#19

WBS: 1.2.1.5
SCPB: 8.3.1.15
QA: L

Civilian Radioactive Waste Management System
Management and Operating Contractor

Waste Isolation Study

B00000000-01717-5705-00062 REV 00/DCN 1

September 9, 1997

Prepared for:

U. S. Department of Energy
Yucca Mountain Site Characterization Office
P.O. Box 30307
North Las Vegas, Nevada 89036-0307

MANAGED COPY
NO. 103924

Prepared by:

TRW Environmental Safety Systems Inc.
1180 Town Center Drive
Las Vegas, Nevada 89134

Under Contract Number
DE-AC01-91RW00134

DF03
102.8
WM-11

WBS: 1.2.1.5
SCPB: 8.3.1.15
QA: L

Civilian Radioactive Waste Management System
Management and Operating Contractor

Waste Isolation Study

B00000000-01717-5705-00062 REV 00/DCN 1

September 9, 1997

Steven F. Saterlie, Study Manager and Lead Document Preparer — Date: 10/29/97

Steven F. Saterlie, Acting Systems Analysis and Modeling Department Manager — Date: 10/29/97

Carl R. Hastings, Acting Systems Engineering Office Manager — Date: 10/29/97

ACKNOWLEDGMENT

This report was prepared from work done by the Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O) under Contract Number DE-AC01-91RW00134 for the U.S. Department of Energy (DOE) office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Project. The major contributors and authors of the study were S. F. Saterlie (TRW), David Sevougian (IN), Vinod Vallikat (IN), Joel Atkins (IN), Bryan Dunlap (IN), Stephen Thornton (LANL).

Thanks are extended to J. C. de la Garza of the Yucca Mountain Site Characterization Office (YMSCO) for his suggestions and oversight of the effort. Special thanks are also extended to J. A. Connell for technical editing and to Technical Publications Management for document preparation.

INTENTIONALLY LEFT BLANK

EXECUTIVE SUMMARY

The waste isolation strategy for emplacement of spent nuclear fuel and high-level radioactive waste in the potential repository in the unsaturated zone at Yucca Mountain, Nevada relies on a defense-in-depth approach using multiple barriers to limit releases of the radioactive wastes to the accessible environment. A successful License Application in 2002 will depend on our ability to establish the effectiveness of the site and the ability of the natural and engineered systems to limit releases. It will require an improved understanding of the performance of the various barriers under the conditions that will occur in the potential repository over long periods of time. In an effort to focus the work necessary to achieve Viability Assessment and License Application, and to assist in gaining an understanding of the performance of the barriers to waste isolation to achieve these objectives, a Waste Isolation Study was initiated to document the current understanding of the performance of the various barriers and to identify which engineered barriers would provide the best performance. The purpose of the study was to estimate the performance of the various barriers, natural and engineered, and, based on that, recommend which barriers should be considered in License Application and what is needed to substantiate the performance of a particular barrier as licensable.

This document reports the work conducted in the Waste Isolation Study from October 1, 1996 to May 15, 1997. The objective of the study was to document our current basis of understanding of the performance of the various barriers considered important to waste isolation, identify the cost of the engineered barriers, identify the relative merit of the various barriers (engineered and natural), and recommend an approach to evaluate the engineering measures that have potential for significant reduction in peak dose at reasonable cost.

One intent of the study was to address the question of what thresholds should be set to significantly reduce the peak dose, at reasonable cost, whenever that peak dose may occur, including time periods up to and beyond 10,000 years. Based on this objective, arguments were advanced in this study that a reduction in peak dose of a factor of 10 is large enough to be considered significant given the uncertainties in the models and measurements. Although subjective, the study recommends that a reasonable cost to achieve this reduction would be $1 Billion dollars or about eight percent of the MGDS costs. This threshold and the cost information were used to provide recommendations as to what engineering measures should be pursued.

The study relied on current work, which is documented in the study, but also drew from previous studies that have provided estimates of performance of the various barriers. The previous studies used were the *Total System Performance Assessment* (CRWMS M&O 1995a), the *Engineered Barrier System Performance Requirements Systems Study Report* (CRWMS M&O 1996a), the *Thermal Loading Study for FY 1996* (CRWMS M&O 1996d), and *Description of Performance Allocation* (CRWMS M&O 1996b). Since those studies, the Project has established that there is a potential for higher moisture fluxes in the unsaturated zone than originally considered. As a consequence, this study conducted some total system performance assessment calculations at higher moisture fluxes to examine the impact of these on performance of selected barriers. It should be noted that significant uncertainties currently exist as to the moisture flux and the transport in the unsaturated zone. In the future, planned or underway testing may reduce these uncertainties. For the most part, this analysis is scoping or preliminary in nature and was not intended to qualify a specific barrier's performance.

Based on the results of the study certain conclusions can be drawn about the performance of the various barriers. Based on our current understanding of the processes and conditions that are believed to exist in Yucca Mountain and on the process models that are currently used, it can be concluded that, as anticipated when Yucca Mountain was chosen as a potential site for disposal of radioactive wastes, the natural barriers provide the preponderance of waste isolation except at early times when the waste is still contained in the waste package. Specifically, the natural barriers provide over a factor of 1000 reduction in releases to the accessible environment.

Evaluations of the performance of engineered barriers were conducted to determine which barriers would provide significant performance (a reduction in dose by a factor of 10 or more) at reasonable cost (a cost of less than $1 billion). As a result of the evaluations it was determined that such barriers as cladding, galvanic protection, and a long lived (regulatory period or longer) drip shield may provide that amount of reduction in dose. However, a number of uncertainties still exist in being able to accurately predict the performance of these barriers. A number of recommendations were developed in the study and as a result of the uncertainties the recommendations include suggestions for activities that could be done to resolve some of those uncertainties. The following recommendations were developed based on the findings of the study.

Recommendations

Based on the synopsis of calculations in this study and the additional performance calculations at the higher moisture flux of 6.2 mm/yr, the following recommendations are offered:

Potential Engineered Barriers

- The performance predictions indicate that zircaloy cladding of the spent nuclear fuel assemblies may provide a significant reduction (about a factor of 10) in peak dose. Based on this study, it is recommended that the Project pursue a course of action that, if successful, will allow taking performance credit for cladding. Licensing issues such as initial integrity of the cladding and subsequent degradation modes needs to be addressed. Performance Assessment should evaluate available measurements of cladding performance done by Pacific Northwest National Laboratories, review the zircaloy corrosion model being developed by the Navy and upgrade/update the Yucca Mountain Project cladding process model. Ongoing materials tests (TR241GBC) on the effects of drips and relative humidity on spent nuclear fuel segments need to be completed and evaluated by Performance Assessment for inclusion in the process model. The updated cladding model should be used in Total System Performance Assessment-Viability Assessment (1998).

- Galvanic protection is another engineered barrier that may produce significant reduction (more than a factor of 10) in dose. However, there are currently significant uncertainties in the number of waste packages that would be protected by galvanic protection and the percent of the corrosion allowance barrier which would have to corrode before the inner corrosion resistant barrier starts to degrade. There are some laboratory tests ongoing (TR251GB7) and some longer term testing planned (long range plan; TR251GBB and TR251FBB) which would examine galvanic protection and potential crevice corrosion. These laboratory tests should be completed and the information used to update the process models for Total System Performance Assessment-Viability Assessment. The longer term tests should be conducted and Performance Assessment should incorporate this information in their models as it becomes available. Other alternatives that decrease the WP degradation rate should be examined as well including developing improved models for degradation of the corrosion resistant barrier and/or to choose a different corrosion resistant material which substantially increases containment lifetime.

- Under the conditions of high flux the performance predictions indicate that drip shields that survive for a long time (the regulatory period or longer) have potential for producing significant reduction in dose. A drip shield will reduce doses during it's lifetime but when it is gone doses return to levels approaching the base case with no drip shield. Long term reduction in dose from the base case at times after the drip shield is gone will required drip shield lifetimes well in excess of 20,000 years. It is unlikely that any man-made materials can be shown to have these very long lifetimes. Some work (TR251GB6) on materials evaluation of titanium and ceramics has been initiated. The testing work, including testing of other candidate material, should be completed and a determination made by Performance Assessment, in coordination with Regulatory and Licensing, as to what is needed for licensing. Based on experiment a range of drip shield life times needs to be used in future calculations.

- An alternative to a drip shield that may offer some merit is a third barrier to the waste package (e.g., a ceramic coating or other). Such an alternative could be evaluated by Waste Package Development with an assessment of drip shields. Evaluations such as constructability and operability to include the increase in waste package weight and thermal effects should be considered in addition to performance.

- Do not consider backfill in the current design concept for the purpose of reducing relative humidity at the WP but do not preclude the use of backfill. The reason not to preclude backfill at this point is that it may be needed to ensure survivability of a drip shield or a ceramic coating on the WP. Performance Assessment should develop a process model to evaluate the evaporative properties of backfill and, if these are found to improve performance, backfill can be reconsidered.

- It is recommended that the testing of cementitious materials planned (Long Range Plan, TR3C5GBB) be completed and evaluated by Performance Assessment and the performance of tunnel liners during heating be examined. If tunnel liners and/or concrete inverts are needed then the pH of the concrete needs to be constrained and it must withstand 200°C for about 100 years while performing its load bearing mission. Once the impacts on performance and on design are known, then a determination is needed as to what pH is acceptable.

- It is recommended that additional work be done on the potential use of apatite as an additive for inverts and backfill. Some additional work on the reversibility of the sorption process should be done and work should also be done to reduce the uncertainties identified in Section 3, in particular the ability of apatite to sorb Np in the presence of other radionuclides. A sensitivity analysis on how much apatite is needed to provide an appreciable reduction in dose should be done. Additionally, the impacts of apatite on the engineered and natural barriers should be examined. Finally, subsurface design should examine the aspects of emplacing this material and the effect of heat on the materials. Envirostone, another material evaluated, was found to not be a practical addition because too large a quantity is required.

- The performance aspects of line loading including inputs of significantly higher local temperatures, should be examined in the Design Basis Modeling effort and conclusions reached as to whether it provides any appreciable advantage in performance.

- It is recommended that an alternate, low thermal loading of 6.2 to 8.9 kgU/m^2 (25 to 36 MTHM/acre) be carried for LA in addition to the current high thermal loading design. In the potentially higher moisture flux that may exist the lower thermal load may significantly increase performance. The low loading must be established as a viable alternative by producing some limited designs, including it in TSPA cases, developing plans to characterize additional area, and providing cost estimates for this case. These plans would not need to be implemented until a decision is reached to change to a low thermal load.

- Solubilities for such key radionuclides as Np and Tc need to be resolved for the most likely compounds.

Natural Barriers

- The predictions indicate that the CHn may provide a significant amount of performance. The new zeolite conceptualization in the three-dimensional geologic model should be incorporated in the Reference Information Base and included in the performance assessment models for Total System Performance Assessment-Viability Assessment. The proportion of fracture flow and matrix flow in the unsaturated zone, including the CHn, should be established. To do this numerous niche tests are underway or planned. Such Summary Account activities as the fracture-matrix interaction tests (TR33124GB5) and transport studies like TR34141FB5 to name a few should be done and used to update the process models.

- The saturated zone provides a significant amount of performance based on the performance predictions. To improve the performance predictions for licensing will require improved estimates of mixing depth, flow velocity, and dispersion properties. The measurements being taken in the C-Well tests should be evaluated. Additional tracer study tests may be needed to obtain the requisite information.

- For the current design concept, the effects of heat on the PTn, CHn, and saturated zone performance need to be understood. The thermomechanical effects on the barriers and the effect resulting from mineral redistribution, dehydration, and porosity changes need to be estimated and a determination made as to whether or not these changes will affect performance.

A factor of 10 reduction in peak dose was established as an appropriate threshold at which to evaluate performance. The evaluations of the natural barriers show that those evaluated (unsaturated zone transport, zeolite in CHn, and saturated zone transport) all were predicted to produce reductions in peak dose of greater than a factor of 10. In fact, the natural barriers provide significant performance, based on the performance calculations done to date (CRWMS M&O 1996b), and both natural and engineered systems will be needed to meet performance requirements. In the case of the engineered barriers the predictions indicate that those that have potential to reduce the peak dose by a factor of about 10 (the threshold for significant reduction) were zircaloy cladding, galvanic protection, and drip shields. The first two barriers exist or may exist without any additional effort and they would require only small to modest expenditures to establish whether or not licensing credit can be obtained for those barriers. Of course, based on further NRC interactions, more work and therefore more cost may be incurred to do what is necessary to make a licensing case for those barriers. At this time these barriers appear to be attainable at "reasonable" cost (less than $1 Billion). In the case of a drip shield if evaluations find a material that is durable and will survive and function for extended periods of time (at least the regulatory period) then the cost for these drip shields appears "reasonable" at about $400 million to produce enough drip shields.

INTENTIONALLY LEFT BLANK

CONTENTS

INTENTIONALLY LEFT BLANK

FIGURES

FIGURES (Continued)

Page

FIGURES (Continued)

Page

INTENTIONALLY LEFT BLANK

TABLES

INTENTIONALLY LEFT BLANK

1. INTRODUCTION

1.1 STUDY OBJECTIVE

This document reports the work conducted in the Waste Isolation Study from October 1, 1996 to May 15, 1997. The objective of the study is, primarily, to provide documentation of the currently estimated performance of the various barriers (engineered and natural) considered important to waste isolation. The study addresses an approach for providing estimates of the engineered barriers that would provide a significant (see Section 2.2) reduction in peak dose and would identify estimates of the cost for such a reduction.

1.2 SCOPE

The Waste Isolation Study addresses a number of issues in the context of both viability assessment (VA) and License Application. These issues are to:

- Document our current basis of understanding of the performance of various barriers
- Utilize the information to identify the relative merit of the various barriers (engineered and natural)
- Identify the cost of engineered barriers
- Conduct limited calculations to determine the performance of backfill under the potentially higher moisture flux conditions which may exist
- Recommend an approach to evaluate engineering measures that have the potential for significant reduction in peak dose at reasonable cost.

The development of the Waste Isolation Study was managed by the Civilian Radioactive Waste Management System (CRWMS) Management and Operating Contractor (M&O) Mined Geologic Disposal System (MGDS) Systems Analysis and Modeling department. The Waste Isolation Study development task potentially affects the following CRWMS M&O organizations:

- Waste Package Development Department
- Waste Package Materials Department
- Site Evaluation Program Operations
- Repository Design Department
- Performance Assessment Department
- Regulatory and Licensing Department
- MGDS Safety Assurance Department
- MGDS Requirements and Integration/Configuration Management Department
- Project Engineering Office.

The Waste Isolation Study was limited in scope due to the availability of certain data and qualified models. Data developed in accordance with an approved quality assurance program were utilized when available. A portion of the existing site data was collected or developed prior to the approval of a quality assurance program. Also, scoping results from total system performance assessments (PAs), e.g., *Total System Performance Assessment* (TSPA-1995) (CRWMS M&O 1995a), the *Engineered Barrier System Performance Requirements Systems Study Report* (CRWMS M&O 1996a), and *Description of Performance Allocation* (CRWMS M&O 1996b) were used. These results and any unqualified data are identified in this Waste Isolation Study report.

1.3 BACKGROUND

The waste isolation strategy for emplacement of spent nuclear fuel (SNF) and high level radioactive waste (HLW) in the potential repository in the unsaturated zone at Yucca Mountain Nevada relies on a defense in-depth approach using multiple barriers to limit releases of the radioactive wastes to the accessible environment. A successful license application will depend on our ability to establish the effectiveness of the site and the ability of the natural and engineered system to limit releases. It will require an improved understanding of the performance of the various barriers under the conditions that will occur in the potential repository over long periods of time. The Project is just starting to understand how some of these barriers will function and the role that they will play in waste isolation. A number of issues have occurred that may influence how the Project implements a waste isolation strategy and achieves a successful license application. Some of these issues are discussed in this study.

A draft Waste Containment and Isolation Strategy is in the development stages and an overview of this strategy has been issued (Yucca Mountain Site Characterization Project (YMP) 1996). The strategy is a multi-barrier, defense in-depth concept. Understanding what is needed to implement this strategy is of critical importance in the next several years to ensure a successful VA in 1998 and LA in 2002.

The waste containment and isolation strategy (WCIS) (YMP 1996) has identified five major system "attributes" that are the most important with respect to performance of the natural and engineered barriers:

- Attribute 1—Rate of water seepage into the repository
- Attribute 2—Waste package (WP) lifetime (containment)
- Attribute 3—Rate of release (mobilization) of radionuclides from breached WPs
- Attribute 4—Radionuclide transport through engineered and natural barriers
- Attribute 5—Dilution in the saturated zone below the repository.

These attributes are a product of the performance of the natural and engineered barriers. Maturing information about the subsurface and improved process models have reached the point where they can be used to make estimates of the performance of the various potential barriers to radionuclide transport. This current study will use these tools as a basis to develop some predictions of the performance of the respective barriers.

In an effort to focus the Project's design efforts and move that activity forward during a time when there has been no upper level Safety Standard because the Environmental Protection Agency standard was remanded. The Project has assumed a specific standard. The interim safety standard [CDA Key 060 (CRWMS M&O 1996i)] is 15 mrem/yr (whole body dose) from all radionuclides released from the repository, through all exposure pathways. This standard is to be applied at 30 km distance and for a time frame of 10,000 years. In addition, there is a groundwater protection standard of 5 pCi/liter from radium-226 and radium-228; 15 pCi/liter from gross alpha particles, including radium-226 but excluding radon and uranium; and 4 mrem/year from gross beta and photon (gamma) particles. While this interim standard provides a work guide, there is uncertainty in what the ultimate standard will require; in particular, where the standard will be applied (at the accessible environment of 5 km or at the location of a postulated critical group at 30 km) and over what time frame (10,000 or 1,000,000 years). Because of these uncertainties, the interim standard indicates that engineering measures that have potential for significant reduction in peak dose and that can be implemented at reasonable cost, should be evaluated. Thus, this study must examine the implications of these uncertainties since some barriers are likely to be more important for one time frame than for another.

Recent evidence of bomb pulse ^{36}Cl and the measurement and evaluation of the geothermal temperature gradient provide indications that fast paths may exist and the percolation flux in the unsaturated zone may be significantly higher than has been previously expected. Additionally, apparent fracture mineral ages tend to support this. These higher fluxes may affect the performance of a particular barrier. The implication of these higher fluxes on performance of the various barriers is addressed in this study.

The regulatory framework in 10 CFR 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*, anticipated that "during the first 300 to 1000 years...[the containment period], emphasis is placed on the ability to contain wastes by WPs within an *engineered barrier system*...following the containment period special emphasis is placed on the ability to achieve isolation of the wastes by virtue of the characteristics of the geologic repository." The Project can only reduce the uncertainty in how these natural barriers operate to isolate waste and show that the presence of waste will not alter these barriers to the extent that the ability of the natural system to isolate waste is degraded. On the other hand, the Project has, to some extent, some flexibility in deciding what engineered barriers should be used. The interim standard encourages an examination of those engineering measures that would provide at a reasonable cost, a significant reduction in releases of radionuclides, to the environment. Additionally, 10 CFR 60.21 mandates that alternatives be evaluated.

Evaluations were performed to determine whether certain engineering approaches can be employed at reasonable cost. Specifically, one issue that was reexamined was the use of backfill. In light of the potentially higher fluxes, it was necessary to evaluate whether backfill can be used in a cost effective manner to substantially reduce releases.

1.4 REPORT ORGANIZATION

The organization of this report follows the outline in the *Technical Document Preparation Plan for the Waste Isolation Requirements Study* (CRWMS M&O 1997a).

- The Executive Summary provides a top-level description of the study and the results.
- Section 1 provides the study objective, scope, background, and organization of the report.
- Section 2 documents the requirements and standards to include quality assurance (QA) requirements and the inputs and assumptions considered. The qualification or lack of qualification of the input data and analyses is discussed.
- Section 3 provides documentation of the currently estimated performance of the engineered and natural barriers. Where known, the uncertainties associated with understanding the performance of those barriers is identified. This section describes new work that was done in support of the study. Performance predictions at potentially higher fluxes are discussed. In addition, a discussion of the work being done on zeolites and invert additives is provided.
- Section 4 provides the cost estimates for the various engineered barriers considered in the study.
- Section 5 provides the study conclusions and recommendations.
- Section 6 gives the references, standards, and regulations used.
- Section 7 contains the acronym list.
- Appendix A documents previous work done on estimating engineered barrier performance.
- Appendix B documents previous work done on estimating natural barrier performance.

2. REQUIREMENTS AND STANDARDS

2.1 QUALITY ASSURANCE

The Quality Assurance (QA) program applies to the development of this technical document. The QAP-2-0 activity evaluation (IOC LV.SEA.RDM.5/97-032, *Perform System Studies*, from R. Memory to Systems Analysis Department, May 1997) was completed. The QAP-2-0 activity evaluation determined that the work performed to develop a system study report is quality affecting because it impacts items that are on the *Q-List* (YMP 1997) by direct inclusion. This study report, as appropriate, will provide recommendations for requirements to be included in the *Repository Design Requirements Document* (RDRD, YMP 1994a) and the *Engineered Barrier Design Requirements Document* (EBDRD, YMP 1994b). Appropriate procedures, QAP-3-5, *Development of Technical Documents*, in particular, were used in the preparation, review, approval and, if necessary, will be used in the revision of the report. Accordingly, a Technical Document Preparation Plan (CRWMS M&O 1997a) for this document was developed, issued, and utilized to guide its preparation. Other applicable procedural controls not specifically discussed in the Technical Document Preparation Plan are listed in the above-mentioned QAP-2-0 activity evaluation.

In some cases, data and/or computational codes of indeterminate quality were used in this study. A portion of the existing site data was collected or developed prior to the approval of a program. Also, in some cases current scoping results from analyses conducted were used and identified in this study. Any data of indeterminate quality were identified as such in the report. Data developed in accordance with an approved A program were utilized when available. Some computer codes are used in this study that were not controlled in accordance with QAP-SI series procedures (see below). The results of any computer programs not controlled by QAPs were identified. Steps are being taken to ensure the codes comply with the appropriate quality procedures. As such, as additional data are obtained under procedures subject to *Quality Assurance Requirements and Description*, DOE/RW-0333P (U. S. Department of Energy [DOE] 1996), requirements and computer codes are validated, it may be necessary to review these results to determine what impact the new results might have and if any changes may be warranted in the conclusions.

The work documented in this study represents scoping analyses with the intention of facilitating the design process by providing requirements for design. Some of the data and many of the models used to support the development of these requirements are not qualified. The quality status of the data and codes is addressed in each section of the text. It is important to note that neither a legal standard nor regulation related to repository long-term performance exists. Therefore, it was not the purpose of this study to confirm the adequacy of the repository design compared to a performance standard, but rather to assess the performance of the various barriers and recommend the use of engineered barriers that have the potential for significant reduction in releases of radionuclides at a reasonable cost. As such, data and assumptions that are identified in this document are for conceptual design and shall be treated as unqualified; these data and assumptions will require subsequent qualification (or superseding data and assumptions) as the testing and design efforts proceed. This document will not directly support any construction, fabrication, or procurement activity and therefore is not required to be procedurally controlled as TBV (to be verified). In addition, the data and assumptions associated with this analysis are not required to be procedurally controlled as TBV. However, use of any data from this analysis for input into documents supporting procurement, fabrication, or construction are required to be controlled as TBV in accordance with the appropriate procedures.

For the analytic models used in the analysis in this report, the appropriate level of documentation per QAP-3-5 is provided. The inputs and output files for the runs that were conducted were saved, submitted to the records system, and referenced in the report.

Computer codes were employed in this study that have not been controlled in accordance with QAP-SI series procedures. The resulting answers from any computer programs not controlled by QAPs will be unqualified. The computer codes used in the work conducted in support of this study are listed below.

TOUGH2, Version 1.0 of March 1991 developed by Pruess (1991) with T2.FOR Module replaced by T2CG1.F Version 1.1 April 1993. The code was used in conjunction with a processor code, CLIN, developed by INTERA in May 1996 to include the geochemical aspects. The code was run on a DEC ALPHA 333 MHZ. The software does not currently have a configuration control number. The code has not been validated but was used over the range for which it was designed and was appropriate for the application.

RIP, The Repository Integration Program 4.05a, November 1995 was used. The code runs on a Personal Computer (PC). The RIP code was developed and verified using ASME NQA-1 and ISO-9000 Standards (Golder Associates 1995). The code was used over the range for which it was designed and is appropriate for the application.

WAPDEG, Version 2.1 developed by INTERA was also used. This code runs on a Hewlett Packard Workstation (735/100). It has not been validated nor is it configuration controlled. The code was used over the range for which it was designed and is appropriate for the application.

NUFT, Version 4-16-96b, 1996. Developed by Lawrence Livermore National Laboratory (LLNL) and runs on an IBM RISC6000 Powerserver Model 375. This code has not been validated nor is it configuration controlled. The code was used over the range for which it was designed and is appropriate for the applications. This code is described in a 1993 report by Nitao (Nitao 1993).

2.2 MGDS AND DESIGN REQUIREMENTS

The input requirements and many of the assumptions for this study were obtained from the RDRD (YMP 1994a), the EBDRD (YMP 1994b), and the *Controlled Design Assumptions Document* (CDA Document) (CRWMS M&O 1996i). These documents contain a number of requirements that impact design and waste isolation. This section presents the applicable requirements and discusses whether analyses were conducted in this study that may impact these requirements.

The interim postclosure standard used in this analysis is the following (CDA Key 060):

- Exposure Limits:

  15 mrem/year from all radionuclides released from the repository, through all exposure pathways (applies to both disturbed and undisturbed conditions, excluding human intrusion), and

  A groundwater protection standard with the following limits (applies to contamination resulting from repository releases via a groundwater pathway only, under undisturbed conditions):

  5 pCi/liter from radium-226 and radium-228

  15 pCi/liter from gross alpha particles, including radium-226 but excluding radon and uranium

  4 mrem/year from gross beta and photon (gamma) particles.

- Focus of Protection

  For the 15 mrem/year primary standard above, radiological exposure is to be calculated for the average individual in a critical group.

  The characteristics of the critical group (including location of the water supply well) will be based on the current day demographics and living habits of individuals living down gradient from the repository, in the Amargosa Valley area (about 30 km from Yucca Mountain).

  For the groundwater protection standard above, evaluation of compliance will be determined both at 20 km and 30 km down gradient from the repository.

- Regulatory Time Frame

  The above limits apply during the first 10,000 years after closure.

  PAs will be conducted past 10,000 years, out to the time of peak dose, in order to gain insight regarding longer-term repository performance.

  These longer-term analyses will be included in the Environmental Impact Statement (EIS).

  Engineering measures that have potential for significantly reducing the peak dose, and could be implemented at reasonable cost, will be evaluated for possible inclusion in the reference design. [CDA Key 060]

The backfill position, which is being evaluated in this study, follows:

Backfill in emplacement drifts is not required. However, the repository design should not preclude the use of emplacement drift backfill at the end of the preclosure period. The specifications for the emplacement drift envelope to accommodate are:

- Level single layer backfill (quartz sand, crushed tuff, or other material of similar favorable thermohydrologic properties)

- Waste packages initially covered with at least 0.6 meters of material. [CDA Key 046]

The following assumption will be used as the basis for establishing the preclosure period of 100 years in the calculations.

> "The repository will be designed to permit waste retrieval for up to 100 years after the initiation of waste emplacement."[CDA Key 016]

This requirement may be changing to 50 years, but the calculations were all done at the existing requirement of 100 years.

The following assumptions address the current thermal loading design.

> Current repository design activities focus on a reference design thermal load that will permit emplacement of at least 70,000 metric tonnes of initial heavy metal (MTHM) within the primary repository area (see RDRD 3.7.2.1.D), and produce dry conditions around the WPs. The current working hypothesis is that a reference areal mass loading of 80-100 MTHM per acre (commercial spent fuel) should produce an average areal thermal loading of about 80 to 100 kW per acre at the time of waste emplacement, and will satisfy both criteria. Prudent levels of flexibility will be maintained by including alternative areal mass loadings through design options and through operational parameters. As laboratory and field test data and more refined analyses become available, a preferred, specific thermal load will be selected.
>
> Risks associated with this approach, such as unexpected and undesirable site responses, will be mitigated by maintaining design and operational flexibility to accommodate a range of areal mass loadings, and by pursuing a performance confirmation program to validate preclosure predictions, to increase confidence in postclosure predictions.
>
> "Thermal Loads. The underground facility shall be designed so that the performance objectives will be met taking into account the predicted thermal and thermomechanical response of the host rock, and surrounding strata, and groundwater system. [10CFR60.133(I)] [RDRD 3.7.5.E.7]"

The following requirements, in part, make it necessary for the natural and engineered barriers to function together within the thermal environment to ensure that satisfactory waste isolation is achieved.

"Mission Requirement. The design of the repository segment shall provide for the disposal of SNF and civilian and DHLW such that the public health and safety and the environment are protected. [NWPA 42USC10131(a)(4)] [NWPA 42USC10131(b)(1)] [1985 Presidential Memo]"

"The Engineered Barrier Segment, shall be designed to ensure that releases of radioactive materials from the Engineered Barrier Segment, and then through the geologic setting to the accessible environment following permanent closure, conform to applicable environmental standards for radioactivity established by the Environmental Protection Agency with respect to both anticipated processes and unanticipated processes and events. [10CFR60.112] [EBDRD 3.7.B]"

"The Engineered Barrier Segment shall be designed, assuming anticipated processes and events affecting the geologic setting, so that containment of radioactive material within the waste packages will be substantially complete for a period to be determined by the NRC but not less than 300 years nor more than 1000 years after permanent closure of the geologic repository <TBV>. [10CFR60.113(a)(1)(ii)(A)] [EBDRD 3.7.D]"

This requirement has been modified in the CDA Document as "the EBS shall be designed, assuming anticipated processes and events, so that containment of radioactive material within the WPs will be substantially complete for 1000 years (with less than one percent of the WPs breached at 1000 years after permanent closure of the geologic repository) and with a mean WP lifetime well in excess of 1000 years. [10CFR60.113(a)(1)(II)(A)] [CDA EBDRD 3.7D]"

"The Engineered Barrier Segment shall be designed, assuming anticipated processes and events, so that 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 that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the NRC; provided, that this requirement does not apply to any radionuclide that is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit is defined to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay. [10CFR60.113(a)(1)(ii)(B)] [EBDRD 3.7.E]"

Although possibly subject to modification if some form of interim storage is established, the analysis done in this study, except for some modification in Section 5 where storage options are examined, uses the following requirement on waste receipt rates.

"The repository shall be capable of receiving waste according to the schedule shown in Table 3-1 of the CRD (Rev 3) [DOE 1996b] [10CFR60.3(a)] [NWPA 42USC10222(a)(5)] [RDRD 3.2.1.2.B][CDA Key 003]"

The total amount of fuel considered for emplacement in the proposed repository at Yucca Mountain is based on the current requirement.

"Assuming the Monitored Retrievable Storage facility is located more than 50 miles from the repository, no quantity of SNF and solidified HLW resulting from the reprocessing of such a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal shall be emplaced in the repository until such time as a second repository is in operation. [RDRD 3.2.1.2.A] [NWPA 42USC1013(d)]"

CDA Document Key Assumption 003 provides for a total of 63,000 MTHM of SNF to be emplaced in the repository.

CDA Document Key Assumption 005 provides for a total of 7000 MTHM of HLW to be emplaced.

Some of the thermomechanical evaluations and considerations for tunnel support are based on a requirement for underground openings.

> "Openings in the underground facility shall be designed to reduce the potential for deleterious rock movement or fracturing of overlying or surrounding rock. [10CFR60.133(e)(2)] [RDRD 3.7.5.E.2]"

Thermal effects need to be considered on WPs.

> "The design of waste packages shall include but not be limited to consideration of the following factors: solubility, oxidation/reduction reactions, corrosion, hydriding, gas generation, thermal effects, mechanical strength, mechanical stress, radiolysis, radiation damage, radionuclide retardation, leaching, fire and explosion hazards, thermal loads, and synergistic interactions. [10CFR60.135(a)(2)] [EBDRD 3.7.1.B]"

> "Limit the fuel cladding temperature to less than 350°C [CDA DCWP 001]"

The current design layouts for the potential repository geometry, layout, and depth were established to satisfy the following.

> "The orientation, geometry, layout, and depth of the underground facility, and the design of any engineered barriers that are part of the underground facility shall contribute to the containment and isolation of radionuclides. [10CFR60.133(a)(1)] [RDRD 3.7.5.E.3]"

The EBDRD (YMP 1994b) originally had a requirement for boreholes to keep the temperature one meter into the drift in a borehole to less than 200 °C to avoid deleterious thermomechanical effects [EBDRD 3.7.G.2]. Based on previous work (CRWMS M&O, 1993) the criterion was changed to keep the emplacement drift wall temperature below 200°C during the preclosure period. This requirement documented in the CDA is as follows.

> "Keep emplacement drift wall temperatures <200°C"[CDA EBDRD 3.7.G.2]

The CDA has a requirement to limit the zeolite in the host rock beneath the potential repository to less than 90°C. The statement of this requirement is the following:

"The temperature at the average top of the zeolite layer beneath the potential emplacement area shall not exceed 90°C. The vertical distance from the emplacement area horizon to the average top of the zeolite layer in the Primary Area is estimated at 170 m. [CDA DCSS 025 in revision]"

An assessment shall be provided to document the predicted effectiveness of engineered and natural barriers, including barriers that may not be themselves a part of the geologic repository operations area, against the release of radioactive material from the WP to the environment. The analysis will also include a comparative evaluation of alternatives to the major design features that are important to waste isolation, with particular attention to the alternatives that would provide longer radionuclide containment and isolation. [RDRD 3.3.1.H] [10CFR60.21(c)(1)(ii)(D)]

The interim standard [CDA Key 060] indicates that engineering measures that have potential for significant reduction in peak dose and can be implemented at reasonable cost, should be evaluated. As such, a study objective was to address the question as to what thresholds should be provided for specifying 'significant' reduction in peak dose for a 'reasonable' cost. The current interim safety standard (see discussion above) requires that peak doses not exceed 15 mrem/yr from all radionuclides released from the repository through all exposure pathways at a distance of 30 km and over a time span of up to 10,000 years. However, in light of the long half-lives of several of the radionuclides, there is a potential that the peak doses will occur beyond the 10,000 year limit. In recognition of this potential, the interim standard also requires consideration of engineering measures with the potential to significantly reduce the dose at reasonable cost for time periods beyond 10,000 years. There is little guidance that we can rely on to help define what is meant by significant reduction. However, predicting doses at long times (10,000 years and beyond) involves a significant level of uncertainty. Therefore, a predicted reduction in dose by a factor of two or three may easily be within the range of uncertainty in predicting the performance of the engineered and natural systems. Hence, the approach proposed in this study is to define a 'significant' reduction threshold as a reduction in peak dose by a factor of 10 or more. Such a predicted reduction should be indicative that a true reduction in the peak doses can be realized. The approach for identifying 'reasonable' cost thresholds can be drawn from the ALARA (as low as reasonably achievable) principle. The objective of ALARA is to limit personnel and environmental radiation exposure to the lowest levels achievable commensurate with sound economic and social considerations. Based on the 1997 Program Cost Estimate (CRWMS M&O 1997b), and adjusting for a 70,000 MTHM repository, the post-Development-and-Evaluation costs for the repository are estimated to be roughly $13 Billion (FY 1997 dollars). Given this total repository cost, an argument can be made that sound economic and social considerations would mandate that significant reductions in peak dose estimates costing less than $1 Billion dollars (about 8 percent of the total cost) should be considered for possible inclusion in the reference design. Specifying such thresholds is, of necessity, subjective and final designation of these thresholds will require further evaluation. However, for the remainder of the report, engineered or natural barriers that can be implemented within these thresholds will be identified.

2.3 INPUTS AND ASSUMPTIONS

This section identifies the reference case, the major input conditions for the study, and the assumptions used. The primary parameters used such as WP size, emplacement tunnel dimensions, and a general repository layout are described below. In many cases specific input parameters and/or assumptions for a given analytic model or for a particular option are discussed in the pertinent section of the report. The origin of the inputs and their quality are identified where appropriate. However, in some cases where these inputs have been used in previous analyses the reader will be directed to the appropriate references for the supporting information. Many inputs and analyses necessary for this study were obtained from the supporting organizations using a QAP-3-12, Design Input Request and Transmittal. This procedure requires supporting organizations to notify us in the future if any changes are made to those inputs. The following identifies the basic assumptions and inputs used in the study.

Thermal Loading—The thermal load in this report will be referred to in terms of an area mass loading (AML) of SNF. The units of the area mass loading are generally kilograms Uranium (taken in this report to be equivalent to kilograms initial heavy metal) per square meter (kgU/m^2) or alternatively metric tonnes Uranium per acre (MTHM/acre). The area mass loading is used since it tends to be most representative of long-term, mountain-scale performance (Buscheck, Nitao, and Saterlie 1994). Another method of identifying the thermal load is in terms of an area power density because it influences the early-time (first few hundred years), drift scale environment.

The base case or reference thermal loading for these calculations was 20.5 kgU/m^2 (83 MTHM/acre). Other thermal loads were considered as necessary for evaluations of alternate thermal loads or to determine when potential thermal effects may arise. An additional case was run at 6.2 kgU/m^2 (25 MTHM/acre) to examine the effect of high fluxes on lower thermal loads. The calculations in this study define MTHM/acre in terms of the amount of SNF, and omit any DHLW in the AML, due to its low heat output. However, that heat is included in the calculations.

Waste Package—A single large WP for pressurized water reactor (PWR) and boiling water reactor (BWR) assemblies was used. Defense high-level waste (DHLW) were included in the three-dimensional calculations. The studies used as the basis for this report used two-dimensional analyses and thus the variance of WP type and spacing in three-dimensions was not possible. As such, the model runs used a single package that had average characteristics representative of a waste stream which would have 40 percent BWR and 60 percent PWR fuel. The burn-up assumptions for the BWR fuel were 31 GWD/MTHM and a 40 GWD/MTHM for the PWR fuel. The average package had fuel loading of 7.1 MTHM/package (CDA Key 004). The WP dimensions were 5.335 m in length and 1.629 m in diameter. Additional details can be found in TSPA-1995 (CRWMS M&O 1995a).

The waste package corrosion models used in the calculations for this study have been updated since TSPA 1995. The Performance Allocation Study and the Engineered Barrier System Performance Requirements Systems Study both use the corrosion model described in TSPA 1995 (CRWMS M&O 1995a).

Waste Stream—As indicated above the calculations in this study used an average waste stream that represented the mix of PWR and BWR fuel anticipated for disposal. This average fuel had characteristics of an oldest fuel first waste stream with an average age of about 26 years.

Subsurface—The reference case is the case established in the *Mined Geologic Disposal System Advanced Conceptual Design Report* (ACD Report, CRWMS M&O 1996e). The potential repository horizon location at Yucca Mountain has been identified as the welded, lithophysae-poor, ash-flow tuffs in the Topopah Spring Tuff of the Paintbrush Group. This rock is unsaturated but the pores (about 13% porosity) are 85 to 95 percent filled with water. The actual rock properties and saturation levels used in the calculations are a function of depth and are based on work compiled by Bodvarsson, et al. (1996). The repository location is defined as the Primary Area and described in the ACD Report (CRWMS M&O 1996e).

The estimated overall average percolation flux (average of fractures and matrix) at the repository horizon is 6.2 mm/yr. This value is about three times the highest percolation flux unaffected by climate change considered in TSPA-95. This updated site-scale unsaturated zone (UZ) flow model is based on an isothermal groundwater flow simulation using a three-dimensional dual permeability model with the most updated hydrogeologic parameters obtained via inverse modeling (ITOUGH). In this model the range of the measured laboratory hydrology data is used to fit the field measured surface infiltration rate and observed saturation profile data (Bodvarsson et al. 1996).

The repository depth below the surface of the mountain and the distance to the water table varies across the repository area. However, for these thermohydrologic calculations an average distance to the water table ranged from 310 to 450 m. The average distance used in the calculations reported in Section 3 was about 350 m. (Note the thickness of zeolite varied from 75 to 145 m).

The subsurface repository considers emplacement of WPs in horizontal emplacement drifts. These emplacement drifts have a diameter of 5 meters. The WP spacing (center to center) was set at 15.4 m and the drift spacings were 22.5 m for the base case. This provided the area mass loading of 20.5 kgU/m^2.

Rock Properties—In some cases for specific evaluations, properties were selected using data obtained from recent subsurface and surface drilling programs and associated laboratory testing. Some of the rock properties were based on analytic calculations using a site scale model where the properties were varied until the results matched such measured values as saturation and ambient thermal gradient. Where this was done, the information is identified and the quality is reported. All other inputs were taken from the *Reference Information Base* (RIB, Rev 04, YMP 1995).

INTENTIONALLY LEFT BLANK

3. PERFORMANCE PREDICTIONS OF POTENTIAL BARRIERS

3.1 INTRODUCTION AND SYSTEM DESCRIPTION

The waste isolation strategy for emplacement of SNF and high-level radioactive waste (HLW) in the potential repository in the unsaturated zone at Yucca Mountain Nevada relies on a defense-in-depth approach using multiple barriers to limit releases of the radioactive wastes to the accessible environment. A successful License Application in 2002 will depend on our ability to establish the effectiveness of the site and the ability of the natural and engineered systems to limit releases. The purpose of this study was to estimate the performance of the various potential barriers, natural and engineered, and, based on that, recommend which barriers should be pursued and what testing or design is needed. This section presents a discussion of the performance calculations that were done in this study and in previous work to identify the performance of various engineered and natural barriers. For the most part this analysis is scoping or preliminary in nature, was not intended to qualify a specific barrier's performance and should be used as a means of evaluating the performance of one barrier relative to another.

The mission of the MGDS is to provide for emplacement and isolation of the nation's commercial SNF and DHLW in such a way that public health and safety are protected. The potential MGDS will be able to accommodate about 70,000 metric tonnes initial heavy metal (MTHM), which currently is assumed to be composed of about 63,000 MTHM of SNF from commercial reactors, about 4,700 MTHM equivalent HLW from reprocessing defense materials, and about 2,300 MTHM of DOE SNF (CRWMS M&O 1996c).

The Code of Federal Regulations requires that for the potential repository the ".....design of any engineered barriers....shall contribute to the containment and isolation of radionuclides" [10 CFR 60.133(a)], and "engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure" [10 CFR 60.133(h)]. Thus, the engineered barriers must work together with the natural system to contain waste. The Project has the flexibility to include or not include some engineered barrier concepts (such as backfill). The engineered and natural barriers considered in this study are listed in Table 3-1 and are discussed below, although full performance calculations were not possible for all of the barriers at this time.

The WCIS (YMP 1996) was developed to assist the YMP in prioritizing testing and analysis activities to focus on the most important remaining issues regarding postclosure safety. The WCIS is designed to help resolve uncertainty in the processes and parameters of greatest significance to long-term performance. The WCIS has identified five major system "attributes" that are the most important with respect to performance of the natural and engineered barriers:

- Rate of water seepage into the repository
- Waste-package lifetime (containment)
- Rate of release (mobilization) of radionuclides from breached WPs
- Radionuclide transport through engineered and natural barriers
- Dilution in the saturated zone below the repository.

Table 3-1 Potential Barriers Considered

Engineered Barriers	Natural Barriers
Cladding	Alluvium/colluvium
Waste Package	PTn
Galvanic Protection	Unsaturated zone transport[1]
Pedestal or WP mount[2]	CHn Sorption
Invert additives	Saturated zone transport
Drip shield	
Backfill	
Richards Barrier Backfill[2]	
Tunnel liner[2]	
Repository configuration	

[1]The unsaturated zone transport includes the CHn
[2]No performance calculations done.

The work in this effort focuses on recent TSPA calculations that were done in this study. These recent calculations were done at the potentially higher percolation flux that are now expected to exist. Some earlier calculations were also done in an *Engineered Barrier System Performance Requirements System Study* (CRWMS M&O 1996a) and a *Description of Performance Allocation Study* (CRWMS M&O 1996b), but in these calculations lower percolation fluxes were used. The results of these previous calculations are documented in the appendices and where necessary are used in this present study. A full suite of sensitivity studies could not be done for this present work and thus, where necessary for a few barriers, calculations of performance for that particular barrier were drawn from the previous work. These cases are identified.

Engineered Barriers

Various types of engineered barriers have been considered. A conceptualization of the various barriers is shown later in Figure 3-44 and is discussed below. Most SNF assemblies have a zircaloy cladding around the fuel pellets. Although the cladding barrier is relatively thin, zircaloy is a very durable material that can provide a barrier to radionuclide transport. At most, up to about one percent of the SNF assemblies have stainless steel claddings and the defense HLW has no cladding but is contained in a glass log within a stainless steel canister. Work has been done to develop cladding degradation models as a result of creep rupture (Chin and Gilbert 1989, Peehs and Fleish 1986). A process model was developed and preliminary performance calculations were done in two recent studies (CRWMS M&O 1996b and 1996d) that showed that cladding may provide a significant barrier to radionuclide transport. One study determined that cladding performance is sensitive to temperature and recommended that a temperature limit of 350°C be retained for cladding (CRWMS M&O 1996d). Durability of cladding under static loads was not adequately addressed, however. The results of the performance calculations are discussed below.

The current design of the WP has two barriers with the outer barrier being a corrosion allowance material of ASTM A 516 steel and an inner barrier of a corrosion resistant nickel alloy, ASTM B 443 (Alloy 625) (CRWMS M&O 1996f). The PA calculations in this study used corrosion curves for Alloy 825 rather than Alloy 625 since the corrosion curves for the new material were not yet available. The thickness of the outer corrosion allowance barrier is 100 mm and the Alloy 625 inner barrier thickness is 20 mm (CRWMS M&O 1996g). The assumptions used for WP degradation are that humid-air general and pitting corrosion of the outer barrier occurs at a relative humidity threshold of between 65 percent and 75 percent. Aqueous general and pitting corrosion of the outer barrier occurs when relative humidities exceed 85 percent to 95 percent. The inner barrier is subject to aqueous localized corrosion. The relative humidity threshold for corrosion initiation of the outer barrier was selected for each WP at random from the range of 65 to 75 percent. For the inner barrier 20 percent was added to the selected number. The corrosion models for these calculations are reported in TSPA 1995 (CRWMS M&O 1995a). It is assumed that corrosion initiation does not begin until the WP surface temperature drops below 100°C. Some estimates of the performance contribution of the WP were developed in the performance allocation study (see Appendix B) but few calculations have been done in which the analysis was run with and without the WP to be able to estimate the contribution of that element of the engineered barriers.

The double-walled WP will result in some degree of galvanic protection of the inner barrier once the outer barrier is breached due to the formation of a galvanic couple between the outer barrier and the inner barrier. The degree of galvanic protection is uncertain until results of tests underway are evaluated. However, using expert judgement, estimates of the amount of galvanic protection were done and these were used to estimate the reduction, if any, in releases to the accessible environment (CRWMS M&O 1995a, CRWMS M&O 1996g, CRWMS M&O 1996b). These results indicate that significant improvement in performance may be achieved with galvanic protection and the specifics of these analyses are described below.

The WPs will be mounted on a pedestal or similar type of holder to keep them centered in the drift and elevated from the emplacement drift floor and away from any potential liquid water that might collect on the floor. No performance calculations have been done for these mounts as yet. Currently no performance is allocated to these. In the TSPA calculations it is assumed that the WP is resting on the floor of the emplacement drift. In addition, the floor of the emplacement drift is assumed to have an invert composed of crushed tuff or other material such as concrete (CRWMS M&O 1996e). In some cases this invert could have minerals or chemicals added to the mix which might provide some sorptivity of radionuclides. Such compounds as apatite (CRWMS M&O 1996a) or envirostone have been suggested as additives. Preliminary scoping estimates of performance were done in the engineered barrier study. Additional work was done to determine just how much additive of the types above would be needed to provide an appreciable increase in performance, and this report provides the results of that work in Section 3.4.

Drip shields over each WP have been suggested as a possible approach to divert water. These drip shields might be used with or without backfill. The long-term survivability of these drip shields has not been evaluated. Estimates of performance for cases with drip shields was done in this study and in the *Engineered Barrier System Performance Requirements Systems Study* (CRWMS M&O 1996a) which is discussed in Appendix A.

Backfill is a concept that has been suggested as a possible component to enhance performance. The *Engineered Barrier System Performance Requirements Systems Study* primarily examined system performance using backfill (CRWMS M&O 1996a). There are several issues that were addressed in the study. Specifically, using backfill can result in WP internal temperatures exceeding 350°C. Methods for limiting these temperatures include limiting backfill thermal conductivity, implacement timing, and WP spacing. Thus, depending on the time that is spent at these higher temperatures, the cladding could be degraded and one may be trading one barrier for another. It should be noted that thermal calculations (CRWMS M&O 1996a) found that most WPs did not exceed the cladding criteria for backfilling at 100 years. Emplacing backfill after the WPs are emplaced was determined to have operational implications and there are increased costs with using backfill. TSPA calculations were performed and these are reported below. The study cited above determined that, although there was estimated to be an order of magnitude improvement in performance with backfill the base case at low fluxes (0.5-2 mm/yr) had significant performance margin without backfill. Thus, it was determined not to include backfill at this time but, because of the uncertainties in the calculations, backfill should not be precluded. These calculations were done at a lower infiltration rate than what is currently believed to exist in Yucca Mountain. Thus, calculations were done in the present study at the higher fluxes anticipated to exist.

Another type of backfill considered in the Engineered Barrier System Study was a Richards Barrier, which is a multi-layer backfill with layers of different porosity. This has an analog in Japanese burial mounds (Conca and Wright 1992). However, emplacing such a multi-layer backfill does not appear to be feasible at this time due to the difficulty emplacing two layers in a hostile environment and confined space (CRWMS M&O 1996a). Some seismic shaking could also result in fingering of sand into the gravel, creating the potential for breakdown of the capillary barrier. Thus, it was not considered in Section 3.2.

A concept being considered by subsurface design is to use concrete tunnel liners to maintain tunnel stability through the 50 to 100 year operational phase. A liner can alter the hydrology but no calculations have yet been done to evaluate this. Some calculations have been done concerning potential impact of cementitious materials in the emplacement drifts and the performance implications will be briefly discussed in this report.

The spacing of drifts and WPs can have implications for waste isolation. These repository configuration issues have been investigated to some extent in previous studies (CRWMS M&O 1996a and 1996d) where the effects of separating WPs by only 1 m or less were examined. The thermal loading or density of the WPs in a given area can be varied and may affect performance. This line load concept did offer some benefit to moderating package-to-package heat variations but there were operational considerations as a result of the higher temperatures. These concepts continue to be investigated and some performance calculations are discussed below.

Natural Barriers and Site Description

The site of the potential repository at Yucca Mountain is located approximately 100 miles northwest of Las Vegas, Nevada in a relatively arid climate. Two of the waste isolation attributes of this site identified in the *Site Characterization Plan* (SCP, DOE 1988) are that the site is located in an area of relatively sparse population and that it is in an arid climate, which would limit recharge of water.

The site is also on the Nevada Test Site which has been used extensively for nuclear testing and the Nellis Air Force Range (DOE 1988). A portion of this site was also Bureau of Land Management land.

The potential repository location in Yucca Mountain currently being evaluated is in the Topopah Spring Member, a welded tuff unit of the Paintbrush tuff (see Figure 3-1). The Topopah Spring Member is approximately 330 m thick and dips from west to east by about six degrees. The potential subsurface layout is primarily in the Topopah Spring welded unit 2 (TSw2), which provides a minimum overburden of 200 m and is a distance of 230 to 380 m above the water table (CRWMS M&O 1996c) for the six calculational columns used in the modeling.

The strata of Yucca Mountain have been generalized into five hydrostratigraphic units that differ from one another in average properties (Montazer and Wilson 1984). These different units, in descending order, are:

- Tiva Canyon welded unit (TCw)
- Paintbrush tuff nonwelded unit (PTn)
- Topopah Spring welded unit (TSw)
- Calico Hills nonwelded unit (CHn)
- Crater Flat undifferentiated unit (CFu), which is composed of the Prow Pass Tuff and the deeper Bullfrog Tuff.

The welded units TCw and TSw have small matrix porosities and permeabilities but have larger bulk permeabilities because these rocks fracture easily. On the other hand, the PTn and CHn have larger porosities with small bulk permeabilities because it is believed they have much fewer fractures. The hydrologic properties of these units are summarized by Bodvarsson et al. (1996).

Because the region is arid, the recharge of groundwater is low and the amount of moving groundwater is also relatively low. Climate changes may occur and these can be estimated through geologic records (Long and Childs 1993).

The first barrier to water infiltration on the mountain, that is present in some allocations, is the unconsolidated alluvium. The alluvium/colluvium has a relatively large storage capacity to retain moisture, which generally allows removal of this moisture by persistent evapotranspiration. However, the alluvium/colluvium is not uniformly distributed and, on side slopes and ridge tops, it may be thin or absent allowing higher infiltration rates (Flint and Flint 1995 and in private communication with those authors).

The significant change in permeabilities between the fractured TCw and the less fractured and hence smaller permeability PTn provide for a significant contrast, which is likely to impede episodic flow of percolating water in the matrix. The contrast in permeabilities between the PTn and underlying TSw layer encourages down-dip diversion of water flow. Earlier studies (CRWMS M&O 1996d) have identified potential thermomechanical issues associated with the PTn and the potential to increase fracture sizes. This effect needs further evaluation.

The TSw matrix in which the potential repository is located has low porosity and permeability with relatively high saturation of the pores of 85 to 95 percent. These conditions would tend to favor imbibition of water from fractures into the rock matrix. However, in some cases this imbibition may be inhibited by mineral deposition in the fractures and the small permeability of the rock. The greatest downward flux of water through this host rock is anticipated to be primarily in the fractures. Studies in the Exploratory Studies Facility have confirmed localized regions of elevated concentrations of ^{36}Cl which tend to confirm fracture flow in certain areas (Fabryka-Martin et al. 1996). This will be further discussed below.

The CHn hydrogeologic unit beneath the potential repository consists of glassy and variably zeolitized nonwelded and partially welded ash flow tuffs and bedded tuffs, extending vertically downward to the water table from the basal vitrophyre in the overlying TSw unit. The zeolites in this layer are predominately clinoptilolite with some mordenite and smectite and, in some deeper areas, analcime. These minerals are hydrous minerals that have a significant affinity for water. In addition, these zeolites, particularly clinoptilolite, have sorptive capacities for a number of radionuclides, particularly Cs, Sr, and to some extent, Np (Meijer 1990). Thus, based on initial studies, they provide significant retardation and increases in travel times for a number of radionuclides (CRWMS M&O 1996d). The location, depths, and concentrations of these zeolites is uncertain at this time since the conceptualizations are based on a limited amount of data from boreholes. This information is summarized in the *Thermal Loading Study for FY 1996* (CRWMS M&O 1996d). Additional borehole data have been or are being analyzed and work is underway to improve the zeolite conceptualizations (this ongoing effort is discussed below).

The saturated zone is the final stage in the path for water soluble radionuclides (nongaseous) to reach locations where there is the potential for drawing water from these regions and exposing the public. Locally, beneath the potential repository, the configuration of the potentiometric field defines a water table that would indicate generally southward flow, joining with eastward flow to produce a southeastward direction of flow away from the repository site based on work done by Robinson (1994) and Luckey et al. (1996). The saturated zone will provide dilution and dispersion of radionuclides during transport. These are functions of the flow velocity and the rock permeability and structural properties of the medium. Additionally, the mixing depth in the saturated zone, which is currently uncertain, will determine the extent of dilution and whether a vertically well-mixed plume of radionuclides results, or a more concentrated plume near the surface of the saturated zone is prevalent. For the calculations in this study, a mixing depth of 50 m, and three values of Darcy flux of 0.1, 0.31, and 1.0 m/yr, were used. More details of the saturated zone flow can be found in the work of Fridrich et al. (1994) and Luckey et al. (1996).

3.2 PERFORMANCE CALCULATIONS

Sensitivity calculations were done by varying a number of parameters. Most of the calculations were done using two-dimensional thermohydrologic calculations. A few calculations were done with a three-dimensional thermohydrologic code. The thermohydrologic code used in both cases was the NUFT code (Nitao 1993). The temporal and spatial distributions of temperature and relative humidity calculated by NUFT were then used in the waste package degradation code, WAPDEG. The dissolution of radionuclides and transport to the accessible environment was calculated with the RIP.

The input and output files for these runs are saved under directory O:\PA\WISSYSTM\ which is located on the server RWYMNS1\DATA1\GROUP. These were saved from about March 11 to March 18, 1997. Additional runs were saved from April 21 to May 2, 1997. The program configurations used (see Section 2) were also saved. These files will be included, on magnetic media, with the final records package. The computer codes used were not qualified, however, the calculations were received via a QAP-3-12 Design Input Transmittal.

As indicated above, the WCIS (YMP 1996) has identified five major system "attributes" that are the most important with respect to performance of the natural and engineered barriers:

- Attribute 1—Rate of water seepage into the repository
- Attribute 2—Waste-package lifetime (containment)
- Attribute 3—Rate of release (mobilization) of radionuclides from breached WPs
- Attribute 4—Radionuclide transport through engineered and natural barriers
- Attribute 5—Dilution in the saturated zone below the repository.

To address the performance effects of these five major system attributes, the TSPA analyses in this study have considered the effect of uncertainty in a number of the most critical system parameters, both design (engineered-system) parameters and natural-system parameters. Calculations were done in this effort to examine the effect that higher potential fluxes might have on the estimates of performance. Table 3-2 lists the parameters that were varied in this study and the WCIS attributes (see above) most affected by the various parameters. Most of these calculations focus on the engineered barriers and engineered system but a couple of calculations done under this TSPA work address natural system attributes.

Table 3-2 Parameters Varied in TSPA Analyses, and Corresponding WCIS Attribute

Parameter	WCIS Attribute
Thermal loading	1,2,3,4,5
Thermal backfill	1,2,3,4
Degree of galvanic protection of inner barrier	2
Integrity of fuel-rod cladding	3,4
Mode of water contact with the waste form and package	2,3,4
Neptunium sorption to zeolite in the unsaturated zone	4
Saturated-zone flux and velocity	5
Down gradient location of wells in saturated zone	5
Time frame of interest	n/a

Assumptions and Base Case

The sensitivity analyses (i.e., sensitivity of system performance to variations in the parameters in Table 3-2) in this study are built upon the analyses in TSPA-95 (CRWMS M&O, 1995a) and the Performance Allocation Study (CRWMS M&O, 1996b), but include updated information on design and site characterization as of this writing. The case chosen to evaluate is considered a conservative case based on the choices of parameters chosen. The major modifications are as follows:

A. Liquid fluxes and velocities (in both fractures and matrix) in the unsaturated-zone (UZ) hydrogeologic units beneath the repository are based upon the most recently calibrated site-scale flow model (Bodvarsson et al. 1996). In fact, the fluxes and velocities in the present study are updates to the published values in the Bodvarsson et al. report. The estimated overall average percolation flux (average of fractures and matrix) at the repository horizon is 6.2 mm/yr. This value is about three times the highest percolation flux unaffected by climate change considered in TSPA-95. This updated site-scale UZ flow model is based upon an isothermal groundwater flow simulation using a three-dimensional dual permeability model with the most updated hydrogeologic parameters obtained via inverse modeling (ITOUGH). In this model the range of the measured laboratory hydrology data is used to fit the field measured surface infiltration rate and observed saturation profile data (Bodvarsson et al. 1996). The simulated liquid saturations and fluxes in the rock matrix and fractures were extracted for six columns of the three-dimensional model, which were selected to represent the footprint of the potential repository. These saturations and fluxes, together with geometric data, were further processed to obtain liquid saturations and fluxes and layer thicknesses for all the layers within each of the six columns from the potential repository horizon to the water table. These processed data, together with the rock matrix permeabilities and porosities at the potential repository horizon, were employed in the total system calculations in the RIP TSPA Code.

B. Backfill properties are updated based upon recent studies by Conca and Wright (1996). In particular, the characteristic curves for crushed TSw2 backfill have been used for the invert properties and for the sensitivity cases that consider overfill as a design option (i.e., backfill).

C. Updated drift-scale thermohydrologic calculations have been performed based on the new calibrated property values in the latest UZ flow model and based on the new Conca and Wright backfill properties. This results in updated values for relative humidity and temperature at the waste-package surface and liquid saturation and temperature in the invert. These temperatures, relative humidities (RH), and liquid saturations were calculated using the computer code NUFT (Nonisothermal Unsaturated-Saturated Flow and Transport) [Nitao 1996]. NUFT is able to simulate the coupled transport of water, vapor, air, and heat in porous media, including conceptual models for fractured porous media. For this work the fractured tuff surrounding the repository drift was conceptualized using the equivalent continuum model (ECM). In the ECM, the assumption of capillary-pressure and thermal equilibrium between fracture and matrix allows the fracture and matrix properties to be pore-volume-averaged into an equivalent continuum. In addition to ECM, which is most effective for steady-state moisture and transient gas, dual-continua methods may be more appropriate for modeling transient pulses, transport, and thermal loading. Each model is a simulation of the processes that occur and each has their advantages and disadvantages.

A two-dimensional grid is used for the thermohydrological simulations that reflects the center-in-drift waste-package emplacement scenario described in the Advanced Conceptual Design (CRWMS M&O 1996e). Symmetrical repository drift and WP spacing is assumed, thus the simulations reflect the conditions of an average WP that is relatively distant from the repository edge and which were smeared along the drift length. The 20.5 kgU/m^2 (83 MTHM/acre) model reflects a drift spacing of 22.5 m. and an in-drift package spacing of 15.4 m, while the 6.2 kgU/m^2 (25 MTHM/acre) model has a drift spacing of 45 m and an in-drift package spacing of 25.56 m. Thermal loading is accomplished by assuming an average fuel age of 26 years and that the boiling water reactor fuel occupies 40 percent of the repository while pressurized water reactor fuel occupies the other 60 percent. Defense high level waste has a negligible impact on the thermal loading. Because of the limitations of the two-dimensional calculations the thermal output from a single WP was smeared over an equivalent area to approximate three-dimensional effects. This will result in somewhat lower WP temperatures and higher relative humidities than what would be seen in three-dimensional models. It should be noted that because the calculations were two-dimensional and used a smeared heat source, the choice of specific WP spacings and drift diameters was not important to the results. Some three-dimensional calculations are presented in Section 3.3.

The simulations begin with a prefill period of 100 years when no backfill is present over the WP. In the no backfill case, gray body radiative heat transport connections are established between the WP and the drift wall with an assumed WP surface emissivity of 0.8. After 100 years, the simulations are continued to 100,000 years with either fully backfilled or no backfill conditions in unchanged drifts. The backfill material uses hydrologic parameters that reflect crushed TSw tuff gravel data (Conca and Wright, 1996) and a thermal conductivity of 0.6 W/m-C.

D. New WP degradation simulations were performed based on new drift-scale thermohydrologic simulations. Since TSPA-95, two major improvements have been made to the waste-package degradation model, WAPDEG. The "corrosion time" concept is used to incorporate the dependency of the corrosion rate on the current corrosion depth. This is more accurate than the approach used in TSPA-95, in which the duration of corrosion was treated directly as a proxy for the corrosion depth. The "patches" approach has also been developed since TSPA-95 to give a more accurate representation of spatial variability on individual WPs. This approach gives a strong correlation between the general corrosion depths at neighboring locations on a specific WP. Both of these improvements are described in the milestone report Engineered Barrier System and Near Field Environment Performance Assessment FY-96 Activities: End-of-Year Summary (WBS 1.2.5.4.2 September 1996)[1]. The corrosion model used, however, still is based on the expert

[1]LLNL report to M&O, LLNL Number LLYMP9705013, May 7, 1997

judgement for Alloy 825 (see TSPA-95, CRWMS M&O 1995a). This should be updated, particularly for the new materials being considered.

E. The degree of galvanic protection has been slightly altered from TSPA-95, but is based upon the same type of physical model. In particular, a threshold is specified to determine the duration of galvanic protection. This threshold is a function of the mass of the outer barrier that has undergone general corrosion. Thus, for some of the analyses in this study (cases 5, 14, and 16—see below), it is assumed that 100 percent of the packages have galvanic protection that will protect the inner barrier until 75 percent of the outer barrier has undergone general corrosion. For most of the cases, however, including the base case, it is assumed that only 50 percent of the packages experience this degree of galvanic protection, while the remaining 50 percent of the packages have no galvanic protection. These choices were based on assumptions made in a previous study (CRWMS M&O 1996b). A few of the analyses (cases 4, 13, and 17) assume no galvanic protection for all packages. It is expected that the 75 percent threshold might be revised after additional laboratory testing results have been obtained.

F. The distribution coefficient (K_D) for Np has been modified to reflect the most recent information in the unsaturated-zone (UZ) site-scale transport model (Robinson et al., 1996). In particular, the expected ^{237}Np K_D is set equal to 2.5 cc/g in zeolitic units , 0 cc/g in nonzeolitic units, and 0 cc/g in the saturated zone, whereas in TSPA-95 the values were 0.5 cc/g, 1 cc/g, and 3 cc/g respectively. The K_D for other radionuclides are unchanged from those reported in TSPA-95 (CRWMS M&O 1995a).

G. The average saturated zone flux is set equal to 0.31 m/yr, as opposed to 2.0 m/yr in TSPA-95, based on updated estimates reported in CRWMS (M&O 1996b).

Table 3-3 is a compendium of the PA sensitivity analyses conducted for this study. The various important parameters listed in Tables 3-2 and 3-3 were varied discretely rather than probabilistically. Based on available information, the variations in these listed parameters are thought to bound the range of probable behavior of the given parameter. The performance results are presented as expected-value time histories for peak dose at the accessible environment (AE) over 100,000 years postclosure. Most of the calculations in this section were only taken to 100,000 years instead of 1,000,000 years. In particular, no peak-dose CCDFs (multiple realization) are presented in this study. Parameters that are modeled stochastically in the TSPA model, e.g., radionuclide solubilities, are assumed to take on the expected value of their probability distribution. The sensitivity analyses listed in Table 3-3 vary the parameters according to the following values:

<u>Thermal loading</u>:
- 20.5 kgU/m^2 (83 MTHM/acre)
- 6.2 kgU/m^2 (25 MTHM/acre)

<u>Backfill</u>:
- With crushed TSw2 backfill emplaced at 100 years
- Without backfill

Table 3-3 Sensitivity Analyses for Waste Isolation Systems Study

(Case 1 is the base case, bold font indicates changes from base case)

Case No.	Thermal Loading (MTHM/acre)	Backfill	Galvanic Protection[1] (%)	Cladding	Drips	Np K_D	SZ q_{sat} (m/yr)	Distance to AE	Time (years)
1	83	No	50/75	No	WP	Low	0.31	5km	100,000
2	**25**	No	50/75	No	WP	Low	0.31	5km	100,000
3	83	**Yes**	50/75	No	WP	Low	0.31	5km	100,000
4	83	No	**0**	No	WP	Low	0.31	5km	100,000
5	83	No	**100/75**	No	WP	Low	0.31	5km	100,000
6	83	No	50/75	**Yes**	WP	Low	0.31	5km	100,000
7	83	No	50/75	No	**WF**	Low	0.31	5km	100,000
8	83	No	50/75	No	WP	**High**	0.31	5km	100,000
9	83	No	50/75	No	WP	Low	**0.1**	5km	100,000
10	83	No	50/75	No	WP	Low	**1.0**	5km	100,000
11	83	No	50/75	No	WP	Low	0.31	**30km**	100,000
12	83	No	50/75	No	WP	Low	0.31	5km	**1,000,000**
13	83	No	**0**	No	**WF**	Low	0.31	5km	100,000
14	83	No	**100/75**	**Yes**	WP	Low	0.31	5km	100,000
15	83	**Yes**	50/75	No	**Shield**	Low	0.31	5km	100,000
16	83	**Yes**	**100/75**	**Yes**	**Shield**	Low	0.31	5km	100,000
17	83	**Yes**	**0**	No	WP	Low	0.31	5km	100,000
18	83	**Yes**	50/75	No	WP	Low	0.31	5km	**1,000,000**
19	83	No	**90/75**	No	WP	Low	0.31	5km	100,000
20	83	No	**99/75**	No	WP	Low	0.31	5km	100,000
21	83	No	**99.9/75**	No	WP	Low	0.31	5km	100,000

[1] The first number refers to the percentage of WPs that have 75 percent (the second number) galvanic protection. A 0 alone indicates no galvanic protection on any package.

Galvanic protection:

- 50/75: 50 percent of the packages have 75 percent galvanic protection of the inner barrier (75 percent of the outer barrier must be gone before the inner barrier starts to degrade) and the remaining 50 percent of the packages have 0 percent galvanic protection.

- 0: 0 percent galvanic protection of the inner barrier for all packages.

- 100/75: 75 percent galvanic protection of the inner barrier with 100 percent of the packages having this protection.

- Other: Cases 19, 20, and 21 had 90, 99, and 99.9 percent of the packages having 75 percent galvanic protection of the inner barrier.

Cladding:

- No credit taken for cladding

- Credit taken for cladding based on the model in Section 3.1 of the performance allocation study which considers only containment credit for rods that didn't unzip (CRWMS M&O 1996b).

In-drift radionuclide release model (see TSPA, CRWMS M&O 1995a):

- Drips on WP: Diffusion from WP and both advection and diffusion from EBS.
- Drips on Waste Form: Diffusion and advection from both WP and EBS.
- Drip Shield: Only diffusion from both WP and EBS.

Neptunium sorption in the unsaturated zone—The sensitivity of Np sorption on the dose at AE is evaluated by discretely varying the ^{237}Np adsorption coefficient (K_D) in the zeolite from 2.5 cc/g to 5 cc/g. The distribution of the zeolite is somewhat different from TSPA-95, and is based on the updated site-scale UZ flow model mentioned above. The average zeolite thickness between the repository and the water table is about 100 m, which is about 27 percent of the total distance from the repository to the water table.

Additionally, the solubility used for the radionuclides was based on the conservative case used in TSPA 95. The solubilities used were 34 gm/m^3 and 1.2 x 10^4 gm/m^3 for ^{237}Np and ^{99}Tc respectively (CRWMS M&O 1995a).

Saturated-zone flux—The sensitivity to saturated zone flux was evaluated as this will affect both the dilution and the travel time to the accessible environment. The extreme values of 0.1 m/yr and 1.0 m/yr for the SZ Darcy flux are thought to bound the range of possible fluxes based upon an equivalent porous medium model. In other words, it is assumed that there is full coupling between fractures and matrix in the SZ with respect to transport, i.e., equilibrium matrix diffusion. An expected value of 0.31 m/yr was used for the base case Darcy flux (CRWMS M&O, 1996b). Saturated-zone velocities are calculated using a porosity of 0.2. The calculations assumed that the radionuclides were distributed uniformly within a 50 m depth of the saturated zone and that there was no lateral dispersion for the doses at 5 km. For longer distances lateral dispersion was considered (CRWMS M&O 1995a). The no lateral dispersion is a conservative bound since lateral dispersion will occur.

Location of accessible environment—Most of the analyses in this study calculated dose based on a water well assumed to be screened over a 50 m interval of the SZ at 5 km down gradient from the repository. All the radionuclide mass in the saturated zone is assumed to be contained and well-mixed in this upper 50 m of the saturated zone. Dose was also based upon 2 L/day of drinking water. However, because of regulatory uncertainty over the definition of the accessible environment, a simulation for the base case (case 1, see below) was performed to determine the dose at 30 km from the repository footprint (Case 11). This calculation is based on an advection-dispersion dilution model discussed in Chapter 7 of TSPA-95.

Time scale—Most of the analyses present results over the first 100,000 years postclosure; however, two of the cases are simulated to 1,000,000 years.

The base case for these performance assessment analyses is Case 1. Cases 2 through 12 represent systematic variations of each of the parameters listed above, while cases 13 through 18 are variations of combinations of the parameters.

Base Case

The base case defined for this study is considered a conservative case and is for a thermal load of 20.5 kgU/m^2 (83 MTHM/acre), no backfill, an average galvanic protection with 50 percent of the WPs having 75 percent galvanic protection, no cladding, EBS conceptual transport model of drips on WP, ^{237}Np K_D=2.5 cc/gm in zeolites, saturated zone flux (q_{sat}) of 0.31 m/yr and the accessible environment is defined to be at 5 km (5 km from the respective columns center). The drinking water dose history at the accessible environment over 100,000 years for the base case is presented in Figure 3-2. The primary contributors to dose are ^{99}Tc, ^{237}Np and ^{129}I. The peak dose after 30,000 years is dominated by ^{237}Np which reaches a maximum value of 0.5 rem/yr. This is different than TSPA-95 analyses in which ^{237}Np did not dominate the peak dose until much later than 30,000 years. The primary cause for the difference is the higher UZ flux and, the lesser degree of Np retardation. (In TSPA-95, all UZ units were assumed to sorb Np.)

Comparison of Sensitivity Cases

The 100,000-year ^{99}Tc, ^{129}I, ^{237}Np, and total peak drinking water dose histories for the sensitivity analyses (cases 2 through 18) are shown in Figures 3-3 through 3-19. Comparisons of the total peak dose for these sensitivity cases to the base case total peak dose are presented in Figures 3-20 through 3-34. These figures give a quantitative indication of the importance of each of the sensitivity parameters relative to overall system performance.

The effect of thermal loading is shown in Figures 3-37 and 3-38. The decrease in thermal loading from 20.5 to 6.2 kgU/m^2 (83 MTHM/acre to 25 MTHM/acre) results in a significant decrease in the total dose value at 5 km. This is attributed to the decrease in temperature, which results in fewer and slower WP failures, and also in a decrease of the spent-fuel dissolution rate. A three-dimensional analysis was found to be needed for these cases (see Figure 3-38 comparing two- and three-dimensional predictions). This is discussed further in Section 3.3. The three-dimensional calculations reported in Section 3.3 will show about an order of magnitude decrease in dose over the base case is predicted. Some small differences were noted in TSPA-95 at the lower fluxes. For those calculations, the lower thermal load case of 6.2 kgU/m^2 was factors of about 2.5 and 1.5 lower doses at 10,000 and 1,000,000 years respectively than the higher thermal load case at 20.5 kgU/m^2 (CRWMS M&O 1995a). In these lower flux cases the dryout of the rock remains for a longer period and the performance differences between thermal loads are not as pronounced.

The effect of thermal backfill on the peak dose is presented in Figures 3-4 and 3-21 (Figure 3-28 shows the doses at one million years). Little effect is seen. This is because the relative humidity and temperature histories are not dramatically different between the backfill and no backfill cases and have a relatively minor effect on the waste-package degradation history. Three-dimensional calculations with backfill being done at the conclusion of this study (Blink, J. 1997 Private Communication) are giving some preliminary indications that some configurations may have improved relative humidity/temperature profiles with backfill than the case without.

The sensitivity analyses for variations in galvanic protection of the inner barrier are presented in Figures 3-5, 3-6, and 3-22. Figure 3-22 shows the comparison for total peak dose. "75 percent galvanic protection" means that for all the packages the inner barrier does not begin pitting until 75 percent of the outer barrier has been degraded, which results in longer life time and slower failure rate of the WPs. In comparison to the base case (for which 50 percent of the packages experience no galvanic protection and 50 percent experience the 75 percent galvanic protection) the case where 100 percent of the packages have 75 percent protection delays the initial release to the AE by 15,000 years compared to the base case and also reduces the dose by almost two orders of magnitude. In comparison to the case where 0 percent of the packages are galvanically protected, the base case with 50 percent of the packages protected does not help in delaying the early WP failures nor the initial dose arrival at the AE, but does reduce the peak dose by a factor of two. A summary plot showing the effect of varying the percentage of WPs that all have 75 percent galvanic protection is given in Figure 3-34. A backfill case without galvanic protection was also run (Figure 3-33) and shows similar trends.

Figures 3-7 and 3-23 show the dose at the AE for the cladding sensitivity analyses. The major contributors to the total dose are similar to the base case, and the total dose for the cladding credit case has decreased by more than an order of magnitude from the no cladding credit case. The fraction of the pins that fail by cladding unzipping was about 0.5 percent, although perforation of the cladding occurred in a number of the assemblies (CRWMS M&O 1996d). The release of ^{99}Tc and ^{129}I is significantly reduced due to the low number of pins unzipped. The effect of cladding is less apparent on the release of ^{237}Np because its dissolution is solubility limited, so it requires longer time to be removed from the WPs in the base case, when there is a large amount of ^{237}Np available for release, than in the cladding case when there is less available for release. Figure 3-23 shows that the performance of the cladding reduces the total peak dose by a factor of between six and 50 in the initial time frame of ten to twenty thousand years and gradually the effect changes to a reduction of dose by a factor of 25 at 100,000 years. The reduction in peak dose was roughly a factor of 8 although uncertainly exists in this number in part due to the large time differences where the peak occurs with and without cladding. Also the performance of cladding results in later release from the WPs as can be seen in the form of delayed release to the AE.

Figures 3-8 and 3-24 show the sensitivity of different conceptualizations of water movement and radionuclide transport in the repository drift. In the "drips on waste form" case the fractures are assumed to drip directly on to the waste form, whereas in the "drips on WP" case (base case) it is assumed that the radionuclides must first diffuse through the corrosion pits in the degraded package. There is little effect on the high solubility radionuclides like ^{99}Tc and ^{129}I, but the dose of low solubility ^{237}Np has increased by more than an order of magnitude, which causes the total dose in Figure 3-24 to increase correspondingly. ^{237}Np is affected more than ^{99}Tc and ^{129}I because its dissolution is solubility limited, so the greater flow of water in the "drips on waste form" model allows more ^{237}Np to be mobilized. This case results in about an order of magnitude increase in total peak dose at the AE. It should be noted that these water contact models do not considered the effects of near-field dry out.

The effect of increasing the distribution coefficient of ^{237}Np in the zeolite is presented in Figures 3-9 and 3-25. Compared to the base case (Figure 3-2) there is not much change in the dose at the accessible environment except for a slight decrease in the dose at later times. This is because most of the flow is in the fractures in the zeolitic units, and currently we assume no retardation in the fracture pathways. This is a significant change over earlier calculations where much of flow was in the matrix. It is important to reduce the uncertainty in how this barrier operates.

The sensitivity to saturated zone flux is presented in Figures 3-10, 3-11, and 3-26. The decrease in the saturated zone flux for the 0.1 m/yr case results in later arrival of the dose at the AE due to decreased liquid velocity, and an increase in the peak dose value due to reduced dilution. The increase in the SZ flux value to 1.0 m/yr results in earlier arrival at the AE because of faster liquid velocity and reduced peak dose at later times due to larger dilution. This linear dependence of peak dose on SZ Darcy flux and arrival time on SZ velocity illustrates the need to better characterize this parameter.

The sensitivity of extending the saturated zone length from 5 km to 30 km is presented in Figures 3-12 and 3-27. Increasing the saturated zone length results in a decrease of the peak dose by a factor of 25 compared to the base case due to the increased dispersion of the plume at the larger distance. This is based on the dilution model in Chapter 7 of TSPA-95 (an analytical advection-dispersion solution in 3-D, combined with a subbasin mixing model) (CRWMS M&O 1995a).

To evaluate the effect of lengthening the time period of analysis, the dose at the AE for a period of 1,000,000 years for the base case is presented in Figures 3-13 and 3-28. Compared to 100,000 years, the dose increases slightly but starts falling off around 600,000 years. The backfill case is also presented in Figure 3-28 and shows very little difference from the no backfill case. Preliminary indications from three-dimensional calculations indicate somewhat better temperature/relative humidity conditions may be realized than what the two-dimensional calculations show.

The combined effect of no galvanic protection and "drips on waste form" (also no cladding) is presented in Figures 3-14 and 3-29. This essentially represents the worst case release scenario from the near-field. Also shown is the case of "drips on waste form" and 50 percent packages with galvanic protection which only shows a factor of two improvement over zero percent WPs with galvanic protection. Thus, most of the difference from the base case appears to be due to the EBS release model, not galvanic protection, especially at later times.

The effect of cladding and galvanic protection on dose are shown in Figures 3-15 and 3-30. The case for 100/75 galvanic protection of all packages, drips on WP, and a high degree of intact cladding has the lowest dose at the AE. This case represents essentially the best case near-field release scenario (a drip shield improves the performance further). It results in two to three orders of magnitude lower total peak dose at the AE than the base case scenario. The galvanic protection leads to later WP failure, which also delays the cladding failure leading to lower doses.

The effect of adding the drip shield functionality on the total drinking water dose is shown in Figures 3-16 and 3-31 for the case with backfill, drips on WP, and average galvanic protection. An ideal drip shield which allows only diffusive releases from the package was used in the calculations. The peak dose for the case with the drip shield is nearly one and a half orders of magnitude lower than the non-drip shield case. The drip shield case represents only diffusive release from the EBS, resulting in significant reduction from the non-drip shield case which has some advective release. The drip shield is assumed to remain intact for 100,000 years. Earlier predictions of drip shield performance (CRWMS M&O 1996a) showed significant delays in releases which are not observed in these calculations. The reason for the difference appears to be due to the fact that in the previous calculations the invert saturation was only 0.1 percent which allowed only extremely small releases. In these calculations the diffusion releases were about two orders of magnitude larger because the invert saturation was considered to be 12 percent (after Conca and Wright 1996).

Additional EBS performance enhancements in the form of cladding and 100/75 galvanic protection serve to delay the release and reduce the dose even further from the previous drip shield case. Figures 3-17 and 3-32 present this case, wherein the peak dose is decreased five orders of magnitude from the backfill only case. This case produces a diffusive release from fewer WPs which fail at later times.

Figure 3-33 presents the 20.5 kgU/m^2 (83 MTHM/acre) backfill case without galvanic protection and with average galvanic protection. The peak dose for the case without galvanic protection is a factor of 2 higher than the average galvanic protection case, due to the more rapid failure of the WPs.

Comparisons of the different sensitivity cases were examined to identify the estimated performance of a particular barrier. For the most part these calculations dealt with engineered components and the performance of these components. For the various cases an Absolute Performance Factor (APF) can be calculated. Additional discussion of APF is given in Appendix A and the Performance Allocation Study (CRWMS M&O 1996b). The APFs were calculated calculated by dividing the estimated peak dose calculated for the base case by the peak dose calculated for the case with the particular barrier included. Thus, a case with an APF of 10 would imply that the particular barrier in question would reduce the peak dose at the accessible environment by a factor of 10. APF can vary as a function of time, particularly in the first 10 to 20 thousand years after emplacement. Plots of APF as a function of time were not included in the report but instead the summary tables report an APF when the dose curves are reasonably constant with time, usually about 100,000 years. The other parameter of interest considered in this study was the delay that a particular barrier provided. The delay was defined as the time that 10 percent of the peak dose occurred at the accessible environment. The base case itself will have a given delay based on the assumptions chosen and the way in which the natural barriers are modeled. The following describes the findings of the calculations according to the particular barrier and focuses predominately on the engineered components since the majority of the calculations focused on those components. Calculations to estimate the natural barrier performance are discussed below.

In summation of the above performance calculations, Table 3-4 provides estimates of the absolute performance factors using the expected dose calculations. In the table a time delay for 10 percent of the peak dose to reach the accessible environment and an APF for the peak dose are shown. The APF was calculated by taking the peak dose for the base case and dividing it by the peak dose for the case with a particular barrier. The base case is shown with its delay time and an APF that is defined as unity. In these calculations the expected value of dose in the drinking water for the base case was divided by the expected value of the corresponding case with the barrier in question included. As before, cladding and galvanic protection provide the most performance although only the 99/75 or greater galvanic protection (99 percent of the packages have galvanic protection where

75 percent of the outer barrier must degrade before the inner barrier starts to fail) provides any appreciable performance compared to the 50/75 case (50 percent of the packages have galvanic protection). At these higher fluxes a drip shield which lasts as long as the time in question also now produces appreciable performance. Finally, predictions at lower thermal loading of 6.2 kgU/m^2 indicate a significant improvement in performance over the higher thermal loading case of 20.5 kgU/m^2. In this latter case, predictions indicate that the moisture returns while the WPs are still hot which results in more corrosion than the lower thermal loading case where the packages stay much cooler.

Table 3-4 Absolute Performance Factors at the Higher Flux Cases

Engineered Barrier Subsystem	Time Delay[1] (years)	APF
Base Case	13k	1
Cladding	15k	8
90% of WPs with 75% galvanic	14k	5
99% of WPs with 75% galvanic	14k	25
99.9% of WPs with 75% galvanic	17k	70
100% of WPs with 75% galvanic	28k	70
Drip Shield	15k	30
Backfill	13k	~1
Repository Configuration (thermal load)[2]	14k	~10

[1]Delay time defined as time to reach 10% of peak dose
[2]Based on 3-D calculations (see Section 3.3)

Table 3-5 presents the total peak dose and concentrations for three scenarios for 10,000 years and 100,000 years. The first scenario is the drinking water scenario, with the exposed individual drinking two liters of water per day from the well as previously defined. Doses for two other scenarios, a subsistence farmer and a residence farmer, were also calculated. The subsistence farmer scenario includes an individual who drinks two liters per day of contaminated groundwater, uses the contaminated water for watering crops and livestock, and consumes only food stuffs that are grown on the farm. The resident farmer scenario includes an individual who drinks two liters per day of contaminated groundwater, uses the contaminated water for watering crops and livestock, and that 50 percent of the food consumed is farm grown. The biosphere dose conversion factors (BDCF) used for the drinking water scenario are presented in TSPA-95 (CRWMS M&O 1995a). The BDCFs for the two scenarios used are presented below for the three key radionuclides. These BDCFs are based on "best estimate," generic, non-site specific parameter values and, therefore, are meant only for illustrative and/or comparative purposes.

Table 3-5 Total Peak Dose

	10,000 Years				100,000 Years			
			All Pathways (mrem/yr)				All Pathways (mrem/yr)	
Distance from Yucca Mountain	Drinking Water[1] (pCi/lit)	Drinking Water (mrem/yr)	Subsistence Farmer[2]	Resident Farmer[3]	Drinking Water (pCi/lit)	Drinking Water (mrem/yr)	Subsistence Farmer	Resident Farmer
5 km	240	0.40	3.8	2.1	180×10^3	490	4200	2500
30 km	0.0	0.0	0.0	0.0	5000	20	130	80

1. Drinking water (2 L/day) based on ICRP-30 (whole body with weighting factors for organs); 40 CFR 141 is based on ICRP-2 (critical organ dose, more conservative)
2. Subsistence farmer drinks 2 L/day and consumes 100% locally grown food
3. Residence farmer drinks 2 L/day and consumes 50% locally grown food

	BDCF(mRem/year/pCi/liter)	
Radionuclide	Subsistence Farmer	Resident Farmer
^{129}I	3.1	1.6
^{99}Tc	0.0079	0.0048
^{237}Np	9.2	6.5

The results shown in Table 3-5 are relatively intuitive, in that the more food the exposed individual consumes from the contaminated source, the greater the dose. Another important result shown in the table is that at 10,000 years for the 30 km accessible environment location, the simulated doses are essentially negligible or zero.

Table 3-6 presents the total peak drinking water dose for the 100,000 year simulation for each of the 18 cases analyzed in this study. Many of the important results from this table have been discussed in the individual sensitivity analyses, but the factors which appear to produce significant reduction in the total peak dose from the base case are:

- Increased galvanic protection
- Drip shield
- Increased distance to AE
- Lower thermal load
- Performance credit for cladding (containment only)
- Higher SZ flux
- Higher Np K_D in zeolite.

Table 3-6 Total Peak Dose in mrem/yr for 100,000 year

Description	Peak Dose (mrem/yr)
Case 1 (base case)	490
Case 2 (low thermal load) (three-dimensional)	50
Case 3 (backfill)	590
Case 4 (no galvanic protection)	960
Case 5 (100% galvanic protection)	6.7
Case 6 (cladding credit)	65
Case 7 (drips on waste form)	12,000
Case 8 (high ^{237}Np K_D in zeolite)	330
Case 9 (q_{sat} = 0.1 m/yr)	1,500
Case 10 (q_{sat} = 1.0 m/yr)	150
Case 11 (AE = 30 km)	19
Case 12 (base case for 10^6 years)	760[1]
Case 13 (no galvanic protection & drips on waste form)	21,000
Case 14 (100% galvanic protection & cladding credit)	1.3
Case 15 (backfill & drip shield)	17
Case 16 (backfill & 100% GP & cladding & drip shield)	0.005
Case 17 (backfill & no galvanic protection)	1,200
Case 18 (backfill for 10^6 years)	910[1]
Case 19 90% of WP with GP	100
Case 20 99% of WP with GP	20
Case 21 99.9% of WP with GP	6.7

[1] Dose at 1,000,000 years

Also, combinations of these factors produce even lower doses. Several factors appear to produce higher total peak doses than the base case. These factors include:

- Alternative EBS release model with assumed drips directly on the waste form
- Lower SZ flux
- No credit for galvanic protection
- Longer analysis time frames (out to 1,000,000 years).

Also, combinations of these factors produce even higher doses.

Natural Barrier Performance based on Performance Allocation

The sensitivity studies above did not focus on the natural barriers and their performance. As such the study must rely to a large extent on previous work that was done. Studies, such as the TSPA conducted most recently in 1995 (CRWMS M&O 1995a), and the *Systems Study of Options for Characterizing the Calico Hills Nonwelded Hydrogeologic Unit at Yucca Mountain, Nevada* (CRWMS M&O 1995b), examined some aspects of performance for natural barriers. The most recent analysis, however, has been the work done in the *Description of Performance Allocation* study (CRWMS M&O 1996b). The majority of the discussion in this section is based on the work performed in that effort. These studies were all conducted at lower percolation fluxes than what are currently anticipated.

More information on these calculations and the natural barriers themselves is contained in the above reports and in Appendix B. Some of the modeling information and assumptions used are reproduced in this section.

The performance allocation study calculated relative and absolute performance factors (APFs) for each natural barrier considered. These factors are somewhat different than the APFs defined above since they were based on calculations of the concentration of radionuclide entering a particular barrier ("source term"), and the concentration of radionuclide which exits the barrier ("downstream mass released"). These concentrations are clearly time dependent and the calculations were carried out to 1,000,000 years. The calculations examined releases from the WP, transport through the invert beneath the WP, transport through the unsaturated zone beneath the WP, which included the CHn, transport through the CHn, and transport through the saturated zone.

The basis of the calculations was the parameter set used in TSPA 1995 (CRWMS M&O 1995a). The base case considered was for a thermal load of 20.5 kgU/m^2 (83 MTHM/acre) and an initial unsaturated zone percolation flux of 1.25 mm/yr. This initial percolation flux is modified according to a climate change model that randomly samples numbers within a period of 100,000 years (CRWMS M&O 1995a). However, the rock properties used in those earlier thermohydrologic calculations which provide estimates of WP lifetime, are those that are consistent with lower flux conditions. The transport calculations in that work were done with a single infiltration rate of 1.25 mm/yr. Recent information, as discussed above, indicates that the percolation flux is probably 1 to 10 mm/yr and under climate changes may be as much as 30 mm/yr.

The assumptions used in the Performance allocation study that are applicable to this effort are primarily those used in the TSPA 1995 work (CRWMS M&O 1995a) and additional details can be found in that reference. A summary of some of those assumptions follows:

- *Waste containers*—the waste containers are emplaced center-in-drift. These containers use the multipurpose container concept with a 100 mm thick corrosion-allowance material, such as mild steel, and a 20 mm thick corrosion-resistant material, such as Alloy 825.

- *Analytic models*—The two-dimensional FEHM code (Zyvoloski et al. 1995) using a smeared heat source was used for the near field calculations of the environment. In conjunction with the more detailed process models the total system performance is calculated using the RIP (Golder Associates 1994). Two dimensional, smeared heat source calculations can underpredict the temperature and overpredict the relative humidity at the WP.

- *Waste Stream*—The waste stream is oldest fuel first with an average age of 26 years and burnup of 39 GWd/MTHM for the PWR fuel.

- *Engineered Barriers*—The WP, backfill, cladding credit, and galvanic protection were considered in some of the cases.

- *Subsurface Design*—The subsurface design considered 5-m diameter emplacement drifts and 22.5-m spacing between drifts. Uniform spacing of identically loaded WPs was approximated by a smeared line source to produce the desired AML of 20.5 kgU/m^2.

- *Fracture Flow and Fracture-Matrix Interaction*— The calculations of seepage flux into the drifts was based on a simplified dual continuum model (refer to TSPA-95) which allowed water to drip on the WPs. The drift scale thermohydrologic model for calculating relative humidity and temperature in the drifts was based on an equivalent continuum model of fracture matrix interactions. The fracture-matrix transport in the far field unsaturated zone uses a dual continuum approach.

The calculations done in the performance allocation report were not done under quality affecting procedures; they are generally scoping calculations. The RIP code was developed and verified using ASME NQA-1 and ISO-9000 standards (Golder Associates 1995). The process models used in the analysis, however, have not been qualified. Those models were used for the purposes intended and over the range for which they were designed. Current uncertainties on unsaturated zone flow and transport are high although the testing program should reduce these uncertainties in the future.

The performance allocation effort calculated the total mass (or mass of a given radionuclide) released from the downstream end of a particular barrier. This was done for each of the barriers considered which were the WP, the engineered barrier system, the Topopah Springs welded unit beneath the repository, the entire unsaturated zone beneath the repository including the CHn and the Prow Pass, and the saturated zone. Although identified as barriers (they limit the amount of flux into the TSw), the performance allocation did not calculate the performance of the alluvium and the PTn.

The mass of radionuclide which exits one barrier is the source for the next barrier just downstream. Based on this, a time-dependent APF for a barrier can be established. This APF is defined (CRWMS M&O 1996b) as the ratio of the input to a particular barrier at any given time to its output. Unless decay produces a radionuclide in a given barrier, the APF is usually greater than or equal to one. Thus, an APF of 10 means that a given barrier reduces the accessible environment dose rate of a radionuclide by a factor of 10 at a particular time. It should be noted that at early time, the APF can indicate a large reduction in dose but at later times the APF can drop as the radionuclide traverses the barrier. An illustrative example of the performance allocation work is provided below in this section.

The estimated peak doses of ^{237}Np at times as large as 1,000,000 years were used to estimate the APF for the various barriers (Figure 2.1-15 and 2.1-17 of CRWMS M&O 1996b). APFs at earlier times were also calculated (see Appendix B) but these were on the steeply rising portion of the curve and are not representative of a steady state performance estimate. Thus only APFs for the one-million year peak dose are reported in Table 3-7. Additionally, a delay time, as was done above, could not be estimated from the existing calculations since the calculations were not run with and without the barrier. Thus, no delay times are reported for the natural barriers. Table 3-7 shows the estimates of these APFs for the three barriers calculated (the first barrier actually includes the second barrier). More discussion on natural barriers is provided in Appendix B.

Table 3-7 Natural Barriers Performance Factors

Natural Barrier Component	Absolute Performance Factor at 1M yrs[1]
Unsaturated zone transport[2]	30
CHn	12
Saturated zone transport	70

[1]Absolute performance stated in terms of the factor that the radionuclide doses exiting the barrier/layer are reduced from the doses entering the barrier
[2]Includes the CHn

Appreciable performance can be attributed to the various natural barriers. This performance is much larger in times earlier than one million years (see Appendix B) when the barrier functions to retard the radionuclide transport. At times of about one-million years, however, the performance is still substantial and these barriers are key components of the system. These predictions depend on the assumptions chosen in the calculations. Specifically, if the adsorption values of zeolites or assumed thicknesses of these minerals change then the performance of the unsaturated zone and the CHn can change. However, there does not appear to be a large sensitivity to the adsorption coefficient (see Figure 3-25) The above calculations assumed that the zeolites were primarily in the unsaturated zone. However, recent evidence indicates that a portion of the saturated zone is in fact zeolitic (see Section 3.5). This could potentially improve performance. Thus, zeolite distributions and adsorption coefficients need to be better known.

The calculations of the saturated zone performance also depend on the assumptions made. A 50 m mixing depth with a mean Darcy flux of 2.0 m/yr were assumed. More recent information indicates the Darcy flux may be more like 0.3 m/yr. The calculations done above show a sensitivity to the Darcy flux in that the lower flux produces less dilution although it delays the plume somewhat from reaching the accessible environment over the higher flux cases. Factors of 2 to 3 lower dose rate were observed as a function of Darcy flow as shown in Figure 3-26. Mixing depth was not varied. Additional information is needed to establish what the characteristics (mixing depth, dispersivity, and Darcy flux or flow velocity) are in the saturated zone.

There are some preliminary calculations that indicate changes may occur in the saturated zone as a result of the effects of heat. LLNL has done an assessment on the effects of hydrothermal flow in the saturated zone. They found that convection cells that have an extent of a few kilometers can develop, causing greater dilution, and flow velocities will increase as a result of the increased temperature in the saturated zone caused by the SNF decay heat. The magnitude of the buoyancy flow increases with increasing temperature (LLNL 1996). There have also been some preliminary scoping analyses of potential mineral and porosity changes in various rock units (unsaturated zone and saturated zone) due to the effects of heating. Mineral redistributions and porosity changes that could result in one to three orders, of magnitude changes in permeability were predicted to occur in the saturated zone. Even larger changes occurred in the unsaturated zone (personal communication from W. Glassley to S. Saterlie, November 1996). Based on this information it is important to understand the heating effects in the various rock units.

3.3 THREE-DIMENSIONAL CALCULATIONS

Two NUFT models were built for the three-dimensional portion of this study, a low (6.2 kgU/m^2 [25 MTHM/acre]) and a high (20.5 kgU/m^2 [83 MTHM/acre]) thermal loading layout. Each model represents a slender column taken out of the center region of a fully loaded repository. Due to use of symmetry assumptions the modeled columns are ½ drift spacing wide and 6 waste packages (with their appropriate in-drift spacings) deep, see Figure 3-35. The height and stratigraphy used in the model is taken from a centrally located column in the 3D site-scale LBL/USGS Yucca Mountain model (Bodvarsson, et al. 1996). Figure 3-36 shows the general layout of the three-dimensional models.

The 20.5 kgU/m^2 Advanced Conceptual Design (ACD) model closely follows the proposed repository drift dimensions and mass loadings described in the Advanced Conceptual Design report, 1996 (CRWMS M&O 1996e). The 6.2 kgU/m^2 design model represents the alternative low thermal loading case, which is basically an expanded ACD design layout. The different dimensions and general layouts of the two models are shown and described in Figure 3-35. Note that the spacing between packages is based on linear mass loading (LML) of the spent nuclear fuel packages only, no package spacing was credited for the defense high level waste packages due to their limited contribution to the thermal output of the repository.

Each three-dimensional model was executed with a no-backfill scenario. The empty drift spaces in the no-backfill scenario are modeled as air type elements and thermal radiation transfer connections are explicitly established between the waste packages out to the drift walls, floor and ceiling. Thermal radiation connections are also modeled between hotter drift floor elements out to the wall and ceiling elements for increased accuracy of the radiative heat transfer calculations.

The three-dimensional thermohydrologic calculations were run for the two thermal loading cases. The temperature and relative humidity profiles, as a function of time, were then input into the waste package degradation model (WAPDEG). Actually, the temperature and relative humidity profiles for three types of WPs were used; a hot design basis package (fuel age of 10 years), an average package (fuel age of 26 years), and a cooler DHLW package. It was not necessary to use all six packages since these three types bound the problem.

The results of the temperature/relative humidity and waste package degradation (pitting distributions) estimated showed that the two-dimensional and three-dimensional calculations were essentially the same for the 20.5 kgU/m^2 (83 MTHM/acre) case. In the case of the 6.2 kgU/m^2 (25 MTHM/acre) the calculations showed, as expected, that the two-dimensional case had underestimated the temperature for the large, hot WPs. The three-dimensional results giving revised WP degradations were used in RIP and the dose at the accessible environment was calculated. The results of these calculations showing total drinking water dose history and the doses for the three long-lived radionuclides are plotted in Figure 3-37. For comparison sake, the total dose histories for the two-dimensional runs at 20.5 and 6.2 kgU/m^2 are plotted in Figure 3-38 with the dose calculated for the three-dimensional run at 6.2 kgU/m^2. The low thermal load with the large WPs (three-dimensional simulation) does not reduce the dose as much as had been predicted in the two-dimensional simulation, however, the dose at the accessible environment for the low thermal load case is still about a factor of 10 lower than for the higher thermal load at 20.5 kgU/m^2.

3.4 INVERT OR BACKFILL ADDITIVES

LANL conducted a series of sorption experiments to assess the sorption potential of two materials (envirostone and apatite) that could be considered as additives to either an invert or backfill materials for the potential repository. The main intent of these experiments was to present a formal analysis of the ability of these two materials to retard the migration of Np and Tc from the repository to the accessible environment and to determine how much of these materials would be needed to obtain a significant reduction in dose for the two radionuclides. A series of experiments on apatite had been conducted in 1996 and those results were reported[2]. In this report[2] Apatite was found to have only small sorption capacity for Tc; however, apatite was identified as a promising sorber for Np. The intent of the limited set of experiments conducted in this effort was to establish a rough estimate of the quantity of apatite or envirostone required to provide significant absorption of the two radionuclides.

Envirostone is a manufactured gypsum-based cementitious material that has a polymer component present, originally intended for use as a solidifying agent for aqueous radioactive waste. Apatite is a naturally occurring calcium phosphate-based mineral that also has various combinations and amounts of fluorine, chlorine, hydroxyl, and carbonate.

The water that was used in the experiment was modified J-13 water. A J-13 water sample from the Nevada Test Site was chemically altered to reflect the type of water that could be encountered by the backfill material in the emplacement drifts. The basis for the alteration is a possible accumulation of salts by evaporation during the initial thermal pulse and/or water contact with the invert or backfill that may be saturated with calcite, silica, aluminosilicates and other phases. To address possible effects from this altered water, the J-13 water sample (Ogard and Kerrisk 1984) had the following chemicals added to it to recreate the possible water that may be found.

[2] Ines Triay and Stephen Thornton, "Sorption Of Radionuclides By apatite As A Backfill Material." Los Alamos National Laboratory, unpublished.

Constituent	J-13 Water mg/l	Modified Water mg/l
SiO_2	30	64
Ca	11.5	70
Mg	1.8	10
Na	45	70
K	5.3	15
Cl	6.4	100
F	2.1	3
SO_4	18	120
HCO_3	—	127
NO_3	10	10

Two radioactive tracers, ^{95}Tc and ^{237}Np were used in this experiment. ^{95}Tc was used because it does not have the very long life that ^{99}Tc has. These isotopes were initially introduced into the J-13 water with radioactive counts of 14000 counts per gm per minute for ^{237}Np and 8000 counts per gm per minute for ^{95}Tc. The tracers were in an aqueous state when they were introduced into the water. Each tracer was put into its own solution and were not combined together.

The general form of the experiment (the first series of tests) was a batch sorption format with envirostone and the apatite mixed in various ratios of solids. The solids were then combined in sample tubes mixed with the tracer tainted J-13 water and sealed. The sealed sample tubes were then placed on a laboratory shaker for a predetermined time period of one, two, three, or four weeks. During the given time period, the envirostone and the apatite should absorb some of the radioactive tracer that was present in the water solution. At the end of the time period the sample tube was weighed, its pH recorded, centrifuged, and then had a measured amount of the J-13 tracer solution placed into a radiation counting vial to see what amount of radioactive tracer remained in the tracer solution. The remaining amount of tracer was then used to calculate the amount of tracer that had been absorbed by the apatite and the envirostone.

A second and brief series of tests were conducted to measure the maximum sorption potential of the two sample materials. These tests were conducted over a number of periods of time until the material would no longer absorb any tracer from the solution the sample was exposed to. For each time period, the 1 gram of sample material was exposed to 20 ml of new tracer solution. At the end of the time period, the old tracer solution was pulled off the sample. The sample was measured for the remaining tracer content and then a new tracer source was introduced to the sample material to start a new sorption time period.

This experimental work was conducted in accordance with the following Los Alamos National Laboratory procedures: LANL-CST-DP-86 R2, LANL-INC-DP-35 R3, LANL-CST-NBK-95-015 p.E2-E21, and LANL-INC-DP-79 R2. The experimental data for this experiment has been recorded and will be maintained in accordance with YMP Procedures. The residence for the experimental data is in the LANL CST-7 files in binder LA-CST-NBK-97-003, section 146. The material data for the apatite and the envirostone is located in the CST-7 files in LA-CST-NBK-97-003, pages AA2 and AA4. The data for the tracer stock preparation is located in the CST-7 files in binder LA-CST-NBK-95-027.

The results of the first series of experiments demonstrated the following results.

The sorption ability of apatite for ^{237}Np and its lack of sorption ability for ^{95}Tc that had been seen in the previous report was confirmed. For envirostone, this experiment showed that it had sorption potential for ^{95}Tc. This potential is believed to be better than any other material tested to date. The potential can be further characterized as increasing with time and increasing in a ramp manner consistent with the increasing percentage of envirostone in the sample mix (Figure 3-39). Also of note, the pH of envirostone in the tracer solution altered the pH of the J-13 solution to be close to that of envirostone. This alteration could raise the solubility limits of the J-13 solution for radionuclides that may come in contact with the envirostone by means of J-13 water.

The Tc data and the Np data with apatite for the second series of tests are shown in Figure 3-40. Envirostone, in the second test as in the first, showed little capability for Np sorption; as such, envirostone's sorption potential can be considered negligible. Envirostone initially showed good sorption capabilities for Tc in the early periods. Its sorption rate however dropped off quickly. At the conclusion of the envirostone sorption process, the data reflects that after an initial sorption capability of 12 to 13 percent of the radionuclides from the solution, in the end the overall sorption of Tc was less than 4 percent of the total amount of Tc that the envirostone was exposed to. Apatite's total sorption ability for Np could not be determined at this time. While apatite showed a marked sorption potential, the maximum potential is difficult to model since both the peak value and determination of reversibility of the sorption were never achieved.

In attempting to answer the question of "how much" of these two materials is needed for use in the repository, only a rough estimate can be made at this time due to the very limited duration of the experiment and the significant number of variables. Uncertainties regarding whether or not the sorptivity reached an asymptote and the amount of radionuclide that will exist lead to only rough estimates of the amount of material necessary to sorb all of a specific radionuclide. For ^{99}Tc estimates of the maximum amount of radionuclide that might exist for 70,000 MTHM of SNF was determined from the decay products in the Characteristics Data Base to be about 9×10^5 Ci (13.1 Ci/MTHM times 70,000 MTHM; average for PWR and BWR). The envirostone (density of 1272 kg/m^3), as indicated above, was not as efficient at absorbing Tc as apatite is at absorbing Np. As such, the estimates found that to absorb all of the ^{99}Tc would require as much as about 1.6×10^7 metric tonnes of envirostone. This amount would need to be spread evenly throughout the emplacement drifts or surrounding each WP. This would amount to about 1100 m^3 or a layer about 130 m deep under each WP to absorb all the Tc, which is not feasible. It does not appear practical to pursue this further.

Apatite (density of 1953 kg/m^3) is fairly efficient at absorption of ^{237}Np. The amount of Np produced is not as easy to estimate as Tc since it is produced as a decay product over a relatively long period of time. However, using a conservative estimate it was estimated that there will be about 8×10^4 Ci of ^{237}Np (1.2 Ci/MTHM times 70,000 MTHM; average for PWR and BWR). As such to absorb all of this Np would require approximately 1.1×10^5 metric tonnes of apatite spread throughout the emplacement drifts. This amounts to about 5 m^3 per each WP which would imply a layer of apatite about 0.6 m deep; this is possible. In this case the apatite was spread under each WP instead of throughout the entire emplacement drift.

Apatite, and possibly envirostone, have some affinity for sorbing other radionuclides in addition to Np. It may be possible that sorption of these radionuclides could diminish the mineral's ability to sorb Np. This needs to be investigated in the laboratory.

Work will continue to determine what the maximum amount of sorption is and the level of reversibility for both apatite and envirostone through continuation of the sorption experiment. If any changes in sorption are found those results will be provided by the QAP-3-12 process to update the report if necessary. Sensitivity studies might indicate that adequate performance may be achieved even if 100 percent of the radionuclide is not absorbed which would mean that less apatite or envirostone could be used. PA should do some of these sensitivity studies to estimate the amounts of material which would provide significant reduction in dose and Subsurface Design should evaluate how to get the additive in the inverts and the cost.

3.5 ZEOLITE CONCEPTUALIZATION

Zeolitized layers beneath the potential repository have long been considered important to waste isolation. Previous analyses (CRWMS M&O 1995a, 1995b, 1996b, and 1996d) have shown potential for appreciable adsorption of some radionuclides by zeolites. As indicated above, the zeolite can provide significant improvement in performance, particularly in the 10,000 year time period. However, significant uncertainties existed in the distribution (sorption) coefficients, K_D's, and the distributions and locations of these minerals beneath the potential repository and their response to heat. These uncertainties need to be reduced to be able to take performance credit for this barrier in LA and it was the intent of this study to document the work being done at reducing the uncertainties.

Preliminary information on the location of the top of the zeolite layer was reported in a previous study (CRWMS M&O 1996d) but that report identified the need to better establish the types and concentrations of zeolite present, the depths at which those zeolite exist, and the continuity of these zeolite-bearing rocks across the repository footprint. As a result of this current study, emphasis has been placed on updating the zeolite conceptualization and including this conceptualization in the RIB (YMP 1995) for ultimate use in future PAs. This subsection provides a brief overview of the status of that work.

The first steps have been taken in the process of developing an updated zeolite conceptualization for use in PAs. Using ongoing work to analyze borehole cores, information on zeolite distributions was obtained. The mineral distributions as a function of depth (for various stratigraphic layers) have been obtained for about 30 boreholes located throughout and around the primary area considered for emplacement. The results of these analyses are primarily described in a milestone report by Los Alamos National Laboratory (Chipera et al. 1997). The core samples were analyzed using quantitative X-ray powder diffraction to obtain information on the mineralogic compositions. The zeolites that are being identified are clinoptilolite, heulandite, mordenite, chabazite, erionite, stellerite, tridymite, and analcime. The zeolite distributions are being incorporated in the RIB (YMP 1995) as a result of this work. Sandia National Laboratories logging studies should yield additional information.

Based on these analyses, clinoptilolite and mordenite are the most abundant zeolite at Yucca Mountain. Until this study, chabazite was considered rare but some concentrations were found in USW SD-7 indicating that the southern end of the primary area may contain concentrations of this mineral. Analcime occurs as a prograde alteration product at greater depths than the other zeolite. Questions have been raised as to whether there are regions in the primary area where "holes" in the zeolite distributions might exist. Such a concern is not supported by the available data and zeolite have been found in all boreholes analyzed to date (Chipera et al. 1997).

The analysis of the borehole data and the identification of the zeolite minerals present is only the first step in the process of developing an updated model for zeolite. The next step, which has been essentially completed by Los Alamos National Laboratory (LANL), is to use interpolation methods between boreholes to develop a three-dimensional model of the mineral distribution. Some plots of these data are shown in Figures 3-41 through 3-43. Figure 3-41 shows the thickness (in meters) of zeolite beneath the potential repository. These plots were generated from the LANL information by taking the percent zeolite observed for each layer depth, multiplying by the layer depth, and then summing between the upper and lower cutoff elevations; thus, the values plotted are equivalent thickness of pure zeolite. Figure 3-41 shows the depths of zeolites from the potential repository horizon to the water table. As indicated above there appear to be no "holes" in the zeolite thicknesses. The largest thicknesses are in the north end of the repository area while the thinner amounts are in the southeast. However, there appears to be at least 60 to 70 meters of zeolite thickness even in those areas. Examining 50 additional meters into the saturated zone, as shown in Figure 3-42 shows that zeolites do exist in the saturated zone with approximately an additional 20 meters depths of zeolite in the north and about 10 to 20 meters depth in the south. Figure 3-43 shows that in the 170 m beneath the repository there are some zeolites, approximately 10 to 20 meters equivalent depth of zeolite. This region would have temperatures above 90°C which could potentially result in these zeolites being altered (CRWMS M&O 1996d). However, those zeolites in the 170 m just beneath the repository horizon are located in the areas that have the most zeolites. In the south portion of the repository block there is negligible difference in zeolite depths between the case where the integration extends to the repository (Figure 3-41) and where it only goes to within 170 m of the repository (Figure 3-43). These conceptualizations are different than what was used in TSPA and even the performance assessment calculations done above. Effort is needed to include these updated conceptualizations in the RIB and include the revised zeolite depths in TSPA 1998.

The zeolite conceptualizations need to be included in the PA models. It is recommended that the Project ensure that the information gets added to the RIB and that the zeolite conceptualizations be incorporated in the next TSPA (TSPA-VA).

3.6 SUMMARY OF PERFORMANCE CALCULATION RESULTS

A variety of design options have been considered for enhancing the EBS performance. An overview of the various options considered is presented in Figure 3-44.

This section summarizes the performance calculations that were done in this study (Section 3) and some of the other work which was documented in the appendices. The results for the various barriers are arranged starting inside the WP and then moving outward through the engineered barriers and then through the natural barriers. Those engineered barriers identified in Figure 3-44 as potential options and for which performance predictions were done are discussed below. If performance predictions were not done this is also noted.

Cladding—Cladding (an option considered as a design option in Figure 3-44) was predicted to provide a significant amount of performance with a factor of about 5 to 50 reduction in dose at the accessible environment depending on time. Cladding primarily only results in a reduction in dose and produces only minimal increase in delay of releases (see Table 3-4). Existing measurements and modeling discussed in the references cited in (CRWMS M&O 1996o) show that zircaloy is very durable and is relatively insensitive to aqueous corrosion over a range of about 2 to 12 pH. Zircaloy cladding, however, was found to be somewhat sensitive to temperature. If the cladding of the temperatures are maintained below 350°C, only about 6 percent of cladding fails from small punctures and a much smaller fraction (about 0.5 percent) of cladding was predicted to undergo a gross rupture (or unzipping). The remainder of the cladding is assumed intact for the entire duration of the calculation. Performance credit, however, could take two forms; containment is the current approach but partial protection of a portion of the SNF exposed may be possible if the fuel rods become broken through static loads. In the latter case the cladding would still provide some protection in the event the fuel rods become broken. An evaluation should be done to determined how much protection this may provide. Tests are planned to be conducted at Argonne on exposure of broken fuel rods to water and measurement of the subsequent dissolution of the SNF.

WPs and Galvanic Protection—Calculations were not specifically done for WP performance since it was an integral part of all the calculations. Nor did this study do an evaluation of different corrosion allowance and corrosion resistant materials. Some performance allocation estimates of the WP are provided in Appendix A. WP sizes were also not examined. These will be done in a Waste Package Size Study which will be conducted during the last half of fiscal year 1997.

Galvanic protection (an MGDS option) is the protection afforded a more noble metal or alloy by the corrosion of a less noble metal or alloy in contact with the same corrosive electrolyte. The performance predictions found that galvanic protection could produce significant performance if a significant number of WPs experience galvanic protection. The base case which had 50 percent of the packages with 75 percent galvanic protection (75 percent of the outer barrier must degrade before the inner barrier starts to degrade) had a factor of about 2 decrease in dose over the case where none of the packages had galvanic protection. Factors of about 5 to 70 occur if the percentage of WPs with galvanic protection increase to between 90 and 100 percent respectively. Significant delays in releases to the accessible environment of as much as 30,000 years could result if 100 percent of the WPs have galvanic protection. Some short term tests are started at LLNL and longer term tests are planned. These need to be conducted to establish the degree of galvanic protection that can be expected and the process models should be updated.

Galvanic protection is a method to increase containment lifetime but other methods may also increase this lifetime. Potential new materials for the WP, some of which have been identified, as well as ceramic coatings could possibly be considered but were not evaluated in this report. Waste package size was not evaluated nor was emplacement options. Additionally, a drip shield could increase lifetime and this is discussed below.

Drip Shield—Using drip shields over the WPs was a potential method considered to keep advective flow from the WPs. The performance predictions found that a drip shield which survived for a long time (the regulatory period or longer) could provide significant performance. Reductions in dose of about a factor of 30 over the base case with no drip shield were estimated. Essentially only minimal increase in delay in dose release time over the base case was predicted. A drip shield will reduce doses during it's lifetime but when it is gone doses return to levels approaching the base case with no drip shield. Long term reduction in dose from the base case at times after the drip shield is gone will required drip shield lifetimes well in excess of 20,000 years. It is unlikely that any man-made materials can be shown to have these very long lifetimes (M. Balady 1997 Personal Communication). Materials such as titanium or ceramic (see Appendix A) might be potential candidates for drip shields. Some tests of material candidates are planned at LLNL and should be conducted to provide estimates of the corrosion of these materials. Durability of the drip shields is a question since rock fall, particularly with ceramics, could damage the drip shield. There are also questions raised about the performance of the drip shield that is in too close a contact with the WP since volume changes in the WP as it corrodes may occur and hydrogen produced in WP corrosion and breaching could degrade titanium. Methods of protecting the drip shield range from putting backfill over the shield to installing the shield as a third barrier to the WP. These various methods of employment were not examined in this study and it is recommended that both Subsurface and Waste Package Design should examine these options, evaluate the operability, and determine the costs.

Backfill—Only a thermal backfill was evaluated in this study. The Richards Barrier or capillary barrier backfill was considered in an earlier study and was summarized in this work. The performance calculations indicate that, for these potentially higher flux conditions, the thermal backfill does not appear to provide any performance advantages. Calculations underway for another effort are indicating the potential for some temperature/relative humidity improved performance based on three-dimensional calculations. These calculations also did not consider whether or not there was an air gap at the top of the backfill which is likely to occur. The previous study concluded that a Richards Barrier would tend to act similar to a drip shield and would likely provide appreciable performance. However, the Richards Barrier was considered by Subsurface Design to not be emplaceable and this opinion has not changed since that time. Although a single layer backfill does not appear to provide any significant performance it should not be precluded at this time. This is due to the fact that it may be needed to provide protection for a drip shield if a decision is made to use such a drip shield. The resolution of whether backfill will be needed or not will need to wait until a decision is made on a drip shield, what material the drip shield is made of, and how the drip shield is installed. Three-dimensional calculations should be done in the Design Basis Modeling Study.

Pedestals or Inverts—Although the study was not able to evaluate the performance of having either a pedestal or invert in the emplacement drift, the study did sponsor an effort to evaluate additives to the invert. Specifically, LANL conducted laboratory tests to evaluate the ability of apatite, a phosphate-based mineral, and envirostone, a gypsum-based cementitious material, to sorb radionuclides. The tests determined that apatite has significant capability to sorb Np but has little affinity for Tc. Envirostone, however, can have appreciable capability to sorb Tc. As a result of these tests, some preliminary estimates were made as to how much apatite and envirostone would be needed to sorb 100 percent of the Np and Tc respectively. Conservative estimates indicate that a layer of apatite about 0.6 m in depth would be needed beneath the WPs to sorb 100 percent of the Np. This implies that 1.1×10^5 metric tonnes of apatite or a total volume of about 5.5×10^4 m^3 would be needed. For envirostone, a layer about 130 m deep would be needed to sorb 100 percent of the Tc which is clearly not achievable. The amount of envirostone needed to absorb all of the Tc would amount to about 1.6×10^7 metric tonnes or a total volume of about 1.3×10^7 m^3. However, it may be that appreciable performance could be achieved with less than 100 percent of the Np and Tc being sorbed. Sensitivity studies should be done to determine how much of these two minerals should be used in inverts. Subsurface Design should evaluate the feasibility of using these materials in inverts and what the costs would be for the treated inverts.

Tunnel Liners—This study did not perform any performance assessments on tunnel liners in the emplacement drifts, nor did it consider any potential liner joints or location of those joints. A preliminary evaluation documented in a status report (see Appendix A) raised questions about the potential for degradation of the ability of the natural barriers to sorb radionuclides due to the impact of the alkaline plume produced by advective flow through cementitious tunnel liners. However, more alkaline conditions may reduce disillusion of the radionuclides from the SNF. Questions continue to be raised concerning the durability of concrete under the thermal environment that will be present. These questions need to be addressed by PA and Subsurface Design and the impacts evaluated prior to including a concrete tunnel liner in the design for licensing.

Repository Configuration—Variations in Area Mass Loading and/or line loading (moving the WPs closer) together may produce improvements in performance. Moving the WPs closer together was evaluated in an earlier study (CRWMS M&O 1996d) and the results are summarized in Appendix A. Some modest performance gains (at most a factor of 2) were noted but the near field was substantially hotter which had operational concerns. This concept should be evaluated further in the Design Basis Modeling work which was recently initiated.

Evaluations of a low thermal load were done in the study. The predictions indicated that at the higher flux conditions which may potentially exist, improved performance (about an order of magnitude reduction in dose) was predicted. The increase in performance appears to result due to the fact that there appears to be somewhat lower WP corrosion at the lower thermal load and the waste is spread over a larger area. The reason for the lower WP corrosion is that for the higher fluxes, the repository rewets faster than earlier predictions such as TSPA 1995 and at the higher thermal load the WPs are hotter than in the low thermal load and hence are in a more aggressive regime for longer periods of time. Further evaluations should be done of a low thermal load option in TSPA 1998 and this alternative should be carried as an option for licensing. Subsurface Design should examine the designs for a low thermal load repository as part of developing alternatives for licensing.

It was beyond the scope of the study to evaluate in-drift versus vertical borehole emplacement. WP sequencing and lag storage are other thermal management tools which could be employed and these were considered to some extent in another earlier effort (CRWMS M&O 1996d). Additionally, for most calculations the layout of the emplacement drift and the slope of that drift were not able to be modeled. Evaluation of these options could possibly be done in the Design Basis Modeling effort that is in progress.

Alluvium/Colluvium—The presence of the alluvium/colluvium was not modeled explicitly. However, it is implicitly modeled because estimates of liquid infiltration into the Tiva Canyon used in the site scale UZ flow model, are based on alluvium cover (Flint and Flint 1996). There is recent evidence (Flint and Flint 1996) that the alluvium has a relatively large storage capacity to retain moisture, which generally allows removal of this moisture by persistent evapotranspiration instead of allowing transport downward into the mountain. However, the alluvium does not uniformly cover the mountain. Some additional discussion is provided in Appendix B.

PTn—The PTn was also not modeled explicitly in this effort but its properties form the basis for the fluxes and velocities derived from the UZ hydrology model. Some preliminary thermomechanical estimates indicate that the fractures in the PTn may increase in aperture by about a factor of two because of the heating (Ho, et. al. 1996). The thermal effects on this layer should be evaluated and the impact on performance examined by Performance Assessment.

Unsaturated Zone Transport—The calculations of unsaturated zone transport that were done in the Performance Allocation Study (see Appendix B) include transport in the CHn. The predictions, although done at a moisture flux of between 0.3 and 2 mm/yr which is somewhat lower than currently anticipated, indicates that the performance is significant. The estimated reduction in dose is about a factor of 30 as a result of transport through the unsaturated zone. Although possibly affecting performance (favorably or unfavorably), the alteration of the fractures as a result of deposition of minerals in the hot refluxing zone was not considered. Certain assumptions were made in the calculations regarding the portion of flow that goes through fractures and the portion that goes through the matrix. A better understanding of this flow and the thermochemical interactions is needed. Additionally, the key radionuclides which were shown to primarily affect the dose are Np and Tc. The solubilities of these radionuclides is uncertain and emerging evidence is indicating that Np solubility could be potentially lower than currently used. The solubilities of these radionuclides needs to be better determined.

CHn—The calculations in the Performance Allocation Study estimated the performance of the CHn separately as a subelement of UZ transport above. The unit was found to have appreciable performance with a factor of 12 reduction in dose. Some of the modeling assumptions have been updated since those calculations were done. Specifically, the dispersion coefficients for the zeolitized regions are now believed to be 2.5 cc/g compared to the 0.5 cc/g used in the Performance Allocation Study. These revised coefficients are documented in the input files which are saved and have been provided via QAP-3-12[3]. Calculations done for this work found that there was some sensitivity to changes in this parameter (see Figure 3-25). The model calculations estimated the zeolite depth as 100 m, however, Section 3.5 provided measurements which showed that the zeolite thickness, above the water table, varied from about 60 to 70 m to over 150 to 160 m. The three-dimensional nature of the zeolites needs to be considered in the next TSPA 1998 using an updated conceptualization. Additionally, the analysis reported in Section 3.5 indicate that there are zeolite concentrations in the saturated zone. These zeolite concentrations are likely to persist to the accessible environment although whether or not they continue much farther to the south beyond that is unknown. These should be considered in the transport model for the saturated zone.

---

[3]Design Input Transmittal (QAP 3-12) from R. Andrews to S. Saterlie, March 31, 1997)

The 1996 thermal study (CRWMS M&O 1996d) determined that the heat produced as a result of the spent nuclear fuel decay can alter the zeolitized layer. Specifically, dehydration and mineral changes are possible. For a thermal loading of less than about 85 MTU/acre mineral changes are unlikely in that the zeolite layer below 170 m under the repository horizon does not exceed 90°C. However, significant dehydration can occur which will produce significant amounts of water in the system. The effects of heat on this layer needs to be better understood.

Saturated Zone Transport—Transport in the saturated zone was also shown to have a significant performance with the saturated zone responsible for reducing doses at the accessible environment by about a factor of 70. This performance is based on the assumptions that were used such as a mixing depth of 50 m. Calculations done in this study found that the flow velocity can change the dose observed at the accessible environment by possibly as much as a factor of ten. Thus, it is important to better understand the mixing depth, flow velocity, and dispersivity in the saturated zone. Tests are underway in the C-wells and the results of these tests should be incorporated into the process models. However, it will take more information over a longer time than those tests to be able to establish the mixing depth and flow velocity. Those tests should be conducted.

Preliminary estimates (personal communication from W. Glassley to S. Saterlie, November 1996) indicate that the temperature of the saturated zone will increase significantly for the thermal load being considered now. This increase in temperature will alter the mineralogy, porosity, and flow velocity in the saturated zone. The effects of heat on the saturated zone need to be further determined and factored into PA calculations.

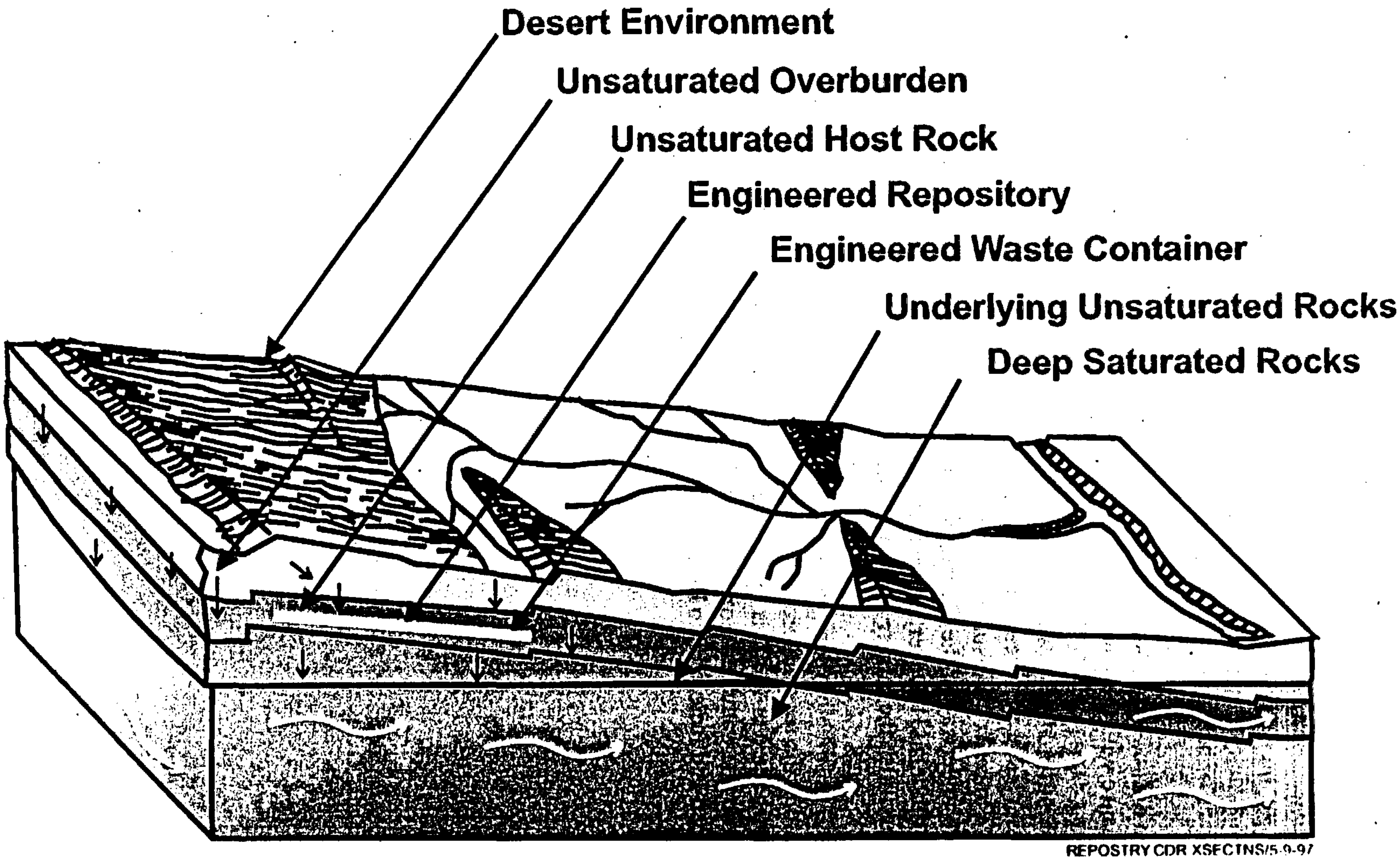


Figure 3-1 Geologic Cross-section

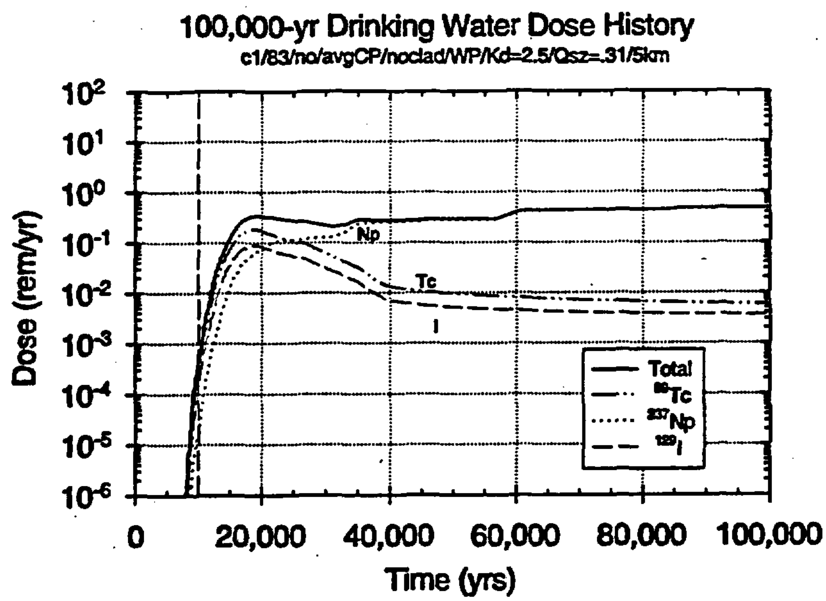


Figure 3-2 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste package (case 1).

DELETED

Figure 3-3 DELETED

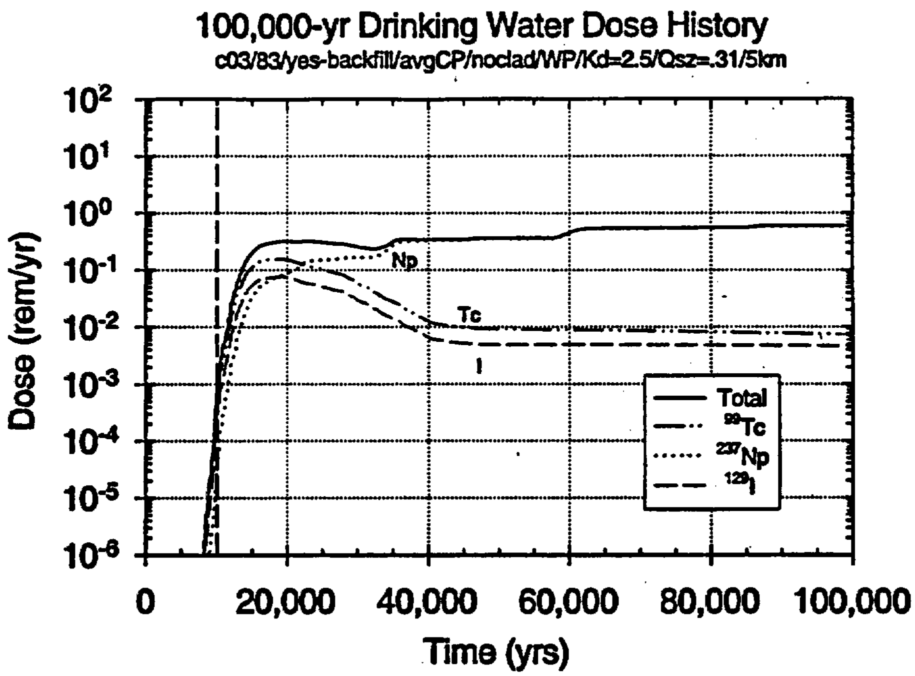


Figure 3-4. Expected-value dose history: 100,000 years, 83 MTHM/acre, backfill, average galvanic protection, drips on waste package (case 3).

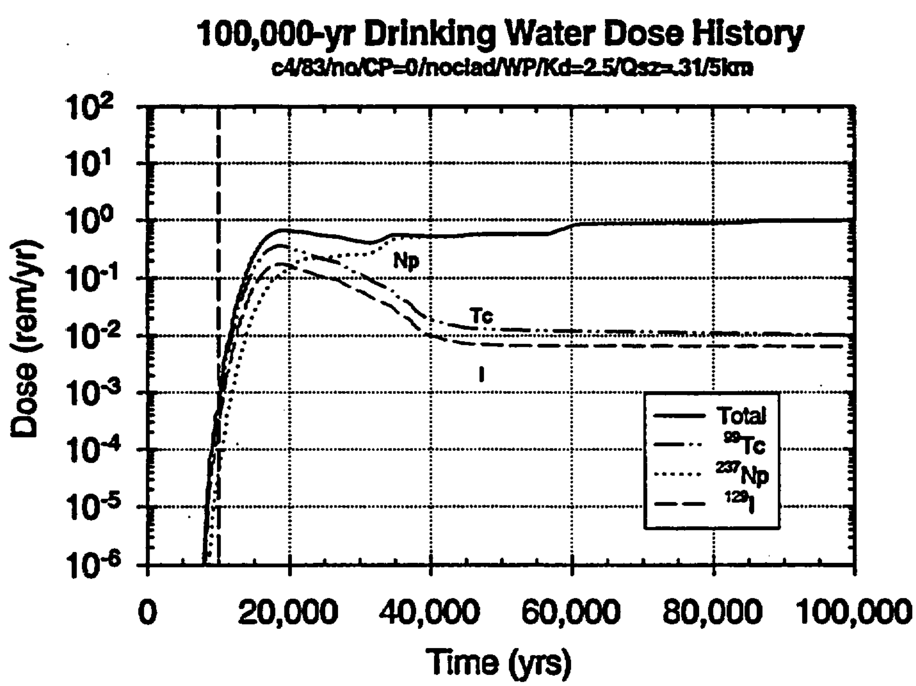


Figure 3-5. Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, no galvanic protection, drips on waste package (case 4).

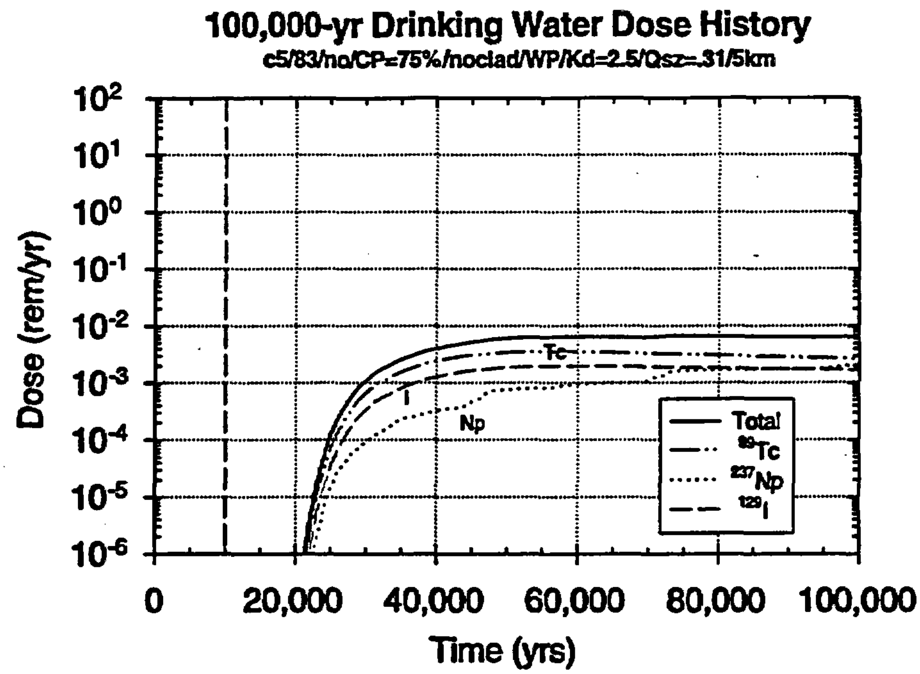


Figure 3-6 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, 100% galvanic protection, drips on waste package (case 5).

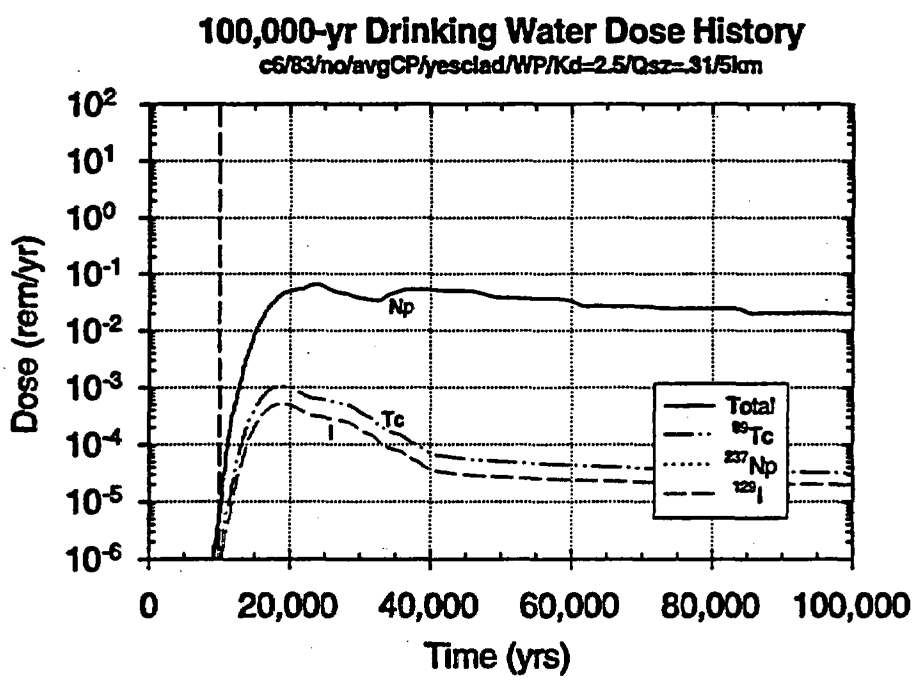


Figure 3-7 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, cladding, drips on waste package (case 6).

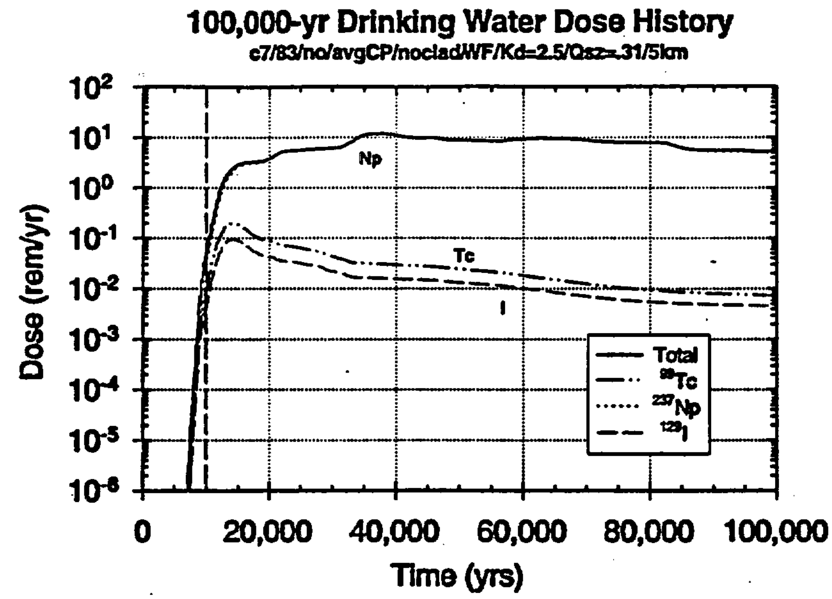


Figure 3-8 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste form (case 7).

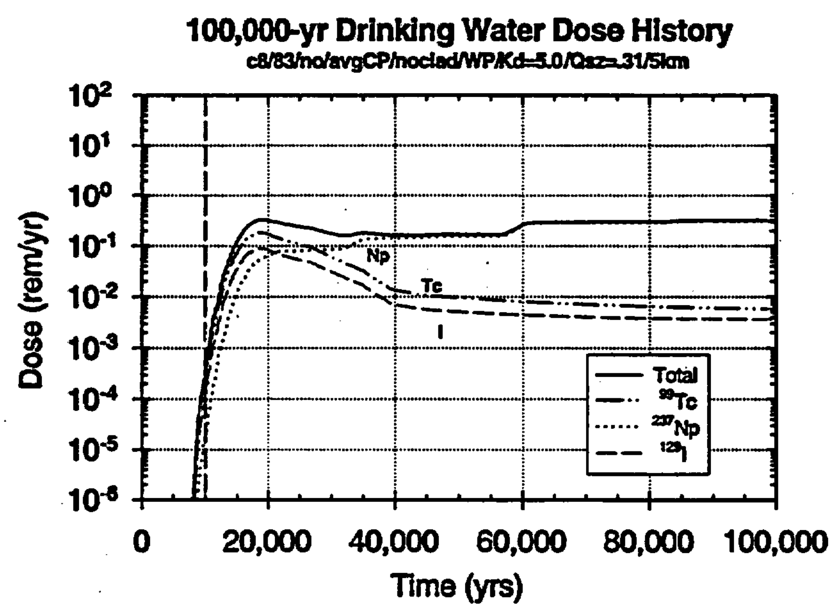


Figure 3-9 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste package, high Kd for Np-237 in zeolites (case 8).

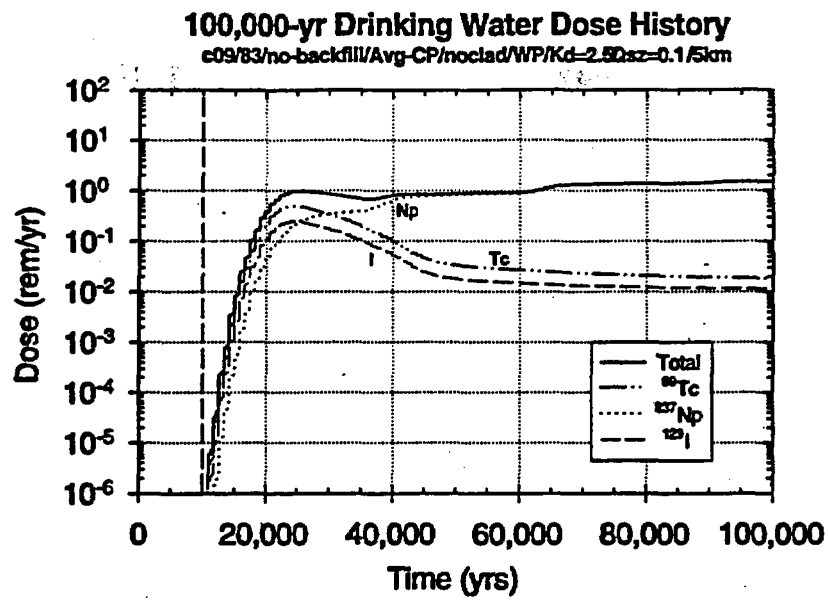


Figure 3-10 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste package, saturated zone flux = 0.1 m/yr (case 9).

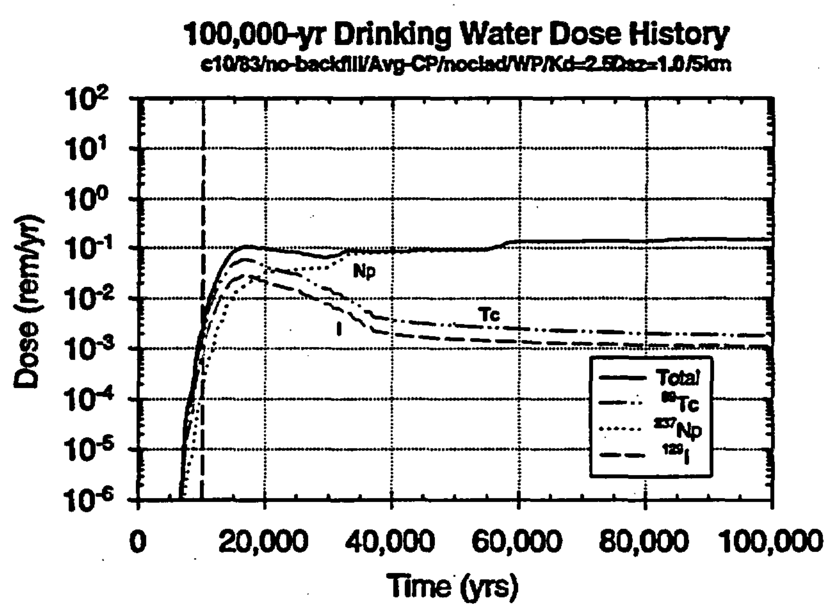


Figure 3-11 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste package, saturate zone flux = 1.0 m/yr (case 10).

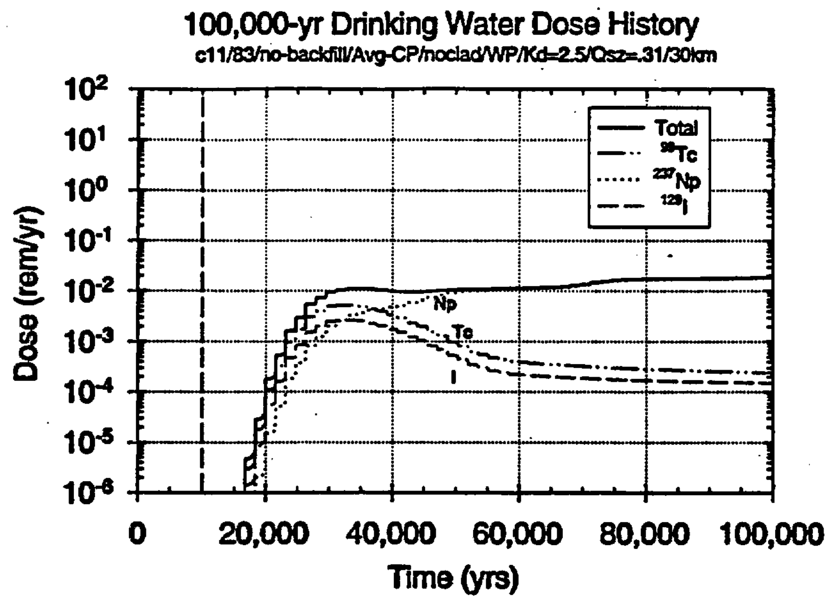


Figure 3-12 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste package, distance from the repository = 30km (case 11).

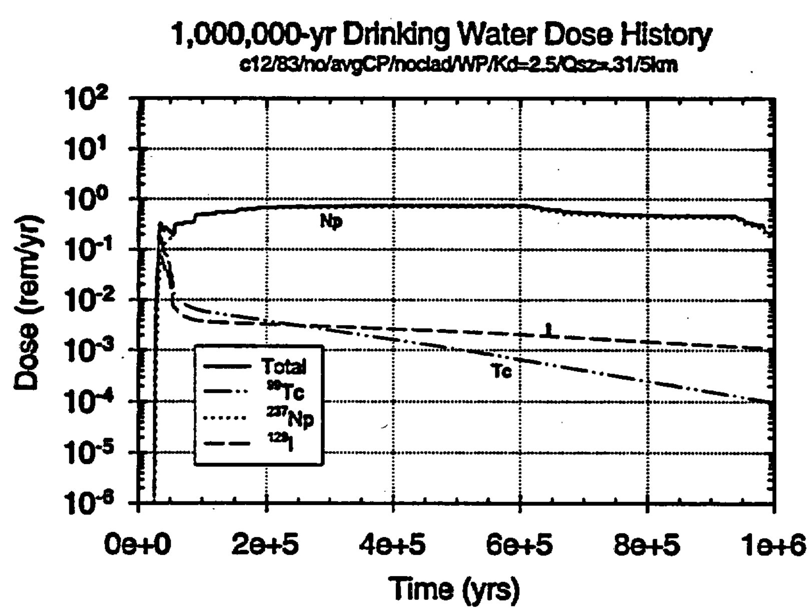


Figure 3-13 Expected-value dose history: 1,000,000 years, 83 MTHM/acre, no backfill, average galvanic protection, drips on waste package (case 12).

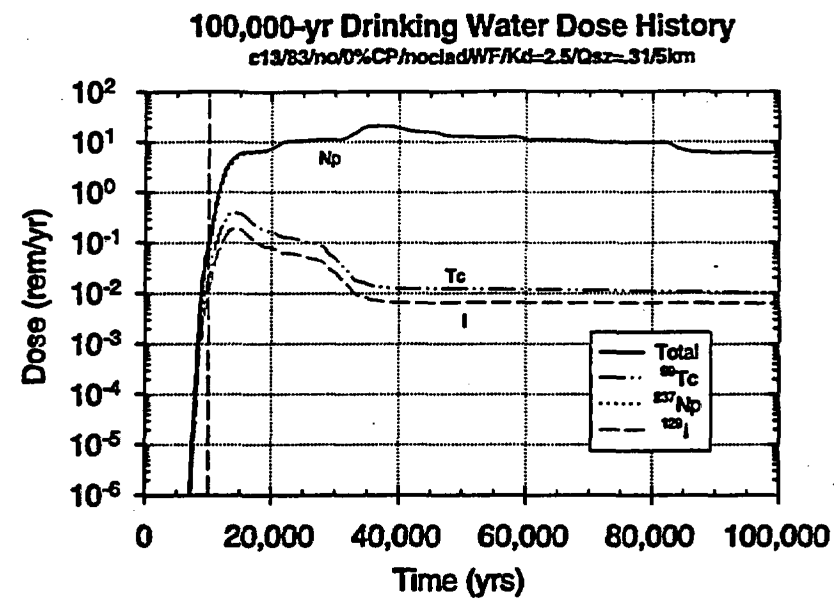


Figure 3-14 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, no galvanic protection, drips on waste form (case 13).

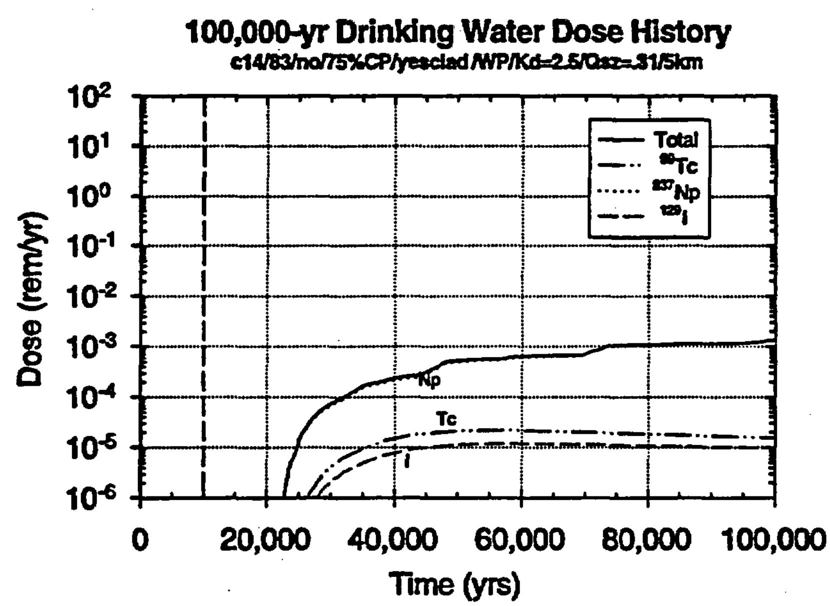


Figure 3-15 Expected-value dose history: 100,000 years, 83 MTHM/acre, no backfill, 100% galvanic protection, drips on waste package, cladding (case 14).

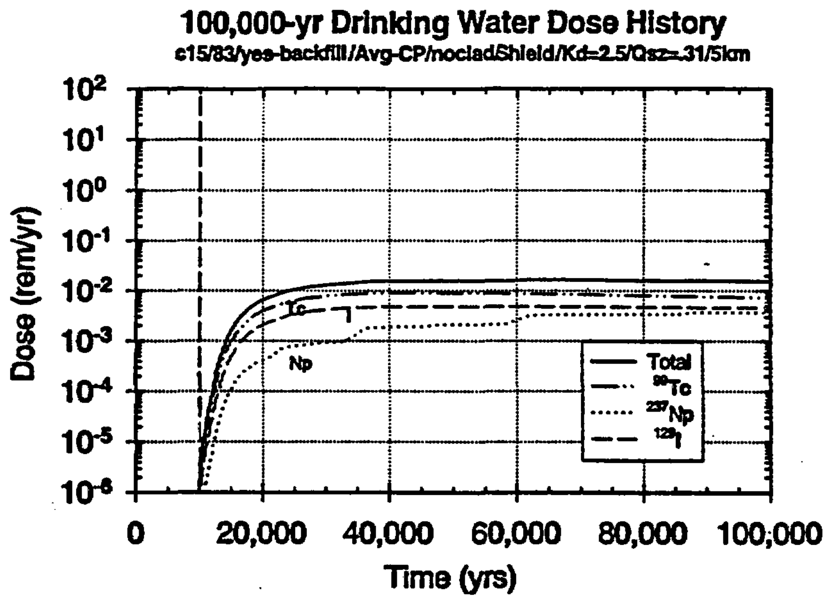


Figure 3-16 Expected-value dose history: 100,000 years, 83 MTHM/acre, backfill, average galvanic protection, drip shield for the entire period (case 15).

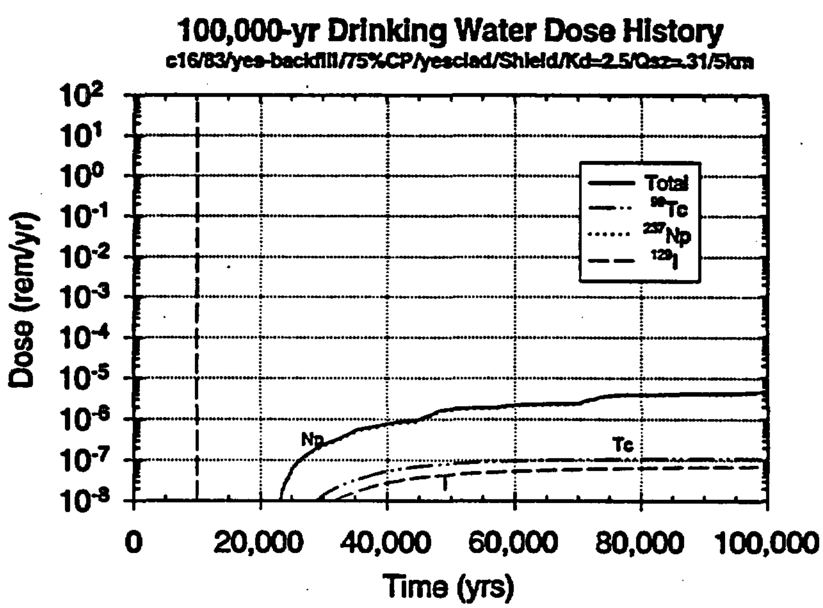


Figure 3-17 Expected-value dose history: 100,000 years, 83 MTHM/acre, backfill, 100% galvanic protection, drip shield for the entire period, cladding (case 16).

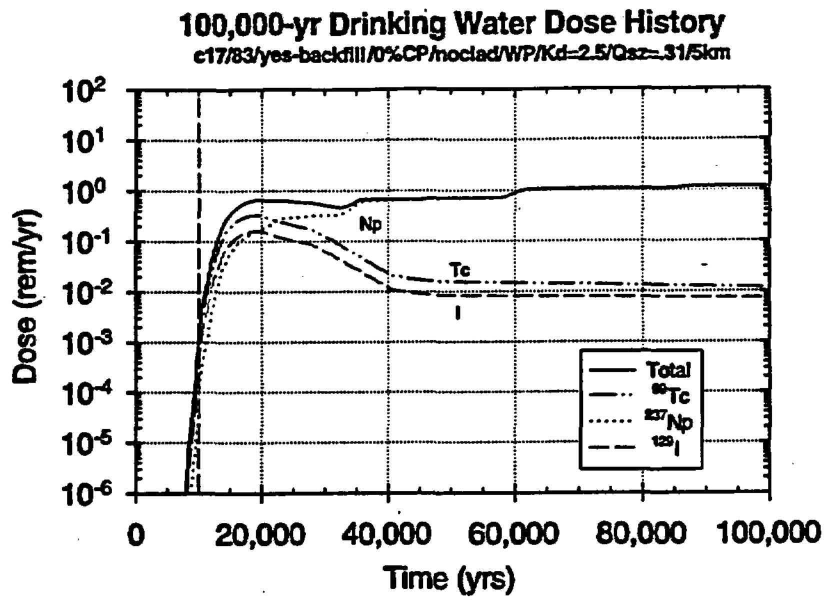


Figure 3-18 Expected-value dose history; 100,000 years, 83 MTHM/acre, backfill, no galvanic protection, drips on waste package (case 17).

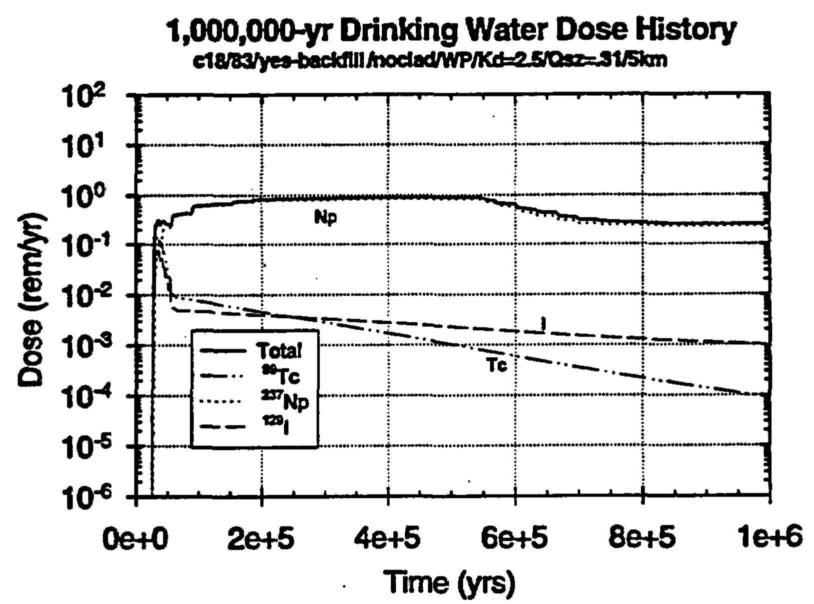


Figure 3-19 Expected-value dose history: 1,000,000 years, 83 MTHM/acre, backfill, average galvanic protection, drips on waste package (case 18).

DELETED

Figure 3-20 **DELETED**

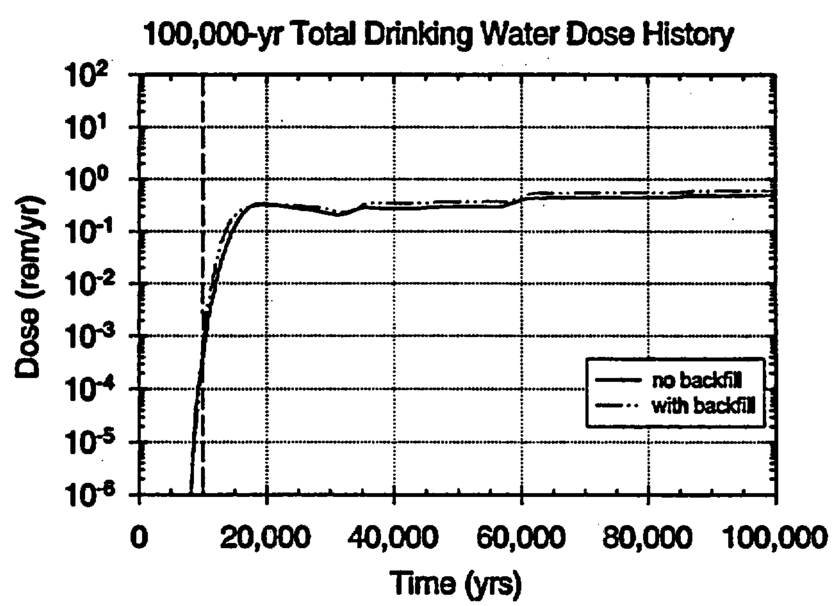


Figure 3-21 Comparison of backfill: 83 MTHM/acre, no backfill *vs* 83 MTHM/acre, with backfill (cases 1 and 3)

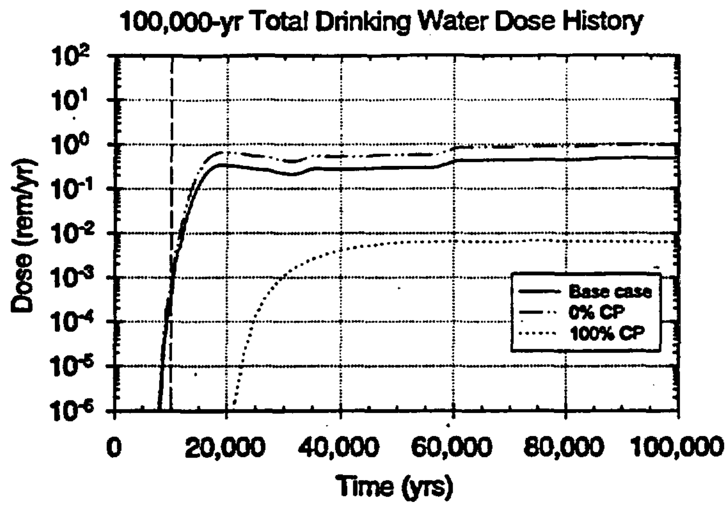


Figure 3-22 Comparison of galvanic protection: 83 MTU/acre, average galvanic protection vs 83 MTU/acre, no galvanic protection vs 83 MTU/acre, 100% galvanic protection (cases 1, 4 and 5)

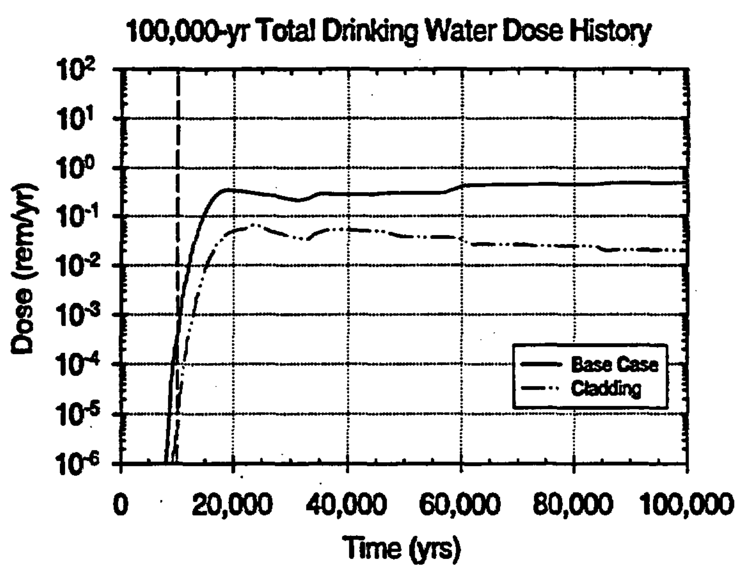


Figure 3-23 Comparison of cladding: 83 MTU/acre no cladding vs 83 MTU/acre, with cladding (cases 1 and 6)

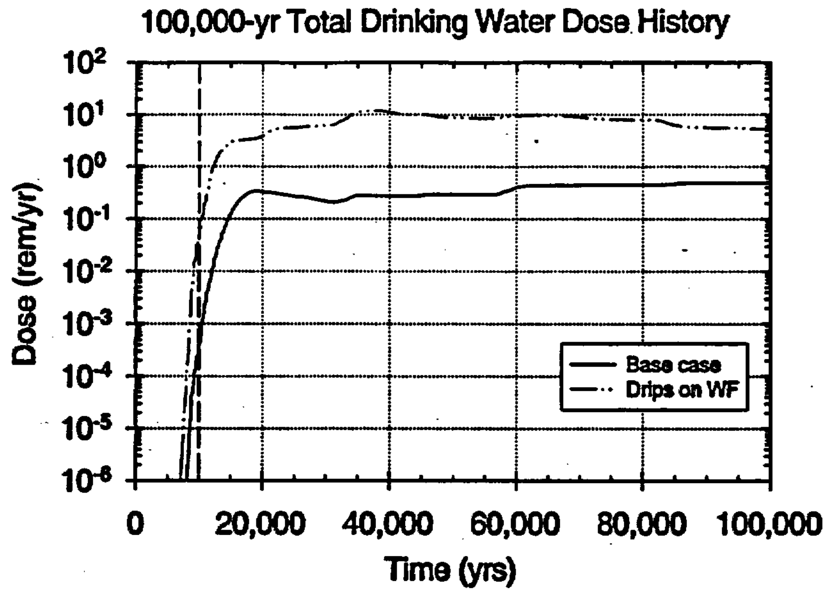


Figure 3-24 Comparison of EBS release: 83 MTU/acre, drips on waste package *vs* 83 MTU/acre, drips on waste form (cases 1 and 7).

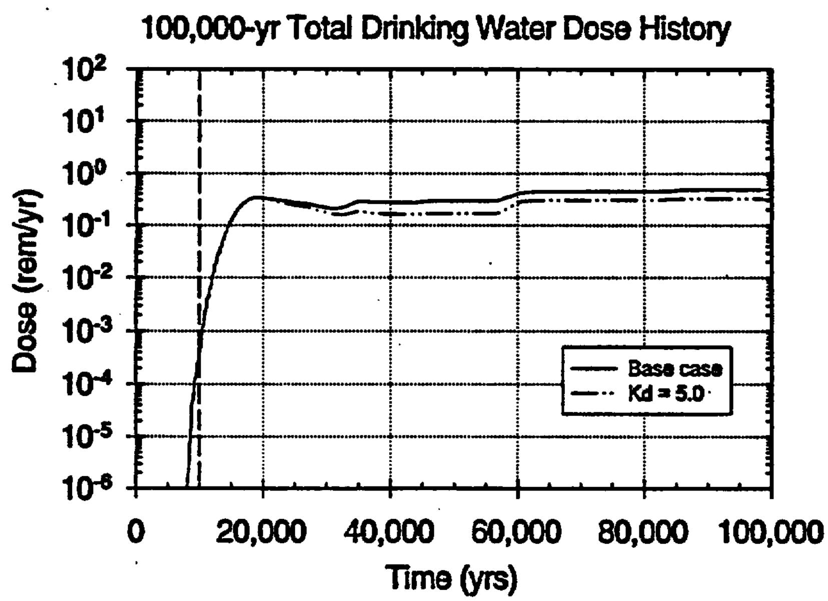


Figure 3-25 Comparison of retardation of Np-237: 83 MTU/acre, Kd=2.5 for Np-237 in the zeolites *vs* 83 MTU/acre, Kd=5.0 for Np-237 in the zeolites (cases 1 and 8).

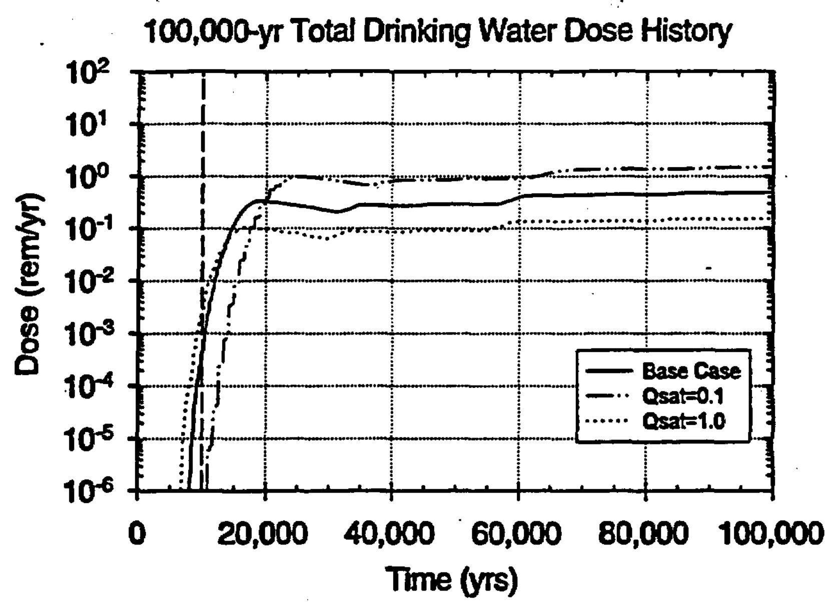


Figure 3-26 Comparison of saturated zone flux: 83 MTU/acre, qsat=0.31 m/yr *vs* 83 MTU/acre, qsat=0.1 m/yr *vs* 83 MTU/acre, qsat=1.0 m/yr (cases 1, 9, and 10).

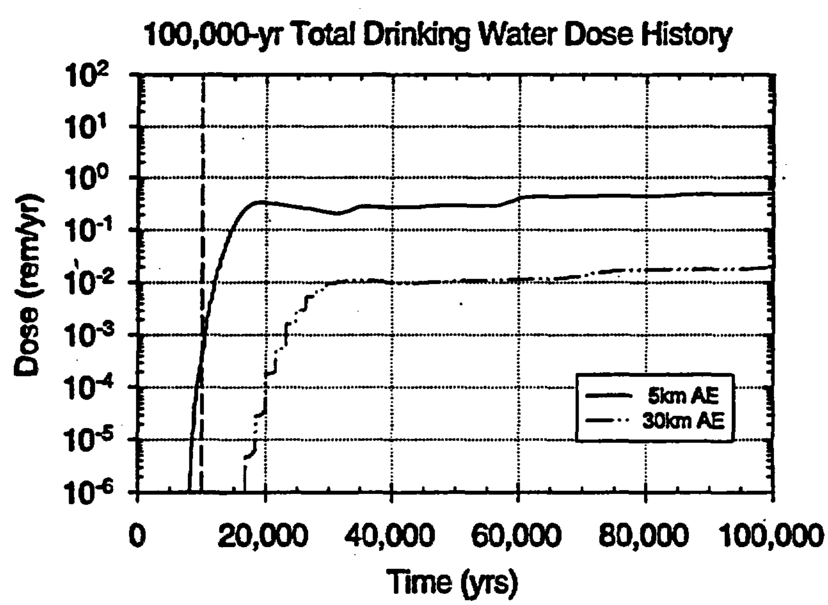


Figure 3-27 Comparison of dose at different distances from the repository; 83 MTU/acre, 5km *vs* 83 MTU/acre, 30 km (cases 1 and 11).

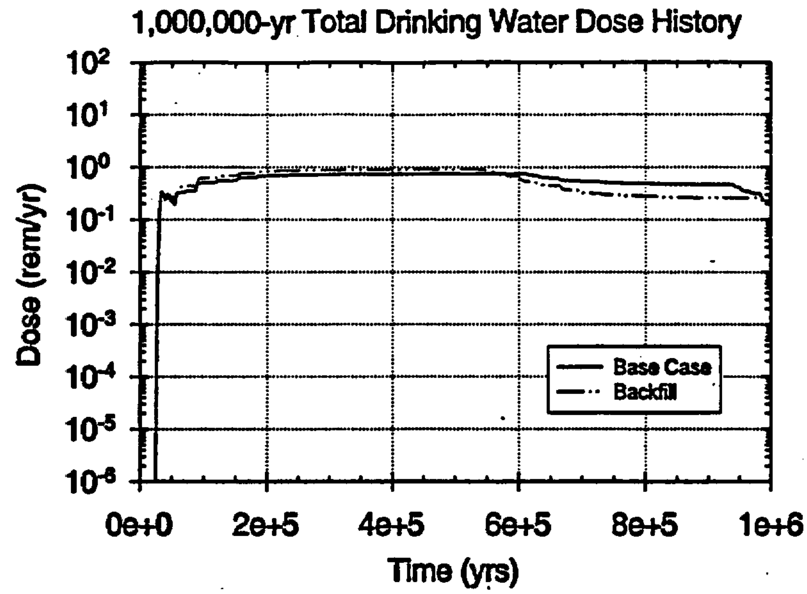


Figure 3-28 Comparison of backfill for a period of 1,000,000 years: 83 MTU/acre, no backfill *vs* 83 MTU/acre, with backfill (cases 12 and 18).

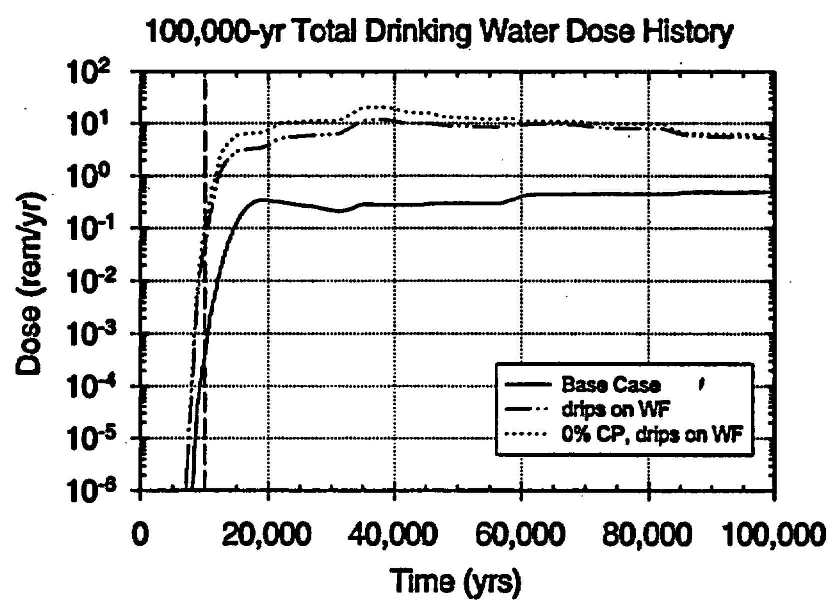


Figure 3-29 Comparison of EBS release and galvanic protection: 83 MTU/acre, drips on waste package, average galvanic protection vs 83 MTU/acre, drips on waste form, average galvanic protection vs 83 MTU/acre, drips on waste form, no galvanic protection (cases 1, 7, and 13).

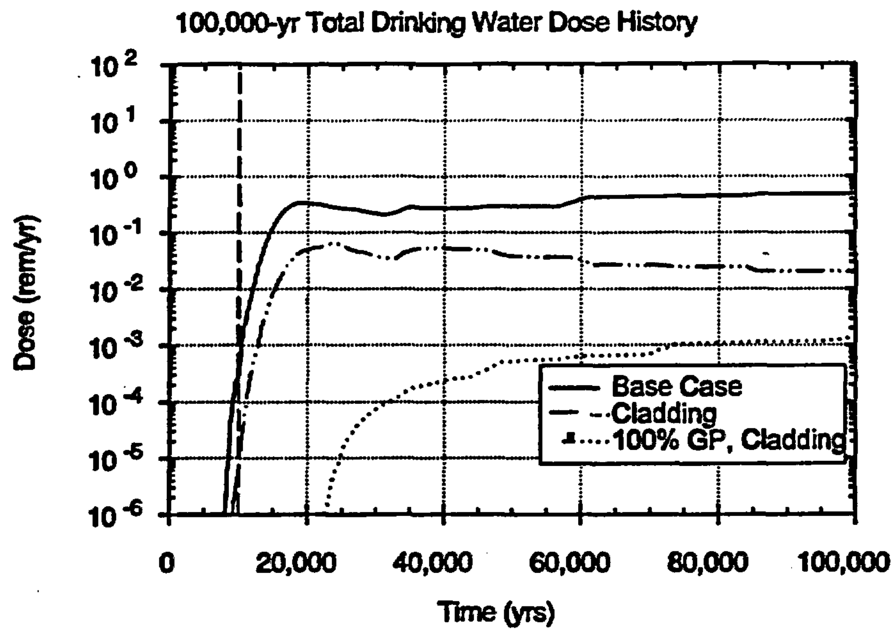


Figure 3-30 Comparison of galvanic protection and cladding: 83 MTU/acre, average galvanic protection, no cladding vs 83 MTU/acre, average galvanic protection, cladding vs 83 MTU/acre, 100% galvanic protection, cladding (cases 1, 6, and 14).

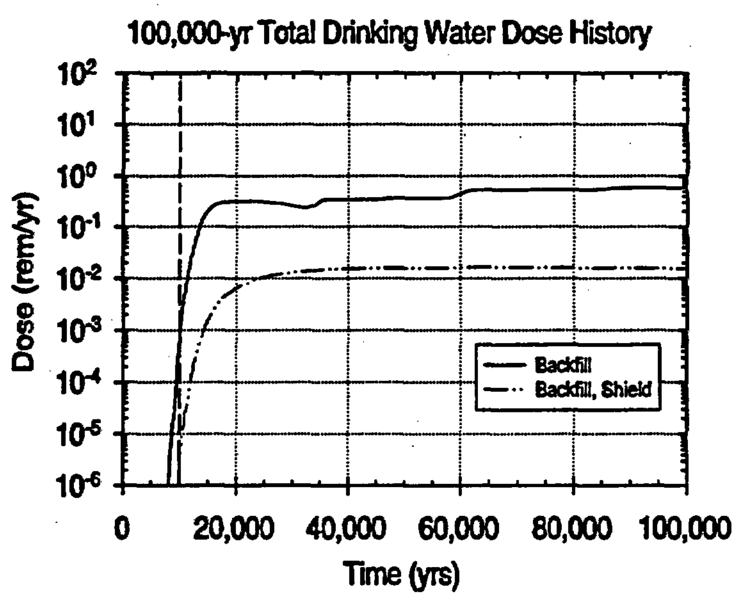


Figure 3-31 Comparison of drip shield: 83 MTU/acre, drips on waste package vs 83 MTU/acre, drip shield (cases 3 and 15).

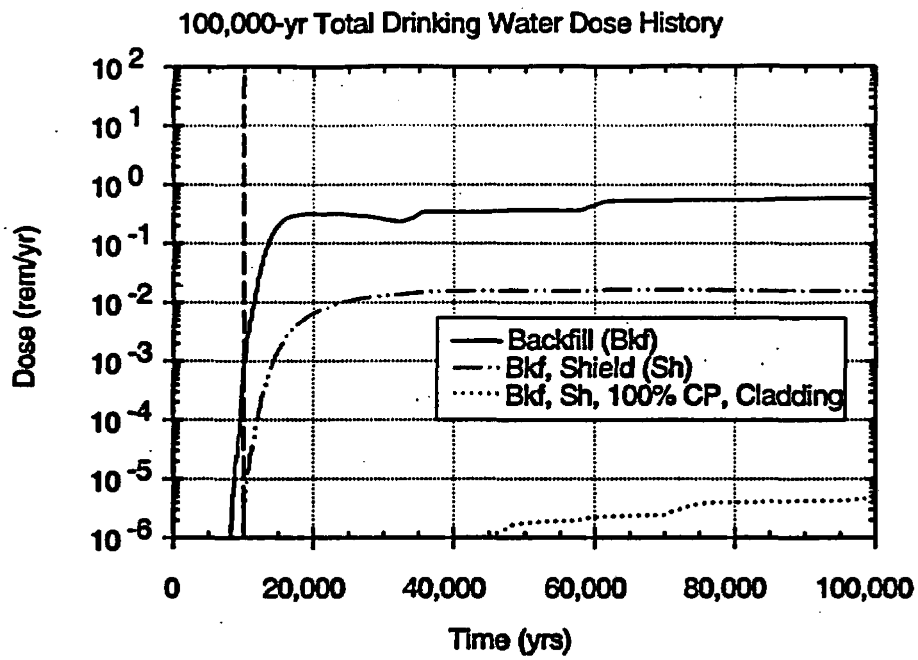


Figure 3-32 Comparison of drip shield, galvanic protection and cladding: 83 MTU/acre, average galvanic protection, no cladding vs 83 MTU/acre, drip shield, average galvanic protection, no cladding vs 83 MTU/acre, drip shield, 100% galvanic protection, cladding (cases 3, 15, and 16).

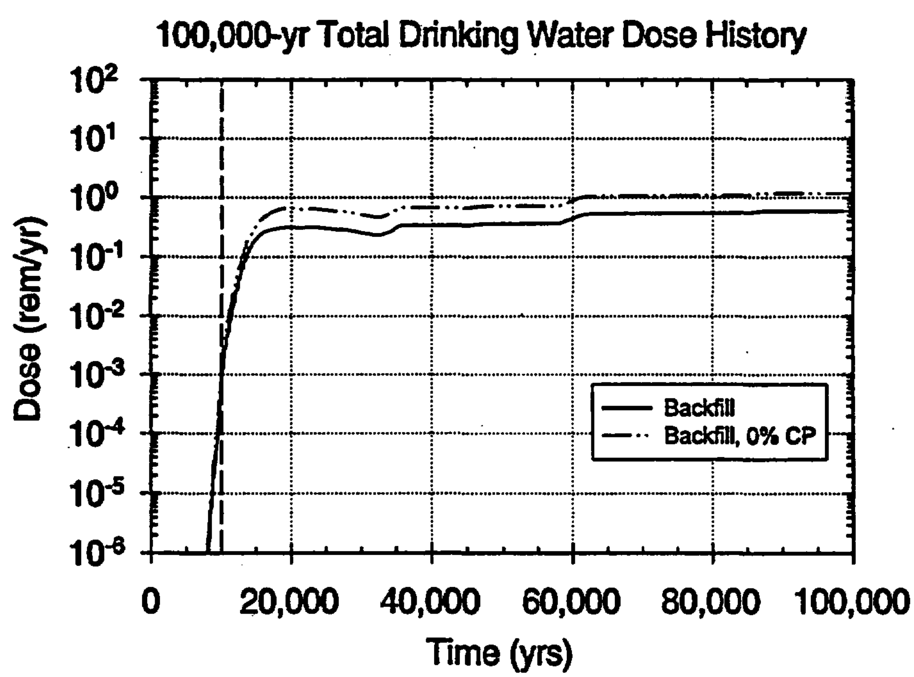


Figure 3-33 Comparison of backfill and galvanic protection: 83 MTU/acre, backfill, average galvanic protection *vs* 83 MTU/acre, backfill, no galvanic protection (cases 3 and 17).

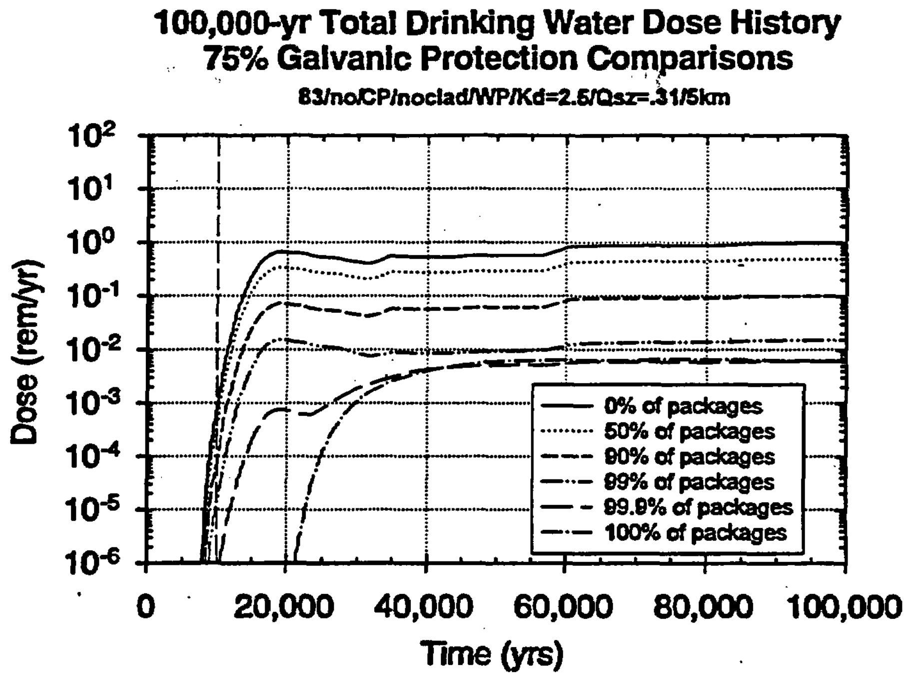
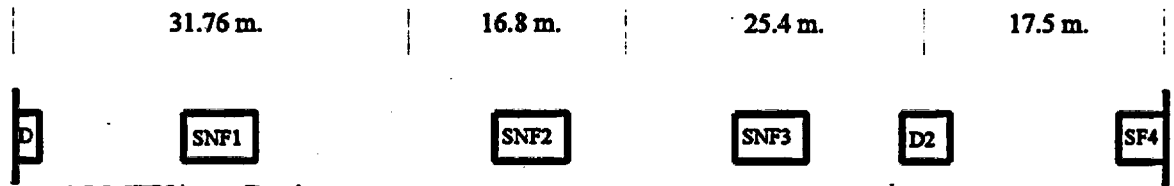
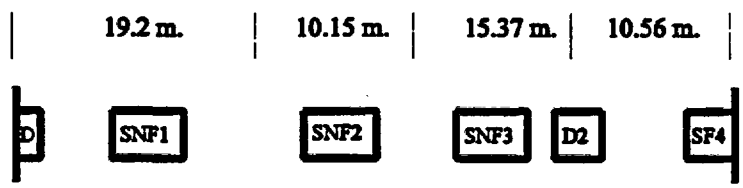


Figure 3-34 Comparison of Various Percentages of WPs that have 75% Galvanic Protection for 83 MTU/Acre



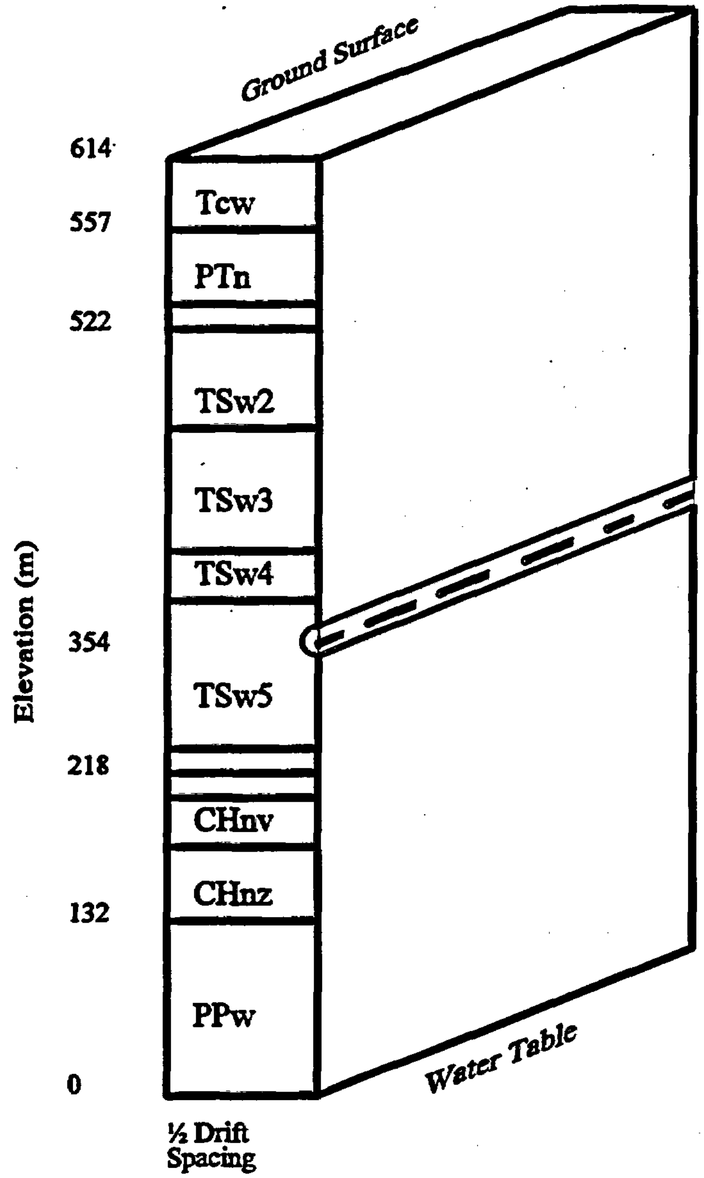
25 MTU/acre Design



83 MTU/acre ACD

	6.2 kgU/m^2	20.5 kgU/m^2
LML (MTHM/m) 0.28	0.28	0.46
Drift Spacing (m)	45.0	22.5
Inner Drift Diameter (m)	5.1	5.0
Length of Drift Section (m)	91.5	55.3

Figure 3-35 Driftwise spacing of waste packages in the three dimensional models. No Linear Mass Loading credit given to defense waste packages



Package Distribution

½ DHLW1

1 SNF1 26 year old 8.83 MTU

1 SNF2 40 year old 4.67 MTU

1 SNF3 26 year old 7.07 MTU

1 DHLW2

½ SNF4 10 year old 9.74 MTU

Figure 3-36 Three-Dimensional Model Description

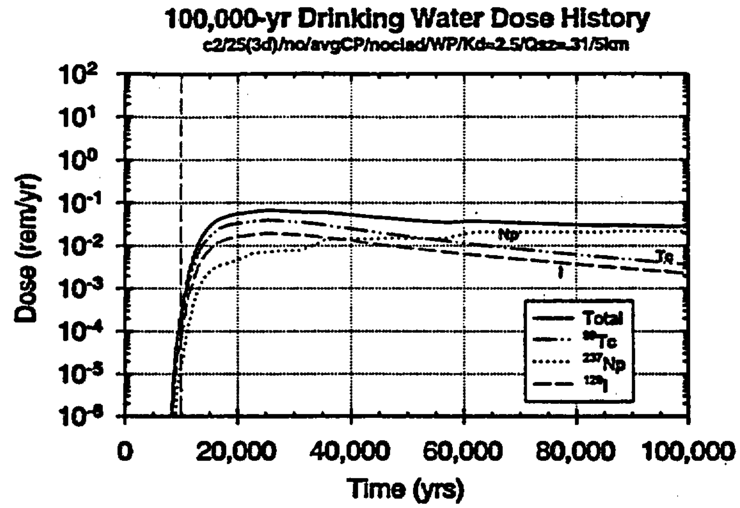


Figure 3-37 Drinking water dose histories for 3-dimensional runs for the low thermal load case of 25 MTHM/acre. The three long-lived radionuclides are plotted as a function of time

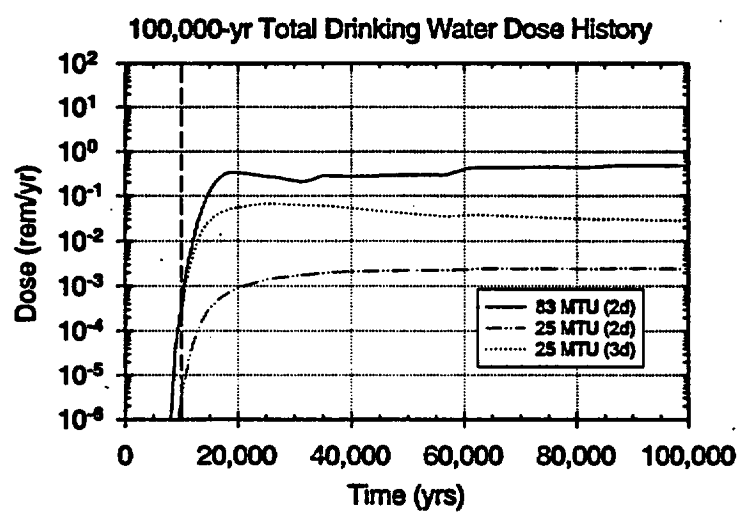


Figure 3-38 Comparison of the total drinking water dose histories for the 2-dimensional cases for 83 and 25 MTHM/acre and the 3-dimensonal case for 25 MTHM/acre

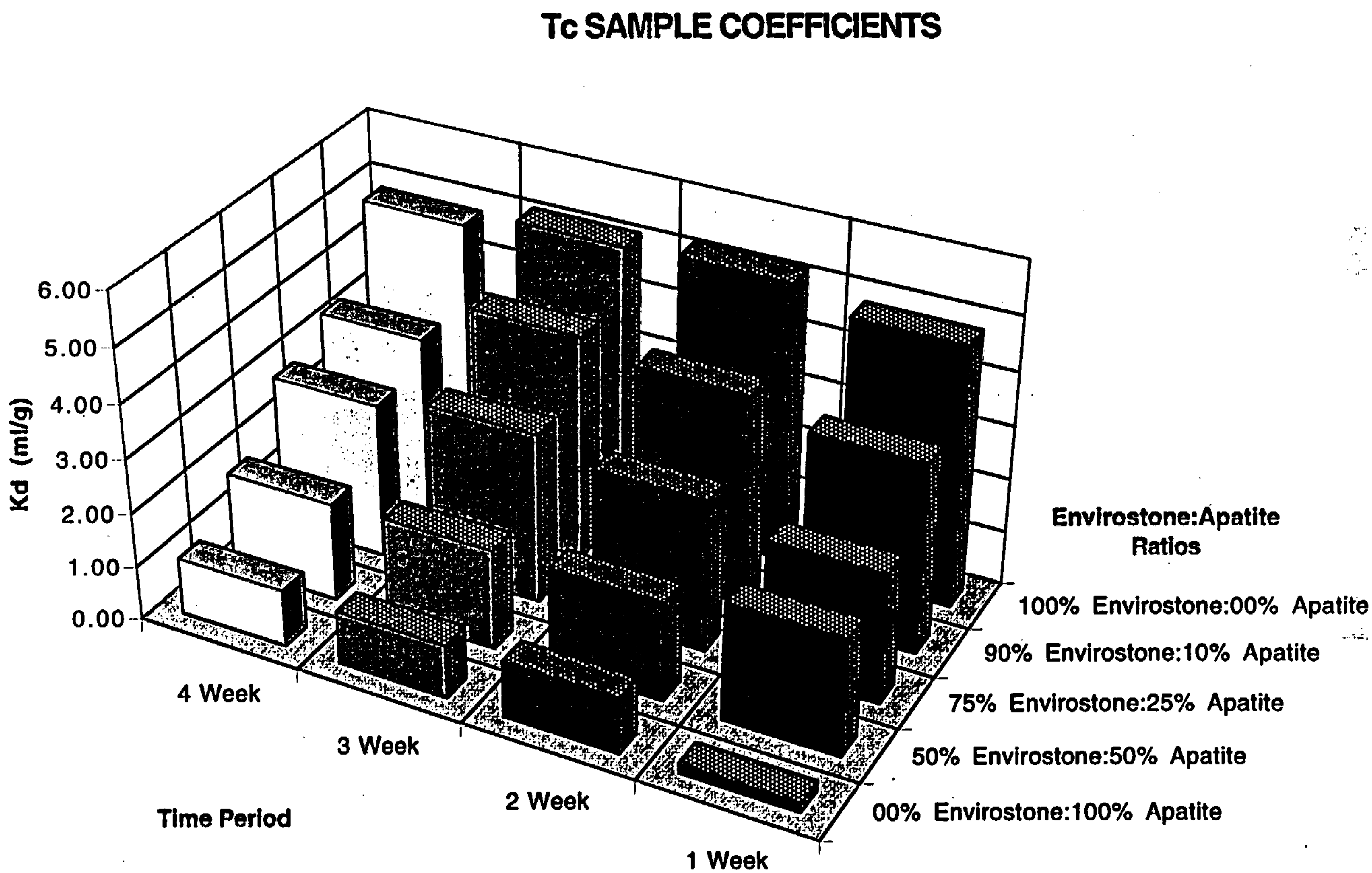


Figure 3-39 Sample Kds of Tc for Envirostone and Apatite

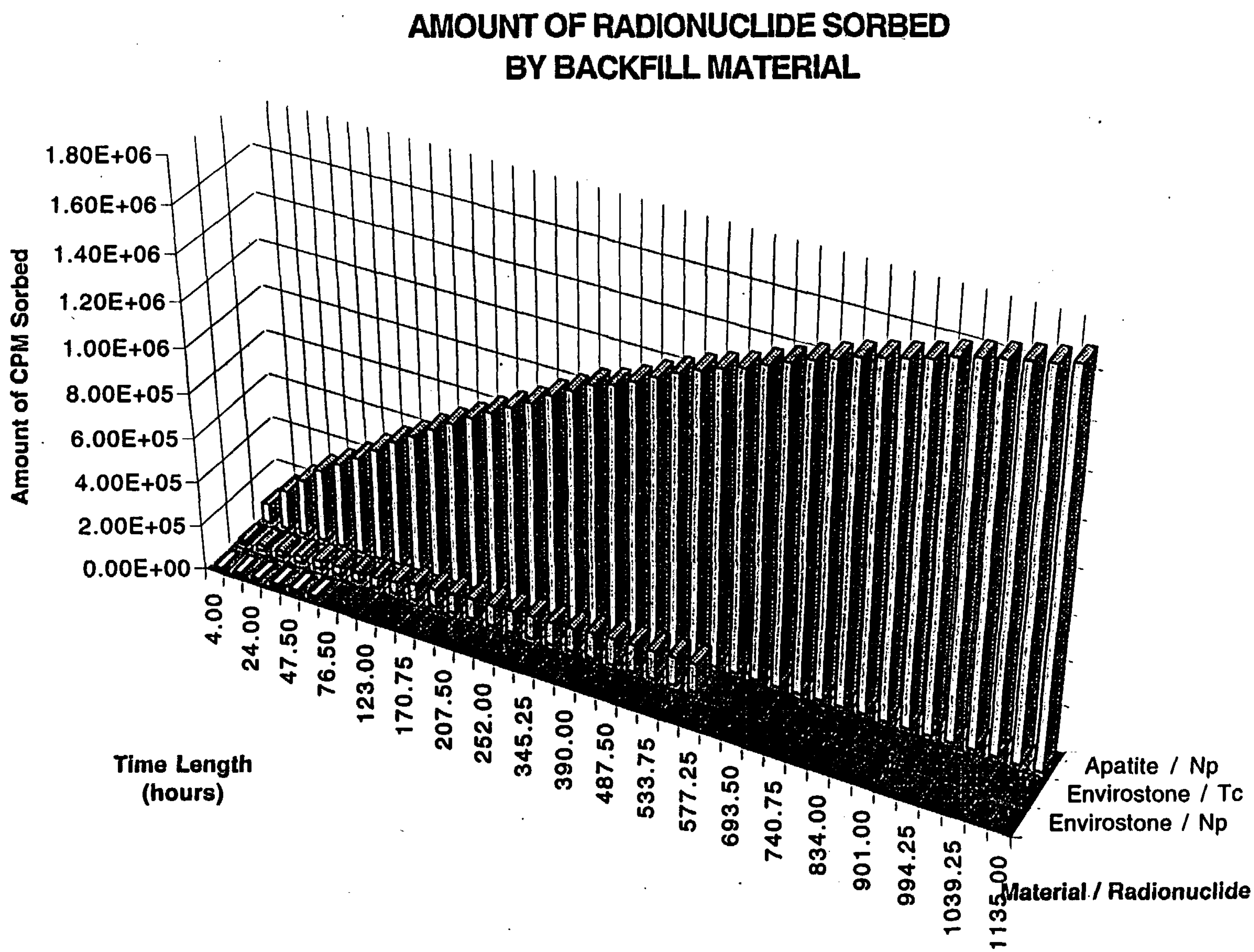


Figure 3-40 Sorption Potential as Function of Time and Backfill Material for Np and Tc

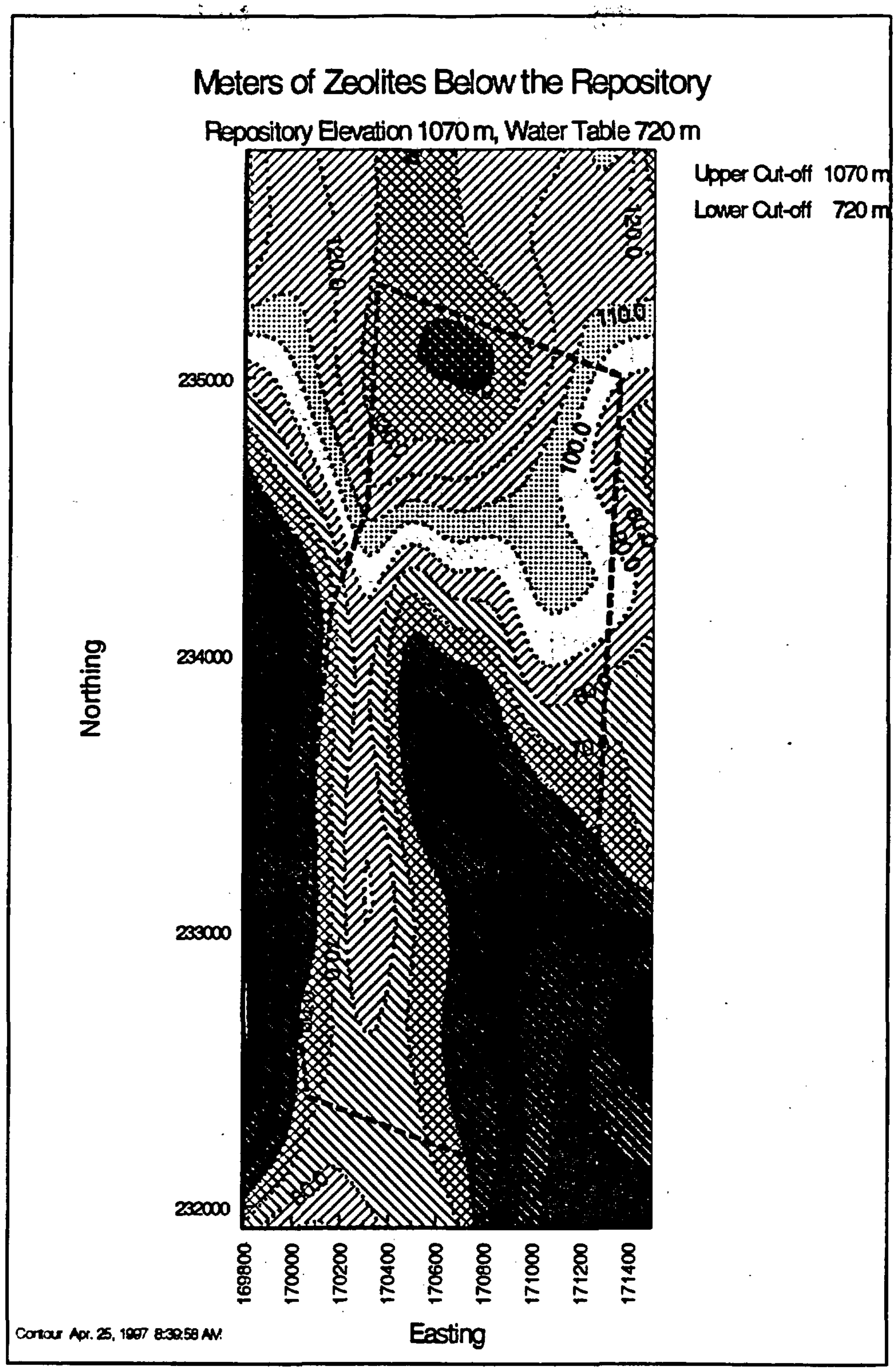


Figure 3-41 .Equivalent Depths of Zeolites Beneath the Potential Repository from the Repository Horizon to the Top of the Water Table

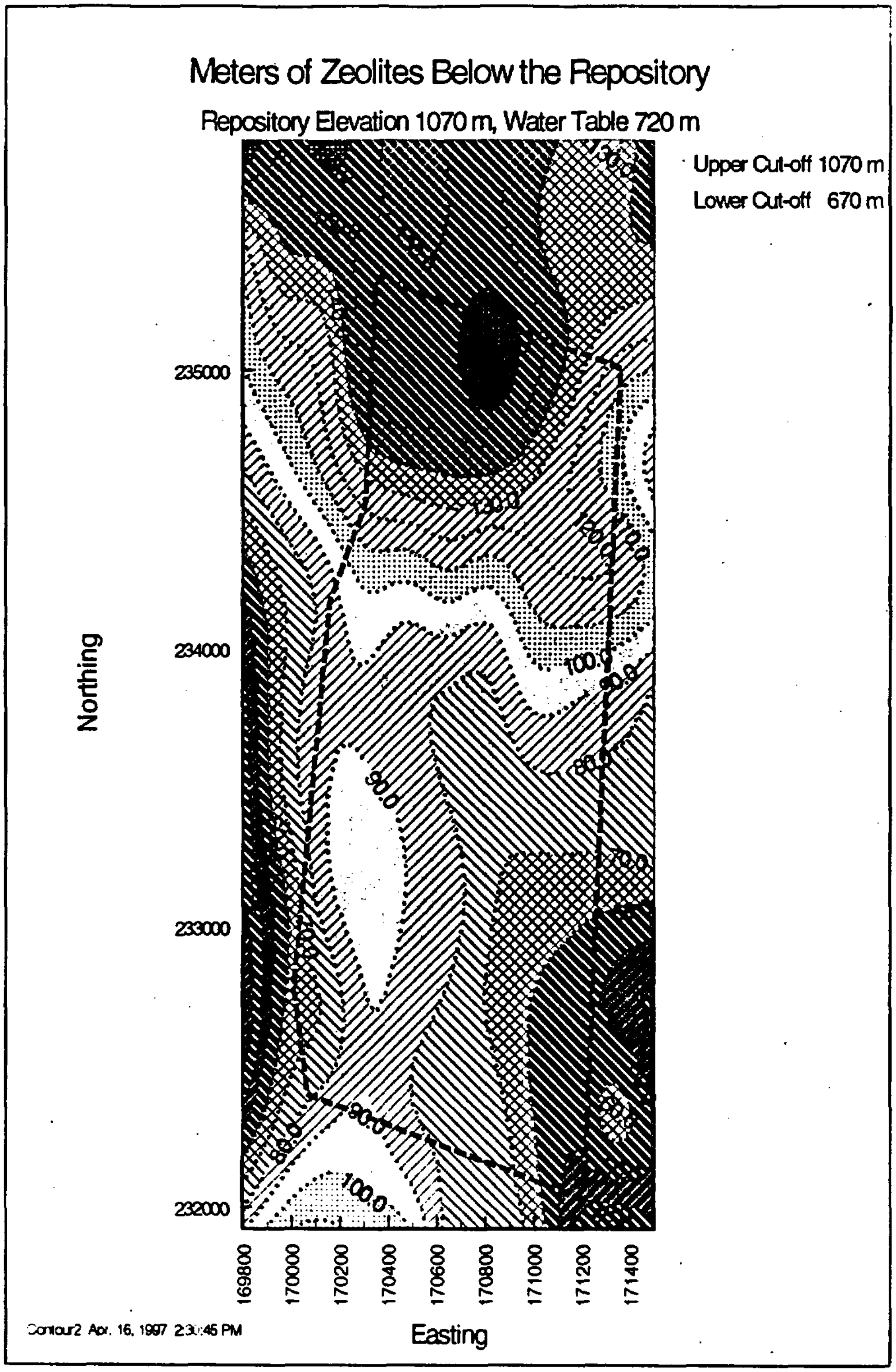


Figure 3-42 Equivalent Depths of Zeolites Beneath the Potential Repository from the Repository Horizon to 50 m into the Saturated Zone

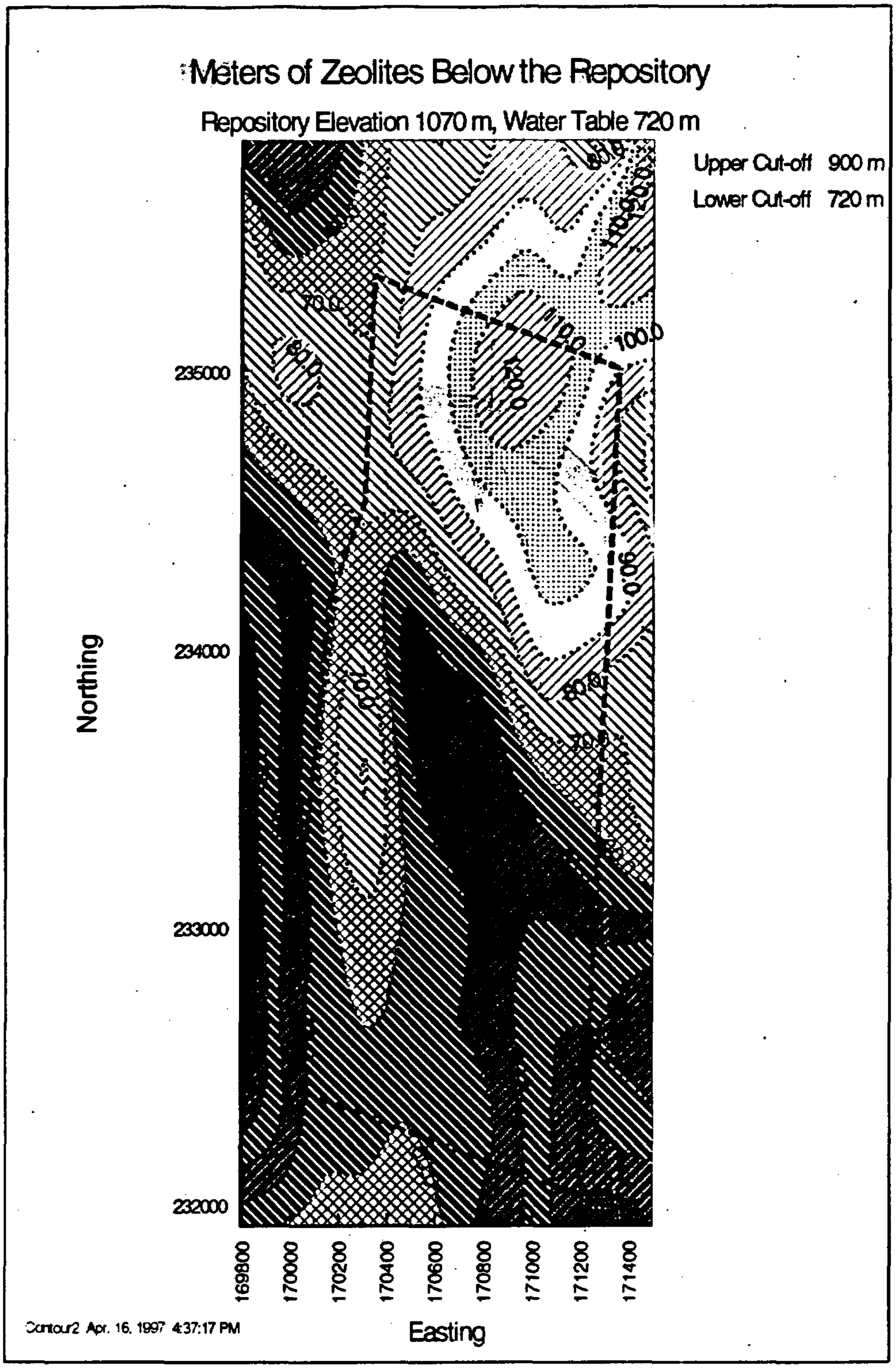
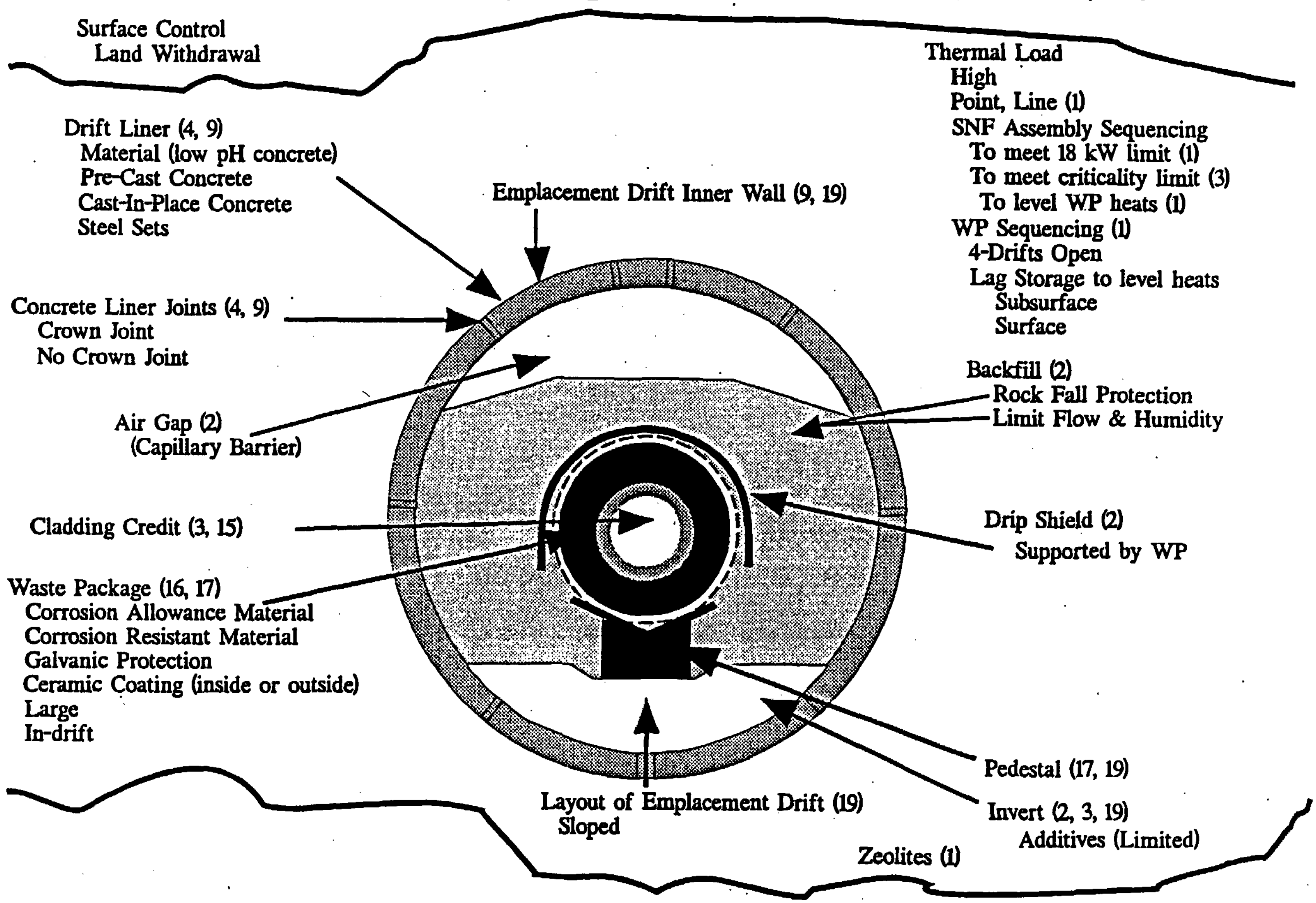


Figure 3-43 Equivalent Depths of Zeolites Beneath the Potential Repository from 170 m Beneath the Repository Horizon to the Top of the Water Table

MGDS VA Design Options for Waste Isolation (10, 18)



Parenthetical numbers are VA design issue numbers, issues 5, 6, 7, 8, 11, 13, 14, 20, 21 not depicted

Figure 3-44 MGDS Desig[illegible]tions for Waste Isolation (10, 18)

4. COST ANALYSIS

4.1 ENGINEERED BARRIER COSTS

An element in the process of deciding which engineered barriers should be pursued is the cost of those engineered barriers. If two engineered barriers provide similar performance but the cost of one is substantially greater than the other then it is important to factor this into the decision-making process. The costs of the various engineered barrier options considered in this report are evaluated in this section. The engineered barrier options considered are:

- WP
- Spent fuel element cladding
- Galvanic protection
- Chemically conditioned invert
- Drip shields
- Backfill
- Concrete invert with WP pedestals
- Tunnel liner.

To support the evaluations of the engineered barrier subsystem alternatives, each will be briefly described and their costs identified in constant 1997 dollars. In some cases components or alternatives were costed even though no performance calculations were made for these cases. The components were considered in discussions but PA calculations could not distinguish between components.

As indicated in Section 1, the study was to address what could be done to provide a threshold for determining a significant reduction in peak dose for a reasonable cost. It is difficult to establish what a "reasonable" cost is since, with the NRC, cost should not be an issue in safety. The Project must work within the existing budgetary constraints and as such wants to determine where they can get the "biggest bang for the buck." Although rather vague it was decided to take guidance from ALARA which says to limit personnel and environmental radiation exposure to lowest levels commensurate with sound economic and social considerations. Thus, "reasonable", although still rather vague, will be established as costs that follow sound economic considerations. Thus, a "reasonable" cost is one that would be economically sound and would reduce dose by at least a factor of ten. Although very subjective it was decided above that a "reasonable" cost estimate at this point might be one billion dollars. The cost analysis in this section will be used to help make such recommendations.

Waste Package—As discussed earlier, the WP design for the SNF is a disposal container that has a double barrier, a corrosion allowance outer barrier of ASTM A 516 steel, and a corrosion resistant nickel alloy 625 (ASTM B 443) as an inner barrier (CRWMS M&O 1996f). There is estimated to be a total of 2,859 large WPs for BWR fuel, 4,137 large WPs for PWR fuel, and 683 small WPs for PWR fuel, for a total of 7,679 containers at an estimated cost of $3,081 million. A total of 3,259 DHLW waste packages at a total cost of $831 million add to a total cost of $3,912 million FY 1997 dollars. The cost is based on a waste stream that was modified from the CDA Rev 4 waste stream to include DOE SNF. This waste stream was used in the modified Project Cost Estimate for 1997.

Spent fuel element cladding—The spent fuel element cladding is the zircaloy cladding surrounding the fuel pellets and is part of the fuel element assemblies to be put in the WPs. The tubes which haven't unzipped contain the radionuclides of the spent fuel and provide the first barrier of containment. Additionally, for elements where the cladding has failed under static loads, some credit may be possible for reducing the area of the SNF available for dissolution. There have been many studies conducted on cladding failure, the mechanism of failure, failure rates, temperature dependency, creep rupture, and quantification of failures in spent fuel. Additional studies regarding the long-term impact of some of the cladding degradation mechanisms on the cladding failure models is desirable. However, developing long-term performance data for the failure models can be both lengthy and costly. Consequently, a practical approach is to use available literature, reports, and data combined and synthesized with theoretical models developed for other programs to support arguments to the U. S. Nuclear Regulatory Commission in obtaining some credit for this barrier in calculating the dose rates expected from the repository (CRWMS M&O 1996d). The Navy is also developing corrosion models for zircaloy and these models could be evaluated. PA models will be modified to take credit for the cladding barrier and include the models for the degradation of the cladding. Degradation models being considered need to ultimately address:

- Delayed hydrogen cracking
- Hydride re-orientation
- Fluorine pitting
- Stress cracking at the repository pressures and temperatures (YMSCO Waste Form Workshop conducted by the M&O on 2/21-24/97).

Because the fuel cladding barrier is an integral part of the spent fuel elements received for the WPs, there is no added physical cost to the engineered barrier system for the barrier. The research effort to utilize existing data, extrapolate it to the repository conditions, develop additional cladding degradation models, and modify the performance models to account for the cladding barrier is estimated to cost around $0.5 million excluding any experimental programs that may be identified later (cost est=6 people x 0.5 yr x $150k=$450k). Not included in this are some short term and long term tests on the effect of relative humidity and dripping on SNF segments. There may also be some related Performance Confirmation costs if the NRC requires further evaluations but the Performance Confirmation Study is not planning any at this time.

Galvanic Protection—Galvanic protection is the protection provided a more-noble metal or alloy by the corrosion of a less noble metal or alloy in electrical contact exposed to the same corrosive electrolyte. This protection provided by the outer corrosion allowance ASTM A 516 steel barrier can delay the corrosion of the WP inner barrier, a corrosion resistant nickel alloy 625 (ASTM B 443). A significant delay in the initiation of the corrosion of the inner barrier will diminish the temperatures of the WPs and hence the temperature dependent corrosion rates. This protection is an added feature that is present because of the two metals and adds no physical costs. However, research, data collection, testing, and model development will be required to verify, with reasonable assurance, the expected galvanic performance provided to the inner liner of the WP during the near term and long term. The FY 1997 and FY 1998 costs estimated for *Electro-chemical Basis for*

Galvanic Testing (Annual Planning Sheets (APS) Item No. TR251FB7-FY97 and TR251GB7-FY98), *Long Term Galvanic Tests* (APS Item No. TR251FBB-FY97 and TR251GBB-FY98), and *Model Development* (approximately 1/5 of total APS Item No. TR251FBE-FY97 and TR251GBE-FY98) identified for developing the galvanic performance predictions for the repository WPs is estimated to be about $1.4 million. Any additional budget will be estimated at the completion of the FY 1998 work.

Chemically conditioned invert—One MGDS alternative is to support the WPs on a crushed tuff invert. Previous research studies have found that apatite minerals contained in phosphate material or other materials, such as envirostone, might prove to be valuable radionuclide retardant additives in the emplacement drift invert, at least for ^{237}Np. Hence, the addition of a phosphate layer was considered a method to sorb the release of ^{237}Np from failed WPs. The thickness of the apatite layer deemed necessary for sorption purposes, while not thwarting the ability of the invert to structurally support the WPs and rail carriage, still needs to be determined. Only the total amounts of apatite and envirostone were estimated. The costs for adding a layer of apatite will range from $24 million for 0.2 m thickness to $338 million for 1.2 m thickness. A layer 0.5 m thick will cost about $94 million (CRWMS M&O 1996a; 1996 dollars converted to 1997 dollars). These thicknesses could affect the current design and should be evaluated by subsurface design. The referenced study was based on repository drifts sufficient for 12,037 WPs and the current design is for 10,938 WPs. Allowing for some potential areas where WPs would not be emplaced, the length of required drifts and inverts would be approximately the same.

Drip shields—Preventing liquid water contact with the WPs is considered a potential function that could extend the WP lifetime based on the assumption that liquid water contact would enhance corrosion rates. Current corrosion models reflect corrosion initiation and corrosion rates of WP surface conditions in terms of relative humidity and temperature only, which is consistent with the expected infiltration rates where only a few packages will see drips. At higher infiltration rates (1.25 mm/year), it is estimated that about 50 percent of the WPs would see drips. Actually the 50% was used in TSPA-95 and for the new database (Bodvarsson et al. 1996) only about 29% of the WPs see drips. One design alternative that would prevent the drips from impinging on the WPs is the installation of a thin Titanium shield fit over the outer containment barrier. This titanium shield may have to be separated somewhat from the WP possibly by a layer of backfill or separators. The longevity of the design is still of concern due to possible damaging rock strikes and chemical interactions.

The cost of providing drip shields for 12,037 WPs is estimated to be about $433 million (CRWMS M&O 1996a) For the current design with 10,938 WPs, the cost for drip shields is estimated by applying the ratio of the WPs to the cost for the 12,037 packages. Hence, the cost for drip shields for the current number of WPs is estimated to be $404 million (Cost=$433M x 10938/12037=$393M in 1996 dollars or $404M in 1997 dollars).

Backfill—The current concept for placing backfill in the emplacement drifts, if determined that it is needed, incorporates the center-in-drift design. The WPs would be emplaced in the drift and backfilling would not be done until just before final closure, which could be 100 years after emplacement. The backfill would be emplaced using existing equipment for emplacing WPs. Backfilling equipment would consist of a rail-mounted train of material supply cars and a locomotive that would transport the material underground to either the individual waste emplacement drifts being backfilled or to side tracks for holding until the material is needed for backfilling. Within a waste emplacement drift the backfill would be emplaced using a self-propelled backfill stower. An additional material supply car with locomotive would shuttle material from the entrance of the waste emplacement drift to the stower. The stower would slowly retreat to the drift entrance as the stowing of backfill is performed. Additional equipment that supports the backfilling would be located on the surface. Because of the hostile thermal and radiation conditions within the emplacement drifts, the use of remote control of all backfilling would be required. Remote-controlled equipment would include operator control stations, wireless communication networks, video monitors, and various sensing devices. The cost for backfilling is estimated to include $158.5 million to $239.4 million for the equipment / stowing operation costs depending on the type of stowing operation utilized and the use of onsite available crushed tuff. If Quartz Sand is used to reduce wicking potential the costs will increase an additional $80.9 million to $118.6 million depending on the type of stowing used to place the backfill (wind row vs. side-to-side) (CRWMS M&O 1996a; 1996 dollars converted to 1997 dollars). No cost analysis was done of a Richards Barrier backfill concept.

Concrete invert with WP pedestal—One of the current drift designs is to use precast concrete inverts to be placed in the bottom of the drift tunnels to support the WPs. There are various options to this design being considered. The most current design is to use precast bottom tunnel segments placed end to end in the bored tunnels to form a continuous bottom lining to the tunnel. About every 1-1.5m a WP support will be placed that locks into the bottom floor segments. These supports will provide a pedestal off the floor with a "V" notch to support the WPs. They may be constructed of all concrete or a combination of a concrete base with a steel "V" support. These pedestal supports will be evenly placed along the repository drifts in order to support all WP spacings required for the various thermal loadings.

The costs of the precast concrete liner and invert components were extracted from the Subsurface Group's (M&O/MK) cost database, which included the cost for "rail, electrification, and communication cost component." After deleting this cost component, as it should be nearly the same for all emplacement alternatives, the costs were divided between those of the concrete invert and the liner. Also the costs for the batch plant, precast yard operations, capital equipment, and general expenses common to both the liner and the invert were split proportional to the cost of the liner and the concrete invert. Hence, the cost estimated for the concrete invert with WP support piers and crushed tuff placed between the piers is about $435 M for the repository.

Tunnel liner—A concept being considered for repository tunnel design includes the use of a concrete tunnel liner. This liner will be made of precast segments placed in the bored tunnel and interlocking with the concrete invert described above to form a continuous tunnel lining. This lining is expected to provide rock protection for the WPs and a barrier to water ingress. However, the longevity after closure and performance implications of the cementitious material need to be resolved. The cost of precast concrete tunnel liners was obtained from the 1997 Program Cost Estimates as $3,930 per meter which when multiplied by the length of tunnel including a 5 percent contingency (163,000 m) yield $640 million for the cost of these liners.

4.2 COST SUMMARY

The methodology for obtaining the costs for the various engineered barriers were discussed above. These costs are summarized in Table 4-1. In summary, the WP costs were estimated using the current concept but these costs could vary substantially if the design were to change, changes in materials, and/or thickness of the various barriers was to change. Cladding and galvanic protection will not require additional cost since these barriers will exist under the current disposal concepts. The costs identified in Table 4-1 are those that would be needed to establish sufficient information to take credit for cladding and/or galvanic protection or at least establish enough information that a determination can be made as to whether licensing credit can be taken for these barriers in a licensing environment. The costs of the remaining barriers would be incurred only if the decision were made to actually employ these barriers.

Table 4-1 Engineered Barrier Costs

Barrier Description	Costs ($millions)
WP	3,912
Zircaloy Fuel Cladding	≥0.5
Galvanic Protection	≥1.4
Chemically Conditioned Inverts	24 to 338
Drip Shields	404
Backfill with Crushed Tuff	158 to 239
Backfill with Quartz Sand	239 to 358
Concrete Invert with Pedestal	435
Tunnel Liner	640

For most of the barriers the largest contribution for the cost is the cost of materials and installation of the barrier. The testing costs to establish performance of a particular barrier for license application is generally a small fraction of the cost. However, in the case of cladding and galvanic protection, the testing and model development are the primary costs. As discussions with NRC develop and proceed the costs to substantiate a barrier could increase. As such for cladding and galvanic protection a greater than or equal sign was used to represent this uncertainty. It should also be noted that no performance confirmation costs were identified for any of the barriers.

INTENTIONALLY LEFT BLANK

5. CONCLUSIONS AND RECOMMENDATIONS

The Waste Isolation Requirements Study was conducted from October 1, 1996 to May 15, 1997. The study examined the performance of the various potential barriers, both natural and engineered, that may contribute to isolation of nuclear wastes emplaced in an underground repository at Yucca Mountain, Nevada. The objectives of the study were to address the following issues:

- Document our current basis of understanding of the performance of various barriers
- Utilize the information to identify the relative merit of the various barriers (engineered and natural)
- Identify the cost of engineered barriers
- Conduct limited calculations to determine the performance of backfill under the potentially higher moisture flux conditions which may exist
- Recommend an approach to evaluate engineering measures that have the potential for significant reduction in peak dose at reasonable cost.

The study relied, for the most part, on work that was done in the *Engineered Barrier System Performance Requirements Systems Study Report* (CRWMS M&O 1996a), the *Description of Performance Allocation* study (CRWMS M&O 1996b), and the *Thermal Loading Study for FY 1996* (CRWMS M&O 1996d). These studies were all conducted at ambient liquid percolation fluxes from about 0.5 to 2 mm/year. Since that work was done, additional measurements and analyses indicate that the percolation flux is potentially between 1 to 10 mm/year. As a result, some additional total system performance calculations were done at a higher flux. Expected value dose calculations were done for a percolation flux of 6.2 mm/year and reported in this study. The main intent of the calculations was to examine the performance of backfill but some calculations with cladding, galvanic protection, and a drip shield were also done. Current uncertainties in unsaturated zone flow and transport are high but should be reduced as a result of tests underway. The study focused on providing performance calculations and document the status of existing performance calculations. For the most part, this analysis is scoping or preliminary in nature and was not intended to qualify a specific barrier's performance.

The interim standard [CDA Key 060] indicates that engineering measures that have potential for significant reduction in peak dose and can be implemented at reasonable cost, should be evaluated. As such, a study objective was to address the question as to what thresholds should be provided for specifying 'significant' reduction in peak dose for a 'reasonable' cost. The current interim safety standard requires that peak doses not exceed 15 mrem/yr from all radionuclides released from the repository through all exposure pathways at a distance of 30 km and a time of up to 10,000 years. However, in light of the long half-lives of several of the radionuclides, the peak doses may occur beyond the 10,000 year limit. In recognition of this potential, the interim standard also requires consideration of engineering measures with the potential to significantly reduce the dose at reasonable cost for time periods beyond 10,000 years. There is little guidance that we can rely on to help define what is meant by significant reduction. However, predicting doses at long times (10,000 years and beyond) involves a significant level of uncertainty. Therefore, a predicted reduction in dose by a factor of two or three may easily be within the range of uncertainty in predicting the performance of the engineered and natural systems. Hence, the approach proposed in this study is to define a 'significant' reduction threshold as a reduction in peak dose by a factor of 10 or more. Such a predicted reduction should be indicative that a true reduction in the peak doses can be realized. The approach for identifying 'reasonable' cost thresholds can be drawn from the ALARA (as low as reasonably achievable) principle. The objective of ALARA is to limit personnel and environmental radiation exposure to the lowest levels achievable commensurate with sound economic and social considerations. Based on the 1997 Program Cost Estimate, and adjusting for a 70,000 MTHM repository, the post-Development-and-Evaluation costs for the repository are estimated to be roughly $13 Billion (FY 1997 dollars). Given this total repository cost, an argument can be made that sound economic and social considerations would mandate that significant reductions in peak dose estimates costing less than $1 Billion (about 8 percent of the total cost) should be considered for possible inclusion in the reference design. Specifying such thresholds is, of necessity, subjective and final designation of these thresholds will require further evaluation. However, this report identified engineered or natural barriers that can be implemented within these thresholds.

The waste isolation strategy for emplacement of spent nuclear fuel and high level radioactive waste (HLW) in the potential repository in the unsaturated zone at Yucca Mountain Nevada relies on a defense-in-depth approach using multiple barriers to limit releases of the radioactive wastes to the accessible environment. The multiple barriers are a combination of natural and engineered barriers which must function together to isolate wastes. Many of the potential barriers that are anticipated to exist or could be engineered were discussed in this study. Performance calculations were done for as many barriers as possible but, in some cases, because of a lack of process models or lack of information on a particular barrier performance calculations were not done. An example of this is for chemical additives to inverts. This study did not perform any performance calculations but laboratory measurements were sponsored where the ability of some minerals to sorb radionuclides was determined. Based on this work and previous work, discussions are provided about what is known about the engineered and natural barriers which are being considered as potential barriers to waste isolation.

A variety of engineered barriers have been postulated and these are summarized in Figure 3-44 in the report. Not all of these design options could be considered in this effort. The potential barriers, both engineered and natural, discussed to some extent in this study are identified in Table 5-1 (reproduced here from an earlier section for convenience). This section provides the conclusions of the study and the recommendations developed. The discussion first focuses on the engineered barriers and then moves away from the engineered barrier system into the natural barriers.

The base case considered in this study is a conservative case. The performance of the various barriers are measured against this base case. The base case was 20.5 kgU/m^2 (83 MTHM/acre), no backfill, an average galvanic protection with 50 percent of the WPs having 75 percent galvanic protection, no cladding, EBS conceptual transport model of drips on WP, ^{237}Np K_D=2.5 cc/gm in zeolites, saturated zone flux (q_{sat}) of 0.31 m/yr and the accessible environment is defined to be at

5 km (5 km from the respective columns center). Variations of a number of these parameters were done for the expected percolation flux of 6.2 mm/yr. The conclusions for the majority of the engineered barriers discussed rely on these recent predictions done in this study. Some of the conclusions for the natural barriers, however, rely on work done earlier, primarily the Performance Allocation Study (CRWMS M&O 1996b). The measure of performance that was used to compare the various options was an absolute performance factor (APF) which indicated the factor that the dose at the accessible environment is reduced from the base case (the base case is defined as having an APF of 1).

Table 5-1 Potential Barriers Considered

Engineered Barriers	Natural Barriers
Cladding	Alluvium/colluvium
Waste Package	PTn
Galvanic Protection	Unsaturated zone transport[1]
Pedestal or WP mount[2]	CHn
Invert with additives	Saturated zone transport
Drip shield	
Backfill	
Richards Barrier Backfill[2]	
Tunnel liner[2]	
Repository configuration	

[1]The unsaturated zone transport includes the CHn
[2]No performance Calculations done

Engineered Barriers

Cladding—Cladding provides a significant amount of performance based on the predictions and it is recommended that credit be taken for this barrier in licensing. Factors of about 5 to 50 depending on time (factor of about 8 reduction in peak dose) in reduction in releases to the accessible environment were estimated. Current estimates of zircaloy performance indicate this material is very long lived and has almost negligible corrosion. However, current uncertainty exists in this performance, in the state of the cladding at emplacement, and the durability of the cladding under static loads. Current information on cladding integrity exists, work on the corrosion performance of zircaloy is being done by the Navy, and tests are underway at Argonne under Summary Account activity TR241GBC to examine transport of radionuclides from broken fuel rods (partial protection). It is judged that obtaining this information and updating the process model will cost about $500,000. This work is underway and should be completed. Based on the results of the work, a determination can be made, in conjunction with licensing, as to the approach needed for licensing.

WP—For at least the first thousand years or so the WP is the primary barrier. Once it starts degrading and releasing radionuclides the performance factors are low or essentially unity. Significant uncertainty currently exists in the corrosion rates of the WPs and corrosion rate information is needed for the thermal and humidity profiles that the packages are to experience. The calculations reported here relied on expert elicitation for the corrosion model and there have also been some material changes which could affect the results. The corrosion models need to be updated for the new material for TSPA 1998.

Galvanic Protection—Galvanic protection is expected to occur and was considered in the base case as occurring for 50 percent of the WPs. Significant performance was predicted to occur if a larger percentage of the WPs have galvanic protection with reduction in doses (APFs) of between 5 to 70. The amount of galvanic protection needs to be established. Some short term tests at LLNL (Summary Account activity TR251GB7) have been started. Based on the results of these tests the process models should be updated and used in TSPA 1998. The cost estimates for this work and some longer term testing (long duration tests over a range of temperatures; TR251GBB and TR251FBB) have been estimated to be about $1.4 million. The longer term tests should be done and a determination made as to how precise the contact tolerance (spacing) between the two layers must be to get consistent, predictable performance. Once this is known, then Waste Package Design should estimate the quality controls that must be placed on manufacturing and the costs for ensuring these quality controls.

Drip Shields—A drip shield which lasts the duration of the period of interest (regulatory period or longer if those times are of concern) was predicted to have significant performance with an APF of about 30. Previous work (CRWMS M&O 1996a) found that doses were delayed and reduced only while the drip shield was intact. The drip shield requires that the radionuclides diffuse away from the WP before they are carried away. Once the drip shield is gone the doses essentially reach values comparable to the cases with no drip shield. The longevity and durability of a candidate drip shield is currently uncertain. It is unlikely that drip shields will achieve the lifetimes required. As part of corrosion work at LLNL (such as Summary Account activities TR256GB2 and TR251GB6), candidate drip shield materials are being tested. In addition to this an assessment is needed as to where the drip shield should be located and how durable it would be. For example if it is incorporated into the WP, which is an option, then such issues as effects of hydrogen gas on titanium need to be addressed. If a drip shield is to be placed over the WP then other issues needed to be addressed such as whether or not the drip shield needs to be protected by something like backfill. Cost estimates for material costs, fabrication, and emplacement of drip shields were estimated at about $400 million. These costs did not include any backfill costs that might be required based on the mode of installation.

Backfill—The backfill calculations done considered only the effect of backfill on the temperature and relative humidity of the WP. Thus, performance assessment did not calculate the effects of backfill as a method to reduce drips on the WP. With these caveats in mind, the results showed that backfill provided little or no performance benefit. At the higher fluxes essentially no performance difference is noted between the case with backfill and the base case without backfill. The cost of backfill is about $160 million to $360 million and there are operational considerations. Thus, backfill is not recommended at this time. However, work is needed to improve the process models for backfill and possibly some testing, currently not planned, is needed to better understand the performance. Additionally, if backfill is needed to protect a drip shield, then the option may need to be retained. More recent three-dimensional calculations with backfill have provided preliminary indications that for some configurations there may be improvements in relative humidity and temperature profiles that should be investigated in the Design Basis Modeling Study.

A Richards barrier backfill was also considered. This backfill acts somewhat as a drip shield. In this respect it has similar performance to the drip shield. Indications are that emplacing a Richards barrier is probably not possible with today's technology and within the context of the current design. This was the position taken in the Engineered Barrier Study (CRWMS M&O 1996a) based on an assessment by Subsurface Design and this was reaffirmed by them in this study.

Tunnel Liners and Inverts—Performance calculations were not done in this study for either tunnel liners or inverts. Some performance calculations have indicated that there is a potential to degrade the performance of the natural barriers as a result of the alkaline plume (CRWMS M&O 1996h). In addition, the long-term survivability/durability of these liners in the high temperature conditions has not been adequately addressed. If concrete is to be used in the drifts the pH needs to be constrained (CRWMS M&O 1996h) and it must withstand 200°C for about 100 years (CRWMS M&O 1996d) while performing its load bearing mission.

Some rough estimates were made as to the total amount of apatite that would be needed in the entire potential repository to absorb all of the ^{237}Np and the amount of envirostone needed to absorb all of the ^{99}Tc. The rough estimates indicate that about 1.6 x 10^7 metric tonnes of envirostone and 1.1 x 10^5 metric tonnes of apatite would be needed. Estimates to absorb all of the Tc indicate a layer of envirostone about 130 m deep beneath each WP which is not feasible. However, for Np about 0.6 m of apatite would be needed which is possible. However, it may be that substantial performance may be obtained without needing to absorb all of the particular radionuclide. These two materials, apatite and envirostone, offer significant promise and should be investigated. Subsurface design should evaluate how to emplace such materials and the impact on design.

Repository Configuration—Some of the configuration changes that have been examined in past studies were such things as line loading, where the packages are moved closer together. This has some potential for small improvements in performance (up to a factor of 2, although calculations were not done at the higher fluxes). However, the localized temperatures are significantly higher and there is some operational concern. More investigation is needed on this to determine the effects of temperatures and the higher fluxes and this could be done in the Design Basis Modeling Study.

Modifying the thermal loading is another configuration change that has been examined. The calculations done in this study at 6.2 kgU/m^2 (25 MTHM/acre) showed significant reduction in dose to the accessible environment over the higher thermal loading base case. Three-dimensional calculations, which considered the localized heat output of the large packages estimated about a factor of ten reduction in dose over the higher thermal loading. This was due primarily to the fact that at the higher fluxes the rock re-wets fairly quickly and at the higher thermal load the WP is hotter so that higher corrosion rates are experienced. With this potential improvement in performance it is recommended that an alternate thermal load of low thermal loading be considered in addition to the current design. Corrosion models of WPs also need to be examined and updated, particularly if new materials are being considered, since this is an important assumption that needs to be verified.

Natural Barriers

Alluvium/Colluvium—The presence of the alluvium/colluvium was not modeled explicitly. However, it is implicitly modeled because estimates of liquid infiltration into the Tiva Canyon, used in the site scale UZ flow model, are based on alluvium cover (Flint and Flint 1996). However, there is recent evidence (Flint and Flint 1995) that this medium has a relatively large storage capacity to retain moisture, which generally allows removal of this moisture by persistent evapotranspiration instead of allowing transport downward into the mountain.

PTn—Sensitivity to uncertainty in PTn properties was not considered, however a best estimate of PTn properties formed the basis for the fluxes and velocities derived from the site scale UZ hydrology model and the drift scale thermohydrology model. There are indications that the PTn significantly contributes to reducing the liquid percolation flux. This reduction was not included in this work. Preliminary calculations indicate that the design basis thermal load will increase fracture size in the PTn by about a factor of two. This study recommends that a better understanding of the PTn performance be developed through sensitivity studies of the effect that high heat loads have on the barrier. If fast paths develop through the barrier performance may be affected. If performance sensitivity is predicted then testing must be developed to validate the effect or the thermal impacts reduced.

Unsaturated Zone Transport—The calculations of unsaturated zone transport that were done in the Performance Allocation Study include transport in the CHn. The calculations indicated that significant performance was achieved with factors of about 30 reduction in dose at the accessible environment. New information has determined that the sorption coefficients have changed significantly over what was used in that study (CRWMS M&O 1996b). The zeolite distributions provided in this study need to be included in future TSPA calculations. This information should be updated in TSPA 1998 and an assessment made of the performance of the unsaturated zone transport. As information becomes available on the amount of fracture flow and matrix flow that occurs this information needs to be added to the TSPA calculations. A key element in the transport is the solubilities of key radionuclides. These solubilities are uncertain and need to be better established.

CHn—The major sorber of radionuclides in the CHn are the zeolitized minerals. The performance allocation study did determine the APF of the CHn layer to be about 12 (included in the factor of 30 in UZ transport) although it can be substantially higher at earlier times (10,000 years). The recent calculations found that varying the distribution coefficient did have some impact on performance. Zeolite concentrations were found to vary across the repository area from about 40 to 140 m thick. These concentrations of zeolites should be considered in TSPA 1998 calculations.

Saturated Zone Transport—Transport in the saturated zone was also shown to have a significant performance with the saturated zone responsible for reducing doses at the accessible environment by about a factor of 70. This performance is based on the assumptions that were used such as a mixing depth of 50 m. Calculations done in this study found that the flow velocity can change the dose observed at the accessible environment by possibly as much as a factor of ten. Thus, it is important to better understand the mixing depth, flow velocity, and dispersivity in the saturated zone. Tests are underway in the C-wells and the results of these tests should be incorporated into the process models. However, it will take more information over a longer time than those tests to be able to establish the mixing depth and flow velocity. Those tests should be conducted.

Preliminary estimates indicate that the temperature of the saturated zone will increase significantly for the thermal load being considered now. This increase in temperature will alter the mineralogy, porosity, and flow velocity in the saturated zone, potentially causing favorable or unfavorable changes in performance. The effects of heat on the saturated zone need to be determined.

RECOMMENDATIONS

The performance predictions determined that certain engineering concepts have potential for significant improvement in performance at reasonable cost. Specifically, cladding, galvanic protection, and a long lived drip shield are the engineering barriers which could, if they function as predicted, produce the reduction in dose. It should be noted that all of these cases assume that the natural barriers are functioning as predicted and even without the engineering enhancements will provide over three orders of magnitude reduction in dose (reduction in dose due to the unsaturated zone transport times the saturated zone transport reduction). Based on the synopsis of calculations in this study and the additional performance calculations at the higher percolation flux of 6.2 mm/yr, the following recommendations are offered:

Potential Engineered Barriers

- The performance predictions indicate that zircaloy cladding of the spent nuclear fuel assemblies may provide a significant reduction (about a factor of 10) in peak dose. Based on this study, it is recommended that the Project pursue a course of action that, if successful, will allow taking performance credit for cladding. Licensing issues such as initial integrity of the cladding and subsequent degradation modes needs to be addressed. Performance Assessment should evaluate available measurements of cladding performance done by Pacific Northwest National Laboratories, review the zircaloy corrosion model being developed by the Navy and upgrade/update the Yucca Mountain Project cladding process model. Ongoing materials tests (TR241GBC) on the effects of drips and relative humidity on spent nuclear fuel segments need to be completed and evaluated by Performance Assessment for inclusion in the process model. The updated cladding model should be used in Total System Performance Assessment-Viability Assessment (1998).

- Galvanic protection is another engineered barrier that may produce significant reduction (more than a factor of 10) in dose. However, there are currently significant uncertainties in the number of waste packages that would be protected by galvanic protection and the percent of the corrosion allowance barrier which would have to corrode before the inner corrosion resistant barrier starts to degrade. There are some laboratory tests ongoing (TR251GB7) and some longer term testing planned (long range plan; TR251GBB and TR251FBB) which would examine galvanic protection and potential crevice corrosion. These laboratory tests should be completed and the information used to update the process models for Total System Performance Assessment-Viability Assessment. The longer term tests should be conducted and Performance Assessment should incorporate this information in their models as it becomes available. Other alternatives that decrease the WP degradation rate should be examined as well including developing improved models for degradation of the corrosion resistant barrier and/or to choose a different corrosion resistant material which substantially increases containment lifetime.

- Under the conditions of high flux the performance predictions indicate that drip shields that survive for a long time (the regulatory period or longer) have potential for producing significant reduction in dose. A drip shield will reduce doses during it's lifetime but when it is gone doses return to levels approaching the base case with no drip shield. Long term reduction in dose from the base case at times after the drip shield is gone will required drip shield lifetimes well in excess of 20,000 years. It is unlikely that any man-made materials can be shown to have these very long lifetimes. Some work (TR251GB6) on materials evaluation of titanium and ceramics has been initiated. The testing work, including testing of other candidate material, should be completed and a determination made by Performance Assessment, in coordination with Regulatory and Licensing, as to what is needed for licensing. Based on experiment a range of drip shield life times needs to be used in future calculations.

- An alternative to a drip shield that may offer some merit is a third barrier to the waste package (e.g., a ceramic coating or other). Such an alternative could be evaluated by Waste Package Development with an assessment of drip shields. Evaluations such as constructability and operability to include the increase in waste package weight and thermal effects should be considered in addition to performance.

- Do not consider backfill in the current design concept for the purpose of reducing relative humidity at the WP but do not preclude the use of backfill. The reason not to preclude backfill at this point is that it may be needed to ensure survivability of a drip shield or a ceramic coating on the WP. Performance Assessment should develop a process model to evaluate the evaporative properties of backfill and, if these are found to improve performance, backfill can be reconsidered.

- It is recommended that the testing of cementitious materials planned (Long Range Plan, TR3C5GBB) be completed and evaluated by Performance Assessment and the performance of tunnel liners during heating be examined. If tunnel liners and/or concrete inverts are needed then the pH of the concrete needs to be constrained and it must withstand 200°C for about 100 years while performing its load bearing mission. Once the impacts on performance and on design are known, then a determination is needed as to what pH is acceptable.

- It is recommended that additional work be done on the potential use of apatite as an additive for inverts and backfill. Some additional work on the reversibility of the sorption process should be done and work should also be done to reduce the uncertainties identified in Section 3, in particular the ability of apatite to sorb Np in the presence of other radionuclides. A sensitivity analysis on how much apatite is needed to provide an appreciable reduction in dose should be done. Additionally, the impacts of apatite on the engineered and natural barriers should be examined. Finally, subsurface design should examine the aspects of emplacing this material and the effect of heat on the materials.

Envirostone, another material evaluated, was found to not be a practical addition because too large a quantity is required.

- The performance aspects of line loading including inputs of significantly higher local temperatures, should be examined in the Design Basis Modeling effort and conclusions reached as to whether it provides any appreciable advantage in performance.

- It is recommended that an alternate, low thermal loading of 6.2 to 8.9 kgU/m^2 (25 to 36 MTHM/acre) be carried for LA in addition to the current high thermal loading design. In the potentially higher moisture flux that may exist the lower thermal load may significantly increase performance. The low loading must be established as a viable alternative by producing some limited designs, including it in TSPA cases, developing plans to characterize additional area, and providing cost estimates for this case. These plans would not need to be implemented until a decision is reached to change to a low thermal load.

- Solubilities for such key radionuclides as Np and Tc need to be resolved for the most likely compounds.

Natural Barriers

- The predictions indicate that the CHn may provide a significant amount of performance. The new zeolite conceptualization in the three-dimensional geologic model should be incorporated in the Reference Information Base and included in the performance assessment models for Total System Performance Assessment-Viability Assessment. The proportion of fracture flow and matrix flow in the unsaturated zone, including the CHn, should be established. To do this numerous niche tests are underway or planned. Such Summary Account activities as the fracture-matrix interaction tests (TR33124GB5) and transport studies like TR34141FB5 to name a few should be done and used to update the process models.

- The saturated zone provides a significant amount of performance based on the performance predictions. To improve the performance predictions for licensing will require improved estimates of mixing depth, flow velocity, and dispersion properties. The measurements being taken in the C-Well tests should be evaluated. Additional tracer study tests may be needed to obtain the requisite information.

- For the current design concept, the effects of heat on the PTn, CHn, and saturated zone performance need to be understood. The thermomechanical effects on the barriers and the effect resulting from mineral redistribution, dehydration, and porosity changes need to be estimated and a determination made as to whether or not these changes will affect performance.

A factor of 10 reduction in peak dose was established as an appropriate threshold at which to evaluate performance. The evaluations of the natural barriers show that those evaluated (unsaturated zone transport, zeolite in CHn, and saturated zone transport) all were predicted to produce reductions in peak dose of greater than a factor of 10. In fact, the natural barriers provide significant performance, based on the performance calculations done to date (CRWMS M&O 1996b), and both natural and engineered systems will be needed to meet performance requirements. In the case of the engineered barriers the predictions indicate that those that have potential to reduce the peak dose by a factor of about 10 (the threshold for significant reduction) were zircaloy cladding, galvanic protection, and drip shields. The first two barriers exist or may exist without any additional effort and they would require only small to modest expenditures to establish whether or not licensing credit can be obtained for those barriers. Of course, based on further NRC interactions, more work and therefore more cost may be incurred to do what is necessary to make a licensing case for those barriers. At this time these barriers appear to be attainable at "reasonable" cost (less than $1 Billion). In the case of a drip shield if evaluations find a material that is durable and will survive and function for extended periods of time (at least the regulatory period) then the cost for these drip shields appears "reasonable" at about $400 million to produce enough drip shields.

6. REFERENCES

6.1 DOCUMENTS CITED

Bodvarsson, G.S.; Bandurraga, T.M.; Chen, G; Haukwa, C.; and Wu, Y.S. 1996. *Development and Calibration of the Three-Dimensional Site-Scale Unsaturated Zone Model of Yucca Mountain, Nevada*. Chapter 8. Berkeley, California: Lawrence Berkeley National Laboratory, Earth Sciences Division.

Buscheck, T.A.; Nitao, J.J.; and Saterlie, S.F. 1994. "Evaluation of Thermohydrological Performance in Support of the Thermal Loading Systems Study." *Proceedings of the International High-Level Radioactive Waste Management Conference*. 1994:592-610. Las Vegas, Nevada: Author.

Chin, B.A. and Gilbert, E.R. 1989. "Prediction of Maximum Allowable Temperatures for Dry Storage of Zircaloy-Clad Spent Fuel in Inert Atmosphere." *Nuclear Technology*. 85:57.

Chipera, S.J.; Carter-Krogh, K.; Vaniman, D.T.; Bish, D.L.; and Carey, J.W. 1997. *Preliminary Three-Dimensional Mineralogic Model of Yucca Mountain, Nevada*. Los Alamos Milestone No. SP321AM4. Los Alamos, New Mexico: Los Alamos National Laboratory.

Conca, J.L and Wright, J.V. 1996. *NESTT Letter Report Subcontract No. 215BA0016-3Y*. Los Alamos National Laboratory Subcontract No. 215BA0016-3Y. Richland, Washington: Northwest Environmental Services, Testing and Training.

Conca, J.L. and Wright, J.V. 1992. "Diffusion and Flow in Gravel, Soil, and Whole Rock" Applied Hydrogeology, Vol 1, pp 5-24.

CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1993. *Site Characterization Plan Thermal Goals Re-evaluation*. B00000000-01717-5705-00005, Rev. 00, Las Vegas, Nevada: Author.

CRWMS M&O 1995a. *Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository*. B00000000-01717-2200-00136 REV 01. Las Vegas, Nevada: Author.

CRWMS M&O 1995b. *Systems Study of Options for Characterizing the Calico Hills Nonwelded Hydrogeologic Unit at Yucca Mountain, Nevada*. B00000000-01717-5705-00021 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1995c. *Thermal Loading Study for FY 1995*. B00000000-01717-5705-00016 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1996a. *Engineered Barrier System Performance Requirements Systems Study Report*. B00000000-01717-5705-00001 REV 02. Las Vegas, Nevada: Author.

CRWMS M&O 1996b. *Description of Performance Allocation.* B00000000-01717-2200-00177 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1996c. *Preliminary Mined Geologic Disposal System Concept of Operations.* B00000000-01717-4200-00004 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1996d. *Thermal Loading Study for FY 1996.* B00000000-01717-5705-00044 REV 1. Las Vegas, Nevada: Author.

CRWMS M&O 1996e. *Mined Geologic Disposal System Advanced Conceptual Design Report.* B00000000-01717-5705-00027 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1996f. *Waste Package Materials Selection Analysis.* BBA000000-01717-2000-00020 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1996g. *Waste Package Fabrication Process Report.* BBA000000-01717-2500-00010 REV 01. Las Vegas, Nevada: Author.

CRWMS M&O 1996i. *Controlled Design Assumptions Document.* B00000000-01717-4600-00032 REV 04. Las Vegas, Nevada: Author.

CRWMS M&O 1997a. *Technical Document Preparation Plan for the Waste Isolation Requirements Study.* B00000000-01717-4600-00080 REV 00. Las Vegas, Nevada: Author.

CRWMS M&O 1997b. *Program Cost Estimate Report.* WAD 97-50, Activity No. PI925D7599. Vienna, Virginia, February 28, 1997.

DOE (U.S. Department of Energy) 1988. *Site Characterization Plan, Yucca Mountain Site.* DOE/RW-0199. Las Vegas, Nevada: Author.

DOE 1996a. *Quality Assurance Requirements and Description.* DOE/RW-0333P, Revision 6. Las Vegas, Nevada: Author.

DOE 1996b. *Civilian Radioactive Waste Management System Requirements Document.* DOE/RW-0406, Revision 3. Las Vegas, Nevada: Author.

Fabryka-Martin, J.T.; Dixon, P.R.; Levy, S.; Liu, B.; Turin, H.J.; and Wolsberg, A.V. 1996. *Summary Report of Chlorine-36 Studies: Systematic Sampling for Chlorine-36 in the Exploratory Studies Facility.* M&O Letter Report 3783AD. Los Alamos, New Mexico: Los Alamos National Laboratory.

Flint, L.E. and Flint, A.L. 1995. *Shallow Infiltration Processes at Yucca Mountain -- Neutron Logging Data 1984-93.* U.S. Geological Survey Water-Resources Investigations Report 95-4035. Denver, Colorado: U. S. Geological Survey.

Flint, A. Hevesi, J. and Flint, L. 1996. *Conceptual and numerical Model of Infiltration for the Yucca Mountain Area.* U.S. Geological Survey Milestone Report 3GUT623M. Denver, Colorado.

Fridrich, C.J.; Dudley, Jr., W.W.; and Stuckless, J.S. 1994. "Hydrogeologic Analysis of the Saturated-zone Ground-water System under Yucca Mountain, Nevada." *Journal of Hydrology,* 154:133-168.

Golder Associates, Inc. 1994. *RIP Performance Assessment and Strategy Evaluation Model: Theory Manual and User's Guide, Version 3.20.* Redmond, Washington: Golder Associates.

Golder Associates, Inc. 1995. *Verification Report for the Repository Integration Program (RIP).* Prepared for the WIPP Technical Assistance Contractor.

Hevesi, J.A. and Flint, A.L. 1996. *Regional Estimation of Precipitation and Recharge in Southern Nevada.* U.S. Geological Survey Water-Resources Investigations Report.

Ho, C.K.; Francis, N.D.; Arnold, B.W.; Xiang, Y.; McKenna, S.A.; Mishra, S.; Barr, G.E.; Altman, S.J.; Yang, X.H.; and Eaton, R.R. 1996. *Thermo-Hydrologic Modeling of the Potential Repository at Yucca Mountain Including the Effects of Heterogeneities and Alternative Conceptual Models of Fractures Porous Media.* Level 3 Milestone T6536 WA-210, Rev. 00.

Long, A. and Childs, S.W. 1993. "Rainfall and Net Infiltration Probabilities for Future Climate Conditions at Yucca Mountain." *Proceedings of the Fourth Annual High Level Radioactive Waste Management Conference, Las Vegas Nevada. April 26-30, 1993.* New York, New York: American Society of Civil Engineers; and La Grange Park, Illinois: American Nuclear Society Inc.

Luckey, R.R.; Tucci, P.; Faunt, C.C.; Erin, E.M.; Steinkampf, W.C.; D'Agnese, F.A.; and Patterson, G.L. 1996. *Status of Understanding of the Saturated-zone Ground-water Flow System at Yucca Mountain, Nevada as of 1995.* Denver, Colorado: U.S. Geological Survey Water Resources Investigations Report.

Meijer, A. 1990. *Yucca Mountain Project Far-Field Sorption Studies and Data Needs.* Report LA-1167 1-MS. Los Alamos, New Mexico: Los Alamos National Laboratory.

Montazer, P. and Wilson, W.E. 1984. *Conceptual Hydrologic Model of Flow in the Unsaturated Zone, Yucca Mountain, Nevada.* USGS-WRI-84-4345. Water Resources Investigations Report. Lakewood, Colorado: U.S. Geological Survey.

NAS (National Academy of Sciences-National Research Council) 1995. *Technical Bases for Yucca Mountain Standards.* Washington, D. C.: National Academy Press.

Nitao, J.J. 1993. "The NUFT Code for Modeling Nonisothermal, Multiphase, Multicomponent Flow and Transport in Porous Media." (3), 313. *EOS Supplement 74.*

Nitao, J.J. 1996. *User's Manual for the USNT Module of the NUFT Code, Version 1.0.* Livermore, California: Lawrence Livermore National Laboratory, Environmental Restoration Division.

Ogard, A.E. and Kerrisk, J.F. 1984. *Groundwater Chemistry Along the Flow Path between a Proposed Repository Site and the Accessible Environment.* Report LA-10188-MS, Los Alamos, New Mexico: Los Alamos National Laboratory. (NNA.870406.0021)

Peehs, M. and Fleisch, J. 1986. "LWR Spent Fuel Storage Behavior." 137:190. *Journal of Nuclear Materials.*

Pruess, K. 1991. *TOUGH2 -A General Purpose Numerical Simulator for Multiphase Fluid and Heat Flow.* LBL-29400. Berkeley, California: Lawrence Berkeley Laboratory.

Pruess, K. and Tsang, Y. 1994. *Thermal Modeling for a Potential High-Level Nuclear Waste Repository at Yucca Mountain, Nevada.* LBL-35381. Berkeley, California: Lawrence Berkeley Laboratory.

Robinson, B.A. 1994. "A Strategy for Validating a Conceptual Model for Radionuclide Migration in the Saturated Zone Beneath Yucca Mountain." *Radioactive Waste Management Environmental Restoration,* 19:73-96.

Robinson, B.A.; Wolfsberg, A.V.; Viswanathan, H.S.; Gable, C.W.; Zyvoloski, G.A.; and Turin, H. J. 1996. *Modeling of Flow, Radionuclide Migration, and Environmental Isotope Distributions at Yucca Mountain.* Draft Report, Milestone 3672. Los Alamos, New Mexico: Los Alamos National Laboratory.

Wilder, D. 1996. *Near-Field and Altered-Zone Environment Report.* UCRL-LR-124998. Livermore, California: Lawrence Livermore National Laboratory.

Yang, I.C.; Rattray, G.W.; and Yu, P. 1996. *Interpretation of Chemical and Isotopic Data from Boreholes in the Unsaturated Zone at Yucca Mountain.* USGS Water Resources Investigations Report, 96-4058.

YMP (Yucca Mountain Site Characterization Project) 1994a. *Repository Design Requirements Document.* YMP/CM-0023. Las Vegas, Nevada.

YMP 1994b. *Engineered Barrier Design Requirements Document.* YMP/CM-0024. Las Vegas, Nevada.

YMP 1995. *Reference Information Base.* YMP/93-02. Las Vegas, Nevada.

YMP 1996. *Highlights of the U.S. Department of Energy's Updated Waste Containment and Isolation Strategy.* YMP/96-01, Rev 0. Las Vegas, Nevada.

YMP 1997. *Q-List.* YMP/90-55Q, Rev. 4. Las Vegas, Nevada.

Wittwer, C.; Chen, G.; Bodvarsson, G.S.; Chornack, M; Flint, A.; Flint, L.; Kwicklis, E.; and Spengler, R. 1995. *Development of the LBL-USGS Three Dimensional Site-Scale Groundwater Flow Model of Yucca Mountain, Nevada*. LBL-37356/UC-814. Berkeley, California: Lawrence Berkeley Laboratory.

Zyvoloski, G.; Dash, Z.; and Kelkar, S. 1995. *FEHM 1.0, Finite Element Heat and Mass Transfer Code*. LA-12062-MS, Rev. 1. Los Alamos, New Mexico: Los Alamos National Laboratory.

6.2 STANDARDS AND REGULATIONS

10 CFR 60. Title 10, *Energy*, Part 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*. Washington D.C.: U.S. Government Printing Office.

INTENTIONALLY LEFT BLANK

7. ACRONYMS

AE	Accessible Environment
ALARA	As low as reasonably achievable
AML	Area Mass Loading
APF	Absolute Performance Factor
BDCF	Biosphere Dose Conversion Factor
BWR	Boiling Water Reactor
CFu	Crater Flat undifferentiated unit
CHn	Calico Hills nonwelded unit
CRWMS	Civilian Radioactive Waste Management System
DHLW	Defense High-Level Waste
DOE	U. S. Department of Energy
ECM	Equivalent Continuum Model
ESF	Exploratory Studies Facility
HLW	High-Level Waste
LANL	Los Alamos National Laboratory
LLNL	Lawrence Livermore National Laboratory
M&O	Management and Operating Contractor
MGDS	Mined Geologic Disposal System
MTHM	Metric tonnes heavy metal
NRC	Nuclear Regulatory Commission
PTn	Paintbrush Tuff nonwelded unit
PWR	Pressurized Water Reactor
QA	Quality Assurance
RH	Relative Humidity
RIP	Repository Integration Program
SNF	Spent Nuclear Fuel
SZ	Saturated Zone
TCw	Tiva Canyon welded unit
TSPA	Total System Performance Assessment
TSw	Topopah Spring welded unit
TSw2	Topopah Spring welded unit 2
UZ	Unsaturated Zone
VA	Viability Assessment
WCIS	Waste Containment and Isolation Strategy
WP	Waste Package
YMP	Yucca Mountain Site Characterization Project

INTENTIONALLY LEFT BLANK

APPENDIX A
ENGINEERED BARRIERS

APPENDIX A
ENGINEERED BARRIERS

A.1 ENGINEERED BARRIERS

The Code of Federal Regulations requires that for the potential repository the ".....design of any engineered barriers....shall contribute to the containment and isolation of radionuclides" [10 CFR 60.133(a)], and "engineered barriers shall be designed to assist the geologic setting in meeting the performance objectives for the period following permanent closure" [10 CFR 60.133(h)]. Thus, the engineered barriers must work together with the natural system to contain waste. The Project has some flexibility to include or not include some engineered barrier concepts, backfill being a prime example. This section presents the facts known about the performance of some candidate engineered barriers and provides some limited updated performance calculations. The first two subsections document Engineered Barrier System Performance Requirements Systems Study.

The WCIS (YMP 1996) was developed to assist the YMP in prioritizing testing and analysis activities to focus on the most important remaining issues regarding postclosure safety. The WCIS is designed to help resolve uncertainty in the processes and parameters of greatest significance to long-term performance and focuses on those attributes of the repository that are thought to have the highest probability of being reasonably bounded prior to the Viability Assessment.

The WCIS has identified five major system "attributes" that are the most important with respect to performance of the natural and engineered barriers:

- Rate of water seepage into the repository
- Waste-package lifetime (containment)
- Rate of release (mobilization) of radionuclides from breached WPs
- Radionuclide transport through engineered and natural barriers
- Dilution in the saturated zone below the repository.

To address the performance effects of these five major system attributes, the TSPA analyses in this study have considered the effect of uncertainty in a number of the most critical system parameters, both design (engineered-system) parameters and natural-system parameters. These TSPA analyses are be discussed in more detail below.

Various types of engineered barriers have been considered. Most SNF assemblies have a zircaloy cladding around the fuel pellets. Although the cladding barrier is relatively thin, zircaloy is a very durable material that can provide a barrier to radionuclide transport. Up to about one percent percent of the SNF assemblies have stainless steel claddings and the defense HLW has no cladding but is contained in a glass log within a stainless steel canister. Work has been done to develop cladding degradation models as a result of creep rupture (Chin and Gilbert 1989, Peehs and Fleish 1986). A process model was developed and preliminary performance calculations were done in two recent studies (CRWMS M&O 1996b and 1996d) that showed that cladding may provide a significant barrier to radionuclide transport. One study determined that cladding performance is sensitive to temperature and recommended that a temperature limit of 350°C be retained for cladding. Durability of cladding under static loads was not adequately addressed, however. The results of the performance calculations are discussed below.

The current design of the WP has two barriers with the outer barrier being a corrosion allowance material of ASTM A 516 steel and an inner barrier of a corrosion resistant nickel alloy, ASTM B 443 (Alloy 625) (CRWMS M&O 1996f). The PA calculations used corrosion curves for Alloy 825 rather than Alloy 625 since the corrosion curves for the new material were not yet available. The thickness of the outer corrosion allowance barrier is 100 mm and the Alloy 625 inner barrier thickness is 20 mm (CRWMS M&O 1996g). The assumptions used for WP degradation are that humid air and pitting corrosion of the outer barrier occurs at a relative humidity threshold of between 65 percent and 75 percent. Aqueous corrosion of the outer barrier occurs when relative humidities exceed 85 percent to 95 percent. The inner barrier is subject to aqueous localized corrosion. Actually, the relative humidity threshold for corrosion initiation of the outer barrier was selected for each WP at random from the range of 65 to 75 percent. For the inner barrier 20 percent was added to the selected number. The corrosion models for these calculations are reported in TSPA 1995 (CRWMS M&O 1995a). It is assumed that corrosion initiation does not begin until the WP surface temperature drops below 100°C. The TSPA calculations (CRWMS M&O 1995a and 1995b) of WP degradation indicate that for the range of infiltration rates from 0.05 to 0.3 mm/yr some of the WPs start to fail at about 2,000 years and this time is not too sensitive to infiltration rate (over the range of TSPA-95 values that does not include current expected values). Some estimates of the performance contribution of the WP were developed in the performance allocation study but few calculations have been done in which the analysis was run with and without the WP to be able to estimate the contribution of that element of the engineered barriers. These calculations are discussed below. (Note that WP degradation simulations at higher infiltration rates are presented in Section 3).

The double-walled WP will result in some degree of galvanic protection of the inner barrier once the outer barrier is breached due to the formation of a galvanic couple between the outer barrier and the inner barrier. The degree of galvanic protection is uncertain until results of tests underway are evaluated. However, using expert elicitation, estimates of the amount of galvanic protection were done and these were used to estimate the reduction, if any, in releases to the accessible environment (CRWMS M&O 1995a, CRWMS M&O 1996g, CRWMS M&O 1996b). These results indicate that significant improvement in performance may be achieved with galvanic protection and the specifics of these analyses are described below.

The WPs will be mounted on a pedestal or similar type of holder to keep them centered in the drift and elevated from the emplacement drift floor and away from any potential liquid water that might collect on the floor. No performance calculations have been done for these mounts as yet. In the TSPA calculations it is assumed that the WP is resting on the floor of the emplacement drift. In addition, the floor of the emplacement drift is assumed to have an invert composed of crushed tuff or other material such as concrete (CRWMS M&O 1996e). In some cases this invert could have minerals or chemicals added to the mix which might provide some sorptivity of radionuclides. Such compounds as apatite (CRWMS M&O 1996a) or envirostone have been suggested as additives. Preliminary scoping estimates of performance were done in the engineered barrier study. Additional work was done to determine just how much additive of the types above would be needed to provide an appreciable increase in performance, and this report provides the results of that work.

Drip shields over each WP have been suggested as a possible approach to divert water. These drip shields might be used with or without backfill. The long-term survivability of these drip shields has not been evaluated. Estimates of performance for cases with drip shields was done in the *Engineered Barrier System Performance Requirements Systems Study* (CRWMS M&O 1996a).

Backfill is a concept that has been suggested as a possible component to enhance performance. The *Engineered Barrier System Performance Requirements Systems Study* primarily examined system performance using backfill (CRWMS M&O 1996a). There are several issues that were addressed in the study. Specifically, using backfill will result in substantially higher WP temperatures, which can result in internal temperatures exceeding 350°C if the backfill does not have a suitable thermal conductivity or backfill is emplaced early. Thus, depending on the time that is spent at these higher temperatures, the cladding could be degraded and one may be trading one barrier for another. It should be noted that thermal calculations (CRWMS M&O 1996a) found that most WPs did not exceed the cladding criteria for backfilling at 100 years. Emplacing backfill after the WPs are emplaced was determined to have operational implications and there are increased costs with using backfill. TSPA calculations were performed and these are reported below. The study cited above determined that, although there was estimated to be an order of magnitude improvement in performance with backfill the base case at low fluxes (0.5-2 mm/yr) had significant performance margin without backfill. Thus, it was determined not to include backfill at this time but, because of the uncertainties in the calculations, backfill should not be precluded. These calculations were done at a lower infiltration rate than what is currently believed to exist in Yucca Mountain. Thus, calculations were done in the present study at the higher fluxes anticipated to exist (see Section 3).

Another type of backfill considered in the Engineered Barrier System Study was a Richards Barrier, which is a multi-layer backfill with layers of different porosity. This has an analog in Japanese burial mounds (Conca and Wright 1992). However, emplacing such a multi-layer backfill does not appear to be feasible at this time (CRWMS M&O 1996a), so it was not considered in this report.

The current subsurface design concept is considering using concrete tunnel liners to maintain tunnel stability through the 100 year operational phase. A liner can alter the hydrology but no calculations have been done to evaluate this. Some calculations have been done concerning potential impact of cementitious materials in the emplacement drifts and the performance implications will be briefly discussed in this report.

The spacing of drifts and WPs can have implications for waste isolation. These repository configuration issues have been investigated to some extent in previous studies (CRWMS M&O 1996a and 1996d) where the effects of separating WPs by only 1 m or less were examined. The thermal loading or density of the WPs in a given area can be varied and may affect performance. This line load concept did offer some benefit to moderating package-to-package heat variations but there were operational considerations as a result of the higher temperatures. These concepts continue to be investigated and some performance calculations are discussed below.

This appendix documents previous work that was done in the Engineering Barrier System Performance Requirements System Study and in the Performance Allocation Study. Those evaluations were done at moisture fluxes of 0.3 to 2 mm/year which were thought to be the most likely values. Section 4.4 provides the results of recent performance calculations done for an expected moisture flux of 6.2 mm/year which is based on the range of moisture flux values of 1 to 10 mm/yr that is now expected to exist in the mountain. Possible higher fluxes that might occur during periods of wetter climates were not considered. Additionally, some new work by Los Alamos National Laboratory on absorption of radionuclide by potential additives for inverts was done for this study and is included in Section 4.4.

A.2 ENGINEERED BARRIER PERFORMANCE

Most of the calculations showing performance were extracted from the *Engineered Barrier System Performance Requirements System Study* (CRWMS M&O 1996a). Where the calculations come from other sources those sources are identified. The Engineered Barrier System Performance Requirements System Study report was prepared in accordance with QAP-3-5, *Development of Technical Documents*. However, most of the calculations are scoping analyses and the codes have not been qualified.

The analyses calculated the WP environment as a function of time used either FEHM for (TSPA-95) or NUFT for the thermohydrologic process model. These results were then used as input into evaluation of WP failure as a function of time. The radionuclide releases and subsequent transport were calculated with the Repository Integration Program, RIP, (Golder Associates 1994).

The majority of the assumptions used in the previous section were used in these calculations. The analytic models, however, were somewhat different. The thermohydrologic calculations were done for percolation fluxes of 0.3 and 1.25 mm/yr. However, those calculations used rock properties that had been developed with site scale models using low infiltration rates (0.3 mm/yr) (Pruess and Tsang 1994). The RIP code then calculate transport using randomly sampled percolation fluxes in the range of 0.5 to 2.0 mm/yr for each of the discrete model runs that were used to develop the cumulative probability distributions. The stratigraphy used is based on a model developed by Wittwer et al. (1995) and the thermal properties are those in Version 4 of the RIB (YMP 1995).

The calculations in this section were all compared against a base case, which did not have any engineered barriers except a WP. A detailed description of the underground layouts and the WPs used in the base case can be found in the ACD Report (CRWMS M&O 1996e). Briefly, this base case has emplacement drift spacings of 22.5 m and axial center-to-center WP spacing of 19.5 m (for PWR packages). The area mass density used was 20.5 kgU/m^3 (83 MTHM/acre). The drift diameters were 5-m diameter emplacement drifts, with center-in-drift emplacement of WPs.

The model assumed a scenario in which there are drips on the waste containers once the temperature has dropped below boiling, independent of humidity. This case is (CRWMS M&O 1995a) more conservative (higher dose) than no drips on the WP since it assumes that once the WP is breached and the radionuclides diffuse through the pits, they are swept away in the water by advection. An even more conservative case, assuming drips on waste form, is considered in the next subsection.

The calculations of APF in this section use a different definition of APF than in the previous section. In particular, in this section APF is calculated by dividing the estimated peak dose for the 50th percentile of the complementary cumulative distribution functions for the base case by the estimated peak dose for the case with the respective engineered barrier included. These were generally done at a distance of 30 km from the proposed repository and at specified times. Table A-1 provides estimates of the APFs for the various engineered barriers considered based on studies in CRWMS M&O 1996a, which were done in accordance with QAP-3-5. The table also provides the environment (percolation flux) at which the thermohydrologic calculations were done and the parameters that the doses or the performance of a particular barrier have shown a sensitivity to in the various calculations. Engineered barriers for which no performance calculations were done are shown with dashed lines.

Table A-1 Engineered Barriers Performance Factors

Engineered Barrier Subsystem	Absolute Performance Factor[1]		Environment at which Calculations Performed[2]	Environmental Parameters which Influence Barrier[3]	Operational Considerations
	10k yrs	1M yrs			
Cladding	20	50	0.3 mm/yr	T; stress	T<350°C
Waste Package	~1	~1	0.3 mm/yr	q(flux), RH, T	—
Galvanic Protection	30	20	0.3 mm/yr	q(flux), RH, T	none
Drip Shield	>1	<1.5[4]	0.3, 1.25 mm/yr	RH; T	*need long life*
Backfill	>1	15	0.3, 1.25 mm/yr	q(flux), RH, T	emplacement, high T
Backfill +galvanic	>1	20	0.3, 1.25 mm/yr	q(flux), RH, T	emplacement, high T
Richards Barrier Backfill	>1	<1.5[4]	0.3, 1.25 mm/yr	q(flux), RH, T	not emplaceable
Repository Configuration	>1	~3	0.3 mm/yr	q(flux), ϕ, T	high T; logistics

[1] Absolute performance is based on dividing the estimated peak dose at 30 km for the ACD base case by the predicted doses with the respective barrier included.
[2] Current measurements indicate a percolation flux in TSw2 of 1 to 10 mm/yr with 5 to 7 mm/yr most likely.
[3] Environment includes such issues as temperature, RH, water chemistry, and percolation flux.
[4] This value could be larger if the barrier is very long lived (>7×10^5 years).

Cladding—A probabilistic cladding degradation analysis tool was developed and analyses were done of potential cladding performance in a recent thermal loading study (CRWMS M&O 1996d). Failure modes for the zircaloy cladding such as creep rupture, delayed hydride cracking, and clad unzipping were considered. The results showed that zircaloy is very durable and is relatively insensitive to aqueous corrosion over a range of about 2 to 12 pH. Zircaloy cladding, however, is somewhat sensitive to temperature and the study concluded that temperatures of the cladding should be kept below 350°C. If these conditions are met, the current probabilistic analyses indicate that under the current design specifications and the expected near-field thermal environments, only about 6 percent of cladding fails from small punctures and a much smaller fraction (about 0.5 percent) of cladding was predicted to undergo a gross rupture (or unzipping) during the period of elevated temperatures. The model assumed the remainder of the cladding was intact for the entire duration of the calculation.

However, the cladding degradation models (e.g., creep rupture, delayed hydride cracking, zircaloy dry oxidation and cladding unzipping) used in the probabilistic analyses were developed for interim dry storage of spent fuel, and the only driving force of cladding degradation in the models is temperature. In other words, the cladding degradation models are for short-term (tens or hundreds of years) cladding behavior and lack the ability to predict long-term (thousands or tens of thousands years) cladding degradation behavior. Thus, confirmation of the cladding degradation models for long-term cladding behavior would be required for a licensing argument, and suitable care should be exercised in using the current analysis results.

Performance credit for cladding may take two forms. The predictions assume that the zircaloy cladding provides containment. However, if the fuel/rods were to break under static loads then a portion of the SNF could be exposed. In this case the remaining cladding could still provide protection by reducing the area of fuel exposed. In this latter case, tests are needed and planned to be conducted at Argonne which would determine how much protection the cladding would provide.

The RIP simulation results in the thermal study indicate that, with the cladding performance that was estimated, the repository performance at the accessible environment is greatly improved (about one to two orders of magnitude decrease in releases to the accessible environment). The 10,000-year total peak release rate at the accessible environment is reduced by about one to two orders of magnitude compared to the case without cladding performance; for the 1,000,000-year performance, the total peak release rate is reduced by one to possibly two orders of magnitude. Similar results are reported for the total peak dose at 30 km distance: with the cladding performance, the 10,000-year total peak dose (50 percentile) is reduced by about a factor of 20 compared to the case without cladding performance, and the reduction in the 1,000,000 year total peak dose rate (50 percentile) is about a factor of 50 (CRWMS M&O 1996d). These values are reported in Table A-1 as the APF. It should be noted that the assumption of no cladding failure at low temperatures accounts for most of the APF.

To claim credit for cladding performance with a high level of confidence in licensing, the current individual cladding degradation models (i.e., creep rupture, delayed hydride cracking, zircaloy dry oxidation, and cladding unzipping) should be re-evaluated, especially for their long-term effects on cladding degradation. Additional cladding degradation processes, not considered in the current study, should be included. For example:

- Stress corrosion cracking (induced primarily by iodine in the interior side of the clad, and induced by the near-field factors such as salt formation on the clad surface)
- Long-term localized corrosion in radiolysis-induced acidic conditions
- Long-term degradation under static loads, which may be caused by a collapse of the internal structure (e.g., basket material)
- Hydride reorientation.

Furthermore, one degradation process is likely to have synergistic effects on the other process(es). For example, stress developed by the internal pressure build-up would make the clad more vulnerable to stress corrosion cracking as well as to creep rupture. One major obstacle to improving the models is a lack of long-term performance data. Development of testing data for such purposes is both lengthy and costly, and may not be practical in view of the time limitation for licensing. The most practical approach for immediate use would utilize the data available in the literature, and combine and synthesize the data with theoretical models that have been developed for other programs. Analysis of cladding that has remained in storage for extended periods may also provide useful information, but the initial state of the cladding should be known. To achieve the objective, a comprehensive compilation of the cladding degradation data currently available in the literature should be considered as a priority effort. Additionally, short term tests are planned to examine whether or not relative humidity and/or dripping water on the exposed end of SNF segments would cause the cladding to split, thus exposing more of the SNF than just the ends. This would simulate fuel rods that might eventually break under static loads and would provide information on whether performance credit for a reduction in exposed surface area could be achieved.

Waste Package—Essentially, all of the calculations to date have been run with a WP as a subsystem component. Thus, it is hard to separate out how much performance the WP provides. The performance allocation study estimated the APF for the WP and found that the WP provides a significant amount of performance in the first few thousand years. However, once the WPs start to fail, the APF drops to a relatively low value. The ^{237}neptunium (Np) APF was found to be about unity, depending on climate fluctuations (CRWMS M&O 1996b, Figure 2-17), for 10,000 years or beyond. Galvanic protection may enhance the lifetime of the WP but is considered as a separate barrier.

Galvanic Protection—Galvanic protection is protection afforded a more-noble metal or alloy by the corrosion of a less-noble metal or alloy in electrical contact and exposed to the same corrosive electrolyte. This protection can delay attack of the more-noble material. The degree of protection is a function of the differences in nobility of the materials. Also important is the amount of polarization or passivation of the surfaces as corrosive attack progresses. The degree of protection needs to be established with tests. Expert elicitation has indicated that some degree of galvanic protection will occur for the two layer WP (CRWMS M&O 1995a). For example, for the work used in the engineered barrier study, 50 percent galvanic protection was assumed which means that 50 percent by mass of the outer, corrosion allowance barrier must be gone before the inner, corrosion resistant barrier begins to corrode.

Table A-1 shows that galvanic protection can provide a significant amount of performance. Specifically, APFs of 20 to 30 were estimated for the two time periods. Tests will be needed to verify the amount of galvanic protection afforded by the actual candidate materials. If credit is to be taken for galvanic protection, some effort is needed to show that extrapolations to long time periods can be done with reasonable assurance. Since both cladding and galvanic protection have the highest performance factors of the engineered system, these concepts merit further study. Thus, this concept appears to merit further efforts.

Galvanic protection is a method to increase the containment lifetime. Other things could also potentially increase this lifetime. Specifically developing a better understanding of the corrosion processes and improving the corrosion materials may lead to predictions of longer lifetime. Also, material has been identified which have the potential for longer lifetimes. Thus, picking an alternative material for the corrosion resistant barrier may provide a benefit.

Pedestals or inverts—No performance assessment calculations were done to consider the presence of pedestals in the emplacement drifts. Some work was done in the engineered barrier study to examine the use of chemically treated inverts to enhance the sorptivity of the invert. Sedimentary apatite ore was considered as a possible addition to the inverts. Using some fairly simple assumptions, rough performance calculations indicated that an additive such as apatite could provide an increase in performance (CRWMS M&O 1996a). What needs to be determined is how much of a mineral like apatite is needed to provide an appreciable increase in performance. Additional discussion on this is provided below.

Drip Shield—The engineered barrier system study examined the possibility of using drip shields to keep advective flow from the WPs. The study evaluated a range of materials to be used including titanium and ceramics. The study concluded that ceramics would probably not withstand possible rock falls and so would not be the best solution for a design without backfill. The titanium, however, appears to be more durable and is fully resistant to water and steam; although if hydrogen is present the material would be degraded. There is, though, the concern that as the WP starts to corrode when the relative humidity increases, the increase in package size could deform the drip shield. Microbial corrosion was not considered (CRWMS M&O 1996a) but is probably not a factor for either ceramics or titanium.

The engineered barrier study evaluated the performance of a drip shield by doing calculations in which there was no advective flow and no drips on the packages, compared to the base case which had drips on sixty percent of the packages. Thus, for releases to occur there had to be diffusion of the radionuclides, except gaseous, beyond the region covered by the drip shield. The results found that the expected doses of water soluble radionuclides, such as Np, to the accessible environment were delayed. For the case of a 100,000 year barrier, the releases are delayed beyond 100,000 years. Significant delay occurs if the barrier has a lifetime of 500,000 years. However, the peak doses at the accessible environment are not changed by a large amount by having a drip shield. As shown in Table A-1, the releases to the accessible environment are reduced by, at most, a factor of 1.5 (for a 500,000 year barrier). If a drip shield can survive 1,000,000 years then significant performance can be achieved (APF=10^7) (Figure 9.3-45 CRWMS M&O 1995a). In all cases, the gaseous releases, such as ^{129}I, are unchanged by having a drip shield or no drip shield thus the APFs at 10,000 years is not much different than unity. At these lower fluxes (0.3 mm/yr), the WP failure does not change significantly with or without a drip shield. The APF at 10,000 years is significantly greater than one for aqueous radionuclides. Most of the releases at 10,000 years are due to gaseous emission. Little of the water soluble radionuclides reach the accessible environment in either the base case or the drip shield case at 10,000 years. The conclusions of the engineered barrier study showed that some performance advantage could be gained by using drip shields but one would have to have a drip shield that would survive over 700,000 years to obtain any reduction in peak dose at 1 million years (CRWMS M&O 1996a). Additional evaluations of a drip shield were done for the potentially higher flux environment and these are discussed below.

A drip shield may be placed over the WPs as an umbrella. However, an alternative method may be to develop a third layer on the WP. This study did not compare the performance or cost benefits of adding a third barrier to the WP.

Backfill—Performance calculations were done (CRWMS M&O 1996a) with and without backfill to determine what advantage there is to using backfill. The results, shown in Table A-1, indicate that for 1,000,000 years backfill may provide an increase in performance by about a factor of 15. At 10,000 years there is some increased performance but the releases to the accessible environment are predicted to be low so the table only indicates the APF is greater than one. That is, the dose predictions were generally predicted to be very low and less than the lower limit of the graph and therefore since backfill provided some lowering of relative humidity an APF of greater than one was indicated. Calculations were also done by considering backfill plus galvanic protection. The results indicate that essentially no difference in APFs was noted between this case and the one with galvanic protection alone. These calculations were done with percolation fluxes of 0.3 and 1.25 mm/yr. Current expectations are that the fluxes are higher than this. Calculations with higher fluxes were done and these are reported in the next subsection.

Backfill use has operational impacts. Backfill is difficult to emplace and also produces a significant increase in WP temperature. In some cases, the cladding thermal criterion of 350°C may be exceeded, although for the average WP and a backfill which has an effective thermal conductivity of greater than 0.4 to 0.5 W/(m·K), the cladding criteria will not be exceeded if the backfill is emplaced at 100 years. Based on the fact that no additional performance margin was deemed necessary, the engineered barrier study recommended that backfill not be used but that the design should not preclude its use (CRWMS M&O 1996a). Further analysis follows in Section 4.3.

Richards Barrier—Multiple layers of different porosity materials can act as a hydraulic barrier to inflow of water from the surrounding environment. Such a barrier is known by a number of names but is most commonly known as a Richards barrier. The Richards barrier acts as a diffusion barrier to the transport of radionuclides by limiting inflow of water to the WPs. The performance calculations (CRWMS M&O 1996a) were the same as those for the drip shield in which there was no advective transport of liquid. The Richards barrier would also provide for higher WP temperatures and decreased relative humidity for a period of time similar to the regular backfill. Thus, its performance should be similar to a drip shield. Based on work done in support of the Engineered Barrier System Performance Requirements System Study, Subsurface Repository Design personnel indicated that emplacing a Richards barrier and ensuring that the layers were adequately placed would not be possible with current technology (CRWMS M&O 1996a).

Tunnel Liner—No performance calculations have been done on tunnel liners since they have not been considered for post-closure performance. Current designs are considering using tunnel liners for the emplacement drifts. Three types of tunnel liners (precast concrete liners, cast-in-place concrete liners, and steel liners) are being considered by design. Precast concrete liners are the most likely candidates at the current time. There are performance issues for such tunnel liners that need to be evaluated. Some examples are the durability and longevity of the liner under the high temperature conditions in the emplacement drifts. The effect that the liners have on moisture flow also needs to be considered. Cementitious tunnel liners may change the chemistry in the drift and of the fluid which is carrying the radionuclides. If the pH of the fluid is increased as a result of the presence of cementitious materials, some preliminary performance assessments have shown that performance may be degraded because the sorptivity of some of the natural barriers may be altered by the alkaline plume (IOC to DOE, Dr. Brocoum from Performance Assessment, *Status/Summary Report for Fiscal Year 1996 Activities within the Performance Assessment Overview Study on the*

Cementitious Material, LV.PA.DCS.09/96-038). However, higher pH may also reduce corrosion and SNF dissolution somewhat. In addition, the impact of heat on the tunnel liners and the consequences on durability, needs to be understood. These effects need to be carefully examined before final decisions are made on tunnel liners. Work is currently underway to examine lower pH cements.

With regard to the heating effects on tunnel liners some predictions have been done of emplacement drift heating. Around the design basis packages and at thermal loads above the existing design of 20.5 kgU/m², drift wall temperatures can approach 200°C. The current design requirement for drift walls is to not exceed temperatures of 200°C (CDA EBDRD 3.7.G.1). This peak is reached about 10 to 30 years after emplacement. The peak temperatures may exist for as much as 50 years. Additionally the period when the emplacement drift supports are needed is currently 100 years, (CDA Key 016). Thus, based on this work it is determined that the tunnel liners will need to perform their load bearing mission for 100 years and survive temperatures as much as 200°C for the majority of this time. The temperature analysis used in this work came from the thermal study (CRWMS M&O 1996d).

Repository Configuration—WP and drift spacing and AML could all potentially affect the performance. Significant package-to-package variation in heat output was found to exist (CRWMS M&O 1995c) which could possibly move moisture from hotter to cooler packages. One method considered to reduce some of the variability was to move the WPs closer together. Instead of about 19 m spacing, cases were evaluated in which the spacing was 1.0 or even 0.1 m between the ends of the packages. This line loading tended to moderate the temperature variability but the near field temperatures increase significantly. The changes had a modest affect on performance. Using line loading increased the APF at 10,000 years but the releases were significantly below the interim standard of 15 mem./yr at 30 km. At 1,000,000 years, the APF for the line load is about three (CRWMS M&O 1996a). Other analyses (CRWMS M&O 1996d) indicated that drift wall temperatures and cladding temperatures could exceed the assumed limits of 200°C and 350°C, respectively. Whether these temperature criteria are exceeded depends to some extent on the model assumptions and the WP sequencing (if the hottest WPs are flanked by DHLWS then temperatures would be lower).

APPENDIX B
NATURAL BARRIERS

APPENDIX B
NATURAL BARRIERS

B.1 SITE DESCRIPTION

The mission of the MGDS is to provide for emplacement and isolation of the nation's commercial SNF and DHLW in such a way that public health and safety are protected. The potential MGDS will be able to accommodate about 70,000 MTHM which currently is assumed to be composed of about 63,000 MTHM of SNF from commercial reactors, about 4,700 MTHM equivalent HLW from reprocessing defense materials, and about 2,300 MTHM of U.S. Department of Energy (DOE) SNF (CRWMS M&O 1996c).

The site of the potential repository at Yucca Mountain is located approximately 100 miles northwest of Las Vegas, Nevada in a relatively arid climate. Two of the waste isolation attributes of this site identified in the *Site Characterization Plan* (SCP, DOE 1988) are that the site is located in an area of relatively sparse population and that it is in an arid climate, which would limit recharge of water. The site is also on the Nevada Test Site which has been used extensively for nuclear testing and the Nellis Air Force Range (DOE 1988). A portion of the site was also Bureau of Land Management land.

The potential repository in Yucca Mountain is in the Topopah Spring Member, a welded tuff unit of the Paintbrush tuff (see Figure 3-1). The Topopah Spring Member is approximately 330 m thick and dips from west to east by about six degrees. The potential subsurface layout is primarily in TSw2, which provides a minimum overburden of 200 m and is a distance of 230 to 380 m above the water table (CRWMS M&O 1996c) for the six calculational columns used.

The strata of Yucca Mountain have been generalized into five hydrostratigraphic units that differ from one another in average properties (Montazer and Wilson 1984). These different units, in descending order, are:

- Tiva Canyon welded unit (TCw),
- Paintbrush tuff nonwelded unit (PTn),
- Topopah Spring welded unit (TSw),
- Calico Hills nonwelded unit (CHn),
- Crater Flat undifferentiated unit (CFu), which is composed of the Prow Pass Tuff and the deeper Bullfrog Tuff.

The welded units TCw and TSw have small matrix porosities and permeabilities but have larger bulk permeabilities because these rocks fracture easily. On the other hand, the PTn and CHn have larger porosities with small permeabilities because it is believed they have much fewer fractures. The hydrologic properties of these units are summarized by Bodvarsson et al. (1996).

The objective of the emerging Project's Waste Containment and Isolation Strategy (YMP 1996), and even earlier in the SCP (DOE 1988), is to contain and isolate the waste so that the public and unrestricted areas are protected against radiation exposure and releases of radionuclides. This is done by using a multibarrier system that is a combination of natural and engineered barriers. These multibarriers must work together to isolate the wastes. This section discusses the performance of the natural barriers and Section 4 discusses the performance of the engineered barriers. This section documents work that was done in the Performance Allocation Study and does not consist of any new work. The last subsection in this section does provide an overview of some new work on estimation of zeolite distributions.

B.2 NATURAL BARRIERS

Because the region is arid, the recharge of groundwater is low and the amount of moving groundwater is also relatively low. Climate changes may occur but these can be estimated with some confidence through geologic records (Long and Childs 1993).

The first barrier to water infiltration on the mountain, that is present in some locations, is the unconsolidated alluvium. The alluvium/colluvium has a relatively large storage capacity to retain moisture, which generally allows removal of this moisture by persistent evapotranspiration. However, the alluvium/colluvium is not uniformly distributed and, on side slopes and ridge tops, it may be thin or absent allowing higher infiltration rates (Flint and Flint 1995 and in private communication with those authors).

The significant change in permeabilities between the fractured TCw and the less fractured and hence smaller permeability PTn provide for a significant contrast, which is likely to impede episodic flow of percolating water in the matrix. The PTn itself has a substantial matrix storage capacity providing for redistribution of water and encouraging down-dip diversion. The contrast in permeabilities between the PTn and underlying TSw layer further encourages down-dip diversion of water flow. Earlier studies (Ho et al. 1996) have identified potential thermomechanical issues associated with the PTn and the potential to increase fracture sizes. This effect needs further evaluation.

The TSw matrix in which the potential repository is located has low porosity and permeability with relatively high saturation of the pores of 85 to 95 percent. These conditions would tend to favor imbibition of water from fractures into the rock matrix. However, in some cases this imbibition may be inhibited by mineral deposition in the fractures and the small permeability of the rock. The greatest downward flux of water through this host rock is anticipated to be primarily in the fractures. Studies in the Exploratory Studies Facility (EHF) have confirmed localized regions of elevated concentrations of ^{36}Cl which tend to confirm fracture flow in certain areas (Fabryka-Martin et al. 1996). This will be further discussed below.

The CHn hydrogeologic unit beneath the potential repository consists of glassy and variably zeolitized nonwelded and partially welded ash flow tuffs and bedded tuffs, extending vertically downward to the water table from the basal vitrophyre in the overlying TSw unit. The zeolite in this layer are predominately clinoptilolite with some mordenite and smectite and, in some deeper areas, analcime. These minerals are hydrous minerals that have a significant affinity for water. In addition, these zeolites, particularly clinoptilolite, have sorptive capacities for a number of radionuclides, particularly Cs, Sr, and to some extent, Np (Meijer 1990). Thus, based on initial studies, they provide significant retardation and increases in travel times for a number of radionuclides (CRWMS M&O 1996d). The location, depths, and concentrations of these zeolites is uncertain at this time since the conceptualizations are based on a limited amount of data from boreholes. This information is summarized in the *Thermal Loading Study for FY 1996* (CRWMS M&O 1996d). Additional borehole data have been or are being analyzed and work is underway to improve the zeolite conceptualizations (this ongoing effort is discussed below).

The saturated zone is the final stage in the path for water soluble radionuclides (nongaseous) to reach locations where there is the potential for drawing water from these regions and exposing the public. Locally, beneath the potential repository, the configuration of the potentiometric field defines a water table that would indicate generally southward flow, joining with eastward flow to produce a southeastward direction of flow away from the repository site based on work done by Robinson (1994) and Luckey et al. (1996). The saturated zone will provide dilution and dispersion of radionuclides during transport. These are functions of the flow velocity and the rock permeability and structural properties of the medium. Additionally, the mixing depth in the saturated zone, which is currently uncertain, will determine the extent of dilution and whether a well mixed plume of radionuclides results, or a more concentrated plume near the surface of the saturated zone is prevalent. For these calculations a mixing depth of 50 m, and two values of Darcy flux of 2 m/yr and 0.31 m/yr, were used (CRWMS M&O 1996b). More details of the saturated zone flow can be found in the work of Fridrich et al. (1994) and Luckey et al. (1996).

B.3 ENVIRONMENT

Performance predictions have been found to depend on the environmental conditions (water percolation flux and temperature) that exist (CRWMS M&O 1995a). This subsection describes the conditions that are believed to exist underground, identifies any differences that might have existed in past calculations, and briefly discusses any potential impacts from these environments. More detailed discussions of the impact of environmental conditions on performance will be provided in later sections.

The annual precipitation at Yucca Mountain averages about 165 mm/year for the prevailing climate (Hevesi and Flint 1996). Potential evaporation is about an order of magnitude greater than this, which implies that for water to escape rapid evaporation and transpiration back to the atmosphere, it must quickly undergo either direct runoff or infiltration. Runoff removes water from the mountain but also results in concentrating the water in local depressions and channels after precipitation. Within a short period of time evapotranspiration consumes most of the water and dries the thicker soils and alluvium which returns most of the water to the atmosphere. Much of the water that infiltrates the mountain is thought to be diverted laterally by the more porous PTn. Based on the assumption of rapid evaporation and lateral diversions, most calculations to date have assumed that the average net percolation rates in the TSw unit are between 0.1 and 1 mm/year (DOE 1988).

Recent studies using measures of the geothermal gradient in the mountain, saturation levels, capillary pressure, and perched water measurements, indicate that the estimates of water percolating below the PTn are between 1 and 10 mm/year (Bodvarsson et al. 1996). In addition, a sampling of rocks underground in the EHF have identified some localized areas where elevated concentrations of ^{36}Cl

have been found (Fabryka-Martin et al. 1996). Strong indications are that the observed ^{36}Cl concentrations are bomb pulse isotopes from nuclear testing and this is further substantiated by a Yang et al. paper (1996) which found bomb-pulse tritium occurring in TSw2. This evidence implies that the bomb pulse isotopes are reliable indicators of rapid fracture flow and that there are localized regions where this rapid fracture flow, at least on an episodic fashion, occurs. This conclusion is based on the fact that the surface nuclear tests occurred about 50 years ago and thus it took less than or equal to 50 years for the flow to reach the TSw units. Additional discussion showing the relevance of these percolation rates is given below.

The current design focus for the potential repository is to emplace the SNF at a density of 20.5 kgU/m^2 (83 MTHM/acre) (CRWMS M&O 1996e). It should be noted that the convention currently being used for the underground emplacement is to specify area mass loadings in kilograms of uranium equivalent. This density of SNF produces a significant amount of heat resulting in significant elevation of the temperatures in the various natural barriers. A thermal loading study (CRWMS M&O 1996d) determined that the heat would result in some dehydration of the zeolite and the resultant production of water, although the process is reversible. The predictions in the study could not determine any impact on performance due to the dehydration and additional water although it was not clear that all of the coupled processes associated with these effects were accounted for. The thermal study also recommended that the temperature of the zeolite not exceed 90°C to minimize the change that the zeolite clinoptilolite may undergo an irreversible mineralogic alteration to analcime. Analcime does not have any appreciable sorptivity for radionuclides. The current design of 20.5 kgU/m^2 does not result in temperatures at the average top of the zeolites that exceed 90°C, based on conduction only predictions. This temperature, at the estimated average top of the zeolite (about 170 m below the potential repository in the Primary Area), is not reached until the area mass loading of SNF is above 22.2 kgU/m^2 (90 MTHM/acre).

The current design area mass loading results in an increase in the saturated zone temperatures from approximately 30°C to as high as 70°C (Ho, et al. 1996). Very preliminary analyses (personal communication from W. Glassley to S. Saterlie, November 1996) indicate that this increase in temperature will result in changes in mineral concentrations, porosity, and, consequently, flow velocity. No estimates have yet been made on the impact of these changes on performance. The calculations to date, and those reported in this study, all assume that conditions similar to the current ambient conditions will be present.

B.4 NATURAL BARRIER PERFORMANCE

Some calculations have been done that provide indications of the performance of some of the natural barriers. Studies, such as the TSPA conducted most recently in 1995 (CRWMS M&O 1995a), and the *Systems Study of Options for Characterizing the Calico Hills Nonwelded Hydrogeologic Unit at Yucca Mountain, Nevada* (CRWMS M&O 1995b), examined some aspects of performance for natural barriers. The most recent analysis, however, has been the work done in the *Description of Performance Allocation* study (CRWMS M&O 1996b). The majority of the discussion in this section is based on the work performed in that effort.

The performance allocation study calculated relative and APFs for each natural barrier considered. These factors were based on calculations of the concentration of radionuclide entering a particular barrier (source term), and the concentration of radionuclide which exits the barrier (downstream mass released). These concentrations are clearly time dependent and the calculations were carried out to 1,000,000 years. The calculations examined releases from the WP, transport through the invert beneath the WP, transport through the unsaturated zone beneath the WP, which included the CHn, transport through the CHn, and transport through the saturated zone. Discussions of WP and invert performance are discussed in Section 3.

The basis of the calculations was the parameter set used in TSPA 1995 (CRWMS M&O 1995a). The base case considered was for a thermal load of 20.5 kgU/m^2 (83 MTHM/acre) and an initial unsaturated zone percolation flux of 1.25 mm/yr. This initial percolation flux is modified according to a climate change model that randomly samples numbers within a period of 100,000 years (CRWMS M&O 1995a). However, the rock properties used in those earlier thermohydrologic calculations which provide estimates of WP lifetime, are those that are consistent with lower flux conditions. The transport calculations in that work were done with a single infiltration rate of 1.25 mm/yr. Recent information indicates that the percolation flux is probably 1 to 10 mm/yr and under climate changes may be as much as 30 mm/yr.

The assumptions used in the Performance allocation study that are applicable to this effort are primarily those used in the TSPA 1995 work (CRWMS M&O 1995a) and additional details can be found in that reference. A summary of some of those assumptions follows:

- *Waste containers*—the waste containers are emplaced center-in-drift. These containers use the multipurpose container concept with a 100 mm thick corrosion-allowance material, such as mild steel, and a 20 mm thick corrosion-resistant material, such as Alloy 825.

- *Analytic models*—The two-dimensional FEHM code (Zyvoloski et al. 1995) using a smeared heat source was used for the near field calculations of the environment. In conjunction with the more detailed process models the total system performance is calculated using the RIP (Golder Associates 1994). Two dimensional, smeared heat source calculations can underpredict the temperature and overpredict the relative humidity at the WP.

- *Waste Stream*—The waste stream is oldest fuel first with an average age of 26 years and burnup of 39 GWd/MTHM for the PWR fuel.

- *Engineered Barriers*—The WP, backfill, cladding credit, and galvanic protection were considered in some of the cases.

- *Subsurface Design*—The subsurface design considered 5-m diameter emplacement drifts and 22.5-m spacing between drifts. Uniform spacing of identically loaded WPs was approximated by a smeared line source to produce the desired AML of 20.5 kgU/m^2.

- *Fracture Flow and Fracture-Matrix Interaction*—The calculations of seepage flux into the drifts was based on a simplified dual continuum model (refer to TSPA-95) which allowed water to drip on the WPs. The drift scale thermohydrologic model for calculating relative humidity and temperature in the drifts was based on an equivalent continuum model of fracture matrix interactions. The fracture-matrix transport in the far field unsaturated zone uses a dual continuum approach.

The calculations done in the performance allocation report were not done under quality affecting procedures; they are generally scoping calculations. The RIP code was developed and verified using ASME NQA-1 and ISO-9000 standards (Golder Associates 1995). The process models used in the analysis, however, have not been qualified. Those models were used for the purposes intended and over the range for which they were designed. Current uncertainties on unsaturated zone flow and transport are high although the testing program should reduce these uncertainties in the future.

The performance allocation effort calculated the total mass (or mass of a given radionuclide) released from the downstream end of a particular barrier. This was done for each of the barriers considered which were the WP, the engineered barrier system, the TSw unit beneath the repository, the entire unsaturated zone beneath the repository, including the CHn and the Prow Pass, and the saturated zone. Although identified as barriers, the performance allocation did not calculate the performance of the alluvium and the PTn. These are barriers because they limit the amount of flux into the TSw.

The mass of radionuclide which exits one barrier is the source for the next barrier just downstream. Based on this, a time-dependent APF for a barrier can be established. This APF is defined (CRWMS M&O 1996b) as the ratio of the input to a particular barrier at any given time to its output. Unless decay produces a radionuclide in a given barrier, the APF is usually greater than or equal to one. Thus, an APF of 10 means that a given barrier reduces the accessible environment dose rate of a radionuclide by a factor of 10 at a particular time. It should be noted that at early times, the APF can indicate a high reduction in dose but at later times the APF can drop as the radionuclide traverses the barrier. An illustrative example of the performance allocation work is provided below in this section.

The estimated doses of ^{237}Np for 10,000 and 1,000,000 years at the various barriers were used to estimate the APF for the various barriers (Figure 2.1-15 and 2.1-17 of CRWMS M&O 1996b). Table B-1 shows the estimates of these APFs for the three barriers (the first barrier actually includes the second barrier) calculated at the two times. In addition, the table provides the environment (percolation flux) at which the calculations were done and the parameters that the doses or the performance of a particular barrier have shown a sensitivity to in the cited calculations. It should be noted that the calculations assumed that the saturated zone was devitrified.

An examination of the table shows that the unsaturated zone transport, including the Calico Hills unit provides the largest performance for releases that affect the performance at 10,000 years. For times of about 1,000,000 years the saturated zone provides the most performance at reducing radionuclides, although the unsaturated zone is of the same order of magnitude. The somewhat lower performance of the CHn may be due to the choice of sorption coefficients for the TSw and CHn (see CRWMS M&O 1995a). For those calculations, the Np sorption coefficient was actually larger in the TSw unit than in the CHn unit. These sorption coefficients have been updated in the calculations done for this study and reported in Section 3. The calculations at 10,000 years are the most uncertain because the predicted releases are relatively small at those times and therefore the calculation had to divide two small numbers.

Table B-1 Natural Barriers Performance Factors

Natural Barrier Component	Absolute Performance Factor[1]		Environment at which Calculations performed[2]	Environmental Parameters which Influence Barrier[3]
	10 k yrs	1M yrs		
Unsaturated Zone Transport[4]	$3x10^7$	30	1.25 mm/yr	q(flux);ϕ
CHn[5]	$1.4x10^6$	12	1.25 mm/yr	q(flux); T; ϕ
Saturated Zone Transport	4000	70	2 m/yr; isothermal	q(flux);T;z(mix) ;ϕ;v(flow)

[1] Absolute performance stated in terms of the factor that the radionuclide doses exiting the barrier/layer are reduced from the doses entering the barrier.
[2] Current measurements indicate a percolation flux in TSw2 of 1 to 10 mm/yr with 5 to 7 mm/yr most likely. Calculations at these higher fluxes are discussed in Section 3.
[3] Environment includes such issues as temperature, RH, water chemistry, percolation flux.
[4] Includes zeolite in CHn.
[5] The performance of this layer was estimated as the difference in doses at the bases of the TSw unit and the base of this unsaturated zone.

The table also provides an indication of what parameters can influence the performance. For all the barriers considered, the percolation flux, labeled q(flux) in the table, can affect the results. In all three of the barriers the permeability and degree of fracture versus matrix flow is also important. Temperature, T, is important for at least two of the barriers if not all three because it can result in mineralogic changes in zeolite (CRWMS M&O 1996d) and it may change flow paths, flow velocities, and mineralogy in the saturated zone. The porosity of the rock, labeled ϕ in the table, is also important to all three barriers. In the saturated zone the mixing depth, z(mix), and the flow velocity, v(flow), are also of interest.

The uncertainty that currently exists about the infiltration rate and the amount of water traveling in fractures needs to be better established and performance predictions made at the expected values. Recent measurements have identified anticipated infiltration rates of 1 to 10 mm/yr with expected values of 5 to 7 mm/yr and possibly as much as four times higher rates as a result of climate changes (Bodvarsson, et al. 1996). Performance calculations are needed at these higher fluxes. Some preliminary calculations made at these higher fluxes are reported in the next section.

Based on the importance of the zeolites to performance, an updated conceptualization is needed. The location of the zeolites, the concentration of zeolites as a function of depth, and the depth of these minerals beneath the repository is essential data. This updated conceptualization is currently being created and a synopsis of the progress to date is provided below. In addition, any updates to the sorptivity coefficients of zeolite for the various radionuclides is important as well.

The current calculations have assumed ambient conditions in the saturated zone. Information needs to be obtained to better estimate the mixing depth (currently 50 m is assumed) and the flow velocities in the saturated zone.

In addition. information is needed as to how some of the saturated zone properties and conditions will change due to thermal effects. Some preliminary calculations have been done as to the effect of heat on the saturated zone. LLNL has done an assessment on the effects of hydrothermal flow in the saturated zone. They found that convection cells that have an extent of a few kilometers can develop and flow velocities will increase as a result of the increased temperature in the saturated zone caused by the SNF decay heat. The magnitude of the buoyancy flow increases with increasing temperature (LLNL 1996). There have also been some preliminary scoping analyses of potential mineral and porosity changes in various rock units (unsaturated zone and saturated zone) due to the effects of heating. Mineral redistributions and porosity changes that could result in one to three orders of magnitude changes in permeability were predicted to occur in the saturated zone. Even larger changes occurred in the unsaturated zone (personal communication from W. Glassley to S. Saterlie, November 1996). Based on this information it is important that heating effects are understood in the various rock units.

MOL.19971110.0137