models of transport of C-14 in the geosphere of Yucca Mountain indicate that the time for C-14 released at the repository level to reach the atmosphere would be on the order of hundreds to a few thousand years, too short a time to depend on decay to diminish the importance of C-14 cumulative releases to the accessible environment (Knapp, 1987, Amter, 1988). Clearly, an assumption of instantaneous release from failed canister is too pessimistic. On the other hand, the assumption that all C-14 is contained in the matrix and released only as the matrix dissolves is too optimistic, because a substantial fraction of the C-14 may be contained in places other than the matrix, e.g, the cladding. Laboratory data on the location of various radionuclides in spent fuel under different conditions will reduce this modeling uncertainty.

5.3 Model Selection and Justification

5.3.1. Model for Dissolved Radionuclides

The Staff has decided to adopt the source term model currently incorporated in NEFTRAN. Radionuclide releases would occur only after failure of the engineered barrier characterized as a single failure time t_f. Upon failure of the engineered barrier, radionuclide release from the waste package will be governed by either the leaching rate determined by the rate of dissolution of the waste form, or limited by the solubility of the individual radionuclides:

for t less than t_f, $R_i(t) = 0$

for t greater than t_f,

$$R_i(t) = \lambda_L M_i(t) \qquad C_i < S_i, \; C_i = \frac{R_i(t)}{IAf} \qquad (5.1)$$

$$R_i(t) = S_i \, IAf \qquad C_i > S_i \qquad (5.2)$$

where M_i = the inventory at time t of the radionuclide in the waste, S_i = solubility of radionuclide i, and C_i = dissolved concentration of radionuclide i.

The leach rate λ_L will be determined by the combination of the infiltration rate I, the fraction of water contacting the waste f, the surface area of the repository A, the solubility of the waste form S_u and the initial inventory of the waste form M_o:

$$\lambda_L = I \times f \times A \times S_u/M_o \qquad (5.3)$$

An assumption inherent in the model is intimate contact between infiltrating water and the spent fuel following the failure of the waste package, and does not take into account any limiting controls on the release that might be afforded by the presence of cladding or other physical structures.

5.3.2 Limitations of Model for Dissolved Radionuclides

The most significant limitations of the dissolved radionuclide source term model are:

- o The model ignores the diffusion-limited case where there might be the buildup of a boundary layer limiting the release of solubility limited radionuclides (this mechanism would apply only if there were a continuous liquid water path between the fuel and the rock).

- o For larger infiltration rates, the model cannot represent phenomena that would tend to limit the rate of release such as the forward rate of reaction for oxidation of the spent fuel, and the possibility that oxidants might not be available in unlimited quantities because they would be consumed within the canisters.

- o The model assumes intimate contact between the groundwater and the waste, ignoring the features such as the air gap designed to prevent such contact. It in effect assumes there is no protection for the fuel from the water, even though the fuel has multiple layers of protection including the air gap, waste package and cladding.

- o The model incorporates a single time to failure, even though it is more likely that waste packages would fail in a distributed manner in time and space.

- o Releases from the matrix of low-solubility radionuclides might result in colloid formation rather than a precipitation.

- o The model does not take into account radionuclides which might not fit neatly into the three compartments (unleached, undissolved and dissolved), such as those collecting in the grain boundaries and in the cladding gap.

- o The model ignores the potentially significant amount of liquid water circulation through evaporation and condensation of groundwater that might be caused by the repository heat, i.e., a "heat-pipe".

The use of this model was based on expediency because the fundamental framework was already in place in the NEFTRAN code and required a minimum of reprogramming to adjust the coefficients to represent the Yucca Mountain case. Adjustment of the coefficients of the model allows a wide latitude of potential source term conditions to characterize either congruent dissolution of the uranium matrix or solubility limited releases.

5.3.~~2~~3 C-14 Release Model

Very little is known about the long-term release of gaseous radionuclides from spent fuel. The only data on releases from spent fuel indicate a rapid, small release of C-14 upon failure of the fuel rod, and very slow release thereafter (van Konyenberg, 1984).

Because of the speculative nature of the C-14 release model, gaseous release pathways were not included into the overall systems analysis, but are treated separately as an auxiliary analysis in Appendix D.

5.4 Source Term Inventory

The inventory of radionuclides assumed for the source term in the Phase 1 study is given in Table 5.1. These radionuclides were chose from an extensive list of fission and activation products found in spent fuel on the basis of half lives, potential inventories and radiotoxicity (in terms of their EPA cumulative release limits).

Table 5.1 - Radionuclide Initial Inventory

Radionuclide	Halflife, yrs	Inventory, Ci
Cm-246	5.50E03	2.45E03
Pu-242	3.79E05	1.12E05
U-238	4.51E09	2.24E04
Cm-245	9.30E03	1.26E04
Pu-241	1.32E01	4.83E09
Am-241	4.58E02	1.12E08
Np-237	2.14E06	2.17E04
U-233	1.62E05	2.66E00
Th-229	7.34E03	1.96E-03
Am-243	7.95E03	9.80E05
Pu-239	2.44E04	2.03E07
U-235	7.10E08	1.12E03
Pu-240	6.58E03	3.15E07
U-236	2.39E07	1.54E04
Pu-238	8.60E01	1.40E08
U-234	2.47E05	5.18E03
Th-230	8.00E04	2.87E-01
Ra-226	1.60E03	5.18E-04
Pb-210	2.23E01	4.90E-05
Cs-137	3.00E01	5.25E09
Cs-135	3.00E06	1.89E04
I-129	1.59E07	2.31E03
Sn-126	1.00E05	3.36E04
Tc-99	2.15E05	9.10E05
Zr-93	9.50E05	1.19E05
Sr-90	2.90E01	3.64E09
Ni-59	8.00E04	2.10E03
C-14	5.73E03	9.80E04

5.5 References

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6.0

6.0 FLOW AND TRANSPORT MODELS

6.1 Introduction

The far-field post-closure simulation of radionuclide transport away from the high-level waste (HLW) repository at Yucca Mountain presents a unique and challenging performance assessment problem in part due to: 1) lack of field and laboratory analyses identifying important processes, 2) the possibility for multiple transport pathways (gas and liquid) available in the unsaturated zone, 3) numerical difficulties in solving the non-linear unsaturated flow problem, 4) data uncertainties and testing limitations in determining the unsaturated zone parameters, and 5) by comparison to porous media, lack of a variety of established computational tools to evaluate radionuclide transport in unsaturated fractured tuff quantitatively.

The selection of a transport model for this phase of the MOU needs to make use of what is currently available. However, the selection process should incorporate information on where improvements are currently being made and should be available in the near future. The process of selecting and implementing a transport model(s) for the performance assessment involved defining the technical issues, reviewing current computer models, selecting computer model(s), evaluating the selected model(s), developing a database, performing support or auxiliary analyses, and making recommendations for future improvements.

6.2 Definition of Issues for Selecting Performance Assessment Transport Models

The selection of models for simulating radionuclide transport should be based on the current and alternative concepts of the Yucca Mountain site, the types of pathways that are envisioned to be analysed, the phenomena that control the flow and transport pathways at Yucca Mountain, and the scenarios expected.

6.2.1 Current Concepts

The Yucca Mountain site is located on and immediately adjacent to the southwestern portion of the Nevada Test Site. Yucca Mountain is a prominent group of north-trending, fault-block ridges. The terrain at the site is controlled by high-angle normal faults and eastward-tilted volcanic rocks. Slopes are locally steep (15 to 30 degrees) on the west-facing side of Yucca Mountain and along some of the valleys that cut into the more gently sloping (5 to 10 degrees) east side of the mountain.

The stratigraphic units of primary interest at Yucca Mountain are primarily comprised of ash-flow and ash-fall tuffs which originated from eruptions during the development of calderas. The amount of welding, fracturing, and chemical alteration varies greatly from one layer to the next. The major stratigraphic units beneath Yucca Mountain starting at the surface are: alluvium, Tiva Canyon welded unit, Paintbrush nonwelded unit, Topopah Springs welded unit, Calico Hills (vitric and zeolitic), Prow Pass (welded and nonwelded), and Bullfrog welded unit (see Figure 6.1).

Three broad categories which describe these tuffs are: densely welded tuffs, nonwelded vitric tuffs, and nonwelded zeolitized tuffs. The densely welded tuffs are highly fractured. These tuffs have a very low saturated matrix conductivity (less than 1 mm/year) and a saturated conductivity for the fracture which is probably several orders of magnitude higher than the matrix value. The nonwelded vitric tuffs have fewer fractures and a high matrix saturated conductivity (100 - 10,000 mm/year). The fractures for this unit would have a relatively low saturated conductivity. The nonwelded zeolitized tuffs have few fractures and low matrix-saturated conductivity (less than 1 mm/year) and low fracture saturated conductivity. The contacts between these units generally tend to occur over short distances and involve large differences in hydrologic properties (Prindle, 1987).

Based on current information on hydrogeologic units and theories of flow at Yucca Mountain, the DOE (from page 3-195 of the Yucca Mountain Site Characterization Plan) described the conceptualization of flow from the Topopah Springs unit to the water table as follows:

1. "Flow in the TSw unit is expected to be essentially vertical and under steady-state conditions to occur as flow within the matrix for fluxes less than some critical value of flux related to the saturated matrix hydraulic conductivity, and predominantly as fracture flow at fluxes higher than the critical value.

2. Lateral flow may be induced in the TSw unit at its contact with the underlying Calico Hills nonwelded unit (CHn). The circumstances under which this may occur depend on the magnitude of the flux in the TSw unit and whether this unit is underlain by the low-conductivity zeolitic facies (CHnz) or the relatively higher-conductivity vitric facies (CHnv) of the CHn unit. At low fluxes within the TSw unit, lateral flow may be produced by capillary-barrier effects within the matrix of the TSw unit where it overlies the CHnv unit. At high fluxes, efficient fracture flow in the TSw unit may produce lateral flow as well as vertical flow where the low-conductivity CHn unit underlies the TSw unit.

3. Flow in both the CHnv and CHnz units is predominantly vertical through the matrix (although a lateral component may occur parallel to the bedding within the vitric CHnv unit) and continues directly to the water table wherever the latter transects the CHn unit. Where the CHn unit lies above the water table, flow is presumed to proceed vertically downward to the water table through the Crater Flat undifferentiated unit (CFu).

4. The nearly vertically oriented fault zones and their associated fracturing may be highly effective pathways for vertical moisture flow, especially in the competent TCw and TSw units. But faults may impede lateral flow and may thus produce perched-water bodies where the faults transect zones or horizons of significant lateral flow."

Additionally, very little data are available on estimated infiltration rates and deep percolation rates past the repository. Estimates of deep percolation rates past the repository horizon are described on page 3-205 of the SCP as:

"Wilson (1985) reviewed available site and regional hydrogeologic data in order to set conservative upper limits on the present, net vertically down- ward moisture flux below the repository horizon at Yucca Mountain and on the present rate of net recharge to the saturated zone in the vicinity of Yucca Mountain. Wilson (1985) concludes (1) that the liquid-water percolation flux, directed vertically downward in the matrix of the TSw unit below the repository horizon, probably is less than 0.2 mm/yr and (2) that the area averaged rate of net recharge to the saturated zone in the vicinity of Yucca Mountain probably is less than 0.5 mm/yr. Although Wilson (1985) considered a number of processes, such as upward water-vapor flow in the fractures of the TSw unit at the repository horizon, these upper bounds on percolation and recharge fluxes must be regarded as preliminary estimates that have as-yet- unknown limits of uncertainty."

Based on the above comments from the SCP, there is currently insufficient data to clearly rule out alternatives to a single conceptual flow model for Yucca Mountain. While the effects of fractures on ground-water flow and of flow diversion at layer boundaries will certainly need to be assessed to determine fluid flux through the repository, the detailed modelling required for this type of determination is beyond the scope of the Phase I analysis. We have assumed for this phase of the MOU that flow will be vertical and flow through the fractures occurs only when the recharge exceeds the matrix saturated conductivity.

It is unclear at this time as to whether or not this conceptualization is conservative. While, flow diversion above the repository due to conductivity contrasts should lead to a lower flux through the repository, flow diversion below the repository could reduce travel time to the water table. Additionally, it is uncertain how fractures should be treated to maintain a certain amount of realism and conservatism. Flow diversion (see Appendix G, Two-Dimensional Flow Model) and the role of fractures are topics that deserve more attention in the second phase of the MOU.

6.2.2 Pathways

The assessment of a repository at Yucca Mountain could involve the following three pathway groups; (1) liquid, (2) gas, and (3) direct. The most obvious release path for radionuclides away from the repository is the liquid pathway. It is anticipated that radionuclides will move vertically in the unsaturated zone towards the water table and then horizontally in the saturated zone (for this phase of the MOU we are only considering transport in the unsaturated zone for the liquid pathway).

Another less obvious pathway is the gas pathway. The gas pathway is a potentially serious concern for the Yucca Mountain repository due to the presence of carbon-14. It is present in quantities at least one order of magnitude greater than allowed for release to the accessible environment. It can exist as one of

several gasses (CO_2, methane, acetylene), and could therefore move relatively rapidly compared to its halflife (5720 years) through the unsaturated fractured rock and along pathways such as access tunnels and excavations. In addition, unlike most of the other radionuclides in the waste, transport in the geosphere is not likely to depend strongly on the influx of water to the repository, and can proceed under totally dry conditions.

A final release pathway could occur as a result of a "direct" release. The "direct" release pathway encompasses a couple of possible scenario types such as a release due to the drilling into the repository and a release due to a disruptive event like a magmatic eruption. For this phase of the MOU it was considered too difficult to consider consequences due to volcanic activity, therefore, the direct release pathway considered only releases due to drilling. Releases resulting from volcanism should be accounted for in the next phase of the MOU.

6.2.3 Flow and Transport Pathway Phenomena

Performance assessment of potential releases of radioactivity from nuclear waste requires an understanding of a number of complicated transport phenomena for the pathways under consideration. The transport pathways to be analysed for Yucca Mountain are the liquid pathway, the gas pathway (primarily involving the transport of carbon-14), and a direct release pathway (due to a drilling scenario or a magmatic scenario).

6.2.3.1 Liquid

This section describes, in a preliminary way, the phenomena associated with the transport of radionuclides in ground water during the post-closure phase of an HLW repository and to assess the relative importance of the identified phenomena. A common starting point in the development of a transport model is a qualitative statement of the conservation of mass in the liquid phase for an elemental volume (Freeze, 1979):

net rate of change of mass within the element	=	flux of solute out of the element	-	flux of solute into the element	+ -	loss or gain of solute mass due to reactions and sinks and sources

The conservation of mass generally leads to the following differential equation that describes the transport of a solute:

$$\partial C/\partial t = D(\partial^2 C/\partial x^2) - v(\partial C/\partial x) + (\rho/n)(\partial S/\partial t) +(-)\ R$$

(mass change) (dispersion) (advection) (reaction) (sink/source)

where; C = solute concentration
t = time
D = dispersion coefficient
x = spatial dimension
v = average linear velocity
ρ = bulk density
n = porosity
S = mass of solute adsorbed per unit mass of rock
R = sink/source term (includes radioactive decay or production)

The processes that control flux into and out of the elemental volume are advection (transport via the bulk motion of the ground water) and hydrodynamic dispersion (transport resulting from mechanical mixing and molecular diffusion). Chemical reactions and radioactive decay will affect the loss or gain of solute mass (for the present analysis phenomena such as Knudsen diffusion and coupled processes are considered of minor importance).

Physical Processes

It is generally assumed that the bulk movement of fluid will be the primary source of transport away from a HLW repository. In a porous medium it is commonly assumed that the average rate of solute transport by advection is equal to the average linear velocity of the fluid times the concentration. The presence of a fracture system complicates the advective flow system. The fracture system provides a high permeability flow path separate from but interacting with the matrix path.

In the unsaturated zone, water is held in the pore space by surface tension forces. Geologic media are comprised of a variety of pore space and fracture dimensions, therefore, these volumes will not empty at the same suction. The large pores (or larger fractures) will empty at low suctions, while small pores (sharper curvature) will empty at higher suctions. Most models of unsaturated flow in fractured media, therefore, assume that under high suction the dominant ground-water pathway will be in the matrix (i.e. the fractures will be dry). However, it is worth pointing out that many factors (transient infiltration rates, fracture coatings, fracture dimensions, and the presence of perched water) can dramatically influence the degree of fracture flow. Many assumptions which preclude fracture flow have not been substantiated by laboratory or field data and, therefore, cannot be ruled out as a possible transport pathway in unsaturated, fractured rocks (Olague, 1989).

Based on the lack of information to support a detailed fracture flow model, we have assumed a steady state flow model where the fractures contribute to flow only when the infiltration rate exceeds the saturated conductivity. Further work will need to determine the degree of conservatism or pessimism in this assumption.

While advection moves solute in the direction of flow, hydrodynamic dispersion and matrix diffusion affect solute concentration along its flow path. Hydrodynamic dispersion includes dilution due to mechanical mixing and molecular diffusion. Mechanical mixing (a direct result of a tortuous path, variation in pore sizes or fracture apertures, and surface roughness) is related to the heterogeneity of the geologic media and is typically characterized by the dispersivity.

For the present analysis we have assumed that dispersivity can be represented with a dispersion length. This treatment was assumed adequate for the present study because the performance measure, cumulative release at the accessible environment over 10,000 years, appears to be rather insensitive to longitudinal dispersion (see Sensitivity and Uncertainty Analyses, Section 9.5).

Matrix diffusion couples the solute concentration in the fracture and matrix systems and is generally thought to provide a retardation of radionuclide transport in the fractures. As with the flow of water across the fracture-matrix interface, a large uncertainty in evaluating this phenomenon is determining the effect of fracture coatings on the diffusion rate. Quantification of the effect of fracture coatings will be needed to better determine the best approach for performance assessment. We currently have conservatively assumed that matrix diffusion does not occur.

Chemical Processes

There are several chemical processes that affect the movement of radionuclides in ground water. One of the most significant chemical processes that occurs is sorption (Olague, 1989). Solute species adsorb to the matrix or fracture surfaces by forming bonds with the molecules on the solid surface. The strength of these bonds and the kinetics depend on many chemical factors such as: 1) electric charge of solute and solid, 2) saturation of bonding sites, 3) pH, 4) oxidation and reduction potential, and 5) temperature and pressure of the hydrogeologic system (Freeze, 1979).

Adsorption can be physical (generally considered a reversible process) or chemical (generally considered an irreversible process). At any moment some of the solute particles are bonded to the solid surface and some are free to move with the ground water. The adsorption-desorption process has typically been represented in most ground- water transport models using a retardation equation that employs a distribution coefficient. The assumptions in this model include instantaneous and reversible adsorption and desorption (equilibrium), linear sorption isotherms, and single-valued sorption isotherms (i.e., no hysteresis effect) (Rasmussen, 1987). Although simplistic this type of model combined with more detailed, supporting geochemical analyses may be the only practical approach for a long term assessment. It is, therefore, the model adopted here.

The model ignores precipitation of radionuclides along the flow path, although solubility is taken into account in the source term. The staff considers that the likelihood of concentration increasing, above solubility limits, along the flow path (e.g., by chain decay of other radionuclides) would be small. Furthermore, this assumption is conservative because it would overestimate the cumulative release.

Table 6.1 Identification of liquid pathway processes and estimated importance in calculating cumulative release from the liquid pathway.

Processes	Estimate of Importance
1. Advection	High
2. Sorption	High
3. Radioactive Decay	High
4. Fracture-Matrix Interactions	High
5. Matrix Diffusion	Medium
6. Precipitation of Radionuclides	Low

7. Dispersion Low

6.2.3.2 Gas

An alternative pathway for radionuclide transport to the accesible environment is possible due to the presence of a gas pathway in the unsaturated tuff. Futhermore, gas phase source terms (i.e., carbon-14, tritium, krypton-85, and iodine-129) could potentially be released from spent fuel buried at Yucca Mountain. Gas phase carbon-14 in the form of carbon dioxide appears to be the most important for considerations of performance assessment. The half-lives of tritium and krypton-85 are relatively short (12.3 years and 10.7 years, respectively) and it is believed that the elemental iodine will quickly partition into the liquid phase. Because of the complexity of the issue and the relatively poor state of knowledge about gaseous release and transport, carbon-14 release to the atmosphere is not included into the total system analysis in Phase 1. An auxiliary analysis for carbon-14 release to the atmosphere is presented in Appendix D.

6.2.3.3 Direct

Potentially significant scenarios for the assessment of repository performance involve the possibility of volcanism in the form of a disruptive event such as a magmatic eruption, or an intrusive event involving human drilling activities. Both scenario classes involve events whose estimated likelihood of occurrence and consequences are very uncertain over the regulatory period of performance for the repository (i.e., 10-100,000 years). Considerations for magmatic events and human intrusion are discussed below. However, due to the complexity in understanding and predicting magmatic events, simulation work in this area was deferred to phase 2.

Magmatic Events

Basaltic eruptions are noted to have occurred near the Yucca Mountain site and west and south of it during the Quaternary period. Observations on basalt flows and cinder cones have been observed on Crater Flat and calderas at Amargosa Valley have deposited ash flows as recently as 200,000 to 300,000 years ago. The consequences assuming that a magmatic eruption occurs are very uncertain; however, it is believed that this class of scenarios would need to consider the following in estimating consequences: (1) entrainment of the waste and deposition on the surface, for example, as a result of a physical (steam) explosion, (2) dispersal of fine-grained ash and radioactivity into the atmosphere, (3) mechanical and thermal loading that can affect rock stresses and permeabilities and flow conditions for radionuclide migration from the repository to the accessible environment, even if the event does not compromise the structural integrity of the repository, (4) the relative amounts of radioactivity that would be released due solely to the occurrence of this natural event, (5) potential barriers to flow or water table level changes and (6) the source term.

The source term depends upon many factors, including:

- mix of waste forms for the repository (spent fuel and high level waste from defense activities)

- spent fuel inventory characteristics (reactor type and burn-up)

- time of emplacement

- emplacement configuration

- rock geochemical properties

- time of eruption or intrusion

- extent, location, and geometry of volcanism

For scenarios involving the interception of waste packages by feeder dikes, estimates of the distribution and size of these dikes (resulting from the feeding of basaltic cinder cones) are needed, in addition to estimates of waiting times for their occurrence (to account for radioactive decay).

Human Intrusion

Human activities such as deep exploratory drilling of boreholes could potentially provide direct releases of radioactivity to the environment. It is believed that this issue is primarily a source term issue which depends on the amount of radioactivity brought to the surface along with the core from grazing or penetrating waste packages during drilling. The waste package material, emplacement configuration, age of waste at time of interception by a drill bit, altogether generally contribute to estimating the radioactive source term. Estimates of radioactivity brought to the surface in contaminated cores from those boreholes that intercept the repository are also needed for a more complete consequence analysis.

In order to estimate the risk one needs to combine the consequence information with a probabilistic analysis of the drilling rate and penetration depth.

6.3 Computer Program Review and Selection

The analysis of any complex system often involves the use of computer implemented mathematical models to assist the analyst in presenting an "accurate" description of the risk or performance of the system. The analysis of hydrologic systems has, over the last twenty years, created an number of computer programs for analyzing a variety of problems (until recently little attention has been paid to an unsaturated, fractured, and uneconomic rock such as tuff). Based on the pathway phenomena and types of scenarios anticipated of Yucca Mountain, computer programs were reviewed for their applicability in a performance assessment.

6.3.1 Liquid Pathway

The evaluation of the liquid pathway could involve a suite of computer programs. The complexity of the Yucca Mountain site could dictate the use of a specific model to evaluate a specific question which then would be used to provide inputs or justify assumptions of simpler models used in a systems model. Some examples of the types of programs needed are: 1) two-phase flow program for analysing thermal effects, 2) two- or three-dimensional program for simulating far-field flow and transport and regional flow, 3) geo-chemical programs for assessing retardation phenomena, 4) a program which includes the influence of fractures or allows for an interaction between fractures and matrix, and 4) an efficient transport program for use in the multiple simulations of a performance assessment.

The review of computer programs is divided into the following four sections: 1) regional or far-field ground-water flow programs, 2) two-phase flow programs, 3) geo-chemical programs, and 4) transport programs. The ability of the various programs to deal with the presence of fractures will be discussed under the individual programs. A summary of the review and the selection rationale is provided in the subsequent sections while individual program summaries are provided in Appendix C.

6.3.1.1 Regional Ground-Water Flow Programs

A number of unsaturated flow programs (e.g., FEMWATER and UNSAT2) were developed approximately 10 years ago to analyze unsaturated flow in near surface soils (Thomas, 1982). NRC participation in the international project HYDROCOIN (Cole, 1987) revealed significant numerical limitations in these programs in simulating unsaturated problems involving large non-linearities (e.g,, infiltration into a dry soil and large permeability contrasts). These and similar type programs were not examined further due to their numerical deficiencies which would be unacceptable in evaluating Yucca Mountain. A new generation of unsaturated flow programs has been developed to better handle the non-linearities encountered in unsaturated flow.

Sandia National Laboratories reviewed 71 computer programs that simulated groundwater flow and transport in the unsaturated zone (Olague, 1989). Based on this review and recently published user manuals, it was decided to provide a description for the computer programs entitled SUTRA, VAM2D, and TRACER3D. The three programs employ similar Darcian approaches to simulating fluid flow in porous media. The ability to simulate fracture flow could only be accommodated through a dual porosity approach. (Currently, there are no existing programs which simulate fracture-matrix interactions with an approach different from dual-porosity. Sandia National Laboratories under RES contract FIN A-1266 is developing a flow program that will account for the fracture-matrix interactions in a more rigorous fashion than is currently available. This program is scheduled for completion in April of 1990.)

The VAM2D program (Huyakorn, 1989) was selected for use in modeling regional flow because of the efficiency of the non-linear numerical techniques employed and the availability of the program for NRC staff use.

6.3.1.2 Two-Phase Flow Programs

Assessing the thermal period of the HLW repository will require programs that can simulate the flow of air, liquid water, and water vapor. TOUGH, NORIA, and PETROS are existing programs which solve the two-phase flow and energy transport problem. A detailed Sandia review of these programs (Updegraff, 1989) discussed the difficulties of running two-phase flow models and the relative strengths and weaknesses of the individual programs. Overall, one program was not superior to the others. However, TOUGH successfully ran most of the test problems while NORIA and PETROS could at best simulate approximately half the test problems.

The TOUGH program (Pruess, 1987) was selected to analyze two-phase flow problems because of its ability to handle a variety of problems (Updegraff, 1989) and the current availability of TOUGH to NRC staff. (Due to the complexity of two-phase flow problems, simulation work was not performed in Phase 1.)

6.3.1.3 Geochemical Programs

The geochemical behavior of the HLW repository could have a very strong effect on the movement of radionuclides. Unfortunately, current geochemical programs are not amenable to most performance assessment systems programs due to their complexity. The primary use of the geochemical programs will be to aid the understanding of the geochemistry of the site and the assignment of lumped parameters in the simpler transport models.

The current phase of the MOU does not involve complex modeling associated with geochemical analyses. Summaries of various programs are included in Appendix C. However, a program has not been selected.

6.3.1.4 Transport Programs

The utilization of a transport program in a systems code for the performance assessment of Yucca Mountain will require a number of simplifications of the real system to accomodate the large number of simulations necessary for sensitivity and uncertainty analyses (see Appendix E, Statistical Convergence for the CCDF). Some of the simplifications being considered are: utilization of a one- or two-dimensional analysis; limited (if any) interaction between fractures and matrix; steady-state flow; and limited geochemistry (typically a lumped retardation factor which is intended to account for all the geochemical interactions).

A number of programs currently exist which employ many of the above simplifications (see code summaries in Appendix C) have been reviewed for utilization in this phase of the MOU. The review included numerical solutions such as SPARTAN, NEFTRAN, and TOSPAC as well as analytic solutions such as the UCB programs. The NEFTRAN (Longsine, 1987) program, developed at Sandia National Laboratories under NRC funding, was selected because: 1) it was available on NRC computer systems, 2) ready access to the Sandia developers, and 3) efficiency of the program and compatibility with the LHS computer program for analysing model sensitivity.

Although all of the reviewed programs did not fully describe fracture-matrix interactions, Sandia is currently modifying NEFTRAN (to be completed by March, 1990 to include fracture-matrix interactions. Staff use with the current version of NEFTRAN will assist technology transfer of the new version of NEFTRAN in 1990.

6.3.2 Gas Pathway

The gas pathway has been treated as an auxiliary analysis in this phase of the MOU (see Appendix D).

6.3.3 Direct Pathway

The staff was unable to acquire computer programs for evaluating the consequences of drilling into a repository in a timely fashion. The staff developed a model that accounts for the anticipated important aspects of a drilling scenario. The model accounts for a drilling rate, radioactive decay, the areal extent of the repository, waste package emplacement orientation (horizontal versus vertical), and boreholes intercepting both the waste package and contaminated rock. A detailed discussion of the drilling model is provided in Appendix H.

6.4 References

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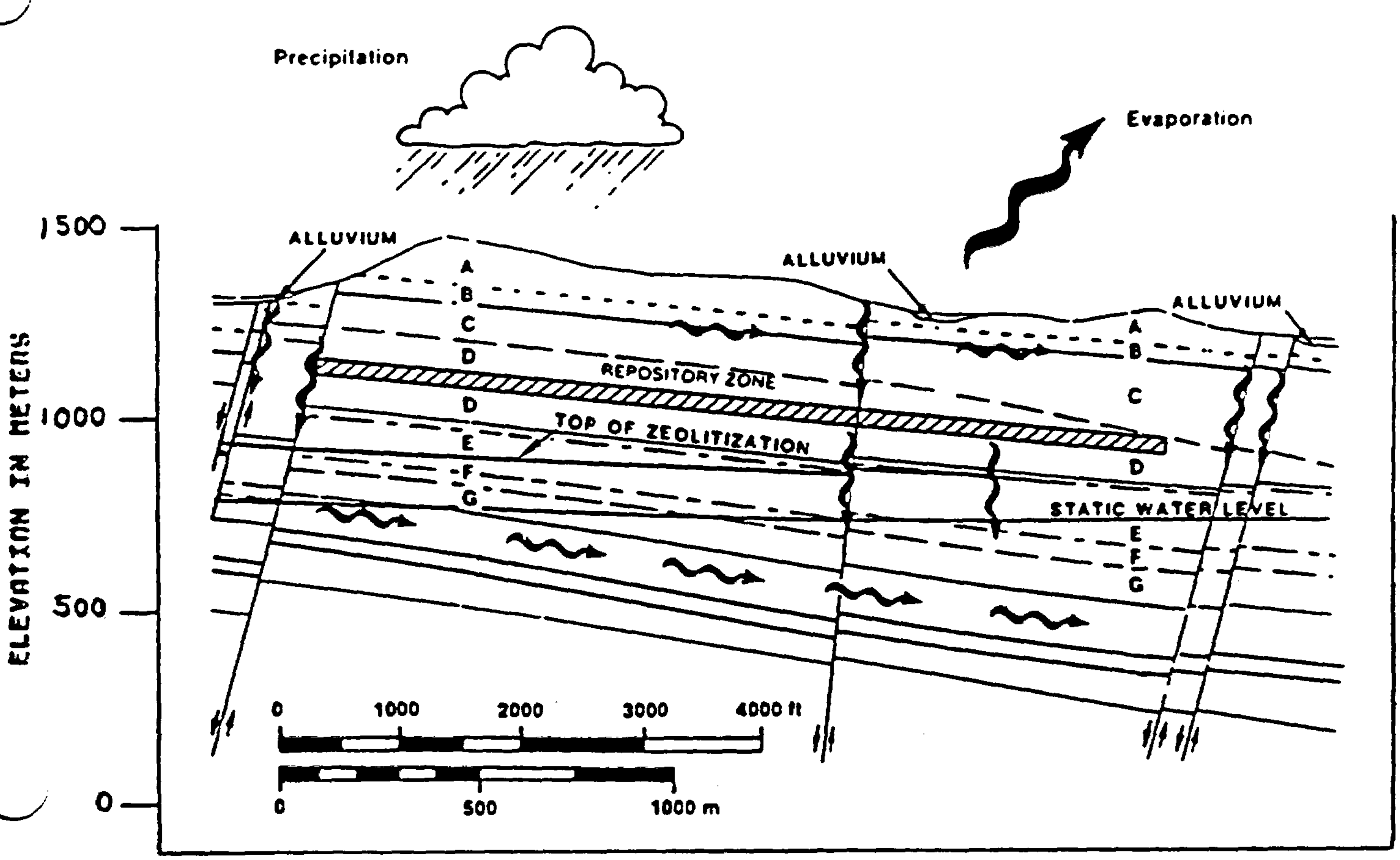

Figure 6.1 Conceptualization of the hydrostratigraphic units at Yucca Mountain (A is the Tiva Canyon unit, B is the Paintbrush unit, C is the Topopah Springs #1 unit, D is the Topopah Springs #2 unit, E is the Calico Hills unit, F is the Prow Pass unit, and G is the Bullfrog unit).

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70

7.0 METHODOLOGY FOR SCENARIO DEVELOPMENT

7.1 Introduction

An important part of a performance assessment for a HLW repository is an evaluation of the uncertainties in projected performance. Two general approaches are available for analyses of uncertainties in repository performance. Such analyses can be carried out by incorporating the uncertainties directly into the model(s) and data base(s) describing the repository system, or uncertainties can be approximated as "scenarios" -- i.e., descriptions of alternative ways in which the repository system might perform in the future. Most analyses use a combination of the two approaches, although there are generally no explicit criteria for which way to treat a specific source of uncertainty. Thus, lists of processes and events to be included in scenarios often include phenomena such as waste canister corrosion, even though such phenomena are likely to be evaluated directly within the repository model(s) and data base(s) rather than as scenarios.

This study distinguished two aspects of an uncertainty analysis: (1) uncertainty about the characteristics of the repository system and its environment as they exist at the time of analysis, and (2) uncertainty about the future evolution of the environment within which the repository will exist far into the future. For the purposes of this study, scenario analysis is limited to the second type of uncertainty. All uncertainties of the first type are assumed to be incorporated directly into the model(s) and data base(s) which describe the repository system.

The term "scenario" is defined here as a description of one of the many alternative ways in which the environment of a repository might evolve in the future. The goal of a scenario analysis is then to identify a set of such scenarios, to be used in uncertainty analyses, which is sufficiently complete to support a regulatory decision regarding the acceptability of the repository.

In this study, phenomena were considered to be either "internal" or "external" depending on the location where they are initiated. Those phenomena initiated in the accessible environment are classified as external perturbations of the repository system, even if the effects of the phenomena occur within the repository. Thus, fault movement within the controlled area of the repository is classified as an external event because the tectonic forces responsible for the movement are external. Similarly, drilling into a repository is classified as an external event because the drilling is initiated outside the controlled area. Phenomena internal to the repository system, such as corrosion of waste canisters, were assumed to be addressed in the development of model(s) and data base(s) describing the repository system, and therefore were excluded from consideration for scenario development.

The boundary of the repository system was chosen to be coincident with the boundary of the accessible environment for two reasons. First, many of the uncertainties within this boundary involve processes rather than discrete disruptive events. Simulation of processes and their associated uncertainties is often fairly simple, sometimes involving no more than specification of a range of values within the data base for the repository (e.g., a range of corrosion rates). On the other hand, phenomena outside this boundary are often rare, discrete events such as fault movement or volcanic activity.

Simulation of such events within the model of the repository system may be awkward, especially when Monte Carlo or related simulation techniques are used. In such cases, the number of simulations needed to obtain a good representation of repository performance may be so large that accurate approximations of repository performance are not practical.

The second, and more important, reason for selection of the repository system boundary involves the way in which the repository is perceived by regulators and by the public. Both groups tend to view the repository system as ending at the accessible environment boundary and to visualize phenomena occurring outside this boundary as external perturbations of the repository. Scrutiny of repository safety tends to take the form of "What if" questions -- e.g., What happens to the repository if a volcano erupts nearby? Evaluation of external phenomena through scenario analysis directly answers such questions, while incorporation of external phenomena into the repository system model(s) or data base(s) would tend to obscure the results of the analysis.

It is important to note differences between the approach adopted here for scenario development versus those proposed by other analysts. Hodgkinson and Sumerling (ref. 1) describe an approach for scenario development in which no distinction is made between "internal" phenomena and those which occur outside the repository. In their approach, processes such as waste canister corrosion would be treated as phenomena to be combined into scenarios for analysis. Because these authors combine internal with external phenomena, their list of "events, features and processes" to be combined into scenarios contains approximately 150 entries and, even after screening out unimportant entries, the number of scenarios that could be constructed from a list of this length would be quite large. Treatment of internal phenomena within the repository system model greatly reduces the potential number of scenarios, keeping the complexity of the repository analysis within manageable bounds.

Hodgkinson and Sumerling also describe an alternative approach, referred to as "environmental simulation," in which an attempt is made to incorporate all identifiable uncertainties into the repository system model. As discussed above, it appears that such an approach would have difficulty satisfying the information needs of regulators, and could require excessive numbers of simulations in order to provide accurate approximations of repository performance.

When this study was initiated, the staff intended to accomplish two objectives: (1) identify a methodology that could be used for scenario development, and (2) demonstrate the utility of the methodology by application to the Yucca Mountain site. Due to limitations on availability of staff resources, only limited progress was made on application of the methodology. This report is, therefore, primarily a status report of on-going work, and consists primarily of a description of the methodology selected by the staff.

7.2 Methodology

The scenario development approach adopted for this study is an adaptation of the event tree approach used in probabilistic risk analyses, and consists of the following steps:

1. Identification of Processes and Events. This step involves identification of a comprehensive set of processes and events that could adversely affect repository performance. Only "external" processes and events occurring (or initiated) in the accessible environment are included. Processes and events internal to the repository system are assumed to be treated as uncertainties within the model(s) or data base(s) describing the repository system and therefore are not included here. When the time of occurrence of a process or event (e.g., volcanic activity) is expected to significantly influence the effects on repository performance, the time is specified as part of the description of the event, and occurrences at several different times may be listed as separate "subevents."

2. Assignment of Probabilities. Probabilities of the processes and events are developed from historical data, models of the processes and events, or expert judgment.

3. Screening of Events and Processes. Where possible, processes and events are eliminated from the list compiled in step 1 using the following screening criteria: a) lack of physical reasonableness, b) low probability of occurrence, and c) insignificant effect on repository performance if the process or event were to occur.

4. Scenario Construction. Processes and events surviving the screening of step 3, above, are combined to form scenarios using the event tree approach described in NUREG/CR-1667. For this study, different permutations of events are not considered to be separate scenarios. Instead, judgment is used to select the most detrimental permutation to be used as a surrogate for all others. For the illustrative purposes of this project, it was planned that the only scenarios to be formed would be those consisting of zero, one or two processes or events -- i.e., scenarios containing three or more events would not be formed.

5. Scenario Probabilities. Scenario probabilities are estimated by combining the probabilities of the processes and events which comprise the scenarios. If the processes and events comprising a scenario are independent, the scenario probability is determined by multiplying the probabilities of the constituents. If the processes and events are not independent, correlations or causal relationships must be considered when determining scenario probabilities.

6. Scenario Screening. Scenarios are screened using the same criteria as for screening processes and events in step 3 above.

Application. Application of the selected scenario development methodology for Yucca Mountain was largely limited to the first step -- identification of processes and events. The primary source of information used to compile a list of processes and events was the staff's knowledge of the Yucca Mountain site, although limited references to literature describing similar scenario development efforts for Yucca Mountain were also made. Some progress was also made on the third step involving screening processes and events. However, because probability assignments were not completed, screening was conducted only on the bases of physical reasonableness and insignificant consequences. Combination of processes and events into scenarios, development of scenario probability estimates, and scenario screening (steps 4 - 6) must await development of probability estimates for the phenomena comprising the scenarios. The following table presents a summary of the candidate list of processes and events identified, including those that were later screened from the list. Following the table is a more detailed description of each process and event and, where appropriate, the basis for screening.

LIST OF PROCESSES AND EVENTS

I. Tectonic

A. Volcanic
1. Extrusive
a. On-site
i. Years 0 - 100
ii. Years 101 - 1,000
iii. Years 1,001 - 10,000
b. Off-site

2. Intrusive
a. Upgradient
b. Downgradient
c. Intersecting repository

B. Regional Uplift & Subsidence
1. Increased rate of uplift
2. Subsidence

C. Fault Movement
1. Fault within controlled area
a. Within underground facility
b. Outside underground facility

2. Fault outside controlled area
a. Location alters groundwater flow
b. Effects limited to ground motion

II. Climatic

A. Current climate - extreme weather phenomena

B. Increase in frequency or intensity of extreme weather phenomena

C. Glaciation
1. Covers site with ice
2. Causes sea level change

D. Change in precipitation
1. Pluvial period
2. Drier period

III. Human-initiated

A. Greenhouse effect
1. Increased precipitation
2. Reduced precipitation

B. Climate control

C. Weapons testing at NTS

D. Drilling
  1. Intersects canister
  2. Misses canisters

E. Mining
  1. Above underground facility
  2. At or below underground facility

F. Withdrawal well(s) at or beyond controlled area
  1. Small, single-family drinking water well
  2. Large drinking water well (addition to Las Vegas supply)
  3. Agricultural irrigation well

IV. Other

A. Meteorite impact

B. ????

DESCRIPTIONS OF PROCESSES AND EVENTS

Process or Event	Description
1 (I.A.1.a.)	On-site extrusive volcanic activity. A basaltic volcano erupts through the underground facility. The volcano is fed through a dike. Waste canisters within the dike mix with the magma, and their contents are erupted. The size of the dike is assumed to be ___________, which is sufficient to eject from the underground facility _____ % of the originally emplaced waste. This size is the worst credible, and is taken to be representative of all less disruptive events. Three "subevents" are defined, based on the assumed time of occurrence. (a) Subevent 1a, occurring immediately after repository closure, represents all occurrences during the first century after closure, (b) Subevent 1b, occurring at year 101, represents all occurrences between year 101 and year 1,000, and (c) Subevent 1c, occurring at year 1,001, represents all occurrences between year 1,001 and year 10,000. Screening on the basis of likelihood is done only on the overall probability of occurrence of the event during 10,000 years -- not on the probabilities of the subevents. The probability of event 1 is estimated to be _______.
2 (I.A.1.b)	Off-site extrusive volcanic activity. Off-site activity is a likely candidate for screening from the list because potentially detrimental effects seem unlikely. However, the event was retained pending a more thorough consideration of potential effects such as alterations of regional or on-site hydrological or geochemical conditions.
3 (I.A.2.a)	Upgradient intrusive volcanic activity. An igneous intrusion at ____________ (location) upgradient from the underground facility forms in a way that alters groundwater flow downgradient from the location of the intrusion. The intrusion is in the form of a dike with dimensions of _______________________, and reaches to a depth of _____ below the ground surface. The location and dimensions are the worst credible values, in terms of effects on repository performance, and are taken to be representative of all less disruptive intrusions. The temperature of the intrusive material is ______, causing thermal alterations of surrounding groundwater flow conditions. The probability of event 3 is estimated to be _________.
4 (I.A.2.b)	Downgradient intrusive volcanic activity. An igneous intrusion forms at ____________ (location) downgradient from the underground facility. Except for location, this intrusion is identical to that of event 3. The probability of event 4 is estimated to be _________.

5 (I.A.2.c) Volcanic intrusion intersects underground facility. An igneous intrusion identical to that of event 3 forms beneath the underground facility, intersecting emplaced waste, but not reaching the ground surface. The probability of event 5 is estimated to be __________.

6 (I.B.1) Increased regional uplift. The existing rate of uplift at the repository site increases to a rate of _______ immediately after repository closure and then remains constant for 10,000 years. This same uniform rate of uplift also occurs within a surrounding area of dimensions ________________. The probability of process 6 is estimated to be _________.

(I.B.2) Subsidence. Subsidence was screened from the list because potentially disruptive effects could not be identified.

7 (I.C.1.a) Fault movement within underground facility. A fault intersecting the underground facility moves immediately after repository closure, resulting in an offset of _______ along the fault. (Should we specify the type of fault, dimensions, etc?) This event is taken to be representative of all similar events with less detrimental effects on waste isolation. The probability of event 7 is estimated to be ___________.

8 Fault movement within controlled area. A fault (I.C.1.b) intersecting the controlled area, but not the underground facility, moves immediately after repository closure, resulting in an offset of _______ along the fault. This event is taken to be representative of all similar events with less detrimental effects on repository performance. The probability of event 8 is estimated to be _________.

9 (I.C.2.a) Fault movement outside controlled area alters groundwater flow. A fault located outside the controlled area moves immediately after repository closure, altering groundwater flow characteristics in a way that potentially influences waste isolation. The location of the fault is _______ and the offset along the fault is _______. This event is taken to be representative of all similar events with less detrimental consequences. The probability of event 9 is estimated to be _________. (NOTE: If both upgradient and downgradient locations of fault movement capable of altering groundwater flow are credible, separate events might need to be defined analogous to events 3 and 4 above.)

10 (I.C.2.b) Fault movement outside controlled area causes ground motion. A fault located outside the controlled area moves causing ground motion at the underground facility and shaft and borehole seals. The maximum acceleration and the frequency of motion are __________. This event is taken to be representative of all similar events with

lower acceleration or less detrimental frequencies. The probability of event 10 is estimated to be ________. (NOTE: It might be possible to combine events 9 and 10 although, in general, these events will be different since event 9 depends strongly on the location of the fault movement, while event 10 is concerned with the ground motion produced by an event at any location.)

(II.A) Current climate -- extreme weather phenomena. Extreme weather phenomena, such as tornados, hurricanes, etc. were screened from the list because potentially detrimental effects on waste isolation could not be identified.

(II.B) Increased frequency or intensity of extreme weather phenomena. These phenomena were also screened from the list because potentially detrimental effects on waste isolation could not be identified.

(II.C) Glaciation covering site with ice or causing sea level change. Glaciation causing the site to be covered with ice was screened from the list because of lack of evidence of occurrence during previous glacial episodes. Sea level changes caused by glaciation were screened from the list because potentially detrimental effects on waste isolation could not be identified.

11 (II.D.1) Pluvial period. A period of increased precipitation begins immediately after repository closure and continues for 10,000 years. Precipitation at the site and throughout the surrounding region is increased by 50% compared to current levels. This event is taken to be representative of all similar events of later onset, shorter duration, or smaller changes in precipitation. The probability of event 11 is estimated to be ________.

12 (II.D.2) Drier period. A period of reduced precipitation begins immediately after repository closure and continues for 10,000 years. Precipitation at the site and throughout the surrounding region is reduced by 50% compared to current levels. This event is taken to be representative of all similar events of later onset, shorter duration, or smaller changes in precipitation. The probability of event 12 is estimated to be ________.

13 (III.A.1) Greenhouse effect -- increased precipitation. The greenhouse effect causes precipitation to increase by 30% above levels that would have otherwise prevailed. The increase begins immediately after repository closure and continues for 10,000 years. This event is taken to be representative of all similar events of later onset, shorter duration, or smaller changes in precipitation. The probability of event 13 is estimated to be ________.

14 (III.A.2) Greenhouse effect -- reduced precipitation. The greenhouse effect reduces precipitation by 30% compared to levels that would have otherwise prevailed. The decrease begins immediately after repository closure and continues for 10,000 years. This event is taken to be representative of all similar events of later onset, shorter duration, or smaller changes in precipitation. The probability of event 14 is estimated to be __________.

(III.B) Climate control. This event was screened from the list because of low likelihood. It is presumed that the institutional controls required by Part 60 will be sufficiently effective to prevent any events of this type that could detrimentally affect waste isolation.

(III.C) Weapons testing at NTS. This event was also screened from the list by presuming that the institutional controls required by Part 60 will be sufficiently effective to prevent any events of this type that could detrimentally affect waste isolation.

15 (III.D.1) Drilling intersects a canister. Wildcat drilling for petroleum breaches a canister allowing part of the canister contents to be brought to the surface in drilling fluids. Wildcat drilling for petroleum is taken to be representative of all potential drilling at the depth of the underground facility. The frequency of drilling at the repository site is estimated to be __________, and the probability that any one drilling event will breach a canister is estimated by the geometric relationship between the area of the waste canisters and the total area of the underground facility.

16 (III.D.2) Drilling misses canisters. Wildcat drilling for petroleum penetrates the underground facility, but misses all canisters. This type of drilling is taken to be representative of all potential drilling at the depth of the underground facility. The frequency of drilling at the repository site is estimated to be __________, and the probability that any one drilling event will miss all canisters is estimated by the geometric relationship between the area of the waste canisters and the total area of the underground facility.

(III.E.1) Mining above the underground facility. This event was screened from the list because effects potentially detrimental to waste isolation could not be identified.

17 (III.E.2) Mining at or below the underground facility. Construction of shafts and other mining activities are assumed to be carried out only if direct contact with wastes does not occur. If wastes are directly contacted, it is assumed that their character will be recognized, mined openings

will be sealed, and mining activities will be abandoned. The frequency of mine construction is estimated to be ____________, and the probability that mining activities will contact waste canisters is estimated by the geometric relationship between the area of the waste canisters and the total area of the underground facility.

18 (III.F.1) Small water well. A small, single-family drinking water well is assumed to be located at the downgradient boundary of the controlled area and is used as a year-round domestic water supply. The well is assumed to be drilled 100 years after repository closure, and is used continually for the next 9900 years. The probability of event 18 is estimated to be ___________.

19 (III.F.2) Municipal drinking water well. A municipal drinking water well is assumed to be drilled at the boundary of the controlled area at year 100 after repository closure, and the well is assumed to be used until year 10,000 after closure (or until depletion of available groundwater supplies). The effect of this well on repository performance is limited to potential alterations of regional groundwater flow characteristics. It is assumed that current requirements for monitoring the quality of municipal water supplies will continue, so that remedial actions will be taken if radioactive contamination of water supplied by the well occurs. The probability of this event is estimated to be __________.

20 (III.F.3) Agricultural irrigation well. The assumptions regarding this well are identical to those for event 19 except that monitoring for potential radioactive contamination of the water is not assumed to occur. Therefore, remedial actions will not be taken to stop potential releases of waste via this well. The probability of this event is estimated to be ______________.

(IV.A) Meteorite impact. This event was screened from the list because of low probability. Several references in the technical literature demonstrate that the probability of impact by a meteorite large enough to disrupt a repository is extremely small.

7.3 Conclusions

Although only limited progress has been made in applying the selected methodology for scenario development, several tentative conclusions have already been reached.

1. The methodology appears to be workable. The distinction between "internal" and "external" processes and events appears to have merit for determining which uncertainties are to be incorporated directly into the model(s) and data base(s) describing the repository system and which are to be addressed in scenario descriptions. This distinction also appears to be capable of limiting the number of processes and events in the scenario analysis to a manageable level.

2. Scenario descriptions are necessarily only approximate descriptions of future repository performance, and must incorporate significant conservatisms in order to limit the number of scenarios to be evaluated. In particular, the time at which a processes or event is assumed to disrupt a repository may be highly conservative. If such conservatism is excessive, definition of "subevents," as was done for volcanism in this analysis, provides a way to remove conservatism and to generate a more realistic approximation of expected repository performance.

3. No way has been found to ensure completeness of the initial list of processes and events from which scenarios are formed. An approach similar to fault tree analysis, in which the repository system is examined to identify potential failure modes, seems a useful way to check on the completeness of process and event identification.

4. Alternative approaches to scenario analysis, such as those described by Hodgkinson and Sumerling, appear to differ primarily in the degree to which they address uncertainties in the model(s) and data base(s) describing the repository system or in scenario descriptions. The approach selected for this study is intermediate between the extremes proposed by others, and appears to be a reasonable trade-off between the desire for a highly simulation of repository performance and the need to limit resources expended on the simulation. The selected approach also appears to have advantages over alternatives for producing information in a form that corresponds to the needs of the NRC's regulatory process.

7.4 Reference

1. Hodgkinson, D. P., and T. J. Sumerling, "A Review of Approaches to Scenario Analysis for Repository Safety Assessment," paper presented at NEA Symposium on Safety Assessments for Repositories, Paris, 1989.

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8.0

8.0 AUXILIARY ANALYSES SUMMARIES

8.1 Introduction

Generally the auxiliary analyses are directed towards the evaluation of the appropriateness and limitations of various computational approaches and the analysis and interpretation of data being used in this phase of the MOU. These analyses include: the two-dimensional flow simulation of Yucca Mountain, the analysis of hydrologic data, and the analysis of statistical convergence for a CCDF. Addtionally, a separate analysis of carbon-14 releases was performed to supplement the liquid and direct pathway analysis. The above auxiliary analyses are discussed in detail in the Appendices. A brief description of the analysis will be given below.

8.2 Carbon-14 Analysis (Appendix D)

The release of carbon-14 from waste packages is a potentially serious problem due to the presence of a fast pathway (gas through the fractures) to the accessible environment. Due to the complexity of the source term considerations of this problem, the analysis was not considered appropriate to be included in the total CCDF. However, it was considered important to perform some simple calculations to obtain a better appreciation and understanding of the magnitude of the problem and some of the concerns.

The analysis identified release mechanisms and the geochemistry of calcite precipitation as areas where data collection and further investigation would be most fruitful.

8.3 Statistical Convergence (Appendix E)

There are rules of thumb for determining the number of Monte Carlo simulations to perform to provide statistically representative results. Due to the highly non-linear problems currently being tackled, it was deemed appropriate to investigate the number of simulations required to obtain statistical convergence.

Approximately an order of magnitude more simulations than the rule of thumb would indicate were required for the current problem. The most likely reason for this result, was the very few simulations which provided a non-zero result in the high consequence part of the CCDF.

8.4 Analysis of Hydrologic Data (Appendix F)

An auxiliary analysis of hydrologic data was conducted to determine if spatial correlations could be identified for porosity and hydraulic conductivity parameters. This analysis did not identify any spatial correlation with depth for saturated hydraulic conductivity data or for Calico Hills unit porosity data. A large scale trend of decreasing porosity with increasing depth was identified for the Topopah Springs unit and a small scale correlation length of less than 40 meters was identified in data from two holes in the Topopah Springs unit.

8.5 Two-Dimensional Flow Simulation (Appendix G)

A two-dimensional flow simulation was conducted to examine the potential for flow diversion at unit interfaces or the propensity for non-vertical flow. The analysis, which considered only matrix flow, showed that considerable non-vertical flow would occur at interfaces where the saturated conductivity of the lower unit was 75 percent or less of the infiltration rate. Future work will need to consider the effect of fractures on non-vertical flow.

SECTION 9.1

TREATMENT OF SCENARIOS

A general approach for analysis of scenarios is discussed in Section 7. Because work on this part of the performance assessment was delayed, a less systematic approach to the treatment of scenarios was taken in the interest of expediency. In particular, the steps of: (1)identification of processes and events, (2)assignment of probabilities, (3)screening of events and processes, (4)scenario class construction, (5)scenario class probability estimation, and (6)scenario class screening were collapsed into a more direct approach. Because of the limited time available to perform the Phase 1 analysis, significant new modeling initiatives were not possible. With this in mind the staff decided to choose a small number of interesting scenario classes to incorporate in the CCDF to demonstrate how this is done and how results from various scenario classes are combined.

Two classes of fundamental events were selected. These events were (1)changes in weather at Yucca Mountain and (2)human intrusion by drilling boreholes. These types of events were selected, in part, because they would demonstrate interesting aspects of repository performance and because the modeling variations needed to accomodate them were not excessive. Thus treatment of the class of weather changes that could be represented by increased infiltration and a rise in the water table at Yucca Mountain were relatively easily accomodated by a small number of modifications to the data used as input to the model representing groundwater transport. Excavation of radioactivity contained either in the repository or in contaminated host rock could be relatively easily modeled to what is believed to be an acceptable degree of accuracy. In addition, excavation of radioacitivity is an archetypical direct release event, representative of the type of modeling anticipated for similar direct release mechanisms like volcanism.

The two classes of fundamental events combine to form four classes of scenarios:

0. base case, no drilling
1. pluvial conditions, no drilling
2. base case, with drilling
3. pluvial case, with drilling.

Consequences for the base case were represented by the output of the NEFTRAN code as described in 6 and 9.2. The pluvial case was represented by the NEFTRAN code, but with input modified to simulate a higher water table and greater infiltration. Because the drilling removed so little radioactivity from either the repository or the host rock, the consequences of drilling, to a first approximation, could be calculated independent of the detailed migration of radionuclides. However, some of the same factors, such as the removal of waste from the repository, influenced both pathways, so parameters important to these factors were used in calculating releases from both pathways. For scenario classes 2 and 3, consequences from both pathways were calculated and subsequently added together by the system code.

The probability of occurence of drilling was considered to be independent of the occurence of pluvial conditions (see Figure 9.6.1). Although drilling boreholes for purposes of acquifer detection or water extraction probably would depend on the climatic conditions at the site, drilling for purposes of mineral exploration would probably not depend on climate. Following the guidance provided by EPA in Appendix B of 40 CFR 191, a constant drilling rate of .0003 boreholes per square kilometer per year, a repository area of 5.1 square kilometers gives 15.3 as the expected number of boreholes over 10,000 years. This means that the probability of no boreholes at the site over the same time period is very small. Using a Poisson distribution to describe drilling, the probabiltiy of no boreholes is estimated to be 2.3×10^{-7}. Thus the probability of drilling is very close to 1.

Had the scenario analysis procedure discussed in Section 7 been followed for this Phase 1 demontration, the event of no drilling (or alternatively the scenarios involving no drilling) would probably have been screened out, because of its low probability of occurence. Alternatively the two scenarios involving no drilling probably would have been sceened out, again because of their low probability of occurence. These non-drilling scenarios were retained in this analysis for demonstration purposes and because the scenario analysis effort had not progressed sufficiently far to use. An interesting result shown in Section 9.6 is that these scenarios, which would in all likelihood been screened out, have a negligible effect on the total CCDF, which is dominated by the scenarios with drilling. Because there were no readily available data, the probability of occurence of pluvial conditions was assumed to be 0.1 and the non-occurence of pluvial conditions was assume to be 0.9.

The two fundamental events selected for treatment here illustrate the striking differences in the importance of various scenarios to the CCDF that are to be expected when the probabilities of occurence or non-occurence of a particular event (such as drilling or pluvial conditions) are nearly equal or are orders of magnitude different. Also note that the treatment of drilling consequences, in combination with consequences from liquid pathway releases, as a separate pathway depended on the viability of the assumptions that: (1)the amount of radioactivity released by drilling is small compared to the total inventory in the repository and host rock and (2)that drilling boreholes had no substantial effect on the mechanisms important to liquid pathway releases. Had these assumptions not been good approximations, a far more complex treatment of the combination of fundamental events would be necessary.

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9.2

9.2 NEFTRAN Source Term Model

NEFTRAN has several built-in source term models; solubility limited, leach limited, and mixing cell. We have adopted the solubility limited and leach limited models for the present analysis.

Engineered barrier lifetime T_{eb}, a randomly sampled variable in the calculation, is the time before which there would be no radionuclides released. Beyond that time, the waste is assumed to be fully accessible to the environment and can be leached and dissolved. Once exposed to the environment, the radionuclides in the waste are assumed to be contained in the uranium dioxide matrix, and to be released at a rate determined by the "Leach time", T_L, which is the time for the matrix to be totally dissolved at a constant rate. The leach time is simply the reciprocal of the leach rate   . We estimate the leach rate on the basis of the total inventory of the matrix M_0 (metric tons), the infiltration rate I (mm/yr), the total surface area of the site, A (m^2), the fraction of infiltrating water contacting the waste $f(L^0)$, and the solubility of the matrix S_0 (grams/gram water):

$$= R \times A \times F \times S/M_0 \qquad (1)$$

The rate of release of nuclides will be governed by either the dissolution rate of the matrix or the radionuclide itself. Most of the radionuclides must first be released from the matrix before their solubilities become limiting. Since more-oxidized fuel is likely to be more soluble, this solubility may be a function of time. The rate of fuel dissolution might be controlled either by the amount of water entering the canister, or if there is ample water, by the solubility of the fuel determined by its oxidation state.

Once released from the waste matrix, the program determines if the concentration of the radionuclide exceed the solubility limit. If so, then the "undissolved inventory" for that radionuclide increases and the flux leaving the source is limited by the solubility. The undissolved inventory can be released later if the concentration leaving the source term drops below the solubility limit. All variables for the source term model except the initial inventories are random, generated externally to the program by the Latin Hypercube Sampling routine.

Several of the radionuclides, notably C-14, I-129 and cesium, are known to collect outside of the uranium oxide matrix, and could be treated as being solubility limited rather than leach limited. We have determined that for the liquid pathway releases, the fraction of the inventories available for immediate release of these radionuclides are not sufficiently great to make changes to the code to facilitate them, so all of their inventories are assumed to be contained in the matrix. However, we do consider the different inventories for C-14 for the gaseous pathway analysis. (In the present phase of this study, the staff has decided to treat C-14 releases separately from the liquid releases of radionuclides (including C-14). Release and transport of C-14 as a gas are covered in Appendix ~~X~~ D.)

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9.3

9.3 Flow and Transport Models

The movement of radionuclides could occur in the liquid, gas and direct pathways. As discussed in Section 6, the liquid pathway was simulated with the NEFTRAN computer program, and a computer code was developed by the staff to simulate the direct pathway (for this phase of the MOU the direct pathway was a drilling scenario). The gas pathway was analyzed as an auxiliary analysis in Appendix D.

9.3.1 Liquid Pathway

The modified NEFTRAN code was set-up to partially overcome the limitations of the one-dimensional structure (i.e., simulate the spatially varying and uncertain conditions at YMP). This specialized implementation can be divided into the following areas: 1) geometry or network set-up, 2) phenomena, and 3) data input.

9.3.1.1 NEFTRAN Network Implementation

The design of the one-dimensional network for NEFTRAN is based on current information on hydrogeologic units and theories of flow at Yucca Mountain. The SCP conceptualizes the flow at Yucca Mountain as essentially vertical and under steady-state conditions within the matrix for fluxes less than k_s, and as fracture flow at higher fluxes. (The potential for lateral flow at the contact between hydrologic units when a higher-conductivity unit is underlaid by a lower-conductivity unit was examined as an auxiliary analysis in Appendix G.)

Based on the assumption of vertical flow and the fact that the repository is envisioned to have a slope similar to the surrounding geologic unit (see Figure 9.3.1), the analysis was comprised of four separate networks. The network (identified as Columns A, B, C, and D) was designed to represent the hydrologic units existing below a portion of the repository and extending to the water table is depicted in Figure 9.3.2 (this representation takes into account the assumption that one end of the repository is 299 meters above the water table while the other end is 155 meters above the water table and different units exist below these two extremes). Additionally, the areal extent of the repository is rather complex (see Figure 9.3.3). The percentage of waste inventory was partitioned among the four columns based on the areal percentage of the repository determined to be above each column (see Figure 9.3.4). The numerical details of the four columns are presented in Table 9.3.1. Each column, labeled A,B,C, and D, defines the fraction of the waste, the thicknesses of the hydrogeologic units present, and the distance to the water table.

Table 9.3.1 - Columns representing YMP

Column	A	B	C	D	$<K_s>$
Topopah Springs Weld	45 m	60 m	55 m	55 m	0.72mm
Calico Hills Vitric	100	50	10	0	107
Calico Hills Zeolitic	20	70	120	100	0.54
Prow Pass Welded	34	45	10	0	88
Prow Pass Nonwelded	90	20	0	0	22
Bullfrog Welded	10	0	0	0	118
Fraction of waste	0.4	0.33	0.17	0.10	-

There are 6 hydrologic units in column A, 5 in B, 4 in C and 2 in D. Note that in Column D, the only layers present have very small average k_s, and that for high infiltration rates, the transport might be dominated by fracture flow, and therefore potentially high rates of transport to the water table. Column C is only slightly better, with two thin layers of the Calico Hills Vitric and Prow Pass Welded present.

Some limitations of this type of network modeling are:

1. It does not take into account lateral flow caused by the diversion of water along interfaces of units or obstructions of flow near faults.
2. Flow in the saturated zone over the substantial distance along the water table to the accessible environment is conservatively neglected.
3. The source term is conservatively considered to start at the boundary of the disturbed zone, 25 meters lower than the plane of the waste emplacement, and therefore closer to the water table.

9.3.1.2 Implementation of Matrix and Fracture Flow in NEFTRAN

This phase of the MOU is primarily involved with demonstrating the current NRC capability to perform a relatively simple analysis of the of Yucca Mountain. Limitations in a regional flow program such as the inability to account for fracture-matrix interactions, were accepted with the understanding that future work will need to remove these constraints. RES funded work at Sandia National Laboratories, FIN A-1266, will provide improved models in FY 1990. However, the staff deemed necessary improvements and specific implementation of the transport program to provide a credible analyses. The purpose of this section is to describe the modifications and the manner in which NEFTRAN is currently being implemented for analysis of HLW at Yucca Mountain.

The NEFTRAN code was developed primarily for saturated repositories in bedded salt and basalt. It represents groundwater flow and transport in a network of flow tubes. The groundwater flux and transport within each flow tube is considered to be fully saturated and at steady state, with each steady state velocity specified by Darcy's Law. In its present form, it is not well suited for the Yucca Mountain case because flow is not likely to be under saturated conditions and may be transient. The NRC staff made modifications to the NEFTRAN code to facilitate the simulation of unsaturated flow and transport.

First, all coding within NEFTRAN that calculated saturated flux through the flow tubes were eliminated. Instead, the flow rates through the network along the path of radionuclide migration are calculated from the infiltration flux. Second, the staff modified the NEFTRAN code to examine predominant downward bifurcated (fracture and matrix) flow.

The current model does not consider diversions from downward flow caused by high saturation or perching along interbeds or faults. However, flow was considered to be either through the matrix or fractures, depending on the rate of infiltration relative to the saturated hydraulic conductivity k_s. At steady state, flow through the vertical column representing the site would be driven by the infiltration rate. Since the column is one dimensional, all flux must pass through each layer. If the infiltration rate is greater than the saturated hydraulic conductivity of the zone, then the fraction of the infiltration exceeding k_s will flow in the fractures. The flow is therefore bifurcated between the fractures and the matrix. The possible subcases for this flow are described below:

a. infiltration lower than saturated hydraulic conductivity

In this case, the staff assumed that because of matrix suction, water will flow entirely within the matrix, so that the velocity of a non-sorbing tracer without dispersion will be equal to the infiltration rate I divided by the water content ϕ; i.e.

$$v = I/\phi \qquad (9.3.1)$$

The water content is related to the unsaturated hydraulic conductivity through a constitutive relationship. In the present case, the Brooks-Corey formula is assumed:

$$\phi = n_e \, (q/k_s)^{1/\varepsilon} \qquad (9.3.2)$$

where ε is the Brooks-Correy factor for each hydrogeologic unit and n_e is the saturated effective porosity.

b. Infiltration Exceeding Saturated Hydraulic Conductivity

In this case the matrix will be incapable of carrying all the flow, a part of the flow will be carried by the interconnected fractures in the tuff. The matrix portion of the flow would have a transport velocity defined

$$v = k_s/n_e \qquad (9.3.3)$$

The fracture portion of the flow would be:

$$v' = (I - k_s)/n_f \qquad (9.3.4)$$

where n_f is the effective porosity of the fracture. This parameter should also depend on the infiltration rate. However, for the present set of calculations n_f will be taken as a constant, 0.0001, representative of a small value leading to short travel times in fractures (Lin, 1986).

9.3.1.3 Implementation of Transport Phenomena within NEFTRAN

Radionuclides will be transported both in the matrix and in the fractures if infiltration exceeds the saturated conductivity If this were to occur, the matrix flow and the fracture flow would be coupled by hydraulic and chemical processes. The effect of matrix diffusion on the transport through the system would depend on the transfer rate of radionuclides between the fractures and the matrix. The net effect of this transfer can be characterized in three ways, depending on the rate:

<u>High transfer rate</u>

At one extreme, transfer between the matrix and fractures would be high, leading to the concentration in the fractures being identical to that in the matrix. For plug flow (i.e., no longitudinal dispersion in the direction of flow) the rate of radionuclide movement would be the flux divided by the total water content ϕ_T, i.e., the total volume of the void water-filled void space:

$$v = I/\phi_T \qquad (9.3.5)$$

<u>No transfer</u>

At the other extreme, no coupling, the transport in the matrix and fracture pathways would be separate, defined by ~~equations~~ Eqs. 9.3.3 and 9.3.4 respectively.

Since most of the potential for retardation and long travel times are in the matrix, a relatively small fraction of flow in the fracture may completely dominate the transport for the bifurcated flow. Therefore, we include only fracture transport for all cases of bifurcated flow, counting only the fraction of the infiltration carried in the fractures.

<u>Partial transfer</u>

For the intermediate case, the concentrations of the matrix and fracture would be coupled by a process allowing the transfer of radionuclides from the higher to lower potential; i.e., if the concentration of radionuclide in the fracture were greater than in the matrix, there would be transport of the radionuclide into the matrix by molecular diffusion. This phenomenon is generally called matrix diffusion.

By judicious choice of parameters, the NEFTRAN code can be made to approximate matrix diffusion using a simple two-zone model (Van Genuchten and Wierenga, 1976). This model assumes that the water contained in the matrix is essentially immobile, because fracture flow is so much faster. The model accounts for the loss of the radionuclide from the mobile fluid to the immobile fluid by transfer across a boundary between the fracture and matrix. The concentration in the matrix and fracture are assumed to be uniform, and do not vary with distance from the interface. The model is only a rough approximation of true matrix diffusion. It ignore concentration gradients lateral to the direction of flow. It may capture salient features of matrix diffusion for our present purposes, however, and maintains the high efficiency of the code. Transport is proportional to a coefficient B. The NEFTRAN manual suggests that the factor B can be approximated from the average fracture spacing a and effective diffusion coefficient D':

$$B = 2D'/(a/2)^2 \qquad (9.3.6)$$

The model does not consider the additional resistance that could be caused by the presence of surface coatings on the fracture. Since fracture coatings are considered to be the norm rather than the exception, the coefficient B should be reduced to take into account the reduction in transfer caused by these barriers.

For the preliminary analyses of the Phase 1 effort, the effects of matrix diffusion are ignored in the cases where infiltration exceeds k_s (the transport strategy is expressed by the "No Transfer" case). The justification for this approach is:

1. The approach is conservative. Transfer from the fractures to the matrix would retard radionuclide transport.

2. Preliminary screening analyses show that for cases where fracture flow is important, the greatest contribution to dose is likely to come from transuranic elements such as plutonium and americium. These elements are known to have a tendency to form colloids. The molecular diffusion coefficient of colloids is orders of magnitude less than for dissolved molecules and ions, so matrix diffusion is not likely to be effective.

3. Fracture coatings on samples of Yucca Mountain tuffs appear to be substantial, and would lead to a diminished effectiveness of both the diffusive transfer of radionuclides and water flow from the fractures to the matrix.

Lacking experimental data on the actual magnitude and rates of matrix diffusion at Yucca Mountain, it is conservative and prudent to discount matrix diffusion in this initial demonstration.

9.3.1.4 Spatial Variability of flow and transport parameters

In order to maintain a high degree of efficiency in the Monte Carlo analyses with NEFTRAN, we represent the complicated spatially varying repository as four vertical columns, each with a small number of hydrogeologic units through which all of the radionuclides must pass. Existing data on tuff layers at the Yucca Mountain site indicates that there is considerable variations in the material properties. Available data do not support long correlation lengths for the transport parameters at the Yucca Mountain site. The data in many cases suggest small spatial correlation, or none at all on the scale for which they were collected. Using constant values of transport parameters in the models therefore would be inappropriate. Assuming perfect spatial correlation within a unit could for example, lead to a false conclusion that conditions leading to short travel time would apply over the whole unit. In actuality, while short travel time might only apply to a small segment of the column and be countered by the presence of a barrier elsewhere in the column. (this analogy applies to a one dimensional analysis only in which the flow must pass through each segment in series). Lin and Tierney (1986) estimated the GWTT distribution on the basis of analyses with parallel columns, varying the correlation length within the columns by changing the spatial step size. The longer the correlation length, the more spread out was the GWTT distribution:

> "The implicit vertical correlation length (10 feet) of the baseline case is much less that the thickness of any of the hydrogeologic units. This results in a large number of independent random variables (travel times through each of the calculational elements) which are added together to obtain a travel time through a column. Consequently there is a low probability that fracture flow will occur through a large number of elements in any single column from the disturbed zone to water table.....
>
> ....Longer correlation lengths affect the travel time distribution, especially at the tail ends of the distribution, because of the increasing probability of fracture flow through a significant number of elements that make up each of the columns... These results indicate high sensitivity of the travel time distribution to the as yet undetermined correlation length for velocity in each hydrogeologic unit. Generally the sensitivity of the travel times to the correlation lengths suggest how prudent it is to perform a carefully designed testing program for determining the correlation length of all key parameters influencing flow velocities."

Long correlation lengths led to an overly broad CCDF for travel time, with some very short travel times at the tail of the distribution. At the other extreme, the assumption of zero correlation length leads to the conclusion that there is only a single groundwater travel time per column. The determination of spatial correlation scales, especially for k_s is therefore important to the analysis.

9.3.1.5 Effective Values of Flow and Transport Coefficients

The NEFTRAN code simulates flow and transport through a network of connected tubes. For the present case, the staff represents the flow and transport model by up to 6 tubes in series, each tube representing a major hydrogeologic unit; e.g., Topopah Springs welded. Each tube is represented by coefficients expressing its physical properties for flow and transport, namely hydraulic conductivity porosity, cross sectional area, and the retardation coefficients for each of the radionuclides.

Flow can be either in the matrix or fractures, depending on the rate of infiltration relative to the capacity of the matrix to support such flow. Since flow will be vertical and under unsaturated conditions, the primary factor for determining whether the flow in the present analysis is in the matrix or the fracture is the saturated hydraulic conductivity k_s. If infiltration exceeds k_s, then the excess will flow in the fractures.

Geostatistical analyses of the k_s data presented in Appendix F indicate that there is no apparent spatial correlation beyond about 10 meters separation distance, the smallest interval evaluated. If we conservatively assume that k_s is completely correlated at a distance of L meters, then we can represent each tube in the column by a connected series of sub-tubes, each of length L. Each sub-tube has uniform properties, but is uncorrelated to the next subtube in the series. The value of k_s for each sub-tube is picked from the lognormal distribution derived from the available core data presented in Table 9.~~x~~. 4.4.

The analysis is based on the assumption that the flux of infiltrating water passes through each of the sub-tubes. The travel time across each sub-tube, depends on whether the flow is greater or less than k_s:

for $I > k_s$

$$\Delta t_i = n_f \, \Delta l/(I k_s) \qquad (9.3.6)$$

$$\Delta t_{ij} = \Delta t_i \, R_{dj,f} \qquad (9.3.7)$$

for $I < k_s$

$$\Delta t_i = \phi_i \, \Delta l/I \qquad (9.3.8)$$

$$\Delta t_{i,j} = \Delta t_i \, R_{dj} \qquad (9.3.9)$$

where Δt_i = the water travel time for subtube i

$\Delta t_{i,j}$ = the travel time for radionuclide j in subtube i

n_f = the effective porosity of the fractures (taken to be 0.0001)

ϕ_i = the water content of the matrix of subtube i

Δl = the length the subtubes

I = the infiltration rate

d

R_{dj} = the matrix retardation coefficient for radionuclie j

$R_{dj,f}$ = the fracture retardation coefficient for radionuclide j

In this analysis, we consider that flow is either totally matrix or totally fracture flow for each sub-tube of length Δl. Even though there will be matrix flow in parallel with the fracture flow, in practice the fracture transport properties can be demonstrated to overwhelm the contributions of the matrix flow and can therefore be safely left out of the analysis.

We then sum the individual travel times and radionuclide travel times to determine effective values of porosity, ϕ_e, and retardation coefficients, R_{dej} for the main tubes representing the hydrogeologic units:

$$\phi_e = \frac{I \sum_{i=1}^{N} \Delta t_i}{N \, \Delta l} \qquad (9.3.10)$$

$$R_{dej} = \frac{\sum_{i=1}^{N} \Delta t_{ij}}{\sum_{i=1}^{N} \Delta t_i} \qquad (9.3.11)$$

There are two levels of sampling:

1. Within each sub-tube we sample for the values of k_s from a lognormal distribution in order to determine tube-averaged properties of effective porosity and retardation coefficients

2. From realization to realization, we sample the mean and standard deviation of the logs of k_s and the sub-tube length L in order to represent the uncertainty in their values from borehole to borehole.

9.3.3 Gas Pathway

The discussion of this pathway is presented in Appendix D.

9.3.2 Direct (Drilling) Pathway

The analysis for the direct drilling pathway is presented in Appendix H. The drilling analysis used parameters specific to drilling (i.e., frequency of drilling) but all other parametric values were obtained by reading the NEFTRAN input files.

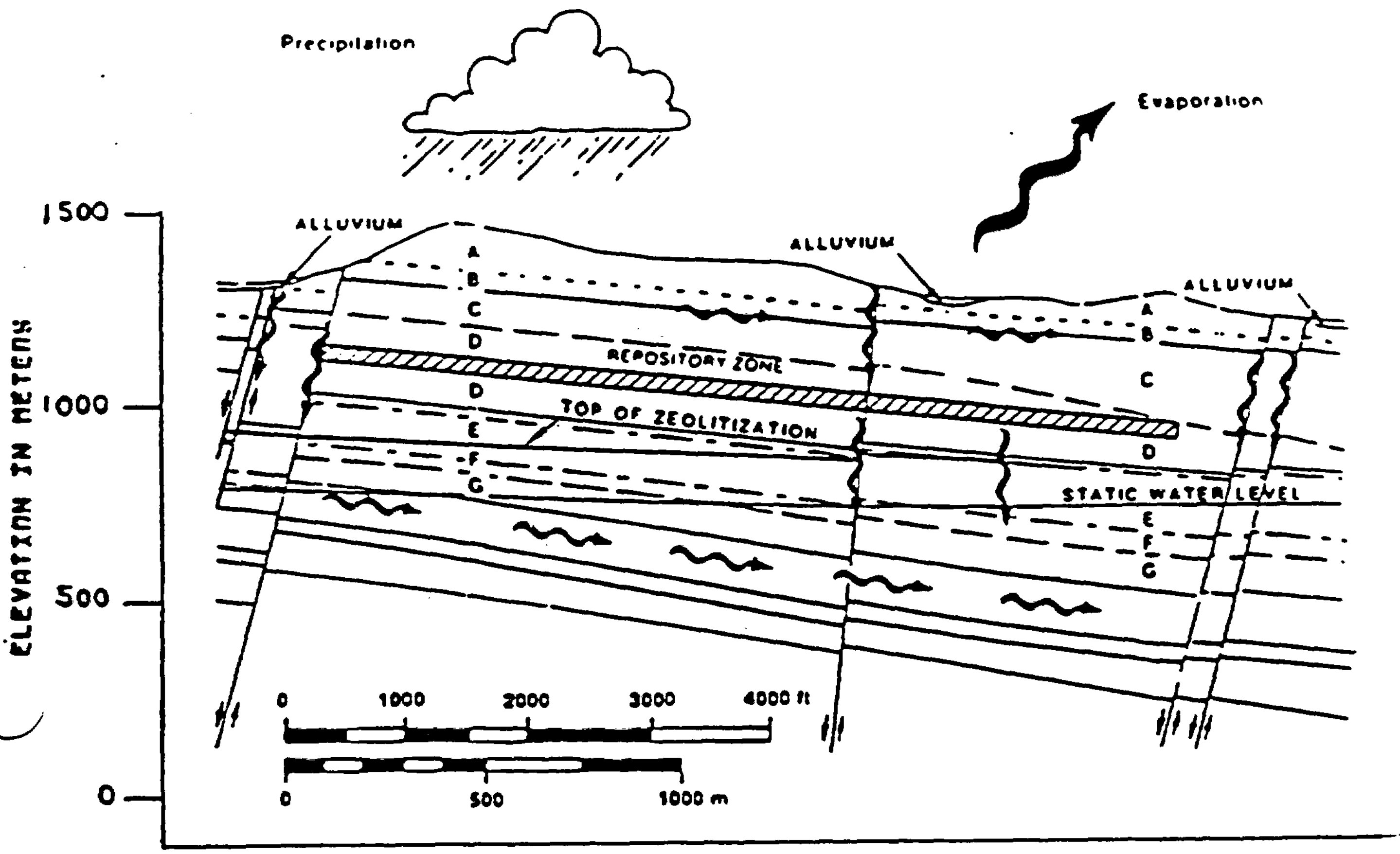

Figure 9.3.1 Conceptualization of the hydrostrategraphic units at Yucca Mountain (A is the Tiva Canyon unit, B is the Paintbrush unit, C is the Topopah Springs #1 unit, D is the Topopah Springs #2 unit, F is the Prow Pass unit, and G is the Bullfrog unit).

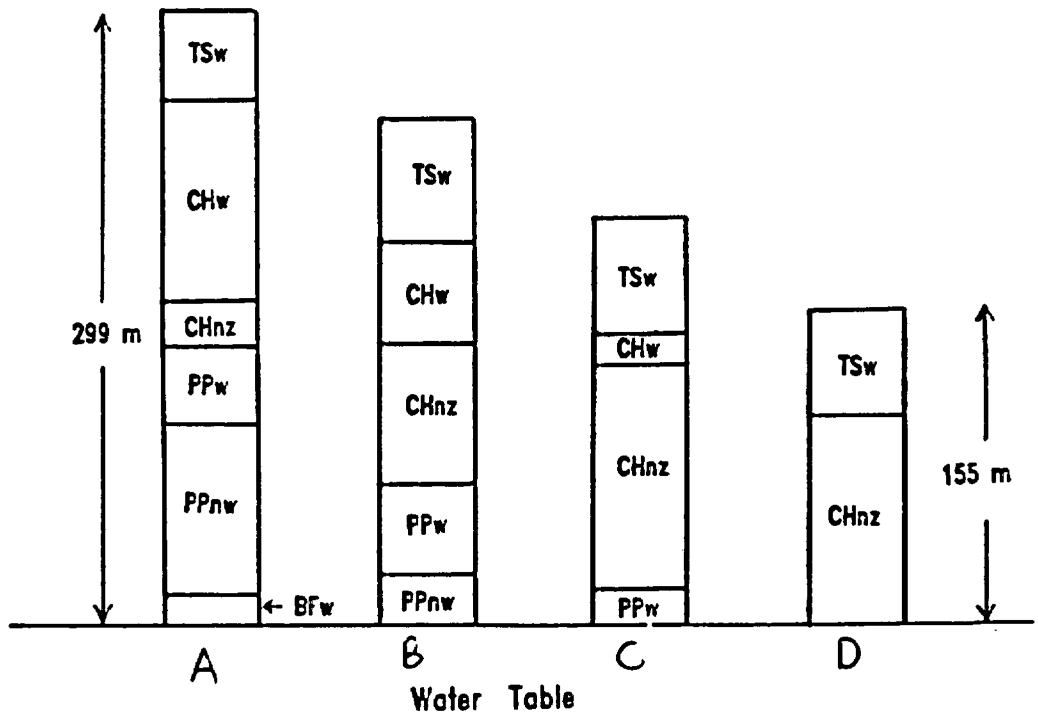

Figure 9.3.2 Hydrostrategraphic units used to simulate the variation in depths and units existing below the repository. The four seperate NEFTRAN representations are identified as A, B, C, and D.

Repository Plan View

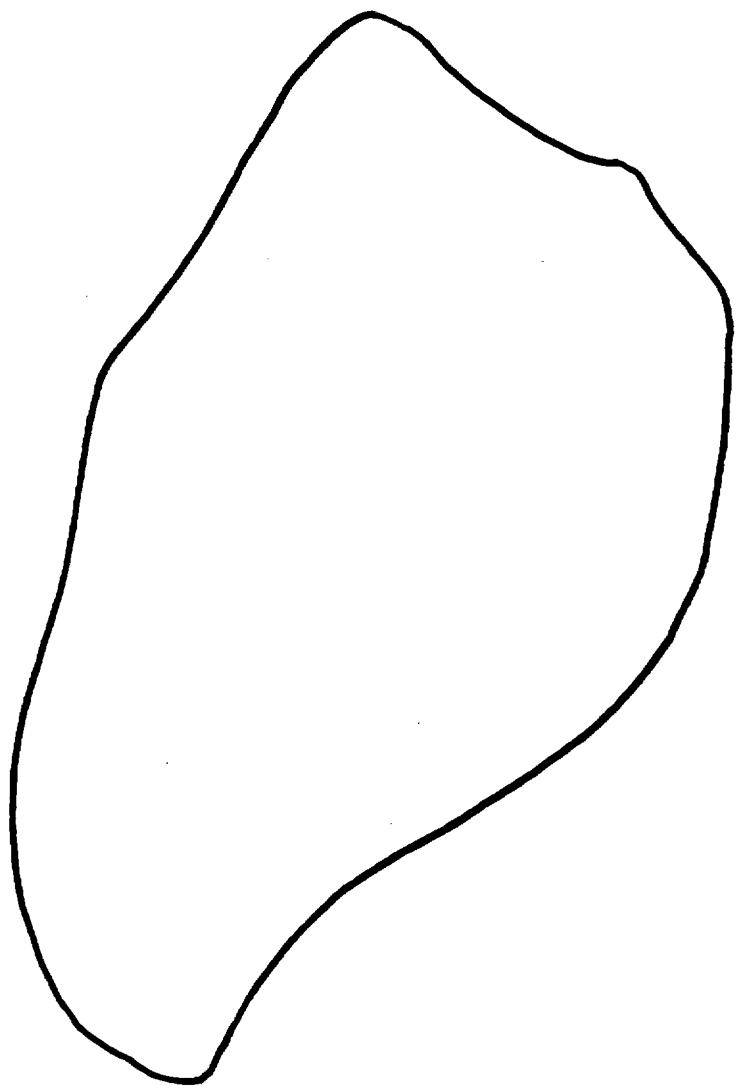

Figure 9.3.3 Areal extent of the repository in plan view.

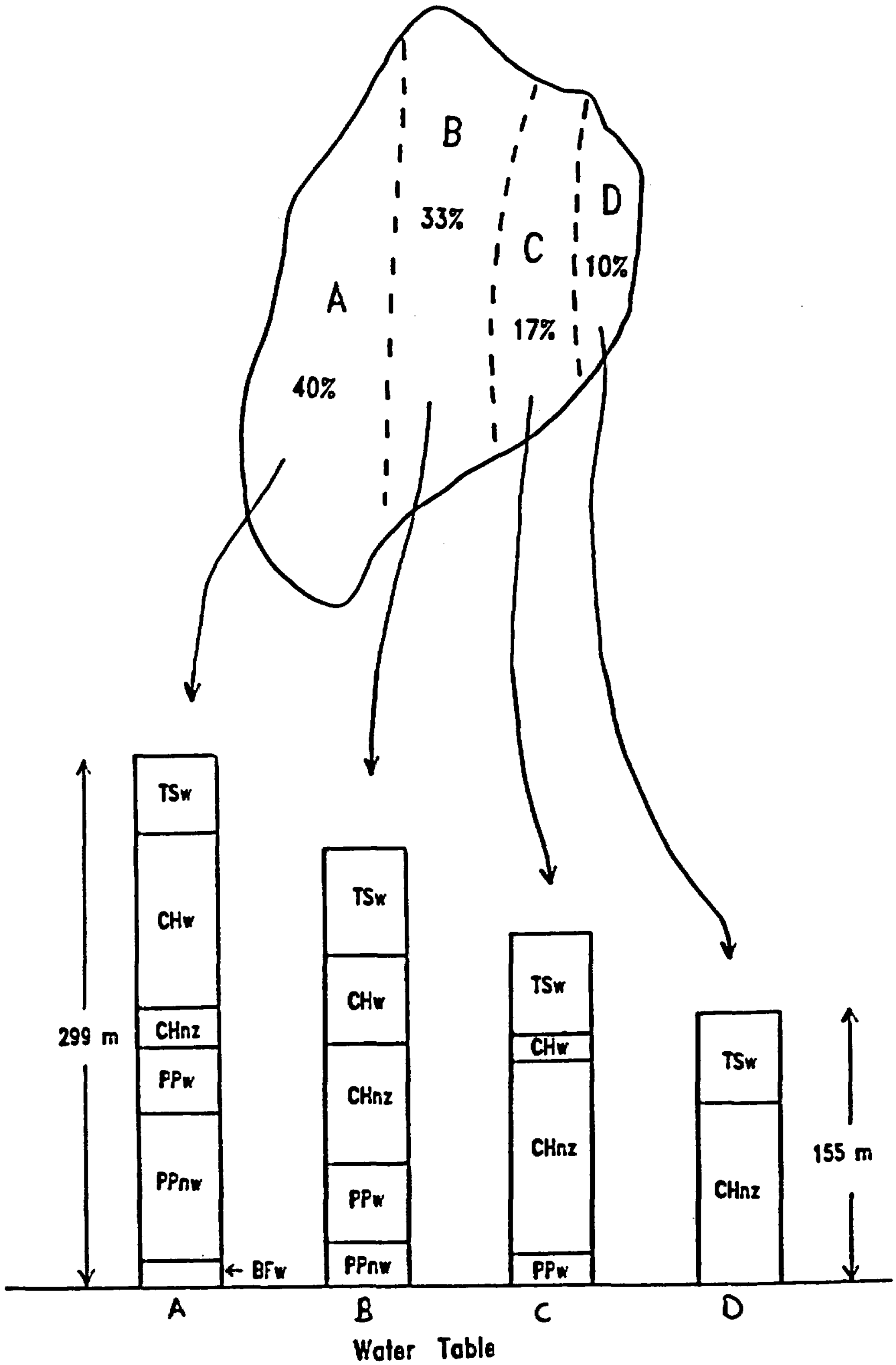

Figure 9.3.4 Representation of the allocation of repository area and inventory of the four NEFTRAN simulations.

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9.4

9.4 Parameters

This section presents the ranges of parametric values used in the liquid and direct pathways. Parameter values used for the gas pathway analysis are presented in Appendix D. The parametric ranges were utilized by the Latin Hypercube Sampling (LHS) program to generate inputs for the source term and transport programs.

9.4.1 Liquid Pathway

Using NEFTRAN to simulate the liquid pathway requires the assignment of the following parameters:

saturated conductivity
porosity
volumetric flux
retardation coefficients
solubility limits
solubility of uranium matrix
waste package lifetime
water contact fraction
dispersivity
correlation length for hydraulic properties

For the liquid pathway analysis, the geologic medium is represented as a series of four vertical units through which all of the radionuclides must pass. Each segment represents a single hydrogeologic unit. Subroutine GETRV in program NEFTRAN contained all of the definitions of source term and transport parameters necessary to make the code emulate the unsaturated flow and transport model.

Inputs to NEFTRAN were generated using the Latin Hypercube Sampling (LHS) program to chose random values from the input parameter ranges. The Latin Hypercube Sampling (LHS) procedure can include correlations between parameters. The input ranges for the NEFTRAN program, as generated by LHS, could include several of the more likely correlations. For the initial calculations of Phase 1, however, correlations between variables are not formally selected in the Monte Carlo input vectors. Several known or suspected correlations are given in Table 9.4.1. Formal inclusion of correlations between variables should be performed in subsequent phases of this study.

Table 9.4.1 - Known and Suspected Correlations

- o Retardation coefficients for similar elements
- o Solubilities of similar elements
- o Solubilities with temperature
- o Temperature of canisters with engineered barrier failure time

- o Uranium matrix decomposition (i.e., oxidation, spallation, dissolution) with waste package failure time
- o Leach rate with infiltration rate and fraction of water contacting waste.
- o Infiltration rate with fraction of water contacting waste form

9.4.2 Sampling Parameters for NEFTRAN Analysis

The staff has collected the parameters necessary for a preliminary analysis of the Yucca Mountain repository from a variety of sources, but primarily published DOE reports, including previous performance assessments for the Yucca Mountain and other repositories. Many of the data are highly uncertain. Nevertheless, the inputs represent the best data available to the staff at the present time. Sensitivity analyses performed following the calculations point out areas where improvement in data would be important in narrowing the ranges of calculated performance. Table 9.4.~~1~~2 shows the input ranges and distributions of parameters for the NEFTRAN and other analyses as generated by the Latin Hypercube Sampling program LHSVAX. The following sections describe the basis for choosing the ranges appearing in the Table.

Table 9.4.~~1~~ 2 - Input to Latin Hypercube Sampling Program

DISTRIBUTION	RANGE	LABEL
NORMAL	100 TO 1000	Engineered Barrier lifetime, years
UNIFORM	1.0E-04 TO 1.0E-03	Sol. of matrix, gm/gm water
NORMAL	0.10 TO 10	Dispersivity, ft
		Infilt. Rate, Ft^3/day
UNIFORM	0.5E+04 TO 0.25E+04	Base Case scenario
UNIFORM	0.25E+04 TO 0.5E+04	Pluvial scenario
UNIFORM	1.0E-04 TO 1.0E-02	Frac. water contact
		Porosity of Matrix
UNIFORM	0.10 TO 0.18	TSw
UNIFORM	0.04 TO 0.14	CHv
UNIFORM	0.28 TO 0.36	CHz
UNIFORM	0.26 TO 0.31	PPw
UNIFORM	0.10 TO 0.18	PPnw
UNIFORM	0.13 TO 0.28	BFw
		Log k_s, mm/yr
UNIFORM	-0.5 TO 0.25	TSw
UNIFORM	-1.4 TO 0.5	CHv
UNIFORM	-0.7 TO 1.2	CHz
UNIFORM	1.4 TO 2.2	PPw
UNIFORM	1.4 TO 2.2	PPnw
UNIFORM	1.5 TO 2.5	BFw
	Standard Deviation of log k_s, mm/yr of log ks, mm/yr	
UNIFORM	0.6 TO 0.75	TSw
UNIFORM	0.7 TO 1	CHv
UNIFORM	0.8 TO 1	CHz
UNIFORM	0.4 TO 0.6	PPw
UNIFORM	0.4 TO 0.6	PPnw
UNIFORM	0.5 TO 0.7	BFw

Table 9.4.2 - Input to Latin Hypercube Sampling Program

(continued)

DISTRIBUTION	RANGE			LABEL
				Retardation Coeff.
UNIFORM	100	TO	1.0E+04	Am
UNIFORM	3000	TO	3.0E+04	Cm
UNIFORM	3	TO	2000	Ni
UNIFORM	5	TO	100	Np
UNIFORM	10	TO	100.	Pu
UNIFORM	0.10E+04	TO	3.5E+04	Ra
UNIFORM	0.20E+04	TO	0.4E+04	Sn
UNIFORM	5	TO	10	Tc
UNIFORM	200	TO	0.50E+04	Th
UNIFORM	5	TO	30	U
UNIFORM	1	TO	1.0E+04	Zr
UNIFORM	20	TO	0.1E+04	Pb
				Solubilities, gm/gm water
UNIFORM	2.0E-10	TO	2.0E-07	Am
UNIFORM	1.0E-09	TO	2.0E-07	Cm
UNIFORM	2.0E-04	TO	1.0E-03	Ni
UNIFORM	2.0E-05	TO	3.0E-04	Np
UNIFORM	5.0E-08	TO	3.0E-06	Pu
UNIFORM	1.0E-08	TO	1.0E-07	Ra
UNIFORM	5.0E-12	TO	5.0E-10	Sn
UNIFORM	0.5	TO	1.0	Tc
UNIFORM	1.0E-11	TO	5.0E-10	Th
UNIFORM	2.0E-11	TO	1.2E-10	Zr
UNIFORM	1.0E-04	TO	2.0E-03	Pb
UNIFORM	20.0	TO	50.0	Corr. length, ft.

9.4.2.1 Waste Package Lifetime

There are no acceptable models to assist us in the choice of the waste package lifetime. The NEFTRAN code is able to accept only a single value of lifetime for each run, even though it is likely that waste package failure would occur in a highly distributed manner.

Waste package lifetime will affect the source term in several ways. First, the package must fail in order for anything to be released at all (although failure does not alone imply that there will be contact between the waste and the water). Second, if the package fails in an essentially dry environment, oxygen from the unsaturated zone will enter, allowing oxidation of the uranium dioxide to commence for a fraction of the fuel rods that have defects. The more-oxidized uranium would have increased solubility over the less-oxidized form. Furthermore, oxidation could cause an increase in volume of the pellets, causing splitting of the cladding and spallation of the pellets and thus possibly increasing surface area. Oxidation might also take place in some of the unfailed canisters because of the presence of small amounts of oxygen, or the dissociation of water caused by ionizing radiation. This radiation could form hydrogen peroxide or nitric oxide which are powerful oxidants. The time to failure of the canister would impact directly on fuel oxidation because the reactions involved temperature and radiation, both of which decrease with time.

For the initial phase of this study, we will simply assume that the fuel solubility is fixed and not a function of time and temperature. Refinements to account for time-dependent oxidation state and temperature may be incorporated into the model in later phases of this study. We also plan to study the isolation afforded the waste by the drying out of the rock.

The staff chose the waste package failure for the liquid pathway analyses as normally distributed with a 0.001 to 0.999 fractiles range for 100 and 1000 years, respectively. For the gas pathway, the staff chose two distributions in order to demonstrate the sensitivity of release of C-14 to waste package lifetime (see Appendix D)

9.4.2.2 Solubility of uranium matrix

Once the canisters and cladding have failed and water penetrates inside, the bulk of the radionuclide release is likely to be from the dissolution of the uranium dioxide waste matrix. The solubility of the waste will be controlled by several factors. Among the more important factors will be the oxidation state of the fuel, which is in turn a function of temperature, oxygen availability and time. We assume that the dissolution rate of the waste is controlled by the rate of disintegration of the uranium dioxide matrix as characterized by a solubility limit (The disintegration of the fuel matrix may not actually be limited by solubility, but by the rate of oxidation). For the present case, the solubility has been chosen to be independent of waste package failure time and temperature and uniformly distributed between 0.0001 and 0.001 grams UO_2 per gram of water.

9.4.2.3 Dispersivity

The dispersivity is a measure of the variance in the transport speed, particularly that caused by variability in material properties. It is not an important consideration in most analyses for cumulative releases. We have chosen the dispersivity to be normally distributed between 0.1 and 10 feet for the 0.001 and 0.999 fractiles respectively.

9.4.2.4 Infiltration Rate

One of the key variables in the analysis is the rate of infiltration which is the main influence on the speed of water movement in the vertical column as well as the amount of water coming into contact with the waste.

Base Case Infiltration

At this time there are no direct measurements of infiltration at Yucca Mountain. Estimates of present day infiltration rates have been calculated from (1) heat flow measurements, (2) precipitation and elevation data, and (3) hydrologic parameters measured from core and in situ in site boreholes. Of the three methods, estimates from hydrologic parameters measured in boreholes are considered generally to be the most accurate, and indicate that the flux through Yucca Mountain is less than 0.5 mm/year, possibly less than 0.2 mm/year. Table 9.3 contains a summary of infiltration estimates cited in the literature. 4.

Most of the previous DOE analyses have employed infiltration rates in the range of 0.1 and 0.5 mm/year. However, because of the considerable uncertainty in the estimates presented in Table 9.3, we have chosen a considerably wider range of infiltration rates. For the base case scenario, infiltration rate is considered to be uniformly distributed between 1.03 and 5.14 mm/year (500 to 2500 cubic feet per day over the total repository area).

Pluvial Scenario

Czarnecki (1985) estimated infiltration for a future pluvial climate scenario for the purpose of calculating the potential rise in the height of the water table. Estimates of future precipitation were based on descriptions of paleoclimates where annual precipitation 12,000 to 9,000 years before present in the modeled area may have been 100 percent greater than modern day annual precipitation. This 100 percent increase in modern-day precipitation was assumed to be the probable maximum increase in the next 10,000 years. He doubled the rainfall estimate of Rush (1970), and then multiplied by the percentage of precipitation occurring as recharge that is associated with that higher precipitation range. He assumed that the increased flux across the northern boundary of the modeled area occurred because of increased precipitation in recharge areas to the north. Vertical infiltration into Fortymile Wash increased because of surface-water runoff from its drainage basin. He calculated that increased recharge from a 100 percent increase in annual precipitation would be 13.7 times greater than estimates of modern day recharge, corresponding to about 7 mm/year infiltration. He also predicted a rise in the water table of 130 meters.

For the purpose of the present study, we estimate the range of infiltration for the pluvial scenario as 5 to 10 mm/year, with an increase in the water table height of 100 meters (We have already chosen a somewhat higher water table than previous DOE analyses).

Draft 11/21/89 A.M.

9.4.3

TABLE ~~1~~ - INFILTRATION ESTIMATES

ESTIMATE	LOCATION	METHODOLOGY	SOURCE
4 mm/yr	Yucca Mt.	Calculated From precip. elevation data	Rice, 1984 Rush, 1970
1-10 mm/yr	Yucca Mt.	Calculated From Drill Hole Thermal data	Sass, 198__
2 mm/yr	Yucca Flat	Calculated From Hydrogeolgic parameters	Winograd, 198__
0.5 mm/yr	Yucca Mt.	Calculated From precip. + Elevation Data	Czarnecki, 1985
<0.5 mm/yr	Yucca Mt.	Calculated From core + Insitu Hydrogeolgic Parameters	& Wilson, 1985
0.5 mm/yr	Yucca Mt.	Calculated From Max. For Matrix k_s Data	Sinnock, 1984
0.1-0.5 mm/yr Tps. ? <1 mm/yr. ?	USW UZ-1	Calculated From core & Insitu Hydrogeolgic Parameters	Montazer, 1985
1E-7-0.2 mm/yr Tps. <1.0 mm/yr	USW UZ-1	Calculated From core & & USW UZ-2 Insitu Hydrogeolgic Parameters	Montazer, 1984

Topopah Springs

9.4.2.5 Fraction of water contacting waste

We characterize the fraction of water infiltrating the site to the fraction actually coming into contact with the waste by a constant factor. The staff performed simple calculations to estimate the fraction of the waste canisters exposed to purely vertical infiltration by taking the ratio of the cross-sectional area of the canisters to the total area of land surface projected by the repository. This ratio is approximately equal to 0.00078. In most cases, infiltrating water could flow around the canisters because of the matrix suction of the unsaturated rock, so this simple figure does not capture the true nature of water contact.

Canisters are likely to be emplaced in the host rock in such a manner as to reduce the likelihood of water coming into contact with the waste. This includes vertical storage and an air gap between the canister and the rock walls. Furthermore, DOE believes that the heat generated by the waste may create a significant zone of dry rock around the canisters, isolating them until cooling of the rock at a later time allows water to rewet the rock. Water may still come into contact with the canisters by several mechanisms:

1. Infiltrating water flowing through fractures and dripping onto the canisters.

2. Loss of the air gap caused by failure of the emplacement holes through mechanical and thermal stresses, or mineral and sediment infilling.

Two additional and potentially important sources of water are (1) lateral inflows from areas of perched water and (2) liquid water circulation caused by heat-driven evaporation and condensation. Lateral inflows would be possible but are unlikely to affect more than a few of the canisters, since the water necessary for this phenomenon to be viable would be diverted from the vertical infiltration available for all canisters. If such a diversion was possible, some canisters might get a greater share of the overall infiltration at the expense of the remaining canisters being exposed to less water. Liquid water circulation caused by heat is potentially important, and is discussed further in Section 5, Source Term.

For the preliminary analysis in this phase of the work, we have chosen the water contact fraction to be 0.002 to 0.01. The analysis in the Environmental Assessment (DOE, 1986) assumed a contact fraction of 0.025, but they specified no basis for this choice.

9.4.2.6 Saturated Hydraulic conductivity

Water flow in the unsaturated fractured rock could proceed through both the matrix of the rock at low rates of infiltration or through the fractures and the matrix at higher rates of infiltration. The switchover from matrix flow to flow in the fractures is probably related to the saturated hydraulic conductivity of the rock matrix. Statistical evaluation of the k_s data presented in Appendix F indicate that it is lognormally distributed. Table 9.4 summarizes the available data on saturated hydraulic conductivities from rock cores at the Yucca Mountain site in terms of its log means and standard deviations where there are sufficient data available.

9.4.2.8 Spatial Correlation of Saturated Hydraulic Conductivity

Geostatistical analyses of the k_s presented in Appendix F indicate that there is no apparent spatial correlation of the core data on saturated hydraulic conductivity of the matrix above the minimum separation distance of 10 meters used in the analysis. Since larger correlation scales are conservative, we assume that there is a correlation scale between 20 and 50 feet. There are insufficient data to determine the distribution of the mean and standard deviation of k_s for, so we assume that they are uniformly distributed from the values calculated for each unit and each borehole. The mean, standard deviation and correlation length of k_s are used to chose representative hydraulic coefficients for each hydrogeologic unit as described in Section 9.3.1.

Table 9.4.~~3~~4 - Log mean and standard deviation of k_s

Unit	Mean of log k_s mm/yr	S.D. of log k_s mm/yr
BFnw	2.22	
	1.38	
	1.71	0.59
BFw	2.08	
CHnv	-1.32	
	0.47	
	0.07	
CHnz	1.16	
	-0.65	0.87
PP	1.44	
	2.09	
TSw	0.22	0.72
	-0.45	0.61

9.4.2.8 Porosity

There are probably more porosity data available from core taken at the YMP site than any other type of data. As used in this study, water velocity and radionuclide transport speed in the matrix are tied closely to the average value of porosity for the columns. We have chosen the porosity ranges from available data averaged over each unit. There are insufficient data to determine the distributions of the average properties, so the averages are assumed to be uniformly distributed. Representative values of porosity for each hydrogeologic unit are sampled from the distribution of mean porosities. Shown in Table 9.5

Table 9.4. 9.5 - Mean Porosity for Units

Unit	Mean Porosity
BFnw	0.2
	0.22
	0.25
BFw	0.13
	0.28
CHn	0.36
	0.2
	0.28
	0.34
	0.29
PPn	0.29
PPw	0.31
	0.31
	0.26
TSw	0.11
	0.13
	0.1
	0.11
	0.18

9.4.2.9 Retardation coefficients

The staff chose values of retardation coefficients for the matrix to reflect reported values for batch and column tests performed by DOE (Refs). For the key radionuclides plutonium and americium, values are chosen on the low end of the range in order to account partially for data that indicate that these substances do not have behave simply, tend to form colloids, and may be difficult to predict under repository conditions. We should hasten to add however that much of the data in column experiments that indicated low retardation for some elements was collected for flow rates 3 to 4 orders of magnitude greater than are likely at YMP, and therefor may be misleadingly pessimistic. Furthermore, sensitivity of total releases to retardation coefficients for plutonium and americium are weak, indicating that factors such as low solubility and long half life may be more important than retardation for these nuclides. Values used in this study are typical of those used previously in Yucca Mountain performance studies (Refs)

Retardation coefficients for the fractures were taken from the study by Lin (1986), and are orders of magnitude smaller than the matrix retardation coefficients. In either the matrix or fracture cases, there was no distinction made for retardation coefficients between different hydrogeologic units. Those units that have low values of saturated hydraulic conductivity however will tend to lower values of effective retardation coefficients because of the greater proportion of the flow to be expected in the fracture zone, as calculated by the procedure presented in Section 9.3.1.5.

9.4.2.10 Solubilities

We have taken the solubilities of radionuclides from several sources, but primarily DOE references, including several preliminary performance assessments. Values used reflect those reported in previous performance assessments from Yucca Mountain.

9.4.3 Direct (Drilling) Pathway

The drilling program was developed to calculate the consequences from the expected number of boreholes intercepting the repository (see Appendix H). The following parametric values were needed: drilling rate, size and number of waste packages, area of repository, time of drilling, and the radius of the borehole. Additionally, the following parametric values from the liquid pathway were used: time of waste package failure, volumetric flux, water contact fraction, and solubility limits (these values were discussed in the above Section and will not be discussed here).

Based on conceptual repository designs the dimensions of the repository system were set as follows: area of repository = 5.1 square km, number of waste packages =18,000, borehole radius = 6 cm, waste package radius = .34 m, and waste package length = 4.3 m. The time for commencement of drilling was set to a arbitrary value of 100 years and the drilling rate was set to .0003 drillings per square km per year based on EPA average drilling rates (reference).

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9.5

9.5 SENSITIVITIES AND UNCERTAINTIES FOR LIQUID PATHWAY ANALYSIS

9.5.1 Introduction

This section covers the sensitivity and uncertainty of the liquid pathway calculations on a scenario by scenario basis. We present the complimentary cumulative distribution functions (CCDF's) for the Base Case and Pluvial scenarios which take into account the uncertainty in the values of the coefficients for each scenario, but not the scenario probabilities. We also present the sensitivity to variations in parameters using rank regression and ad hoc variations of single parameters, including those parameters relating to the NRC guidelines of 10CFR60.113. Total system results, which also take into account the scenario probabilities, are covered in Section 9.5, but we have not performed formal sensitivity and uncertainty analyses on these results.

9.5.2 Statistical uncertainty analysis

An important part of conducting a performance assessment of a waste repository for high level waste is quantifying the uncertainties associated with the probabilities of occurrence of credible scenarios and those associated with the offsite and onsite consequences (both radiological and nonradiological).

Many risk and environmental impact assessments apply single or best estimate values for model parameters and assert that these valuations are reasonable and conservative (i.e., lead to overpredictions) without quantifying the degree of conservatism inherent in the assessments. A variety of techniques is available to quantify the uncertainty in complex models for assessing radiological impact upon man and the environment that may include nonlinearities and time-varying phenomena (1,2). These include: the Monte Carlo (Helton, 1961), fractional factorial design (Cochran, 1963), Latin hypercube sampling (Cranwell, 1981, Iman, 1979, McKay, 1979), response surface (Meyers, 1971), differential sensitivity analysis, (e.g., adjoint (Baybutt, 1981, Oblow, 1978, Cacuci, 1980)) and Fast Probabilistic Performance Assessment (CNWRA, 1988) methodologies. A preferred technical approach would be flexible, economical to use, easy to implement, provide a capability to estimate an output distribution function and rank input variables by different criteria.

9.5.2.1 Latin Hypercube Sampling

In this study the Latin hypercube sampling (LHS) scheme was chosen to be implemented on the flow and transport model in the performance assessment of the high level waste repository. The advantages and properties of the Latin hypercube sampling techniques are:

- The full range of each input variable is sampled and correlation coefficients between all pair-wise input variables can be specified.

- It provides unbiased estimates of the parameters (means and variances) of cumulative distribution functions and means for model output under moderate assumptions.

The LHS method is a member of the class of sampling techniques which include Monte Carlo and stratified random sampling. Several risk assessments for nuclear waste repositories (Campbell, 1979) have applied LHS techniques. Furthermore, LHS has been applied to the model for atmospheric transport of reactor accident consequences and recently used for the severe reactor accident calculations in NUREG-1150 (NRC, 1989). We remark that one may wish to distinguish between different types of uncertainty associated with modelling of physiochemical processes - in particular:

- The statistical uncertainty due to inherent random nature of the processes, and

- The state of (perhaps "lack-of") knowledge uncertainty.

This latter state-of-knowledge uncertainty may be further subdivided into model and parameter uncertainty. The parameter uncertainty is due to insufficient knowledge about what the input to the code should be. This study documented herein deals primarily with parameter uncertainty. The modeling uncertainty is due to simplifying assumptions and the fact that the models used may not accurately model the true physical process.

As shown in Table 9.5.1, first a set of key parameters in the model under study needs to be identified. For each chosen variable, a set of quantitative information is developed regarding the range of variation, probability distribution, as well as, correlations among the variables. For our study, we did not use any correlations between input variables. The data input to the LHS program is given in Table 9.4.~~1~~.2, which shows the distribution and range of input for each variable. The basis for choosing these inputs is discussed in Section 9.4. This information is used as input to the Latin hypercube sampling code (Iman, 1984a,b). LHS is used to generate what is called a design matrix. Specifically, if N computer runs are to be made with k parameters under study, the design matrix has dimensions N x k. Each row of this matrix contains the input valuations of each of the chosen k parameters (independent variables) for the N computer runs. Experience has shown that N equal to (4/3)k is often a sufficient number of samples to generate a CCDF (Sandia,), but in our particular case many more samples were needed for statistical convergence (See Appendix 2.2).

9.5.3 Ad Hoc Sensitivities

In this section, we present results of the NEFTRAN runs for the base case and pluvial liquid pathway scenarios with the intent of demonstrating the effects of individual variables on the resultant cumulative radionuclide releases to the accessible environment.

The NEFTRAN computer code as modified for the Yucca Mountain case was run for the base case scenario to calculate cumulative releases for either 10,000 years or 100,000 years, and the pluvial scenario for 10,000 years. For each simulation, we generated a list of 47 variables using Latin Hypercube Sampling. The list of variables for each simulation is called a "vector". The input constants, ranges and distributions for generating the vectors are presented in Tables 9.2.

9.5.3.1 Sensitivity to Infiltration

Figures 9.5.1 and 9.5.2 show the resultant conditional CCDF's for the base case scenario at 10,000 and 100,000 years, respectively. Also plotted on these figures are CCDF's composed only from vectors having infiltration rates less than limits set at 2.0 or 1.0 mm/yr in order to demonstrate the particular significance of this parameter to repository performance.

The great sensitivity to infiltration rate can be partially explained by the next two figures. Figures 9.5.3 and 9.5.4 show the CCDF's for the base case scenario at 10,000 and 100,000 years respectively, comparing the contribution of column D to the contribution from all 4 columns. Column D contains just 10% of the waste, but has the shortest pathway to the water table. In addition, column D contains just two units; the Topopah Springs welded and Calico Hills zeolitic. Both of these units have relatively low saturated hydraulic conductivities, k_s, which would make them prone to fracture flow for higher infiltration rates. Fracture flow leads to both short travel times for liquid water and low retardation coefficients. Figure 9.4.3 shows the effect most dramatically, where virtually all of the contribution to the high-impact portion of the curve would be caused by Column D alone. Retarded radionuclides have not yet started to arrive from the other columns. Travel times through the other three columns would be too long to contribute much to the CCDF within 10,000 years. Figure 9.4.4 shows that more of the contribution to the CCDF comes from the other three columns over the 100,000 year period, because the long-lived radionuclide start arriving.

Figure 9.5.5 shows the CCDF for the pluvial scenario. In this case, the water table is shallower and infiltration rates are higher than the base case scenario, so travel times are shorter for all columns. Relatively more of the cumulative EPA ratio comes from column A, B and C than was the case for the base case scenario. These scenarios are not directly comparable however, because long computer run times led to the necessity of reducing the number of vectors from 500 to 98 for the pluvial scenario.

9.5.4 Sensitivity Analysis using Regression

The next step in the process involves performing a sensitivity analysis on the calculated results. The aim is to determine and quantify the relative contributions of the kth variable toward the output variability. Sensitivity analyses can be very fruitful in preliminary studies such as this one, since it can help to identify which parameters and models should be refined in future studies. In addition, sensitivity analyses may allow the analyst to check his intuition about the importance of the parameters and phenomena of the model, and lends insight on the meaning of the modeling.

Sensitivity can be determined by performing step-wise linear regression analyses on either the raw results of the model analysis (i.e., the EPA ratios) or the ranks of the raw results (i.e., replacing the "raw" data values by their ranks). Ranks may be preferred when highly nonlinear relationships are present between the model outputs and inputs, but the correlations obtained have less significance than those using the raw data. Both graphical analyses and statistical distribution fitting procedures may also be useful in identifying patterns in the data. The present report shows only the regression analyses on raw results; i.e., EPA release ratio.

We analyzed the sensitivity of the cumulative release for several cases using a modified version of the STEPWISE program from Sandia National Laboratories. We modified the STEPWISE program to read the data file of input vectors generated by the LHS sampling procedure and the combined results for columns A through D generated by NEFTRAN for those inputs. The regression coefficients are presented in Table 9.5.2 for the base case and pluvial scenarios. There were 500 vectors for the base case scenario, but because of excessively long run times, there were only 98 vectors for the pluvial scenario. This led to more equivocal results for the pluvial scenario. We chose to show only those regression coefficients with the highest significance, or in some cases parameters that would be important for their apparent lack of sensitivity.

The sensitivity analyses proved to be very revealing, both for the sensitivities to some parameters and apparent lack of sensitivities to others. The most consistently sensitive parameters seem to be contact fraction, infiltration rate, solubility of the matrix, and the saturated hydraulic conductivity of the Calico Hills vitric unit. Of these, high values of infiltration rate and saturated hydraulic conductivity lead to fast fracture flow pathways and low retardation in Column D, which contributes most of the releases in the base case scenario.

9.5.5 Average Importance of Radionuclides

We also evaluated the average contribution of the radionuclides to the EPA ratio for the scenarios. This parameter was calculated by taking the average contribution by radionuclide to the EPA ratio for all vectors. We present the average contribution by radionuclide in Table 9.5.3 for the base case scenario at 10,000 and 100,000 years, and the pluvial scenario at 10,000 years. In addition, we present further results for the base case scenario including only those vectors that have infiltrations less than 1.0 mm/yr or 2.0 mm/yr to demonstrate sensitivity of the result to infiltration. The isotopes Pu-239 and Pu-240 stand out as the most important contributors to the EPA ratio because of their large inventory in the source term, long half lives and potentially low retardation in the rock. Nearly all of the contribution of these radionuclides comes from inventory in the source term rather than from chain decay of heavier radionuclides (e.g., Am-243). Other radionuclides are important in a few cases. I-129 appears for the 100,000 year case with infiltration of less than 1.0 mm/yr because of its exceedingly long half life. The isotopes I-129, C-14 and Tc-99 would take on high relative importance if the groundwater flow were always restricted to matrix rather than fracture flow. This would have been the case except for column D for the base case scenarios, as the saturated k_s of most of the units in the other columns was sufficient to assure retention of most of the significant but retarded radionuclides.

9.5.6 <u>Sensitivity to NRC Performance Criteria</u>

NRC defines a set of performance criteria for particular barriers in 10CFR60.113:

"60.113(a)1(ii)(A) Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission.....that such period shall not be less than 300 years nor more than 1,000 years after permanent closure of the geologic repository...."

"60.113(a)1(ii)(B) The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of the radionuclide calculated to be present at 1000 years following permanent closure......."

"60.113(a)2 The geologic repository shall be located so that the pre-waste-emplacement groundwater time along the fastest path of likely radionuclide travel from the disturbed zone to the accessible environment shall be at least 1000 years......"

These limitations imposed by NRC have the intent of providing a set of criteria for the repository independent of the EPA release limits specified in 40CFR191, and prevent reliance on a single barrier to the release of radionuclides to the accessible environment.

9.5.6.1 Effects of NRC Performance Criteria on CCDF

We examined how compliance with the NRC standards relate to the outcome of the performance assessment calculations in terms of compliance with the cumulative release limits. The conditional CCDF for the base case scenario was recalculated by using the original set of 500 input vectors and output releases, but screening out those vectors which did not comply with the NRC criteria stated above. The subset of vectors that "passed" the criteria were then used to plot a CCDF and compared to the CCDF plotted from all of the vectors for the base-case scenario unconditionally. The screening procedure is described below:

Substantially complete containment - Vectors with engineered barrier lifetime less than a specified time were screened out. For the sake of this demonstration, we chose only a single representative cutoff time of 500 years.

Release rate limitation - The release rate model in NEFTRAN looks only at the congruent release of radionuclides contained in the uranium dioxide fuel. The maximum rate is controlled by the dissolution rate of the matrix. The NRC performance criterion was specified as less than 10^{-5}/yr of the radionuclide inventory present at 1000 years. For this demonstration, we assumed that the release rate was equivalent to the dissolution rate of the matrix. Releases of some of the radionuclides might actually be smaller than the congruent dissolution rate because they are solubility limited, so the screening criterion might be slightly overrestrictive. The dissolution rate calculated in NEFTRAN is a function of uranium solubility, infiltration and water contact fraction.

Groundwater travel time limitation - The model is based on the assumption that transport occurs in four separate pathways, columns A, B, C and D, in order to partially simulate the spatial variability inherent in the Yucca Mountain repository. Clearly, column D is both the shortest pathway and the one most likely to saturate, with correspondingly faster flow and lower retardation. Therefore we take "groundwater travel time along the fastest pathway of likely radionuclide travel" as the mean travel time along column D. Groundwater travel time is a function of infiltration rate, porosity, saturated hydraulic conductivity and correlation length.

Figure 9.5.6 shows the conditional CCDF for the base case scenario for unrestricted vectors, and vectors limited by either waste package lifetime or release rate or groundwater travel time. It assumes no relationship between waste package lifetime and engineered barrier system release rate. For the present case, all 500 vectors had release rates less than 10^{-5}/yr, so that CCDF curve is coincident with the unrestricted curve. There is a significant benefit shown for an engineered barrier lifetime of 500 years or greater, but only for the low probability, high impact releases.

The most dramatic effect is for the screening on the basis of groundwater travel times. All of the high-impact release were essentially eliminated when travel times shorter than 1000 years were eliminated from the CCDF. The explanation for this effect is that flow along column D is controlled by fractures for infiltration rates higher than the saturated hydraulic conductivity. Fracture flow is both faster and leads to conditions of lower radionuclide retardation. Eliminating the cases leading to saturation allows only releases through the rock matrix under unsaturated conditions, with commensurately greater retardation.

9.5.6.2 Average Contributions by Radionuclide

Table 9.5.4 illustrates the average contribution by radionuclide for the unrestricted releases and the releases complying with the NRC performance criteria. All cases except the one restricted by groundwater travel time show the main contributions coming from isotopes Pu-239 and Pu-240, which would be expected to be retarded in the matrix and greatly attenuated. For the releases restricted by the 1000 year groundwater travel time however, the main contributors are the radionuclides C-14 and I-129, which are unretarded and can therefore move relatively quickly through the matrix.

Table 9.5.4 - Relative Radionuclide importance for Unrestricted Vectors and those Restricted by NRC Performance Criteria

Radionuclide	Unrestricted vectors	500 yr W.P. Lifetime	1000 yr. GWTT
Pu-240	0.41	0.40	0
Pu-239	0.39	0.37	0
C-14	0.094	0.13	0.94
Am-241	0.077	0.062	0
Am-243	0.014	0.014	0
I-129	0.005	0.007	0.05

9.5.6.3 Ad Hoc Sensitivities to NRC Criteria

We also plotted the results of the 500 runs versus the values of the individual NRC criteria of groundwater travel time, waste package lifetime and release rate from the engineered barrier. The results, shown in Figures 9.5.7, 9.5.8, and 9.5.9 all demonstrate that imposing the NRC criteria would have a favorable impact on the total releases to the accessible environment. Imposing the 1000 year groundwater travel time limitation virtually eliminates any non-compliance with the EPA containment requirement. None of the vectors yielded release rates from the engineered barrier that exceeded the NRC limit of 10^{-5}/yr, but the EPA release increases with increasing engineered barrier release. There was also a noticeable decrease in EPA release with increasing engineered barrier lifetime.

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Table 9.5.1
Steps to Perform Uncertainty and Sensitivity Analysis

- o Maximum-Minimum Ranges of Probabilities
- o Correlation Matrix

2. Run Latin Hypercube Sampling Code

3. Run Source Term and Flow and Transport Models

4. Statistical Analysis

    Fitting Distributions
    Regression Analysis
    Graphical Display and Analysis

Table 9.5.2
Regression of YMP Liquid Pathway Cumulative Releases
(Raw data correlations)

Variable	Base Case 10,000 yrs	Base Case 100,000 yrs	Pluvial 10,000 yrs
W.P. LIFETIME	-.045	-.049	-
SOLUBILITY UO_2	0.09	0.13	0.32
INFILTRATION	0.1	0.31	0.23
CONTACT FRACTION	-	0.18	0.44
MEAN LOG K_S TSW	-	-.11	-
MEAN LOG K_S CHNZ	-.14	-.22	-.28
RD CM	-	-	-.2
RD PU	-	-0.23	-.22
RD RA	-	-	0.18
SOL. CM	-	-	0.17
SOL. PU	-	-	-.27
CORR. LENGTH	0.11	-	-

STAT 11/16/89

Table 9.~~4~~5.3 - Average Importance of Radionuclides to EPA Release Limits
(Only if greater than 0.01 contribution, bold if greater than 0.05)

Radionuclide	Base Case	Base Case	Base Case	Base Case	Base Case	Base Case	Pluvial Case
Time	10^4 yr	10^4 yr	10^4 yr	10^5 yr	10^5 yr	10^5 yr	10^4 yr
Infilt.	<5.14 mm	<2.0 mm	<1.0 mm	<5.14 mm	<2.0 mm	<1.0 mm	
Am-241	**0.077**	**0.061**	**0.069**		0.014	0.017	0.069
Am-243	0.014	0.016	0.016			0.013	
C-14	**0.094**	0.013		0.015	0.031	**0.061**	
I-129	**0.05**				0.037	**0.229**	
Np-237	0.01			0.015	0.014		
Pu-238	0.010						
Pu-239	**0.39**	**0.438**	**0.438**	**0.726**	**0.589**	**0.183**	**0.443**
Pu-240	**0.41**	**0.463**	**0.465**	**0.069**	**0.181**	**0.442**	**0.459**
Pu-241	0.02						
Pu-242				0.024	0.011		
Tc-99				0.016	0.022	0.013	
Th-230					0.011	0.011	
U-233				0.012			
U-234	0.02			0.048	0.034	0.010	
U-236				0.026	0.018		
U-238				0.024	0.018		

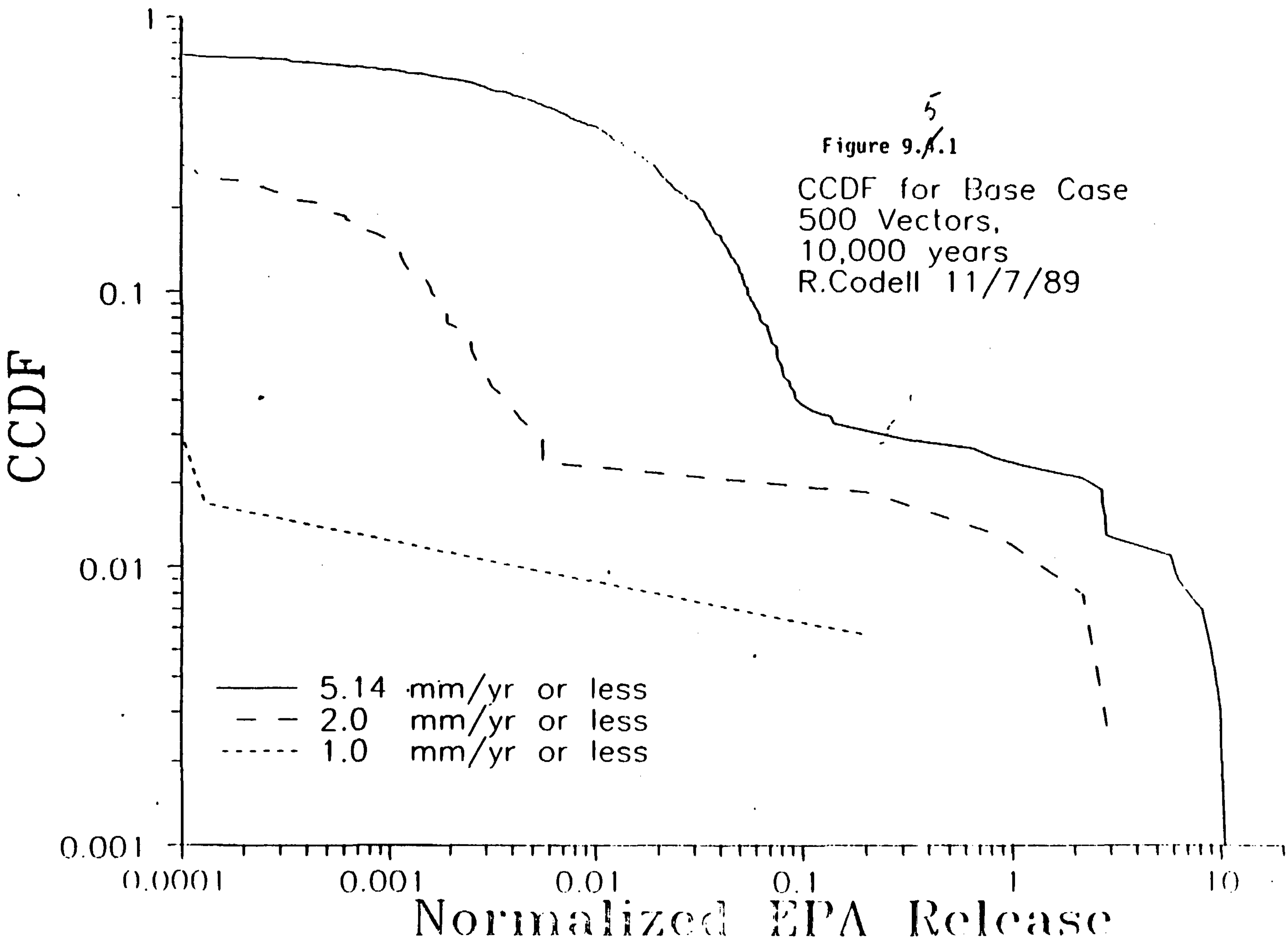
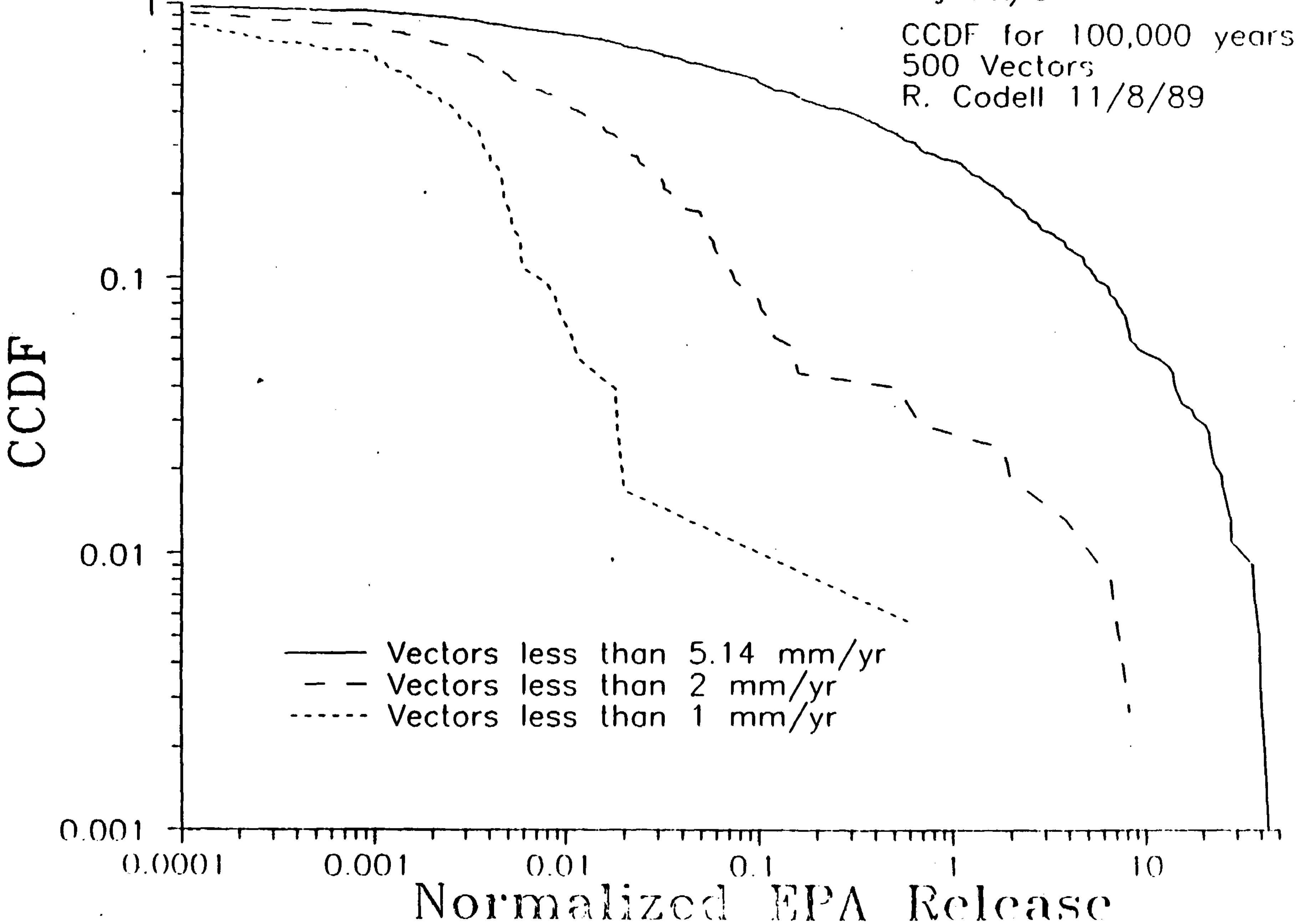


Figure 9.5.1
CCDF for Base Case
500 Vectors,
10,000 years
R.Codell 11/7/89
CCDF
1
0.1
0.01
0.001
0.0001
0.001
0.01
0.1
1
10
Normalized EPA Release
5.14 mm/yr or less
2.0 mm/yr or less
1.0 mm/yr or less

Figure 9.~~4~~5.2



CCDF for 100,000 years
500 Vectors
R. Codell 11/8/89
CCDF
1
0.1
0.01
0.001
0.0001
0.001
0.01
0.1
1
10
Normalized EPA Release
Vectors less than 5.14 mm/yr
Vectors less than 2 mm/yr
Vectors less than 1 mm/yr

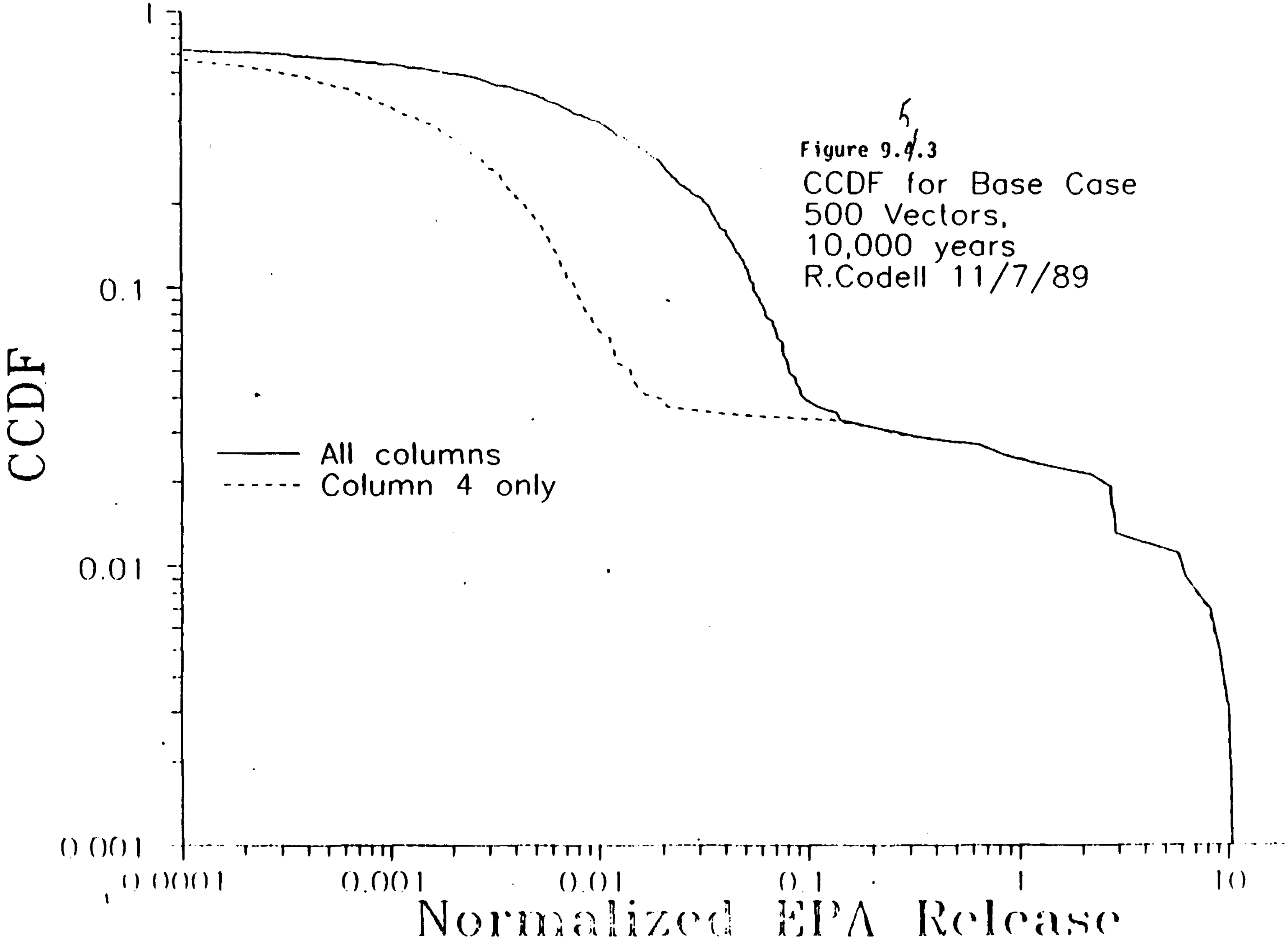


Figure 9.~~4~~5.3
CCDF for Base Case
500 Vectors,
10,000 years
R.Codell 11/7/89
All columns
Column 4 only
CCDF
1
0.1
0.01
0.001
0.0001
0.001
0.01
0.1
1
10
Normalized EPA Release

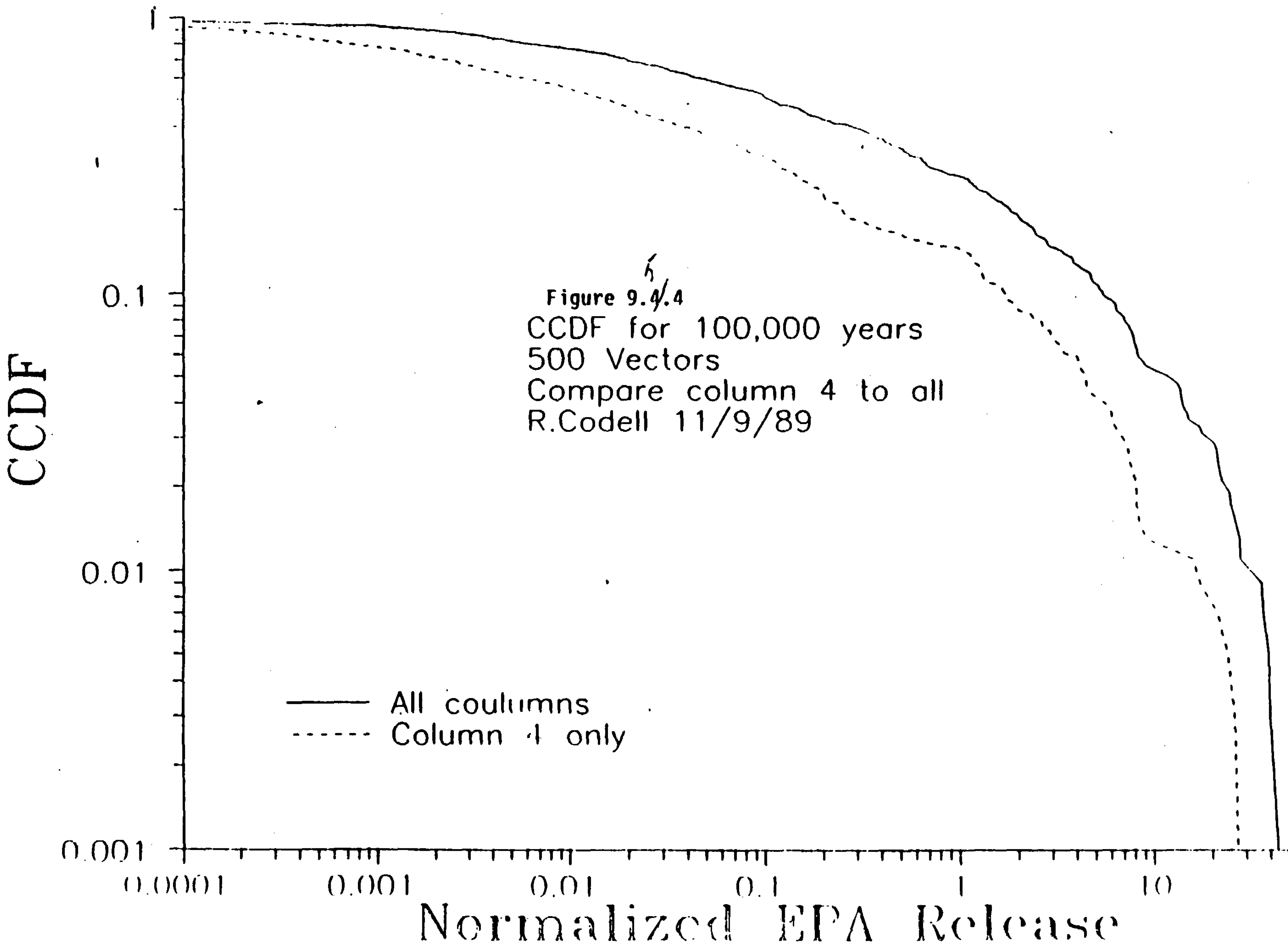


Figure 9.5.4
CCDF for 100,000 years
500 Vectors
Compare column 4 to all
R.Codell 11/9/89
All coulumns
Column 4 only
CCDF
1
0.1
0.01
0.001
0.0001
0.001
0.01
0.1
1
10
Normalized EPA Release

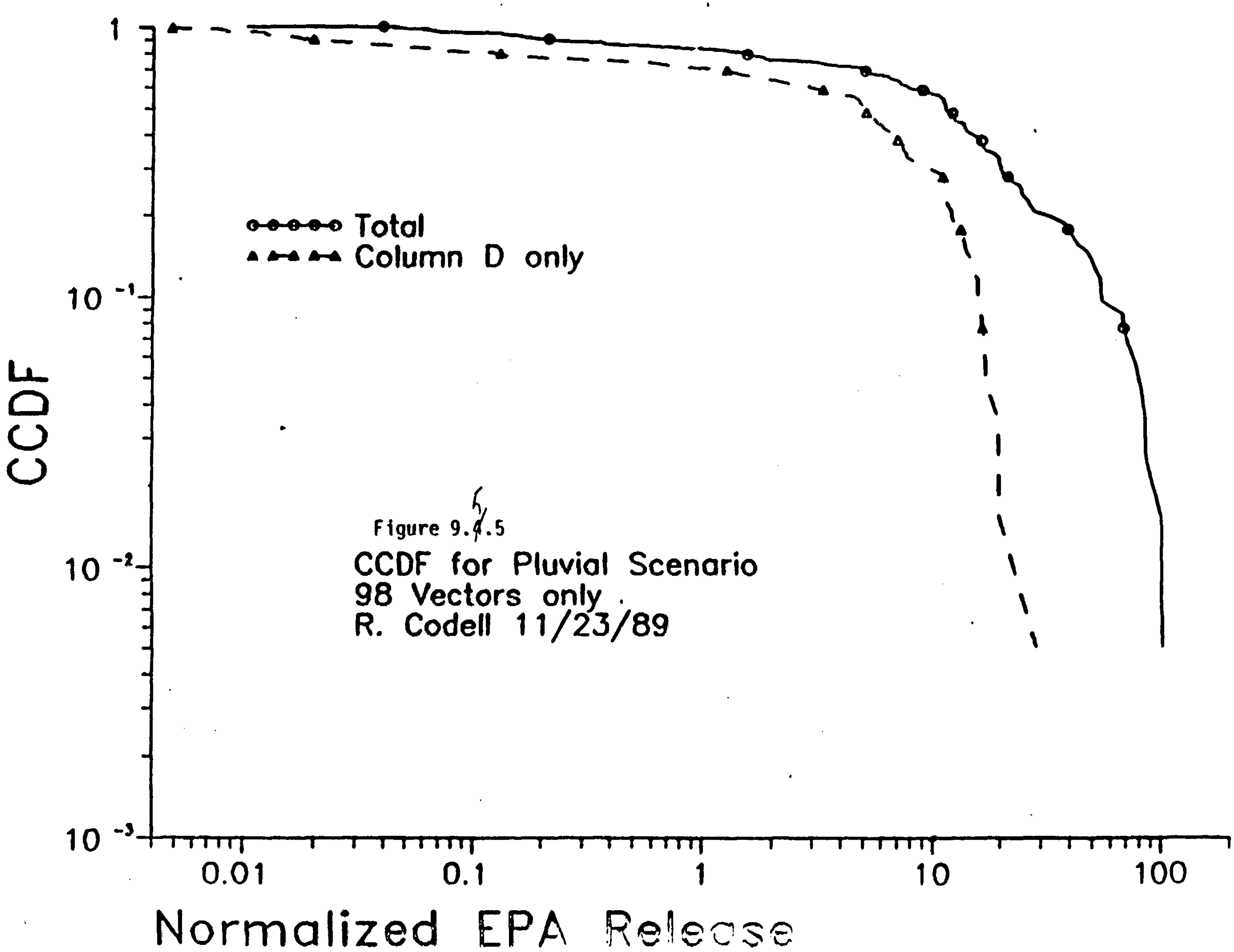


Figure 9.~~4~~.5
CCDF for Pluvial Scenario
98 Vectors only
R. Codell 11/23/89

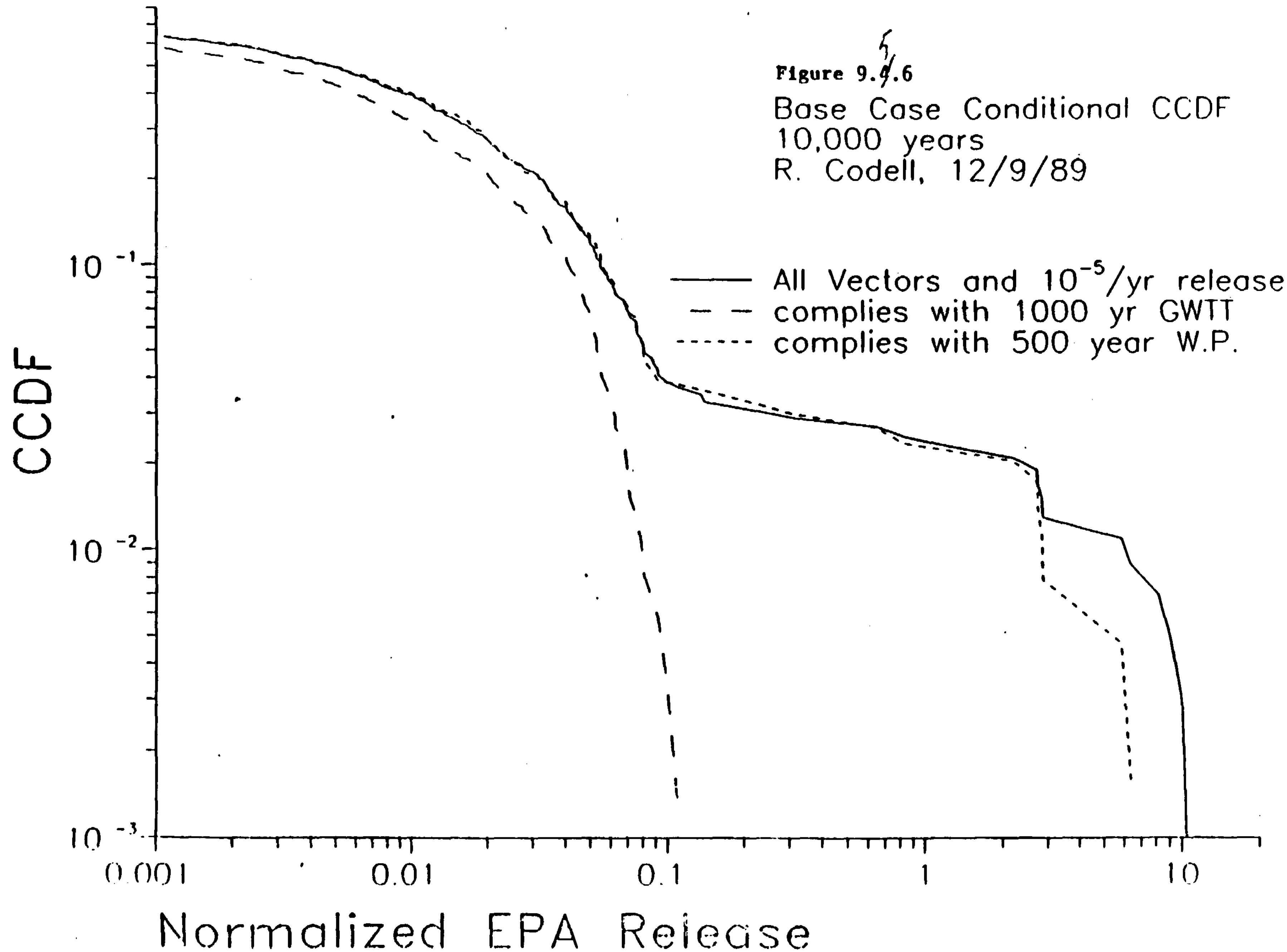


Figure 9.~~4~~5.6
Base Case Conditional CCDF
10,000 years
R. Codell, 12/9/89

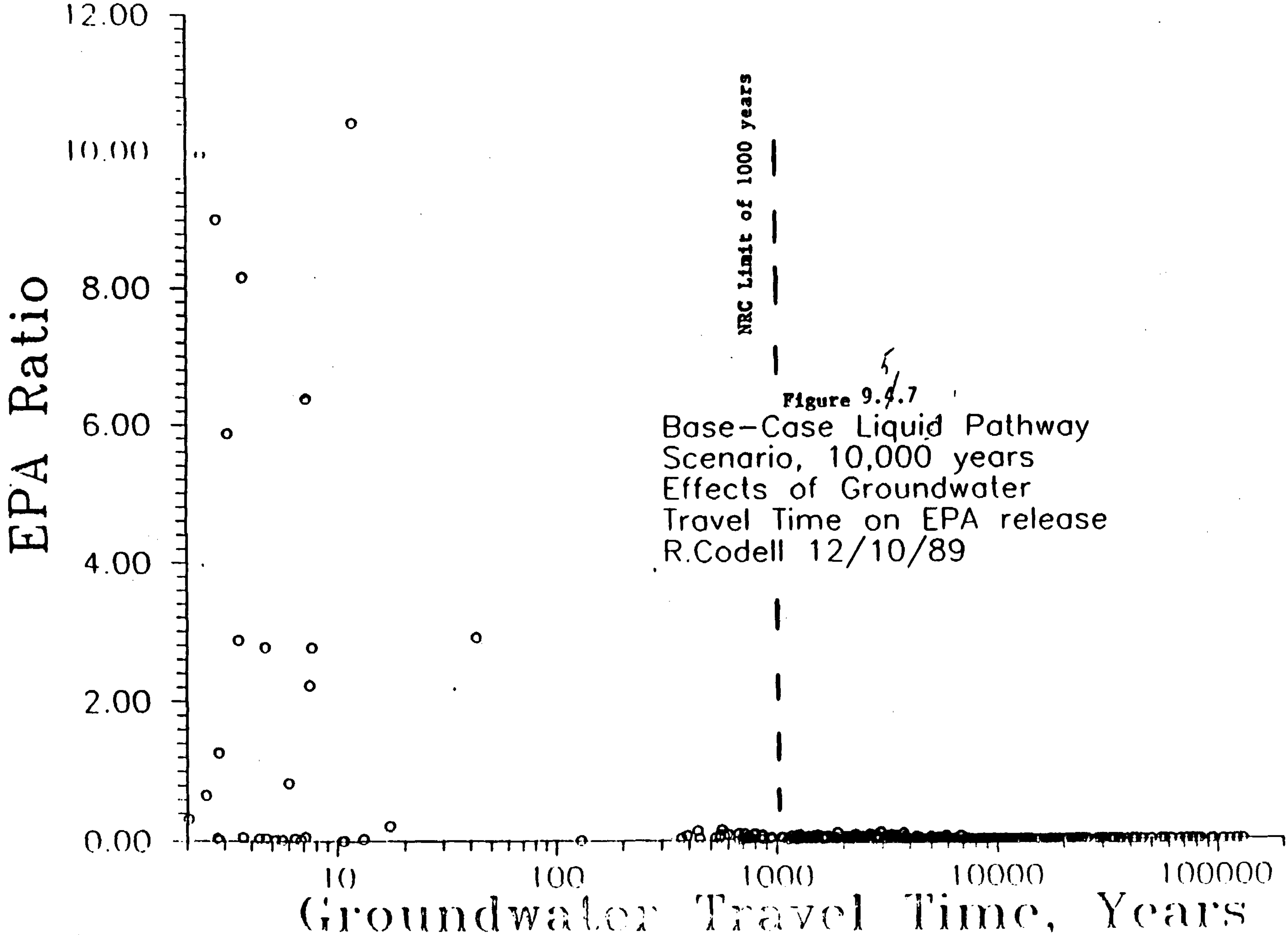


Figure 9.~~4~~5.7
Base-Case Liquid Pathway
Scenario, 10,000 years
Effects of Groundwater
Travel Time on EPA release
R.Codell 12/10/89

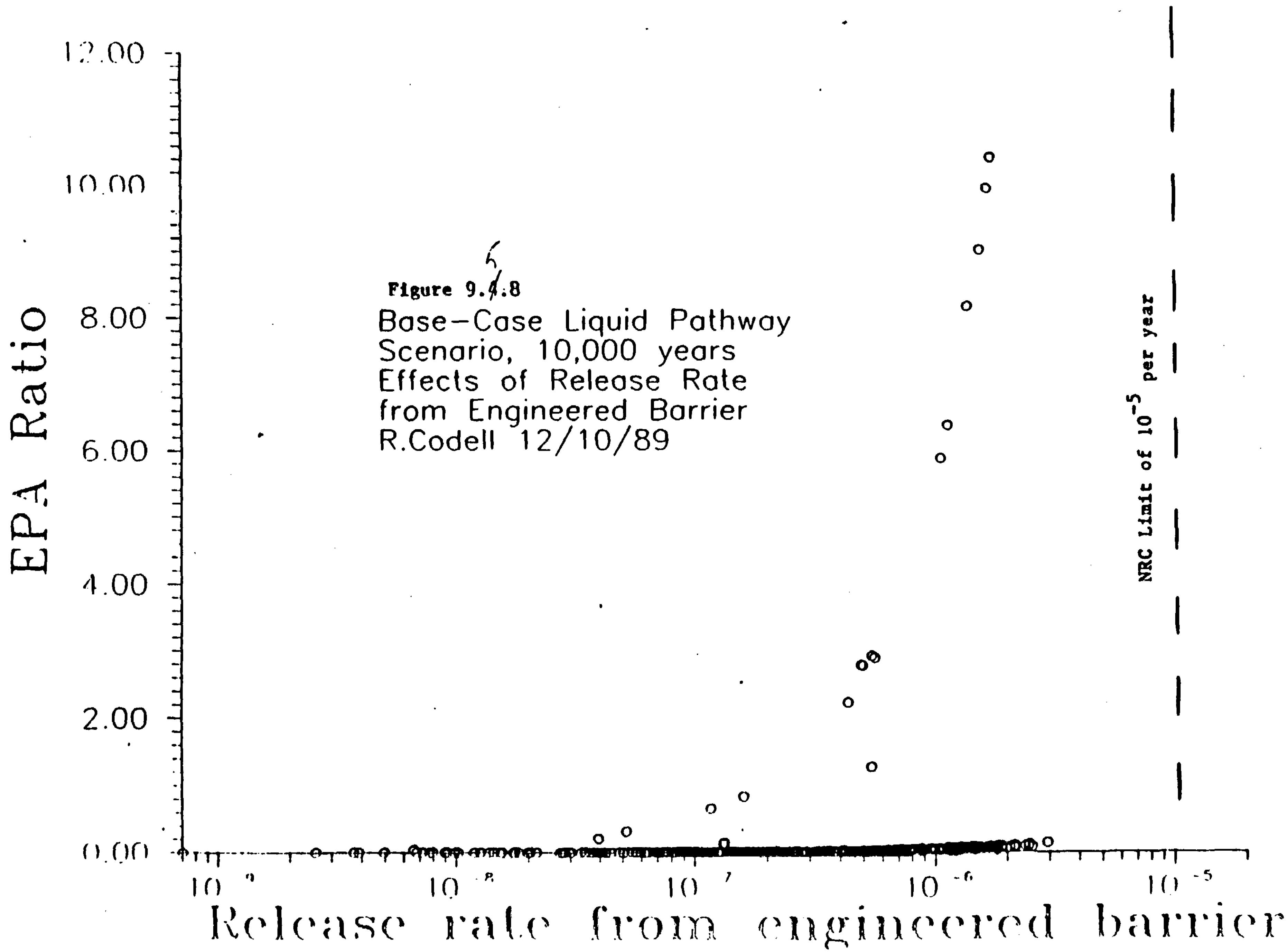


Figure 9.5.8
Base-Case Liquid Pathway
Scenario, 10,000 years
Effects of Release Rate
from Engineered Barrier
R.Codell 12/10/89
EPA Ratio
12.00
10.00
8.00
6.00
4.00
2.00
0.00
10^{-9}
10^{-8}
10^{-7}
10^{-6}
10^{-5}
Release rate from engineered barrier
NRC Limit of 10^{-5} per year

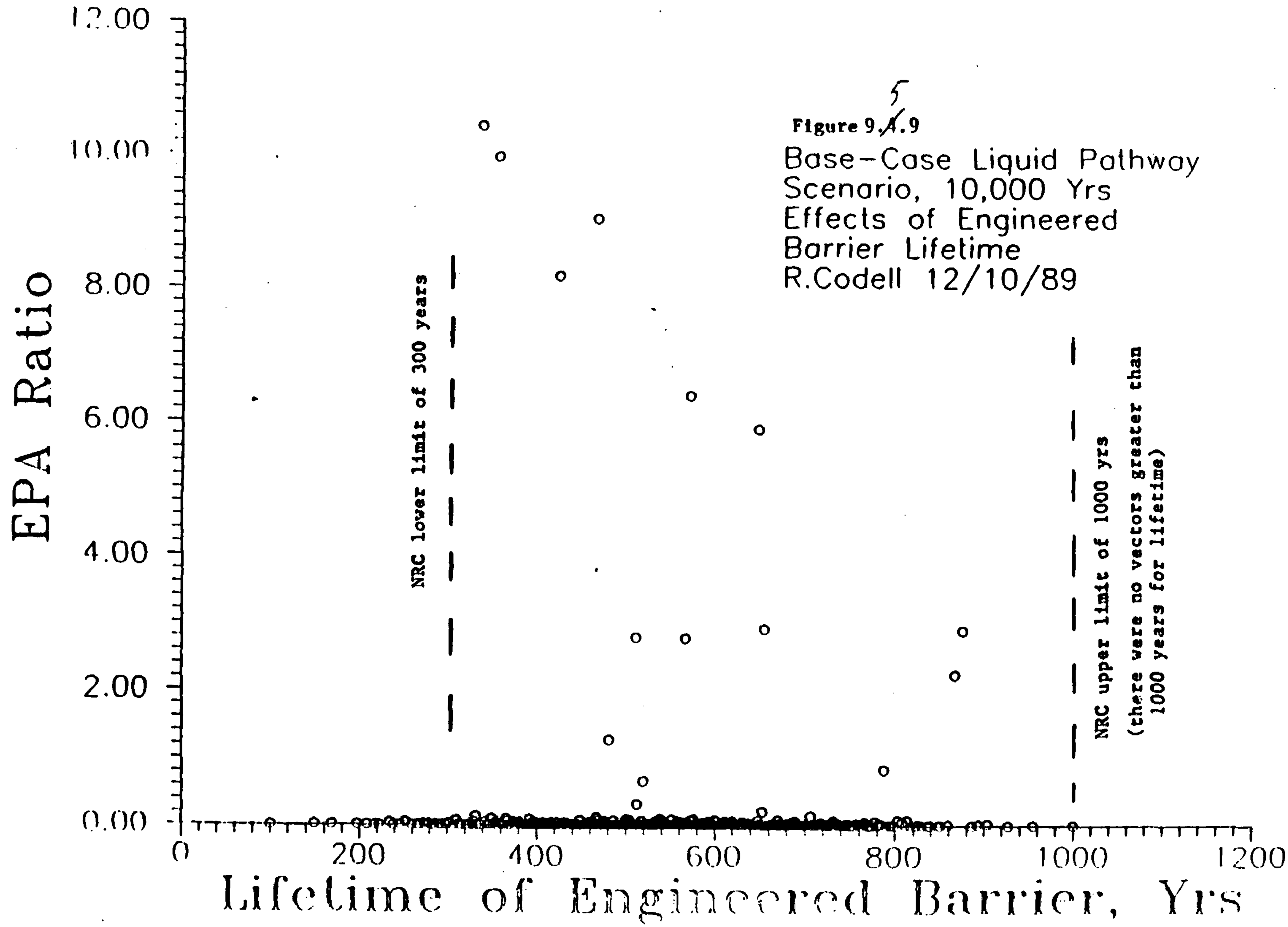


Figure 9.5.9
Base-Case Liquid Pathway
Scenario, 10,000 Yrs
Effects of Engineered
Barrier Lifetime
R.Codell 12/10/89
NRC lower limit of 300 years
NRC upper limit of 1000 yrs
(there were no vectors greater than
1000 years for lifetime)
EPA Ratio
12.00
10.00
8.00
6.00
4.00
2.00
0.00
0
200
400
600
800
1000
1200
Lifetime of Engineered Barrier, Yrs

Section 9.6

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MOU CHAPTER ON TOTAL CCDF

9.6 Total CCDF

9.6.1 Introduction

The results presented here can only be considered as a preliminary performance assessment and should not be taken as representative of the performance of a repository at Yucca Mountain, Nevada. Among the most important limitations of the study are:

1) the lack of sufficient site data,
2) the large uncertainties in the data now in use,
3) the use of only four scenarios to characterize future states at the site,
4) uncertainties in the site conceptual model, and
5) uncertainties in modeling the physicochemical processes leading to radionuclide release and migration in the geosphere.

For this MOU demonstration, the staff concentrates on four scenario classes:

1) an undisturbed or base case,
2) pluvial conditions,
3) drilling under undisturbed conditions, and
4) drilling under pluvial conditions.

As shown in Figure 9.6.1, these particular scenarios arise from the possible combinations of two fundamental events: a pluvial period (or not) and drilling at the site (or not). Probabilities for each of the scenario classes are determined by multiplying the probabilities of their independent constituent events. The likelihood of each event is based upon staff judgment in the case of the pluvial/nonpluvial events, and 40 CFR Part 191 Appendix B for the human intrusion events. 40 CFR 191 assumes a likelihood of drilling at the site as a set number of boreholes per unit area over 10,000 years based upon the geologic formations in which the repository is located.

There are two important points to note in Figure 9.6.1. First, the case in which conditions at the repository over the next 10,000 years remain as they are today appears highly unlikely. Secondly, the addition of a drilling event to both the undisturbed and pluvial cases has two effects. It increases the overall probability of the scenario class, and it also adds slightly to the consequences at the high probability/low consequence end of the graph. This will be more apparent on the graphed CCDFs for the undisturbed case than for the pluvial conditions for reasons discussed later.

The partial CCDFs for each of the scenario classes are shown in Figures 9.6.2 through 9.6.5. These differ from the distribution of consequence figures shown earlier (in the flow and transport section of the report) in that these partial CCDFs incorporate the probabilities of the scenarios themselves. For this reason, the cumulative probability of any single scenario presented here never reaches 1.00, as it will for the total CCDF, which is a composite of all four scenario classes.

9.6.2 Partial CCDF Results

9.6.2.1 The Undisturbed Case

The log-log plot of summed normalized EPA release versus cumulative probability for undisturbed conditions (Figure 9.6.2) shows the characteristic concave downwards shape for a CCDF. As will be the case in each of the graphs, the curve meets the y-axis at the likelihood of the scenario; here the likelihood is equal to 2.3×10^{-7}. Although the maximum EPA ratio is slightly greater than 10.0, the probability of this occurrence is small enough as to be ignored, since the plot lies well below the EPA standard presented in 40 CFR 191. The jog in the curve may be due to the transition from matrix flow to fracture flow as the saturated conductivity of the rock is exceeded.

9.6.2.2 Pluvial Conditions

Consequences from the pluvial case (Figure 9.6.3) range from EPA ratios of .01 to in excess of 100. Yet, as for the undisturbed conditions, the overall probability of the scenario is too low to warrant further consideration of the consequences. In fact, it is highly likely that both undisturbed and pluvial scenarios would have been screened out in a full scenario analysis.

Note: An inordinate amount of computer time required on the CRAY supercomputer limited the pluvial and drilling under pluvial conditions scenarios to only 98 vectors. Furthermore, because a sample of 200 input vectors was planned and generated with the LHS sampling routine to represent this pathway and scenario class, a subset of 98 vectors might lead to spurious correlations and an inadequate representation of the parameter space.

9.6.2.3 Drilling Under Undisturbed Conditions

The effects of the drilling event discussed above are readily apparent in the partial CCDFs for the two human intrusion scenarios. The CCDF for drilling under undisturbed conditions (Figure 9.6.4) shows the slight step attributable to consequences from the drilling in the low consequence/high probability end of the curve. The higher consequence/lower probability portion of the CCDF is dominated by releases via the liquid pathway.

More importantly though, drilling increases the overall probability of the scenario to 0.9. Thus, EPA ratios for this scenario greater than 10.0 lie outside the EPA standard, which appears as a step function in the figure.

9.6.2.4 Drilling Under Pluvial Conditions

The shape of the partial CCDF for drilling under pluvial conditions (Figure 9.6.5) does not exhibit the effects of the drilling. This is because these consequences are in the range of .0001, and are therefore negligible when factored into overall consequences of .01 to 100. However, the difference between this curve and the pluvial partial CCDF lies again in the alteration of the probability of the scenario class itself. With the overall likelihood of drilling under pluvial conditions equal to 0.1, the high consequence/low probability end of the curve violates the standard set by the EPA.

9.6.3 Results for the Total CCDF

Figure 9.6.6 demonstrates how each of the four individual scenarios contributes to the total CCDF. It is clear from this figure that the undisturbed and pluvial scenarios can be ignored in the calculation of the overall CCDF, because their respective probabilities are negligible despite the high consequences involved. The same cannot be said however of the two scenario classes involving drilling. It is readily apparent that these two scenario classes dominate the total CCDF under the given probabilities and conditions.

The total CCDF for the four scenario classes modeled is compared against the EPA standard in Figure 9.6.7. This comparison shows that the standard is exceeded in two locations, with the more pronounced violation at the high consequence/low probability end, particularly between EPA ratios of 10 and 100.

This result should not be taken as representative of the performance of a repository at Yucca Mountain, Nevada. Rather, it should be used as an indication of the importance of the assumptions and modeling of fracture/matrix interactions and the physical parameters, such as infiltration rate, important in such modeling.

DETERMINATION OF SCENARIO PROBABILITIES FROM THE PROBABILITIES OF FUNDAMENTAL EVENTS

	$\bar{P}$ 0.9	P 0.1
$\bar{D}$ 2.3×10^{-7}	scenario class # 0 probability $= 2.0 \times 10^{-7}$	scenario class # 1 probability $= 2.3 \times 10^{-8}$
D ~ 1.0	scenario class # 2 probability ~ 0.9	scenario class # 3 probability ~ 0.1

$\bar{P}$ is not pluvial

P is pluvial

$\bar{D}$ is no drilling

D is drilling

scenario class # 0 is no drilling, not pluvial

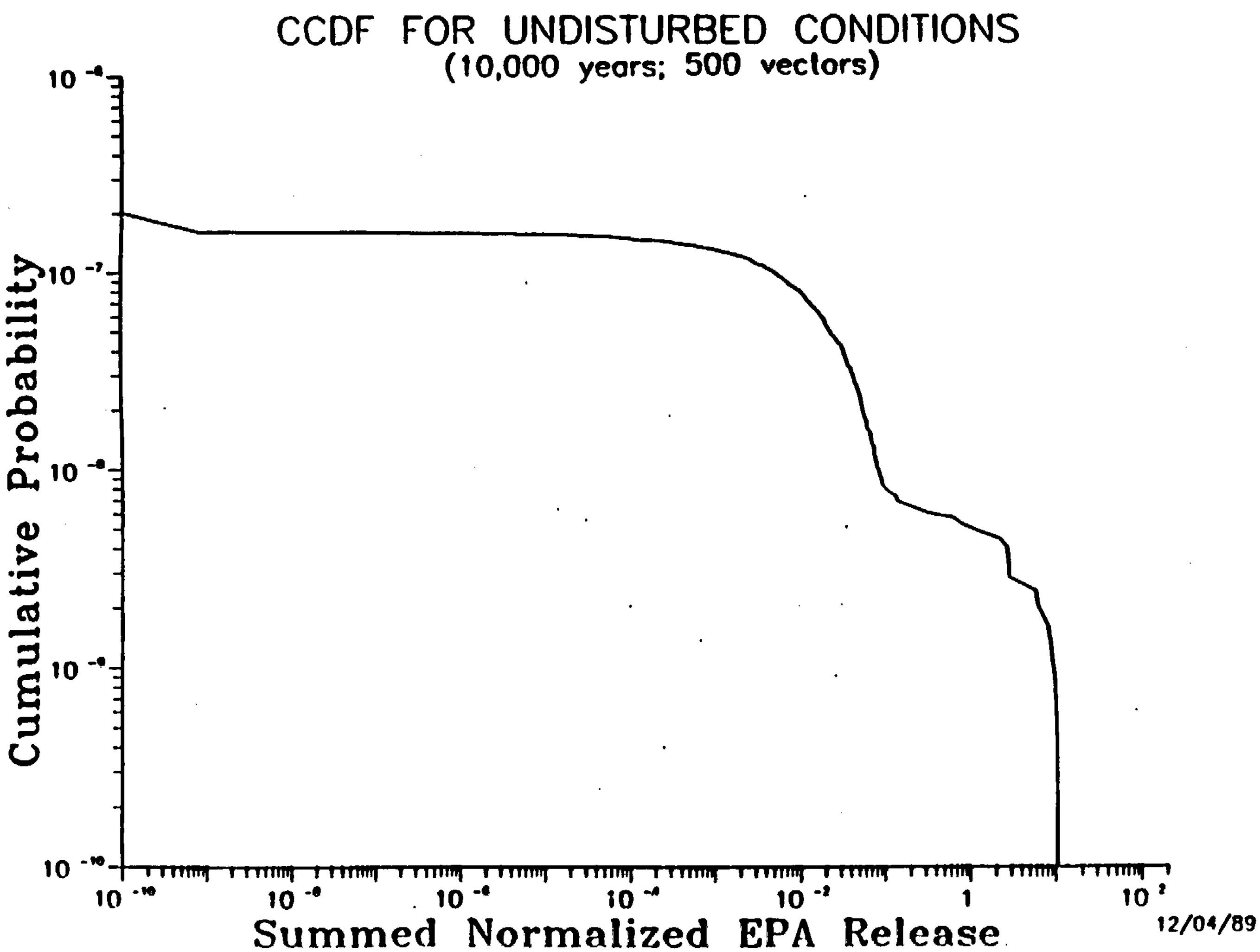
scenario class # 1 is no drilling, with pluvial

scenario class # 2 is drilling, not pluvial

scenario class # 3 is drilling and pluvial

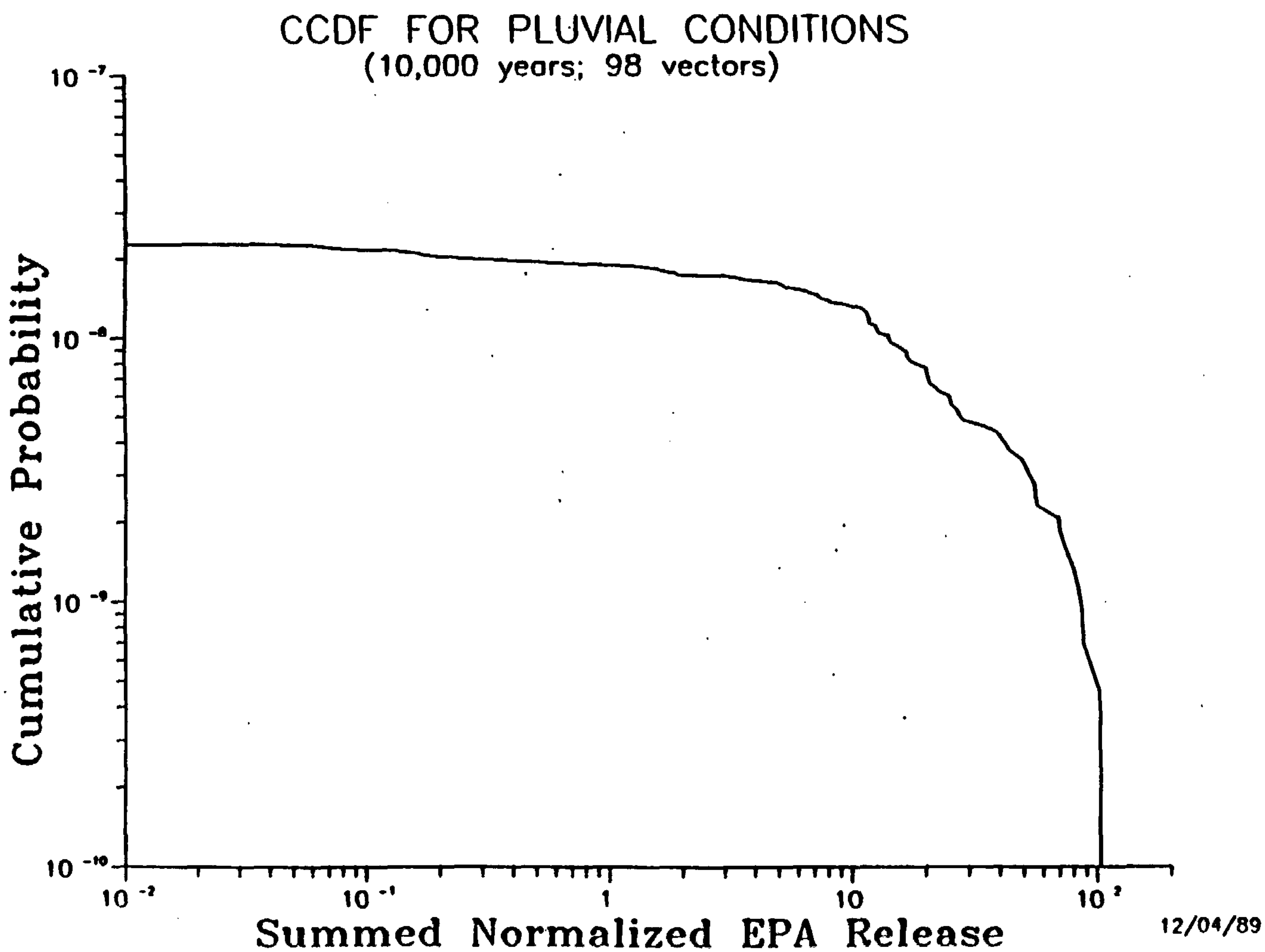
Note: Probability combinations assume that fundamental events have independent probabilities of occurence; this is not a general restriction.

Figure 9.6.1



CCDF FOR UNDISTURBED CONDITIONS
(10,000 years; 500 vectors)
Cumulative Probability
Summed Normalized EPA Release
12/04/89

Figure 9.6.2



CCDF FOR PLUVIAL CONDITIONS
(10,000 years; 98 vectors)
Cumulative Probability
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{-2}
10^{-1}
1
10
10^{2}
Summed Normalized EPA Release
12/04/89

Figure 9.6.3

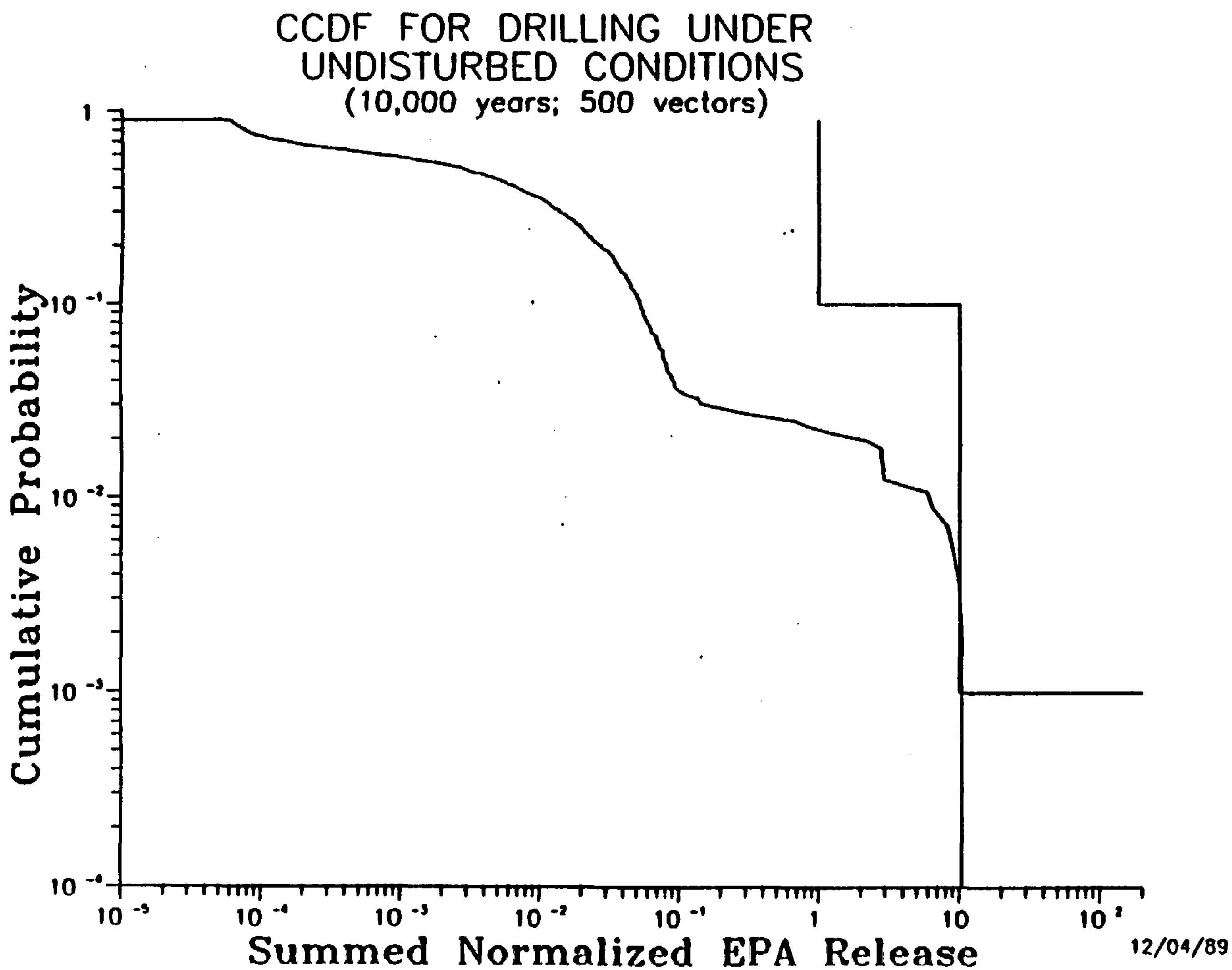

CCDF FOR DRILLING UNDER
UNDISTURBED CONDITIONS
(10,000 years; 500 vectors)
Cumulative Probability
1
10⁻¹
10⁻²
10⁻³
10⁻⁴
10⁻⁵
10⁻⁴
10⁻³
10⁻²
10⁻¹
1
10
10²
Summed Normalized EPA Release
12/04/89

Figure 9.6.4

CCDF FOR DRILLING UNDER
PLUVIAL CONDITIONS
(10,000 years; 98 vectors)

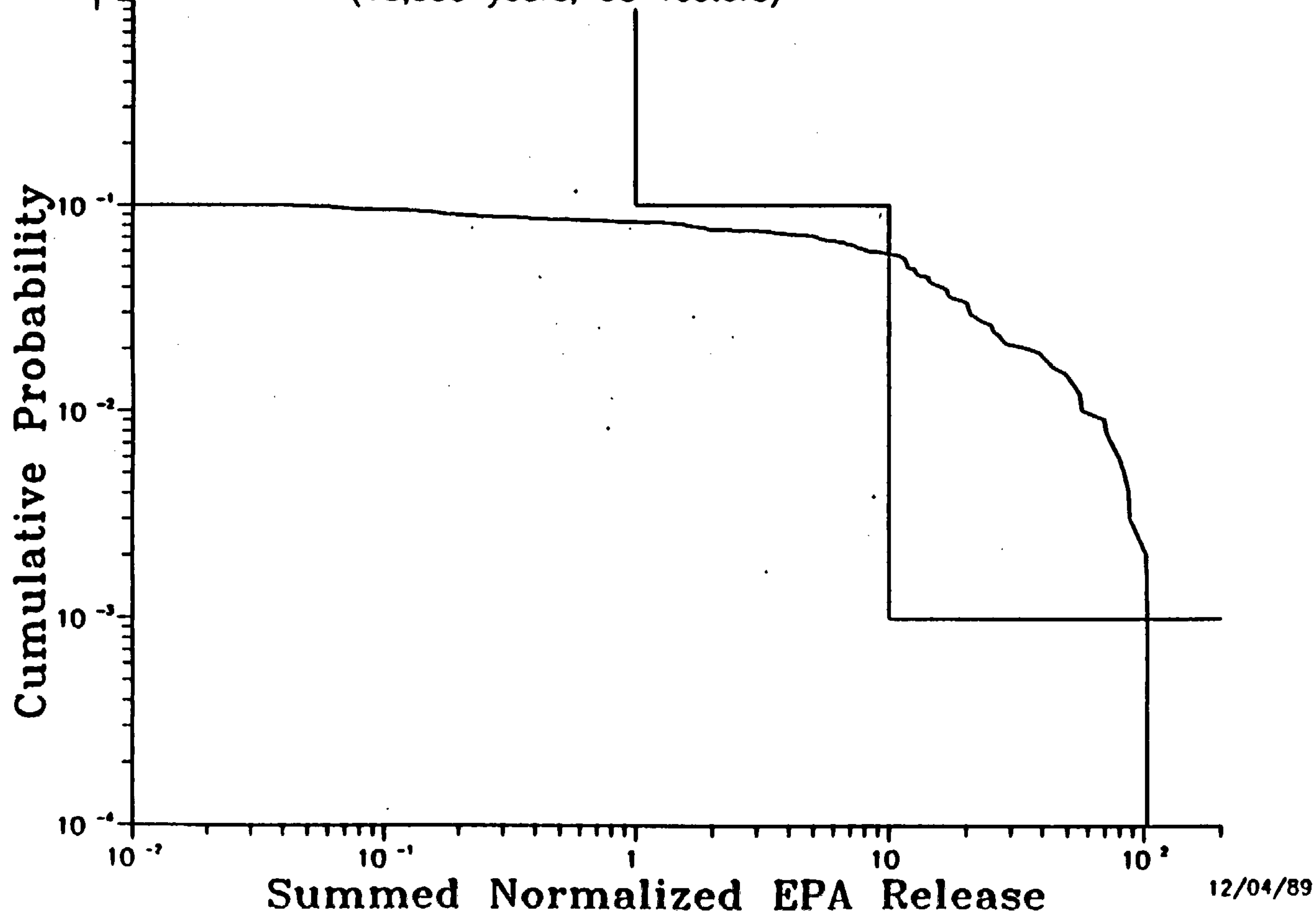
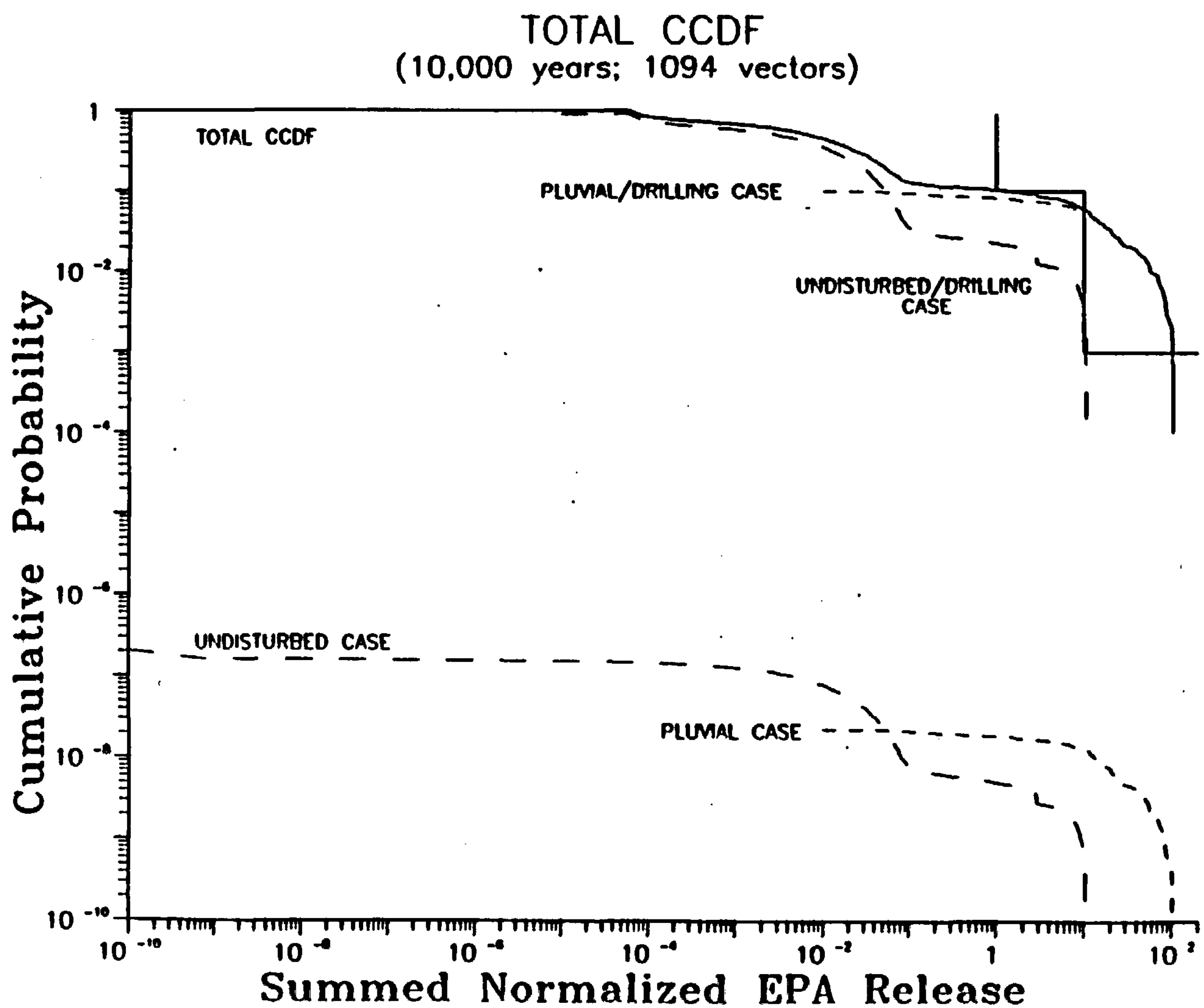


Figure 9.6.5

Figure 9.6.6


TOTAL CCDF
(10,000 years; 1094 vectors)
TOTAL CCDF
PLUVIAL/DRILLING CASE
UNDISTURBED/DRILLING CASE
UNDISTURBED CASE
PLUVIAL CASE
Cumulative Probability
Summed Normalized EPA Release
1
10^{-2}
10^{-4}
10^{-6}
10^{-8}
10^{-10}
10^{-10}
10^{-8}
10^{-6}
10^{-4}
10^{-2}
1
10^{2}

TOTAL CCDF
(10,000 years; 1094 vectors)

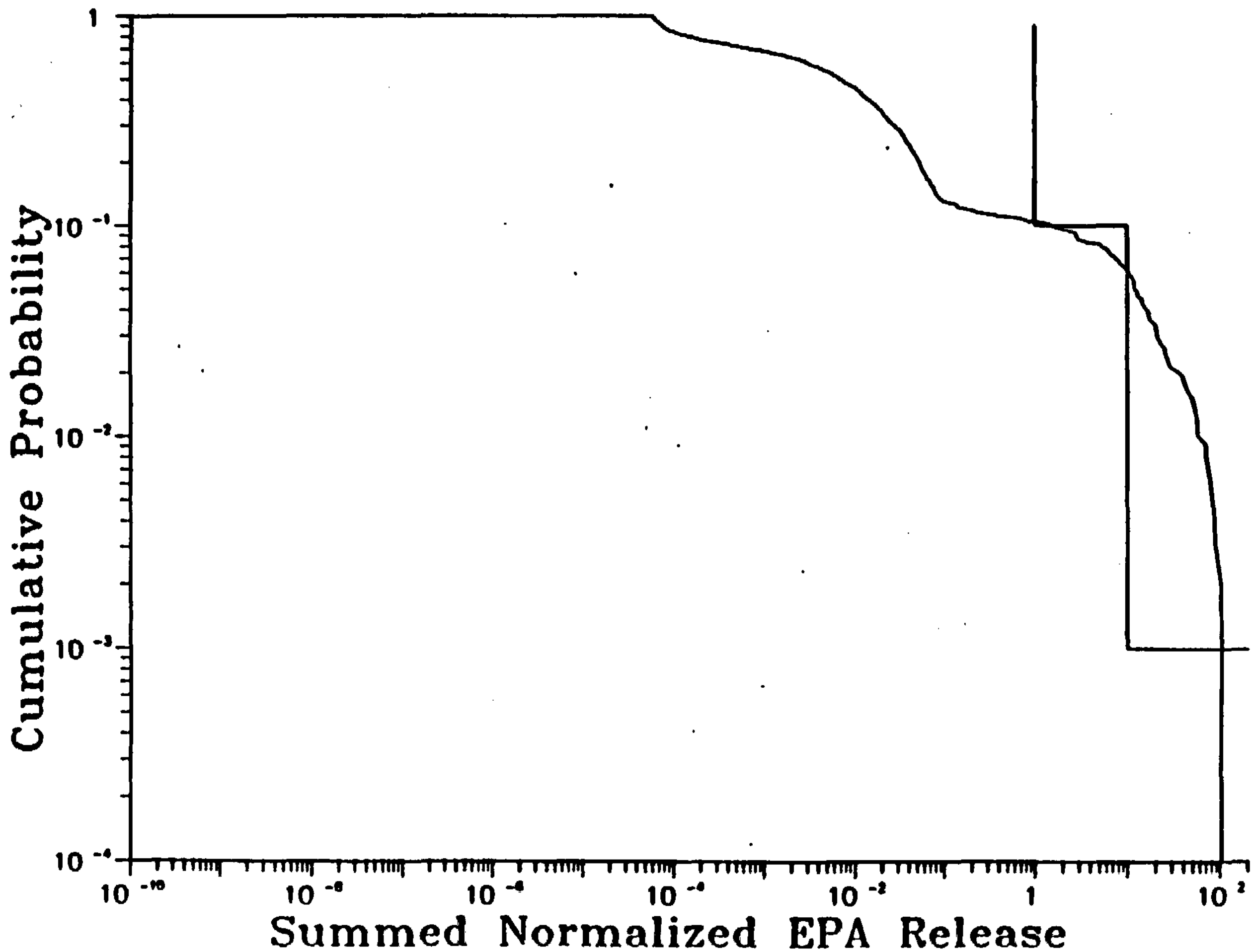


Figure 9.6.7

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Author's Name:
tjm

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section 10

SECTION 10

CONCLUSIONS AND RECOMMENDATIONS

Work Performed.

In order to perform this preliminary performance assessment and demonstrate the staff capability to conduct such work, the following types of activities were performed:

1. Computations & Support
   - data input
   - model setup
   - code development & testing
   - code execution
   - output analysis

2. Auxiliary analyses
   - evaluation of assumptions
   - preprocessing raw data

3. Documentation
   - Draft report

By conducting the activities listed above, the NRC staff achieved the following major accomplishments:

1. The NRC staff demonstrated its capability to conduct independently performance assessments for a HLW repository; in doing so the staff gained insight into the performance of the Yucca Mountain repository and increased its insight into the capabilities and limitations of the currently available performance assessment methodology.

2. Developed a CCDF to describe performance of a Yucca Mountain HLW repository for a limited set of scenario classes, using preliminary data.

3. Modeled the liquid pathway of the undisturbed scenario class for the Yucca Mountain repository using:

   (1) the NEFTRAN computer code to simulate transport in the unsaturated zone

   (2) four vertical transport legs under the repository to account for spatial variability

   (3) an improved treatment of waste form dissolution

   (4) a nonmechanistic model of waste package failure

   This liquid pathway modeling was extended to treat pluvial conditions

4. Developed and used a total system code.

5. Developed a model and the corresponding computer code for human-intrusion by drilling.

6. Performed a preliminary statistical analysis of results (sensitivity and uncertainty) using several techniques including Latin Hypercube Sampling (LHS) and regression analysis methods.

7. Executed several auxiliary analyses:

   - potential for non-vertical flow
   - sampling requirements for CCDF generation
   - consequences of C-14 gaseous releases
   - statistical analysis of available hydrologic data for input to flow and transport models

Tenatative Results

In presenting some tentative results, the authors want to state some important caveats to be kept in mind when contemplating these results. Taking these tentative results of a preliminary analysis out of context or separating these tentative results from these caveats, may lead to the inappropriate interpretation and use of the results, for which the authors of this report cannot be responsible.

1. The results presented here have had limited peer review, has numerous simplifying assumptions, and are based on limited data; therefore, THE NUMERICAL RESULTS SHOULD NOT BE TAKEN AS REPRESENTATIVE OF THE PERFORMANCE OF A REPOSITORY AT YUCCA MOUNTAIN, NV.

2. The analysis is replete with UNCERTAINTIES regarding:

   o conceptual models
   o data
   o physiochemical models
   o models and data for predicting scenarios

3. Only a limited set of scenario classes were incorporated in the modeling, so the total CCDF presented in this report cannot truly represent total system performance.

4. The modeling of waste package failure is nonmechanistic and rudimentary; therefore, this aspect of repository performance is probably not adequately represented.

5. The flow and transport models used attempt to simulate key aspects of the performance of a repository at Yucca Mountain, but do so indirectly through modifications of transport analysis for saturated rock. A more direct representation of flow and transport in partially saturated, fractured rock is needed to assure an adequate level of confidence in the results.

Given the caveats stated above, the authors wish to remind the reader that the tentative conclusions stated below should be used only with these substantial limitations kept in mind. Based on a preliminary analysis, the staff has reached th following tentative major conclusions:

1. The areal extent of the Yucca Mountain repository appears to be an important aspect determining performance and should be included in models of performance; important aspects appear to be areal variability of:

- o waste package failure
- o depth of rock to water table
- o potential of rock units to sustain fracture flow

2. The gaseous release of C-14 could be an important issue in repository performance, but more analysis and data are needed (DOE is primarily responsible for gathering the needed data).

3. The potential for nonvertical flow at Yucca Mountain appears to be great and could have a substantive effect on the performance of a repository there. There could be perching of water along interbeds and diversion of water to shorter paths to the water table. More analyses and additional data collection by DOE are required to evaluate this significant alternative to DOE's preferred conceptual model of predominantly vertical flow.

4. For the "liquid pathway" scenario class, the most significant contributors to the consequences represented by the CCDF are isotopes of plutonium. Because plutonium behavior is poorly understood, large uncertainties exist regarding:

   - o colloids
   - o retrograde solubility
   - o sensitivity of chemistry to oxidation state

5. For the "liquid pathway" scenario class, the important input parameters appear to be:

   - o infiltration flux
   - o fraction of infiltrating groundwater contacting the waste
   - o uranium matrix solubility
   - o saturated hydraulic conductivity for the Calico Hills Vitric unit

6. Consequence codes currently used may not be sufficiently efficient to allow analyzing many scenarios each with many input parameter vectors, so that total system performance is adequately characterized.

Preliminary Recommendations

Based on this preliminary analysis and the limitations noted, the staff has some preliminary recommendations regarding the directions for further technical work to take. These recommendations for technical improvements include improvements to (1) modeling used to estimate performance, (2) analyses used to support the estimates of performance, (3) scientific input and research needed to provide a better basis for the estimates of performance.

Recommended improvements to modeling of performance:

General

1. Add the capability for modeling additional scenario classes.

3. Acquire, test, and evaluate codes developed by SNL for a repository in the unsaturated zone.

4. Explore, with the CNWRA, the adaptation of the FPPA (Fast Probabilistic Performance Assessment) methodology to generate the total system CCDF.

5. Evaluate additional codes, which could not be acquired and evaluated during this short-time effort, to determine whether existing codes can meet the NRC modeling needs or whether additional code development is needed.

Flow and Transport

1. Refine groundwater modeling (e.g., by considering higher dimensions).

2. Incorporate a model of gas-pathway transport in the calculation of the CCDF.

3. Include flow and transport through the saturated zone.

4. Directly model transport through a partially saturated, fractured rock, instead of the indirect, approximate representation used in Phase 1.

5. Explicitly model fracture/matrix coupling.

Source Term

1. Attempt to develop or use a previously developed mechanistic model of waste package failure

2. Develop a mechanistic model of contact between groundwater and the waste

3. Treat the repository as a source of radionuclides distributed in time and space, instead of as a point source

Recommended improvements to and extensions of auxiliary analyses:

1. Perform detailed geochemical analyses to investigate:

   - use of K_D's (distribution coefficients)
   - effects of spatially varying saturation on radionuclide migration
   - waste form, groundwater, tuff reactions
   - waste package degradation
   - oxidation of the spent fuel matrix
   - plutonium behavior

2. Evaluate heat effects at early time periods; estimate the thermal, hydrologic, and geochemical environment of the repository at early times.

3. Evaluate safety and reliability value added by NRC subsystem requirements beyond total system requirements of EPA (some work of this type is discussed in Section 9.4).

4. Estimate health effects from releases to evaluate adequacy of 40 CFR 191.

5. Evaluate importance of thermally and barometrically driven air flow on repository performance at Yucca Mountain.

6. Perform detailed hydrologic analysis for Yucca Mountain, to provide a better input t the transport analysis and to examine, in more detail, various alternative hypotheses regarding hydrology at Yucca Mountain.

Recommendations for additional scientific input (some of these items could be performed by either the DOE or NRC, while others are clearly the responsibility of DOE):

1. Develop and demonstrate a mathematically rigorous, scientifically robust method for scenario analysis.

2. Obtain geoscience input for modeling vulcanism.

3. Obtain geoscience and hydrologic input to modeling faulting, uplift, and subsidence at Yucca Mountain.

4. Obtain laboratory chemical analysis to determine the partitioning of radionuclides in various compartments of the spent fuel waste form.

5. Obtain field and laboratory data on phenomena important to the near-field behavior of the repository, especially the effects of heat.

6. Obtain more data on plutonium geochemistry.

7. Obtain a better understanding of waste package corrosion in the unsaturated zone.

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App A