EEG-3
EEG-2 (Appendix III)

Radiological Health Review
of the Draft Environmental Impact Statement (DOE/EIS-0026-D)
Waste Isolation Pilot Plant, U. S. Department of Energy

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*The notation (5) refers to Chapter 5 in the DEIS.

APPENDIX

FOREWORD

The purpose of the Environmental Evaluation Group (EEG) is to conduct an independent technical evaluation of the potential radiation exposure to people from the proposed Federal radioactive Waste Isolation Pilot Plant (WIPP) near Carlsbad, in order to protect the public health and safety and ensure that there is no environmental degradation. The EEG is part of the Environmental Improvement Division, a component of the New Mexico Health and Environment Department — the agency charged with the primary responsibility for protecting the health of the citizens of New Mexico.

The Group is neither a proponent nor an opponent of WIPP.

Analyses are conducted of reports issued by the U.S. Department of Energy (DOE) and its contractors, other Federal agencies and other organizations, as they relate to the potential health, safety and environmental impacts from WIPP.

The project is funded entirely by the U.S. Department of Energy through Contract DE-AC04-79AL10752 with the New Mexico Health and Environment Department.

Robert H. Neill
Director

INTRODUCTION

These comments are limited to the radiological health and safety and environmentally related aspects of the Draft Environmental Impact Statement (DEIS), Waste Isolation Pilot Plant (DOE/EIS-0026-D) and the background material used by DOE, with the primary focus of the comments on those aspects that have potential effects on the State of New Mexico.

While the Environmental Evaluation Group (EEG) recognizes that some later data have become available and that there may be significant changes in the mission of the WIPP, the review has been confined to the DEIS WIPP Reference Case (1-2; 6).* In those cases where options on WIPP are still open or data was not provided, final evaluation will await that information.

This evaluation includes:

(1) checking the calculations in the DEIS with the assumptions and methods used;

(2) checking computations by alternate (usually simplified) approaches;

(3) evaluating the assumptions and methodology used;

(4) considering possible omissions;

(5) evaluating conclusions reached; and

(6) recommending additional actions to be taken.

Alternative locations to the proposed WIPP site have not been evaluated since they are beyond the scope of EEG's mission.

Several meetings were held with the DOE and it's contractors to clarify some of the assumptions, input parameters, and numerical procedures used in various analyses.

*The notation (1-2; 6) refers to Chapter 1, page 2, paragraph 6 of the DEIS.

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SUMMARY

General

The Department of Energy is to be commended for making a major effort to determine the environmental impact of WIPP.

This review of radiological health considerations contains a number of concerns, questions and recommendations that should be addressed by the Department of Energy in the final Environmental Impact Statement (EIS).

Using the assumptions contained in the Draft Environmental Impact Statement (DEIS), the EEG calculated a number of radiation doses and the results were found to be in general agreement with those presented in the DEIS. The doses resulting from the operational and long-range releases from WIPP to the general population are no more than a fraction of existing radiation doses to the public. However, there are a number of technical considerations in the assessment of radiation exposure that were not adequately evaluated in the DEIS. They are discussed in this review.

A number of additional dosage estimates have been identified that need to be calculated by both DOE and EEG.

As the DEIS did not contain estimates of the amounts of radioactivity to be permanently located in the repository, it was necessary to calculate these amounts.

Estimated Plutonium Inventory in TRU Wastes*

Radionuclide	Activity (Curies)
Pu-238	35,000
Pu-239	480,000
Pu-240	120,000
Pu-241	1,200,000

*after 30 years of repository operation

It is apparent from our analyses that additional information and evaluations will be necessary in the future if the WIPP project proceeds. Consequently, the DEIS and its review are only the beginning of the health and safety evaluations that need to be performed.

The DOE stated in the DEIS that the WIPP repository should be licensed by the Nuclear Regulatory Commission (NRC). Recent developments suggest that the WIPP may not be licensed by that organization. EEG recommends that the proposed facility be subjected to the full scrutiny of health and safety considerations afforded by the licensing procedures of the NRC.

Health Effects

The DEIS did not estimate health effects to people from either the expected or potential radiation exposure but used dose as a presumptive index of hazard. Although not as informative as health effects, it has been a common practice in radiation protection work. Various radiation standards-setting organizations such as the International Commission on Radiological Protection (ICRP), United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR) and the Biological Effects of Ionizing Radiation Committee (BEIR) of the National Academy of Sciences have developed models for mortality risk coefficients from ionizing radiation. In order to do expected mortality calculations, it is necessary to know not only the magnitude of the radiation exposure but the size of the population exposed as well as the probability of such an occurrence. DOE should address the issue of health effects in the final EIS. EEG will generate those estimates when the required information has been developed for the various population groups from both normal and accidental exposures.

Transportation

The equations used in the DEIS calculations of radiation dose from the normal transportation of the radioactive wastes were derived by the EEG and the calculated doses were found to be in agreement with those presented in the DEIS. These exposures to the general population are small additions to those from natural background and other man-made radiation sources. However, a critical evaluation of the assumptions used on potential accidents in the DEIS raises the following issues.

Radiation exposures from deliberate acts of sabotage in the transportation of radioactive materials could be considerably higher than those from traffic and rail accidents but the DEIS assumed there would not be a difference.

Some of the DEIS assumptions for accidents may not be conservative. Examples are:

1) A fire occurring during a rail accident involving contact-handled transuranic wastes (CH-TRU).

2) Leakages of remote-handled transuranic wastes (RH-TRU) from a container following a rail accident.

3) Ingestion of radioactive material following an airborne release.

Consideration should be given to shipping all the radioactive waste by rail wherever the calculations show that the actual and potential radiation exposures to people will be reduced. This is consistent with the concept in radiological health that all unnecessary radiation exposure be avoided and exposures kept as low as reasonably achievable (ALARA). Consideration should also be given to restricting shipments in icy weather.

Waste Acceptance Criteria

A full evaluation of the radiological consequences of operations and accidents cannot be completed until the waste acceptance criteria are developed by the DOE Waste Acceptance Criteria Steering Committee. DOE has been furnishing that material to EEG for review as it becomes available. There are a number of criteria that must be specified such as the degree of combustibility of the wastes, the amount of gas that can be generated through decomposition of organic materials, the amount of pyrophoric material, and the amount and type of non-radioactive hazardous material to be stored.

Site Characterization

There are uncertainties regarding several geologic and hydrologic aspects of the area surrounding the WIPP site. DOE is continuing to gather and analyze data relevant to these features and processes. The final EIS should include a more detailed analysis of the following:

1) Brine reservoirs, apparently large and under high pressure, which have been encountered in at least 7 wells within 9 miles of the periphery of the WIPP site.

2) Deep dissolution; i.e. dissolution of lower and intermediate levels of the salt beds.

3) Breccia pipes, which may be localized deep dissolution features, starting in the lower portion of the salt beds and migrating upward.

4) Variations and uncertainties in ground water flow rates and flow paths.

5) The effect of the presence of impurities (e.g. clay, anhydrite, and polyhalite) on the physical, hydrological, thermal and strength characteristics of rock salt from the repository horizons.

Site Selection Criteria

In the absence of regulatory standards by the Nuclear Regulatory Commission and the Environmental Protection Agency for the permanent disposal of radioactive wastes, reliance has been placed by the Department of Energy on establishing criteria that a repository should meet.

In light of this fact, we recommend that the Department of Energy formally request the involved federal agencies and other bodies of technical expertise to comment on the reasonableness and adequacy of the site selection criteria so that a consensus can be achieved. In this manner, any allegation that the criteria were unilaterally established by DOE can be avoided.

A failure of the proposed repository to meet a given design criterion does not in itself mean there is a hazard. It does identify or flag those areas that need to be thoroughly analyzed to determine whether or not the consequences of failure could result in radiation exposure to people.

Operational Exposure

The information on occupational radiation exposure is incomplete in the DEIS and presumably will be covered in more detail in the Preliminary Safety Analysis Report (PSAR).

The operational accident scenarios evaluated in the DEIS appear to be fairly complete in scope and the EEG calculations agreed with the DEIS when the same assumptions were used. Some of the assumptions may underestimate the amount of radioactivity released from damaged containers.

It is also unclear whether the exhaust air from the underground waste handling facility will pass through the HEPA filters before being released to the environment.

From the information in the DEIS, there is a question whether the non-radiological Ambient Air Quality Standards of New Mexico will be met in Zones II, III and IV. A more detailed analysis is necessary to determine the control measures that will be required.

Experimental Waste Program

It is recognized that the experimental high level radioactive waste program will provide empirical evidence for many of the theoretically derived geological parameters. However, in order to evaluate the potential radiation exposure to workers and the public it will be necessary to know the radionuclides involved, the amounts of radioactivity, the waste form, the details of the experiments and the plans for retrieval of the radioactive material. The experimental waste program could contain 9 to 90 million curies of radioactivity if the full-sized commercial high level waste canisters are used.

Long Term Radiation Releases

The DEIS considers a number of scenarios which could lead to release of radioactivity after the repository has been sealed. Based on the assumptions used in the DEIS analyses of long-term release scenarios, EEG's results are in reasonable agreement with dose rates and radionuclide migration times presented in the text. Except for the drilling scenario, the dose rates are small. However, the scenarios considered were limited and the EEG has identified additional scenarios and calculations which should be considered such as the potential contamination of well water or the role of pressurized gas in bringing radioactive material to the surface. EEG has considered the ranges over which some of the parameters relevant to the movement of radioactivity in ground water can vary and the effects of these variations on the DEIS results. Therefore, EEG recommends that the detailed sensitivity analysis currently being conducted by DOE should be included in the final EIS.

Retrievability

It is essential that the ability to retrieve the radioactive wastes be examined in detail as to criteria, procedures, logistics, canister integrity, hazards to workers and hazards to the general population.

Decommissioning

Decommissioning options are discussed in satisfactory detail in the DEIS. However, the related issue of the degree and longevity of site control after decommissioning must be addressed. This is important since an uncontrolled site would be subject to various human actions, especially drilling, that could violate site integrity. The advantage and feasibility of site control for periods greater than 100 years should be evaluated.

HEALTH EFFECTS

The DEIS neither estimated nor discussed health effects from the potential radiation exposure to the population but used doses instead as an index of hazard. In the definition of risk in the glossary, the DEIS defined "consequence" of exposure as "population dose" and not "health effects". Although this is not as informative as health effects, it has been a common practice in radiation protection work.

Estimated health effects from WIPP should be included in the final EIS. It is recognized that there are uncertainties associated with such estimates that include the anticipated size of the future population at risk from WIPP, the probability of accidents and the frequency distribution of those accidents, the magnitude of the population dose for various conditions and indeed the basic applicability of a linear correlation of health effects with doses at such low dose rates. EEG plans to undertake these calculations in the future and to also include comparisons with presumed deaths from natural background and other radiation sources in the environment.

EEG intends to use the following approach. ICRP (Publications 26, 27) has developed a set of risk coefficients for various somatic biological end points and tissues that are based on currently available data (Ref. 1, 2). For a uniform whole-body irradiation (averaged over both sexes and all ages) their report indicates a mortality risk coefficient of approximately 10^{-4} rem^{-1} (a probability of 1 death per 10,000 person-rem). In 1977 UNSCEAR gave more detailed information on the basis for this numerical value and pointed out that such coefficients are obtained for mortalities induced at doses in excess of 100 rads (Ref. 3, p. 414). There is disagreement over the numerical values of the risk coefficients and their applicability for different types of radiation and for different population groups and the results should be considered as approximations. Table 1 indicates the steps in developing risk estimates.

As indicated in the first column, populations at risk of 10^4 (10,000) and 10^6 (1,000,000) are assumed. In column 2, the uniform whole-body equivalent dose received by each member of that population is indicated to be either 10^{-6} (0.000001), 10^{-4} (0.0001), or 10^{-2} (0.01) rems. These ranges of values generally cover the estimated average dose equivalents received and the population sizes according to the calculations contained in the DEIS. It is assumed for purposes of this table that these dose equivalents are received throughout the body of each of the persons in the populations. The products of a value in column 1 and the value in column 2, for a given line, gives the population dose equivalent in person rems indicated in the third column. If the risk coefficient is assumed to be 10^{-4} deaths/rem, the product of a numerical value in the third column and 10^{-4} gives the number of deaths that are presumed to occur as a result of the irradiation. One needs to remember that these are the number of deaths throughout the lifetime of the individuals involved. It is seen that for a population of 1 million persons, uniformly exposed to a dose equivalent of 0.01 rem, one would estimate 1 death from radiation induced cancer during the entire lifetime of all members of that population.

Table 1

Illustration of Method to Calculate Radiation Induced Deaths*

If Population is	and Dose Equivalent is (rem)	Population Dose Equivalent (person-rem)	Presumed Deaths
10,000	0.000001	0.01	0.000001
10,000	0.0001	1	0.0001
10,000	0.01	100	0.01
1,000,000	0.0001	100	0.01
1,000,000	0.01	10,000	1

*The numbers used for this example are for illustrative purposes only and are not directly applicable to WIPP.

Comparison with Natural Background

In the absence of information on health effects, it is customary to compare man-made radiation exposure to that which occurs from natural background and the DEIS has done this. Dose commitments from ionizing radiation are presented in the DEIS for time periods ranging from a few days to one year and fifty years. These dose commitments are compared to the dose equivalent from natural background over a few hours, one year, fifty years and seventy years. In some cases, the DEIS inappropriately used dissimilar time periods.

Doses in which the radiation is absorbed over one year should only be compared to natural background radiation over a similar time period. Similarly, doses from radiation that occur over fifty years can be compared to fifty years cumulative total from natural background radiation exposure. Examples where this has not been done include Tables 9-18, 9-19, 9-25 and in the discussion of Table 6-13.

References

1. International Commission on Radiological Protection. Recommendations of the International Commission on Radiological Protection (ICRP Publication 26), 1977.

2. International Commission on Radiological Protection. Problems Involved in Developing an Index of Harm (ICRP Publication 27), 1977.

3. United Nations Scientific Committee on the Effects of Atomic Radiation. Sources and Effects of Ionizing Radiation, 1977.

INVENTORY OF RADIOACTIVITY

(DEIS Chapters 2, 6, 9, E)

TRU Waste Inventory

The DEIS stated that: "The quantities of waste stored at various storage locations are not precisely known; that is, the estimations of these quantities...have large uncertainties associated with them. In addition, it has not yet been decided which locations will actually be shipping waste to the WIPP reference repository" (6-11;3). EEG recognizes DOE's difficulty in obtaining an accurate inventory of TRU waste to be stored at the WIPP. The calculated inventories used by EEG are based on information in the DEIS. If this information is incomplete or incorrect on the quantity or isotopic composition, then the dose and concentration estimates will also be in error.

Activity Estimates

Since the DEIS did not include estimates of the total volume or activity of transuranic (TRU) waste, EEG has prepared estimates of the amounts to be located in the repository (see Tables 2, 3 and 5) and the inventories for truck and rail shipments (see Tables 3 and 4) and recommends that such information be included in the final EIS.

EEG's estimates are based on information in the DEIS, particularly waste volume and shipment projections in Chapter 6 and radionuclide concentrations in Appendix E. The period of repository operation was taken to be thirty years, because of the DEIS statement that "...the plant is designed for a useful life of at least 30 years" (1-4;8). Details of the calculations appear in Appendix 1.

Radionuclide Concentrations

It is recognized that the amounts of radionuclides present in containers of a given type differ greatly, making it difficult to get an accurate inventory. However, there are inconsistencies in the DEIS. The average plutonium content listed in Appendix E is 8 grams per box and 13 grams per drum for CH-TRU waste (Tables E-1, E-2, pp. E-2,3), whereas Table 9-43 (9-103) in the DEIS leads to higher estimates. This table gives projected CH-TRU waste isotopic concentrations (in Ci/liter and g/liter) 100 and 1000 years after burial. The DEIS notes that "the inventory listed in these tables is not precisely the same as that shown in Appendix E" and states that "actual assay data from the Idaho National Engineering Laboratory" were used (9-102). The data in Table 9-43 appear consistent with results of an INEL assay reported by Bingham and Barr (SAND 78-1730).

Table 6 summarizes the differing actinide concentrations obtained from Tables E-1, E-2 and 9-43. Box concentrations obtained from Table E-2 are an order of magnitude lower than drum concentrations obtained from Table E-1. The INEL assay concentrations are slightly higher than the Table E-1 drum concentrations. Does this mean that new data suggest higher box and drum concentrations than those given in Tables E-1 and E-2?

Spent Fuel Inventory

The spent fuel inventory in the DEIS agrees with other published inventories (references 1, 2, 3). The computer program used to derive these inventories was ORIGEN (ref.4). ORIGEN has been evaluated, tested and distributed by the U. S. Department of Energy, Radiation Shielding Information Center (ref. 5) and is being used worldwide as an accepted inventory code by the nuclear field. Correlation between measurements and calculations has generally been good (ref. 6).

EEG notes that the activation product Carbon-14 was not included in the spent fuel inventory. It will be present in greater quantities than I-129 (ref. 7), has a half-life of 5730 years, and is very mobile in the environment. It has been projected to cause the major part of the population dose from nuclear reactors (ref. 8). This omission should be explained.

Table 2

Estimated 30-year Repository TRU Waste Inventory*

Isotope	Activity (Ci)
Plutonium-238	3.5×10^4
Plutonium-239	4.8×10^5
Plutonium-240	1.2×10^5
Plutonium-241**	1.2×10^6
Americium-241**	5.5×10^4

*These estimates include the effects of decay and ingrowth.

**Plutonium-241 (half-life = 13 years) is a beta emitter which decays to Americium-241 (half-life = 460 years).

Table 3

Inventory of Radioactivity*

CH-TRU

Isotope	Repository Total (Ci)
Pu-238	4.0×10^4
Pu-239	4.7×10^5
Pu-240	1.2×10^5
Pu-241**	2.8×10^6
Am-241	7.7×10^3
Total	3.4×10^6

RH-TRU

Isotope	Repository Total (Ci)	Activity in a rail shipment (Ci)	Activity in a truck shipment (Ci)
Sr-90/Y-90	2.8×10^6	2100.	420.
Co-60	1.7×10^4	13.	2.6
Ru-106/Rh-106	2.5×10^4	18.5	3.7
Cs-137/Ba-137m	1.4×10^4	10.5	2.1
Eu-152	3.6×10^3	2.7	.53
Eu-154	1.4×10^4	10.5	2.1
Th-232	8.0	.006	.001
U-234	6.5×10^{-2}	(4.9×10^{-5})	(9.7×10^{-6})
U-235	2.7	.002	(4.1×10^{-4})
U-238	6.0×10	.044	.009
Pu-238	7.4×10^2	.55	.11
Pu-239	8.7×10^3	6.5	1.3
Pu-240	2.0×10^3	1.5	.3
Pu-241**	5.2×10^4	39.	7.8
Am-241	1.4×10^2	.11	.02
Cm-244	3.6×10^4	27.	5.3
Total	3.0×10^6	2.2×10^3	4.5×10^2

*30 years of new production are added to the backlog (see DEIS Tables 6-2, 6-6). The effects of decay and ingrowth are not included in these estimates.

**Beta emitter with a 13 year half-life.

Table 4

CH-TRU Shipment Inventories

Isotope	Ci/drum (Table E-1)	Ci/box (Table E-2)	Ci/rail shipment of drums	Ci/truck shipment of drums	Ci/rail shipment of boxes	Ci/truck shipment of boxes
Pu-238	4.1×10^{-2}	6.5×10^{-2}	4.9	1.7	1.6	.52
Pu-239	4.8×10^{-1}	7.5×10^{-1}	58.	20.2	18.	6.0
Pu-240	1.2×10^{-1}	1.8×10^{-1}	14.	5.0	4.3	1.4
Pu-241	2.9	4.6	350.	120.0	110.	37.
Am-241	7.8×10^{-3}	1.2×10^{-2}	0.94	0.33	.29	.10

CH-TRU Shipment Volumes*

Mode	Container	Volume of container (ft^3)	Containers per shipment	Waste volume per shipment (ft^3)
Rail[a]	Box	112	24	2700
Rail	Drum	7.4	120	930
Truck[b]	Box	112	8	900
Truck	Drum	7.4	42	310

[a]ATMX railcar assumed for rail shipment.
[b]Type B container for truck shipment assumed to hold 8 boxes.

*DEIS, Table 6-3(6-12).

Table 5

TRU Waste Volumes ($ft.^3$): 30 Year Repository Totals

Type of Waste	Containers	Backlog*	New waste production per year*	New waste production in 30 years	Backlog + 30 year production
CH	Boxes	7.0×10^5	9.0×10^4	2.7×10^6	3.4×10^6
CH	Drums	2.4×10^6	1.5×10^5	4.5×10^6	6.9×10^6
RH	Canisters	7.7×10^4	6.9×10^3	2.1×10^5	2.8×10^5

* DEIS, Tables 6-2, 6-6 (6-12,6-14).

Table 6

CH TRU Radionuclide Concentrations

Isotope	Half-life (yrs.)	Radionuclide concentrations (µCi/l): From Table E-1,* in drums	From Table E-2,* in boxes	From INEL assay SAND 78-1730 (p. 87)	From Table 9-43 at 100 years
Pu-238	8.8×10	2.0×10^2	2.0×10	2.4×10^2	1.1×10^2
Pu-239	2.4×10^4	2.3×10^3	2.3×10^2	2.8×10^3	2.8×10^3
Pu-240	6.5×10^3	5.8×10^2	5.6×10	6.8×10^2	6.7×10^2
Pu-241	1.3×10	1.4×10^4	1.4×10^3	1.7×10^4	---
Am-241	4.6×10^2	3.8×10	3.8	4.6×10	4.7×10^2

*Calculated by dividing the expected activities (Ci/drum and Ci/box) in Tables E-1 and E-2 by the container volumes (208 liters for a drum and 3.2×10^3 liters for a box), and multiplying by 10^6 µCi/Ci.

References

1. U.S. Nuclear Regulatory Commission. Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0404), March 1978.

2. U.S. Energy Research and Development Administration. Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle (ERDA-76-43), May 1976.

3. U.S. Environmental Protection Agency. Significant Actinide Activities in the LWR and LMFBR Nuclear Fuel Cycles (EPA-520/3-75-006), 1974.

4. Bell, M.J. ORIGEN - The ORNL Isotope Generation and Depletion Code (ORNL-4628), 1973.

5. Oak Ridge National Laboratory. Radiation Shielding Information Center. ORIGEN. Isotope Generation and Depletion Code - Matrix Exponential Method (CCC-127), October 1978.

6. Industridepartment Energikommissionen. Disposal of High Active Nuclear Fuel Waste. A Critical Review of the Nuclear Fuel Safety (KBS) Project on Final Disposal of Vitrified High Active Fuel Wastes.

7. U.S. Department of Energy. Management of Commerically Generated Radioactive Waste (DOE/EIS-0046D), Draft Environmental Impact Statement, April 1979.

8. U.S. Nuclear Regulatory Commission. Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Cooled Reactors (NUREG-0002), 1976.

WASTE ACCEPTANCE CRITERIA
(DEIS Chapter 5)

Major Conclusions

1) Since the waste acceptance criteria are under active development by the DOE, they are not in the DEIS. However, DOE has been furnishing EEG with material on the criteria as they are being developed by the Waste Acceptance Criteria Steering Committee for review.

   Until such time as waste acceptance criteria are defined, the radiological consequences of operations and accidents cannot be fully analyzed. Three major concerns of both the DOE and the EEG are the presence in the TRU waste of:

   a) gas from organic decomposition

   b) combustible materials; and

   c) respirable particles.

2) Some interim criteria on RH-TRU waste in Table 5-1 are less stringent than criteria for CH-TRU waste.

3) It is essential that the retrievability of the radioactive wastes be examined in detail as to criteria, logistics, procedures, integrity of containers, hazards to workers, and hazards to the general population.

CH and RH-TRU Criteria

The review of the interim waste acceptance criteria for CH and RH-TRU waste (5-4, 5-5) led to several concerns:

1) Combustibility. A limit has not been placed on the amount of combustible materials which may be placed in individual containers or collectively in the underground storage rooms. EEG is

concerned since fire is listed among the possible accidents.

2) Gas Generation. Gas-generating materials in CH-TRU waste are limited to 10% by weight in any single storage room. No limit is given for RH-TRU waste. The 10% limit shown would not provide meaningful guidance to the individuals packing the containers. How much gas-generating waste will be accepted? How much gas and what type can be generated? What will be the long term effects of gas generation? On pp. 9-133 to 9-136, gas generation and its possible effects on the repository are discussed. These problems are being investigated by the DOE. Calculations have been carried out which indicate that gas pressures in the repository "might exceed lithostatic pressures at the repository depths" (9-136; details are not given).

   The statement is made that "To insure that evolved gases will not fracture the rock overlying the reference repository, the waste acceptance criteria will limit the amount of gas-producing material in the waste accepted for burial" (9-136;3). Gas generation criteria should be very specific and should be supported by evidence of their adequacy. Detailed guidance should be provided to waste-generating facilities in order to help them meet the criteria. It is not clear how waste-generating facilities will determine the content of gas-producing materials in previously stored wastes.

3) Pyrophorics. EEG believes criteria should specify the amounts of pyrophoric material permitted in both CH and RH-TRU waste.

4) Hazardous Material. What non-radioactive hazardous materials must WIPP be prepared to handle, and in what total quantity? What criteria will the WIPP operator use in authorizing such material? (See the reference to "Hazardous materials" in Table 5.1). What calculations have been done on the potential reentry of these materials to the biosphere? Some of them could be hazardous for periods longer than the radioactive wastes.

5) Thermal Power. A criterion of 0.1 W/ft^3 is given for color coding and identification for the CH-TRU waste. A criterion should be given for RH-TRU as well.

The explanation given for not restricting combustibility, gas generation or thermal power for RH-TRU waste is that "quantities [of RH-TRU waste] are insignificant, and processing will probably not be available" (5-4). The DEIS refers to RH-TRU waste as constituting "a small fraction (about 2% by volume) of the TRU waste generated by the DOE complex" (5-6;1), and goes on to state that "Even if all the RH-TRU waste were gas-producing or combustible, there would probably not be enough to cause significant problems at the WIPP reference repository" (5-6;2). EEG's estimates of total respository TRU wastes volumes (see Table 5, p. 20) are 1.0×10^7 cubic feet of CH-TRU and 2.8×10^5 cubic feet of RH-TRU. If this amount of RH-TRU material is to be considered insignificant, calculations in the final EIS should support this conclusion. Furthermore, EEG estimates the average level of radioactivity of material in a shipment of RH-TRU waste to be 2.2×10^3 Ci/rail shipment and 4.5×10^2 Ci/truck shipment (see Table 3, p. 18). The degree of mobility and combustibility of wastes will be a factor in determining the consequences of a transportation accident.

Previously Stored TRU Wastes

To what extent will previously stored TRU waste be re-examined, treated as may be necessary (incineration, immobilization of ash, etc.) and repackaged prior to shipment to WIPP? Some of the wastes proposed for WIPP may have been in storage as long as 20 years. The characteristics of the wastes and the containers could have changed substantially in that time, rendering either the wastes, the containers, or both unsuitable for storage at the WIPP. According to reference 16, it is doubtful that 17C drums would meet the leak test requirements of the ANSI standard 14.5, particularly after a decade of storage. The integrity of the drums without polyethylene liners is particularly suspect. Is the no-leak requirement of ANSI 14.5 or the requirements of 10

CFR 71.42 (b) applicable to the packages to be shipped to the WIPP?

Information in reference 11 indicated that there was non-uniformity among the various suppliers of the TRU wastes in the way in which wastes were stored and data recorded. The waste acceptance criteria should clearly establish uniform practices which are consistent with the needs of WIPP. For example, reference 22 indicated that all Pu-238 contaminated waste in drums which have been previously stored for significant time periods should be considered potentially explosive until individual drum analyses are conducted. This would imply that such drums would not meet a criterion prohibiting explosive material in CH and RH-TRU waste containers.

Impacts of Processing

Processing of CH-TRU waste by slagging pyrolysis was presented in the DEIS as a strong possibility (5-9). This raises certain questions. Would slagging pyrolysis facilities be set up at all sites from which waste would be sent? Would pyrolysis take place only at INEL? In this case, would waste from other locations be sent to Idaho for processing or would the waste acceptance criteria be relaxed for waste from other locations? Will some of the waste be processed at the repository site? This would have implications in the area of transportation and operational exposures.

The statement on page 5-7;3 that "the waste-acceptance criteria finally selected will produce smaller impacts than the impacts calculated from the assumed criteria" seems premature.

Detailed Comments

5-3, 5-4 What is the rationale behind "Large suppliers must observe another limit: the surface-dose rate of their shipment averaged over 3 months, must be no higher than 10 mrem/hr" (5-3:7)? What is the limit for small suppliers? Also, what are the surface contamination limits (Ref. 49 CFR 143.398)?

5-4 Have criteria been developed for spent fuel and High Level Waste?

5-5 Surface Contamination Criteria reference should be 49 CFR 173.398 instead of 49 CFR 73.398 in Table 5-1.

5-4,5-7;4 Table 5-1 stated that small quantities of pyrophoric material will be acceptable. Page 5-7 stated that environmental impacts were assessed under the assumption that no pyrophoric material would be accepted.

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20. "Comments on Proposed Interim Waste Acceptance Criteria for Contact-Handled TRU Waste," a letter from D.E. Large to H.H. Irby, October 5, 1978.

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22. "An Interim Summary of Experimental Programs for the WIPP TRU Waste Acceptance Criteria," (Draft), January 1979.

23. U.S. Department of Energy. Project Overview. Waste Isolation Pilot Plant (WIPP-DOE-21), January 10, 1979.

TRANSPORTATION
(DEIS Chapter 6)

Major Conclusions

1) The equations used in the calculations of radiation doses from the normal transportation of the radioactive wastes have been derived by the EEG and the calculated doses were found to be in agreement with those presented in the DEIS. EEG has made a critical evaluation of the assumptions used in order to determine the validity of these dose estimates. These doses would represent small additions to the general population radiation exposure in comparison to other man-made radiation sources and natural background.

2) EEG has identified a number of additional dosage calculations to be performed and these are listed on pages 90-92.

3) Radiation exposures from deliberate acts of sabotage in the transportation of radioactive materials could be considerably higher than those from conventional traffic and rail accidents. The DEIS assumed there would not be a difference.

4) Some of the assumptions for accidents may not be sufficiently conservative. The following possibilities were not included in the DEIS calculations.

   a) A fire occurring during a rail accident involving contact handled transuranic wastes (CH-TRU).

   b) Leakage of remote handled transuranic wastes (RH-TRU) from a container following a rail accident.

   c) Ingestion of radioactive material following an airborne release.

5) Consideration should be given to shipping all the radioactive waste by rail wherever the calculations show that the potential radiation exposures to people would be reduced. This is consistent with the concept in radiological health that all unnecessary radiation exposure be avoided and exposures kept as low as reasonably achievable (ALARA). Consideration should also be given to restricting shipments in icy weather.

6) The maximum dose to people from atmospheric dispersion can be closer than the 0.5 miles calculated in the DEIS if the plume does not rise to a height of 20 meters at the time of release or if more unstable atmospheric conditions occur.

Radiation Doses from the Normal Transportation of Radioactive Wastes

The radionuclide inventories for truck and rail shipments of both CH-TRU and RH-TRU wastes were calculated by EEG and are shown in Tables 3 and 4. Derivations of the equations used in the calculation of radiation exposure from the normal transportation of radioactive wastes are shown in Appendix II.

Comparison of calculated doses in Table 7 and 8 show substantial agreement between the annual doses in the DEIS and those calculated by the EEG using NUREG-0170. These exposures represent small additions to normal background radiation and man-made radiation exposure.

The doses given in the DEIS are population doses. No information was presented on potential doses to individuals. Projections of maximum individual doses during normal transport should be provided.

TABLE 7

Calculated Radiation Doses from Normal Transportation of CH-TRU Waste

Annual Population Dose (man-rem)

Origin and Mode	Population surrounding route while moving			Passing Motorists			Population surrounding route while stopped			Crew		
	EIS	EEG	EIS/EEG*	EIS	EEG	EIS/EEG*	EIS	EEG	EIS/EEG*	EIS	EEG	EIS/EEG*
INEL(box) Truck	.096	.14	.68	.049	.044	1.1	.16	.16	1.0	2.4	2.6	.92
Rail	.34	.21	1.6				.007	.007	1.0	.01	.009	1.1
INEL(drum) Truck	.59	.88	.68	.31	.27	1.1	.99	.99	1.0	15	16	.93
Rail	2.1	1.3	1.7				.04	.04	1.0	.08	.05	1.6
Hanford Truck	.52	.75	.69	.27	.24	1.1				13	14	.92
Rail	1.6	1.0	1.6							.06	.04	1.5
LASL Truck	.15	.22	.68									
SRP Truck	.06	.09	.67									
Rail	.16	.10	1.6									

*Ratio of EIS Dose to EEG Dose.

TABLE 8

Calculated Radiation Doses from Normal Transportation of RH-TRU Waste

Origin and Mode	Annual Population Dose (man-rem)		
	Population surrounding route while moving		
	EIS	EEG	EIS/EEG*
INEL			
Truck	.29	.44	.66
Rail	.26	.37	.70
Hanford			
Truck	.16	.25	.67
Rail	.13	.19	.68

*Ratio of EIS Dose to EEG Dose.

Radiation Doses from Transportation Accidents

According to the DEIS, the barriers limiting the release of radioactivity to the environment following an accident result in only 0.004% of the radioactive material being respirable following a rail accident with CH-TRU wastes (6-23) and 0.015% following a truck accident (6-24). The only radioactive material that would be released in a rail accident involving RH-TRU waste would be 0.1% of the Cs-137 activity (6-25). The references from which the values of these barriers were selected in many instances do not show the basis on which they were derived. Empirical evidence needs to be developed under experimental conditions to confirm the reasonableness of many of these values.

Consolidated calculations for atmospheric dispersion coefficients (χ/Q) are to be found in Appendix V and are in reasonable agreement with those presented in the DEIS when the same assumptions are used.

If the plume in an airborne release does not rise to a height of 20 meters, then larger χ/Q values can be obtained and the maximum dose can occur at distances closer than 0.5 miles.

The following calculations were performed for a rail accident with spent fuel, using the assumptions shown in the DEIS. The results obtained were in substantive agreement with those in the DEIS.

Table 9

Dose to an Individual[a]

	Dose Commitment, (rem)		
Organ	DEIS	EEG	DEIS/EEG
Bone	1.2	1.2	1.0
Lung	0.3	0.2	1.5
Whole Body	1.1	0.9	1.2

[a]Maximum dose to an individual one-half mile from the accident. Details are shown in Appendix II.

Although the CH and RH-TRU doses to the general population from normal transportation are not considered to be of public health significance in comparison to other radiation sources in the environment, serious consideration should be given to shipping all the radioactive waste by rail, wherever the calculations show that the actual and potential radiation exposures will be reduced. This is consistent with the concept in radiological health that all unnecessary radiation exposure be avoided and exposures kept as low as reasonably achievable (ALARA).

The approximate total distance to be driven by the trucks will be: (600 shipments/year) x (1000 miles/shipment) x (30 years) = 18 million truck-miles. Calculations of injuries and accidents unrelated to radiation should be performed for rail and truck shipments. Consideration should also be given to restricting shipments in icy weather.

The following need to be clarified:

1) Who is responsible for accident response?

2) What response capability exists now and is planned for the future?

3) What state and local assistance is required?

4) Who equips, trains and funds the people?

5) Who pays for deployment, if required?

6) Who assumes financial risk for accidents?

Additional Dosage Estimates

There are a number of additional dosage estimates that need to be calculated:

1) Radiation exposure from acts of sabotage in the transportation of radioactive waste materials. The amounts of radioactivity released could be greater than those released in accidents. Are there any sabotage scenarios that could produce criticality? (Occupational, General Population)

2) Radiation exposure to emergency workers such as police and firemen following a transportation accident. (Occupational)

3) Exposure to a person stopped in an automobile next to a radioactive waste truck at a red light or in a traffic jam. (General Population)

4) Exposure from shipments of retrieved radioactive waste following the completion of the high level waste experiments. Containers could be bent, damaged or under pressure from gas generated by decomposed organic material. (Occupational, General Population)

5) Ingestion from contamination of a water supply or crops following an airborne release. (General Population)

6) Material resulting from decommissioning and dismantling of weapons production facilities in Hanford. While the DEIS assumes that none of the 5 to 95 million cubic feet of material will be shipped to WIPP, it notes that the WIPP will have the capacity to receive some of this TRU waste (2-22;2). (Occupational, General Population)

7) Consideration of a diffuse source of radioactivity rather than a point source in transportation calculations.

8) Population dose estimates were provided in man-rems. They do not identify the maximum dose to an individual.

Detailed Comments

6-4 Consideration should be given to limiting truck shipments during icy weather from sites such as LASL.

6-7;5 According to the DEIS (5-7), the shipping containers will not contain pyrophoric material. Can depleted uranium be pyrophoric under certain circumstances?

6-8 The interaction of the pyrophoric material permitted on page 5-4 and the hydrogenous material layered in the cask construction is not addressed in the transportation fire scenarios.

6-9 Will DOE or the carrier select the routes to be taken? Are there always two drivers or could the shipment be left unattended during stops?

6-12, 6-14 No information is provided on waste used in HLW experiments such as:

radionuclides
amount of radioactivity
type of container
form of material

What quantity, types, configuration of non-radioactive wastes are expected to be shipped as a contaminant in the radioactive waste?

6-17;2 The last line should read "from natural background" and the time period should be one year.

6-18;1 The numerical value (1.0 person-rem) does not agree with the value shown in Table 6-10, and unit "person-rem dose" is inappropriate. We are not able to confirm the figure of 0.02%.

6-18 Tables of doses include values for occupational and general population. They should be separated since different criteria apply to them.

6-23 What is the basis for assuming no ingestion of radioactivity from an airborne release following a transportation accident? Also, a body of water could be contaminated.

The assumptions for meteorology coupled with a release height of 20 meters for the aerosol result in a maximum dose occurring 0.5 miles downwind. Other assumptions can produce larger exposures at closer distances.

6-23;4 What is the basis for the assumption that contaminated food would immediately be taken out of distribution? Such administrative action has not always been possible or necessary.

6-23;5 The hypothetical rail accident involving CH-TRU waste calculates that only 0.004% of the radioactive material in the shipment would be airborne and respirable in a release. What is the basis for each of the factors in the calculation?

6-24;2 The 1978 Shefelbine reference is not adequate to justify the assumption that 10% of the waste is in powder form.

6-25, 6-26 Would radionuclides other than Kr-85 and Cs be volatilized in the fuel element accidents involving fire?

6-25;3 The hypothetical rail accident of a violent wreck with a fire for one hour involving RH-TRU waste assumes that only 0.1% of the Cs-137 would be released. No other radionuclide listed in Table E-3 on page E-4 would be released to the environment. What is the basis and rationale for these numerical values?

6-25;5 The hypothetical rail accident of a violent wreck with a fire for one hour involving spent fuel waste assumes that only 30% of Kr-85 and 0.1% of Cs-134/137 would be released. No other radionuclide listed in Table E-3 on page E-4 would be released to the environment. What is the basis for these statements?

6-26 The hypothetical accident involving the shipment of spent fuel only considers cesium and krypton being released; but the operational accident for spent fuel on 9-37;1 notes that tritium, Krypton-85, and Iodine-129 are easily released.

6-26, 6-23 It was assumed that there was no route of exposure except inhalation for the accident. Administrative control cannot be relied upon in this type of incident and other routes of exposure must be considered.

6-26 The assumptions that many nuclides including tritium, Iodine-129 (and others) are released from a damaged spent-fuel assembly in the WIPP above ground facility, are different than the assumptions discussed on 6-26 for a rail accident. These differences should be resolved.

6-26, Tables 6-14,6-15 We were able to reproduce the spent fuel bone dose of 4200 man-rem shown in Table 6-14. We were unable to reproduce the population dose commitments in Table 6-15.

6-27 Drums were considered in the scenarios involving transportation accidents but boxes were not. An explanation is needed.

Surface contamination tests upon arrival at the repository are needed.

6-27, Table 6-3 Using the assumptions of the spent fuel transportation accident outlined in the DEIS, calculations by EEG were in general agreement with the dose to individuals given in Table 6-3.

6-27, 6-18 The various radiation exposures from the shipment by truck are greater than by rail (annual man-rem doses from transportation of CH-TRU, RH-TRU and spent fuel, pp. 6-18 and 6-19). The same is true for accidents (p. 6-28). Consideration should be given to transporting all the radioactive wastes by rail which would reduce the expected and potential radiation exposure in accordance with the ALARA (as low as reasonably achievable) concept.

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SITE CHARACTERIZATION

(DEIS Chapter 7)

Major Conclusions

The EEG has evaluated the Geological Characterization Report (GCR) Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico (SAND 78-1596), December, 1978, which is the source of most of the geological and hydrological data in the DEIS; the complete review is in Appendix III. Conclusions and summary statements in the DEIS, as well as in the GCR, did not take into account certain important problems related to geologic and hydrologic factors. The following is a summary of EEG's major concerns:

1) Seven wells within nine miles of the periphery of the WIPP site have encountered brine reservoirs under artesian pressure. The origin, evolution, frequency of occurrence and size of these high pressure brine reservoirs were not adequately addressed in either the DEIS or the GCR.

2) There is at least one confirmed occurrence of a "chimney... with clay cemented brecciated rock", commonly called a breccia pipe, approximately seven miles from the WIPP site (Mississippi Chemical Corporation potash mine). Several other possible breccia pipes are under various stages of investigation. The origin, evolution and frequency of occurrence of these features must be better understood. They may be localized deep dissolution features which originate in the lower portion of the evaporites and migrate upward. Such localized dissolution features could now exist or develop later beneath the proposed site.

3) The DEIS and the GCR assumed that surface or shallow dissolution is the dominant process of salt removal from the evaporite beds. However, deep dissolution may be causing a preferential removal of the salt horizon which is proposed for the repository.

4) The lithology of the repository horizons is described on page 7-24 of the DEIS and parts of the lithologic log of the ERDA-9 hole are shown in Figures 1 and 2 of this section (from Fig. 4.3-3B of the GCR). These sections of the logs describe the lithology of the repository horizons for CH and RH zones as shown on Figure 4.3-3A of the GCR. The logs show the presence of clay, anhydrite and polyhalite in addition to halite, as the constituents of both repository horizons. The presence of these impurities should be taken into account in evaluating physical, hydrological, thermal and strength characteristics of "rock salt" from the repository horizons.

5) The values of hydrologic parameters (e.g. hydraulic conductivity, distribution coefficients, and effective porosity) can vary over a large range and the DEIS provides such information. In addition, potentiometric surface maps, hydraulic gradients and flow paths have been constructed on the basis of limited data. In some cases (e.g. hydraulic conductivity) the DEIS gives a range of measured values. Ranges should be assessed in all cases, particularly for distribution coefficients. The distribution coefficient (K_d), which affects the speed with which a given radionuclide is transported in groundwater, can be affected by rock type, extent of fracture permeability, water quality characteristics, competing ion effects, and the chemical form of the radionuclide of interest. Values obtained for a given nuclide in a given rock formation have been observed to vary by several orders of magnitude.

6) More information should be given in the final EIS on surface water hydrology in the region surrounding the WIPP site.

Items 1, 2 and 3 have been discussed in detail in Appendix III (EEG's Review Comments on the GCR). No new information on these items is presented in the DEIS. The DEIS concluded that there was no evidence of brine reservoirs or ongoing deep dissolution at the WIPP site. EEG questions the basis of these conclusions in the Review of the GCR.

Lithology of Proposed Emplacement Horizons

According to the DEIS (7-21;4), the repository horizons were selected due to the presence of relatively pure salt layers. When the NAS-NRC Committee (Ref. 2) recommended salt as the most likely geologic medium for radioactive waste disposal, it placed strong emphasis on the "purity" of a bedded salt formation so that its thermal and physical properties could be predicted. The presence of impurities can affect the properties of bedded salt. Examples are:

1) Argillaceous (clayey) layers in bedded salt may provide conduits for the migration of water to and from the repository. While some of the impurities found in bedded salt have lower permeabilities than halite, a path for migration of water may be created along the contact between two layers of differing lithology.

2) A subgroup of the Interagency Review Group on Nuclear Waste Management commented on salt formations: "The hydrologic regimes in which anhydrite occurs are characterized by flows along bedded planes, but locally channeling (cavern formation) occurs in anhydrite similar to that in limestone and gypsum" (Ref. 1).

3) The chemical reactions which may take place in the vicinity of high level waste, accelerated by elevated temperatures and high pressures become more complex and unpredictable when the host rock is heterogeneous.

4) Because thermal conductivities of clays and polyhalite are very different from that of halite, the dissipation of heat resulting from the high level wastes will not be uniform around the waste. This may result in cracking, parting of seams and uneven concentration of moisture.

These potential problems are not discussed in the DEIS, although the lithology of the repository horizons is presented as follows: "The basic mineral of both repository horizons is halite. Also present are anhydrite, polyhalite, quartz and a suite of clay minerals (illite, chlorite, talc, serpentine, and expendable clays). Halite beds within the emplacement horizons are about 97% halite. Most of the remainder is anhydrite" (7-24;5). Note that the last line quoted refers to 97% halite in halite beds and not in the total repository horizon. The lithologic log for the CH repository horizons (Figure 1) shows anhydrite beds which are 0.2, 0.7 and 0.9 feet thick and most halite layers are "argillic and polyhalitic". The RH repository rocks are mostly "anhydritic and argillic halite" (Figure 2). The bottom 20 feet of the RH zone is primarily "dense anhydrite".

Unidentified Structures

A lamprophyre dike or a series of en-echelon dikes were reported within six miles of the periphery of the WIPP site. If associated igneous bodies underlie the WIPP site, they could affect the integrity of the salt beds. The cross-section on Figure 4.4-5 in the GCR shows faults in the Castile directly below the WIPP site and the contour map on Figure 4.4-6 shows confined faults on top of the Castile. These should be explained.

Surface Water Hydrology

There is not enough information given on surface water hydrology in the region around the site to enable one to adequately evaluate the effect of the site on local water resources. Since surface runoff is a potential pathway to spread contamination, it needs to be evaluated in much more detail. This evaluation should include runoff from floods with a 100-year and 1,000-year return period. The fate of this runoff water after it reaches Nash Draw (or elsewhere) needs to be evaluated. A description of

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Ground Water Hydrology

The ground water data was largely obtained on the Rustler and deeper aquifers and was used to evalute the role of these aquifers in transporting radionuclides away from the site. Another pathway of exposure would be from wells drilled into the Rustler, Santa Rosa Sandstone or other shallow lenses near the site, and used for individual water supplies, gardens or stock watering. More information is needed on present and potential well water use, quantities of water available, effect of surface recharge, and potential for the well water to be contaminated by the Bell Canyon or Rustler aquifers.

Climatic Changes

Based on the evidence presented on page H-62 and H-63 of the DEIS, the present interglacial period may last another 4,000-5,000 years followed by a cooling trend culminating in another glacial age. In that case, the climate near the WIPP site may be significantly cooler and wetter in 10,000-15,000 years. EEG recommends that long range modeling take into account plausible future climatic changes in hydrological regime.

Figure 1

From GCR, Figure 4.3-3B
CH repository horizon from 2074 to 2176 feet

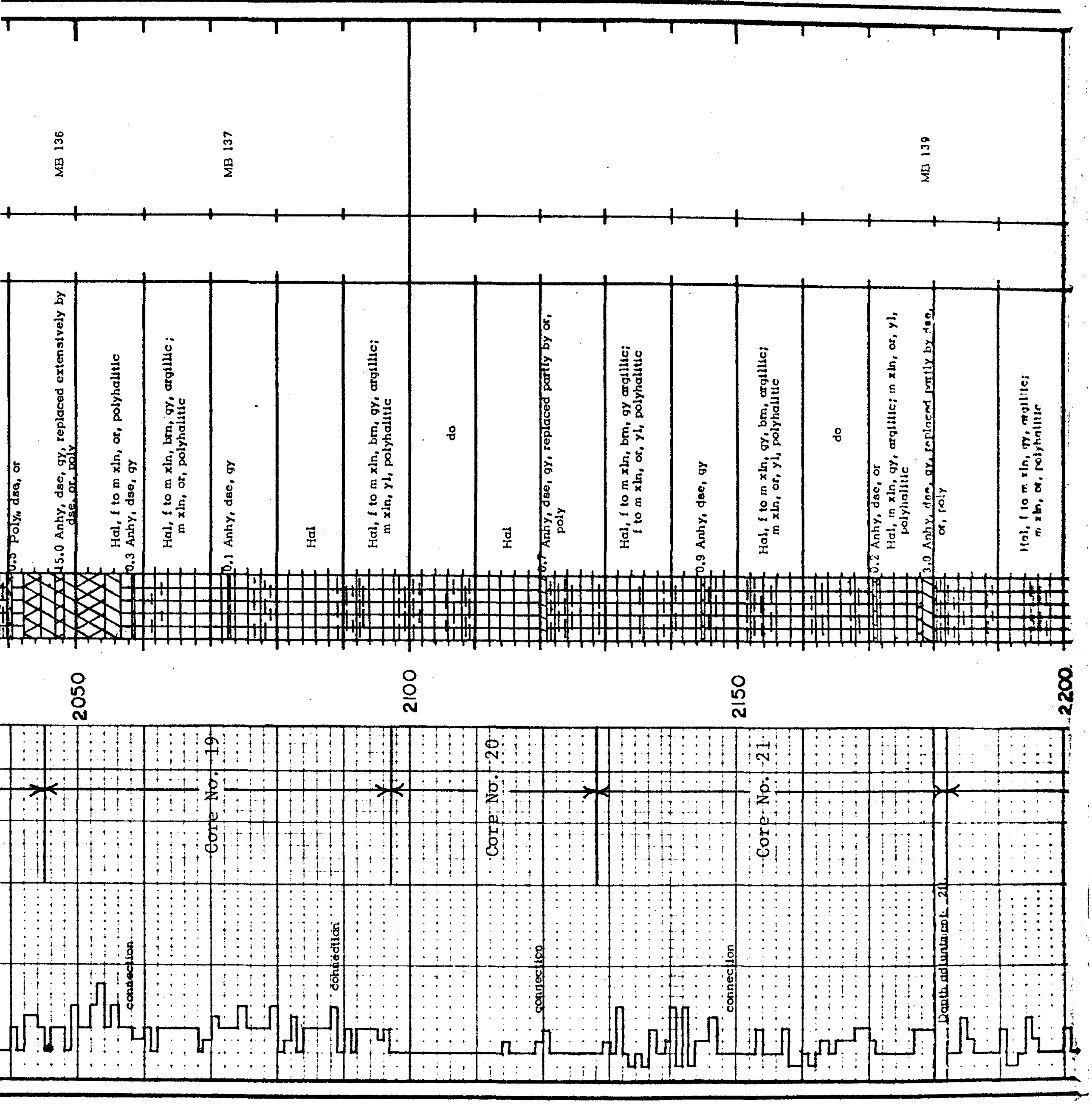
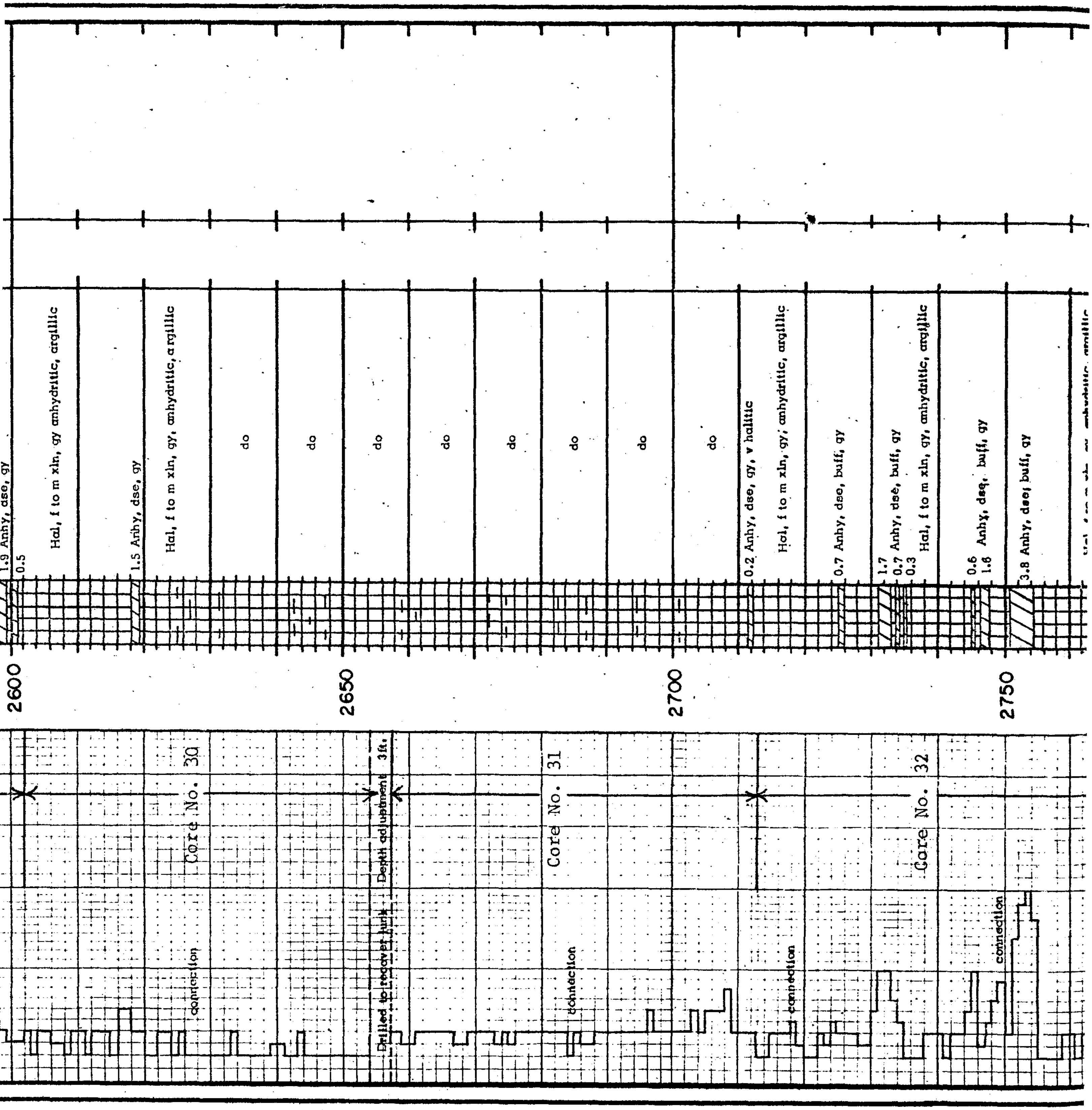


Figure 2

From GCR, Figure 4.3-3B
RH repository horizon from 2620 to 2730 feet



Detailed Comments

7-17;2 Section 17, T22S, R31E is in control Zone III.

7-26;7 What is a "depositional-growth fault" and what is its significance?

7-30;9 What is the status of the investigation of these faults?

7-47;3 It should be "approximately 3300 square miles" instead of "1 million sq. mi." (See Fig. 7-17).

7-65;1 A map and a cross-section showing locations of brine pockets encountered and their stratigraphic locations should be included in the final EIS.

7-65;4 Why has the model assumed a flow path in the Rustler directly to Malaga Bend to the Pecos River. (See Fig. K-5). Couldn't the water seep out in Nash Draw since the top of the Salado is exposed in Nash Draw?

7-69;1 Why is it assumed that the water will come out at Malaga Bend? Why not Laguna Grande de la Sal? What is the origin of the water in Laguna Grande de la Sal?

7-72;6 What is the basis for the assumption of the origin of water for Laguna Plata and Laguna Gatuna?

7-74;1 "Weaver Pipe" could be an example of a breccia pipe which has no surface expression. Could there be such breccia pipes at the WIPP site which have no surface expression?

7-75;3 Late Pleistocene (Wisconsin) was from 40,000 years to 15,000 years B.P. Such climate changes can occur in the future as well.

7-75;7 The GCR (August 1978) referred to current and future studies to evaluate deep dissolution (6-46;1). The EIS (April 1979) concluded "In any case, deep dissolution does not occur near the site". Have these studies been concluded?

References

1. Interagency Review Group on Nuclear Waste Management, Subgroup for Alternative Technology Strategies. "Isolation of Radioactive Waste in Geologic Repositories: Status of Scientific and Technological Knowledge," Subgroup Report on Alternative Technology Strategies for the Isolation of Nuclear Waste (TID-28818 Draft), Appendix A, October 1978.

2. National Academy of Sciences-National Research Council. Committee on Waste Disposal. The Disposal of Radioactive Waste on Land (Publication 519), April 1957.

OPERATION OF THE REPOSITORY
(DEIS Chapter 8)

Major Conclusions

1) Present plans would permit public access to Zones II, III and IV during operations. Also, there is private land 2.8 miles downwind (northwest) of the center of the site where building could occur. Consideration should be given to the radiological air quality and noise environments at these locations in addition to those at the James Ranch.

2) Calculations of radionuclide releases from routine operations agree with those in the DEIS when the same assumptions are used. However, several assumptions used in estimating the amount of radioactivity released are unverified.

3) Radon emissions from natural radioactivity in the repository have not been measured in soil, mined rock, and the proposed waste horizons. Radon should be measured to see if levels might be high enough to be a problem for underground workers and a source of radiation exposure to the public from the excavated salt.

4) Other than radon, the present radiological monitoring program appears satisfactory for the next several years.

5) From the limited information provided in the DEIS on the high level waste experimental program, 9-90 million curies of radioactivity may be involved in the experiments with full-size canisters. In this case, the experimental waste could be the most significant factor in the analysis of potential radiation exposures during the operational phase of the repository. This was not considered in the DEIS calculation of radiation doses.

6) All high level waste used in experiments is scheduled to be retrieved and all TRU waste and spent fuel elements are to be in a

retrievable condition. The DEIS does not address such important items as the criteria for retrieval, the hazards to workers, and hazards to the public. More information is needed before the feasibility of retrievability can be evaluated.

7) The options of decommissioning are adequately covered for the present. The advantage and feasibility of control for periods greater than 100 years should be included in this evaluation.

8) From the material presented in the DEIS, one could conclude that Ambient Air Quality Standards may be violated in Zones II, III, and IV, unless certain measures are taken to insure that the standards are met.

9) There will be some degradation of the noise environment due to repository operations and the traffic related to it. More attention needs to be given to mitigating noise.

10) Due to WIPP-induced population growth in surrounding communities, there will be some impact on water quality, water supply, and solid and hazardous waste conditions. The EEG agrees with the DEIS conclusion that, with proper planning, the existing systems are adequate to absorb the increase.

<u>The Site and Its Environs</u>

The entire area of the site and much of the land immediately outside of Zone IV are owned by the Federal or State government. The James Ranch, located 3 miles south-southwest of the site center, is privately owned and occupied and was used to calculate the maximum individual exposures to radioactivity and noise in the DEIS.

However, it may not be conservative to assume that this is the location of the maximum exposed individual, for the following reasons:

1) Private land is located just outside the northwest boundary, 2.8 miles downwind from the site center. From the atmospheric

dispersion coefficients given in Table H-36, calculations of concentrations of airborne effluents would be about 5 times higher than at James Ranch.

2) Plans indicate public access to parts of the site for ranching, recreation and resource extraction. Consequently, people may be as close as one-half mile from the site center.

Analysis of the effects of repository operation on individuals in the population should consider these locations where people will be permitted to live or visit as well as where they live now.

Normal Radiation Releases

Radon Emissions

Radon, a naturally occurring radioactive noble gas is contained in air exhausted from underground mines. Because radon concentrations could be high enough to be a hazard to some underground workers and could result in measurable off-site exposures, the concentrations should be measured.

The DEIS recognized that radon will be present in exhaust air but did not consider the radon from the mined rock storage pile on the surface. Radon concentrations in the DEIS were based on concentrations reported in the Nuclear Regulatory Commission's Final Generic Impact Statement on Mixed Oxide Fuels (GESMO).

Appendix IV contains a more detailed discussion of the possible radon exposure and calculates dosages to the bronchial epithelium as well as the pulmonary lung dose at 0.5 miles, 2.8 miles (NW) and 3.0 miles (SSW). These calculated doses (which are average, rather than upper-bound limits) suggest that potential radon exposures are high enough to require direct measurement at the site to determine actual concentrations.

Operational Releases of Radioactivity

Starting with the assumptions used in Chapter 8, the radiation releases to the environment were calculated by EEG and they agreed with the results listed in the DEIS (Table 8-6). Agreement was also obtained with the quantity of radioactivity collected on HEPA filters reported in Table 8-7.

However, one area of potential disagreement relates to the calculation of releases of radioactivity from underground storage of contaminated containers. The EEG calculations assumed a 4-year accumulation of boxes and drums that were releasing 1% per year of their remaining surface contamination. The EEG calculated release was approximately four times the quantity of released TRU wastes listed in Table 8-6. Was the calculation in the DEIS based on one year's accumulation of containers?

It is not clear from the description in the DEIS what becomes of spilled material from damaged drums and boxes. The DEIS, assumed that 1.47 curies of TRU waste would be spilled per year and 0.1% would become airborne. The remainder is unaccounted for. The quantity of TRU wastes on ion exchange resins reported in Table 8-7 was only 0.04 curies per year.

Assumptions for Release

A number of factors are involved in the chain of events that must occur before radioactivity is released at the site. Most of the factors that are used to calculate the release of the CH-TRU waste are assumed (surface contamination, non-fixed surface contamination, number of boxes and drums damaged, percent of the surface area that is cracked). Is there a data base for the surface contaminated or damaged drums and boxes that have been packaged, shipped and stored over the years? It is difficult to determine if the assumptions are conservative based on the following information presented in the DEIS:

1) The DEIS assumed that one spent fuel assembly and its canister are damaged in a four-year period (8-32:1) as described in the NRC's <u>Reactor Safety Study</u>, 1975. A more recent report reviewing the history of spent fuel assembly accidents by Johnson (Ref. 1), presented data which suggest that one or more accidents per year might be more realistic.

2) All calculations assumed an average percentage of powder and average radionuclide concentration in each drum. Calculations should also be made using boxes, which contain more waste than drums and have higher levels of radioactivity.

3) Specific data are absent on design, testing and experience with remote handled TRU waste casks and canisters.

Due to the ranges of possible values of factors involved in a potential release of radioactivity, a sensitivity analyses should be performed to determine their effect on potential doses.

High Level Waste Experiments

The DEIS did not state the amount and types of radionuclides that will be brought into the repository for these experiments. Twenty to 200 bare waste experiments are to be conducted (8-47). The number of full sized canisters emplaced was estimated to be between 20 and 200 (8-48), but "...these numbers, like the estimates of bare waste reaction chambers, may change by as much as a factor of 2" (8-48;2). "The source of the waste to be used in these experiments is not as yet defined" (2-24;3). The possibilities of using laboratory produced commercial reactor wastes, aged defense HLW or wastes fortified with Strontium-90 or Cesium-137 are then discussed.

Even with these few details it is apparent that the quantities of radioactivity brought into the repository in the experimental program could be large if one assumes that the full sized canisters described on page 8-48 are the same as the high level waste canisters described for commercial high level waste in Table E-4 on page E-5. The estimated amount of radioactivity in a high level waste canister is about 460,000 curies (Table E-4) which would give a total of 9-92 million curies from the canisters alone.

Retrievability

Present plans are to retrieve all high level wastes (HLW) experiments after completion and to have the ability to retrieve contact-handled and remote-handled TRU wastes and spent fuel. The periods of retrievability are apparently 10 years for TRU wastes and 20 years for spent fuel (the times are reversed in the statements on 2-18;7 and 2-19;1). Container life for TRU wastes is designed for 10 years so it can be retrieved (5-4). A possible need for repackaging retrieved containers is recognized (9-52;5) and apparently it is planned to do this underground. Another reference (9-49;2) stated that accidents during retrieval are expected to be no worse than could occur during emplacement.

The DEIS did not provide guidance on the criteria for retrieval of TRU and spent fuel wastes. Details were not provided on how retrieval would be conducted and on the contamination and exposure problems that are expected. The retrieved containers could be damaged during emplacement, storage, and retrieval. Also, chemical action of the salt environment for periods of 10 or 20 years could produce deterioration in the integrity of the canisters. Retrieval of high level waste experiments will be further complicated by bare wastes and contaminated salt.

While retrieval is possible, the removal of radioactive waste from the repository will involve more problems than emplacement. The extent of this difference has not been adequately addressed in the DEIS and should be expanded upon in either the final EIS or PSAR (Preliminary Safety Analysis Report).

EEG believes that retrieval will be a complex operation with the potential for significant radiation exposure to workers and for possible releases to the environment. It is necessary for retrievability to be evaluated in detail for procedures, logistics, and criteria before conclusions can be drawn about its feasibility.

Decommissioning

The discussion on decommissioning of the WIPP site repository (8-53 to 8-57) covers various alternatives and contains adequate detail at this time. Any of the alternatives listed on page 8-54 should be acceptable if carried out properly. There are two issues that have the potential to increase the probability of long-term problems:

1) administrative control over the site; and

2) borehole plugging.

Possible industrial use of the site is indicated (8-53;3). The land area is expected to be returned to its natural state in several decades unless the mothballing option is taken (11-1). Also, Scenario 5 (9-124) assumes administrative control is lost after 100 years and unregulated drilling can occur. This scenario results in a high dose to well drillers. A detailed evaluation should be made of the degree of control needed at the site after decommissioning and should include:

1) the possibility of control for periods longer than 100 years;

2) the long-term controls over shallow-well drilling in Zone III and resource extraction in Zone IV; and

3) details of the long-term radiological monitoring program.

Radiological Monitoring Program

Pages J-24 to J-41 of Appendix J of the DEIS describe the present radiological monitoring program, the tentative pre-operational monitoring program, the proposed operational monitoring program, and the post-operational monitoring program.

While it is realized that these future programs are necessarily tentative, the following comments are offered.

Present Program

This program appears adequate for several years, with one exception. Measurements of radon and its short-lived daughter product concentrations are needed from the soil, from mined rock, and in the underground mine.

Radon monitoring should be done as soon as possible because the presence of high levels could influence the design of underground ventilation.

It will be necessary to obtain sufficient samples and analyses before operation to insure that the variations in the background (naturally occuring and from weapons testing fallout) levels of actinides, tritium, Carbon-14 and fission products are adequately known. These values are needed in order to be able to detect contamination from site operations.

Pre-operational Program

It is noted that no air particulate station is planned for Hobbs. Since it is a major population center, with a calculated long-term χ/Q only 10% lower than at Eunice, this omission should be reconsidered. Also, the three days per week of sampling should be randomized in order to measure levels on work days, and non-work days.

Consideration should also be given to monitoring radioactivity in rainfall and runoff (when it occurs) at the site as well as surface water and biota in Nash Draw. Several additional shallow wells, whether presently used for human consumption or not, should also be sampled on an annual or biennial basis.

In several cases in Table J-4 (Appendix J), the types of analyses are not specific enough. Gross analysis is useful as a screening mechanism for detecting significant contamination. However, it usually will not detect trace migration of radionuclides. All media being sampled should have periodic analyses of the actinides, tritium, Carbon-14 and long-lived fission products. Consideration should be given to developing and maintaining a capability of measuring Iodine-129 in case of accidents (J-27;2).

Operational Monitoring Program

The same considerations expressed for the pre-operational program are applicable for the operational program. No further comments are offered at this time.

Post-operational Program.

The outline of a post-operational program presented in Table J-7 appears reasonable. However, the borehole radionuclide analyses should be for specific radionuclides rather than gross alpha and beta for the reasons discussed above.

Non-Radiological Hazards

Air Quality

The EEG analyzed the data presented in the DEIS to determine if a potential exists to exceed the Ambient Air Quality Standards at the Reference Site. It was concluded that standards for several of the criteria pollutants could be violated during construction and operation of the WIPP. This conclusion, which differs from that implied in the DEIS, is due to the following factors:

1) When calculating ambient concentrations, it is appropriate to consider locations where the public has access rather than county-wide averages. The distance from the WIPP site may be less than 0.5 miles.

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1) When calculating ambient concentrations, it is appropriate to consider locations where the public has access rather than county-wide averages. The distance from the WIPP site may be less than 0.5 miles.

2) Appropriate allowance must be made for the number of shifts that will operate at the site. There is a discrepancy in the DEIS. Table 8-9 assumes only one shift operation, whereas page 2-19 and 8-27 mention three-shift operation to calculate radiation releases.

3) For three shift, five-day per week operation, the nitrogen oxide and sulfur dioxide concentrations at 0.5 miles were calculated to exceed the annual average concentrations permitted by the State of New Mexico if the χ/Q values in Table H-36 are used. The annual nitrogen dioxide Standard would also be exceeded during construction.

4) During construction, the fugitive dust emissions shown on page 9-8 would exceed the permissible 24-hour level at a distance of two miles when the background concentration of approximately 30 $\mu g/m^3$ is added.

5) Particulate emissions during the operating phase are dominated by releases from the salt pile and from the salt drying unit. The magnitude of salt pile emissions has a range of uncertainty. Experience with the potash industry suggests that the pile emissions will be negligible except for the periods when salt is being reclaimed for drying and use as backfill. Emissions from the salt drying unit (other than from combustion) were not estimated in the DEIS. This source has been found to be significant in the potash industry.

Although the EEG analysis concludes that construction and operation of the site will violate the New Mexico Ambient Air Quality Standards, experience with the potash industry suggests that it should be possible to meet these standards with proper engineering controls, elevated releases, and other mitigating measures.

A more detailed analysis of the air quality aspects needs to be performed. This analysis should include one-hour and three-hour analyses as well as 24-hour and annual values. A more precise estimate of emissions from the salt pile and dryer is needed. The analysis should consider such factors as elevated releases, non-point source emissions, cloud depletion, control technology, and other mitigating measures that will be taken. The final EIS should contain the results of this re-evaluation and indicate the measures that will be provided to insure that Ambient Air Quality Standards are not exceeded.

Noise

The DEIS makes predictions on the noise levels from construction and operation at the WIPP site. For the most part these projections appear reasonable. However, the conclusions emphasize the fact that ambient levels will still be well below various standards and suggest there is no problem. Actually, the noise environment will be degraded both during the construction and operating phase and some residents, off-site and near transportation arteries, and users of Zones II, III and IV will be exposed to more noise than at present. Furthermore, while the DEIS makes reference to measures that could minimize noise exposure, no commitments are made to implement specific measures.

Several items in the DEIS requiring clarifications are:

1) Traffic noise impact from WIPP-related commuter and truck traffic cannot be estimated without knowing the projected traffic volume (of both trucks and autos) with and without the project. This needs to include the effect of night-time traffic which will be present during three-shift operation and construction.

2) The assumption of a peak dBA of 84 at 50 feet from diesel trucks is optimistic since the Federal standard for Inter-State trucks permits 90 dBA and many intrastate trucks cannot meet this standard.

3) References on pages 9-4 and 9-27 imply that noise levels of about 45 dBA will be inaudible at the James Ranch. Actually, if the ambient is 26-28 dBA, sound pressure levels of less than 35 dBA will be clearly audible.

4) It is unclear from the description of the mined-rock storage just what noise sources are included and how they might vary with time.

5) Operational noise near the site would be expected to alter the present mix of wildlife species. The conclusion that this would be minor and insignificant should be documented.

The final EIS should include more precise analysis of just how much the noise level is expected to rise from site construction and operation. Also, consideration should be given to mitigating measures such as:

1) busing of workers to drastically reduce auto traffic;

2) muffling of construction equipment and use of low noise products where available;

3) a requirement that all trucks meet the Federal noise regulations required for Inter-State Commerce; and

4) housing of various equipment and operations.

Water Quality

Several aspects of the WIPP site operation may have an effect on water quality. Primary impacts (on site) could occur from:

1) the sewage plant effluent and sludge;

2) reclaimed water use on-site;

3) runoff and leaching from the salt pile; and

4) general site runoff.

Secondary impacts could occur from the WIPP induced population growth in Eddy and Lea counties. The most likely problem is from septic tank contamination in unsewered areas and is recognized in the DEIS (9-91;2). Both primary and secondary impacts appear to be manageable with proper planning.

Solid and Hazardous Waste Control

Construction, operation and decommissioning of the WIPP site will result in the generation of substantial quantities of solid waste and unestimated amounts of non-radiological hazardous wastes. There will also be some secondary impacts in Carlsbad and Hobbs due to the WIPP induced population growth.

It is not possible to evaluate the hazardous waste situation from the limited information in the DEIS. Under present New Mexico regulations, it is permissible to dispose of hazardous wastes on the site without a permit where they are generated. However, Federal regulations are expected to be in effect prior to the beginning of site construction and they will probably require regulation whether disposal is on or off-site. The types and quantities of hazardous waste expected to be generated on-site need to be determined more precisely.

Metals and discarded equipment are scheduled to be recycled with a commercial salvage company (8-35;6). An appropriate control system should be established to insure that this recycling does not lead to off-site radiological contamination.

Water Supply

Since the WIPP plant operators propose to purchase its water supply from the City of Carlsbad, the State of New Mexico would be involved in regulatory procedures only indirectly. In addition, the project could be exempted from State regulations under Part I., Section 102, Water Supply Regulations.

A portion of the population growth could take place outside of incorporated city limits. Water supplies for these families would probably come from individual wells. Local and regional governmental agencies should be aware of potential water quality problems related to the increased number of wells and their proximity to septic systems since the State Environmental Improvement Division does not regulate individual water supply systems.

Detailed Comments

8-15;3 No mention is made in the DEIS of the management or organization of the health physics program. EEG assumes this will be covered in the PSAR.

8-17;5 A contamination check should be made on empty CH waste containers before they are "reloaded onto vehicles leaving the plant".

8-23 Consideration should be given to isolating the High Level Waste experimental area from the remainder of the mine in case of accident. It is unclear how the isolation of the air flow will be accomplished from the description on pages 8-20, 8-22 and 8-23.

8-28;1 The DEIS stated that 10% of surface activity is released and becomes airborne. What data is this based on?

8-28;2 The DEIS states that 30 drums and five boxes per year may be received in a damaged condition. This is .019% of the drums and 0.21% of the boxes. Are these numbers predicated on actual experience?

8-28;3 The DEIS states that cracks generated by dropping a 55-gallon drum will be less than 1% of the total area of the drum surface. Is there a reference for this assumption?

8-28;4 The assumptions of an airborne fraction of 0.00023 per hour and a decontamination factor of 10^6 are referenced and the airborne fraction is taken from an experiment utilizing a road-like surface. Both larger and smaller fractions were observed from other experiments by the authors (Mishima and Schwendiman, 1973).

8-31;4 The DEIS assumed that one canister per year will be cracked, the crack is 1% of the area, and that the release is proportional to the crack. Is there a reference for these assumptions? Mishima and Schwendiman in BNWL-1732, 1973 do not cover these items.

8-32;1 It is assumed that one spent fuel assembly and its canister are damaged in a four-year period. From the data presented in a review of the history of spent fuel assembly accidents (ref. 1) it appears the assumed rate of one accident per 1000 assemblies handled might be too low.

The NRC's Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (ref. 2) stated that both NFS (Nuclear Fuel Service) and AGNS (Allied-General Nuclear Services) included in their safety analysis reports on underwater fuel drop accidents in which it is assumed that all of the fuel pins in a fuel assembly were ruptured. It appears that when the DEIS assumptions on released fractions from fuel assemblies are compared to other references (ref. 1 and 2) the DEIS assumptions might be too low.

8-31;5 Since experiments with high level waste are planned with bare sources, this paragraph should be clarified.

8-32;1 The assumptions that many nuclides including H-3, Kr-85, I-129, tellurium and selenium are released from the damaged spent fuel are much different from the assumptions discussed on page 6-26 for a rail accident. These differences should be resolved or the rationale explained.

In addition, C-14 has been consistently omitted from all inventories, releases and dose calculations pertaining to spent fuel. The DOE in its Draft Environmental Impact Statement Management of Commercially Generated

Radioactive Waste (Ref. 3), consistently lists C-14 in its inventory (as on page 2.1.16) and outlines the calculations for C-14 (in Appendix D).

8-34;1,3	The DEIS stated that there are 200 HEPA filters in "parallel". This differs with statements made in 8-26;4 where it says there are two stages of filters in series. If the statement on page 8-34 is correct, then there is only a decontamination factor of 10^3 rather than the 10^6 used throughout the report.
8-34 Table 8-7	The total radioactivity per drum in Table 8-7 totals 6.7-4 Ci, not 5.7-4.
8-36;3	How effective will the protective action of spraying the salt pile with water be in containing the contents?
8-37;2	265 $\mu g/m^3$.
8-38 Table 8-9	While the table shows the total emissions of pollutants at the site, it would be helpful to show the maximum expected emission rates and when they occur.
8-43;3	Laboratory decontamination agents with EDTA may be present in the TRU waste. If EDTA is present, it may drastically alter the migration of actinides through the soil and effectively alter the K_d values in the long range release scenarios.
8-47;4	Are the "reaction chambers" merely drilled holes in the salt of the mine? If so, how does one collect "gaseous samples" without having such samples contaminated by the ambient air?

8-49;7 If the canistered spent fuel assembly is placed inside a sleeved hole, can this assembly do anything more than produce a temperature gradient outside the sleeve? If not, why use a spent fuel assembly?

8-49;5 There is no reference or backup information given to substantiate the statement "Sufficient air quantities will be provided to support the mining and storage operations as well as to remove fission gases that might escape from unsealed storage rooms". Uranium mining experience indicated adequate ventilation can be difficult to provide.

8-51 The plans for retrieval have not addressed the problem of radiation protection.

8-52;2 Have the contamination limits been established, and what is "an acceptable level"? The potential contamination problem for "retrieval after backfilling" could be extremely troublesome. More information on personnel exposure control and contamination limits should be provided. Where will the radioactive waste and contaminated salt be taken?

8-55;3 The DEIS stated that the site might be used after decommissioning as an industrial site. No scenarios cover this possibility of future use.

8-57;1 The DEIS stated that the results obtained so far give the DOE confidence that newly developed plugging methods will be available in decommissioning the repository. What are the references?

References

1. Johnson, A.B., Jr. Behavior of Spent Nuclear Fuel in Water Pool Storage (BNWL-2256), 1977.

2. U.S. Nuclear Regulatory Commission. Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0404), March 1978.

3. U.S. Department of Energy. Management of Commer ally Generated Radioactive Waste (DOE/EIS-0046D), Draft Environmental Impact Statement, April 1979.

4. U.S. Energy Research and Development Administration. Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle (ERDA-76-43), May 1976.

5. Bell, M.J. ORIGEN - The ORNL Isotope Generation and Depletion Code (ORNL-4628), 1973.

6. Oak Ridge National Laboratory. Radiation Shielding Information Center. ORIGEN. Isotope Generation and Depletion Code - Matrix Exponential Method (CCC-127), October 1978.

7. Industridepartment Energikommissionen. Disposal of High Active Nuclear Fuel Waste. A Critical Review of the Nuclear Fuel Safety (KBS) Project on Final Disposal of Vitrified High Active Fuel Wastes.

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1. Johnson, A.B., Jr. Behavior of Spent Nuclear Fuel in Water Pool Storage (BNWL-2256), 1977.

2. U.S. Nuclear Regulatory Commission. Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0404), March 1978.

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5. Bell, M.J. ORIGEN - The ORNL Isotope Generation and Depletion Code (ORNL-4628), 1973.

6. Oak Ridge National Laboratory. Radiation Shielding Information Center. ORIGEN. Isotope Generation and Depletion Code - Matrix Exponential Method (CCC-127), October 1978.

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RADIOLOGICAL IMPACTS OF THE REPOSITORY
(DEIS Chapter 9)

Major Conclusions

Operational Releases

1) Atmospheric dispersion coefficients were calculated by EEG, checked with those listed in the DEIS and found to be in agreement. However, the DEIS did not use a consistent approach in calculating those coefficients for operational releases of radioactivity, air quality emissions and transportation accidents.

2) EEG obtained close agreement with the DEIS inhalation dose calculations from both normal and accidental releases to an individual at the James Ranch. Consideration should be given to inhalation doses received by transient people in Zones II, III and IV and to potential residents near the northwest site boundary.

3) The detailed assumptions used in evaluating accident scenarios may underestimate the amounts of radioactivity that could be released.

4) An accident scenario involving a methane gas pocket should be considered.

5) The assumption that contaminated food will be taken out of distribution has not always been possible or necessary.

6) The Chapter 9 assumption that exhaust air from underground waste handling and storage areas passes through HEPA filters is inconsistent with statements in Chapter 8. Since the absence of filters can result in a substantial increase in doses from particulate radioactivity, it is important to clarify this point.

7) Occupational radiation exposure has not been evaluated by EEG because of lack of necessary data in the DEIS. More information is needed on waste operations, the environmental control systems, and the health physics program. It is anticipated that this information will be provided in the Preliminary Safety Analysis Report.

Long Term Releases

1) EEG has identified a number of repository breach scenarios which also should be considered and evaluated in the final DEIS:

   a) well water becomes contaminated and is used for irrigation or stock watering;

   b) gas, generated by organic decomposition of the waste, acts as a driving mechanism in bringing waste to the surface;

   c) a connection develops between the repository, a high pressure brine reservoir and the surface;

   d) solution mining for salt takes place.

2) EEG has checked many of the DEIS dose calculations for the long term release scenarios considered, and the EEG and DEIS results are in agreement. Since the hydrologic parameters on which these dose estimates are based can vary by several orders of magnitude, the effect of parameter variation on dose estimates should be evaluated.

3) Unacceptably high radiation doses could occur to well drillers from a scenario 5 type incident. Control measures should be considered to prevent such an event.

Operational Releases

Atmospheric Dispersion Coefficients

Several key χ/Q values were calculated by a simplified hand calculation (see Appendix V for details) and compared with those used in the DEIS. The EEG model differed somewhat from the MESODIF Code used in the DEIS in not allowing the plume to be blown back over the source to contribute on a "second pass". The values calculated for the long term average χ/Q were lower than those used in the DEIS by factors of 3.3 to 4.4 (Table H-36). Lower values would be expected in the prevailing downwind direction from the model difference, although the magnitude of the difference cannot be estimated from the data available. Values calculated for the one-hour frequency $(\chi/Q)_{5\%}$ and $(\chi/Q)_{50\%}$ (H-Annex 1, Table 21) varied from 1.0-4.1 times those in the DEIS with agreement being best at 0.5 miles. EEG concluded that the short-term and long-term χ/Q values used for the site are reasonable.

The DEIS analysis did not use atmospheric dispersion coefficients to compute annual concentrations for non-radiological air pollutants other than a 24-hour value for particulates. In this case, another equation was used and an effective $(\chi/Q)_{24\ hrs.}$ of $2.1 \times 10^{-6}\ s/m^3$ was obtained. This is to be compared to a $(\chi/Q)_{50\%}$ of about $15 \times 10^{-6}\ s/m^3$ (H-Annex 1, Table 21) and an annual average of $5.9 \times 10^{-6}\ s/m^3$ (in downwind maximum sector from Table H-36). This calculation is inconsistent with that for site radionuclide releases.

Slightly different dispersion coefficients were used in the DEIS to compute doses from transportation accidents and an elevated release was assumed. This assumption produces lower dose estimates and is consequently less conservative than assuming a surface release. The DEIS did not explain its procedures for determining that 0.5 miles from an accident was the maximum concentration. EEG made this calculation and was in general agreement when the same assumptions were used. However, it was noted that concentrations were not negligible closer to the site and that if other atmospheric stability categories were assumed the highest values occurred closer than 0.5 miles.

Any inconsistencies in the use of atmospheric dispersion coefficients and assumptions (especially the rationale for assuming an elevated release in transportation accidents that do not involve fires, while assuming surface releases from the site) should be explained in the final EIS.

Radiological Doses to the Public

The calculated doses (Table 9-18) received by an individual at the James Ranch were checked using the releases from Table 8-6 (see Appendix VII). In all cases, the results agreed with the DEIS within 20%. The doses are small and well below existing standards that apply to other types of nuclear facilities. However, there are some uncertainties in the release fractions assumed (see Chapter 8 discussion) and in the source term.

There is a question whether the maximum exposed individual would necessarily be a resident of the James Ranch. If he were to reside on private land 2.8 miles northwest of the center of the site, he would receive an inhalation dose five times as great as at the James Ranch. Also, individuals spending some time in Zones II, III and IV would be exposed to higher concentrations while on the site. For example, average concentrations at 0.5 miles from the site center would be 16-145 times those at the James Ranch.

The assumptions on living patterns in Table 9-17 appear reasonable for the average person residing in each subsector. The calculation of a maximum dose to an individual should consider a person with a family cow that provides over 1% of his milk, or a garden that provides over 10% of his vegetables, or cattle that provide over 50% of his meat. Additionally, there may be game killed on the site and consumed by area residents. The final EIS should state the assumptions and calculate the maximum ingestion dose to an individual at both the James Ranch and at the northwest boundary. The inhalation doses should be listed separately from the ingestion doses.

Environmental Effects of Accidents During Operations

The operational accident scenarios evaluated in the DEIS (Table 9-21, pp. 9-45 to 9-48) appear to be fairly complete. However, it appears that the DEIS may underestimate the duration of fires, the number of containers involved and the clean-up time involved. An operational scenario not addressed was that of encountering a methane gas pocket during the mining operations resulting in an underground explosion involving multiple drums and/or boxes or spent fuel canisters. EEG recommends that such a severe operational accident be investigated by DOE in the final EIS.

Where referenced, estimates of release fractions (pp. 9-49 and 9-50) have been reviewed and in some cases the values used in the DEIS may be questioned. For example, it is stated (9-49;5) that Shefelbine supports a conservative assumption that 10% of the waste is in powder form and that 25% of the waste is combustible (Ref. 1). In reviewing the Shefelbine report, no mention of powder was found. The 10% powder figure might be deduced from the data indirectly. Shefelbine does reference Dieckhoner (1978) as the source of the information that 25% of the waste is combustible. In fact, Shefelbine states that "this data should be used with caution because there seems to be a consensus that, in spite of regulations, considerable mixing of combustibles and non-combustibles occurred in the past" (Ref. 1, p. 25). The presence of combustibles directly affects the severity of a postulated fire.

A second example concerns the percentage of activity which is released and respirable during accidents. It was stated that 1% was used as an intermediate value based on Mishima and Schwendiman (Ref. 2, 3). Mishima and Schwendiman described that as much as 1% of the plutonium was airborne during the combustion of flammable contaminated materials (Ref. 2, p.6). They also found that 10-40% of uranium oxide became airborne after being mixed with combustible material and ignited. Uranium oxide was used to simulate plutonium in these experiments.

In the spent fuel accident, a gap activity of 30% of the gaseous activity (H-3, Kr-85, I-129) was chosen. Although there is little information in the literature on the gap activity of fuel assemblies older than 10 years, gap activities as high as 45% have been observed (Ref. 4). Also, the quantity of Carbon-14 released was ignored.

The computer code AIRDOS-II was used in the DEIS to calculate resulting doses and dose commitments. It traces each nuclide from the point of release through the biosphere to man. AIRDOS-II is listed with the U.S. Department of Energy Radiation Shielding Information Center and has been tested and evaluated by this group prior to distribution for general use. Hand calculations by EEG using the DEIS assumptions and standard formulas gave results which generally agreed with those reported in the DEIS.

The assumed distribution of radionuclides released to the environment during operational accidents is sometimes different than the assumed releases from transportation accidents. For example, the RH-TRU waste railroad accident involving impact and fire considers only the release of cesium-137. The surface fire at the facility (Accident R-11) has cesium-137 as less than 1% of the total release. It is recommended that a consistent release fraction be used in calculations throughout the final EIS.

AIRDOS-II considers only the inhalation and external pathways of exposure. In some situations (such as surface runoff) the water pathway could be significant and should be considered. The assumption that contaminated food will be taken out of distribution has not always been possible or necessary. Existing Federal Protective Action Guides do not recommend removal of food or milk from commerce unless the projected dose commitment is 5 rem to the whole body or 15 rem to the thyroid.

Using the releases and the atmospheric dispersion coefficients (χ/Q) for accident scenario C-7 described in the DEIS, EEG calculated doses for a person living at the James Ranch. The results agreed with those given in Table 9-25 of the DEIS and are shown on the following Table 10.

Table 10

Dose Estimates at the James Ranch from Accident C7

Reference	χ/Q Type	χ/Q (s/m^3)	50 yr. Bone Dose Commitment (rem) EEG	EIS
Table 21 p. 26	50% one-hour frequency	0.58×10^{-5}	7.4×10^{-9}	5.5×10^{-9}
Table H-36 p. H-59	Annual	0.62×10^{-6}	7.9×10^{-10}	
Table 21 p. 26	5%	0.568×10^{-4}	7.2×10^{-8}	

EEG also calculated doses from the spent fuel hoist drop accident scenario (R15) using the assumptions of the DEIS. The results, presented in Table 11, are in agreement with the DEIS calculations. The only question raised concerned the assumptions of release fractions for various accidents. The margin of error in the assumptions is not well known and EEG recommends that the basis of the assumptions be discussed in the final EIS.

The Chapter 9 assumption that exhaust air from underground waste handling and storage areas passes through HEPA filters is inconsistent with statements in Chapter 8.

Occupational Radiation Exposure

Occupational radiation exposure at WIPP is scheduled to be covered in the Preliminary Safety Analysis Report (PSAR). Therefore, no attempt will be made to evaluate it here. The following are examples of the kind of information needed to adequately evaluate occupational exposure:

1) The analysis should consider estimates of maximum individual doses, the expected distribution of doses among workers, and the population dose to the entire work force. An evaluation should be made of whether these doses are as low as reasonably achievable (ALARA).

2) Additional information will be needed on the environmental control systems, other physical facilities, and pertinent equipment, both above and below ground, so that the reasonableness of projected doses can be evaluated. More data are needed on the actual radiation levels (average and range) that workers will be exposed to from remote handled TRU waste, spent fuel assemblies and on the high level waste

Table 11

Operational Accident Scenario R-15
Hoist Drop - Spent Fuel - 6 hr. Release

Nuclide	50 year Dose Committment (rem)				
	Bone	Lung	Whole Body	Skin	Thyroid
H-3		9.54×10^{-6}	7.66×10^{-6}		0.77×10^{-5}
Kr-85		2.52×10^{-6}		7.04×10^{-5}	
Sr-90	7.43×10^{-6}	2.78×10^{-8}	0.46×10^{-6}		
Ru-106		5.97×10^{-12}			
I-129	0.02×10^{-6}		0.06×10^{-6}		4.82×10^{-5}
Cs-134		2.14×10^{-10}			
Cs-137		3.96×10^{-10}			
Pm-147	7.65×10^{-12}	2.4×10^{-13}			
Eu-154	8.22×10^{-11}	1.4×10^{-12}			
Np-237	5.89×10^{-12}	1.0×10^{-14}			
Pu-238	8.53×10^{-8}	1.0×10^{-10}			
Pu-239	1.04×10^{-8}	5.5×10^{-11}			
Pu-240	1.66×10^{-8}	8.75×10^{-11}			
Pu-241	1.49×10^{-10}	1.06×10^{-11}			
Pu-242		2.48×10^{-13}			
Am-241	1.59×10^{-8}	2.87×10^{-11}			
Am-242m					
Am-243	2.2×10^{-10}				
Cm-243	3.11×10^{-11}				
Cm-244	1.29×10^{-9}				
EEG Totals	7.58×10^{-6}	1.2×10^{-5}	8.28×10^{-6}	7.04×10^{-5}	5.59×10^{-5}
DEIS Totals*					
R-15 Spent Fuel	8.7×10^{-6}	1×10^{-5}	8.3×10^{-6}	2.2×10^{-4}	3.2×10^{-5}

*Table 9-25, page 9-56

experiments (including retrieval operations). Also, it will be necessary to describe the management organization (including health physics activities) that will be used to operate the facility and provide health and safety control.

Long-Term Releases

Five repository breach "scenarios" were analyzed in the DEIS (section 9.5.1). Scenarios 1-4 all resulted in dissolution of the waste, passage of the waste into the Rustler aquifer, and passage through the aquifer into the Pecos River. Scenario 5 involved direct access by drilling.

Dose Calculation Methodology

The radionuclide concentrations and resultant radiation doses reported in the DEIS were obtained by using large computer codes. EEG was able to check many of these computer calculations by hand. To check results of the hydrologic model used to describe nuclide transport in the liquid breach scenarios 1-4, EEG used a simpler model. To check dose calculations, EEG used standard formulas and conversion factors. The calculations are discussed in Appendices VI and VII, and the results compared with those in the DEIS. In its calculations, EEG used the hydrologic parameters, radionuclide inventory, and scenario descriptions in the DEIS. The EEG and DEIS results agree closely.

While these calculation checks tend to support the validity of the methods employed in the DEIS to calculate nuclide concentrations, ingestion doses and external gamma radiation doses, they do not provide checks on the validity of the assumptions used or the appropriateness of the situations analyzed.

Parameter Values

One must consider the key parameters that lead to the nuclide concentrations and ingestion doses calculated. They include the distribution coefficients (K_d) values which are responsible for holding back such nuclides as plutonium, neptunium and thorium, as well as the basic driving parameter, $\bar{v}$, the assumed groundwater flow velocity. The last quantity depends on hydraulic conductivity, porosity and hydraulic gradient. All of these parameters have large uncertainties associated with them because of natural variation and difficulties in measurement. Both K_d values and hydraulic conductivities can vary by several orders of magnitude. A thorough review of these uncertainties and of their impact on radiation doses must be made, and should be included in the final EIS.

EEG has done some calculations relevant to the effect of variations in K_d and hydraulic conductivity values and they appear in Appendix VII.

Flow Paths

All of the hydrologic breach scenarios assumed a flow along the Rustler aquifer and release at the Malaga Bend of the Pecos River. However, the interface between the Rustler and Salado formations is exposed at Nash Draw. A spring at the north edge of Laguna Grande de la Sal is fed by water from the Rustler aquifers. The calculated hydraulic potentials for the Rustler formation (Figure K-5, p. k-13) indicate that the shortest release path is 15 miles from WIPP to Malaga Bend. However, the measured hydraulic potential contours in the Rustler formation (Figure K-3, p. K-12) indicate that the shortest flow path is 9 miles to Laguna Grande de la Sal. The dosage calculations should take this shorter path into consideration.

Scenario 5

Dose estimates in the DEIS for both external radiation to a drill crew member (Table 9-47) and inhalation to a resident (Tables 9-48 and 9-49) were checked using the assumptions in the DEIS (see Appendix VII). EEG calculated a dose of 71 rem to a drill crew-member, compared to ∿90 rem in the DEIS. In either case the dose is high enough to warrant serious consideration of control measures to prevent such an occurrence.

Alternate Scenarios

It is not clear that the scenarios used in the DEIS are indeed upper limits or bounding cases. EEG has identified a number of scenarios which also should be considered in the final EIS:

1) well water becomes contaminated and is used for irrigation or stock watering;

2) gas, generated by organic decomposition of the waste, acts as a driving mechanism in bringing waste to the surface;

3) a connection develops between the repository, a high pressure brine reservoir and the surface; and

4) solution mining for salt takes place.

The April 1979 Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste (DOE/EIS-0046D) discussed solution mining for salt as the most likely of several repository breach scenarios. The main pathway of exposure was considered to be ingestion of contaminated salt. The presence of salt contaminated with radioactive materials was not expected to go undetected for long, due to quality control checks. The DOE assumed the contamination went undetected for one year, and obtained radiation doses orders of magnitude higher than those obtained for the scenarios considered in the WIPP DEIS.

Detailed Comments

9-4 Table 9-2	The construction equipment noise levels given in Table 9-2 are achievable but will require proper equipment and noise control procedures to obtain.
9-26 Table 9-13	The Department of Housing and Urban Development Criteria for Noise Assessment given in Table 9-13 were revised in 1979 and those should be listed in the final EIS.
9-33 Fig. 9-3	The schematic diagram (Fig. 9-3) shows only the air pathway of exposure. Surface runoff from contaminated surface areas, wildlife contamination from surface lagoon, and ingestion of drinking water are not discussed. Although these pathways may not be the primary ones, they should be considered.
9-33;4	Each wedge of the study area was divided into 10 subsectors, not 14 as stated.
9-34 Fig. 9-4	According to page H-6, Figure H-1 and page H-8, Table H-4, the population size of 50 would not be within the 5 mile or 10 mile radius. There also appears to be some discrepancy with some of the other numbers. For example, the population of 230 given within the 10 mile radius in the WNW direction does not agree with the numbers given in Appendix H. Do the numbers include workers at potash mills?
9-37;2-4	The DOE <u>Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste</u> (Ref. 6) consistently used a 70 year dose commitment instead of the 50 year dose commitment used in the WIPP DEIS. The 70 year dose commitment seems most appropriate when discussing population dose commitments.

9-37;6 Would one expect the dose from krypton to be about 25% of the total from spent fuel when one considers the whole body, lungs and bone? The krypton irradiation involves primarily submersion in the gas because little of the krypton will circulate in the blood and therefore irradiate bone.

9-38 Tables 9-18, 9-19 It should be indicated whether the tritium that seems to be included in the spent fuel group is a gaseous molecule or incorporated in a water molecule. If the tritium is a diatomic molecule, then the dose is received only from immersion, but if it is a part of a water molecule, then it has an effective half life of about 12 days (indicated in ICRP II, or in ICRP Publication 10). Since the values indicate similar doses for bone, lungs and whole body, one must assume that the calculations were for tritium incorporated in a water molecule. In this case, the values given are for dose equivalent and not dose equivalent commitment.

It is not clear what nuclides are contained within "structural materials, fission products, actinides, and spent-fuel". One can assume that the actinides are specified in Table 8-6, page 8-30; however, it is not clear how one separates the structural material, fission products and spent-fuel.

EEG questions the comparison of dose commitment and 50 year dose equivalent from natural background in assessing the acceptability of such releases.

9-38 A dose calculation was made from the actinides (largest dose contributor) using Pu-239 as the primary isotope and agreement with Table 9-18 was obtained when the information in Table 8-6, page 8-30, was used as the source term.

9-40 Table 9-20 The surface dose rate should be 10mrem/hour not per year. The 1900 mrem/year should be subjected to the "as low as reasonable achievable" (ALARA) principle.

9-40;1 The statement was made that the radiation dose to workers on the RH waste portion has not been computed. This type of waste is perhaps the source of highest individual exposures and should be carefully monitored.

9-42;5 The information for the environmental control for the rock pile is inadequate.

9-51;7 The use of filters on exhaust air from the underground storage areas to the atmosphere can reduce the radioactive concentration of particulates by a factor of 10^6. It is unclear from the following two statements whether the storage room air is actually filtered:

(1) page 8-29, Table 8-5, footnote a - "Except for underground operations, effluent treatment is provided by filters in the ventilation system (decontamination factor = 10^6)".

(2) page 8-33;4 - "Airborne surface activity in the underground storage area will be released to the atmosphere unfiltered".

Operational Accident Scenarios C13 and C22 both use 10^6 decontamination factor for the HEPA filters.

9-53 Table 9-24 The DEIS used different distributions of radionuclides released to the environment in transportation accidents and operational accidents involving RH-TRU canisters.

9-54;3 The DEIS assumed 30% of the Kr-85, H-3 and I-129 present in the fuel cladding gap were available for release. NUREG-0404 assumed that 10% of the Kr-85 and 1% of the I-129 present is in the fuel cladding gap and available for release (ref. 9, p. 4-19). A General Electric document (ref. 4) predicted that fission gas release fractions range from 20 to 45%. This report further quotes studies which report 3% of iodine found in the gas plenum and is available for release. The 45% for Krypton is somewhat higher than the 30% predicted by the Reactor Safety Study (ref. 8) used in the DEIS. The General Electric figures of 20 - 45% are also higher than ref. 2 results of 10% release of Kr-85 and 1% of I-129. The doses and dose commitments are directly proportional to the release fractions chosen for the calculations, and the DEIS value of 30% appears reasonable.

Why is the 50 percentile χ/Q used for the accidents? Would the 5 percentile value be more reasonable? The (χ/Q) 5% in NW downwind sector is approximately 16 times the (χ/Q) 50% in SSW sector used in the DEIS.

9-56;1 All calculated doses appear to be adult doses. Were the doses to infants, children and teenagers considered in the computer programs used?

9-56 9-57 One should not compare the radiation received from natural background over 50 years with doses that occur over shorter times (e.g. tritium and I-129).

9-107;3 How was the pressure difference of 7.5 psi obtained? How reliable is that pressure difference? See EEG's question in the GCR review (Appendix III) on the reliability of the head data in the Delaware Mountain Group aquifer.

9-108 In the flow calculations for scenario 1, what value of transmissivity was used for the DMG? Was it 50 $ft.^2$/day? How reliable is the value?

In the flow calculations for scenario 2, what was the basis for the assumption that the wellbore has a hydraulic conductivity K = 50 ft./day?

9-112;5 Should this reference be to figure K-5 rather than K-6?

9-114, 9-115 Table 9-45 is inconsistent with Table 9-46. The transport rates given for I-129, Ra-226, and U-235 in scenario 2 (with the upper transmissivity assumption), do not agree.

	Table 9-45	Table 9-46
I-129, g/yr	2.6 x 10	3.3×10^{-1}
I-129, Ci/yr	4.5×10^{-3}	5.8×10^{-5}
Ra-226, g/yr	2.0×10^{-14}	2.9×10^{-7}
Ra-226, Ci/yr	2.0×10^{-14}	2.9×10^{-7}
U-235, g/yr	2.9 x 10	2.2 x 10
U-235, Ci/yr	6.2×10^{-5}	4.8×10^{-5}

9-116 9-117	The graphs appear to be incorrectly plotted. For example, in Figure 9-15 the maximum concentration of 3.1×10^{-6} should be plotted at between .5 and .6 of the distance from 1 towards 10^{-10}. Tables would provide a more accurate presentation.
9-117	The graphs terminate while the dose rate is still rising in Figure 9-16 a, d, and f. The final EIS should show the maximum dose rates and when they will occur.
9-133, 9-134	EEG agrees that the generation of gas from organic material in the radioactive waste can be important for both transportation accident and long-term storage, and the question posed in the DEIS of the gas generation effects upon the repository must be resolved. Large amounts of gas could be generated. Pressures exceed lithostatic pressure (9-136), i.e. 2000 psi. These large pressures could cause fracturing of the overlying rocks and would possibly release gas with radioactive contaminants directly to the atmosphere through fractures or through a well drilled into the repository.
9-136;4	Will the brine migration induced by heat-emitting radionuclides cause difficulty in retrieval of sources?
9-153 Table 9-53	The probability of fire is assumed to be 10^{-3} and of a dropped container, 10^{-2}. The reasoning behind these numbers should be in the EIS.

9-154 What is the expected distribution of annual dose equivalents received by the radiation workers retrieving stored waste?

9-157 The ratio of population doses to maximally exposed persons is constant in Table 9-55, except for the whole body dose. Why?

References

1. Shefelbine, Henry C. Preliminary Evaluation of the Characteristics of Defense Transuranic Wastes (SAND 78-1850), November 1978.

2. Mishima, J., and Schwendiman, L.C. The Amount and Characteristics of Plutonium Made Airborne Under Thermal Stress (BNWL-SA-3379), October 1970.

3. ________. Some Experimental Measurements of Airborne Uranium (Representing Plutonium) in Transportation Accidents (BNWL-1732), August 1973.

4. General Electric Company. Safety Evaluation Report for Morris Operation Fuel Storage Expansion. Phase I (NEDO-20825), March 1975.

5. Oak Ridge National Laboratory, Radiation Shielding Information Center. AIRDOS-II, Estimation of Radiation Doses Caused by Airborne Radionuclides in Areas Surrounding Nuclear Facilities (CCC-304), 1977.

6. U.S. Department of Energy. Draft Environmental Impact Statement, Management of Commercially Generated Radioactive Waste (DOE/EIS-0046D), April 1979.

7. U.S. Nuclear Regulatory Commission. Draft Generic Environmental Impact Statement, Uranium Milling (NUREG-0511), April 1979.

8. U.S. Nuclear Regulatory Commission. Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH-1400), 1975.

9. U.S. Nuclear Regulatory Commission. Draft Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0404), March 1978.

ADDITIONAL DOSE ESTIMATES

The following lists additional dosage estimates that should be considered in the final EIS.

Long Term

1) Build-up in the environment from radionuclides in water removed from the Pecos River for irrigation, incorporated into soil and plants, and cycled in food and man over long periods of time.
   (General Population)

2) Generate dosage estimates using the DOE generic Waste Isolation Safety Assessment Program (WISAP) model currently under development by the Battelle Northwest Laboratories.
   (General Population)

3) A connection is made between the Delaware Mountain Group aquifer, the repository and the surface.
   (General Population)

4) A connection is made between the repository, a high pressure brine reservoir and the surface.

5) Effects of high pressure gas formation on the release of radionuclides to the environment.
   (General Population)

Transportation

1) Radiation exposure from acts of sabotage in the transportation of radioactive waste materials. The amounts of radioactive material released could be greater than those released in accidents.
   (General Population, Occupational)

2) Radiation exposure to emergency workers such as police and firemen following a transportation accident.
   (Occupational)

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   (General Population)

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   (General Population, Occupational)

2) Radiation exposure to emergency workers such as police and firemen following a transportation accident.
   (Occupational)

3) Exposure to a person in an automobile stopped next to a radioactive waste truck at a red light or in a highway traffic jam. (General Population)

4) Exposures from shipments of retrieved radioactive wastes following the completion of high level waste experiments. Containers could be bent, damaged or under pressure from gas generated by decomposed organic material. (Occupational, General Population)

5) Contamination of a water supply or crops following an airborne release. (General Population)

6) Potential radiation exposure from transportation of material resulting from decommissioning and dismantling of weapons production facilities. While the DEIS assumes that none of the 5 to 95 million cubic feet of material will be shipped to WIPP, it notes that the WIPP will have the capacity to receive some of the TRU waste (2-22;2). (Occupational, General Population)

7) Calculation of individual doses as well as population doses. (Occupational, General Population)

8) Consideration of a diffuse source of radioactivity rather than a point source.

Construction and Operation

1) Radon-222 from the mined salt and from the walls of the underground repository. (Occupational, General Population)

2) Radiation exposure from decommissioning and dismantling of the above ground facility. (Occupational)

3) Pressurized brine breaches the mine, damages containers and flows up the shaft to the surface.
(Occupational, General Population)

4) A methane gas pocket leaks into mine and explodes.

5) Acts of sabotage at surface processing facility.
(Occupational, General Population)

6) On-site exposure.
(General Population)

7) Ingestion doses from operational releases at the site.
(General Population)

Mineral Extraction

1) Radiation exposure to workers who may bring minerals to the surface (oil drilling, solution mining) and to the public using the products.
(Occupational, General Population)

2) Exposure from burning natural gas obtained from formations below the site. Radioactive waste material could move downward as pressure is decreased with the removal of gas.
(General Population)

3) Exposure to people who may use well water from the Culebra, Magenta, or Santa Rosa sandstone aquifers contaminated with radionuclides.
(General Population)

APPENDICES

APPENDIX I

Radioactivity Inventory Calculations

Tables 2-5 in the Inventory of Radioactivity section were prepared as follows:

Volume estimates (cu. ft.), Table 5, were calculated to equal:

$$\left(\begin{array}{c}\text{backlog}\\ \text{of waste,}\\ \text{cu.ft.}\end{array}\right) + \left(\begin{array}{c}\text{30}\\ \text{yr.}\end{array} \times \begin{array}{c}\text{new waste}\\ \text{production,}\\ \text{cu. ft./yr.}\end{array}\right)$$

where backlog and new production volumes were taken from Table 6-2 for CH waste and Table 6-6 for RH waste.

Total repository CH and RH-TRU activities (Curies), Table 3, were obtained by adding total box, drum and RH-TRU canister activities, without considering decay or ingrowth. The total activity of a given nuclide in a given type of container was calculated as:

$$\frac{(\text{curies per container}) \times (\text{total cu. ft. of waste in containers})}{(\text{cu. ft. per container})}$$

where the container activities were taken from Table E-1 (drums), E-2 (boxes) and E-3 (RH canisters); single container volumes were taken from Table 6-3 and 6-5; and total waste volumes in given types of containers were taken from Table 5 of this review.

Shipment activities (Curies), Tables 3 and 4, were obtained by multiplying activities per container (Tables E-1, E-2, E-3) by the number of containers in a shipment (Tables 6-3, 6-5).

The total repository actinide inventory (Curies), Table 2, which includes the effects of decay and ingrowth, after 30 years of repository operation, was calculated as follows. The Plutonium-239 and 240 inventories are not affected by decay over a 30 year period, and these activities are found by adding the CH and RH-TRU activities given in Table 3. Plutonium-238 and 241 have half-lives of 88 and 13 years, respectively, and so these inventories decay significantly in 30 years. The activity after 30 years, A(30), is found from:

$$A(30) = A(0)e^{-\lambda(30)} + \frac{C}{\lambda}\left(1 - e^{-\lambda(30)}\right)$$

where λ is the nuclide decay constant, A(0) is the nuclide activity in the waste backlog, and C is the activity in the new waste produced each year.

Americium-241 is affected by both decay and ingrowth (from Plutonium-241) in a 30 year period. The final inventory was estimated by adding the total Table 3 Americium-241 inventory to the ingrowth term:

$$\frac{13}{460}\left[\begin{array}{l}\text{Pu-241 activity in Table 3 (without decay)} \\ \quad - \text{Pu-241 activity in Table 2 (with decay)}\end{array}\right]$$

where 13/460 is the ratio of the Pu-241 and Am-241 half-lives. Then this undecayed total was multiplied by a 30-year Am-241 decay factor of .96. This method underestimates slightly the amount of Am-241, since it assumes that all of the Am-241 (from the 30 year repository inventory and from the decay of Pu-241) has been present for the full 30 years.

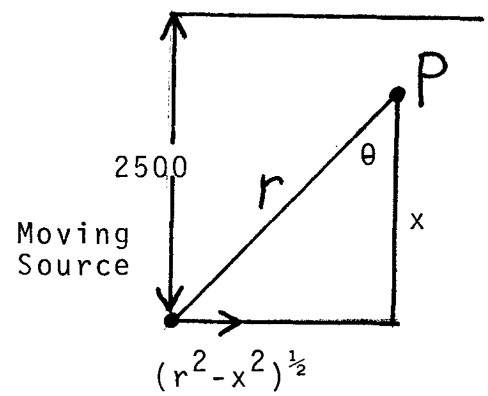
A more complete discussion of decay and ingrowth estimates is included in Appendix VI.

APPENDIX II

TRANSPORTATION CALCULATIONS
(Chapter 6)

To Develop An Expression For Dose To A Person At Point P From A Moving Source

(1) Dose at Point P $= \int D_r \, dt$



where D = dose (rads)

D_r = dose rate (rads/h)

t = time (h)

(2) $D = \int D_{r6} \frac{6^2}{r^2} e^{-\mu r} \, dt$

where D_{r6} = dose rate at 6 feet

r = distance (feet)

μ = linear absorption coefficient (feet^{-1})

Now $s = (r^2 - x^2)^{\frac{1}{2}}$. Thus:

(3) $v = \frac{ds}{dt} = \frac{1}{2} (r^2 - x^2)^{-\frac{1}{2}} \; 2r\frac{dr}{dt}$;

substituting (3) for dt into (2):

$$D = \int D_{r6} \frac{6^2}{r^2} e^{-\mu r} \frac{r\,dr}{v(r^2 - x^2)^{\frac{1}{2}}}$$

To evaluate the integral

$$\int \frac{dr}{r(r^2-x^2)^{\frac{1}{2}}} ,$$

The following change of variable is made:

$$\frac{r}{x} = \sec \theta$$

$$r = x\sec \theta$$

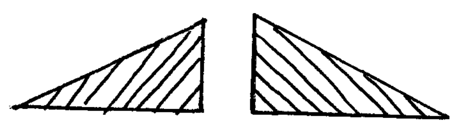
$$dr = x\sec \theta \tan \theta \, d \theta$$

$$\tan \theta = \frac{(r^2 - x^2)^{1/2}}{x}$$

$$(5) \int \frac{x\sec \theta \tan \theta \, d \theta}{x\sec \theta \; x \; \tan \theta} = \int_{\theta = \Pi/2}^{\theta = 0} \frac{d\theta}{x} = \frac{\pi}{2x} .$$

Therefore,

$$D = \frac{D_{r6}' \; 36 \; \pi 2}{v \; 2x}$$

for both sides of  .

$$\text{Dose to a Person at Point P} = \frac{D_{r6} \; 36\pi}{v \; x}$$

To Calculate Total Man Rem Dose to People

$$\underset{\text{(Man-Rem)}}{\text{Dose}} = 2 \; (P.D.)L \int_{Min}^{d} \frac{D_{r6} \; 36\pi \; dx}{v \quad x}$$

$$\text{Dose} = \frac{2 \; (P.D.)L \quad D_{r6} \; 36\Pi}{v} \quad \ln \; (d/min)$$

where

P.D. = population density (people/mi^2)

L = length of shipment path (miles)

v = velocity (mile/hour)

d = distance to people (miles)

min = minimum distance (miles)

correcting for different population densities in urban (u), suburban (s) and rural (r) and velocities and using format of NUREG 0170, p. D-6, Ref. 4 and RADTRAN P. 19, Ref. 3.

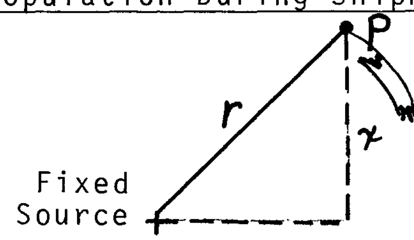
$$\frac{\text{pop. dens.}}{V} = \left[\frac{f_r PD_r}{V_r} + \frac{f_s PD_s}{V_s} + \frac{f_u\ PD_u}{V_u} \quad (f_o + 1.636f_1)\right]$$

incorporating PPS = Packages per shipment
SPY = Shipments per year
L = Distance of Shipment (miles)
TI = Transport Index (mr/h)
K = Dose Rate at Distance d
k = Ko (TI)

$$\text{Dose} = (TI)(Ko)(PPS)(SPY)(L)\pi\ \ln(d/min)\ x$$

$$x\left[\frac{f_r PD_r}{V_r} + \frac{f_s PD_s}{V_s} + \frac{f_u PD_u}{V_u} \quad (f_o + 1.636f_1)\right]$$

<u>To Develop An Expression For Dose To a Population During Shipment Stops</u>.



Annulus=2Πrdr

$$\text{Dose} = \int D_r\ dt$$

$$= Ko(TI)\Delta T(\text{Pop. Dens.}) \int \frac{2\Pi r dr\ e^{-\mu r}\ B(r)}{r^2}$$

$$= Ko(TI)\Delta T(\text{Pop.Dens}) \int_x^d \frac{2\Pi e^{-\mu r}\ B(r)\ dr}{R}$$

Evaluating the integral

$$\int_{r=x}^{r=d} \frac{e^{-\mu r}\ B(r)\ dr}{r^2} = \ln\ (d/x) - \ln(d-x) - \frac{(\mu t)^2}{22!} - \frac{(\mu r)^3}{33!} + \frac{(\mu r)^4}{44!}$$

$$= 3.845$$

$$= (Q_1)Ko(TI)(Shipments/y)\left[\Delta T_r (PD)_r + \Delta T_s\ PD_s + \Delta T_u\ PD_u\right]$$

<u>To Develop An Expression for Dose To Crew</u>

$$Dose = D_r\ \Delta t$$

$$= Ko(TI)S\ \ N_c\ \frac{e^{-\mu d}\ B(d)}{d^2}\ \Delta t_{shipment}$$

where N_c = Number in crew
d = Average distance to crew (feet)
FM = Distance/shipment (miles
Δt = Average time for shipment (hours)
S = Shipments/year

$$\Delta t = \left[\frac{f_r}{V_r} + \frac{f_s}{V_s} + \frac{f_u}{V_u}\right] FM$$

$$Dose = Ko(TI)S\ N_c\ \ e^{-\mu d}\ B(d)\ \left[\frac{f_r}{V_r} + \frac{f_s}{V_s} + \frac{f_u}{V_u}\right] FM,$$

Accident Calculations

Inhalation Dose

$$D_{INH} = K_{INH} \times C_{INH}$$

$\left(\frac{pCi/}{cm^3}\right)$ (rem/μCi inhaled) EIS App. G-5

External γ Dose

$$D_{IMM} = K_{IMM} \times C_{imm}$$

$\left(\frac{pCi/}{cm^3}\right)$ (rem-cm^3/μCi-hr) EIS App. G-5

Immersion Dose

$$D_{wimm} = K_{wimm} \frac{R_t}{d} \left(\frac{1 - e^{-\lambda Tt}}{\lambda t} \right) C_{wimm}$$

(rem cm^3/μCi-hr.)

R_t = surface dep. pCi/cm^2-sec

EIS App. 6-6

D_{wimm} = water immersion dose

d = depth of water

λt = effective λ

t = build up in water

Atmospheric Dispersion

Using Gaussian Plume Dispersion for Ground Level Concentrations, EIS G-1 Equation of Pasquill Reduces to

$$x = \frac{Q}{\Pi \sigma_y \sigma_z \mu} e^{\left[-\frac{1}{2} \left(\frac{H}{T_z}\right)^2\right]}$$ EIS p. G-3

$X = pCi/m^3$
T_y = horiz. dispersion coef. (m)
T_z = vert. dispersion coef. (m)
H = effective ht. of plume
Q = emission rate (pCi/sec)

AIRDOS-II
(FORTRAN on CDC 6600)

<u>To Calculate Curies (Release)</u>

NUREG 0170 Vol. 1 G-1

$Q^1 = (n_i)$ (RF) (AER) (RESP) (E) (DF)

Q^1 =	(n_i)	(RF)	(AER)	(RESP)	(E)	(DF)
	(Ci/ shipment)	(Fraction Released)	(Fraction as Aerosol)	(Fraction Respirable in Aerosol)	(Particle Size Dust Factor)	(Dilution Factor)

Radioactivity Released in Transportation Accidents

Type of Accident	Common Assumptions	Source	Fraction Released From Drum	Fraction Released to Environment	Fraction in Air	Fraction Entrained in Air	Fraction Respirable	Amt of Radioactivity
Rail CH-TRU	Pasquill Stability Factor=F	1 Flat Bed Car 3 Type B Packages 42 Drums/Package 126 Drums	50% in Drums Released to Packages 63	10% 6.3	10% .64 Drum	1.4% .0090	62% .0055 Drum	See Inventory of Radioactivity
Truck CH-TRU	Wind Speed = 1 m/sec	1 truck 42 Drums 42 Drums	50% 21	10% 2.1	25% .5	---------	---1%-------- .0065 Drum	See Inventory of Radioactivity
Rail RH-TRU	Inversion Layer = 1000 m Release Height = 20 m	1 Flat Bed Car 5 Canisters	1% (Only Volatile fission products)	10%				.01 Ci Cs-137
Rail Spent Fuel		1 Cask 10 Canister/cask	30% Kr-85 1% Cs	100% Kr-85 10% Cs				7800 Ci Kr-85 440 Ci volatile f/p

Transportation X/Q Factors

at 0.5 mi, F condition
μ= 1 m/sec, H = 20

p. G-3 $\frac{X}{Q} = \frac{1}{\Pi \sigma_Y \sigma_Z \mu} e \left[-1/2 \left(\frac{H}{z} \right)^2 \right]$

$\sigma_z = .016d(1 + .003d)^{-1} = .016(805)[1 + (3-4)(8+2)] = 10.39$

$\sigma_y = .04d(1 + .001d)^{-1/2} = 32.2[1 + .08]^{-1/2} = 31.0$

$\frac{X}{Q} = \frac{1}{\Pi(10.4)(31.0)(1)} \quad e{-1/2}\left(\frac{20}{10.4}\right)^2 = 9.9\text{-}4(.158) = \underline{\underline{1.6\ \text{-}4}}$

As comparison:

Worst annual at 0.5 mi = 9.0 - 5 (Table H-36) 10 x lower
(X/Q)5% for 1 hour, 0.5 mi = 4.3 - 4 (Table 21, App. H)
3 x higher (X/Q) max for 1 hour, 0.5 mi = 1.1 - 3

The X/Q for gound level release is between 5% and max one-hour X/Q so is conservative enough. Justification for using effective height of 20m is not obvious.

Also, can't check whether X max at 1/2 mile is correct.
See additional comments on χ/Q in Chapter 8.

APPENDIX III

EEG-2

Review Comments on
Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico
SAND 78-1596, Volumes I and II, December 1978

Environmental Evaluation Group
Environmental Improvement Division
Health and Environment Department
State of New Mexico

August, 1979

Second Edition
10/79

CONTENTS

FOREWORD

The purpose of the Environmental Evaluation Group (EEG) is to conduct an independent technical evaluation of the potential radiation exposure to people from the proposed Federal radioactive Waste Isolation Pilot Plant (WIPP) near Carlsbad, in order to protect the public health and safety and ensure that there is no environmental degradation. The EEG is part of the Environmental Improvement Division, a component of the New Mexico Health and Environment Department - the agency charged with the primary responsibility for protecting the health of the citizens of New Mexico.

The Group is neither a proponent nor an opponent of WIPP.

Analyses are conducted of reports issued by the U.S. Department of Energy (DOE) and its contractors, other Federal agencies and other organizations, as they relate to the potential health, safety and environmental impacts from WIPP.

The project is funded entirely by the U.S. Department of Energy through Contract DE-AC04-79AL10752 with the New Mexico Health and Environment Department.

Robert H. Neill
Robert H. Neill
Director

INTRODUCTION

The purpose of this document is to review and evaluate the scientific information contained in the Geological Characterization Report, Waste Isolation Pilot Plant (WIPP) Site, Southeastern New Mexico, SAND 78-1596, Volume I and II, December, 1978, (herein referred to as the GCR), and its supporting references, as it pertains to the environmental, health and safety aspects of the radioactive waste repository proposed near Carlsbad, New Mexico.

These evaluations and interpretations are based on reviews by the Environmental Evaluation Group (EEG) staff and several consultants with expertise on geological aspects of the site. EEG also convened two technical meetings to explore some topics of particular concern.

The review focused on some major concerns regarding areas in which more data or more detailed analysis appeared necessary and did not comment on those areas with which EEG agrees such as Seismology, or on areas which fall outside the scope of EEG's mission such as Resources. In this sense, the review may appear negative in tone and does not acknowledge those areas that have been thoroughly investigated by Sandia, and the U.S. Geological Survey and other contractors of the U.S. Department of Energy. It is recognized that additional data have been obtained since August, 1978, which may answer some of the questions raised.

The review describes in detail the important geotechnical issues on which there are questions or differences in interpretation and comments on the technical basis for certain conclusions presented in the GCR.

ACKNOWLEDGEMENTS

This document reflects the efforts of a number of individuals. The review is a synthesis of comment by the EEG staff consisting of

Robert H. Neill, Director
James K. Channell, Environmental Engineer
Lynn W. Gelhar, Hydrologist*
Carla Wofsy, Mathematician

and the following consultants who have reviewed and commented on portions of the GCR:

Mary Anderson, Assistant Professor of Geology, University of Wisconsin
Roger Anderson, Professor of Geology, University of New Mexico
Lokesh Chaturvedi, Associate Professor of Geological Engineering, New Mexico State University
Stanley Davis, Professor of Hydrology and Water Resources, University of Arizona
George Griswold, President, Tecolote Corporation
Gerardo P. Gross, Professor Geophysics, New Mexico Institute of Mining & Technology
William Hiss, Geologist, U.S. Geological Survey
Gary Landis, Assistant Professor of Geology, University of New Mexico

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*Dr. Lynn Gelhar served as a part-time member of the EEG staff from February 15 to July 1, 1979, during which time he reviewed aspects of the Geological Characterization Report (GCR) and prepared an early draft of the EEG comments on that report. He was not involved in any way in the preparation of the final version of EEG's comments on the GCR.

- Donald Langmuir, Professor of Geochemistry, Colorado School of Mines
- Allan Sanford, Professor of Geophysics, New Mexico Institute of Mining & Technology.

Each of the consultants had an opportunity to review and comment on an early draft of this document. In the process of the evaluation there were informal contacts and a meeting on March 22 and August 20, 1979 with personnel from Sandia Laboratories and the U.S. Geological Survey who are associated with the WIPP project. Their assistance in clarifying interpretations in the GCR and providing reference materials is acknowledged. They also indicated that a considerable amount of data has been obtained by the Department of Energy since the GCR was prepared over a year ago that will be published shortly.

PRINCIPAL CONCERNS

The following geotechnical questions, each of which bears on the short-term or long-term integrity of the site, are not resolved by the August 1978 information in the GCR and its references. DOE is continuing to gather and analyze data relevant to these features and processes.

1. What is the origin, evolution and occurrence of the high-pressure brine-reservoirs which were encountered in the upper part of the Castile formation in ERDA No. 6 and in at least 6 wells within 9 miles of the site? (See Section 2).

2. What is the origin, evolution and occurrence of the "breccia pipes" which have been encountered in the area? They may be localized deep dissolution features which originate in the lower portion of the evaporites and migrate upward. Such localized dissolution features could now exist or develop later beneath the proposed site (see Section 3).

3. What are the processes and rates of deep dissolution of salt near the site? There may be a preferential removal of the salt horizon which is proposed for the repository (see Section 3).

4. What are the regional and site hydrologic conditions for the aquifers above and below the evaporites? The hydrologic information is necessary to assess any possible long-term release of radioactive material from the repository (see Section 5).

Additional information on geological phenomena will be required by EEG in order to assess their significance in any potential release of radioactive materials to the surface and any effect on the health and safety of people and on the environment.

REVIEW OF GEOTECHNICAL ISSUES

1. Site Selection Procedures and Criteria

The following references suggested endorsement by different agencies in the selection of sites for an underground radioactive waste repository. Were they official recommendations by those agencies or were they made by individual staff members?

a) "...the USGS and ORNL selected the Permian Basin in New Mexico as best satisfying their site selection guidelines." (2-5.1)*?

b) "In the opinion of both ORNL and USGS, the two core holes, AEC 7 and 8, indicated acceptable subsurface geology at the ORNL site." (2-5.4)

c) "On November 14, 1975, the USGS recommended an area about seven miles southwest of the ORNL site for further examination"? (2-7.1)

There are several instances in which criteria appear to have been developed or altered to satisfy the condition of the WIPP site. A failure of the proposed repository to meet a given design criterion does not in itself mean there is a hazard. It does identify or flag those areas that need to be thoroughly analyzed to determine whether or not the consequences of failure could result in radiation exposure to people. The requirement (2-12.3) that the site be located at least one mile from a borehole penetrating the Salado formation was changed from two miles to one mile after the site at the ERDA No. 6 borehole was discovered to be unacceptable.

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*The notation (2-5.1) refers to Chapter 2, page 5, paragraph 1 of the GCR.

a) The GCR states that the studies of Snow and Chang (1975) and Walters (1975) allowed a more quantitative judgment on the question (2-6.2).

   1) What specific results of these studies justify the statement that "This buffer would assure more than a quarter of a million years of isolation using very conservative flow assumptions"?

   2) How are the conditions of those studies pertinent to the WIPP situation?

b) A report by Fader (1973) is also cited (2-12.3) as justification of the one mile criterion. However, this study was based on field observations of surface subsidence near abandoned wells in a Kansas salt bed and indicated that borehole dissolution can develop very rapidly in terms of geologic time. The area of surface subsidence was found to be approximately 1000 feet in diameter after about 31 years and subsidence of over 10 feet was observed. In view of these other studies, are there any analyses to substantiate the "quarter million years of isolation" and justify the one-mile criterion?

c) Griswold (1977, p. 12) is also cited (2-12.3) as providing justification for the one-mile criterion but Griswold's report only had a general statement that "This change, which resulted from studies performed for ORNL on the dissolutioning effects in boreholes, was made desirable by the extensive deep gas-exploration drilling in the Delaware Basin."

The ORNL criterion "no active mining within 5 miles (2-10.3) was also changed to "minimize existing potash lease rights in Zones I and II" (2-22.1).

The criterion that the site should be located one mile from a dissolution front (2-21) appears to be arbitrary in view of the uncertainty of the mechanism and rates of dissolution. According to Fig. 2-9, the dissolution front at the top of the Salado is located slightly inside the western boundary of Zone IV of the site. This would be about 1.8 miles west of the boundary of Zone II, the limits of underground storage in the proposed WIPP repository (4-39.1). How accurately is the location of the dissolution front and the rate of 6-8 miles per million years known?

2. Brine Reservoirs in the Evaporites

As noted on (2-11.2), an artesian brine flow was encountered at the original site at ERDA No. 6. Aspects of this brine occurrence are discussed in several locations in the report (e.g. 1-16.3, 1-31.4, 2-11.2, 4-67.3, 4-69.3, 6-19.4, 7-75.2, 7-90.1, 7-99.3, 7-102.4, 8-5.1). This approach together with some omissions has made it difficult to assess. The ERDA No. 6 brine, accompanied by concentrations of H_2S exceeding OSHA's standards for occupational exposure, was encountered on the flank of an extreme localized upthrusting structure from the middle of the Castile; dips as high as 70° were seen in the core and the middle anhydrite unit (A-11) has been displaced vertically by as much as 950 feet (Anderson and Powers, 1978,p. 79). According to a report to Sandia Laboratories [Tiab, 1977, p. 1]*, the well flowed at 662 barrels/day, but this data is not in the GCR. Tiab [1977, p. 6] also reported that the volume of the reservoir at ERDA No. 6 could be as large as 2 million barrels of water. The GCR reports Griswold's estimate of 100,000 to 1 million barrels in the discussion of lithium resources (8-5.1).

Seven wells have encountered brine reservoirs within a distance of 9 miles from the site. Griswold's (1977, page 42) Table XII (see below) gave data on four nearby oil wells

*References in brackets are listed in the end of this document; all other references are given in the GCR.

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Griswold (1977)

TABLE XII

Brine Flows From Nearby Wells

Well Name	Location	Flow Rate (bbl/day)	Depth of Flow (ft.)
ERDA No. 6	Sec. 35, T21S, R31E	600	2709
Hudson Federal	Sec. 1, T23S, R30E	12,000	2802
Culbertson-Erwin	Sec. 26, T22S, R32E	Strong	3515
Bootlegger Ridge	Sec. 36, T22S, R32E	20,000	3671
Gulf 1-A	Sec. 25, T22S, R32E	36,000	3600

The Castile brine reservoir encountered about 1/4 mile from the southwest corner of the outer boundary of the site at the Belco Hudson Federal well is discussed briefly (4-68.1). This occurrence, with an estimated flow of 12,000 barrels/day (Griswold, 1977), demonstrated that such brine flows are not always associated with the highly deformed region near the which have also encountered artesian brine reservoirs; these flows are typically an order to magnitude larger than that of ERDA No. 6. Griswold (personal communication to Lynn Gelhar, April 6, 1979) confirmed that two additional wells (Masco No. 1 & 2, Sec. 20, T22S, R33E) encountered artesian brine in the Castile. The GCR gives only sketchy information on these additional brine flows. Locations of "artesian brine flows" are shown in Figure 2-5 but the text (2-13.3) that describes the figure simply refers to "..., brine flow anticlines,..." without discussion. A more comprehensive discussion of these brine reservoirs should have been included in Chapter 6 on hydrology.

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Capitan reef. Griswold's Figure 6 (see Figure 1) graphically demonstrated the stratigraphic position of these brine flows in relationship to the proposed repository.

Several statements are made in the report implying that Castile brine flows are generally associated with anticlinal structures; these include: use of the term "...brine flow anticline..." (2-13.3), the criterion on anticlinal structures (2-22.1), the discussion of structure of the 124 marker bed on (4-69.3) and discussion of lithium resources (8-5.1). The structure of the top of the Castile (Figure 4.4-6) indicates anticlinal features within the immediate site area (e.g. at the north edge of Zone II) which are as severe as that associated with the Belco Hudson Federal brine flow. In view of the common association of the brine flows and anticlinal structures there appears to be little justification for the statement on (4-73.1); "There is no suggestion here of deformation of the type associated with artesian brine reservoirs."

The GCR makes no attempt to evaluate the possibility that geopressurized brine with H_2S may occur within the evaporites in places without anticlinal structures. Anderson (1976, p. 21-22) described such an occurrence at the UNM-Pokorney No. 1 location and noted that dissolution effects were similar to those observed at ERDA No. 6.

Brine reservoirs occur in the Castile formation which is directly below the Salado formation where the repository would be located and the brine reservoir at the ERDA No. 6 location has intruded locally up to the level of the lower part of the Infracowden salt. The origin, occurrence and configuration of these brine reservoirs are not adequately addressed in the GCR. For example, the geochemical analyses of the ERDA No. 6 brine are generally inconclusive with regard to the origin and period of isolation of the brine (see Section 6). It is conceivable that stress changes due to repository con-

Capitan reef. Griswold's Figure 6 (see Figure 1) graphically demonstrated the stratigraphic position of these brine flows in relationship to the proposed repository.

Several statements are made in the report implying that Castile brine flows are generally associated with anticlinal structures; these include: use of the term "...brine flow anticline..." (2-13.3), the criterion on anticlinal structures (2-22.1), the discussion of structure of the 124 marker bed on (4-69.3) and discussion of lithium resources (8-5.1). The structure of the top of the Castile (Figure 4.4-6) indicates anticlinal features within the immediate site area (e.g. at the north edge of Zone II) which are as severe as that associated with the Belco Hudson Federal brine flow. In view of the common association of the brine flows and anticlinal structures there appears to be little justification for the statement on (4-73.1); "There is no suggestion here of deformation of the type associated with artesian brine reservoirs."

The GCR makes no attempt to evaluate the possibility that geo-pressurized brine with H_2S may occur within the evaporites in places without anticlinal structures. Anderson (1976, p. 21-22) described such an occurrence at the UNM-Pokorney No. 1 location and noted that dissolution effects were similar to those observed at ERDA No. 6.

Brine reservoirs occur in the Castile formation which is directly below the Salado formation where the repository would be located and the brine reservoir at the ERDA No. 6 location has intruded locally up to the level of the lower part of the Infracowden salt. The origin, occurrence and configuration of these brine reservoirs are not adequately addressed in the GCR. For example, the geochemical analyses of the ERDA No. 6 brine are generally inconclusive with regard to the origin and period of isolation of the brine (see Section 6). It is conceivable that stress changes due to repository construction and/or heat generating wastes in combination with the high pressure in an underlying brine reservoir could induce deformation and fracturing which would release brine into the repository. In view of the large flows and volumes, high pressures, and accompanying H_2S of these reservoirs, they need to be characterized in detail in order to assess their role as a potential threat to health, safety and environment. Finally, future resource exploration could conceivably penetrate an underlying brine reservoir and bring pressurized brine to the surface.

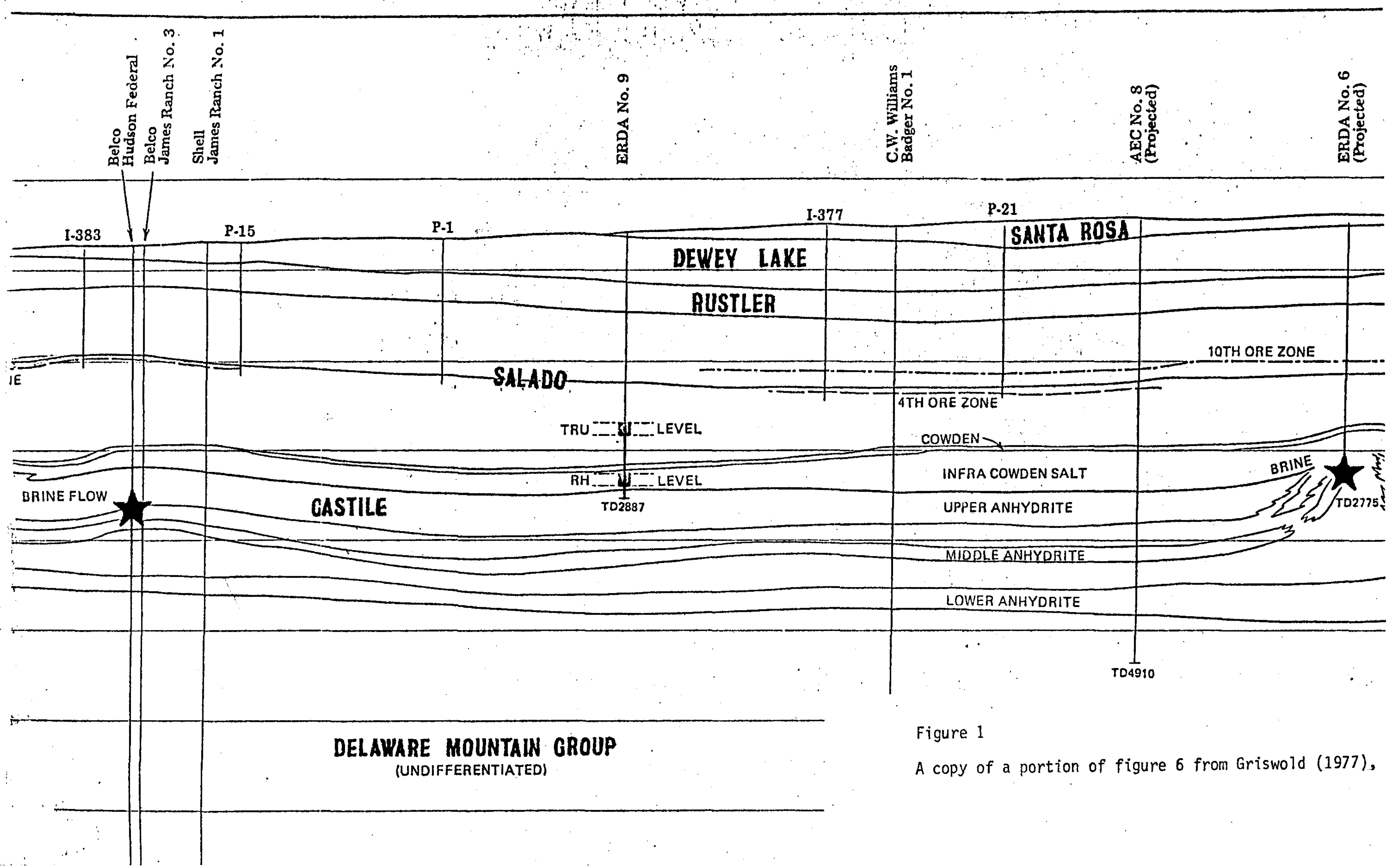

Figure 1

A copy of a portion of figure 6 from Griswold (1977),

3. Dissolution Processes and Rates

Salt dissolution can occur at different depths in the salt beds. The GCR presumes that the dissolution at the top surface of the Salado (shallow dissolution) is the most significant (1-26.2, 6-38). However, the presence of on-going deep dissolution, near the levels of the repository, has been suggested as a possible threat to the site. Both types of dissolution are discussed separately below.

Shallow Dissolution.

The GCR adopts the estimate by Bachman and Johnson (1973) that the lateral shallow dissolution front located 2 miles west of the site is approaching the site at 6 to 8 miles per million years and would take 225,000 years to reach a point 1,500 feet above the repository. However, the dissolution is probably not advancing eastward at a uniform rate. The front is envisioned as a "feather edge" (4-37.2) and certain tongues of the edge may move faster than others. Vertical dissolution is estimated at 0.33 to 0.5 feet per 1,000 years which would require 3 million years to remove the 1,500 feet of evaporites above the repository. Much of the salt in the Rustler formation directly above the repository has already been removed (4-41.3 and Figure 6.3-7) by dissolution along this front. How accurately is the location of the dissolution front and the rate of dissolution known?

Deep Dissolution.

The question arises whether deep dissolution is an on-going process in or near the horizons proposed for the repository and may play a role in helping to bring radioactive waste to the surface. EEG is presenting the following discussion in an attempt to help resolve the issue.

The GCR stated that exploratory holes, including ERDA 10 and ERDA 11 (6-2.1) as well as other data, indicate that deep dissolution is not taking place near the site. Anderson (letter to EEG, 4/24/79) has provided a photograph of a portion of the ERDA 10 core which shows a possible dissolution breccia in the lower anhydrite of the Salado formation. Anderson (letter to EEG, 5/14/79) also provided a photograph of core from WIPP 11 at a depth of 3100 feet showing clear replacement halite indicative of solution activity. The photographs are not included in this review since they do not reproduce clearly. EEG is sending them to the DOE for their interpretation. Why is ERDA 10 considered to be "...the nearest probable location of regional deep dissolution..." (6-42.1)?

According to Anderson (1978) deep dissolution within or below the salt has occurred extensively in the Delaware Basin. The GCR quoted Anderson's (1978) estimate that 50 percent of the original salt of the Delaware Basin evaporites has been removed (6-37.14) and that the salt from that unit will be gone from the basin in about another million years (6-45.3) but does not include his estimate that 73 percent of the lower Salado salt has been removed. Anderson's data indicate that the lower Salado salt beds, in which the disposal horizons are located, have been the most active zone of dissolution in the basin.

The GCR (6-45.2) appears to question Anderson's (1978) conclusion that dissolution is mainly a Cenozoic process and that it could remove the entire lower Salado salt in another million years, hypothesizing that deep dissolution may have been important in the Jurassic period. Anderson (1978) concluded that "The advancing effects of lateral dissolution can be expected to reach the disposal site before removal of the overlying salt beds." Are there any recent data to resolve this difference between Anderson and the GCR?

The GCR stated that "The proposed site is in an area of the Delaware Basin that is free of regional deep dissolution, but localized features are present in the vicinity" (6-41.3). What are these localized features?

The following geological phenomena may be related to deep dissolution.

Feature 5 mi. SE of site. The GCR discussed a map of Anderson on deep-seated dissolution features in the northern Delaware Basin and stated "The nearest of these deep mid-basin features to the proposed WIPP site...occurs about 5 miles southeast of the site..." (4-64.2). The sonic log of the Perry Federal #1-31 well in that area (Sec. 31, T22S, R32E, Fed-1, Figure 4.1-2) indicated that 200 feet of Infracowden salt was missing (R. Anderson, personal communication to Lynn Gelhar, 4/24/79). Furthermore, a major structural depression appeared in the 124 marker bed in that same location (see Figure 4.4-7). A similar depression in the 124 marker bed was found in the northern part of the WIPP site (see Figure 4.4-7, Sec. 9, T22S, R31E). The GCR suggests that this feature "...is not significant to the WIPP site" (4-70.1), citing isopach maps of Anderson (1978), and seismic reflection data. However, none of this information resolves the question of whether this depression is a deep-dissolution feature. The isopach maps (Figs. 4.3-4 through 4.3-7) do not cover the lower portion of the Salado or the Castile where the deep dissolution would be expected.

Anderson's regional isopach maps obviously are not going to resolve this feature because they are based on deep well data which are sparse in the site area. Seismic reflection data for the next horizon below the 124 marker bed (top of the Castile, Fig. 4.4-6) are inclusive in that area which is identified as "highly disturbed area".

Thinning of salt. Anderson's (1978) Figure 16 also showed a possible dissolution feature at the southwest edge of the site. The Infracowden salt (the high level waste horizon)

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Thinning of salt. Anderson's (1978) Figure 16 also showed a possible dissolution feature at the southwest edge of the site. The Infracowden salt (the high level waste horizon) thins rapidly toward the southwest and disappears completely about three miles from ERDA No. 9 in the area of this feature (Anderson 1978, Figure 7). The GCR (4-35.1) presents Jones' and Anderson's different interpretations of this thinning. Griswold (1977, p. 42) has suggested that brine pockets may be left behind a dissolution (suberosion) front. Might some of this thinning of Infracowden salt be associated with dissolution and brine flows in the area of the Belco Hudson Federal well?

The GCR (4-35.1) indicates that the lower member of the Salado is 1,195 feet thick at ERDA No. 9 and "...thins to 430 feet near the northeast corner of the area,..., due to beds missing at corrosion surfaces...". Where was the 430 foot thickness measured? "Corrosion" surfaces in ERDA No. 6 core samples are associated with dissolution (Anderson and Powers 1978). Is this thinning primarily due to faulting or is dissolution also a factor?

Origin of San Simon Sink. In discussing the origin of San Simon Sink, the GCR indicates that "shallow dissolution is a factor in the development of this sink" (6-40.3). In view of the depth of collapse, the scale of the feature, and its location along the reef margin, why is deeper dissolution not considered an equally good possibility?

Breccia pipes. Breccia pipes or "domal karst features" (Vine, 1960) are thought to be a result of localized removal of salt and are discussed on pages 2-17.2, 3-18, 4-7.1, 4-41.2 and 10-12.1. The second paragraph of the section on page 3-17 discussed Anderson's hypothesis that breccia pipes are caused by localized deep dissolution and brine density flow through fractures connected to an underlying aquifer. Anderson (1978) suggested that breccia pipes are formed by dissolution from below followed by collapse of the overlying units. Breccia pipes would then migrate upward and might eventually penetrate the surface. The deep dissolution theory of breccia pipe formation is consistent with conclusions reached in investigating the Michigan Basin [Landes, Ehlers, Stanley 1954] and collapse structures of the Prairie Formation Saskatchewan

[DeMille _et_ _al_, 1964]. Anderson suggested that these processes are ongoing and that breccia pipes are presently being formed or could be activated in the basin. If Anderson's ideas are correct, and in view of the presence of dissolution features near the site, it is not inconceivable that breccia pipes may exist or may develop beneath the proposed repository.

The GCR indicated that recent drilling (WIPP 13) of a suspected breccia pipe has shown that the resistivity anamoly is not caused by dissolutioning (4-7). What data formed the basis for this interpretation? Also, what additional studies are planned to resolve the origin, occurrence and significance of breccia pipes? Anderson's (1978) concept, as well as others, of the origin of dissolution features in the salt beds in the Delaware Basin should be treated in detail.

The nature and occurrence of deep dissolution has not been resolved by the information in the GCR. Anderson (1978) has presented evidence indicating that deep dissolution can play a significant role in the removal of salt and concluded "Extensive regional and localized dissolution in the Delaware Basin and the random distribution and on-going nature of localized dissolution suggests that this particular basin may have already progressed to a stage of dissolution where geological estimates of site integrity may not be obtained with the required degree of certainty" (page IV). How will the mechanism and rate of deep dissolution be determined?

Future course of the Pecos River. Another aspect of salt dissolution is its possible effect on the future course of the Pecos River. The thickness of the salt section decreases on the order of 1000 feet from the WIPP site to a point at the Pecos River directly west of the site and the corresponding difference in surface elevation is around 400 feet. If dissolution causes subsidence east of the Pecos River, it could cause the river to migrate eastward toward the site. If the course of the Pecos River is so altered, this could lead to accelerated dissolution near the site.

The GCR does note briefly that the course of the Pecos River may be affected by solution features (3-10), but does not consider possible eastward migration of the river or its role in ongoing and future dissolution. What is being done to evaluate this possibility?

4. Site Structure and Geophysical Exploration

Information on structural features within the evaporites in and around the WIPP site is needed to evaluate potential hazards such as brine reservoirs or breccia pipes which may be associated with deformation or dissolution in the evaporites. Potash exploration holes provide some detail on shallow features but the only direct information on the disposal levels in the lower Salado is from a single well, ERDA No. 9. All information in the GCR on the underlying Castile formation in the site area appears to be based on seismic reflection surveys.

The GCR identified several anomalous features which may be of concern including:

a) A resistivity anomaly in the northern portion of Zone II bearing some resemblance to the patterns associated with breccia pipes (4-7.1). The GCR said it is a shallow surficial feature with no disturbance of underlying beds, based on drilling WIPP 13.

b) A 70-foot depression at the top of the Salado formation, two miles NE of the site (4-7.1).

c) A depression in the 124-marker bed in the Salado formation on the northern edge of the site, one mile west of feature b) (Figure 4.4-7). According to Anderson (1978,

p. 78), this feature could be associated with faulting in the Bell Canyon formation. If so, this would suggest that the depression was formed by deep dissolution, possibly caused by movement of water upward from the Bell Canyon formation.

d) A missing section of Infracowden salt (base of the Salado formation) 5 miles southeast of the site which also shows up higher in the Salado formation as a depression in the 124-marker bed.

e) A salt anticline located on the northern boundary of the site originating in the Castile formation. WIPP 11 was drilled through this structure to the lower anhydrite of the Castile and did not encounter brine (4-68.3, 4-69).

f) Seismic reflection data suggested the presence of faults at the top of the Salado formation north of ERDA No. 9 but data from WIPP 18, 19, 21 and 22 showed no apparent faulting (4-72).

Several wells were drilled to evaluate some of these features (WIPP 11, 13, 18, 19, 21, 22) and in each case the interpretation was that the anomaly was not significant. No information is given in the GCR on the type of data that was collected from the wells or the depth of the wells. What data are available for these wells?

The configuration of the top of the Castile (Figure 4.4-6) is important because that horizon may reflect deformation related to brine reservoirs or deep dissolution. Figure 4.4-6 indicated several faults with vertical displacement up to 300 feet, a possible anticline northeast of the site, and a "highly disturbed area" extending to the northeast. The significance of these structures is recognized in the GCR (4-73.1). "Among aspects needing further investigation,

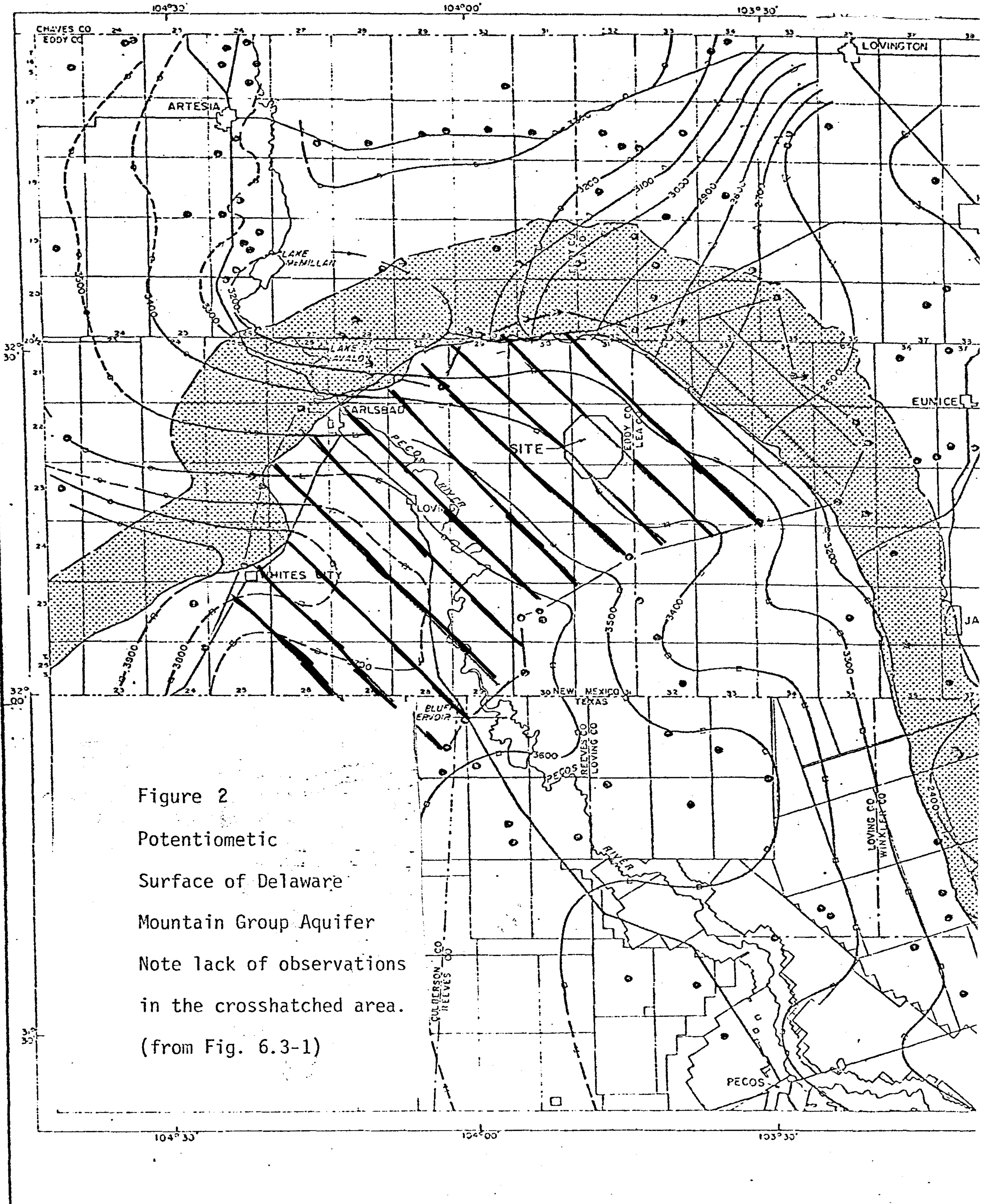


Figure 2

Potentiometic

Surface of Delaware

Mountain Group Aquifer

Note lack of observations

in the crosshatched area.

(from Fig. 6.3-1)

perhaps the most significant is a determination of the extent to which the upper levels of the Castile closest to the repository levels have been deformed by any salt deformation that may have taken place in the lower halite units of the Castile." Because of these uncertainties, the following statements in paragraph (4-73.1), (1) "There is no suggestion here of deformation of the type associated with brine reservoirs..." and (2) "This knowledge will permit a more detailed assessment relative to the location, design and construction of the storage facility but is not believed necessary for a general qualification of the site." do not appear warranted on the basis of information in the GCR or its references. In view of the complex structural feature indicated by Figure 4.4-6, what detailed justification is available?

Because of the possibility of localized deep-dissolution features (breccia pipes) in the site area, there is a need to define the ability of the seismic reflection method to detect such features. What information is available on the ability of the seismic reflection method or other geophysical techniques to detect breccia pipes or brine reservoirs?

5. Hydrology

Aquifers exist above, below and adjacent to the evaporite beds which are proposed for the repository. Therefore, subsurface hydrology at the site and in the region is a major concern because ground water flow controls the process of salt dissolution and is a primary transport mechanism for the release of radioactive material from the repository.

The underlying aquifer (Delaware Mountain Group) is discussed on (6-14); the potentiometric map for this aquifer (Fig. 6.3-1) is based on Hiss (1975). As shown in Figure 3, there is a large area around the site in which no well data were available. Are there additional data which could be included in this map? What about AEC 8? Also, would it be more useful to map the potentiometric surface of Delaware Mountain Group aquifers using a density near that of the aquifer waters?

The GCR (6-16.1) indicates that recharge to the Delaware Mountain Group occurs via precipitation on the outcrops and downward leakage through the younger rocks where the evaporites have been removed. If there is to be any assessment of the effects of climate change on this aquifer, it is necessary to know more specifically the locations and conditions in the recharge area(s) of the Delaware Mountain Group aquifer.

Data on the overlying aquifers are more extensive but the regional potentiometric map for the Rustler aquifers (Figure 6.3-2) contains no data from east of the site. The degree of connection between the brine aquifer and the Rustler aquifers east of the site should be clarified. Do the two aquifers have the same potentiometric surface (Figure 6.3-2)? The GCR contains no specific information on the nature of recharge to the Rustler aquifer; on (6-9.4) it states "...is presumed to be from precipitation in outcrop areas or from overlying formations." How much recharge occurs and what fraction of this enters the aquifer from the outcrop areas, and what fraction from the overlying formations? How will the recharge and flow in the Rustler aquifer be affected by climatic change? The summary of the Mercer and Orr (1978) report in the GCR (6-30.2) noted that head distributions in the Culebra Dolomite indicate fluid movement to the southeast across the site. This observation is not consistent with the regional potentiometric map (Figure 6.3-2) which indicated flow toward the southwest. How are these differences reconciled?

The summary stated "...measurements of the effective porosity...are very difficult to obtain". Have ranges been established of the effective porosity in the site area? This parameter is important in the hydrologic transport modeling of the long-term release of radioactive material from the repository. Aquifers adjacent to the evaporite beds are also of importance especially in relation to possible deep-dissolution in the evaporites. The discussion of the Capitan aquifer (6-17) does not consider the possibility of some lateral connection between the reef aquifer and the salt beds with the resulting development of deep-dissolution wedges and reef-margin dissolution features (Anderson, 1978).

The statement on (6-10) referring to the Castile formation "...as an aquiclude separating the Salado from the underlying sandstones of the Delaware Mountain Group" does not consider the possibility of fracture permeability which may be associated with deformation in the Castile, as suggested by Anderson (1978) in relation to deep dissolution.

The question of the effect of climate change on the hydrologic regime is not addressed.

Generally the hydrologic data for the region and the site is not adequate to characterize the long-term hydrologic transport of radioactive materials or to define the mechanism and rates of salt dissolution.

6. Geochemical Analyses

The analyses of waters from ERDA No. 6 (Sections 7.7 and 7.9) are intended to resolve questions about the nature and origin of this brine reservoir; the appropriate questions are stated on (7-90). On (7-102.4) the GCR concludes, on the basis of the uranium disequilibrium model, that the ERDA No. 6 reservoir "...has no connection with any other known ground water, and has been in its present environment for at least 880,000 years." This conclusion is unjustified. A more qualified statement is found on (7-99.1) that assumes an original connection to the Capitan reef and an original activity ratio equal to that of the Capitan reef water at "Carlsbad 7", $\alpha_0 = 5.14$.

One could assume as an initial value the results for the Middleton (α_0 = 1.81) or Hackberry (α_0 = 1.22) waters (Table 7.27) which also come from the Capitan aquifer and are geographically closer to ERDA No. 6 (see Fig. 7.18). From Figure 7.20 these values of α_0 yield much smaller ages; about 300,000 years for Middletone and a "negative" age for Hackberry. The results of the uranium-disequilibrium dating are stongly dependent on an assumed α_0 and do not provide any definitive information about the age or degree of isolation of the ERDA No. 6 reservoir.

On (7-99.1) it is indicated that the solid, liquid and gaseous phases are not in chemical equilibrium in the ERDA No. 6 brine reservoir. A static body of water would be expected to come to chemical equilibrium with surrounding rocks within a relatively short time frame. (Opinions expressed to EEG ranged from weeks to decades). Yet this presumably very old water was not found to be in chemical equilibrium with the host rock. One or more of the following is implied:

a) The laboratory analyses of brine, gas and stable isotopes are in error.

b) The ERDA No. 6 water is not "old", i.e., the interpretation of the uranium-disequilibrium method in the GCR (7-99) is invalid.

c) Water which has been in contact with other types of rocks is mixing with the brine reservoir fluids and preventing equilibrium.

d) Biogenic reactions which prevent equilibrium are occurring.

e) The surrounding rocks are not halite or anhydrite, contrary to core data.

As stated on (7-78) "The isotopic composition of ERDA No. 6 brine is consistent with an approach to isotopic equilibrium between water and clay minerals, not necessarily in the Castile." This observation is an additional indication that the ERDA No. 6 brine may not be indigenous to the Castile; this possibility is further supported by the following:

a) Gas analyses from ERDA No. 6 (7-76) predict a calculated P_{O_2} of 8.8×10^{-83} using CO_2-CH_4 redox equilibria. This value is not compatible with sulfate evaporite mineral assemblages.

b) Results of the synthesis of ERDA No. 6 water chemistry (Sample 14, Table 7.20) using the U.S. Geological Survey computer program WATEQF indicate that the water is undersaturated with halite, anhydrite and gypsum, but is saturated with calcite and dolomite.

c) WATEQF calculates a P_{CO_2} of $10^{-2.09}$; P_{CO_2} calculated from the basis of calcite saturation is $10^{-1.98}$ indicating that ERDA No. 6 water is in equilibrium with calcite.

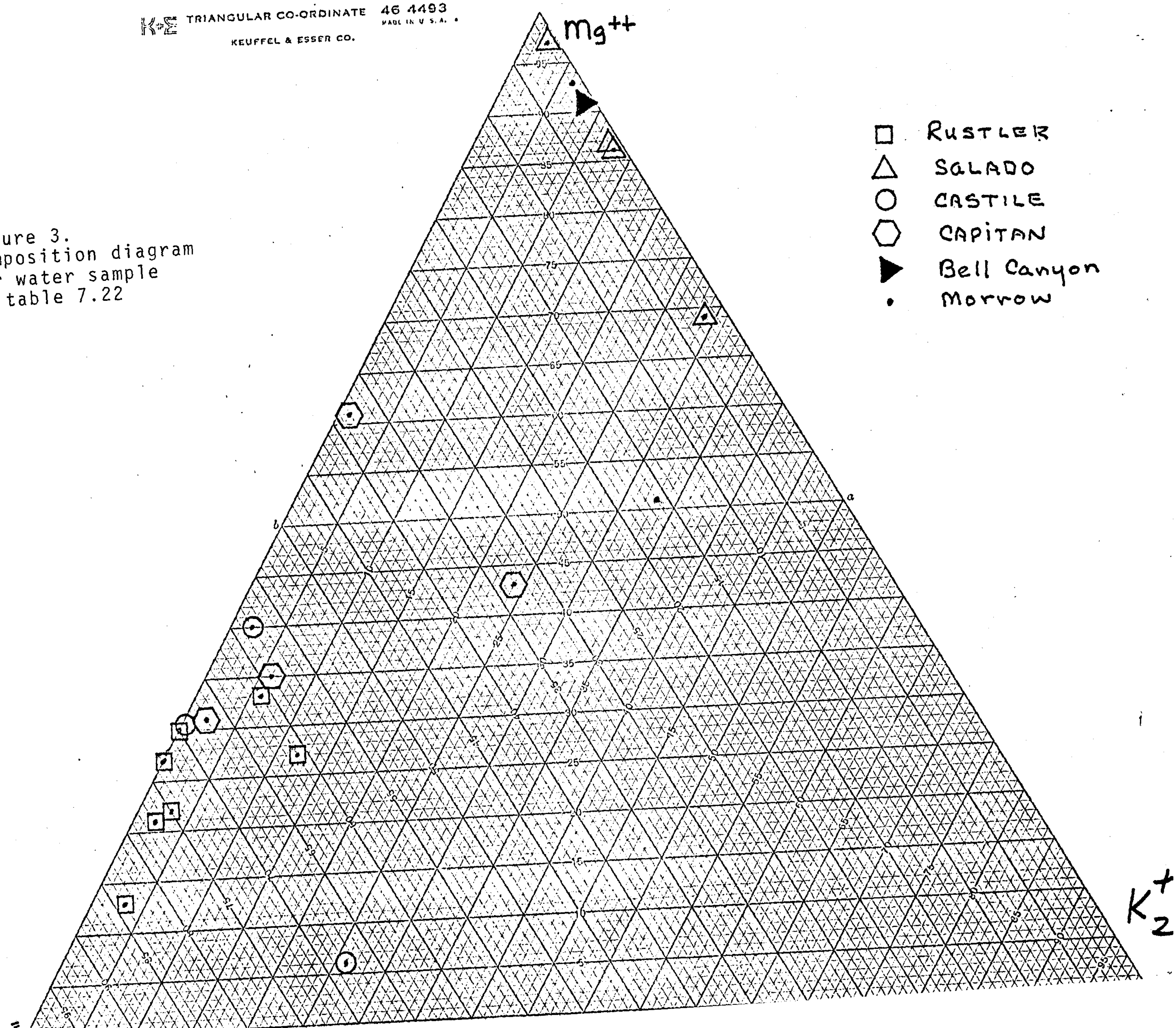
These points suggest the possibility that ERDA No. 6 water has recently been in contact with dolomite and calcite, e.g. the Capitan limestone and associated dolomites (Figure 3.3-2) or the Bell Canyon formation which "contains considerable limestone interbeds and lime rich intervals" and whose top unit consists of limestone (Figure 4.3-2). Two possible routes for water movement from the Capitan reef were suggested by Anderson (1978, pp. 69, 72): along the contact between the Castile and Salado formation, or moving upward from the Bell Canyon formation through fractures in the Castile formation.

The stable isotope data for ERDA No. 6 water (Figure 7.12) indicated isotopic composition which is distinct from modern meteoric waters. However, these data do not give a positive indication of the origin of the water in the brine reservoir.

Regional water chemistry information is important because it can help to understand salt dissolution conditions and clarify the interpretation of the regional hydrology. However, the data are limited and are not systematically related to the possible flow system. There are two water samples from the Bell Canyon aquifer listed in the GCR (AEC 8 on Table 6-3.3 and Sample No. 9 on Table 7.20) and they have different chemical compositions. The composition diagram of Figure 3 was prepared from concentrations of 3 of the 8 ions listed on Table 7.20. Only one Bell Canyon water sample is included (No. 9). Figure 3 suggests that this Bell Canyon water sample has a composition similar to the three water samples from the Salado. This is consistent with the observation in the GCR that "this brine probably did not originate in the Bell Canyon, but its solutes probably came from nearby evaporites" (7-74.3). A possible interpretation of these observations is that solutes in the Bell Canyon aquifer have originated in the overlying evaporites. While this is not conclusive, it supports the Anderson (1978) hypothesis of deep dissolution through connection with the underlying aquifer.

None of the questions posed on (7-90) concerning the ERDA No. 6 reservoir are resolved with any degree of certainty by the geochemical information in Chapter 7; all that can be said is that the origin and evolution of such brine reservoirs in the evaporites remains a mystery.

Figure 3.
Composition diagram
for water sample
of table 7.22



7. Rock Properties and Special Studies

The special studies of Chapter 9 are not being reviewed at this time. These studies are very important for the evaluation of the long-term integrity of a proposed repository, but Chapter 9 generally covers ongoing work which has not come to definitive conclusions. The following are some general concerns which will have to be resolved before the long-term integrity of the site can be evaluated.

a) The thermo-mechanical properties of salt and other adjacent rock are very complex especially when one is concerned with the regional-scale deformation induced by a proposed repository. The *in situ* properties of rocks at this scale can be strongly affected by impurities, heterogeneity and fracturing which is not accounted for in laboratory experiments of the kind described in Chapter 9. This problem is recognized on (9-2.3).

b) The radionuclide sorption properties of rocks are important in the hydrologic transport modeling of the long-term radioactive releases from a proposed repository. The static experiments with powdered rock samples (9-26) may not be relevant to the flow in fractured rock that would be involved in the Rustler aquifer (6-30.4).

   Additional laboratory and field work will be required to realistically describe the *in situ* sorptive properties of rocks at the site.

c) Migration of brine inclusions within the salt under the influence of waste induced thermal gradients could lead to undesirable accumulations of water in the repository. On (7-68.2) "Accurate predictions of the behavior of *in situ* inclusions in the thermal gradient around a canister in salt cannot be obtained at this time...". More information is required on this phenomenon.

8. General Comments

"The purpose of the GCR is to provide an account of the known geotechnical information considered relevant to site selection (see Section 2-3) for the proposed WIPP site" (2-2.2). Further discussion on (2-2.2) states "...for the most part, specific judgments regarding the suitability of the site are not made." A conclusion is presented on (2-7.1) that "Sufficient information has now been developed to allow the site to be adequately characterized for site selection purposes," and on (2-23.6) the statement is made that "...much basic information has been gathered indicating no major technical problems with the site as it is now understood."

Based on our review of the GCR and its references, additional data collection and analyses will be required in order for EEG to conduct its assessment of any potential release of radioactive materials to the surface and any subsequent effect on the health and safety of people and on the environment. Critical analyses of the role of localized geological perturbations such as breccia pipes, brine pockets and varying rates of dissolution are required for hydrologic transport modeling.

References

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De Mille, G., et al. "Collapse Structures Related to Evaporites of the Prairie Formation, Saskatchwan," Geological Society of America, v. 75, (1964), pp. 307-316.

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APPENDIX IV

RADON RELEASES FROM WIPP

Summary

The quantities of radon expected to be released from operation of the WIPP site were calculated without any local measurements. The situation is discussed in greater detail below.

WIPP DEIS Radon Emanation Estimates

The DEIS uses a value of 0.9 curies/year of Radon-222 and 0.04 curies/year for Radon-220 obtained from the Nuclear Regulatory Commission's Final Generic Impact Statement on Mixed Oxide Fuels (GESMO). The DEIS also uses a typical concentration of 1×10^{-9} curies/cubic meter in exhaust air. These values are apparently estimates since they are unreferenced in the GESMO document.

There are no data in the DEIS on either uranium levels or radon at the surface or in the Salado formation where extensive mining would occur. In general, the Carlsbad area has a slightly lower than average terrestrial radiation, which suggests low uranium concentrations in the soil. While salt beds are usually low in uranium content, there are significant variations in different areas or geological formations and averages cannot be presumed. It is interesting to note that Carlsbad Caverns has high enough radon levels (up to 0.25 Working Levels) for the National Park Service to be concerned about exposure to their employees. (Ref. 1).

Since concentrations of radon from mines can vary over two to three orders of magnitude, it is necessary to obtain uranium, radium and radon data underground and at the surface. Three areas of environmental concern include:

(1) the amount of radon exhausted from the repository;

(2) the quantity of radon that emanates from the salt stored above ground; and

(3) the radon naturally emanating from the salt storage area (so that it can be estimated whether salt storage increases or reduces net emanation).

The possible doses from radon, while low, could be significant compared to other releases from the repository. See the attached calculations.

Radon Dose Estimates

DEIS Estimates

p. 8-33 $Q = 0.90$ Ci/y ^{222}Rn, 0.04 Ci/y ^{220}Rn

p. H-59 $\frac{X}{Q} = 6.2 \times 10^{-7} \frac{s}{m^3}$ at James Ranch (3 miles SSW of site)

$$\frac{pCi}{m^3} = \frac{9 \times 10^{11}\ pCi/y}{3.15 \times 10^{7}\ \frac{s}{y}} \left(6.2 \times 10^{-7}\ \frac{s}{m^3}\right) = \underline{\underline{0.18\ pCi/m^3}}$$

$$\text{Dose, mrem/y} = 0.18\ \frac{pCi}{m^3} \left(\frac{0.625\ \text{mrem/y}}{\frac{pCi}{m^3}}\right) = \underline{\underline{0.011\ \text{mrem/y}}}$$

Bronchial Epithelium

[conversion factor is from G-44 in GEIS Uranium Mills and is the average dose a resident indoors would get from an outside concentration of 1 pCi/m^3 of ^{222}Rn. This dose is to bronchial epithelium. Dose to pulmonary lung is only 4-11% of this (pages 6-39 and 6-67 of GEIS)]

p. 9-39 The DEIS uses a dose of 2.5×10^{-4} mrem/y to lung. This can be checked by the above information and the Dose Committment Factor from NUREG-0172, Table 8.

$$\text{Dose} = \frac{pCi}{m^3} \left(\frac{m^3}{y}\ \text{inhalation}\right) \left(\text{DCF}\ \frac{\text{mrem}}{\text{pCi inhaled}}\right)$$

$$= 0.18\ (7300)\ (2.05 \times 10^{-6}) = 2.6 \times 10^{-4}\ \frac{\text{mrem}}{y}$$

Reasonableness of DEIS Estimate

Assume an emanation rate from the mine equal to the average rate (1 pCi/sec-m^2) and that there may be as much as 5×10^5 m^2 exposed area in the tunnels and storage rooms (this could be high by factor of 2-3, depending on the plan for opening and closing rooms in the repository).

Then:

$$\text{Ci/y} = 5 \times 10^5 \text{ m}^2 \left(10^{-12} \frac{\text{Ci}}{\text{m}^2\text{-s}}\right) 3.15 \times 10^7 \frac{\text{s}}{\text{y}} = \underline{\underline{15.7}} \text{ Ci/y}$$

From Mine

No estimate has been made from possible salt pile radon. If radon emanation is even 1 pCi/m^2 - s above background, then for a 30 acre pile:

$$\text{Ci/y} = 30 \text{ Ac. } (4.05 \times 10^3 \frac{\text{m}^2}{\text{AC}}) \; 3.15 \times 10^7 \frac{\text{s}}{\text{y}} \; (10^{-12} \frac{\text{Ci}}{\text{s-m}^2})$$

$$\text{Ci/y} = \underline{\underline{3.8}} \text{ Ci/y} \quad \text{from pile (per 1 pCi/m}^2 \text{ - s)}$$

These values may be higher than exist at the site but they are by no means an upper bound, e.g. a variety of metal mines have been observed to have radon concentrations 2-40 times those assumed here. Evaporite deposits are typically an order of magnitude lower, yet in some cases they contain commercial ore deposits. Also, there is poor correlation between radon levels and ore geochemistry.

Possible Radon Doses

Assume 20 Ci/yr. radon release and check doses at two closest privately owned areas (James Ranch, use 3.0 mile SSW; and 2.8 mile NW) where residences are possible. Also check dosage at maximum 0.5 mile sector, because these areas are open to people.

Inhalation Dose

$$\text{millirem/yr.} = \frac{PCi}{s} \left(\frac{\chi}{Q} \; \frac{s}{m^3}\right) \frac{0.625 \text{ mrem/y}}{PCi/m^3}$$

$$= \frac{20 \times 10^{12} \frac{PCi}{y}}{3.15 \times 10^7 \frac{s}{y}} \quad (6.2 \times 10^{-7}) \; 0.625 = \underline{\underline{0.25}} \text{ mr/yr.} \quad 3.0 \text{ mi S}$$

$$= 6.34 \times 10^5 \quad (2.9 \times 10^{-6}) \; .625 = 1.1 \text{ mr/yr.} \quad 2.8 \text{ mi N}$$

$$= 6.34 \times 10^5 \quad (9 \times 10^{-5}) \; .625 = \underline{\underline{35.7}} \text{ mr/yr.} \quad 0.5 \text{ mi W}$$

Regional Annual Radon Dose Committment

A crude estimate of the total dose to the regional population from all pathways can be obtained by prorating the doses shown on Table 6-15 (attached) from the Generic Environmental Impact Statement on Uranium Milling (NUREG-0511). Adjustments need to be made for the difference in regional population (94,050 for WIPP versus the 57,300 used in Table 6-15) and for the curies of radon released (7,000 versus 20 curies/year).

$$\text{Adjustment Factor} = \frac{94{,}050}{57{,}300} \; \frac{20}{7000} = .00469$$

$$\text{Bronchial Epithehium Dose} = 138 \text{ man-rem } (.0047) = \underline{\underline{0.65}} \text{ man-rem}$$

$$\text{Bone Dose} = 53.3 \; (.33) \; (.0047) = \underline{\underline{0.08}} \text{ man-rem}$$

$$\text{Whole Body Dose} = 6.47 \; (.35) \; (.0047) = \underline{\underline{0.01}} \text{ man-rem}$$

$$\text{Lung Dose} = 12.8 \; (.50) \; (.0047) = \underline{\underline{0.03}} \text{ man-rem}$$

Dose Comment

The regional doses from a 20 curie/year radon release are small. They are, however, similar in magnitude to the total doses presented in Table 9-19, not trivial as suggested on page 9-39.

It is also noted that the individual doses calculated in Table 9-18 are for James Ranch. Areas outside the security fence have χ/Q values as large as 9×10^{-5} s/m^3 (WNW at 0.5 mile) which is 145 times as large as used here. This needs further consideration.

Table 6.15 Annual Population and Environmental Dose Commitments Resulting from Operation of the Model Mill

Exposure Pathway	Annual Population Dose Commitments, person-rem/yr[a]							
	Total Dose Commitments				Doses Received by the Regional Population[b]			
	Whole Body	Bone	Lung	Bronchial Epithelium[c]	Whole Body	Bone	Lung	Bronchial Epithelium[c]
External from ground	0.511	0.511	0.511	-	0.511	0.511	0.511	-
External from cloud	2.36	2.36	2.36	-	2.36	2.36	2.36	-
Inhalation	0.170	4.61	6.48	138.	0.170	4.61	6.50	138.
Vegetable Ingestion	3.58	50.0	3.58	-	2.74	38.3	2.74	-
Meat Ingestion	2.94	30.24	2.94	-	0.437	4.49	0.437	-
Milk Ingestion	0.458	5.5	0.458	-	0.253	3.04	0.253	-
TOTALS	10.0	93.2	16.3	138.	6.47	53.3	12.8	138.

Exposure Pathway	Annual Environmental Dose Commitments, person-rem/yr							
	Total Dose Commitments				Doses Received by the Regional Population[b]			
	Whole Body	Bone	Lung	Bronchial Epithelium[c]	Whole Body	Bone	Lung	Bronchial Epithelium[c]
External from ground	1.97	1.97	1.97	-	1.97	1.97	1.97	-
External from cloud	2.36	2.36	2.36	-	2.36	2.36	2.36	-
Inhalation	0.170	4.61	6.50	138.	0.170	4.61	6.50	138.
Vegetable Ingestion	4.52	59.6	4.52	-	3.46	45.6	3.46	-
Meat Ingestion	4.78	48.7	4.78	-	0.710	7.24	0.710	-
Milk Ingestion	0.725	8.45	0.725	-	0.400	4.66	0.400	-
TOTALS[d]	14.6	125.7	20.9	138.	9.07	66.4	15.4	138.

[a]Based on exposure during the final year of mill operation.

[b]Doses received by the regional population are less than total doses for ingestion pathways because the regional population consumes only 76.5%, 14.9% and 55.2% of regionally produced vegetables, meat, and milk, respectively.

[c]Doses presented for the bronchial epithelium are those resulting from inhalation of short-lived Rn-222 daughters.

[d]The following percentages of annual dose commitments received by the region are due to annual radon releases (7.0 kCi): whole body, 35%; bone, 33%; pulmonary lung, 50%; and bronchial epithelium, 100%.

References - Appendix IV

1. U.S. Environmental Protection Agency. Radiation Protection Activities (EPA-520/4-77-005), 1976.

2. U.S. Nuclear Regulatory Commission. Draft Generic Environmental Impact Statement on Uranium Milling (NUREG-0511), April 1979.

3. U.S. Environmental Protection Agency. Radioactivity in Selected Mineral Extraction Industries, A Literature Review (ORP/LVF-79-1), November 1978.

APPENDIX V

ATMOSPHERIC DISPERSION COEFFICIENTS

I Long Term Averages

Procedure Used. Several key χ/Q values were calculated by EEG by a separate hand calculation to determine if the values from Table H-36 in the Draft EIS were reasonable. The procedure used differed in several ways from the MESODIF Code:

(1) It did not account for a plume being blown back over the source to contribute on the "second pass".

(2) It did not include the effect of releases during the 0.7% of the time that winds are below 0.3 meters per second ("calm conditions").

(3) The horizontal and vertical dispersion coefficients (σy, σz) for Category F were used for Category G conditions.

All of these differences would tend to make the calculated values of χ/Q less than those obtained from MESODIF.

The basic expression used was:[1]

$$\frac{\bar{\chi}}{Q}\text{ long-term} = \Sigma_i = A,B,C,D,E,F \; \frac{\left(\frac{2}{\pi}\right)^{\frac{1}{2}} fF_i}{\sigma z_i \, \mu_i \, \frac{2\pi x}{n}} \; e^{-\frac{h^2}{2\sigma z^2}}$$

where

$\bar{\chi}$ = annual average downwind concentration in the sector of interest at distance x downwind, $\mu Ci/m^3$

Q = annual average discharge rate, $\mu Ci/s$

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(1) From: Fowler, Ted W., "Air Submersion Skin Surface Dose Rate from Noble Gases", EPA Radiological Review Guidelines No. 3, May 1973.

f = frequency of the time that wind blows in a given sector

$\frac{2\pi x}{n}$ = sector width where x is the downwind distance and n is number of sectors

F_i = annual stability persistence frequency for the f_i th meterological conditions (A,B,C,D,E, and F)

σ_{z_i} = vertical stability parameter for the i th meteorological condition at distance x downwind, m

$\bar{\mu}_i$ = wind speed at ground level for the i th meteorological condition, m/s

h = effective stack height, m

if elevated releases are ignored and 16 sectors are used this reduces to:

$$\frac{\bar{\chi}}{Q} = \Sigma_{i = A,B,C,D,E,F} \frac{2.032 fFi}{\sigma_{z_i} \mu_i x}$$

The normal use of this equation would be where only the frequency of stability categories (in Table H-35 for the WIPP site) and the annual distribution of wind direction (Table 20, Annex 1, of Appendix H) is known. Average values of wind speed ($\bar{\mu}_i$) are used for each stability category (from Fowler). A sample calculation is shown below for the northwest downwind sector at 2.8 miles from the site.

Stability Category	F_i	$\bar{\mu}_i$ $\left(\frac{m}{s}\right)$	σ_z (m)	$\frac{F_i}{\sigma_{z_i}\ \bar{\mu}_i}$ $\left(\frac{s}{m^2}\right)$
A	.255	2.46	2600	4.00 - 5
B	.033	2.69	570	2.11 - 5
C	.031	3.98	240	3.34 - 5
D	.136	5.86	83	2.77 - 4
E	.105	3.68	53	5.28 - 4
F	.119	2.00	33	5.07 - 3
G	.216			
			=	5.97 - 3

For NW downwind sector (f = 0.182)

$$\frac{\chi}{Q} = \frac{2.032\ (.182)(5.97-3)}{4.5+3} = \underline{\underline{(4.9 - 7)}}\ \frac{s}{m^3}$$

DEIS value from H-36, by interpolation = $(3.3 - 6)\frac{s}{m^3}$

A similar calculation at 3.0 miles SSW of the site gives a value of $(7.8 - 8)\frac{s}{m^3}$ compared to the DEIS value of $(6.2 - 7)\frac{s}{m^3}$

The DEIS contains more meteorological data than needed for the above simplified calculation. Use can be made of these data by using actual data for wind speed and stability category frequency in each downwind sector. An example is shown below for 2.8 miles northwest of the site. Wind speed frequency for each stability category is obtained from Tables 13-20 in Annex 1, Appendix H for the southeast wind direction.

Stability Category	μ_i (m/s)	fF_i	σ_z (m)	$\frac{fF_i}{\sigma_z \mu_i} \left(\frac{s}{m^2}\right)$	
G & F	0.85	.033	33	1.28 - 3	
	2.25	.038	.	0.51 - 3	
	4.05	.002		0.02 -3	1.81 - 3
E	0.85	.003	53	0.67 - 4	
	2.25	.028		2.34 - 4	
	4.05	.022		1.02 - 4	
	6.55	.005		0.14 - 4	0.42 - 3
D	0.85	.001	83	0.14 - 4	
	2.25	.009		0.48 - 4	
	4.05	.005		0.15 - 4	0.08 - 3
C	2.25	.001	240	0.02 - 4	
	4.05	.001		0.01 - 4	0.00 - 3
B & A	Negligible by inspection				
					Σ = 2.31 - 3

$$\left(\frac{\chi}{Q}\right)_{2.8 \text{ mi NW}} = \frac{2.032\ (2.31 - 3)}{4.5\ +3} = (1.0 - 6)\frac{s}{m^3}$$

DEIS value from H-36, by interpolation $= (3.3 - 6)\frac{s}{m^3}$

A similar calculation at 3.0 mi SSW of the site gives a value of $(1.4 - 7)\frac{s}{m^3}$ compared to the DEIS value of $(6.2 - 7)\frac{s}{m^3}$.

Also, a calculation for 0.5 mi NW downwind sector was $(1.6 - 5)\frac{s}{m^3}$ compared to $(6.2 - 5)\frac{s}{m^3}$ in DEIS.

II Short Term Averages

The $(\frac{\chi}{Q})_{5\%}$ and $(\frac{\chi}{Q})_{50\%}$ one hour frequencies were calculated from inspection of Tables 16-19, in Annex 1, Appendix H. The key factor in determining each value is to find the $\sigma_{z_i} \mu_i$ value that is minimum at a given distance. This is done in the tabulation below at 800 m.

Stability Category	σ_{z_i} (m)	μ_i $(\frac{m}{s})$	$\sigma_{z_i} \mu_i$ m^2/s	Inverse Ranking
G & F	12	0.85	10	1
		2.25	27	4
		4.05	49	7
		6.55	79	
E	18	0.85	15	2
		2.25	41	5
		4.05	73	
		6.55	117	
D	27	0.8	23	3
		2.25	61	8
C	50	0.8	42	6

A SE wind direction (NW downwind sector) occurs 18.2% of the time. Consequently, the 5% frequency is 0.91% of the total time; the 50% frequency is 9.1%.

Percent occurrence: $G_{.85} = 2.6$; $F_{.85} = 0.7$; $D_{.85} = 0.1$; $E_{.85} = 0.3$; $G_{2.25} = 1.8$; $F_{2.25} = 2.0$; $E_{2.25} = 2.8$.

$\therefore \ (\frac{\chi}{Q})_{5\%}$ = G stability, 0.85 $\frac{m}{s}$ speed.

$$= \frac{2.032\ (1)}{10(805)} = (2.5 - 4)\frac{s}{m^3}$$

DEIS has $(2.8 - 4)\frac{s}{m^3}$

$(\frac{\chi}{Q})_{50\%}$ = E stability, 2.25 $\frac{m}{s}$

$$= \frac{2.032\ (1)}{41(805)} = (6.2 - 5)\frac{s}{m^3}$$

DEIS has $(6.3 - 5)\frac{s}{m^3}$

III Summary of Results

ATMOSPHERIC DISPERSION COEFFICIENTS
(Comparison of EEG and DEIS values)

Downwind Sector	Long-Term Values DEIS s/m^3	EEG s/m^3	RATIO DEIS/EEG		One-hour Values DEIS s/m^3	EEG s/m^3	RATIO DEIS/EEG
SSW, 3.0 mi	6.2-7	(a) 7.8-8	7.9	(c)	5.6-5	1.5-5	3.7
		(b) 1.4-7	4.4	(d)	5.8-6	2.1-6	2.8
NW, 2.8 mi	3.3-6	(a) 4.9-7	6.7	(c)	6.5-5	1.6-5	4.1
		(b) 1.0-6	3.3	(d)	8.3-6	3.8-6	2.2
NW, 0.5 mi	6.2-5	(b) 1.6-5	3.9	(c)	2.9-4	2.5-4	1.2
				(d)	6.3-5	6.2-5	1.0

(a) Short method, using average stability category frequencies from H-35 and wind speed.

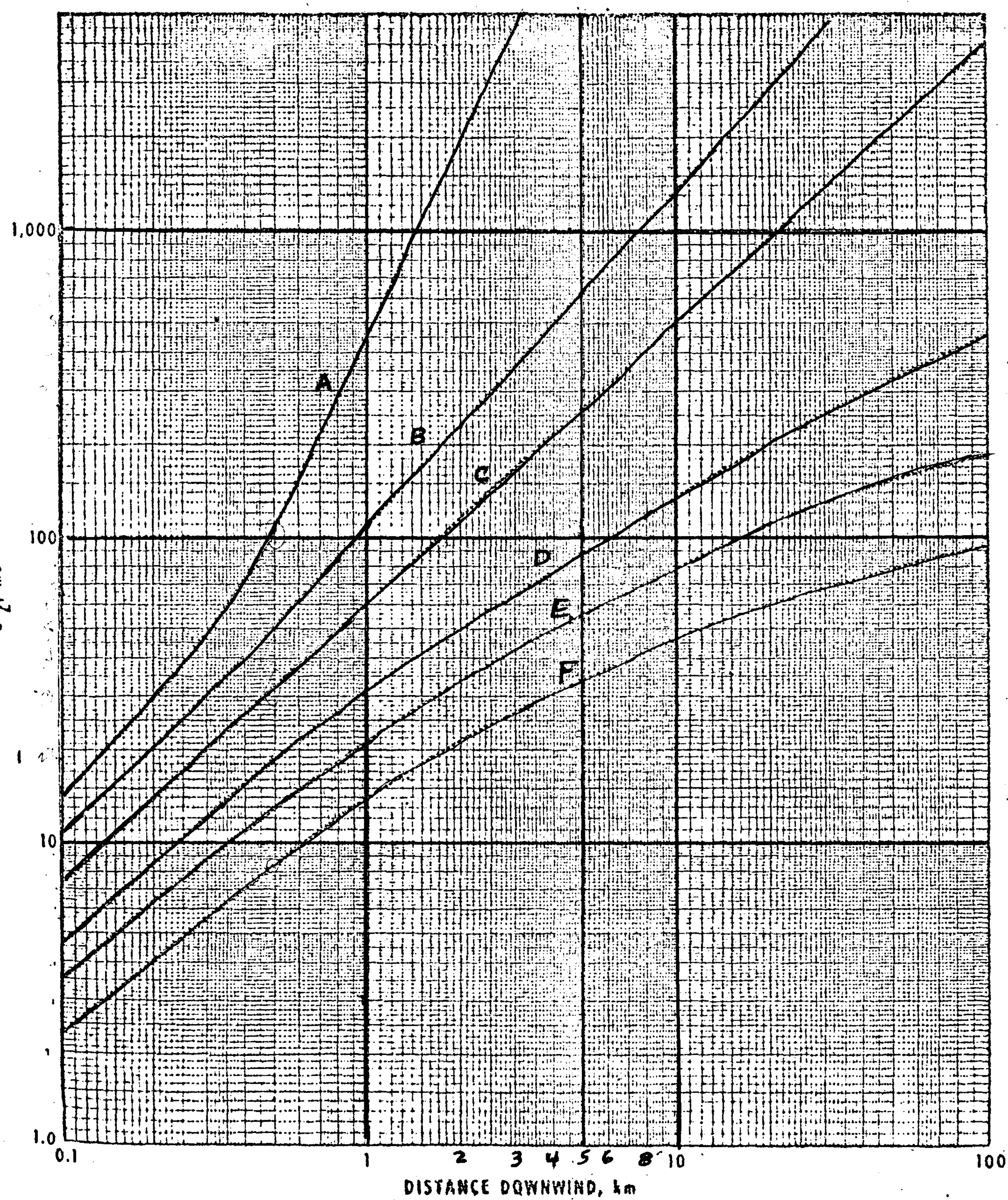
(b) Procedure using specific meteorological data in Annex 1, Appendix H.

(c) 5% occurrence, one hour frequency (Table 21, Annex 1, Appendix H).

(d) 50% occurrence, one hour frequency (Table 21, Annex 1, Appendix H).

The agreement is not too close in most cases with the values in the DEIS being consistently higher. Higher values in the DEIS are to be expected from the differences in models. The use of stability Category G parameters and the incorporating of calm winds into the model might reasonably increase the calculations by more than two-fold. Including the plume being blown back on the source would tend to increase the values some also. This is considered satisfactory agreement and it is concluded that the values used in the DEIS are reasonable.

Figure 1 - Vertical dispersion coefficient as a function of downwind distance from the source (2)



(2) Turner, D. Bruce, "Workbook of Atmospheric Dispersion Estimates", U.S. Department of Health, Education, and Welfare, Public Health Service, 1967.

IV Air Quality χ/Q

For calculation of the 24-hour ground-level concentration of particulates, the DEIS (p. 9-8) used the following equations (from Turner, 1969; PEDCo, 1973):

$$= \frac{0.36Q}{\pi\mu\sigma_y\sigma_z}$$

and assumed "restrictive dispersion conditions" that gave $(\sigma_y\sigma_z) = 17{,}000\ m^2$ at 2 miles. This is equivalent to category D conditions as computed by the formulas in Table G-1. This gives an "effective χ/Q" of:

$$\frac{\chi}{Q} = \frac{0.36}{\pi\mu(17{,}000)} = \frac{0.36}{\pi(3.14)17{,}000} = (2.1 - 6)\frac{s}{m^3}$$

This value can be compared to the χ/Q values calculated in Appendix H for long term and one-hour frequency.

$$\left(\frac{\chi}{Q}\right)_{50\%} = (15 - 6)\frac{s}{m^3} \quad \text{(SW downwind sector)}$$

$$\left(\frac{\chi}{Q}\right)_{\text{long term}} = (5.9 - 6)\frac{s}{m^3} \quad \text{(NW downwind sector)}$$

Since the 24-hour χ/Q by this calculation is only about one-third of the annual χ/Q calculated in Appendix H, it is concluded that the two methods are inconsistent.

V Transportation χ/Q Factors

From page G-3:

$$\frac{\chi}{Q} = \frac{1}{\pi \sigma_y \sigma_z \mu} \, e\left[-\tfrac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right]$$

Assumptions: distance = 0.5 mi, F stability conditions,
μ = 1 m/s, h = 20m

$$\sigma_z = .016d(1 + .0003d)^{-1} = .016(805)[1 + (3-4)(8+2)]^{-1} = 10.39$$

$$\sigma_y = .04d(1 + .0001d)^{-\frac{1}{2}} = 32.2[1+ .08]^{-\frac{1}{2}} = 31.0$$

$$\frac{\chi}{Q} = \frac{1}{\pi(10.4)(31.0)(1)} \, e^{-\frac{1}{2}\left(\frac{20}{10.4}\right)} = 9.9\text{-}4(.158) = \underline{\underline{(1.6 - 4)}}\frac{s}{m^3}$$

As comparison:

Worst annual (χ/Q) at 0.5 mi = $(9.0 - 5)\frac{s}{m^3}$ (Table H-36).
9% of above.

$(\frac{\chi}{Q})_{5\%}$ for 1 hour, 0.5 mi = $(4.3 - 4)\frac{s}{m^3}$ (Table 21, App. H).
44% of above.

$(\frac{\chi}{Q})_{max}$ for 1 hour, 0.5 mi = $(1.1 - 3)\frac{s}{m^3}$. 111% of above.

The $(\frac{\chi}{Q})$ for ground level release is between 5% and max one-hour $\frac{\chi}{Q}$ so is conservative compared to values used elsewhere in the DEIS.

Since the assumption of an elevated release significantly reduces the calculated doses, there should be more of a justification for its use, especially in an accident where no fire is assumed. Furthermore, the assumption of stability F conditions is not especially conservative since less stable conditions can result in similar concentrations occurring closer than 0.5 miles from the source. This analysis should consider other distances and stability categories and also the case of a ground level or other elevated release height.

APPENDIX VI

Simple Model for Estimating Maximum Radionuclide Concentrations in the Pecos River, and Associated Ingestion Doses, due to the Release of Radioactivity from the WIPP Repository

by Moses A. Greenfield

A. The Square Wave Model

The model used in the Draft Environmental Impact Statement (DEIS, 1979) is based on a system of "three coupled partial differential equations describing the behavior of a liquid phase injected into an aquifer system" (INTERA, Sept. 1977). The equations are based on conservation of liquid mass, energy and the mass of a contaminant dissolved in the fluid. Additionally there are equations for each radioactive nuclide which conserve mass for each species and take account of radioactive decay.

Solutions for this complex system of coupled non-linear partial differential equations are obtained by developing finite-difference approximations in a three-diminsional grid (INTERA, Sept. 1977). There is also interest in developing relatively simpler solutions based on a one-dimensional approximation. The authors of the INTERA report checked the adequacy of their programmed trace component equations by comparison with known one-dimensional analytical solutions for radioactive chains (Lester et al, 1975). One-dimensional transport models have been developed by a number of writers and used to test parameter dependence and to compare different nuclide behaviors (Holly et al, 1971; Borg et al, 1976; Barr, 1979). It is helpful to give an elementary derivation to gain insight into the importance of the various terms that appear.

In Holly's treatment (loc. cit.) account is taken of nuclide transport due to water flow in a homogeneous, isotropic, porous aquifer with hydrodynamic dispersion, adsorption and desorption, and radioactive decay, with a one-dimensional spatial coordinate.

Figure I depicts a homogeneous, isotropic aquifer with groundwater flow rate $\bar{v}$, porosity Θ, single spatial coordinate x, and L and S representing the liquid and solid phases respectively. C_L, C_S represent the concentrations of a radionuclide in the two phases. An equation representing material balance in the element Δx after a time passage, Δt, may be written as follows:

I. $\theta \cdot \Delta x \cdot \Delta C_L + (1 - \theta) \cdot \Delta x \cdot \Delta C_S \quad = \quad C_L \cdot \theta \cdot \bar{v} \cdot \Delta t$

(expresses change in quantity of nuclide after a time passage, Δt, in volume element Δx.)

(flow in)

$$-\left(C_L + \frac{\partial C_L}{\partial x} \Delta x\right) \cdot \theta \cdot \bar{v} \cdot \Delta t$$

$\cdot$(flow out)

$$- \lambda \Delta t \; [\theta \cdot \Delta x \cdot C_L + (1 - \theta) \cdot \Delta x \; C_S]$$

(decay Fraction) (inventory in L,S.)

$$-D_L \frac{\partial C_L}{\partial x} \cdot \theta \Delta t + D_L \left[\frac{\partial C_L}{\partial x} + \frac{\partial^2 C_L}{\partial x^2} \Delta x\right] \theta \Delta t$$

(flow associated with dispersion at entrance, exit of element; D_L assumed independent of x.)

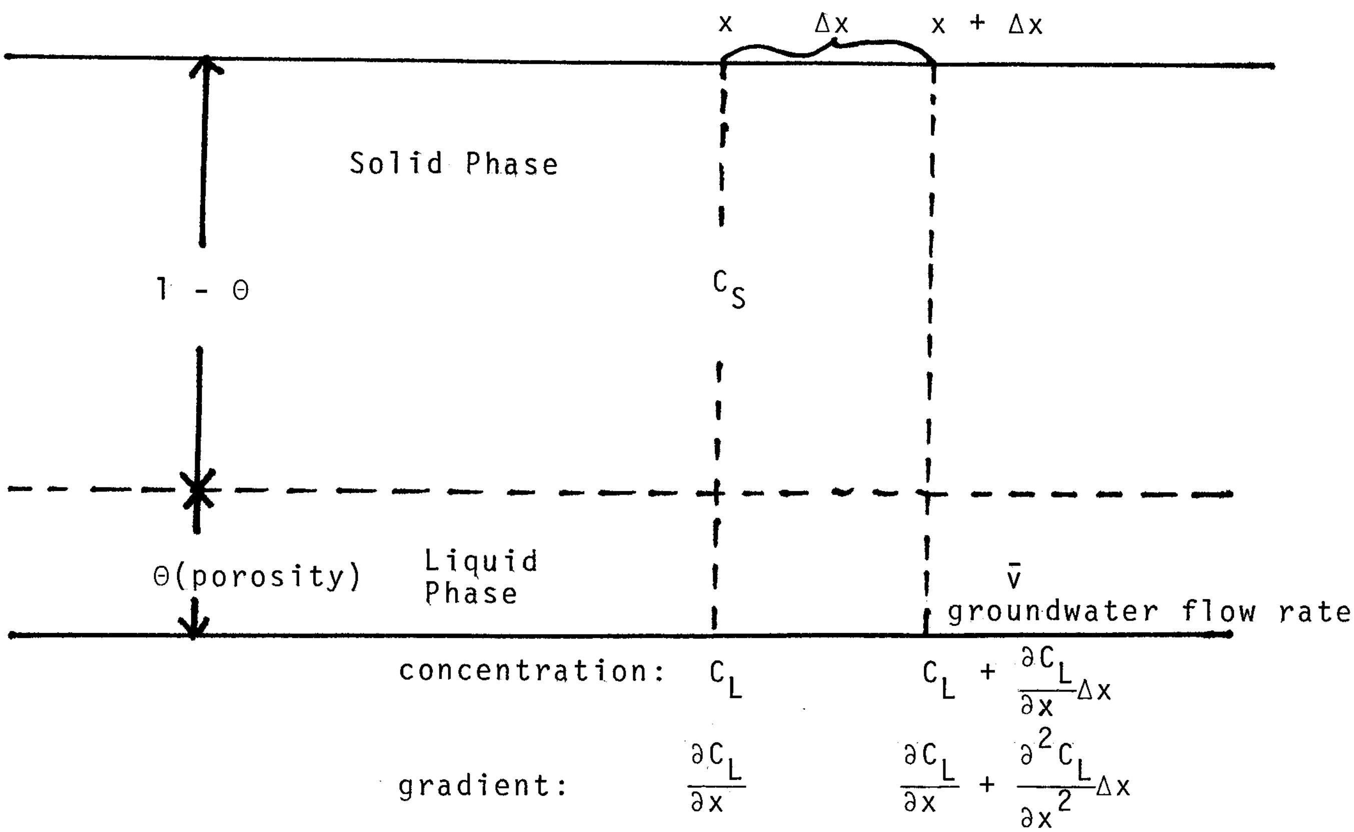
Equation I represents the material balance for any nuclide; λ is the decay constant and D_L is the coefficient of hydrodynamic dispersion.

Combining terms and dividing by $\theta \cdot \Delta x \cdot \Delta t$, gives:

II. $$\frac{\partial C_L}{\partial t} + \frac{1 - \theta}{\theta} \frac{\partial Cs}{\partial t} = - \bar{v} \frac{\partial C_L}{\partial x} + D_L \frac{\partial^2 C_L}{\partial x^2} - \lambda \left[C_L + \frac{1 - \theta}{\theta} Cs\right]$$

Adsorption may be introduced by assuming instantaneous equilibrium and reversibility between liquid and solid phases, with:

$$C_s = K'_d C_L \quad (K'_d \text{ dimensionless})$$



Volume Elements:

Liquid $\Theta\Delta x$

Solid $(1-\Theta)\Delta x$

C_L, C_S in $\frac{\mu Ci}{ml}$ or similar units

Figure 1

Also, let $K_d' = \rho_B K_d$, with ρ_B(g/ml) = bulk density of aquifer.

Then $\frac{Cs}{\rho_B} = K_d C_L$; ($\frac{Cs}{\rho_B}$ in μCi/g, K_d in ml/g).

K_d is called the distribution coefficient, and is a measure of the extent to which the solid adsorbs the nuclide. Values for K_d are listed in the DEIS, Volume II, Table K-3, page K-20, and were used as input parameters for the calculations made in the report.

Replace C_s in II by $\rho_B K_d C_L$. Then

III. $$\left(\frac{\partial C_L}{\partial t} + \lambda C_L\right)\left(1 + \frac{1-\Theta}{\Theta} \rho_B K_d\right) = -\bar{v}\frac{\partial C_L}{\partial x} + D_L \frac{\partial^2 C_L}{\partial x^2}.$$

IV. Let $B = 1 + \frac{1-\Theta}{\Theta}\rho_B K_d$; $K_L = e^{\lambda t} C_L$.

Then:

V. $$B\frac{\partial K_L}{\partial t} = -\bar{v}\frac{\partial K_L}{\partial x} + D_L \frac{\partial^2 K_L}{\partial x^2}$$

Let $t' = t/B$.

Then:

VI. $$\frac{\partial K_L}{\partial t'} = -\bar{v}\frac{\partial K_L}{\partial x} + D_L \frac{\partial^2 K_L}{\partial x^2}.$$

Values for $D_L = 185\ m^2/yr$ (P. Brannen, personal communication to M. Greenfield, 1979) and $\bar{v}$ = 4.45 m/yr (=0.04 ft/day)

(DEIS, II, K-14; also DEIS, I, 9-112) have been used in the report calculations.

Note that $\frac{D_L}{\bar{v}}$ = 41 m = 136 ft.

If the concentration gradient changes are small in ~40 meters, perhaps $D_L \frac{\partial^2 K_L}{\partial x^2}$ can be neglected in some approximation compared to $\bar{v}\frac{\partial K_L}{\partial x}$.

VII $$\frac{\partial K_L}{\partial t'} + \bar{v}\,\frac{\partial K_L}{\partial x} = 0$$

Any function of the form $f\,(x - \bar{v}t')$ or $f[x - (\frac{\bar{v}}{B})\,t]$ will satisfy this partial differential equation; it represents a traveling wave form, f, moving with the speed $\frac{x}{t} = \frac{\bar{v}}{B}$ from repository site towards the river. Thus if B>>1, then the consequences will be a retarded nuclide movement. The actual concentration C_L is of the form $e^{-\lambda t}f$. For a collection of nuclides, indexed by i, each will move with its own velocity, $\bar{v}/B_i$, and will be modified by its own decay factor, $e^{-\lambda_i t}$, and the appropriate inventory activity, A_i.

If one assumes a common dissolution time, T_d, then $\frac{A_i}{T_d}$ is the time rate of release from the repository, acting as a source term. Figure 2 depicts the "spectral" composition of nuclides traveling from the repository to the Pecos River.

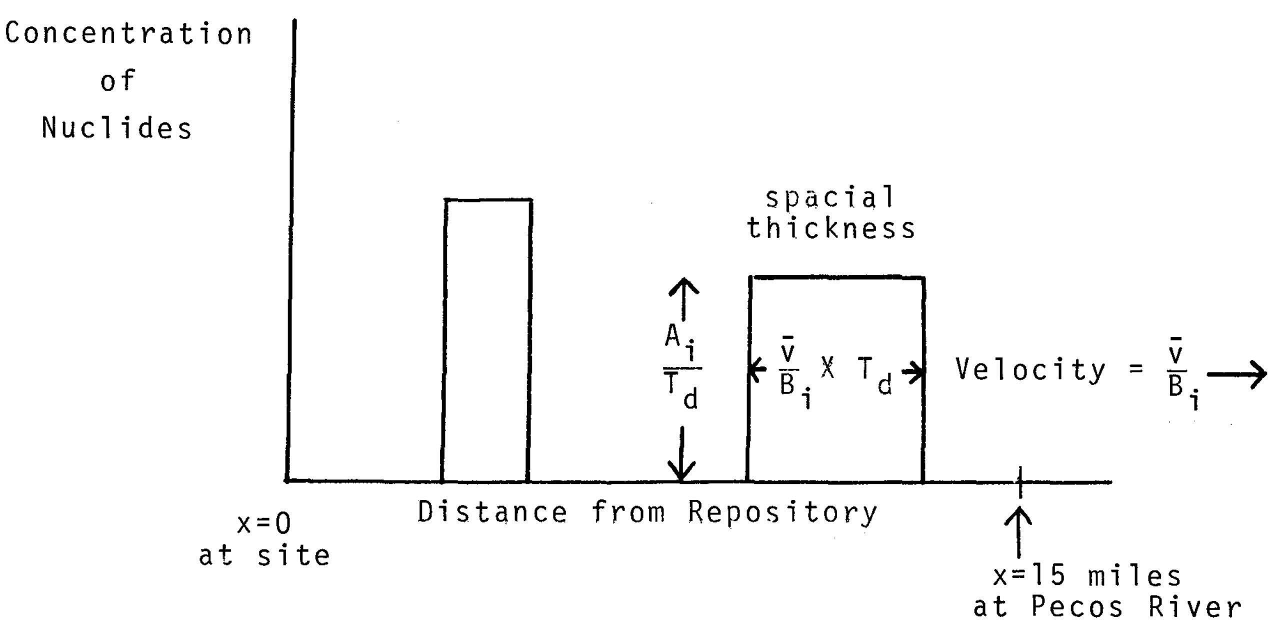


Figure 2.

Arrival time $(AT)_i$ to Pecos River is $\frac{15 \text{ miles}}{\bar{v}/B_i}$.

Release Rate from Repository (RRR) is $\frac{Ai}{T_d}$..

Discharge Rate at Pecos River (DR_{PR}) is $\frac{Ai}{T_d} e^{-\lambda_i (AT)_i}$

If the flow rate of the Pecos River is F, then the concentration of nuclides at the Pecos River, (C_{PR}), is $\frac{Ai}{T_d} \cdot \frac{1}{F} \cdot e^{-\lambda_i (AT)_i}$, at the arrival time $(AT)_i$.

The purpose of constructing this rather crude and simple "square wave" model is to use a zero order approximation approach for comparison with the calculations in the DEIS reports. Since the model is one-dimensional and dispersivity is neglected, one may reasonably expect concentrations and doses to be greater than those based on a more elaborate 3 dimensional approach. Clearly this comparison is intended only as a rough check on the validity of the methodology used in the DEIS calculations. If there is an approximate agreement, then attention may be turned to the assumed values for the input parameters, notably $\bar{v}$ the groundwater flow velocity and the K_d values.

B. Parent-Daughter Decay Chain

The preceeding analysis assumes that repository inventory accounts for all the nuclides that potentially may travel towards the Pecos River. Actually one must consider daughter decay products as additional sources. In some instances the production of the nuclide via decay may be much greater than the source from the repository. Another reason for the importance of decay products stems from the retardation values, B, for some nuclides. For example, the K_d value for Thorium is given as 2200 $(\frac{ml}{g})$ (DEIS, II, Table K-3, p. K-20). This leads to a B value of the order of 10^4, which precludes the travel of Thorium from the repository (Th-230 and Th-229). However, Th-229 is the daughter decay product of U-233 which is present in the repository inventory. The listed value of K_d for uranium is 1 (ml/g) (DEIS, II, Table K-3, p. K-20) which leads to a B value for uranium of about 19. Thus U-233 travels from the repository to the Pecos River in a time interval of approximately $(AT)_{U-233}$ = 19(15 mi./.04 ft./day) = 10^5 years. The Th-229 in effect gets a "piggy-back" ride and makes the journey in the same time interval. The following analysis develops the relations needed to compute release rates into the Pecos River and consequent concentrations.

Assume a single daughter, D, from a parent, P, with decay constants λ_D, λ_P. The initial parent activity (in the repository) is $A_{P,0} = \lambda_P N_{P,0}$ ($N_{P,0}$ is the number of parent atoms initially in the repository). Assume the daughter activity initially is $A_{D,0} = 0$.

In general.

I.
$$\frac{dN_D}{dt} = \lambda_P N_P - \lambda_D N_D = A_P - A_D$$

$$N_P = N_{P,0} e^{-\lambda_P t}$$

Thus, at t=0:

II.
$$\left.\frac{dN_D}{dt}\right|_0 = \lambda_P N_{P,0} = A_{P,0}$$

Thus, the initial rate of production of daughter activity is:

III.
$$\lambda_D \left.\frac{dN_D}{dt}\right|_0 = \left.\frac{dA_D}{dt}\right|_0 = \lambda_D A_{P,0}.$$

For a very long lived parent, III is a convenient approximation for computing the rate of production of a daughter.

The general solution for I is:

IV.
$$N_D = \frac{A_{P,0}}{\lambda_D - \lambda_P} \left[e^{-\lambda_P t} - e^{-\lambda_D t} \right]$$

The general form for $\frac{dA_D}{dt} = \lambda_D \frac{dN_D}{dt}$ can be stated as:

V. $$\frac{dA_D}{dt} = \lambda_D A_{P,0} \left[\frac{\lambda_D e^{-\lambda_D t} - \lambda_P e^{-\lambda_P t}}{\lambda_D - \lambda_P}\right]$$

One may also compute the ratio (A_D/A_P):

VI. $$\frac{A_D}{A_P} = \frac{\lambda_D}{\lambda_D - \lambda_P}\left[1 - e^{-(\lambda_D - \lambda_P)t}\right]$$

For sufficiently long times (with $\lambda_D t >> 1$), VI becomes:

VII. $$\frac{A_D}{A_P} = \frac{\lambda_D}{\lambda_D - \lambda_P}$$

An approximation for the daughter discharge rate into the Pecos River can be derived. Assume that $B_D >> B_P$. One may picture the "square wave" concentration of the parent making its journey to the Pecos River, dropping or leaving behind the daughter product distributed along the pathway from repository to the Pecos River. Compute the total activity of the daughter at the arrival time, AT_P, of the parent ($AT_P = \frac{15 \text{ miles}}{\bar{v}/Bp}$), distributed spatially along the 15 miles.

From VI:

VIII. $$A_D = \frac{\lambda_D}{\lambda_D - \lambda_P}\left[1 - e^{-(\lambda_D - \lambda_P)\ (AT)_P}\right] \times A_{P,0} e^{-\lambda_P (AT)_P}$$

If one makes the approximation that this activity of the daughter is uniformly spread over the 15 miles (actually there is a small gradient which is computed later in this report for an actual case), then the spatial distribution at the time AT_P is:

$$\frac{\Delta A_D}{\Delta x} = \left.\frac{A_D}{15}\right|_{t=AT_P}$$

Now one may compute the linear velocity of this distributed activity as:

$$V_D = \frac{\bar{v}}{B_D}$$

Combining the spatial distribution with the velocity yields the discharge rate into the Pecos River (DR_{PR}):

IX. $$DR_{PR} = \frac{A_D}{15} \times \frac{\bar{v}}{B_D}\Bigg|_{t=AT_p}$$

C. <u>Computation of Discharge Rates into the Pecos River and of Concentrations; Comparison of Peak Values (and Times) With Intera Calculations</u>

Scenario 4 mandates the full flow of Rustler aquifer through the repository, and assumes the complete dissolution of the spent fuel assemblies along with the bedded salt. The DEIS states that this proposed event is a bounding case, and for this reason it was chosen for calculation using the linear, square-wave model.

Column 1 of Table 1 lists the nuclide and Column 2 lists the inventory activity, A, in curies contained in the 1000 spent fuel assemblies in the repository. These numbers are based on the tabulated values of Ci/liter for each nuclide, 10 years after discharge from the reactor, (DEIS, I, Table 9-44, p. 9-104), and the computed volume per assembly (or canister) of 485 liters. This number is based on the stated dimensions of the assembly as 14 inch diameter by 16 foot length.

$$Vol_{can} = \frac{\pi D^2 H}{4} = \frac{\pi}{4}\left(\frac{14}{12}\right)^2 \quad 16 \text{ ft.}^3$$

$$= 17.1 \text{ ft.}^3 = 485 \text{ l.}$$

Thus the inventory listing for Tc-99 is obtained as:

$$1.4 \times 10^{-2} \text{ Ci/l} \times 485 \text{ l/can} \times 10^3 \text{ can}$$

$$= 6.8 \times 10^3 \text{ Ci.}$$

Th-232 is omitted in column I because the inventory is very small, a factor of 10^4 less than the next larger amount (Th-229).

Column 3 lists the release rate from the repository (RRR) in (μCi/sec), obtained by dividing the activity in column 2 by the dissolution time T_d = 4650 yrs. = 1.47×10^{11} sec.

This value for T_d is based on a dissolution rate of 21.4 ft.^{3}/day (salt plus radioactive material), and a repository volume of (930 x 930 x 42)ft.^{3} = 3.63×10^7 ft.^{3}. Thus $T_d = \frac{3.63 \times 10^7}{21.4}$ days = 1.70×10^6 days = 4,650 yrs. = 1.47×10^{22} sec.

(DEIS, I, p. 9-112; II, p. K-14.)

Column 4 lists the values for the retardation factor, $B = 1 + \frac{1-\theta}{\theta}\rho_B K_d$ (equation IV, section A). The porosity, Θ, for the Rustler aquifer is taken as 0.1 (DEIS, II, K-18): ρ_B is assumed to be 2g/m^3. Values for K_d are taken from DEIS, II, Table K-3, p. K-20.

Column 5 lists arrival times, AT, at the Pecos River computed as:

$$(AT)_i = B_i \text{ (5280) years.}$$

Since $(AT)_i = 15 \text{ mi.}/(\bar{v}/B_i)$ and $\bar{v}$ = .04 ft./day or 15 ft./yr.

Column 6 lists half- lives (DEIS, I, Table 9-44, 9-104).

Column 7 lists computed discharge rates into the Pecos River (DR_{PR}) in units of (μCi/sec), based on the release rate from repository with an appropriate decay factor: $(DR_{PR}) = (RRR)e^{-\lambda(AT)}$.

Column 8 lists the concentration in the Pecos River computed by dividing the discharge rate by the volume flow rate of the river, 515 l/s (DEIS I, 9-116).

Column 9 lists peak values for concentrations in the Pecos River as computed by INTERA and listed in BDM/TAC 79-156-TR, Appendix B, P. B-4 through B-27.

D. Discharge Rates into Pecos River, Concentrations for Daughter Products

Discharge rates for a number of nuclides were not computed (column 7) either because of relatively short half-lives, or rather large B (K_d) values or both (Ra-226, Th-229, Th-230, Np-237, Pu-239, Pu-240, Pu-242, Am-243). However a number of these are produced as daughter products of "traveling" parents and computations are then made on that basis. For example:

$$\text{U-234 (P)} \rightarrow\rightarrow\rightarrow \text{Th-230 (D);}$$
$$\text{Th-239 (P)} \rightarrow\rightarrow\rightarrow \text{Ra-226 (D);}$$

Thus both Ra-226 and Th-230 will piggy-back on the U-234.

In the case of U-233, the repository is a source. Np-237 in the repository doesn't travel because of a large B(K_d) value. However Np-237 does produce U-233, which constitutes a second (and larger) source for that nuclide. Additionally:

$$\text{U-233 (P)} \rightarrow\rightarrow\rightarrow \text{Th-229(D).}$$

In this case the Th-229 will piggy-back on a traveling parent, U-233.

D.1. Consider this series:

$$\begin{array}{r} \text{U-238 } (4.8 \times 10^{9} \text{ yrs.}) \\ \downarrow \\ \text{U-234 } (2.4 \times 10^{5} \text{ yrs.}) \\ \downarrow \\ \text{Th-230 } (0.77 \times 10^{5} \text{ yrs.}) \\ \downarrow \\ \text{Ra-226 } (1.6 \times 10^{3} \text{ yrs.}) \end{array}$$

For the first pair (U-238, U-234):

$$A_{U-238} \text{ (repository)} = 146\text{Ci}$$
$$\text{RRR U-234} = 5.9 \times 10^{-3} \ (\mu\text{Ci/sec})$$

(from Table 1, column 2 and 3). From section B, equation III,

$$\lambda_{U-234} \frac{dN_U}{dt} = \frac{0.693}{2.4 \times 10^5 \text{ yr.}} \times 146 \text{ Ci} \times 10^6 \text{ µCi/Ci}$$

$$\times \frac{\text{yr.}}{0.315 \times 10^8 \text{ sec.}}$$

$$= 1.3 \times 10^{-5} \text{ µCi/sec.}$$

This is <1/4% of RRR U-234, and is unimportant as a source term.

D.2. Now consider U-234→ Th-230; A_{U-234} = 870 Ci.

Compare the initial rate of growth of Th-230 activity with the repository source, RRR_{Th-230}. From section B, equation III,

$$\lambda_{Th-230} \frac{dN_{Th-230}}{dt} = \frac{0.693}{0.77 \times 10^5 \text{ yr.}} \times 8.7 \times 10^2 \text{ Ci}$$

$$\times 10^6 \text{ µCi/Ci} \times \frac{\text{yr.}}{0.315 \times 10^8 \text{ sec}}$$

$$= 2.5 \times 10^{-4} \text{ µCi/sec.}$$

This is about 5 x RRR_{Th-230} (= 4.6×10^{-5} µCi/sec).

Thus not only is the radioactive decay of U-234 a larger source term to produce Th-230 (compared to repository), but the uranium also acts as a carrier.

The method outlined in section B is now used to compute the discharge rate into the Pecos River.

First compute the total activity of the Th-230, distributed along the 15 miles. The time of interest is the arrival time, AT, for the parent of U-234, which is 1.0×10^5 yrs. (Table 1, column 5). At t = 10^5 yrs., the activity of U-234 is:

$$A_{U-234(t=o)}\ e^{-\lambda_{U-234}(AT)}$$

$$= 870 \quad Ci \times exp\left\{\frac{-0.693 \times 10^5}{2.4 \times 10^5}\right\}$$

$$= 870 \times 0.75$$

$$= 652\ Ci.$$

The activity of daughter Th-230 is given by (section B, equation VIII):

$$\frac{A_{Th-230}}{A_{U-234}} = \frac{0.693/0.77}{(0.693/0.77) - (0.693/2.4)} \times \left[1 - e^{-\left(\frac{0.693}{0.77} - \frac{0.693}{2.4}\right)}\right]$$

$$= 0.67.$$

Thus, $A_{Th-230} = 652 \times 0.67 = 437$ Ci at $5=10^5$ yrs.

The distribution of this activity spatially over the 15 miles is not uniform, but the gradient is not large. It can be estimated by computing the rate of formation in (Ci/Kyr) at the Pecos River ($t=10^5$ yrs.) vs. the rate of formation at the repository with an allowance for a time interval of 10^5 hrs. to elapse.

Using section B, equation III, at the repository, at t=0:

$$\frac{dA_{Th-230}}{dt} = \frac{0.693}{0.77 \times 10^2\ Kyr} (870Ci) = 7.8\ Ci/Kyr$$

Thus at the repository, at $t=10^5$ yrs., this would in effect be diminished by the decay factor:

$$exp\left\{\frac{-.693 \times 10^5}{0.77 \times 10^5}\right\} = 0.407$$

or $$\frac{dA_{Th-230}}{dt} = 7.8(.407) = 3.2\ Ci/Kyr$$ at the repository at $t=10^5$ yrs.

However at the Pecos River, at $t=10^5$ yrs, from section B, equation V:

$$\frac{dA_{Th-230}}{dt} = \left(7.8 \frac{Ci}{yr.}\right) \left[\frac{\frac{0.407}{0.77} - \frac{0.749}{2.4}}{\frac{1}{0.77} - \frac{1}{2.4}}\right] = 1.9 \text{ Ci/Kyr}$$

Thus an inventory of 437 Ci of Th-230 is distributed along the path from the repository to the Pecos River, with a spatial gradient such that the relative concentration (Ci/mile) at the repository is just $\frac{3.2}{1.9}$ = 1.7 x the value at the Pecos River. If the variation is assumed to be linear with distance it can be shown that the spatial gradient varies from 36.7 $\frac{Ci}{mile}$ at the repository to 21.7 Ci/mile at the Pecos River. (If uniformity had been assumed, the value would be $\frac{437 \text{ Ci}}{15 \text{ miles}} = 29.1 \frac{Ci}{mile}$). The linear speed of this activity is computed as:

$$V_{Th-230} = \frac{\bar{v}}{B_{Th-230}} = \frac{0.04 \text{ ft/day}}{0.396 \times 10^5} \left[\frac{1}{0.528 \times 10^4 \text{ ft./mile}}\right]$$

$$\times \left[\frac{1}{0.864 \times 10^5 \text{ sec/day}}\right]$$

$$= 0.221 \times 10^{-14} \text{ miles/sec.}$$

Combining the speed with spatial gradient one computes the discharge rate:

$$DR_{Th-230} = 21.7 \frac{Ci}{mile} \left(10^6 \frac{\mu Ci}{Ci}\right) 0.221 \times 10^{-14} \frac{miles}{sec}$$

$$= 4.80 \times 10^{-8} \frac{\mu Ci}{sec.}$$

Dividing by the flow rate of the Pecos River, $F = 515 \frac{\ell}{sec}$:

$$C_{PR,\ Th\text{-}230} = \frac{4.80 \times 10^{-8}\ \frac{\mu Ci}{sec}}{5.15 \times 10^{2}\ \frac{\ell}{sec}}$$

$$= 0.93 \times 10^{-10}\ \mu Ci/\ell.$$

The comparison value (Intera) listed in BDM-TAC 79-156-TR, Appendix B, B-12 is:

$$1.01 \times 10^{-10}\ \frac{\mu Ci}{\ell} \quad \text{at} \quad t = 10^5 \text{ years.}$$

D.3. The calculation for radium-226, the last daughter of interest in the U-238 chain can only be approximated with the crude model being used, in part because of its relatively short half-life. However an upper limit can be estimated.

Since Ra-226 is so short-lived, it may be assumed to be in secular equilibrium with its parent, Th-230. At or close to the time $t = 10^5$ years, the parent activity is $A_{Th\text{-}230} = 437$ Ci (see D.2.). In this case, use section B equation VII, to compute the daughter activity:

$$A_{Ra\text{-}226} = 437\ (1.02) = 446 \text{ Ci.}$$

The spatial distribution will be the same, virtually, as for the Th-230. Thus at or near the Pecos River the spatial gradient will be $21.7 \times 1.02 = 22.1\ \frac{Ci}{mile}$. If one assumes that the Th-230 distribution has advanced to or near the Pecos River, then the Ra-226 would move with greater speed, since its B value is much lower (~ 450) than that for Th-230 (39,600). The potential discharge rate into the Pecos River would be the product of the spatial gradient and the speed of advance. Thus:

$$DR_{Ra\text{-}226} = 22.1 \times 10^{+6}\ \frac{\mu Ci}{mile} \times \frac{0.04\ \frac{ft.}{day}}{450} \times \frac{1}{5.28 \times 10^3\ ft./mile}$$

$$\times \frac{1}{8.64 \times 10^4\ sec./day} = 4.31 \times 10^{-6}\ \mu Ci/sec.$$

This must be considered an upper limit, since it is clear that prior to the arrival of Th-230 to the Pecos River, the production of the Ra-226 from the leading edge of the Th-230 would "race" ahead of the thorium column, but would also undergo relatively rapid decay (radium has a short half-life in this context of 1600 yrs.). Thus there would actually be some build-up of Ra-226 at or near the Pecos River over some period of time.

The discharge rate computed above must be compared with the total production rate of Ra-226 from Th-230, to ascertain whether that would constitute a rate-limiting process. Using section B, equation III:

$$\frac{dA_{Ra\text{-}226}}{dt} = \frac{0.693}{1600 \text{ yrs.}} (437 \text{ Ci}) \left(10^{+6} \frac{\mu\text{Ci}}{\text{Ci}}\right) \frac{1}{0.315 \times 10^{8} \frac{\text{sec}}{\text{yr.}}}$$

$$= 0.60 \times 10^{-2} \ \mu\text{Ci/sec.}$$

Thus Th-230 is producing Ra-226 at a rate which is orders of magnitude greater than the rate at which Ra-226 is leaving Th-230 as a discharge into the Pecos River.

An upper limit value for the concentration in the Pecos River is computed by dividing the discharge rate by the river's flow rate, F:

$$C_{PR,\ Ra\text{-}226} = \frac{0.431 \times 10^{-5} (\mu\text{Ci/sec})}{0.515 \times 10^{3} \ (\text{l/s})}$$

$$= 0.84 \times 10^{-8} \ (\mu\text{Ci/l})$$

$$= 8.4 \times 10^{-9} \ (\mu\text{Ci/l}).$$

The listed concentrations in BDM/TAC 79-156-TR, Appendix B, B-13, show increasing values up to the last tabulation for $t=10^5$ yrs. of 7.5×10^{-10} (μCi/l).

D.4. Consider Np-237 $\rightarrow\rightarrow\rightarrow$ U-233 as a source term for the U-233. The repository produces a release rate for U-233 of 0.99×10^{-5} µCi/sec, and continues this for a dissolution time DT = 4650 yrs. For this same time interval one may compute the activity of U-233 produced by decay from the Np-237 inventory of 440 Ci.

From Section B, equation VI:

$$\frac{A_D}{A_P} = \frac{\lambda_D}{\lambda_D - \lambda_P} \quad [1 - e^{-(\lambda_D - \lambda_P)t}].$$

In this case t (=4650 yrs.) is sufficiently small that $(\lambda_D - \lambda_P)t \ll 1$. Thus one may rewrite the above as:

$$\frac{A_D}{A_P} = \lambda_D \cdot t,$$

(by expanding the exponential).

$$\text{Thus:} \quad A_{U-233} = A_{Np-237} \times \frac{0.693 \times 4560}{1.6 \times 10^5}$$

$$= 440 \text{ Ci} \times 0.020$$

$$= 8.86 \text{ Ci at } t = 4650 \text{ yrs.}$$

This is in effect to be added to the repository inventory of 1.46 Ci for U-233. Thus the correction to the previously computed concentration in the Pecos River of 1.2×10^{-8} µCi/l is:

$$1.2 \times 10^{-8} \times \frac{8.86 + 1.46}{1.46} = 8.5 \times 10^{-8} \left(\frac{\mu Ci}{l}\right).$$

See the entry in column 8, Table 1 for U-233. Compare with the BDM, Appendix B, B-25 value of 6.2×10^{-8} $\left(\frac{\mu Ci}{l}\right)$.

D.5. Consider Th-229 as the daughter of U-233. As the U-233 moves to the Pecos River, it produces Th-229 which is then distributed over the 15 miles from the repository to the river (because of the large B value for Thorium). To obtain the total activity of Th-229 produced by decay, it is first necessary to compute the total activity of U-233 produced in $t = 10^5$ years by decay from Np-237. From section B, equation VI:

$$\frac{A_{U-233}}{A_{Np-237}} = \frac{\lambda_{U-233}}{\lambda_{U-233} - \lambda_{Np-237}} \times \left[1 - e^{-(\lambda_{U-233} - \lambda_{Np-237})t}\right]$$

at $t = 10^5$ yrs., with appropriate values for the λ's:

$$\frac{A_{U-233}}{A_{Np-237}} = 0.356;$$

$$\text{at } t = 10^5 \text{ yrs., } A_{Np-237} = 440 \times \exp\left\{\frac{-0.693}{21}\right\}$$

$$= 426 \text{ Ci.}$$

Thus $A_{U-233} = 151$ Ci at $t = 10^5$ yrs. Since Th-229 is a short lived daughter, it will be in transient equilibrium with U-233 and have virtually the same activity of 151 Ci.

Thus the spatial distribution of the Th-229 is $\frac{151}{15} \frac{\text{Ci}}{\text{mile}} = 10.0 \frac{\text{Ci}}{\text{mile}}$

$$= 10 \times 10^6 \frac{\mu\text{Ci}}{\text{mile}} .$$

The linear speed of the Th-229 is $0.221 \times 10^{-14} \frac{\text{miles}}{\text{sec}}$ (see similar calculations for Th-230).

Therefore: $DR_{PR} = 10.0 \times 0.221 \times 10^{-14} \times 10^{+6}$

$$= 2.21 \times 10^{-8} \ \mu Ci/sec$$

$$\text{and } C_{PR,\ Th\text{-}229} = \frac{2.21 \times 10^{-8}}{0.515 \times 10^{+3}} \quad \frac{\mu Ci}{l}$$

$$= 4.3 \times 10^{-11} \ \frac{\mu Ci}{l} .$$

Compare with BDM, Appendix B, B-26 value of 1.5×10^{-11} (μCi/l).

E. <u>Comparison of I-129 and TC-99 with DEIS, I, for Discharge Rate and Maximum Concentrations</u>

The computed value for the discharge rate into the Pecos River for I-129 is $1.0 \times 10^{-4} \ \frac{\mu Ci}{sec} \times 0.864 \times 10^{5} \ \frac{sec}{day} = (\mu Ci/day)$.

The value given as a maximum in DEIS, I, Fig. 9-14(b), 9-113 is 12.2 (μCi/day).

One may picture the time sequence for the square-wave calculation and for the DEIS computation as follows:

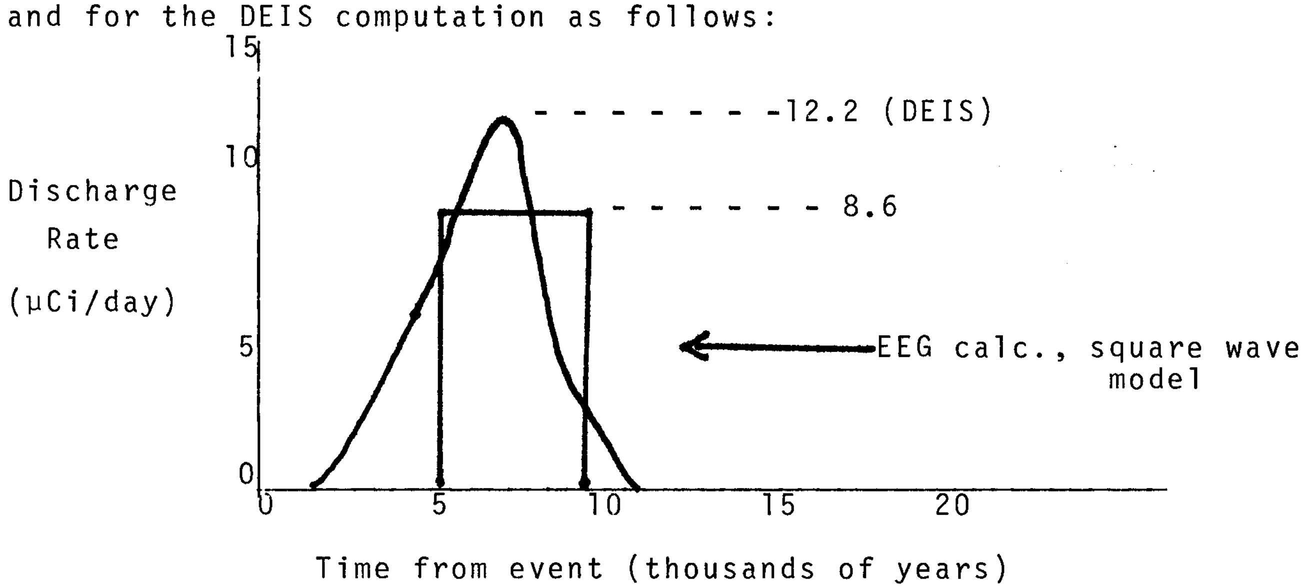


Figure 3

For the concentration at the Pecos River, one may picture the time sequence as follows:

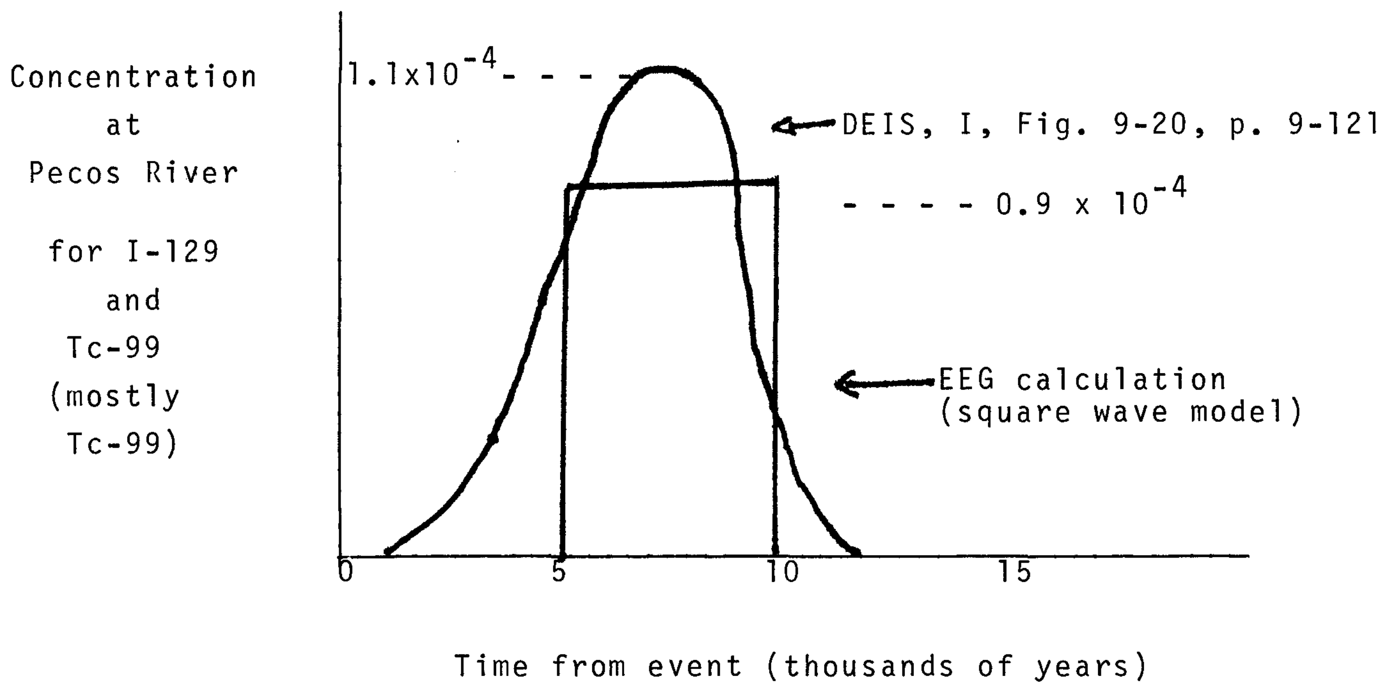


Figure 4

F. Use of "Square Wave" Model to Calculate Dosages to Whole Body and Organs

Scenario 4, Spent Fuel Assemblies at Malaga Bend

One may test the square wave model by combining Pecos River concentrations (µCi/l) with the assumed ingestion rate of 730 (l/yr.) to obtain µCi ingested. The mrem dose will be spread over a time interval depending on the t_{eff} (effective half-life) for the nuclide in the whole body (or organ of interest). Conventionally a 50 year commitment period is utilized. For nuclides like Tc-99 with rapid elimination ($t_{eff} \sim 5$ to 30 days), the dosage interval virtually coincides with the ingestion time. For radium and thorium with very slow elimination, the body (organ) content diminishes slowly, and the dosage is spread over the 50 years. Thus, the final computed dose is equivalent to mrem/yr. only for rapidly equilibrated nuclides (Tc-99). For others, the designation is the unavoidably awkward: mrem/per 50 yr. per µCi ingested. If the ingestion is assumed to re-occur a second year, a third year, etc., then the mrem/50 yr. will increase approximately linearly with the total ingested number of µCi's, for radium and thorium but not for Tc-99 (see Figure 5).

Note the temporal relation between uptake rate I (µCi/yr.) and body or organ content q (µCi) for various nuclides.

Note that the DEIS, pages 9-122 and 9-123, uses the notation "Dose rate (rem/yr.)" which is inappropriate. To be consistent with its own source material it should read: "Dose Commitment/ one year intake" or equivalent.

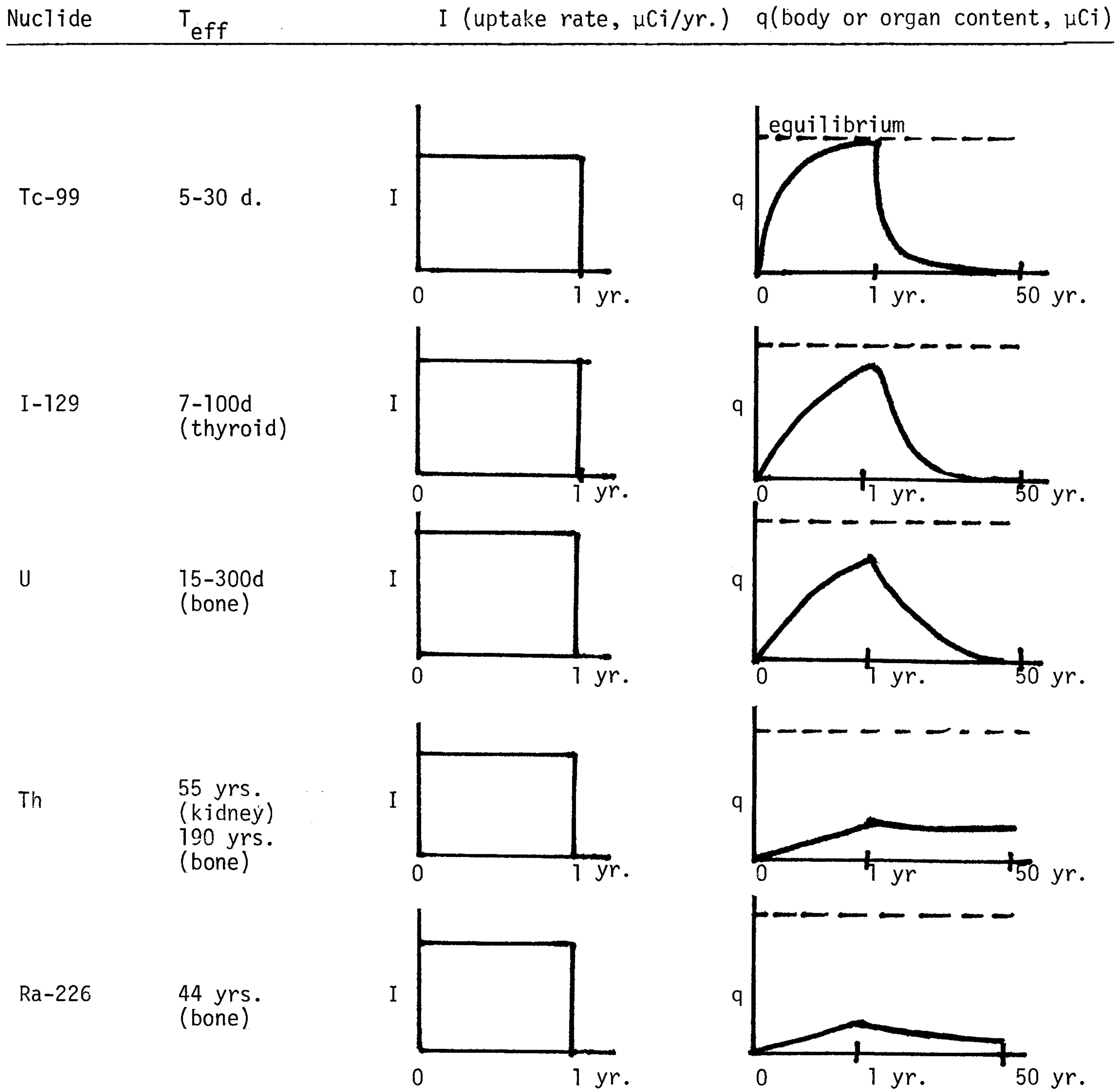

Nuclide
T_{eff}
I (uptake rate, μCi/yr.)
q(body or organ content, μCi)
Tc-99
5-30 d.
I
0
1 yr.
equilibrium
q
0
1 yr.
50 yr.
I-129
7-100d
(thyroid)
I
0
1 yr.
q
0
1 yr.
50 yr.
U
15-300d
(bone)
I
0
1 yr.
q
0
1 yr.
50 yr.
Th
55 yrs.
(kidney)
190 yrs.
(bone)
I
0
1 yr.
q
0
1 yr
50 yr.
Ra-226
44 yrs.
(bone)
I
0
1 yr.
q
0
1 yr.
50 yr.

Figure 5

Tables 2-8 include the following, column by column:

column 1: nuclide

column 2: concentration at the Pecos River in (μCi/l), as computed by square wave model

column 3: mrem/μCi ingested, as tabulated in NUREG 0172 for whole body (or organ); based on ICRP Reports 2 (1959), 6 (1962) and 10 (1967).

column 4: col. (2) x col. (3) x 730 ℓ/yr = mrem/one year intake

column 5: compare with BDM/TAC 79-156-TR, Appendix B listings, pages designated

bottom of page : compare with EIS-I; 9-122, 123, maximum dose, upper transmissivity assumption.

The tables give comparisons for 10 nuclides and for 8 organ systems, where appropriate (e.g. of the 10 nuclides only I-129 contributes to thyroid dose). The nature of the concordance between columns (4) and (5) follows that obtained previously between Pecos River concentrations, "square wave" model vs. computer listings in BDM, Appendix B print-outs. The ratios of "square wave" to computer listings for doses vary from a minimum of 1.2 to a maximum of 5.0 with a mean value of 4.0.

Comparisons (for internal consistency) between DEIS maximum values and BDM sums are generally good, with one exception: for Lower Large Intestine [LLI], the sum of BDM listings is $0.124 \frac{\text{mrem}}{\text{1 yr. intake}}$ vs. 0.158 for the DEIS maximum value, a difference of 24%.

G. Comments

The square wave model yields results which differ from INTERA - BDM listings (e.g. for maximum concentrations in Pecos River) by factors ranging from 1.1 to 11.2 (for Radium-226) with a mean value of 4.2 (see Table 1). Considering the crudity of the one-dimensional square wave model, this may be considered fair agreement. The same degree of variation is present in the comparisons of total body and organ doses (see Tables 2-8). Usually the square wave model leads to a higher estimate, as expected.

Perhaps the most useful outcome of the calculations made is that it permits one to put as a lower priority the question of the validity of the methodology employed in the DEIS calculations of nuclide concentrations and doses by ingestion. Instead one may consider the key parameters that lead to the final results. This would include the K_d values which are responsible for holding-back such nuclides as Pu, N_p, and Thorium. Also, one would include the basic driving parameter, $\bar{v}$, the assumed ground water flow velocity. A thorough review appears to be appropriate to permit an assessment to be made of the validity of the values actually employed in the DEIS. Additionally, it is probably useful to examine the scenarios used in the DEIS to consider whether indeed 'bounding cases' have been included, as stated in the report.

TABLE 1

MOVEMENT OF NUCLIDES FROM REPOSITORY TO PECOS RIVER (15 mi.)

Comparison of Peak Values with Intera (BDM** listings)

NUCLIDES	ACTIVITY Ci	RELEASE RATE FROM REPOSITORY RRR (µci/s)	RETARDATION FACTOR B	ARRIVAL TIME AT PECOS RIVER AT (YRS.)	HALF-LIFE (YRS.)	DISCHARGE RATE INTO PECOS RIVER DR_{pr} (µci/s)	CONCENTRATION IN PECOS RIVER C_{pr} (µci/l)	BDM PEAK VALUE (time, yrs)
Tc-99	6.8×10^{3}	4.6×10^{-2}	1	5.4×10^{3}	2.1×10^{5}	4.5×10^{-2}	0.87×10^{-4}	1.2×10^{-4} (7000)
I-129	15	1.02×10^{-4}	1	5.4×10^{3}	1.6×10^{7}	1.0×10^{-4}	1.9×10^{-7}	2.7×10^{-7} (7000)
Cs-135	1.5×10^{2}	1.02×10^{-3}	270	1.5×10^{6}	2.3×10^{-6}	0.65×10^{-3}	1.3×10^{-6}	
Ra-226	1.26	0.86×10^{-5}	450	2.4×10^{6}	1.6×10^{3}			
Ra-226*				1×10^{5}		4.3×10^{-6}	8.4×10^{-9}	0.75×10^{-9} (10^{5})
Th-229	0.06	0.4×10^{-6}	39,600	2.1×10^{8}	7.2×10^{3}			
Th-229*				1×10^{5}		2.2×10^{-8}	4.3×10^{-11}	1.5×10^{-11} (10^{5})
U-233	1.46	0.99×10^{-5}	19	1.0×10^{5}	1.6×10^{5}	0.64×10^{-5}	1.2×10^{-8}	
U-233*'			19	1×10^{5}			8.5×10^{-8}	6.2×10^{-8} (10^{5})
U-234	8.7×10^{2}	5.9×10^{-3}	19	1×10^{5}	2.4×10^{5}	4.4×10^{-3}	8.5×10^{-6}	1.8×10^{-6} (10^{5})
U-235	7.8	5.3×10^{-5}	19	1×10^{5}	7×10^{8}	5.3×10^{-5}	1.0×10^{-7}	0.24×10^{-7} (10^{5})
U-236	1.4×10^{2}	0.95×10^{-3}	19	1×10^{5}	2.3×10^{7}	0.95×10^{-3}	1.8×10^{-6}	0.47×10^{-6} (10^{5})
U-238	1.46×10^{2}	0.99×10^{-3}	19	1.0×10^{5}	4.8×10^{9}	0.99×10^{-3}	1.9×10^{-6}	0.37×10^{-6} (10^{5})
Th-230*			39,600	1.0×10^{5}	0.77×10^{5}	4.8×10^{-8}	0.93×10^{-10}	1.01×10^{-10} (10^{5})

*as a daughter product 'repository plus parent **BDM-TAC-79-156-TR, Appendix B, B-4 to B-27

TABLE 2

Whole Body - Adult - Drinking 730 l/yr

NUCLIDE	C_{pr} (µci/l)	NUREG 1072 m rem/50y per µci/yr		(1)(2)x730(l/yr) m rem/1 yr intake	BDM 79-156-TR App. B m rem/1 yr intake		
I-129	1.9×10^{-7}	9.21	p. 21	1.28×10^{-3}	2.5×10^{-3}	p. B-100	t=6,000 to 8,000 yrs.
Tc-99	0.87×10^{-4}	5.02×10^{-2}	21	3.19×10^{-3}	5.5×10^{-3}	B-108	
Cs-135	1.3×10^{-6}	7.99	22	7.58×10^{-3} 1.5×10^{6} yrs			
U-238	1.9×10^{-6}	4.54x10 +D*	23	6.30×10^{-2}	1.27×10^{-2}	B-124	t = 10^5 yrs.
U-234	8.5×10^{-6}	5.17x10	23	$32.\times10^{-2}$	7.0×10^{-2}	B-132	
Th-230	0.89×10^{-10}	5.70x10	23	3.7×10^{-6}	7.9×10^{-6}	B-140	
Ra-226	0.84×10^{-08}	2.2×10^{5}	23	1.35	0.29	B-148	
U-236	1.8×10^{-6}	4.96x10	23	6.52×10^{-2}	1.8×10^{-2}	B-172	
U-235	1.0×10^{-7}	4.86x10 +D	23	3.55×10^{-3}	0.98×10^{-3}	B-204	
U-233	8.5×10^{-8}	5.28x10	23	3.28×10^{-3}	2.65×10^{-3}	B-228	
Th-229	0.88×10^{-10}	3.91×10^{2}	23	2.51×10^{-5}	0.81×10^{-5}	B-236	
				COMPARE			
Sum				1.80 $\frac{\text{m rem}}{\text{1 yr int}}$	0.394 $\frac{\text{m rem}}{\text{1 yr int}}$		

Also compare EIS, 9-122(a): 0.399 $\frac{\text{m rem}}{\text{1 yr int}}$

Note: Ra-226 accounts for 75% of total!!

*Includes effect of daughters

TABLE 3

Bone - Adult - Drinking 730 l/yr.

NUCLIDE	C_{pr} (μci/l)	NUREG 0172 m rem/μci intake	p.	(1)(2)x730(l/yr) m rem/1 yr intake	BDM 79-156-TR App. B.	p.	
I-129	1.9×10^{-7}	3.27	21	4.54×10^{-4}	9.45×10^{-4}	B-98	t=6,000 to 8,000 yrs.
Tc-99	0.87×10^{-4}	1.25×10^{-1}	21	0.79×10^{-2}	1.38×10^{-2}	B-106	
U-233	8.5×10^{-8}	8.71×10^{2}	23	5.40×10^{-2}	4.16×10^{-1}	B-226	$t = 10^5$ yrs.
U-234	8.5×10^{-6}	8.36×10^{2}	23	5.19	1.13	B-130	
U-235	1.0×10^{-7}	8.01×10^{2}	23	5.85×10^{-2}	1.50×10^{-2}	B-202	
U-236	1.8×10^{-6}	8.01×10^{2}	23	1.05	0.289	B-170	
U-238	1.9×10^{-6}	7.67×10^{2}	23	1.06	0.214	B-122	
Th-229	0.88×10^{-10}	7.98×10^{3}	23	5.13×10^{-4}	1.66×10^{-4}	B-234	
Th-230	0.89×10^{-10}	2.06×10^{3}	23	1.83×10^{-4}	2.84×10^{-4}	B-138	
$Ra\text{-}226_D$	0.84×10^{-8}	3.02×10^{5}	23	1.85	0.403	B-146	
				COMPARE			
Sum				$9.26 \frac{\text{m rem}}{\text{1 yr int}}$	$2.09 \frac{\text{m rem}}{\text{1 yr int}}$		
DEIS-I, 9-122(d):					$2.09 \frac{\text{m rem}}{\text{1 yr int}}$		

U-234 accounts for 55% of total.

TABLE 4

Thyroid - Adult - Drinking 730 l/yr.

NUCLIDE	C_{pr} (μci/l)	NUREG 0172 m rem/μci intake	(1)(2)x730(l/yr) m rem/1 yr intake	BDM 79-156-TR App. B
I-129	1.9×10^{-7}	7.23×10^{3} p. 21	1.00	1.86 p. B-101

COMPARE

EIS-I, 9-122(b): 1.88 $\frac{\text{m rem}}{\text{yr}}$

t=6,000 to 8,000 yrs.

TABLE 5

Liver - Adult - Drinking 730 l/yr.

NUCLIDE	C_{pr} (μci/l)	NUREG 0172 m rem/μci intake	(1)(2)x730(l/yr) m rem/1 yr intake	BDM 79-156-TR App. B m rem / 1 yr intake	p.	
I-129	1.9×10^{-7}	2.81	3.90×10^{-4}	8.26×10^{-4}	B-99	t=6,000 to 8,000 yrs.
Tc-99	0.87×10^{-4}	1.86×10^{-1}	1.18×10^{-2}	2.05×10^{-2}	B-107	
Th-229	0.88×10^{-10}	1.19×10^{2}	0.76×10^{-5}	0.25×10^{-5}	B-235	t = 10^5 yrs.
Th-230	0.89×10^{-10}	1.17×10^{2}	0.76×10^{-5}	1.61×10^{-5}	B-139	
Ra-226	0.84×10^{-8}	5.74	3.52×10^{-5}	1.66×10^{-5}	B-147	
U-238*	1.9×10^{-6}			3.68×10^{-5}		
U-234*	8.5×10^{-6}			1.19×10^{-6}		
U-236*	1.8×10^{-6}			9.0×10^{-9}		
U-235*	1.0×10^{-7}			7.08×10^{-5}		
U-233*	1.2×10^{-8}			1.31×10^{-4}		
			COMPARE			
Sum				2.58×10^{-4}	at t=10^5 yrs.	
EIS-I, 9-123(a):				2.72×10^{-4}		

*No values for liver for U are listed in NUREG 0172.

TABLE 6

<u>Kidney</u> - Adult - Drinking 730 l/yr.

NUCLIDE	C_{pr} (μci/l)	NUREG 0172 m rem/μci intake	p.	(1)(2)x730(l/yr) m rem/ 1 yr intake	BDM 79-156-TR App. B m rem/1 yr intake	p.	
I-129	1.9×10^{-7}	6.04	21	8.4×10^{-4}	16.5×10^{-4}	B-102	t=6,000 to 8,000 yrs.
Tc-99	0.87×10^{-4}	2.34	21	1.49×10^{-1}	2.58×10^{-1}	B-110	
U-233	8.5×10^{-8}	2.03×10^{2}	23	1.26×10^{-2}	0.98×10^{-2}	B-230	t = 10^5 yrs.
U-234	8.5×10^{-6}	1.99×10^{2}	23	1.23	0.27	B-134	
U-235	1.0×10^{-7}	1.87×10^{2}	23	1.36×10^{-2}	0.36×10^{-2}	B-206	
U-236	1.8×10^{-6}	1.91×10^{2}	23	2.51×10^{-1}	0.69×10^{-1}	B-174	
U-238	1.9×10^{-6}	1.75×10^{2}	23	2.43×10^{-1}	0.49×10^{-1}	B-126	
Th-229	0.88×10^{-10}	5.75×10^{2}	23	3.69×10^{-5}	1.20×10^{-5}	B-238	
Th-230	0.89×10^{-10}	5.65×10^{2}	23	3.67×10^{-5}	7.78×10^{-5}	B-142	
Ra-226	0.84×10^{-8}	1.63×10^{2}	23	1.00×10^{-3}	0.23×10^{-3}	B-150	
				COMPARE			
<u>Sum</u>				1.75 ⟷	0.402		
EIS-I, 9-123(b):					0.408		

U-234 accounts for 67 to 70% of total.

TABLE 7

Lung - Adult - Drinking 730 l/yr

NUCLIDE	C_{pr} (μci/l)	NUREG 0172 m rem/μci intake	(1)(2)x730(l/yr) m rem/1 yr intake	BDM 79-156-TR App. B 1 yr intake	
I-129	1.9×10^{-7}	* p.	*	1.01×10^{-4}	p. B-103
Tc-99	0.87×10^{-4}	1.58×10^{-2} 21	1.00×10^{-3}	1.74×10^{-3}	B-111
			COMPARE		

t=6000 to 8000 yrs.

*No value listed for I-129/Lung in NUREG 0172.

EIS-I, 9-123(c) 2.45×10^{-4} at $t=10^5$ yrs.

NUCLIDE	C_{pr} (μci/l)	BDM 79-156-TR App. B 1 yr intake	
U-238	1.9×10^{-6}	3.68×10^{-5}	p. B-127
U-234	8.5×10^{-6}	1.19×10^{-6}	B-135
Th-230	0.89×10^{-10}	7.3×10^{-10}	B-143
Ra-226	0.84×10^{-8}	8.99×10^{-6}	B-151
U-235	1.0×10^{-7}	7.08×10^{-5}	B-207
U-233	8.5×10^{-8}	1.31×10^{-4}	B-231
Th-229	0.88×10^{-10}	3.11×10^{-8}	B-239
U-236	1.8×10^{-6}	9.0×10^{-9}	B-176
Sum		2.49×10^{-4}	

No values of m rem/μci intake for Lung are listed for U in NUREG 0172.

TABLE 8

Lower Large Intestine (LLI) - Adult - Drinking 730 l/yr.

NUCLIDE	C_{pr} (µci/l)	NUREG 0172 m rem/µci intake	(1)(2)x730(l/yr) m rem/l yr intake	BDM 79-156-TR m rem/l yr intake	p.	
I-129	1.9×10^{-7}	4.44×10^{-1}	6.16×10^{-5}	21.5×10^{-5}	B-104	t=6000 to 8000 yrs.
Tc-99	0.87×10^{-4}	6.08	3.86×10^{-1}	6.71×10^{-1}	B-112	t=6000 to 8000 yrs.
U-233	8.5×10^{-8}	6.27×10^{1}	3.89×10^{-3}	3.12×10^{-3}	B-232	t=10^5 yrs.
U-234	8.5×10^{-6}	6.14×10^{1}	3.81×10^{-1}	0.83×10^{-1}	B-136	t=10^5 yrs.
U-235	1.0×10^{-7}	7.81×10^{1}	5.70×10^{-3}	1.53×10^{-3}	B-208	t=10^5 yrs.
U-236	1.8×10^{-6}	5.76×10^{1}	7.57×10^{-2}	2.08×10^{-2}	B-175	t=10^5 yrs.
U-238	1.9×10^{-6}	5.50×10^{1}	7.63×10^{-2}	1.54×10^{-2}	B-128	t=10^5 yrs.
Th-229	0.88×10^{-10}	5.12×10^{2}	3.29×10^{-5}	1.06×10^{-5}	B-240	t=10^5 yrs.
Th-230	0.89×10^{-10}	6.02×10^{1}	3.91×10^{-6}	8.30×10^{-6}	B-144	t=10^5 yrs.
Ra-226	0.84×10^{-8}	3.32×10^{2}	2.04×10^{-3}	0.452×10^{-3}	B-152	t=10^5 yrs.
			COMPARE			
Sum			0.545	0.124		
EIS-I, 9-123(d):				0.158 $\frac{\text{m rem}}{\text{yr}}$		

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APPENDIX VII

OPERATIONAL AND LONG TERM RELEASE CALCULATIONS

Contents

APPENDIX VII

OPERATIONAL AND LONG TERM RELEASE CALCULATIONS

Item VII-A

Annual Dose Commitment at James Ranch from Routine Operation Releases

The Draft EIS uses the Releases to the Environment tabulated in Table 8-6 to calculate doses at James Ranch (3.0 miles south-southwest of the center of the site) in Table 9-18. The following calculation is based on Table 8-6 and the extrapolated $\bar{\chi}/Q$ value from Table H-36.

$$\text{Dose} = \left(\frac{\chi}{Q}\right) Q \text{ (inhalation per year) (Dose Commitment Factor)}$$

adjusting for units

$$\text{Dose} = \left(\frac{\chi}{Q}\,\frac{s}{m^3}\right)\left(\frac{m^3}{y}\text{ intake}\right)\frac{pCi}{Ci}\,\frac{1}{\frac{s}{y}}\left[Q\,\frac{Ci}{y}\left(\frac{mrem}{pCi\text{ intake}}\right)\right]$$

$$\text{Dose} = \frac{(6.2\text{-}7)(7.3\text{+}3)(1\text{+}12)}{(3.14\text{+}7)}\,\frac{pCi}{Ci}\,\left[(Q \times DCF)\,\frac{Ci\ mrem}{pCi\ y}\right]$$

$$\text{Dose} = (1.44\text{ +}2)\,(Q \times DCF)\,\frac{mrem}{y}$$

Dose commitment factors are from NUREG-0172, Table 8.

Values of Q x DCF are tabulated on the following page for significant radionuclides from Table 8-6.

TABLE VII-A

Organ Dose Commitments at James Ranch from Routine Releases

Nuclide	Q (Ci/y)	Bone DCF (mrem/pCi)	Bone Q(DCF) (Ci/pCi mrem/y)	Lungs DCF (mrem/pCi)	Lungs Q(DCF) (Ci/pCi mrem/y)	Whole Body DCF (Ci/pCi)	Whole Body Q(DCF) (Ci/pCi mrem/y)
^{3}H	0.45	-	-	1.58-7	7.1-8	1.58-7	7.1-8
^{85}Kr	7.8	-	-	2.41-9	1.9-8	-	-
^{238}Pu	2.3-5	2.74	6.3-5	.182	4.3-6	6.92-2	1.6-6
^{239}Pu	2.6-4	3.19	8.3-4	.172	4.5-5	7.75-2	2.0-5
^{240}Pu	6.4-5	3.18	2.0-4	.172	1.1-6	7.73-2	4.9-6
^{241}Pu	3.5-3	6.41-2	2.2-4	1.52-4	5.2-7	1.29-3	4.5-6
^{241}Am	4.3-6	1.01	4.3-6	6.06-2	2.6-7	6.71-2	2.9-7
^{244}Cm	9.5-7	5.90-1	5.6-7	6.06-2	5.8-8	3.51-2	3.4-8
Σ(DCF)Q			1.31-3		5.12-5		3.13-5
50 year dose commitment = (1.44+2)(QxDCF) =			0.19 mrem		7.4-3 mrem		4.5-3 mrem
=			1.9-4 rem		7.4-6 rem		4.5-6 rem
DEIS values (Table 9-18) =			1.5-4 rem		7.1-6 rem		3.8-6 rem
Ratio $\frac{\text{DEIS value}}{\text{EEG value}}$			0.8		0.96		0.84

Item VII- B

Operation Accidents - CH-TRU Waste
Scenario C-7

The activity released and the resulting doses due to operational accident Scenario C-7, Surface Fire (1 hr.), have been evaluated.

The basic assumptions or model of the scenario are given on page 9-50, DEIS and summarized below.

Assumptions

1) It takes one hour to put out the fire.

2) 25% of a typical drum is combustible.

3) 1% of the activity in the combustible waste is released in respirable form per hour.

4) One drum burns, the two adjacent drums burst exposing contents which do not burn, only 10% of spilled contents is powder.

5) A total of 0.0014% of each of the two adjacent drums is respirable and released.

6) The double HEPA filter bank has a decontamination factor of 10^6.

Based on the CH-TRU inventory given on page E-2, DEIS, the following analysis may be made:

Analysis

Table VII-B

Amount of Radioactivity Released in C-7 Surface Fire

Isotope	Ci/drum (Pg. E-2)	Respirable Release Fraction [0.25% +	Adjacent 2 Drums 2(0.0014%)] x	Decon Factor 10^{-6} =	Ci Released
Pu-238	4.1×10^{-2}	1×10^{-4}	1.148×10^{-6}	10^{-6}	1×10^{-10}
Pu-239	4.8×10^{-1}	1.2×10^{-3}	1.34×10^{-5}	10^{-6}	1.2×10^{-9}
Pu-240	1.2×10^{-1}	3.0×10^{-4}	3.36×10^{-6}	10^{-6}	3.0×10^{-10}
Pu-241	2.9	7.25×10^{-3}	8.12×10^{-5}	10^{-6}	7.3×10^{-9}
Am-241	7.8×10^{-3}	1.95×10^{-5}	2.18×10^{-7}	10^{-6}	1.97×10^{-11}
Total	3.5				8.9×10^{-9}
			DEIS (Table 9-23, Page 9-51)		8.8×10^{-9}

The curies released due to the accident scenario C-7, Surface Fire calculated above is the product of the curies per drum times the sum of the fraction released from the burned drum and two adjacent damaged drums times the decontamination factor.

The curies released to the environment are then dispersed and diluted by using AIRDOS-II in the DOE analysis (page 9-54, DEIS). In order to evaluate the doses due to the releases given in Table 9-25, page 9-56, DEIS, an independent calculation was made using the same procedure as in Item VII-1, with dose commitment factors from NUREG-0172. For a $(\chi/Q)_{50\%}$ of $(5.8\text{-}6)\frac{s}{m^3}$ the dose is:

$$\text{Dose: } \frac{(5.8\text{-}6)(0.83)(1+12)}{(3.6+3)} [Q(DCF)] = (1.34 + 3) [Q(DCF)]$$

Doses are calculated on Table VII - 3 for the James Ranch. The maximum dose at 0.5 miles is also calculated $(\frac{\chi}{Q})_{max}$ because the public could be at this location.

Table VII-C

Dose Received at James Ranch From
Radioactivity Releases in C-7 Surface Fire

Nuclide	Q Ci	Bone Dose		Lung		Whole Body	
		DCF	Q(DCF)	DCF	Q(DCF)	DCF	Q(DCF)
^{238}Pu	1.0-10	2.74	2.74-10	.182	1.82-11	6.92-2	6.92-12
^{239}Pu	1.2-9	3.19	3.83-9	.172	2.06-10	7.75-2	9.30-11
^{240}Pu	3.0-10	3.18	9.54-10	.172	5.16-11	7.73-2	2.32-11
^{241}Pu	7.3-9	0.064	4.68-10	1.52-4	1.11-12	1.29-3	9.42-12
^{241}Am	1.9-11	1.01	1.92-11	6.06-2	1.15-12	6.71-2	1.27-12
		$\Sigma Q(DCF)$ =	5.54-9		2.78-10		1.34-10
50 year dose commitment		=	7.4-6 mrem		3.7-7 mrem		1.8-7 mrem
		=	7.4-9 rem		3.7-10 rem		1.8-10 rem
DEIS value (Table 9-25)		=	5.5-9 rem		2.7-10 rem		1.3-10 rem
Ratio $\frac{\text{DEIS value}}{\text{EEG value}}$		=	0.74		0.73		0.72
Dose at 0.5 mile		=	1.4-6 rem		7.1-8 rem		3.4-8 rem

Item VII-C

Operational Accidents
Underground Container Failure (hoist drop - R15)

One of the potentially more serious operational accidents with WIPP is described in the DOE DEIS is the hoist drop accident involving a spent fuel canister. The assumptions made are:

1) 0.1% of contents crushed into particles 10 microns or less;

2) Duration of accident - 6 hours;

3) Multiplying 0.84% (6 x 0.14%/h) by the powder inventory will give the airborne and respirable release of all isotopes except H-3, Kr-85, I-129;

4) 30% of H-3, Kr-85, I-129 released during first hour.

5) Gases not retained by filters; and

6) Double HEPA filter bank gives a decontamination factor of 10^6.

Table VII-D, entitled "Hoist Drop Accident", details the calculations going from the spent fuel isotopic inventory to curies released. The results agree with those presented in the DEIS (Table 9-24) for the hoist-drop - spent fuel accident.

The dose that would be received by an individual at the James Ranch was then calculated using the assumptions in the DEIS. The equations used in this calculation are given below. The resulting doses are shown in Table 11 of the main report.

Intake and Dose Equivalent or Dose Commitment Equations for R-15 Accident

Intake

Io = intake in uCi

$$Io = Q(\frac{Ci}{s})\ \chi/Q\ (\frac{s}{m^3})\ B\ (\frac{20\ m^3}{d} \times \frac{6}{24}) \times 10^6(\frac{uCi}{Ci})\ fa$$

Where Q = quantity in curies released divided by the sec of release.

$X/Q = 0.58 \times 10^{-5}\ s/m^3$ given in Table 21, page 26 of Appendix H, Annex 1 DEIS.

B = air intake, breathing rate 20 m^3/d release of 6 hours = (6/24). 20 = $5m^3$

fa = fraction inhaled which reaches critical organ.

Dose Commitment

DE = 50 year dose commitment from short term intake.

$$DE = \frac{74\ Io\ T\ \Sigma E(RBE)n}{m}\ (1 - e^{\frac{-0.693\ t}{T}})$$

Where DE = rem (50 year)

$74 = (\frac{51}{0.693}\ \frac{rem.dis.g}{MeV.d.uCi})$

T = effective half life, days.

$\Sigma E(RBE)n$ = effective energy

t = time of exposure 50 years, (50 x 365 days)

m = mass of critical organ.

Table VII-D

Hoist Drop Accident R-15

Isotope	Table E-5 Ci/Canister	Respirable Dust 0.001 (Ci)	Respirable air-borne Dust 0.0084 (Ci)	gas fraction 0.3	10^{-6} decon*	Ci released
H-3	150	-	-	-		45
Kr-85	2600	-	-	-		780
Sr-90/Y-90	$3x10^{4}$	$3x10^{1}$	$2.52x10^{-1}$		-	$2.5x10^{-7}$
Ru-106/Rh-106	$2.3x10^{2}$	$2.3x10^{-1}$	$1.93x10^{-3}$		-	$1.9x10^{-9}$
I-129	$1.5x10^{-2}$	-	-	-	-	$4.5x10^{-3}$
Cs-134	$4.3x10^{3}$	$4.3x10^{0}$	$3.6x10^{-2}$		-	$3.6x10^{-8}$
Cs-137/Ba-137m	$4.0x10^{4}$	$4.0x10^{1}$	$3.36x10^{-1}$		-	$3.36x10^{-7}$
Pm-147	$3.6x10^{3}$	$3.6x10^{0}$	$3.02x10^{-2}$		-	$3.0x10^{-8}$
Eu-154	$2.4x10^{3}$	$2.4x10^{0}$	$2x10^{-2}$		-	$2.0x10^{-8}$
Np-237	$1.6x10^{-1}$	$1.6x10^{-4}$	$1.34x10^{-6}$		-	$1.3x10^{-12}$
Pu-238	$1.3x10^{3}$	$1.3x10^{0}$	$1.09x10^{-2}$		-	$1.1x10^{-8}$
Pu-239	$1.5x10^{2}$	$1.5x10^{-1}$	$1.26x10^{-3}$		-	$1.26x10^{-9}$
Pu-240	$2.2x10^{2}$	$2.2x10^{-1}$	$1.84x10^{-3}$		-	$1.8x10^{-9}$
Pu-241	$3.1x10^{4}$	$3.1x10^{1}$	$2.6x10^{-1}$		-	$2.6x10^{-7}$
Pu-242	$6.7x10^{-1}$	$6.7x10^{-4}$	$5.6x10^{-6}$		-	$5.6x10^{-12}$
Am-241	$6.7x10^{2}$	$6.7x10^{-1}$	$5.6x10^{-3}$		-	$5.6x10^{-9}$
Am-242m	4.0	$4.0x10^{-3}$	$3.36x10^{-5}$		-	$3.36x10^{-11}$
Am-243	9.1	$9.1x10^{-3}$	$7.6x10^{-5}$		-	$7.6x10^{-11}$
Cm-243	1.5	$1.5x10^{-3}$	$1.3x10^{-5}$		-	$1.3x10^{-11}$
Cm-244	$8.8x10^{2}$	$8.8x10^{-1}$	$7.39x10^{-3}$		-	$7.4x10^{-9}$

Compares to table 9-24 Spent Fuel Hoist drop.

*Particulate Decontamination (10^{-6})

Item VII-D

Scenario 5 - Indirect Pathways Calculation

The DEIS calculated the doses from inhalation and ingestion pathways to an individual living 500 meters downwind from a drilling mud pit that was contaminated with CH-TRU or spent fuel. A procedure is presented on page K-22, K-23 for this calculation. However, the following calculation will attempt to check this result by an alternate procedure, using χ/Q values from H-36.

For the CH-TRU waste case with a 10-inch drill hole, the following assumptions are used:

ρ, do, K, A: same as on K-23

μ : 2.25 m/s

χ/Q : $(2.3\text{-}4)\,s/m^3$ extrapolation from 800m to 500m assuming $\left(\frac{d_2}{d_1}\right)^2$ relationship

Ci/g sample: average concentration from Table E-1 times a sample size of 142ℓ (9.12 inch diameter hole drilling through 11 feet of drums).

Quantity of mud: 100 tons (p. 9-124).

$$\text{Curies/gram of Pu-239:} \quad \frac{\frac{0.48\,Ci}{208\,\ell}(142\,\ell)}{100\text{ tons}\quad \left(2000\,\frac{lb}{ton}\right)454\frac{g}{lb}}$$

$$\text{Ci Pu-239 in top cm of mud:} \quad 66.9\,m^2\ \left(10^4\ \frac{cm^2}{m^2}\right)(1\text{ cm depth})^2\ \frac{g}{cm^3}\left(3.61\text{-}9\frac{Ci}{g}\right)$$

$$= (4.83\text{-}3)\,Ci$$

$$\text{Source Term, Pu-239}\ \frac{Ci}{s} = (4.83\text{-}3)\,Ci\ \left(10^{-13}\right)\frac{1}{s}\ \left(\frac{2.25}{1.0}\right)^3 = (5.51\text{-}15)\frac{Ci}{s}$$

$$\text{Dose} = \frac{\chi}{Q}\,(Q)(\text{inhalation})\text{DCF}$$

$$= (2.3\text{-}4\,\frac{s}{m^3})(5.51\text{-}3\,\frac{pCi}{s})(7.3\text{+}3\,\frac{m^3}{y})(3.19\,\frac{mrem}{pCi}) = (2.95\text{-}2)\,\frac{mrem}{y}$$

$$\underline{(3.05\text{-}5)}\,\frac{rem}{y}$$

Dose from other actinides, besides Pu-239, are tabulated below:

Table VII-E

Indirect Pathways Doses - Scenario 5

Nuclide	Ci/drum t = 0	Ci/drum t = +100y	Ci/142ℓ	(DCF)	Q(DCF)
Pu-238	.041	.018	.012	2.74	3.29-2
Pu-240	.12	.12	.082	3.18	2.61-1
Pu-241	2.9	.015	.010	.064	6.4-4
Am-241	.0078	.078	.053	1.01	5.3-2
				ΣQ(DCF) =	3.47-1
Pu-239	.48	.48	.327	3.19	1.04

Total Bone Dose: $(2.95\text{-}5)\frac{rem}{y}\left(\frac{1.39}{1.04}\right) = (3.93\text{-}5) = (3.9\text{-}5)\frac{rem}{y}$ Bone Dose.

This value is close to the value of (3.6-5) rem/y used in Table 9-48 of the DEIS.

It is noted that the χ_1 factors listed on the bottom of page K-23 cannot be obtained from the equation and assumptions at the top of the page without choosing a value for μ, the mean wind speed. The value of 2.25 m/s used in the above calculation is reasonable based on the data in Appendix H (although perhaps lower than the site average) and gives agreement within 10-15%.

Item VII-E

Scenario 5 - Drill Back Accident
Occupational Dose Evaluation

The accident scenario described on pages 9-124 to 9-126 was evaluated to check the reasonableness of the calculations. Data, and source inventories from the DEIS are used and referenced.

Mineral Exploration Case - Spent Fuel

1) The volume of waste in a geological core with a 14 foot fuel rod is:

$$V = \frac{\pi}{4}\left(\frac{2.12}{12}\right)^2 ft^2 \; [14 \; ft] \; 28.3 \; \frac{\ell}{ft^3} = 9.84 \text{ liters}$$

which agrees with the 10 liters used in 9-124.

2) The drill recovers only a fraction of the fuel in an assembly. If the dimensions of an assembly are 8.5 inches square (NUREG-0116, p. 3-8) then

$$F = \frac{\frac{\pi}{4}(2.12)^2}{(8.5)^2} = 0.049 \text{ of contents in one assembly.}$$

3) The dose rate was established by using the inventory in Table E-5, page E-6 of the DEIS and calculating the amount present at 100 years after emplacement (i.e. t + 110 years after removal from reactor). Exposures at 1 meter per curie per hour were obtained from the Radiological Health Handbook, 1979 edition, page 130 or the relationship:

(1) R_{hr} at 1 foot $\simeq$ 6 Ci (mev/γ)(No. of γ/disintegration)

4) Bremsstrahlung was also calculated for Sr-90, Y-90 and Cs-137 and considered for other Beta emitters. The expression:

(2) $f = 3.5 \times 10^{-4}\ ZE$ where f = fraction energy to photons
Z = atomic number absorber
E = maximum Beta energy

gives the amount of γ energy from the beta decay. This value is used in expression (1) to calculate the dose rate at 1 meter. Uranium Oxide, with a Z = 82.2 was used for Z.

5) Ingrowth of 241_{Am} from decay of 241_{Pu} was also calculated using the expression:

$$(3)\ A_2 = \frac{\lambda_{2Ai}}{\lambda_2 - \lambda_1}(e^{-\lambda_1 t} - e^{-\lambda_2 t}) + A_2^{\circ}\ e^{-\lambda_2 t}$$

for 241_{Pu}, $A_1^{\circ} = 3.1 \times 10^4$ Ci; for 241_{Am}, $A_2^{\circ} = 6.7 \times 10^2$ Ci

at + 100 years:

$$A_2 = \frac{0.00151}{-.0510}(3.1 \times 10^4)[.005 - .860] + 6.7 \times 10^2(.86) =$$

$\underline{\underline{1350}}$ Ci of 241_{Am}

6) Most radionuclides in Table E-5 were eliminated by inspection, because of short half-lives and concentrations that appeared to make the contribution to the does rate negligible.

The above procedure gives an external radiation dose of about 71 rem to the maximum exposed individual with the assumption given in Scenario 5. This is about 20% below the value given in Table 9-47 as an approximation. The reason for being this far below is not known; there appear to be no other nuclides in the inventory that would make much difference.

It is noted that if the time were taken as 100 years after removal from the reactor (rather than 110 years as used here) then the total would be 90 rem.

Conclusion. Agreement on the maximum dose to a drilling crew member is sufficiently close so that the conclusions drawn about the seriousness of such an accident remain valid.

Table VII-F

External Dose Rate From Fuel Assembly Radionuclides

Nuclide	$Ci_{(R+10)}$	$T_{\frac{1}{2}}(y)$	$Ci_{(R+110y)}$	Exp. Factor R/h-Ci	Dose Rates - R/h Assembly	Sample
^{135}Cs	-	-	1.5 - 4	-	-	-
^{137}Cs	4.0 + 4	30.1	4.0 + 3	0.33	1320.	64.6
^{154}Eu	2.4 + 3	16	3.1 + 1	0.62	19.2	0.9
^{85}Kr	2.6 + 3	10.8	4.2 + 0	0.013	.05	-
^{241}Am	6.7 + 2	458	1.35 + 3	0.012	16.3	0.8
^{244}Cm	8.8 + 2	17.6	1.7 + 1	0.00001	-	-
^{234}U	3.9 - 1	2.5 + 5	3.9 - 1	0.010	.004	-
Bremsstrahlung Radiation						
^{90}Sr	3.0 + 4	28.1	2.55 + 3	0.0016	4.2	0.2
^{90}Y	3.0 + 4	(28.1)	2.55 + 3	0.034	85.8	4.2
^{137}Cs	4.0 + 4	30.1	4.0 + 3	0.0034	7.0	0.4
					Dose =	71.1 rem

Mineral Exploration Core - CH-TRU Waste

1) Check volumes used on 9-124 (8 liters in sample.
Assume drill through either 4 levels of drums or 3 of boxes.

$$V_{drum} = [\frac{\Pi}{4}(\frac{2.12}{12})^2 \text{ ft}^2 \text{ core}][2.75 \frac{\text{ft}}{\text{drum}} (4 \text{ drums})]28.3\frac{\ell}{\text{ft}^3} = 7.64\ell$$

$$V_{box} = 7.64\ell[\frac{3.8 \text{ ft of box (3 boxes)}}{11.0 \text{ ft drums}}] = 7.94\ell$$

So the use of an 8 liter sample is reasonable.

2) Since the average drum (p. E-2) has a higher concentration of plutonium than the average box, the drum case will be used. From p. E-6 the only external dose would come from the Am-241 that is present or which ingrows from Pu-241.

$$A_{241_{Am}} = \frac{0.00151}{-.0510} (Ci_{241_{Pu}})[.005 - .860] + (7.8 - 3)\ .86\ Ci_{241_{Am}}$$

$$Ci\ 241_{Pu} \text{ in waste} = \frac{2.9Ci}{208\ell} (8\ell) = 0.112$$

$$Ci\ 241_{Am} \text{ in waste} = \frac{0.0078}{209} (8) = 0.0003$$

at t+100 yrs $A_{241_{Am}} = .00284 + .00026 = \underline{.0030}$ Ci

Dose: $\frac{(3.0 - 3)}{(1.35 + 3)} (16.3) = (35.6 -6) = (0.036 - 3)$ Rem from 241_{Am}

(Table 9-47 uses 1.0 - 3).
Note (from SAND 78-1850 pp. 21-23) a drum could have ≤200 gm Pu or 25 times the average. If one of those were struck +3 average drums the dose would be:

$(3.6-5)(\frac{224}{32}) = (2.5 - 4)$ Rem from 241_{Am}.

An inspection of nuclides distribution on pages E-36 and E-37 indicates that the only non TRU gamma emitter of possible significance is ^{137}Cs. 6.08 Ci of ^{137}Cs are distributed in 467,323 cubic feet.

$$^{137}Cs \text{ per } 8\ell = \frac{6.08 \text{ Ci } (8\ell)}{(4.67 + 5 \text{ ft}^3)(2.83+1\frac{\ell}{\text{ft}^3})} = (3.68\text{-}6)\text{Ci } ^{137}Cs \text{ in sample}$$

at +100 years = (3.7-7)Ci in sample.

Dose = (3.7-7)0.33 = (1.2-7)Rem from ^{137}Cs

If concentration is $(\frac{224}{32})$ average; ^{137}Cs = $(8.6\text{-}7)\frac{s}{m^3}$

So ^{137}Cs dose is negligible compared to 241Am.

Conclusion. The calculated dose in Table 9-47 for CH waste appears to be accurate and perhaps slightly conservative.

Item VII-F

Scenarios 1-4 (Hydrologic Breach)

Scenarios 1-4 (section 9.5.1) are similar in that all involve the formation of a hydrologic connection between the repository and the Rustler aquifer, after the repository is sealed. In each case, the breach results in dissolution of the waste, passage of the waste into the Rustler aquifer, and passage through the Rustler into the Pecos River.

The three-dimensional model used in the DEIS analysis of nuclide transport in the Rustler was developed by Intera Environmental Consultants. EEG used a simple "square wave" model, described in Appendix VI, to gain a better understanding of the key features of the Intera model and to check some of the DEIS results. Appendix VI includes an application of the square wave model to scenario 4. In this section, the square wave model is outlined briefly and its application to scenarios 1-4 is discussed.

The model assumes that the waste dissolves at a constant rate and enters the Rustler at this rate. Each nuclide then moves toward the Pecos River at a rate equal to the Rustler velocity $\bar{v}$, divided by a retardation factor:

$$B = \frac{1 - \theta}{\theta} \rho K_d$$

where θ is the porosity of the Rustler aquifer, ρ is the bulk density of the Rustler aquifer and K_d is the distribution coefficient associated with adsorption of the given nuclide onto Rustler rocks. The values used for θ and ρ are $\theta = 0.1$ and $\rho = 2$ (g/ml).

Distribution coefficients used in the DEIS are listed in Table K-3, p. K-20. (The large uncertainties associated with these parameters are discussed briefly in Item VII-I of this Appendix.)

Thus each radionuclide has an arrival time (AT) at the Pecos River, 15 miles from the repository:

$$AT = 15 \text{ mi}/(\bar{v}/B).$$

(Earlier arrival times for nuclides produced by radioactive decay of other nuclides are discussed in Appendix VI.)

Each radionuclide has a release rate from the repository (RRR):

$$RRR = \frac{\text{repository inventory activity (Ci)}}{\text{Total dissolution time (sec)}}$$

where the dissolution time (DT) is given by:

$$DT = \frac{\text{repository volume}}{\text{dissolution rate}} .$$

The repository volumes from the CH and RH levels are:

CH volume = 9,000 x 1,200 x 16.5 ft (K-22)
RH volume = 930 x 930 x 42 ft (K-21).

Radionuclide concentrations in the Pecos River (C_{PR}) at the nuclide's arrival time (AT) are found from:

$$C_{PR} = \frac{RRR \times e^{-\lambda(AT)}}{515 \text{ liters/second}}$$

where λ is the nuclide's decay constant and 515 l/sec is the flow rate of the Pecos River (9-116).

Table VII-G summarizes the Rustler velocities and dissolution rates characteristic of the different scenarios. Table VII-H lists dissolution times and selected arrival times for scenarios 1 and 4.

Table VII-G

Rustler Velocities and Dissolution Rates

Scenario	Fluid Velocity $\bar{v}$ in Rustler (ft/yr)*	Dissolution Rates (ft^3/day)** CH	RH
1	17.5	.33	.84
2	15	.15	.39
3	15	.057	.012
4	15	81	21.4

*"Fluid velocity through the Rustler aquifer for the upper-transmissivity bound is roughly 0.04 ft./day" (p. 9-112); that is 15 ft./year. "In scenario 1, some fluid from the Bell Canyon aquifer is added to the Rustler aquifer; after this addition the fluid velocity in the Rustler aquifer increases slightly -- roughly by a factor of 1/6 (p.9-109). Thus the Rustler fluid velocity in scenario 1 becomes (15 ft/yr) x (7/6) = 17.5 ft/yr.

**In scenario 1, a borehole through the repository connects the Rustler (upper) and Delaware Mountain Group (lower) aquifers. Water flows from the lower to the upper aquifer, dissolving 54 ft^3/day of the salt and waste bordering the borehole (9-111). The fraction of dissolved material which is from the CH repository level is the ratio of the CH level height (16.5 ft.) to the borehole length (2,700 ft.); i.e. 54 x (16.5/2700) ft^3/day or 0.33 ft^3/day of material from the CH level is dissolved. Similarly, 54 x (42/2700) ft^3/day or .84 ft^3/day of material from the RH level is dissolved. For scenario 2, the DEIS states that "The waste-dissolution rate for Scenario 2 was calculated to be less than that for Scenario 1 by a factor of 2.17" (9-109). This was used to calculate the scenario 2 dissolution rates given in Table VII-G. However, these rates add up to 0.54 rather than the total 0.64 ft^3/day given in (9-112; 2). In scenario 3, the rates at which diffusion brings waste and salt into the Rustler are given as 0.057 ft^3/day for the CH repository level and .012 ft^3/day for the RH level (9-112). Scenario 4 is discussed in Appendix VI.

Table VII - H

Migration and Dissolution Times

Scenario	Arrival time (AT) at Pecos River, (yrs.) if: $K_d=0$	$K_d=1$	$K_d=10$	Dissolution time (yrs.) CH	RH
1	4.5×10^3	8.6×10^4	8.2×10^5	1.5×10^6	1.2×10^5
4	5.3×10^3	1.0×10^5	9.6×10^5	6.0×10^3	4.7×10^3

Item VII- G

Hydraulic Conductivity, Interstitial Velocity in the Rustler Aquifer

<u>Notation</u>: $\bar{v}$ = interstitial velocity [ft/day]
K = hydraulic conductivity [ft/day]
θ = porosity
Δh = change in head (difference between heights of potential lines) [ft]
$\Delta \ell$ = distance over which Δh is calculated. [ft]

<u>Formula for interstitial velocity</u>:

$$\bar{v} = \frac{K}{\theta} \cdot \frac{\Delta h}{\Delta \ell}$$

(see, for example, Mercer and Orr, 1977, p. 17).

<u>Calculation of $\bar{v}$, from DEIS information</u>:

Let K = 1 ft/day (Fig. K-7, vicinity of WIPP)
θ = 0.1 (Table K-2)

$$\left.\begin{aligned} \Delta h &= 3200\text{-}2900\,\text{ft} \\ &= 300\ \text{ft} \\ \Delta \ell &= 15\ \text{mi} \\ &= 79{,}200\,\text{ft.} \end{aligned}\right\} \quad \text{(Fig. K-3, Fig. K-5)}$$

$$\text{Then } \bar{v} = \frac{1}{0.1} \times \frac{300}{79{,}200}\ \text{ft/day} = 0.038\ \text{ft/day.}$$

Compare with the DEIS value of 0.04 ft/day (9-112).

This calculation suggests that the interstitial Rustler aquifer velocity of .04 ft/day, taken as an upper bound in the DEIS, is not conservative.

1) Figure K-7 gives hydraulic conductivity values in the Rustler which are lowest (1 ft/day) at the WIPP site and increase after that.

2) A 1977 Mercer and Orr report used in the DEIS transport modeling, gives the average hydraulic conductivity in the rustler as 16 ft/day and the average interstitial velocity as 0.5 ft/day.

3) Transmissivities reported in a later Mercer and Orr report (1979) range from 10^{-4} to 140 ft^2/day and translate to a hydraulic conductivity range of 5×10^{-6} to 7 ft/day. The highest conductivity value measured in a well near WIPP is 2 ft/day (for hydrologic hole H-3 in WIPP Zone II).

Item VII-I

Variability in K_d Values; Effect on Radiation Exposure

The low radiation doses calculated for the hydrologic breach scenarios result in part from the fact that the K_d value assumed for plutonium retards the rate at which plutonium travels in groundwater by a factor of 37,800.* Thus, by the time it reaches the Pecos River, the large initial inventory of plutonium (in spent fuel or transuranic waste) has decayed to an insignificant quantity. In fact, the only significant nuclides that are not retarded by adsorption are iodine and technetium, and the smallest retardation factor for other nuclides is B = 19 (when K_d = 1) for uranium.

However, much smaller K_d values than those used in the DEIS have been measured using rocks from the Rustler formation (ref. 1, 2). More importantly, laboratory measurements of K_d values probably do not reflect field conditions. For example, small amounts of plutonium or other elements with the capacity of the rocks to adsorb more plutonium; thus a "loading effect" can reduce K_d values. Chelating agents like EDTA can also reduce K_d values, as can temperature, pH and other physical and chemical factors. Finally the equilibrium model in which K_d's make sense may be inappropriate for flow in a fractured medium.

Table VII-I summarizes K_d information for different nuclides and Table VII-J indicates the maximum nuclide concentrations in the Pecos River in scenarios 1 and 4 when low K_d values (K_d = 0 or 1) are assumed.

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*The retardation factor B is: 1 + 18 K_d (see Appendix VI).

Note that essentially the same effect on nuclide speed can be obtained by lowering the K_d value by a given factor or raising the hydraulic conductivity or interstitial velocity by the same factor. (See Item VII-H for a discussion of variations in hydraulic conductivity and interstitial velocity.)

Table VII-I

K_d Information

Nuclide	K_d used in DEIS	K_d range in refs. 1, 2
Tc-99	0	0.15-6.7
I-129	0	1
Cs-135	15	1 to 6,540
Ra-226	25	-
Thorium	2200	-
Uranium	1	-0.9 to 6.7
Np-237	700	8-23
Plutonium	2100	17.6 to 5400

Table VII-J

Effect of K_d's on Plutonium Concentration in Pecos River, Scenarios 1 and 4, Spent Fuel

Nuclide	Inventory (Ci) at 1000 yrs.	Half-Life Years	Scenario	Release Rate[1] from Repository (μCi/sec)	Decay factor[2] at arrival time, if			Maximum concentration[3] in Pecos River (μCi/l), if		
					K_d=0	K_d=1	K_d=18	K_d=0	K_d=1	K_d=18
Pu-239	1.5×10^5	2.4×10^4	1	3.8×10^2	.88	.08	0	6.5×10^{-5}	1.2×10^{-6}	0
			4	9.7×10^{-1}	.86	.06	0	1.6×10^{-3}	1.1×10^{-4}	0
Pu-240	2.0×10^5	6.5×10^3	1	5.3×10^{-2}	.62	1.0×10^{-9}	0	6.2×10^{-5}	1.0×10^{-8}	0
			4	1.3	.57	2.3×10^{-5}	0	1.4×10^{-3}	5.8×10^{-8}	0

1 $\frac{\text{Inventory (Ci)} \times 10^6\ \mu\text{Ci/Ci}}{\text{Dissolution time (sec)}}$ (see Item VII-F, Table VII-H , for dissolution times)

2 $e^{-(\ln 2/\text{half-life}) \times (\text{arrival time})}$ (see Item VII-F, Table VII-H , for arrival times).

3 $\frac{\text{Release rate from repository } (\mu\text{Ci/sec}) \times \text{decay factor}}{515\ \text{l/sec flow of Pecos River}}$

References

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