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Division of Hazardous Substances Regulation
Bureau of Radiation

Supplement to the July 1987 Draft Environmental Impact Statement for Promulgation of 6NYCRR Part 382:

REGULATIONS FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FACILITIES

Modeling and Dose Assessment of Alternative Low-Level Radioactive Waste Disposal Methods in New York State

August, 1988

New York State/Department of Environmental Conservation

Supplement to the July 1987
Draft Environmental Impact Statement for
Promulgation of 6 NYCRR Part 382:
Regulations for Low-Level Radioactive Waste
Disposal Facilities

**MODELLING AND DOSE ASSESSMENT
OF ALTERNATIVE LOW-LEVEL RADIOACTIVE
WASTE DISPOSAL METHODS
IN NEW YORK STATE**

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New York State Department of Environmental Conservation
Division of Hazardous Substances Regulation
Bureau of Radiation
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1. ACTION: The New York State Department of Environmental Conservation proposes to supplement the July 1987 draft environmental impact statement for promulgation of Part 382, Subparts A, B, C, D, E and J of Title 6 of the Official Compilation of Codes, Rules and Regulations of the State of New York, entitled "Regulations for Low-Level Radioactive Waste Disposal Facilities". The supplement presents the results of additional environmental pathway analysis and dose assessments to those presented in the draft environmental impact statement.

2. AREA AFFECTED BY ACTION: New York State

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Supplement to the July 1987 Draft Environmental Impact Statement for Promulgation of 6 NYCRR Part 382: Regulations for Low-Level Radioactive Waste Disposal Facilities

Modelling and Dose Assessment of Alternative Low-Level Radioactive Waste Disposal Methods in New York State

ABSTRACT

This document supplements the subject EIS. It serves to document the environmental pathway and dose assessment analyses in the July 1987 Department of Environmental Conservation (DEC) report entitled "Draft Environmental Impact Statement for Promulgation of 6 NYCRR Part 382: Regulations for Low-Level Radioactive Waste Disposal Facilities (Certification of Proposed Sites and Disposal Methods)." In response to discussions on the July 1987 DEIS calculations, some changes have been made in certain hydrogeological parameters and in nuclide transfer parameters in the biosphere. The opportunity was taken to recalculate pathway dose-conversion factors, especially for carbon-14.

The DEC adopted 10CFR Part 61 Performance Objectives in 6 NYCRR Part 382. The results show that the performance objective for an effective dose equivalent of 25 millirem per year for the whole body can be met for all designs depending upon location, and that the performance objective for a dose equivalent of 75 millirem per year to the thyroid will be more difficult to meet. Repository design, site characteristics, and better identification of the iodine-129 source term will play significant roles in meeting that limit.

This report was prepared by Atomic Energy of Canada (AECL) and Acres International Corporation, under contract to and input/review by the DEC.

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1. Introduction

The Low-Level Radioactive Waste (LLRW) Management Act of 1986 defined the activities that New York State will undertake to develop a permanent disposal facility by 1993. As part of those activities, the New York State Department of Environmental Conservation has promulgated regulations (6 NYCRR Part 382) which establish minimum siting and technology criteria for the disposal facility. A Final Environmental Impact Statement (FEIS) entitled "Final Environmental Impact Statement for Promulgation of 6 NYCRR Part 382: Regulations for Low-Level Radioactive Waste Disposal Facilities (Certification of Proposed Sites and Disposal Methods)" was issued in December, 1987, which addresses this action.

In the FEIS, various practicable LLRW disposal alternatives are described. The alternatives are grouped into three general categories:

(a) Abovegroun engineered disposal - This concept (Figure 1) involves engineered structures placed on the ground surface to contain and isolate the wastes.

(b) Belowground, engineered disposal - This alternative (Figure 2) uses both engineered features and the natural site characteristics to contain and isolate the wastes within 100 feet (30 meters) of the ground surface.

(c) Underground, mined repository - This alternative (Figure 3) involves the use of an existing or new mined cavity at a depth greater than 100 feet (30 meters) that is specifically engineered to isolate and contain LLRW.

The same three alternatives were addressed in the July 1987 Draft Environmental Impact Statement (DEIS) entitled Draft Environmental Impact Statement for Promulgation of 6 NYCRR Part 382: Regulations for Low-Level Radioactive Disposal Facilities, and the draft regulations also promulgated in July 1987. Since there was little actual disposal experience with any of the alternatives, it was necessary to determine if the concepts were viable, and to examine the features of each that were to be regulated. These functions were achieved in part by assessments of the potential doses that might be incurred by individuals residing near hypothetical disposal facilities based on each concept, and located in the three physiographic provinces in New York State (Figure 8).

The assessments were done by a technique called modelling in which the major characteristics of the concepts and the site were expressed mathematically. The mathematical models were then converted into computer codes that allowed potential doses to individuals to be assessed at each of the generic sites. The method and the results of these calculations were summarized in the Draft Environmental Impact Statement (DEIS) entitled "Draft Environmental Impact Statement for Promulgation of 6 NYCRR Part 382: Regulations for Low-Level Radioactive Waste Disposal Facilities (Certification of Proposed Sites and Disposal Technologies)" which was issued in July, 1987.

The major conclusions from the DEIS were:

- of the three alternative technologies, the underground mined repository provided the most protection against future radiation exposure of nearby residents;
- for the near-surface vaults, the predicted peak annual effective dose equivalents exceeded the performance objective in all but one case;
- the three radionuclides contributing significant exposures were, in descending order, carbon-14, iodine-129, and technetium-99;
- belowground vaults performed slightly better than aboveground vaults on the same site for modelling of subsurface flow only.

Comments on the DEIS submitted by various organizations and individuals indicated that additional information on the assumptions, procedures and the results was desirable. In addition, some issue was taken with the details of the methods and the assumptions made in their use. This report updates and expands the assessment work in the DEIS with those comments in mind and, as well, provides the updated environmental pathway analyses and dose assessments.

The major conclusions from this present report are:

- the specific design of the LLRW disposal facility will have very significant effects on its ability to meet the performance objectives;
- New York State (NYS) should seek an accurate estimate of the radionuclide inventory to be disposed in the LLRW facility so that the source term for the site may be more accurately projected;
- NYS should carefully review LLRW disposal facility proposals to assure that performance objectives can be met; the review of AGVs should consider surface contamination and airborne pathways which were not modelled for this report;
- assumptions made in conceptual modelling are critical to environmental pathway analysis and dose assessment; since many of the assumptions in this work were very conservative, an actual facility can be expected to result in lower doses to the general population.

This report begins by presenting some background on modelling as an assessment technique and on the potential health effects of exposure to ionizing radiation. The models, and the corresponding computer codes, COSMOS and SYVAC, used in the assessments, are described primarily in an Appendix. Additional discussion is provided on the options and reasons for selecting particular assumptions, data, and scenarios. The numerical results of the assessments are presented both as effective doses to individuals and doses to specific organs of those individuals. The effective doses reported here again show that the best performance is offered by the underground mined repository. However, because of major reductions in the estimated doses from carbon-14 (compared to those reported in the DEIS), the near-surface technologies both appear capable of satisfying the performance objectives at a variety of sites after detailed site and design parameters are considered. The radionuclide iodine-129 remains the only major contributor to committed doses to individual organs.

This report continues with a discussion of the results and of additional factors which may be important. The final section reviews the differences in the codes, assumptions, and parameter values used in the DEIS and this report, and the consequent changes in the predicted performance of the alternative technologies in the three physiographic provinces.

2. Objectives and Limitations of Modelling

Modelling is a tool for converting a physical real-life system into a mathematical form so that it can be studied to determine its characteristics. If that real-life system is a proposed or existing radioactive waste disposal facility, then an assessment model will reflect processes occurring over an extended time period into the future and spread over a significant areal extent. The model will provide a means for examining the expected performance of the facility. Actual testing of the system by measurement and monitoring is, of course, not possible, because of the very slow rates of change. The model thus becomes the only method of predicting the performance and changes, but can only accomplish that end within the bounds of our understanding of the system and of the future conditions. However, an actual LLRW disposal facility must be constructed to applicable engineering codes and monitored throughout operation and closure in order to assure conformance to the performance objectives in 6 NYCRR Part 382, to check the modelling results, and to enable corrective action to be taken if necessary.

The application here of modelling to low-level radioactive waste disposal is to generate predictions of radiation doses. These doses might be received by members of a critical group of the general population, those receiving potentially maximum exposures, that could be exposed as a result of the presence of the disposal facility. In the current case, models are used to represent three disposal concepts, as hypothetical facilities, each located in three different hydrogeological settings which might exist in various parts of New York State. The models thus illustrate some of the influences of the differences between concepts and locations, and identify some of the radionuclides that could be more important in determining the potential radiation doses. However, because they are based on generic sites and designs that have specifications that are the least conservative yet potentially permitable under the proposed regulations, they do not include specific beneficial features that are likely to be present in a fully developed disposal facility at a carefully chosen site.

Once the development of an actual LLRW disposal facility is undertaken, additional factors are likely to be included in the models that will be used to evaluate the expected performance of specific designs in well-characterized settings. The evaluations will include sensitivity analyses to identify those design, siting and operational aspects most important in ensuring that the disposal facility will successfully achieve safe isolation of the LLRW over the extended period of potential hazard. The modelling will also provide a guide as to the selection of monitoring installations that can provide assurance that the intended performance is being attained.

The limitations in the ability of modelling to fulfill the above-mentioned objectives arise in each of the steps towards the development of a model. The steps in the development of a reliable model are:

- the understanding of the processes and factors important to the functioning of the real system;
- the definition of the physical model that can describe the characteristics of those processes and factors;
- the creation of a set of mathematical relationships that are equivalent in characteristics to the physical model;
- the programming of computer codes that can calculate the numerical results that represent predictions of future conditions in the modelled facility and its surroundings;
- the assembly of sets of data that describe the parameters and properties of the components of the real system;
- the verification of the modelling results; and
- the documentation and validation of the model through use, publication and peer review.

If the real system is too complex to be represented in detail by a single mathematical model, it may be divided into submodels that can be applied independently to produce concise inputs to a central framework or computer code that derives the overall results.

Once the whole system has been modelled by a computer code, checks should be done to confirm that the code can reflect the performance characteristics of the real system. The first step in obtaining this assurance is the verification that the computer code does accurately calculate values representing the mathematical relationships that were created as models of the physical system. The second, and much more difficult step, is a validation that the mathematical relationships and the verified equivalent computer code do in fact characterize the controlling processes and factors in the real system. An overall validation is often not possible because of the system complexities and the extended time periods involved. Validation might then be limited to demonstrations that key components of the model adequately reflect the apparent behavior of the corresponding portion of a real system over a limited time period or under magnified conditions that accelerate the processes of change.

Each of the steps in the development and application of the model and codes is likely to involve coping with uncertainties and unknowns. Thus, assumptions and approximations are introduced to allow evaluation of one or more complete scenarios, that is, continuous sets of processes and migration pathways connecting the radionuclide source to an eventual radiation dose to an individual. These assumptions are the bridges that allow the gaps in knowledge of the disposal system to be spanned and thus enable a prediction of future performance to be made.

Some of the assumptions are necessary to cover areas of future events that can be estimated but which are not knowable. These include the amount and nature of LLRW that will be generated during the next several decades, and the changes in land use and lifestyle near the disposal site during the next several centuries. Some assumptions are required to reduce complexities to a manageable level. For example, the hydrogeological inhomogeneities in the site can be so detailed to be beyond complete characterization, let alone modelling, and thus require approximation in order to attain a result. Some assumptions are needed where knowledge of natural processes is incomplete and cannot be overcome by studies in a reasonable time period with the resources available. An example is the behavior of natural stable iodine in the environment, which is not yet fully understood. Because of this, the modelling of the migration of radioiodine I-129 from the LLRW must also be somewhat uncertain.

The assumptions and approximations that are made will usually be selected to be as realistic as possible, but with a bias in the direction of greater safety. Sensitivity analyses, that indicate the degree to which a specified change in an assumption or parameter value affects the predicted result, can be a guide as to the potential level of uncertainty or error that may result from the assumption.

Uncertainties in parameter values may be handled in two ways. The most straightforward is to use a consistent set of single "best-estimate" values in a "deterministic" run of the code. The alternative is to use a "stochastic" approach in which a range is selected for the values of each parameter. The probable distribution of values within that range, e.g. normal, log-normal, or uniform is assumed. Then, a large number of runs of the code are made, for each of which single parameter values are chosen randomly from each range according to the assumed distribution. This procedure leads to a number of individual results, the magnitude and distribution of which give an opportunity to estimate the potential level of variability in the consequent doses.

In this present report, deterministic modelling has been applied to various LLRW disposal options in a generic, rather than specific, manner. Interpretation of the results presented should take into account the resultant limitations of this modelling approach. Firstly, the scenarios chosen to be evaluated were those expected, based primarily on experience, to be dominant in the potential for radiation exposure of members of the public. Secondly, the characteristics of the disposal facility, including those of the site, the disposal method, and the affected population, were selected as representing minimum levels of compliance with the disposal regulations. More designs are possible, with advantageous characteristics that would ensure lesser consequences, but their specification is most efficiently accomplished once specific facilities are to be evaluated.

The models used here were intended to represent the expected performance of several basic approaches to disposal. No estimates were made of the consequences of unanticipated conditions or low-probability events, such as premature severe deterioration of the concrete structures, particularly for the aboveground vault, or of seismic events much more severe than considered in the design.

Finally, the predicted individual doses are strongly influenced by the mean radionuclide concentrations assumed to be present in the disposed waste. These values were adopted from the U.S. Nuclear Regulatory Commission document NUREG-0782, entitled "Draft Environmental Impact Statement on 10 CFR Part 61 'Licensing Requirements for Land Disposal of Radioactive Waste', based on early surveys of a variety of waste streams. The actual concentrations in the waste disposed over the next thirty years may be significantly different from those assumed, because of changing factors related to the sources of the waste, economic influences, and changes in technology. If current trends continue, the waste quantities and their contained radioactivity will be less than the values assumed in these assessments.

In summary, modelling provides a tool for predicting the potential radiation exposures resulting from a LLRW disposal facility. If grounded on a basic understanding of the processes involved, modelling can bridge gaps in knowledge, but cannot avoid a corresponding level of uncertainty. More detailed information can lead to more accurate predictions. It cannot, unfortunately, overcome the uncertainties of unforeseen events or trends in future situations.

3. Assessment of Radiological Effects

The numerical output of the modelling for these assessment studies is the potential radiation dose to the most exposed members of the population that may reside near the disposal facilities. Although radiation dose is basically a measure of the amount of energy absorbed in the body, or in an organ or tissue of the body, per unit mass, the more important concern is the effect of that energy on health. The relationship between the energy absorbed and the effect on health is a complex one, and depends on such factors as the type of radiation (alpha, beta, gamma, X-ray), the tissues involved, and the dose rate, as well as the amount of dose.

Radiological protection can involve a broad range of conditions, but many of the concerns are not pertinent to the type of radiation exposures which might occur for the public as a result of a LLRW disposal facility. For example, large doses can cause early and acute effects which in the extreme can be lethal within days or weeks. However, with the possible exception of occupational accidents at the facility, radionuclide concentrations sufficient to produce such acute doses will never occur outside the disposal units. Rather, the studies reported here must be concerned with relatively low doses accumulated over long periods as a result of chronic exposure.

The potential health effects of such doses are restricted to the induction of cancers in the exposed person, or the induction of genetic defects in the descendants of the exposed person if the dose is received prior to reproduction. The conservative assumption commonly made is that the probability of such effects actually occurring is proportional to the dose received, even down to extremely low doses.

Also, only the effect on people is assessed because, in general, sensitivity to radiation increases with biological complexity. From this the International Commission on Radiological Protection (ICRP) concluded (ICRP, 1977) that if individual persons are adequately protected from radiation, then other living things are also likely to be sufficiently protected

Individuals may receive radiation doses from external or internal sources of radionuclides. Exposure to external sources might occur, for example, from standing on contaminated ground or swimming in contaminated water. At levels of contamination which could conceivably be experienced from a well-designed LLRW disposal facility, external doses from such sources should be expected to be very small relative to the average doses to the public as a whole from natural, medical, and lifestyle exposures. Average radiation doses received by the U.S. public are summarized in Table 1 (NCRP, 1987).

As shown by the modelling results presented later in this report, doses from radionuclides ingested as a result of potential groundwater pathways from the waste are expected to be a more important source of long-term exposure to residents near the disposal facility than are external sources. Ingested (or inhaled) radionuclides can result in a dose to the whole body and to specific organs or tissues, depending on the physiological behavior of the individual nuclides. The magnitude of the dose is expressed in several ways. The terms are defined (adapted from CSA, 1987) as follows:

Absorbed dose - the amount of energy absorbed in the body, or in an organ or tissue of the body, due to exposure to ionizing radiation, divided by the respective mass of the body, organ or tissue; the unit of absorbed dose is the rad and is equal to 0.01 joule per kilogram.

Dose equivalent (H) - a quantity (in units of rem) which is equivalent to the absorbed dose (in units of rad) multiplied by a quality factor to account for the different potential for injury of different types of radiation. In this report, the term "dose" means "dose equivalent" unless qualified otherwise.

Committed dose equivalent (H_{50}) - the dose equivalent that will be accumulated over 50 years following a single intake of radioactive material into the body. It is numerically equal to the annual dose equivalent at steady state resulting from a chronic intake of the same magnitude each year.

Weighted dose equivalent - means the dose equivalent to the body, or to an organ or tissue of the body, multiplied by the appropriate weighting factor (i.e., weighted dose equivalent = $w_T H_T$, using the symbols defined under "effective dose equivalent").

Effective dose equivalent (H_E) - the sum of the weighted dose equivalents received by the organs and tissues of the body, which may be expressed in mathematical form as follows:

$$H_E = \sum_T w_T H_T, \text{ where}$$

w_T - is a weighting factor (listed below) for organ or tissue T,
H_T - is the dose equivalent received by organ or tissue T, that includes the dose equivalent from external sources of radiation plus the committed dose equivalent from radioactive substances in the body.
Committed effective dose equivalent - the sum of the committed dose equivalents received by the organs and tissues of the body, weighted according to the weighting factors given below.

Weighting factor - the ratio of the risk of fatal cancer or inheritable injury arising from a dose equivalent received by an organ or tissue, to the total risk of such stochastic effects from the receipt of an equal dose equivalent when the whole body is irradiated uniformly.

The following weighting factor values are used for radiation protection and regulatory purposes (CSA,1987):

Organ or Tissue	Weighting Factor w_T
Gonads (testes and ovaries)	0.25
Breast*	0.15
Red bone marrow	0.12
Lungs	0.12
Bone surfaces	0.03
Thyroid gland	0.03
Other organs or tissues (0.06 for each of the five other organs or tissues receiving the highest dose equivalents)**	0.30
Whole body (or trunk and body)	1.00

* The w_T value is an average for males and females.
** The extremities, eye lenses, and sometimes the skin are excluded.

In the case of irradiation of the gastrointestinal tract, the stomach, small intestine, upper large intestine, and lower large intestine are considered as separate organs.

The performance objectives in Section 382.11 of the 6 NYCRR Part 382 regulations (which duplicates Section 61.41 of 10 CFR Part 61) are expressed as limits of annual dose of "an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public".

In the past, the term "whole-body dose" has been used in two ways. The first was the dose received from a uniform irradiation of the whole body, and the second was the average dose to the whole body from the nonuniform irradiation of the body from radionuclides incorporated in single organs or tissues. Unfortunately, these two definitions do not correspond to comparable risks for doses of the same numerical value. To overcome this inconsistency, the ICRP recommended (ICRP,1977) the use of "effective dose equivalents", as defined above, which provides a consistent method of expressing the total risk to a person (e.g. a member of the public) from both internal radionuclide sources and from external irradiation.

The results of the assessments have, therefore been presented as the peak annual dose rate to the thyroid and r. seven other organs and tissues, as well as the annual effective dose equivalents calculated using the weighting factors tabulated above. The worldwide trend by regulatory agencies is to express dose limits to members of the public in terms of this effective dose equivalent alone without reference to individual organ doses. The Environmental Protection Agency and the Nuclear Regulatory Commission have indicated that they intend to adopt this method in the near future. Such a change will assign less impact to radionuclides which concentrate in a single organ. For example, a thyroid annual dose of 75 mrem from iodine-129 would contribute only 2.25 mrem toward the effective dose equivalent annual limit of 25 mrem.

The assessments described in this report derive values for the annual doses from the committed dose equivalents to organs, such as the thyroid, lungs, lower large intestine, and red bone marrow, and the committed effective dose equivalent which is the weighted sum of these committed organ dose equivalents. The method involves the application of "dose conversion factors" (Johnson, J.R. and D.W. Dunford, 1983; and Johnson, J.R., 1982), which relate the annual dose equivalents at equilibrium to the chronic annual intake rates of radionuclides.

This procedure takes into account the buildup of radionuclides that have a long residence time in the body, and gives a conservative estimate of the annual doses.

4. Scenario Selection

A scenario is a set of pathway and source descriptions sufficient to specify complete histories of nuclides migrating from source to irradiation of humans. Changing one component of a pathway will define a different scenario, but changing the value of a parameter (except perhaps for very large changes) will not. A scenario description, therefore, includes: the various kinds of waste, significant site properties, vault geometries, engineered migration barriers, legislated controls, and the group of humans most at risk.

The scenarios modelled in this work, which exclude surface pathways, are generic in the sense that the results are meant only to help define or supplement regulations regarding the expected, post-closure behavior of three basic concepts of disposal facility in three different types of location, making nine possible combinations or scenarios. The conceptual disposal facilities examined consist of aboveground structures and belowground structures located above the water table, and deep bedrock mined cavities. The scenarios are illustrated schematically in Figures 4, 5 and 6, and the site plan for the near-surface vaults in Figure 7. The three locations represent sites typical of three regions of New York State shown in Figure 8: the Valley and Ridge (Province II), the Appalachian Plateau (Province IV), and the Interior Plains (Province V).

Each scenario is described by a set of parameters chosen to represent the more significant physical processes, especially those that can be controlled by choice of site and facility type, since these are the variable quantities of this study. Single values for each of these parameters are chosen to represent typically one of the facility type and region combinations. No allowance is made for the variations which may be possible from detailed field examinations within the regions or through detailed engineering design.

Adequacy of performance for each scenario can be determined by comparison with an accepted standard, but making a decision to accept or reject any scenario at this stage is premature. Data from a thorough investigation of a specific site can be very different from typical values. Also, based on such thorough data, many accident situations can be anticipated, and detailed engineering specifications can be made to compensate for these. For this reason, only the expected performance of the facility is considered; no catastrophic failures or intrusion situations are modelled. In particular, there is no modelling of any catastrophic failure of any vault structure that could lead to contamination of the ground surface.

To make dose calculations, based on the set of available data describing the scenario, presupposes that the relevant physical principles are identified and represented in the modelling in sufficient detail. The degree of model detail, however, depends upon the degree of detail in the available data. For the generic study being considered, detailed data are not available, thus the degree of modelling need only be basic. The remainder of this section, therefore, is to describe the basic level of model detail appropriate to the data.

The ideal radioactive waste disposal facility would contain the radionuclides at a location away from any human presence until the radioactivity has decayed to innocuous levels. Containment may be thought to consist of not only engineered structures but also of natural barriers, making both the site and the structure important and interdependent.

Because radionuclides may take thousands or even millions of years to decay to innocuous levels, radioactivity releases can be prepared for by directing the transport of radioactive contamination away from humans or by dispersing contamination widely enough so that radioactivity levels at any given point are innocuous. If this is still not adequate then control of the inventory is needed. One can regulate the types or amounts of radionuclides that are placed in the disposal facility so that they decay to insignificant levels while the facility remains intact.

In this work, the assessments are based on radioactivity inventories of the 24 radionuclides in the 36 waste streams assumed in the DEIS for 10 CFR Part 61, with one exception: the carbon-14 content of one waste stream (N-SOURCES) is omitted, since it greatly exceeds Class C limits and would not qualify for disposal in the types of facilities considered. The radionuclide inventory is listed in Table 2. Many of these radionuclides are long-lived, hence releases must be anticipated.

It is assumed for this study that the most probable pathway of radionuclide release, and the pathway of major significance, involves the entry of water into the disposal vault, dissolution of the waste form, and transport in groundwater of radionuclides out of the vault, into an aquifer, and into a well and/or river where the water is used by humans. In order to limit the amount of contamination released in this manner, two general criteria are: 1) the waste is encapsulated and packaged to prevent water from dissolving the waste form, and 2) contaminant transport is delayed so that it becomes a long, slow process.

The first criterion can be met in the above- and belowground cases by constructing the vaults above the water table* and by maintaining the vault interior at a low water content by restricting water from entering and by placing waste materials in containers, such as steel drums. If water does enter, making the vault free draining can minimize water contacting the waste. One of the features that can assist in achieving

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*As used in this supplement, the terms water table, saturated, and unsaturated are defined as follows:

Water Table - is the elevation of the groundwater surface, at equilibrium, in a hole drilled into the upper part of the saturated zone of the overburden or rock.

Saturated - means that all of the pores are filled with water.

Unsaturated - means that some pores are not full of water.

the second criterion is a layer of adsorbent buffer material located in the drainage path. Retardation by buffer can be an important contribution to containment of the shorter-lived radionuclides.

In the case of the mined cavity, the vault is saturated, but water flow rates can be restricted. Leach-resistant waste forms can contribute to meeting the first criterion. The second criterion is met by requiring that the waste vaults be at a sufficient depth that the water must pass through an appropriately large thickness of bedrock. Although the flow path is assumed saturated along its entire length, groundwater flow in the bedrock region is slow. With the long path lengths that can be involved, moreover, radionuclide residence times of many thousands of years are possible. Significant decay of radionuclides while in transit can thus occur.

No significant gaseous releases are considered. In the case of the above- and belowground vaults, the dry conditions would cause the biodegradation processes to be much slower and different chemically than if wet conditions prevailed. There would be no buildup of gas pressure because the vault is designed to be free draining into the ground. Since the gas generated is expected to be primarily carbon dioxide, interaction with soil and groundwater will effectively prevent its direct escape to the atmosphere. In the case of mine cavities, the vault is assumed flooded immediately after closure, resulting in a saturated environment under several atmospheres pressure. Any gas produced will be compressed and have increased solubility. If excess gas accumulates, it will tend to displace water from void spaces and could reduce flow around the waste.

Additionally, since failure situations beyond normal deficiencies are not modelled, releases to the atmosphere through waste suspension are not considered. Along with no significant gaseous release this means that no atmospheric dispersion and dilution processes are considered. However, natural deterioration processes are expected to result eventually in some loss of vault structural integrity. Although surface pathways have not been modelled, exposures by such routes could occur, particularly from aboveground vaults.

The only chemistry-related phenomenon that was considered appears in the radionuclide transport modelling in the use of distribution coefficients for sorption effects. The distribution coefficient defines the ratio, at equilibrium, between the amount of nuclide adsorbed on the solid and the concentration in the liquid system. The chemical compounds containing radionuclides in the waste form are assumed to be readily soluble in groundwater (although the leach-resistant matrix is not).

Modelling of chemical speciation phenomena is not performed because of the complexity involved and because of the differences possible depending on design-related and site-specific considerations. The materials of the waste form and matrix, vault structure and backfill could be chosen, for example, to enhance the containment properties of the vault.

No solubility limits are imposed. This is certainly a conservative approach, but not overly so, since for low-level wastes in bales or wastes in a leach-resistant form, the amounts of dissolved ions containing radionuclides can be below solubility limit concentrations.

Radionuclide transport via unsaturated flow taking place under the above- and belowground facilities is represented in a simple fashion using bulk parameters. At the ground surface, where water movement takes place continually, the infiltration rates from precipitation data (Climatic Atlas, 1954) were used directly, as average inflow of water. For nuclide migration through unsaturated layers in the vault and ground below, a different approach was taken. Here, the water in the partially filled pores was assumed to be static, and the migration process was a diffusive one, through the films of water on the pore surfaces. The usual diffusion/retardation equation was used, but the values of the parameters in it were adjusted to allow for the fact that the pores were not full of water.

In practice, unsaturated hydraulic conductivity and dispersion depend on hydraulic head or moisture content with hysteresis effects evident. These all depend on position and time, the soil type and stratigraphy, and on the amount, time and duration of precipitation. All of these quantities are highly site-specific and require much experimentation to obtain.

Once the contaminated groundwater has percolated through the unsaturated zone or welled up from the bedrock, it enters the faster-flowing aquifer, which is the saturated zone of the overburden. This provides a dilution mechanism. It is assumed that the aquifer is large enough to support a well to fill the needs of one self-sufficient household.

The well is situated at the boundary of the disposal facility, where this self-sufficient household is located. Any contamination that remains in the aquifer downstream of the well is assumed to enter surface water, such as a small river, where further dilution can occur.

The consideration of human activity in the context of assessment modelling of radioactive waste disposal has to be assumed. The approach taken is to assume a critical individual: living at the site boundary, growing his or her own food, raising his or her own stock, and spending all his or her time in the immediate vicinity. Such an individual, moreover, is assumed to live at this location and in this manner for all time. A diagram of the nuclide migration pathways which were modelled is given in Figure 9.

Although perhaps unreasonably conservative, this assumption is made for the reason that, should dose levels calculated for this situation be found acceptable, they would also be found acceptable for any more realistic human behavior. If on the other hand doses are too high, trying to account for the individual's lifestyle, such as going to the store for food and growing only a small garden (if at all), getting water through a municipal supply, spending weekends in another location, having self-imposed dietary restrictions for whatever reason, would bring down doses, but be impossible to assure for periods of decades, much less thousands of years. In practice, one would think about choosing a better site or improving the design to assure conformance with the performance objectives in 6 NYCRR Part 382.

5. Assumptions Made for Modelling the Scenarios

To pass from the mostly qualitative descriptions of scenarios in Section 4, to assessments by COSMOS and SYVAC (or any other assessment codes) some subsidiary assumptions are necessary to describe the scenarios in quantitative terms. Where practicable, consistent and generally conservative assumptions were made for all sites and technologies, so that comparison of results would give some general guidance on the relative merits of locations and disposal methods. The specific details of the proposed location, design, and operational procedures could significantly change the differences in dose estimated by the generic analyses. Specific sites could provide more delay and dilution of radionuclides escaping the disposal units. Of equal importance, the features of the disposal units, and the form and packaging of the wastes they contain, might contribute to less release over the period of concern.

This section describes the assumptions, and the resulting data are listed in Section 6.

a) Waste Forms and Inventory

For the waste, it was necessary to identify likely forms of waste processing and likely container sizes so that leaching could be described. Two forms were modelled: the A stream, assumed to be compacted; and the combined B and C streams, assumed to be solidified in cement. As a "generic" assumption, both waste forms were contained in 210-liter (55-gallon) mild-steel drums, of the size and quality specified in DOT-17H, to be stacked with a reasonable packing fraction. To avoid being site-specific about loading sequences and vault locations in the facility, it was conservatively assumed that the whole waste inventory was contained in one row of vaults and was loaded in a short time. No radioactive decay of the nuclide inventory was assumed to occur until facility closure had been achieved. However, the subroutine SEERA was also called by COSMOS to give estimates of the reductions in population doses that reasonable assumptions about loading and location patterns might imply (see Appendix A).

b) Container Failures

The environment was assumed to be similar in all vaults and mined cavities and the same container failure function (Figure 10) was used for all of the scenarios, representative of a vault that was wet but not always full of water. A specific calculation of vault environment would, of course, permit the choice of a corresponding container failure function.

c) Vault Geometry

For assessment calculations, a row of vaults was represented by one long vault of the same width and a height that would permit the stacking of a reasonable number of layers of containers. Backfill, eventually composed of porous sand, was supposed to fill the spaces between the containers and spaces between layers. Below the bottom layer of containers was a layer of buffer mixture consisting of 10 percent clay and 90 percent sand. The buffer was supposed to be carefully packed, to permit easy draining of the vault, but to be self-sealing so that cracking which would allow water to bypass the buffer would be extremely unlikely. The elevations of the vaults are discussed below in the section dealing with overburden.

d) Water Flow through Vault and Surrounding Overburden

The permeability of the backfill, buffer, and underlying drainage path was assumed to be sufficiently high that the chance of a 'bath tub' effect from a vault filling with water was assumed to be negligible. Hydraulic movement inside the vault and into the underlying ground, was assumed to be vertical. Although horizontal conductivity in the overburden could be quite high, the neighborhood of the vault is likely to be backfilled and will be more compacted. In addition, the hydraulic forces that could lead to horizontal flow will be very small. Hence, it was assumed that flow through the overburden to the aquifer would be vertical, and horizontal dispersion would be small in comparison.

e) Infiltration

For a belowground vault, percolating water would move through the backfill piled over the roof but would not be able to penetrate the vault until the roof started to fail. It was assumed that the volume of water available to pass through roof cracks in belowground vaults was the infiltration typical of the particular province. Prudent design would probably include additional infiltration barriers, but none was assumed for the assessment. The situation is quite different for the aboveground vault, with no backfill over the roof. The full amount of precipitation can fall on the roof, without being reduced by evaporation from the soil surface, but there can be direct evaporation and presumably the roof would be shaped to encourage runoff. It was assumed that the volume of infiltration water available to pass through roof cracks, in a particular province, was the same for aboveground and belowground vaults. Infiltration rates for the three provinces were taken from (Climatic Atlas, 1954). The same values were used in the DEIS.

f) Roof Failures

Concrete caps were assumed to maintain their structural integrity for hundreds of years. However, long before their structural failure, small areas would crack and begin to permit the entry of water. The failure rate depends on the environment to which the concrete is exposed. Aboveground structures are assumed to be maintained in good condition over the first hundred years after closure with subsequent progressive failure, perhaps as a result of freeze-thaw cycles. After the maintenance period, the roof remains leak-free for 100 years, and no infiltration enters the vault until 200 years after closure. The roof is designed to resist structural failure for 600 years, at which point 22 percent of the roof leaks. Henceforth the roof fails at an increasing rate until 100 percent of the roof leaks at 1,500 years.

Near-surface buried structures may be subject to chemical attack depending on the type of soil but conditions are generally regarded as benign as far as concrete is concerned. For the lime-rich overburden of Province II, chemical action is assumed to be very small. No leaks were assumed to occur until 500 years after closure. Soil erosion and funneling of infiltration toward joints occur to a maximum of 5 percent at 750 years. The failure rate was assumed to increase linearly to 40 percent at 1200 years. The failure rate then begins to level off, perhaps because of particulates carried by the infiltration, water filling in cracks, so that 50 percent of the roof has failed at 1,500 years and beyond.

Chemical action is the assumed failure mechanism in overburden consisting of acidic glacial till, occurring in both Provinces IV and V. The following treatment of this process is conservative, however, since the concrete itself has a buffering effect on soil acidity. Leakage is assumed to start at 500 years, increases non-linearly to 20 percent at 1,000 years, then linearly to about 60 percent at 1,300 years, and finally tapers off to 75 percent at about 1,500 years.

Water access to the mined repository was assumed to occur immediately after closure, and roof-failure modelling was not needed.

g) Nuclide Transfer in Aboveground and Belowground Vaults

The transfer of nuclides from the waste to water which can carry them through the food chain is perhaps the most critical process in near-surface disposal systems. These systems do not have the benefit of the very long transit times which can be achieved from a mined repository. In the period when some drums have failed, but before the vault leaks, there is assumed to be a sufficient water film on the waste that radionuclides slowly diffuse out into the water film and thence eventually to the groundwater. This slow process continues throughout all vaults in those sections w ch remain unaffected by leakage. Once water begins to leak into the vault, the percolation of water around the waste in the wetted areas moves the nuclides in the water films more quickly toward the bottom of the vault. A section through a below-ground vault is illustrated schematically in Figure 11. The degree of flushing is controlled by the process of advection* which depends on total flow and drainage rates. The contaminated water passes through the layer of buffer material (sand plus adsorbents) and out into the unsaturated soil beneath the vault. The buffer material retards the migration of most of the nuclides, but its interaction with the mobile nuclides is significantly less. The nuclides that emerge from the buffer are carried into the soil by the water flowing from the vaults. To the extent that each part of this sequence of processes is understood scientifically, it is described mathematically in the model, and the overall result is calculated. Information from tests that help characterize the behavior of the various nuclides in each of the processes is used in the calculation.

It follows that several processes will be taking place simultaneously in a vault. First, there will be diffusion-controlled leaching of the readily leachable wastes and, at a slower rate, from the leach-resistant wastes, as their containers rust. Second, as the roof begins to leak, parts of the vault will be subject to flow of water downward, with increased migration of nuclides: the remaining fractions will be much drier, with much lower migration rates, but as leakage increases, the 'dry' areas will become fewer.

Leaching moves nuclides out of an effective slab into a thin layer of water on its surface. The estimation of the slab thickness is

*The migration of radionuclides in the water phase is modelled as occurring under the influence of diffusional processes within the liquid and, in addition, if the water is flowing, by advection, i.e., by being carried within the moving mass of water. The latter is dominant in those parts of the vault receiving leakage through the roof.

described in Appendix A. The water layer was taken to be very thin, but its actual thickness is not important and cancels out in the normalizations. A value of 0.001 m was used. The average thickness of backfill, between drums and between layers, is obtained very simply as the ratio of (volume of backfill per drum)/(surface area of a drum). The effective area over which leached nuclides are transferred to the infiltrating water is obtained by following the water as it spreads uniformly over the top and bottom surfaces of the drums and flows between drums to pass from layer to layer. For example, suppose that there are N cylindrical drums in the vault, of height H, and radius R. If F is the packing fraction, the integration yields for the transfer area the value,

$$N[2\pi RH + F\pi R^2].$$

If there are n layers of drums, a similar calculation of the average water path over which transfer occurs gives the length,

$$n[H + (2/3)FR].$$

h) Nuclide Transfer in Mined Repositories

Mines are assumed to be infiltrated by water from the lower bedrock immediately after closure. Leaching takes place in the same fashion as in the vaults and nuclides are then moved by diffusion and advection through the infiltrating water, into the bedrock. The water velocities in the lower bedrock are considerably smaller than infiltrations near the surface and typical values were chosen for the three provinces. A buffer layer, 0.5 m thick, surrounds the stacks of drums, and was the same as for the vaults.

i) Vault and Mined Cavity Environment

Apart from the assumptions in g) and h) above, no attempt was made to predict the environment in vaults or mined cavities in more detail, and in particular, modelling of chemical processes was reserved for a site- and waste material-specific assessment. It was assumed that gaseous decomposition and corresponding gas escapes were not significant. As a consequence of this, the atmospheric dispersion models in the codes were not utilized.

j) Overburden and Aquifers

The average depths of overburden vary quite considerably from province to province, and within the overburden the unsaturated layers are quite different too. The aquifer velocities were taken to correspond to conductivities and hydraulic heads that lay well within the ranges quoted in the DEIS. The thickness of the aquifer layers were taken to vary in the same fashion as the overburden, with the requirement that they be at least sufficient to support a water flow that would yield the needs of the small farm described below. The effective width of the aquifer is the length of the row of vaults that lies across it, and an additional requirement is that it can supply the well without its drawdown exceeding that width. If this is satisfied, the total flow of water and nuclides from vault into aquifer is diluted by the whole aquifer flow. The drawdown is estimated by the methods of NUREG/CR 1759, Vol.3.

It was assumed that a distance of at least 2 m should be maintained between the bottom of the vault and the top of the aquifer. The vault should then be safe from rises in aquifer level, even for one-in-ten thousand, or less frequent, years of heavy precipitation.

The result of all the above restrictions is that 'belowground' vaults as modelled in Provinces II and IV can be buried below original surface level to only one half of their height, and in Province V to only one tenth. They can, of course, still be mounded over by backfill.

k) Bedrock

Bedrock was assumed to lie in two layers below the overburden (see Figure 6, for example), labelled upper and lower bedrock respectively in the tables of Section 6. Their dimensions were chosen to be representative of the three provinces.

The groundwater flow path from the mined chamber into the overburden at the surface was assumed to be directly upward as a result of an hydraulic gradient that was vertical. This is a very conservative assumption since in most candidate areas, the gradient would be expected to be primarily horizontal at the depths involved. This assumption leads to minimum calculated transit times from the waste to the dose receptor. The contaminated flow emerges from the upper layer of rock into the aquifer, and there forms a horizontal contaminant plume of the same width as produced by the near-surface vault array.

1) Diffusion Parameters

To calculate migration by diffusion, four parameters are needed for each nuclide. Two are simply material-dependent: the tortuosity, and the porosity factors. Two depend upon the nuclide and the form of it present in the pore water of the material: the molecular diffusion coefficient in the pore water, and the retardation factor.

The diffusivities or coefficients of diffusion of the radionuclides were taken from several sources such as the "International Critical Tables" and the "Handbook of Chemistry and Physics". All the radionuclides were as a first approximation assumed to be strong electrolytes, i.e., highly dissociated salts, quite mobile and capable of diffusing to infinite dilution. This assumption is conservative in its approach, because not all radioisotopes will be highly dissociated, for instance the carbonate salts of strontium, uranium and other heavy metals. As well, the diffusivity literature values for the electrolytes are given at 25°C, and they are likely to be lower at the lower temperatures expected in a repository environment.

It was assumed that for the purposes of the present study, the motion in an unsaturated medium could be represented by taking the above parameters but with a reduced value of the diffusion coefficient, and the values were all reduced by the same factor, five. The backfill material around the waste containers was assumed to be largely inert (e.g. sand), and no credit was taken for adsorption of the nuclides.

m) Populations at Risk

The hypothetically maximum exposed individuals in the general population for the study was a family group situated at the buffer zone boundary, taking all its water from a well drilled into the aquifer, the water of which had either passed beneath the array of vaults or up from around the mined repository. The extracted water was used for all domestic purposes, for irrigating vegetable crops, and for farm animals.

The animals fed in summer on grass, and in winter on cereals that had been grown in summer. Although the vegetable crops for the humans were irrigated when necessary, the grass and cereals were not.

The hypothetical family group used nearby surface waters, partly fed by the aquifer, for fishing and recreation.

The lifestyle of the hypothetical family group differed somewhat from the description in NUREG/CR-1759 and NUREG/CR-4370 because it was based on a more recent census, with more precise definitions of 'rural' dwellers and their water uses (U.S. Dept of Commerce, 1984).

It is quite likely that certain chemicals might be taken as natural isotopes in certain foodstuffs and might serve as buffers to reduce the intake of radioactive isotopes of the same chemical form. An example would be the use of iodized compounds for the cattle to prevent hoof-rot. However, to maintain the conservative approach, no credit was taken for such possibilities.

The modelling has thus focussed on the above group at maximum risk, taking all its water from the aquifer. A larger population group would need to supplement the water available from some of the aquifers studied, and hence would dilute the radionuclide concentrations in the contaminant plume and reduce the corresponding doses accordingly. Two dilution factors taken from NUREG/CR-4370 were considered: for a 'population well', and for a 'surface water' supply.

n) Dose Conversion Factors and Dose Calculations

The dose conversion factors (DCF's) that were used for this report and are in the data banks of COSMOS and SYVAC, are different from the set used for the DEIS document. Most of the changes are the result of new calculations by Johnson and Dunford (1979, 1981, 1982, 1983), and Dunford and Johnson (1988), but two changes deserve special mention in this section.

Originally it was assumed that carbon-14 was present in an insoluble form that would be retained in the body for very long periods, but the more realistic assumption is now made that it is dissolved in water as carbon dioxide. The corresponding DCF for ingested water was thus reduced by a factor of 40.

The question of the uptake of carbon-14 by plants is still not settled. The following mechanisms are possibilities.

i) Through the roots directly. This could occur by inorganic forms of carbon (either as carbonates or carbon dioxide in solution), moving with the soil water, and taken up by plants. This is real, but it is likely to be small.

ii) Volatilization of inorganic carbon through chemical reactions in the soil. The mechanism results in carbon dioxide diffusing from the soil to the plant foliage via the atmosphere.

iii) Active biological breakdown of carbonates in the soil by microbial action. This is likely small since most soil microbes feed on organic material.

iv) Passive biological breakdown where carbonates are inadvertently incorporated into biological cells which eventually decompose, releasing carbon dioxide.

Although the above processes are not completely absent, in keeping with recommendations by J.R. Johnson of AECL, and the EPA comments on the DEIS, their effects will be assumed to be negligible, until definitive experiments have been made.

For grass and cereals, the question of root uptake is minimized for this study because, in the provinces of New York State, it is highly unlikely that the grass or cereal crops will be irrigated, and hence their source of water containing carbon-14 has been removed from the model that was used for the DEIS.

The withdrawal of irrigation from grass and cereal removes a source of tritium in the water and the eventual tritium dose rate is reduced by a factor of about 1.5.

No atmospheric pathways were included in the modelling for the following reasons. Potential exposure via the atmosphere might result from inhalation of gaseous nuclides from waste decomposition or of dust suspended by the wind from a contaminated ground surface. Significant rates of gaseous release are not expected because waste decomposition will be relatively slow in the sheltered environment of the vaults during the first few centuries after closure. Because of the variability of wind directions, any releases which do occur are likely to be blown only occasionally toward the hypothetical family group which is maximally exposed by the groundwater pathway.

The other potential atmospheric pathway of a generic nature is from ground contamination in areas irrigated by contaminated well water. However, the potential exposure from this source will be significantly less than from the more direct drinking water and food pathways. One further factor which has not been included is the natural deterioration of the structure of an aboveground vault which could eventually contribute to additional ground-surface contamination.

o) Time Intervals, Time Ranges, and Nuclide Grouping

For calculations, it is necessary to specify the time-steps and time-range for each run, and some parameters are affected by the size of step.

An important factor is the half-life of a nuclide, and time-steps should not be larger than this, or else the fine detail of migration peaks may be lost. The time-range depends upon transit delays and must be large enough to cover the peak of concentration or dose, at the location of interest in a pathway.

The description of the failure functions for containers and roofs is affected by the choice of time-step. In the computer models, the information about successive failures cannot be conveyed on a finer time scale than the chosen interval, and it may be necessary to revise the description of the function so that its general behavior is still transmitted correctly - as the nuclide 'sees it'.

Nuclides have been divided into 3 groups according to the combined effects of half-life and transit time. The corresponding calculations covered: 3,000 years at 10-year steps; 24,000 years at 80-year steps; and 300,000 years at 1,000-year steps.

p) Assembly of Parameters for an Assessment

To summarize, for each scenario, it is necessary to provide, among other parameters, the following information for COSMOS:

i) The waste forms, activity and radionuclide inventories, containment, and packing.

ii) Vault dimensions and engineered barriers.

iii) Geometric parameters for leaching, and transfer to infiltrating water.

iv) Container failure functions.

v) Roof failure functions.

vi) Infiltration and aquifer velocities.

vii) Overburden and aquifer dimensions.

viii) Diffusion parameters for waste-forms, buffer, overburden, and aquifer.

ix) Definition of the population at risk and its lifestyle.

x) Definition of the well extraction rate and any surface sources.

xi) Mathematical decisions for the calculation, such as time intervals and ranges.

For mined repositories, the parameters are similar except that they include additional rock layers which lie between the top of the mined cavity and the lower boundary of the aquifer.

The numerical values of the parameters used in the modelling are discussed in Section 6.

6. Data Used in the Calculations

Data representing the parameters considered in the post-closure assessment of doses from the aboveground and belowground disposal options are described and presented in this Section. For completeness, some information included in Sections 4 and 5 is repeated.

In general, for consistency with the computer input and output, powers of ten are expressed in exponent form. For example, 0.0023 may appear as 2.30E-03. Dimensions are given in meters (m) and time in years (a). Doses can be expressed in sieverts (Sv) per year, but for this study they have been converted to the older unit, roentgen-equivalent-man (rem) per year, where the conversion is 1 Sv = 100 rem.

Table 2 lists the assumed radionuclide activities and inventories (assuming a total waste volume of 220,000 cubic meters) in 30 years accumulation of Class A and Classes B+C waste streams. Estimates were based on NUREG-0782. There are 24 different nuclides, ranging from short-lived iron-55 to several very long-lived actinides. These have been arranged into three groups for computational purposes, depending on likely peak-dose locations in time. The activities of plutonium-239 and -240 are listed together under the entry for Pu-239. In order to calculate inventory, decay losses while in transit, and dispersion rates in the biosphere, it is conservative to use the larger half-life of the two, hence the smaller decay factor.

For aboveground or belowground disposal, a typical vault unit is taken to be a concrete box (internal dimensions: 50 m long, 20 m wide and 5 m deep) covered by a one-meter thick concrete cap (Figures 1 & 2). The vault bottom is well drained to the ground layers beneath. Each unit houses four months' waste, with a packing fraction of 50 percent. The units are closely spaced in rows of nine; that is, three years' capacity in each row 450 m long (Figure 7).

Calculations are simplified by assuming that all the radioactivity from three years' production is present in one long vault (450 m long, 20 m wide, 5 m high) located at right angles to the groundwater flow, and at the downstream edge of the disposal facility. The loading is then multiplied by 10, to account for the full 30-year inventory (see Table 2).

In an actual disposal facility, radionuclides escaping from the rows of vaults located upstream with respect to the aquifer must migrate over a slightly longer path length, and hence remain in groundwater for a longer time and diminish in activity through radioactive decay. The doses resulting from the calculation, therefore, represent an upper bound. Dose reduction factors to account for this effect were estimated by the sub-routine SEERA and are listed in Table 3.

The underground mined repository (Figure 3) is assumed to be an array of chambers (chamber dimensions: 50 m long, 20 m wide, 10 m high). As a conservative and simplifying assumption for the calculations, all of the 30-years' inventory of radionuclides is assumed to be concentrated in a single chamber.

Class A waste is assumed to be compacted but readily leachable, while B and C wastes, considered together, are encapsulated in a leach-resistant cement-matrix. All wastes are contained in 210-liter (55-gallon) steel drums (cylindrical dimensions: 0.28 m radius, 0.86 m high). Their failure function is listed in Table 4, and shown in Figure 12. Briefly, failure starts at 40 years, reaches 35 percent at 120 years, and 100 percent at 240 years. As explained in Section 5, part b, the description must be appropriate to the time steps. The drums are piled in five layers with the bottom layer resting on a buffer layer 0.5 m thick. The buffer in the near-surface vaults is a mixture of 10 percent clay and 90 percent sand, a composition chosen to combine the good adsorption properties of clay with the free-draining characteristics of sand; the same mixture surrounds the waste mass on all sides in the mined repository. The buffer layers serve to retard migration of radionuclides away from the waste by a combination of adsorption and effects on flow patterns.

The wastes in drums have a mean-chord-length (four times the volume divided by the surface area) of 0.42 m. A slab having the same mean-chord-length (equal to twice the thickness, which is therefore equal to 0.21 m) is used to represent the waste form in the modelling of waste-form leaching, once the drums have failed.

In the cases of aboveground or belowground disposal, water is assumed to enter the vault through cracks in the roof, and to pass through the backfill where it can contact waste. Roof failure is modelled by treating a fraction of the roof area, increasing with time, as if it consisted of overburden material, thus allowing infiltration into the vault. Infiltration rates, and precipitation data leading to these, are presented in Table 5. For a mined repository, roof failure modelling does not apply.

The roof failure mechanisms were discussed in Section 5, and the resulting failure functions are listed in Tables 6, 7 and 8, and shown in Figures 12 and 13. As explained in Section 5, Part o, the descriptions are appropriate to the time steps.

i) Aboveground in all three provinces - leakage starts at 200 years, reaches about 50 percent by 800 years, and 100 percent by about 1500 years.

ii) Belowground in Province II - leakage starts at about 500 years, reaches about 20 percent at 1000 years, and levels off at 50 percent by about 1600 years.

iii) Belowground in Provinces IV and V there is assumed to be a more aggressive environment than in Province II - leakage starts at about 500 years, reaches about 50 percent at 1200 years, and levels off at 75 percent by about 1600 years.

Dimensions and flow parameters governing radionuclide migration from the surface of the waste form, through backfill and buffer, and through the surrounding ground (including bedrock in the mined disposal scenario), to a well and to surface water, are presented in Tables 9, 10, 11, 12, and 13.

The various layers preceding the aquifer, in the cases of aboveground and belowground disposal, are considered to be in saturated and unsaturated states, so that radionuclide transport is dominated respectively by advection and diffusion. The migration pathway for the mined disposal case is saturated along its entire length.

The aquifer path length of 300 m in overburden represents the horizontal distance from the most downstream row of disposal vaults to the site boundary, or from the discharge point from bedrock in the case of the mined cavity disposal option. This represents the width of the buffer zone between the vaults and the site fence. At this point, it is assumed that groundwater is drawn up through a well that supplies the needs of a farming family, denoted as the critical family. Any radionuclide contamination not taken up into the well is assumed discharged into surface water in the form of a river 10 m further on.

Well water is assumed to fill all the needs of the critical family; supplying water for drinking, bathing, and watering vegetable crops and stock. Since the critical family is assumed to be entirely self-sufficient in its diet, all food and drink (excepting marine seafood) is assumed to contain radionuclides. Contamination in surface water accounts for doses associated with eating fresh-water fish and includes exposure to river sediment.

Well demand rates for the critical family, for a larger population, and a flow rate representing a typical surface stream, are presented in Table 14. If they are divided by the aquifer flows of Table 9, they provide the dose dilution factors of Table 15. Annual doses for the other two water-demand situations may be obtained by reducing the doses to adults in the critical family by these factors.

Annual adult doses for the critical family are calculated by multiplying the radionuclide concentrations in the well and surface waters by the relevant Pathway Dose Conversion Factor (PDCF) from Tables 16 and 17, which consist of the product of radionuclide transfer coefficients in the food chain and dose conversion factors.

The transfer coefficients are based on the food-chain pathways, transfer models, and diet and lifestyle data (samples appear in Tables 18, 19 and 20) as found in Canadian Standard (CSA, 1987). Additional information was provided for this work by the original contributors (see Section A.1.5 of Appendix A), in order to obtain Dose Conversion Factors (DCF's) to represent committed dose equivalents for all of the following organs:

LNG: lung,
STO: stomach wall,
LLI: lower large intestine,
KID: kidney,
LVR: liver,
RBM: red bone marrow,
BOS: bone surface,
THY: thyroid.

In addition, the Effective Committed Dose Equivalents (EDE), were updated from the Canadian Standard (CSA, 1987). The new calculations used organ dosimetry data from Oak Ridge National Laboratories (Snyder et al., 1974), and the code GENMOD developed at Chalk River Nuclear Laboratories (Dunford and Johnson, 1988). The changes from those in the DEIS are listed below.

Differences in the PDCF's between the old and new EDE-values have been noted for the nuclides listed below. These differences in value originate from differences in DCF's and transfer model assumptions. Comments appear where changes not related to the new DCF's were made.

H-3	: down by factor of 1.5	(no watering of grass/cereals)
C-14	: down " " " 1300	(about 32 for no root uptake, times 40 for assuming C-14 in CO_2 only)
Ni-59	: up by factor of 2	
Ni-63	: up " " " 2	
Tc-99	: up " " " 4	
Np-237	: down " " " 10	
Pu-238	: up " " " 8	
Pu-239	: up " " " 2	
Pu-241	: up " " " 8	
Pu-242	: up " " " 8	
Am-241	: down " " " 60	(error in half-life corrected)
Am-243	: up " " " 1.5	
Cm-243	: up " " " 2	
Cm-244	: up " " " 1.5	

Errors of up to a factor of ten seem generally to be within uncertainty errors expected for these quantities. Errors larger than a factor of ten are attributed to changes resulting from comments received concerning biosphere assumptions, or corrections to fundamental constants. As an example of the latter, half-lives were checked and corrected against those found in IAEA, 1987.

7. Results and Their Interpretation and Application

7.1 Presentation

As was done in the DEIS, dose assessments have been calculated, for the three reference disposal technologies, situated on hypothetical sites which might be typical of each of the three most suitable geologic provinces in New York State, for the scenarios described in previous sections. The results are summarized in Tables 21, 22, and 23. As discussed in Appendix B, results in this report are revised from those reported in the DEIS as a consequence of some updating of assumptions and data. The updating reflects additional review and development done since issuance of the DEIS in July, 1987, some of which was in response to comments received on that report. Also, the calculated doses to individual organs have been included in this document to provide additional basis for evaluation related to the dose performance objectives in 6 NYCRR Part 382.

For the aboveground and belowground vaults, the dose predictions split quite naturally into three groups, according to the times at which their peaks occur, after closure. The times are: Group 1, after a few hundred years; Group 2, after a few thousand years; Group 3, from a few thousands to some hundreds of thousands of years. For the mined repositories, the delays are all increased, the earliest group is rendered completely negligible, and peaks from Group 2 can be divided into two periods: after some tens of thousands of years; after some hundreds of thousands. Migration of Group 3 to the surface is expected to take millions of years.

Effective committed dose equivalents, and committed dose equivalents for eight organs, were calculated for adults and infants in the hypothetical farm family chosen to be the 'most exposed population'. However, the infant doses do not show any significant differences from the adult doses, and certainly would not lead to any different conclusions, so they have not been included in this report.

Tables 24, 25, and 26 list dose rates for aboveground vaults in Provinces II, IV and V, respectively.

Tables 27, 28, and 29 list dose rates for belowground vaults in Provinces II, IV and V, respectively.

Tables 30, 31, and 32 list dose rates for mined repositories in Provinces II, IV and V, respectively.

Whenever a particular nuclide, or nuclides, can be identified as making the most significant contribution to a total dose, this is indicated in the tables.

The dose rates that were estimated at very long times for the very long-lived nuclides in Group 3 deserve attention, more for their implications than their actual values. They are given separate discussion in Section 8.

Figures 14 and 15 permit an inter-provincial comparison of aboveground and belowground vaults, respectively.

Figures 16, 17, and 18 compare the three disposal technologies, in each province in turn.

Figures 19 through 24 show the contributions of the most significant nuclides in each province, for aboveground and belowground vaults, in turn.

An indication of the magnitude of total doses (essentially from I-129 only) from mined repositories in the very long term is given in Figure 25.

7.2 Comparison with studies made elsewhere

It is natural to compare these results with earlier calculations made for the .S. Nuclear Regulatory Commission.

An important difference between the methods of COSMOS/SYVAC and the NRC calculations (NUREG-0782, NUREG-0945, and NUREG/CR-4370) is the modelling of the transfer of radionuclides from waste to leachate within the vault.

The calculations in NUREG-0782 that were used in assessments for 10 CFR Part 61 were quite empirically based. An estimate of the rate of infiltration of water into the top of the waste trench was combined with measurements at Maxey Flats of the ratio (concentrations of several nuclides in the water in a flooded trench)/(their concentration in the waste) to give an estimate of the maximum rate of leaching. This maximum rate was then reduced by a correction factor which was estimated from the expected time of water contact after each precipitation event. The estimate for the correction factor under conditions in the northeastern states ranged from 0.0013 to 0.0054. As a result of comments received after the method was published, an update report (NUREG/CR-4370) suggested an alternative range for the correction factor, from 0.2 to 0.6, which would have increased the dose to individuals in the hypothetical farm family by almost two orders of magnitude. Although some intermediate value was recommended, based on specific site conditions, the published generic results were based on the original factors, and were not changed.

The water-usage factors of the critical farm family in the NRC calculations were based on an earlier U.S. census. The figures for total water consumption in rural areas were essentially correct in the census, but the category of 'rural dweller' was not well defined, and too few such dwellers were reported, with the consequence that the implied average use of water per capita was too high. The census of 1984 (United States Dept. of Commerce, 1984) defined the category more precisely, the number of rural dwellers increased, and the per-capita use of water was reduced. The calculations here have used the later figures, for a hypothetical farm family of four people.

Some of the PDCF's in the COSMOS/SYVAC analyses are higher than the NUREG values. In the DEIS calculations, the carbon-14 numbers were considerably higher, but revised values for this report are not (the carbon-14 doses were discussed in considerable detail in Sections 5 and 6). The most significant difference remaining is for iodine-129, where the factor is about two orders of magnitude higher. It is not easy to find all the assumptions in the NRC calculations. The AECL values used in this report are considerably closer to the ICRP values (quoted in NUREG/CR-4370) and, in short, there is no obvious reason for change.

Comparisons were made by the U.S. Environmental Protection Agency (EPA) with the estimates of the DEIS, (see EPA comments in the FEIS). Their estimates, using a version of the PRESTO code, were in good agreement with COSMOS/SYVAC except for carbon-14, where the latter were about 30 times higher. The difference arose because COSMOS/SYVAC assumed the uptake of carbon-14 by roots and PRESTO-EPA did not. Scientific opinion is still very much undecided on precise values, but it is probably fair to say that a majority opinion would opt for very small uptake. The present version of SYVAC/COSMOS DCF's, used in this report but not in the DEIS, have assumed zero uptake and the resulting doses should now be in good agreement with the EPA estimates. DCF's are discussed in detail in Sections 5 and 6.

7.3 Some general comments

The modelling has evaluated the expected performance of LLRW disposal, on the assumption that no unanticipated system failures have occurred-for example, as a result of extreme environmental conditions, or inadequate quality control. In a site- and design-specific study, these would have to be assumed based on scientific knowledge and their probabilities and effects assessed.

The models describing radionuclide migration are one-dimensional. For the underground mined repository, water flow from the waste is assumed to rise vertically through the rock layers, i.e., the shortest path to the overlying aquifer. In the aquifer, the flow is horizontal to the well. For the near-surface vaults, water escaping beneath the units is assumed to flow directly down through the unsaturated zone to the aquifer, and then horizontally to the well. On none of these paths is allowance made in the calculation for the spreading out of the contaminant plume. Dispersion transverse to the flow would cause dilution of the contaminants and thus lead to a reduction in doses. Such dispersion might be promoted by horizontal geologic strata that cause directional differences in the permeability to groundwater flow. Since such strata might be present in the rock above the mined repository, and in the unsaturated zone below the vaults, their omission from the calculations is an additional conservative factor.

Once one departs from the assumption that unacceptable levels of radionuclides will be kept away from the exposed population (with completely unacceptable doses), there is no such thing as a completely 'conservative' scenario, because it will always be possible to perturb it so as to produce higher doses. In practice then, scenarios are a mixture of 'conservative' and 'realistic' assumptions (bearing in mind that these are subjective terms!) and the present modelling is no exception in that some of its assumptions are 'less conservative' than others. However, an attempt has been made to spell out relevant assumptions and the reader who would make them differently should be able to assess the resulting changes in assessments.

Appendix B contains a detailed discussion of the differences, in modelling and data, between this Support Study and the one reported in the DEIS.

A quick scan of the tables and figures shows immediately that all of the scenarios for the three disposal technologies have estimated doses that never exceed the regulating limit by large factors and are usually well below them. The studies are thus well suited to be a departure point for future refinement of modelling, improvement of data, or changes to represent specific design details or locations.

7.4 Some detailed comments

This section contains several detailed comments on particular issues.

a) A comparison of the doses for aboveground and belowground vaults shows the value of the overburden. One might expect that the aboveground performance would be worse because the roof failure function allows water to infiltrate the vault in larger quantities and at earlier times. However, in the modelling for this study, the depths of total overburden were reduced considerably (in response to comments received on the DEIS), and the remaining thicknesses between aquifer top surface and ground level were not large (see Table 9). The aboveground vaults, sitting on the surface, thus had appreciably thicker layers between vault bottom and aquifer than the belowground vaults, which were partly buried. The retardation in these extra thicknesses has been enough to overcome the accelerated migration that came from the worse roof performance. In regions of deeper (non-average) overburden one would expect the belowground vaults to have the superior performance.

b) Iodine-129 is a very significant contributor, particularly to effective dose equivalent and thyroid dose, because of its long half-life and its mobility. However, this contribution arises from very small quantities in the inventory and the measure of those amounts is not at all certain. It has been conservatively assumed to be present at the lowest limit of measurement, but arguments based on its production processes suggest that it is, in fact, much less plentiful. The question is under study at several establishments, and was a topic at an Electric Power Research Institute Workshop in Palo Alto, CA, in November 1987.

Similar comments apply to the estimated very low levels of uranium-234, uranium-235, and uranium-238, which may give rise to significant doses at very long times. Other long-lived nuclides which were shown to be potential dose contributors via groundwater at very long times were nickel-59, cesium-135, and plutonium-242.

c) The uptake of carbon-14 has been discussed in several sections, and is mentioned again in Appendix B. At the moment, it is not the most significant contributor to dose, but if future work indicates that a small uptake rate should be assumed, then it will rise in importance, particularly if the discussion of the previous paragraph, b), results in a reduction of iodine inventory.

d) There are still discrepancies, sometimes as much as an order of magnitude, in the PDCF values of different compilations - notably for carbon-14, technetium-99, and iodine-129. The Canadian CSA and the ICRP numbers tend to be higher than the NRC numbers of NUREG-0782.

7.5 Some extensions of the results

If the assumptions are appropriate in the SEERA calculations of dose-reduction factors, the results, which are listed in Table 3, can be used to multiply the detailed organ doses and significant-nuclide doses to allow for vault loading delays and the vault siting pattern. The effects range from reduction by a factor of 10 for the Group 1 nuclides to negligible corrections for the Groups 2 and 3 nuclides. The correction factors depend only upon vault loading delays and the spacing between vault rows and hence are the same for aboveground and below-ground vaults in the same Province.

Simple sensitivity calculations will indicate the effectiveness of increases in buffer thickness or overburden thickness, or a change in aquifer velocity. For instance, for an increase in thickness by X, in a region with porosity of P, and bulk velocity v, the increase in transit time for a nuclide peak is approximately, T = RXP/V, where R is the retardation factor for that nuclide. If the nuclide decay factor is lambda, then the increased time will result in reduction in peak height by approximately exp(-lambda.T). Similar calculations can be made to allow for changes in velocity.

Changes in bulk flow rates in the aquifers, in pumping rates at the well, or flow rates of surface water, will produce changes in dilution factors that can be applied to the dose rates.

The effect on modelling results of changes in infiltration velocities are not as readily predicted as those from changes in aquifer velocities because they appear in non-linear fashion in mass-transfer factors, but a rough idea of their effect could be obtained by the sensitivity analysis described above.

8. Overall Conclusions

Overall, the new results lead to conclusions similar to those in the DEIS; some differences in detail are, however, evident. Because of changes to some parameters, such as in pathway dose conversion factors, especially the changes related to carbon-14 behavior, many of the peak values of effective dose rate equivalent have been reduced significantly.

Of the nine combinations of disposal methods and site characteristics evaluated (Table 21, and Figures 16, 17, and 18), seven yielded peak effective dose rate equivalents that met the performance objective of keeping whole body doses below 25 mrem per year. Only the aboveground and belowground vaults in the hypothetical setting in Province V yielded greater values (by about a factor of 2).

As well as the effective dose rate equivalents, the committed dose rate equivalents for the eight specific organs and tissues can be compared with the performance objectives of 75 mrem per year to the thyroid and 25 mrem to any other organ. The results in Table 22 show that doses to the thyroid exceed the performance objective for all technologies in Province II, and for the aboveground and belowground vaults in Province V. Estimated doses to the other organs (Table 23) are less than the objective for all technologies and all site characteristics.

As observed in the DEIS results, the significant dose contributions result from the three mobile long-lived radionuclides, carbon-14, technetium-99, and iodine-129. Although these nuclides account for only 0.008 percent of the initial radioactivity in the wastes, they have a very large influence on the peak annual doses because of their relatively low retardation in the environment, and their long half lives (that for carbon-14 is 5730 years, technetium-99 is 213,000 years, and iodine-129 is 16,000,000 years). In fact, among the results in this report, iodine-129 alone is responsible for all doses which exceed the performance objective.

Because of the large differences in the relative importance of specific radionuclides, the assessment results have been related to three groups of nuclides which have similar behavior because of a combination of their half lives and migration rates. In the listing in Tables 24 through 29, the potential for Group 1 radionuclides to cause exposure is limited to a few hundred years. Group 2, which contains only the mobile nuclides mentioned above, produces exposures from the near-surface vaults which peak in the 1000 to 10,000 year period, and is the most important group. Group 3 are well retarded but long-lived, and thus could potentially result in minor exposures in the long-term beyond 10,000 years.

In Group 1, tritium, as tritiated water, is the only nuclide which is estimated to result in even minor exposures. Although the tritium accounts for 58 percent of the initial radioactive inventory and migrates at the same rate as water, the maximum predicted dose from it is only 0.006 mrem/year. Since its half life is 12.3 years, it decays almost completely in several hundred years.

For the three nuclides in Group 2, their importance in descending order is iodine-129, carbon-14, and technetium-99. As can be seen from Figures 19 to 24, the annual effective dose equivalents from iodine-129 are an order of magnitude or more greater than those from carbon-14, and a factor of 1000 greater than those from technetium-99. The estimated doses from iodine-129 are believed to be quite conservative since pessimistic assumptions were made in accounting for areas of uncertainty. Uncertainties currently exist in the estimates of iodine-129 concentrations in the waste, the importance of dilution by stable iodine-127 in the environment, and the mobility of the chemical species of iodine in the waste. If, as more information on these topics becomes available, the degree of conservatism in assumptions is found not to be excessive, specific steps may be warranted to ensure that iodine-129 inventories in the facilities are well controlled.

None of the Group 3 nuclides produced significant doses in the first 10,000 years. However, some of the nuclides in that group are sufficiently long-lived that, even though their migration in the disposal system is greatly retarded, there was concern that they may eventually produce significant doses. Although the levels of uncertainty become rather large because of the potential climatic and demographic changes, the codes were used to estimate doses for times of 300,000 years or more. Cesium-135 and some of the actinides appeared to be potentially significant, but none of the peak annual effective dose equivalents exceeded the values estimated to occur at earlier times from the Group 2 nuclides. One factor which should be kept in mind, however, is that no allowance was made in the modelling for the losses in stability of the disposal units in these very long time scales. For example, some surface pathways of importance are likely to develop from the aboveground vaults, but were not assessed.

The results of all of the assessments indicate that the three disposal technologies provide a range of radiological protection, with the greatest barrier to biosphere contamination provided by underground mined repository disposal. All of the combinations of disposal method and typical site characteristics, even with quite conservative assumptions, showed excellent isolation of those radionuclides constituting the major sources of radioactivity expected in the wastes, that is Groups 1 and 3. However, the small amounts of the long-lived mobile radionuclides in Group 2 could result in dose levels from the near-surface facilities (aboveground and belowground vaults under assumed Province V conditions) which exceed the performance objectives at times beyond 1200 years.

The results of the assessments for the deep underground mined repositories show that groundwater transit times are long enough and flows small enough that even poorly retarded nuclides with very long half lives reach the well withdrawing water from the overburden at only very low concentrations. As a result, doses to individuals are significantly lower than those for the aboveground and belowground vaults.

Also, the potential concerns of loss of stability and inadvertent intrusion associated with near-surface vaults, are not significant for mined repositories. The only significant doses to the public are from iodine-129 and technetium-99 (carbon-14 has decayed). The peak doses from these radionuclides do not occur before 10,000 years, a length of time considered by the NRC to be an appropriate limit for assessments such as these. The differences in the doses from the mined repositories in the three provinces result primarily from a combination of effects of the various site characteristics, both favorable and less favorable, rather than a single dominant factor. Thus, a discrimination between candidate provinces based on these results would be very tentative. The characteristics of a particular site within any one of the provinces could vary significantly from those modelled. Also, the modelling did indicate that the engineered barriers associated with waste emplacement, as well as the natural features of the site, were important contributors to the system performance.

The predictions for the near-surface vaults, both aboveground and belowground, show that the doses from the principal radionuclides (Groups 1 and 3) are consistently very small, less than 0.001 percent of natural background. However, as mentioned above, the calculated doses from the long-lived mobile radionuclides are significant since the doses in some instances exceed the present performance objectives. If NRC standards are changed (as expected) to omit dose to individual organs from the performance objectives, and focus on a limit on effective dose equivalent only, then it seems probable that all of the technologies could be made to satisfy the objective.

The expected performance of the underground mined repositories is considerably better than the near-surface disposal methods. However, based on the estimated doses in this report, there is not a large difference between the performances of the aboveground and belowground vaults. This is, perhaps, not surprising since the distinctions between them in the models were relatively small because of the limitations imposed by the newly assumed site characteristics. Since the depths to the water table were reduced considerably from those assumed for the DEIS, there was little difference between the elevations of the aboveground and belowground vaults relative to the original site grade (Table 9). In order to maintain a zone of unsaturation below them, the belowground vaults had to project above the original grade when constructed, and then be buried by mounding at the time of closure. The unsaturated zone was thicker beneath the aboveground vaults than the belowground vaults, and thus compensated for the somewhat earlier and greater roof leakage into the former.

Although the results of the modelling indicate that facilities using any one of the technologies may be capable of satisfying the performance objectives, a number of factors not considered in the calculations could have important effects on the actual performance. The facilities described by the models were defined with features that provided isolation at roughly the minimum level demanded by the regulations. The specific details of the proposed location, design, and operational procedures could significantly change the differences in dose estimated by the generic analyses. Specific sites could provide more delay and dilution of radionuclides escaping the disposal units. Of equal importance, the features of the disposal units, and the form and packaging of the wastes they contain, might contribute to less release over the period of concern.

Also, the modelling was directed at an evaluation of the expected performance of the facility assuming that no unanticipated system failure occurred as a result of extreme environmental conditions (e.g. exceptional earthquake), inadequate quality control, or operational accidents. Some of the factors not analyzed but which could be important in determining the overall risk are discussed below. The discussion is meant to be illustrative rather than an exhaustive analysis of all such factors.

The transfer of nuclides from the waste to water which can carry them through the food chain is perhaps the most critical process in near-surface disposal systems. Three submodels in the COSMOS code describe a sequence of processes that control the nuclide transfer to groundwater. The first control is the protection from water contact afforded by the vault structure and the drums containing the waste. The deterioration of each is modelled in a conservative way. In the period when some drums have failed but before the vault leaks, there is assumed to be a sufficient water film on the waste that radionuclides slowly diffuse out into the water film and thence eventually to the groundwater. This slow process continues throughout all vaults in those sections which remain unaffected by leakage, but contributes a relatively small fraction of the predicted dose. Once water begins to leak into the vault, the percolation of water around the waste in the wetted areas flushes the nuclides in the water film toward the bottom of the vault. The degree of flushing is controlled by the process of advection which depends on the total flow and drainage rates. The contaminated water passes through the layer of buffer material (e.g. sand plus adsorbents) and out into the unsaturated soil beneath the vault. The buffer material and the unsaturated soil retard the migration of most of the nuclides, but their interaction with the mobile nuclides is significantly less. The nuclides that emerge from the buffer are carried into and through the soil by the water flowing from the vaults.

Since the peak doses result from the water that enters the vaults, the factors that control the roof leakage rates are very important. For the aboveground vault (AGV), design features which promote precipitation runoff away from areas of potential cracking can reduce leakage to less than the rates assumed in the calculations. Deterioration of an AGV concrete roof would be more rapid than that of a belowground vault (BGV) because of its exposure to the more aggressive environment in the atmosphere. However, because of the accessibility of an AGV, roof repairs could be made as long as institutional control was maintained. The buried roof of a BGV would be difficult to repair, but would be expected to deteriorate significantly more slowly. Also, prudent design of a BGV would incorporate supplementary barriers to roof leakage, such as an overlay of low permeability clay and a water-shedding membrane. However, the models assume that no infiltration resistance, beyond normal site soil, controls water infiltration to the BGV roof.

Of similar importance to the control of water leakage into the vaults is the reliability of the systems to remove water from the vaults. If drainage rates from the waste zone cannot accommodate the rate at which water is infiltrating, the waste becomes completely saturated, and the excess infiltration overflows, carrying leached radionuclides with it. The consequence is most serious if the overflow is on the ground surface, since it can cause rapid migration of the contained nuclides by surface flow including those nuclides normally well retarded by adsorption on the soil. This phenomenon, known as "bathtubbing", was illustrated by past experience with shallow land burial at West Valley. Exposures from such a system failure will tend to be less, the deeper the waste is buried.

Although, in general, radionuclides in the unsaturated soil zone will tend to migrate downward, release of nuclides into the uppermost soil region may, under the action of plant roots, temperature gradients, and diffusion, move upward toward the surface. Additional potential for dose would then result. Again, deeper burial provides an advantage.

"Bathtubbing" in an AGV would be particularly serious because, not only would escape likely carry nuclides onto the ground surface, but also there would be a significant risk that freezing of the trapped water would damage the vault. These concerns led to the incorporation in the regulations of a necessity to ensure that the design of the disposal units provides adequate drainage to the subsurface. Supplementary protection beyond a leak-tight roof could be provided during institutional control by a system to pump and treat any water that leaked into the vaults. Since the regulations require vault design that avoids "bathtubbing", the potential consequences of the phenomenon have not been included in the assessments.

All three of the disposal technologies evaluated in the modelling included a 0.5 m thick layer of "buffer" adjacent to the waste in the direction of potential nuclide migration. The buffer would be a permeable natural material, e.g. a mixture of sand and clay, chosen for its good adsorption properties to retard most radionuclides, but allowing the flow of water as required. In an actual disposal unit, the thickness and properties of the buffer could be chosen to complement the isolation characteristics of the site, and, in effect, close the migration path against escape of specific radionuclides.

The models describing radionuclide migration are one-dimensional. For the underground mined repository, water flow from the waste is assumed to rise vertically through the rock layers, i.e., the shortest path to the overlying aquifer. In the aquifer, the flow is horizontal to the well. For the near-surface vaults, water escaping beneath the units is assumed to flow directly down through the unsaturated zone to the aquifer, and then horizontally to the well. Along none of these paths is allowance made in the calculation for the spreading out of the contaminant plume. Dispersion transverse to the flow would cause dilution of the contaminants and thus lead to a reduction in doses. Such dispersion might be promoted by horizontal geologic strata that cause directional differences in the permeability to groundwater flow. Since such strata might be present in the rock above the mined repository, and in the unsaturated zone below the vaults, but are not included in the modelling, the calculations contain this additional conservative factor.

As indicated in Figures 14 through 18, the performance of each technology varies considerably with the characteristics of the site in which it is situated. Although the assumed site characteristics have been associated with specific physiographic provinces in New York State, the performance of an actual site would, of course, depend on the combination of controlling characteristics, no matter where in the State it was located. Some indications of which characteristics are important can be obtained by comparing the results from each Province.

Of the mined repositories, only that in Province II results in exposures which exceed the performance objectives during the first 10,000 years. Since the assumed retardation factors were the same in all provinces (with a few exceptions), and the only significant doses are associated with iodine-129, the controlling factors must be the groundwater flow rates in the rock and in the aquifer. A higher flow in the rock leads to a greater transfer rate of very long-lived nuclides to the aquifer. A lower flow in the aquifer leads to less dilution, higher concentration in the well, and hence to greater predicted exposures. These conditions are most evident in the Province II data.

For the aboveground and belowground vaults, the best performance is predicted for the conditions assumed in Province IV, the worst for Province V. Although differences were assumed between provinces in parameters such as infiltration rate, soil porosity and tortuosity, and unsaturated zone thickness, the most important is the 20-fold difference in aquifer flow rate, the high flow again being responsible for greater dilution. However, higher flow rates also mean shorter transit time across the buffer zone. The advantage of the greater dilution is only fully effective if the vault design ensures a low nuclide escape rate, particularly for the nuclide tritium. Although no calculations were done for an extremely low aquifer flow, it is probable that the inability of a well to supply normal demands would also result in low exposures.

The overall conclusions which can be drawn from this modelling report are:

1. The specific design of the LLRW disposal facility will have very significant effects on its ability to meet the performance objectives.
2. New York State (NYS) should seek to obtain an accurate estimate of the radionuclide inventory to be disposed in the LLRW facility so

that the source term for the site may be more accurately projected.

3. NYS should carefully review LLRW disposal facility proposals to assure that the performance objectives can be met and if, with conservative assumptions, the facility cannot meet the performance objectives of 6 NYCRR Part 382, the State must consider facility or site design modifications and/or restricting the inventory of radionuclides which would lead to failure to meet the performance objectives.
4. Since surface contamination and airborne pathways were not modelled, NYS should very carefully review proposals for AGV's.
5. Assumptions made in conceptual modelling are critical to environmental pathway analysis and dose assessment, e.g. the thickness of overburden assumed will significantly affect dose estimates.
6. Most assumptions in the conceptual environmental pathway analysis were very conservative and thus an actual facility design and site can be expected to result in lower doses to the general population, and many facilities can be expected to meet the performance objectives.
7. The results and assumptions contributing to these conclusions indicate that:
   i. Based on the modelling and the fact that surface contamination was not assessed, AGV's show the least ability of the three methodologies considered to meet the performance objectives.
   ii. Site-specific design information will be necessary for the Siting Commission to obtain a certification from the New York State Department of Environmental Conservation (DEC) pursuant to 6 NYCRR Part 382.
   iii. Special attention should be given by the Siting Commission, New York State Energy Research and Development Authority and DEC to assuring the dose performance objectives are met for the mobile long-lived radionuclides of C-14, Tc-99 and I-129 in groundwater which were the major contributors to dose in this study.

iv. The Siting Commission must obtain the necessary data to be able to model the site to prove the location and facility design can meet the performance objectives. This will be a major undertaking.

This study showed that the actinides were not significant contributors to dose in the first 10,000 years after closure. However, for surface-water flow, which was not modelled herein, it is expected that they will be significant dose contributors.

9. <u>References</u>

Andres, T.H., 1986. Features of the SYVAC Executive. Proceedings of Twentieth Information Meeting of the Canadian Nuclear Fuel Waste Program. Atomic Energy of Canada Limited Technical Record, TR-375, Pinawa, Manitoba, 1986.

Atomic Energy Control Board of Canada, 1987. Regulatory Objectives, Requirements, and Guidelines for the Disposal of Radioactive Wastes - Long Term Aspects. Regulatory Document R-104, 1987.

Canadian Standards Association, 1987. Guidelines for Evaluating Derived Release Limits for Radioactive Effluents for Normal Operation of Nuclear Facilities. Canadian Standard 1987.

Cheung, S.C.H., and T. Chan, 1983. Parameter Sensitivity Analysis of Near Field Radionuclide Transport in Buffer Material and Rock for an Underground Nuclear Fuel Waste Vault. Atomic Energy of Canada Limited Report, AECL-7801, Pinawa, Manitoba, 1983.

Climatic Atlas of the United States, 1954. Harvard Press, Princeton, N.J., 1954.

Culkowski, W.M., and M.R. Patterson, 1976. A Comprehensive Atmospheric Transport and Diffusion Model. Oak Ridge National Laboratory, Tennessee, USA, Report ORNL/NSF/EATC-17, 1976.

Dirac, P.A.M., 1953. The Chord Method. Introduction to the Theory of Neutron Diffusion, Vol. 1. Edited by K.M. Case, F. de Hoffman, and G. Placzek. Published by Los Alamos Scientific Lab, New Mexico, USA, 1953.

Dormuth, K.D., and G.R. Sherman, 1981. SYVAC - A Computer Program for Assessment of Nuclear Fuel Waste Management Systems, Incorporating

Parameter Variability. Atomic Energy of Canada Limited, Report AECL-6814, Pinawa, Manitoba, August 1981.

Dunford, D.W., and J.R. Johnson, 1988. GENMOD - A Program for Internal Dosimetry Calculations. Atomic Energy of Canada Limited, Report AECL-9434, in printing January 1988.

Fraser, J.S., and R.G. Jarvis, 1985a. A Mathematical Model, and Code LIXY, for Leaching of Radionuclides from Containment. Atomic Energy of Canada Limited, Report AECL-8827, Chalk River, Ontario, 1985.

Fraser, J.S., and R.G. Jarvis, 1985b. A Mathematical Model, and Code HADES, for Migration of Radionuclides from a Shallow Repository. Atomic Energy of Canada Limited, Report AECL-8828, Chalk River, Ontario, 1985.

Goodwin, B.W., T.H. Andres, P.A. Davis, D.M. Leneveu, T.W. Melnyk, G.P. Sherman, and D.M. Wuschke, 1987. Post-Closure Environmental Assessment of a Concept for the Disposal of Nuclear Fuel Waste, submitted to Int. Journal of Waste Management and the Nuclear Fuel Cycle.

Heinrich, W.F., and T.H. Andres, 1985. Response Functions for the Convection-Dispersion Equations Describing Radionuclide Migration in a Semi-Finite Medium. Ann. Nucl. Energy, Vol. 12, No. 12, pp. 685-691, 1985.

IAEA, 1987. Handbook on Nuclear Activation Data, Technical Report Series #273. International Atomic Energy Agency, Vienna.

ICRP, 1977. Recommendations of the International Commission on Radiological Protection, ICRP Publication #26, Pergamon Press, N.Y., 1977.

Jarvis, R.G., R.Y. Adam, C.I. Bretzlaff, J.M. Laurens, and S.R. Wilkinson, 1986. The COSMOS-S/D Assessment Code Complex for a SLB Repository at CRNL. Proceedings of the Second International Conference on Radioactive Waste Management, Winnipeg, Manitoba, 1986 September 7-12. Atomic Energy of Canada Limited, Report AECL-9350, Chalk River, Ontario 1986.

Johnson, J.R., D.G. Stewart, and M.B. Carver, 1979. Committed Effective Dose Equivalent Conversion Factors for Intake of Selected Radionuclides by Infants and Adults. Atomic Energy of Canada Limited, Report AECL-6540, Chalk River, Ontario, November 1979.

Johnson, J.R., 1982. Dose Conversion Factors Used in the Current Canadian High Level Waste Disposal Assessment Study. Radiation Protection Dosimetry, Vol. 3, No. 1/2, pp. 47-50, 1982. Atomic Energy of Canada Limited, Report AECL-7869, Chalk River, Ontario, 1982.

Johnson, J.R., and D.W. Dunford, 1983. Dose Conversion Factors for Intakes of Selected Radionuclides by Infants and Adults. Atomic Energy of Canada Limited, Report AECL-7919, Chalk River, Ontario January 1983.

Laurens, J.M., 1985. A Computer Model of the Biosphere, to Estimate Stochastic and Non-Stochastic Effects of Radionuclides on Humans. Atomic Energy of Canada Limited, Report AECL-8645, Chalk River, Ontario, 1985.

Laurens, J.M., and C.I. Bretzlaff, 1986. Model of a Failure Function for Containers. AECL/CRNL Technical Memorandum MWM-008, 1986.

LeNeveu, D.M., 1987. Response Functions of the Convection Diffusion Equations Describing Radionuclide Migration in a Finite Medium. Ann. Nucl. Energy, Vol. 14, No. 2. pp. 77-82, 1987.

Luikov, A.V. 1968. Analytical Heat Diffusion Theory. pp 288 et seq. Analysis of the Generalized Solution. Academic Press, 1968.

NCRP 1987. Ionizing Radiation Exposure of the Population of the United States. National Council for Radiation Protection. Report NCRP #93, 1987.

NUREG-0782, 1981. Draft Environmental Impact Statement on 10CFR Part 61. United States Nuclear Regulatory Commission, 1981.

NUREG/CR-1759, 1981. Data Base for Radioactive Waste Management. United States Nuclear Regulatory Commission, 1981.

NUREG/CR-4370, 1986. Update of Part 61 Impact Analysis Methodology. United States Nuclear Regulatory Commission, 1986.

Palmer, J.F., 1981. Derived Release Limits for Airborne and Liquid Effluents for CRNL During Normal Operations. Atomic Energy of Canada Limited, Report AECL-7243, Chalk River, Ontario, 1981.

Shamir, U.R., and D.R.F. Harleman, 1967. Dispersion in Layered Porous Media, Proc. of the Am. Soc. of Civil Eng., Journal of Hydraulic Division, Vol. 93, pp. 237-260, 1967.

Sherman, G.R., D.C. Donahue, S.G. King, and A. So., 1986. SYVAC2- A Systems Variability Analysis Code for Assessment of Nuclear Fuel Waste Disposal. Atomic Energy of Canada Limited Technical Record, TR-317, Pinawa, Manitoba, October 1986.

Snyder et al., 1974. A Tabulation of Dose Equivalent per Micro-Curie Day for Source and Target Organs of an Adult for Various Radionuclides. Oak Ridge National Laboratory, Report ORNL 5000, Vols. 1 and 2, 1974.

United States Dept. of Commerce, 1984. Statistical Abstract of the United States, Washington, D.C., 1984.

United States National Oceanic and Atmospheric Administration (NOAA), 1974. Climates of the States: Volume 1 - Eastern States, U.S Department of Commerce.

US NRC 10CFR/Part61. United States Nuclear Regulatory Commission, '10CFR Part-61 - Licensing Requirements for Land Disposal of Radioactive Waste', Federal Register, Vol. 47, No. 248, pp 57446-57477, 27 December, 1982.

Wilkinson, S.R., 1987a. ATMO, A Model of Radionuclide Migration in the Atmosphere. Atomic Energy of Canada Limited, Report AECL-9507, Chalk River, Ontario, 1987.

Wilkinson, S.R., 1987b. HYDROS, A Model of Radionuclide Migration in Surface Waters. Atomic Energy of Canada Limited, Report AECL-9506, Chalk River, Ontario, 1987.

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Mr. Philipose is a Development Engineer in the Prototype Engineering Section of AECL, and is responsible for the conceptual design and specification of prototype intrusion resistant underground disposal facilities. He has over 20 years experience in design, design supervision and project engineering, as well as extensive experience in the computer-aided stress analysis and design of structures.

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Ms. Ruddock is a system programmer in the Safety Studies Section and has assisted the Modelling Section. Her experience includes work on data base management, and applications programming for waste management.

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Mrs. Evraire has over 15 years' experience in office administration, including material purchasing, marketing research studies, education course administration, and for the last two years, has been senior secretary in the Waste Management Technology Division. Her experience and professional attitude contributed to the finalization of this report.

TABLE 1

Average Annual Radiation Doses Received by Individuals

in the U.S.* from Various Sources

	Effective Dose Equivalent mrem/year
Natural Sources	
Inhaled radon daughters	200**
Cosmic	30
Terrestrial	30
Internal natural radionuclides	40
Man-Made Sources	
Medical, dental X-rays	39
Nuclear medicine	14
Consumer products	9
All other sources (including occupational, fallout, nuclear fuel cycle)	< 3
ROUNDED TOTAL	~360

*From "Ionizing Radiation Exposure of the Population of the United States", (NCRP, 1987).

**"At a level of 4 picocurie/liter (indoor radon) people would receive about 7,700 mrem to the sensitive cells in the lung, or about 1,000 mrem whole body dose equivalent each year if they spent 75% of their time in the structure." Richard J. Guimond, Director of Radon Action Program, U.S. Environmental Protection Agency (Health Physics Society Newsletter, January 1988, Vol. XVI No. 1).

Radon measurements conducted in 2043 homes (year-round, single unit, owner occupied) throughout New York State by the New York State Energy Research and Development Authority found an average annual radon concentration of 1.13 picocurie/liter (range=0.0-38.3 pCi/l) on the first floor. This translates to an average annual whole body dose equivalent of 283 mrem. NYSERDA News Release, November 4, 1987.

TABLE 2

RADIONUCLIDE INVENTORY IN 30 YEARS ACCUMULATION OF WASTE

WASTE TYPE	A	B+C	A+(B+C)
TOTAL ACTIVITY (Ci)	5.87E+04	2.62E+06	2.68E+06
TOTAL VOLUME (m^3)	1.90E+05	2.74E+04	2.17E+05

NUCLIDE	HALF LIFE (Years)	LAMBDA (1/a)	ACTIVITY (Ci) Type A	ACTIVITY (Ci) Type B+C	INVENTORY (Nuclei) Type A	INVENTORY (Nuclei) Type B+C
Group 1						
H-3	1.23E+01	5.62E-02	2.98E+03	1.55E+06	6.20E+22	3.22E+25
Fe-55	2.68E+00	2.59E-01	2.62E+04	4.24E+05	1.18E+23	1.91E+24
Co-60	5.27E+00	1.32E-01	2.37E+04	4.32E+05	2.10F 23	3.83E+24
Sr-90	2.90E+01	2.39E-02	1.08E+02	3.31E+04	5.28E+21	1.62E+24
Pu-238	8.77E+01	7.90E-03	5.43E+00	2.44E+03	8.04E+20	3.61E+23
Pu-241	1.44E+01	4.81E-02	1.70E+02	6.48E+04	4.13E+21	1.58E+24
Cm-243	2.85E+01	2.43E-02	4.90E-03	8.44E-01	2.36E+17	4.06E+19
Cm-244	1.81E+01	3.83E-02	4.07E+00	7.68E+00	1.24E+20	2.34E+20
Group 2						
C-14	5.73E+03	1.21E-04	1.71E+02	5.01E+01	1.65E+24	4.84E+23
Tc-99	2.13E+05	3.25E-06	7.98E-02	1.13E+00	2.87E+22	4.06E+23
I-129	1.57E+07	4.33E-08	2.24E-01	1.90E+00	6.05E+24	5.13E+25
Group 3						
Ni-59	7.60E+04	9.12E-06	1.71E+01	3.06E+02	2.19E+24	3.92E+25
Ni-63	1.00E+02	6.93E-03	3.03E+03	4.31E+04	5.11E+23	7.27E+24
Nb-94	2.00E+04	3.47E-05	2.22E-01	4.21E+00	7.48E+21	1.42E+23
Cs-135	3.00E+06	2.31E-07	7.97E-02	1.13E+00	4.03E+23	5.72E+24
Cs-137	3.02E+01	2.30E-02	2.27E+03	4.86E+04	1.15E+23	2.47E+24
U-234*	2.44E+05	2.84E-06	1.53E+01	3.60E+00	6.29E+24	1.48E+24
U-235	7.04E+08	9.85E-10	5.74E-01	1.40E-01	6.81E+26	1.66E+26
U-238	4.47E+09	1.55E-10	2.15E+00	1.05E+00	1.62E+28	7.92E+27
Np-237	2.14E+06	3.24E-07	7.45E-07	2.44E-05	2.69E+18	8.80E+19
Pu-239/40	6.56E+03	1.06E-04	3.72E+00	3.23E+03	4.10E+22	3.56E+25
Pu-242	3.76E+05	1.84E-06	8.16E-03	7.05E+00	5.18E+21	4.48E+24
Am-241	4.32E+02	1.60E-03	3.42E+00	1.25E+04	2.50E+21	9.13E+24
Am-243	7.37E+03	9.40E-05	2.31E-01	7.34E-01	2.87E+21	9.03E+21

* U-234 is expected to be present in wastes containing U-235 but was not included in the source information used in NUREG-0782. An estimate of the U-234 inventory is included in this table, but was not used in the modelling.

TABLE 3

DOSE REDUCTION FACTORS

Allowance for delays in vault loadings and gaps between vault rows.

10 rows of vaults, with 10 m gaps. 3 year loading delay between rows.

	Dose Reduction Factor		
Nuclide	Province II	Province IV	Province V
H-3	0.224	0.480	0.236
C-14	0.959	0.996	0.963
Fe-55	0.100	0.100	0.100
Co-60	0.100	0.100	0.100
Ni-59	0.619	0.963	0.646
Ni-63	0.100	0.100	0.100
Sr-90	0.100	0.244	0.100
Nb-94	0.143	0.704	0.151
Tc-99	1.000	1.000	1.000
I-129	1.000	1.000	1.000
Cs-135	0.994	1.000	0.998
Cs-137	0.100	0.101	0.100
U-235	1.000	1.000	1.000
U-238	1.000	1.000	1.000
Np-237	0.987	0.999	0.988
Pu-238	0.100	0.100	0.100
Pu-239/40	0.107	0.482	0.109
Pu-241	0.100	0.100	0.100
Pu-242	0.812	0.985	0.828
Am-241	0.100	0.157	0.100
Am-243	0.175	0.776	0.188
Cm-243	0.100	0.100	0.100
Cm-244	0.100	0.100	0.100

TABLE 4

CONTAINER FAILURE FUNCTION FOR STEEL DRUMS
AS IMPLEMENTED FOR TIME-SCALES OF

3,000 years		24,000 years		300,000 years	
TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED
0	0.0	0	0.0	0	1.0
40	0.001	80	0.001		
80	0.06	160	0.8		
120	0.35	240	1.0		
160	0.80				
200	0.97				
240	1.0				

TABLE 5

ANNUAL INFILTRATION RATES (METERS PER YEAR)

	PROVINCE		
	II	IV	V
Precipitation	1.143	0.965	0.965
Evaporation + Transpiration	0.584	0.508	0.533
Runoff + Infiltration (R + I)	0.559	0.457	0.432
Infiltration	0.140	0.069	0.086
Infiltration as % of (R+I)	25.	15.	20.

TABLE 6

ROOF FAILURE FUNCTION FOR ABOVEGROUND VAULTS
AS IMPLEMENTED FOR TIME-SCALES OF

3,000 years		24,000 years		300,000 years	
TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED
0	0.0	0	0.0	0	0.0
200	0.001	240	0.001	1000	1.0
280	0.028	400	0.070		
360	0.052	560	0.170		
440	0.090	800	0.504		
520	0.136	1040	0.840		
600	0.212	1200	0.960		
800	0.504	1280	0.980		
1000	0.800	1440	1.0		
1080	0.884				
1160	0.940				
1240	0.972				
1320	0.988				
1480	1.0				

TABLE 7

ROOF FAILURE FUNCTION FOR BELOWGROUND VAULTS IN PROVINCE II
AS IMPLEMENTED FOR TIME-SCALES OF

3,000 years		24,000 years		300,000 years	
TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED
0	0.0	0	0.0	0	0.0
480	0.001	480	0.001	1000	0.50
560	0.012	560	0.012		
680	0.034	720	0.040		
800	0.070	800	0.070		
920	0.140	960	0.170		
1080	0.290	1080	0.290		
1200	0.400	1200	0.400		
1280	0.448	1280	0.448		
1400	0.480	1440	0.486		
1600	0.500	1600	0.500		

TABLE 8

ROOF FAILURE FUNCTION FOR BELOWGROUND VAULTS IN PROVINCES IV & V
AS IMPLEMENTED FOR TIME-SCALES OF

3,000 years		24,000 years		300,000 years	
TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED	TIME (Years)	FRACTION FAILED
0	0.0	0	0.0	0	0.0
480	0.001	480	0.001	1000	0.2
560	0.012	560	0.012	2000	0.75
680	0.030	800	0.052		
800	0.052	960	0.130		
920	0.100	1120	0.360		
1080	0.292	1200	0.500		
1200	0.500	1280	0.632		
1280	0.632	1360	0.696		
1360	0.696	1440	0.728		
1440	0.728	1520	0.750		
1600	0.750				

TABLE 9

FLOW RATE AND DIMENSION DATA (USING TRANSFER WIDTH OF 450 m)

	PROVINCE II	PROVINCE IV	PROVINCE V
LEACHING			
Waste Slab Thickness (m)	0.21	0.21	0.21
Effective Thickness of Water Layer for Leaching from Drums (m)	0.001	0.001	0.001
TRANSFER TO INFILTRATION WATERS			
Transfer Area (m^2)	1.495E+5	1.495E+5	1.495E+5
Transfer Length (m)	4.77	4.77	4.77
BACKFILL & BUFFER			
Effective Thickness of Backfill Over and Between Drums (m)	0.14	0.14	0.14
Buffer Thickness (m)	0.5	0.5	0.5
UNSATURATED ZONE			
Range of Thickness Given (m)	9 - 10	6 - 9	1.5 - 3
Thickness Chosen for Aboveground Scenario (m)	9.5	7.5	2.5
Depth of Vault Buried (from grade to bottom of buffer) for Belowground Scenario (m)	2.5	2.5	0.5
Thickness Chosen for Belowground Scenario (m)	7.0	5.0	2.0
LOWER BEDROCK			
Path Length (m)	95.	385.	250.
Velocity (m/y)	9.00E-05	3.00E-07	3.00E-07
Effective Cross-Section Area (m^2)	9000.	9000.	9000.
Porosity	0.005	0.005	0.005
UPPER BEDROCK			
Path Length (m)	30.	30.	30.
Velocity (m/y)	9.00E-03	9.00E-03	9.00E-04
Effective Cross-Section Area (m^2)	9000.	9000.	9000.
Porosity	0.01	0.01	0.01
AQUIFER			
Path Length (m)	300.	300.	300.
Depth (m)	10.	10.	5.
Effective Cross-Section Area (m^2)	4500.	4500.	2250.
Flow rate (m^3/y) (see Table 10 for velocity)	9000.	90000.	4500.

TABLE 10

HYDRAULIC CONDUCTIVITIES, HYDRAULIC GRADIENTS, HENCE FLOW VELOCITIES IN AQUIFERS

	Province II	Province IV	Province V
Conductivity Range (m/y)	1.E-1 -- 3.E+3	1.E+0 -- 3.E+3	1.E-1 -- 3.E+3
Gradient Range	1.E-4 -- 1.E-2	1.E-3 -- 1.E-1	1.E-4 -- 1.E-2
Velocity Range (m/y)	1.E-5 -- 3.E+1	1.E-3 -- 3.E+2	1.E-5 -- 3.E+1
Velocity Chosen (m/y)	2.0	20.0	2.0

TABLE 11

PORE-WATER DIFFUSION COEFFICIENTS (METERS SQUARED PER YEAR) AND RETARDATION FACTORS

NUCLIDE	PORE-WATER DIFF. CO. (Sat.)	(Unsat.)	RETARDATION FACTORS (Cmprsd.)	(Solid)	(Buffer)	(Ground)
Group 1						
H-3	7.69E-02	1.54E-02	1.0	1.0	1.0	1.0
Fe-55	5.87E-02	1.17E-02	1.33E+04	1.33E+03	2.00E+03	2.64E+03
Co-60	5.87E-02	1.17E-02	7.88E+03	1.33E+03	1.00E+03	1.75E+03
Sr-90	6.12E-02	1.22E-02	8.22E+03	1.39E+03	8.50E+01	3.60E+01
Pu-238	5.93E-02	1.19E-02	7.71E+03	7.71E+03	3.52E+03	3.52E+03
Pu-241	5.93E-02	1.19E-02	7.71E+03	7.71E+03	3.52E+03	3.52E+03
Cm-243	5.58E-02	1.12E-02	1.52E+04	3.05E+03	1.20E+03	1.20E+03
Cm-244	5.58E-02	1.12E-02	1.52E+04	3.05E+03	1.20E+03	1.20E+03
Group 2						
C-14	3.84E-02	7.68E-03	4.99E+01	5.99E+05	1.00E+01	1.00E+01
Tc-99	4.76E-02	9.52E-03	1.0	1.0	4.0	4.0
I-129	4.76E-02	9.52E-03	1.0	1.0	4.0	4.0
Group 3						
Ni-59	5.87E-02	1.17E-02	7.88E+03	1.33E+03	1.00E+03	1.75E+03
Ni-63	5.87E-02	1.17E-02	7.88E+03	1.33E+03	1.00E+03	1.75E+03
Nb-94	4.76E-02	9.52E-03	1.30E+05	1.30E+05	4.64E+03	4.64E+03
Cs-135	5.40E-02	1.08E-02	3.63E+02	4.71E+02	4.35E+03	3.50E+02
Cs-137	5.40E-02	1.08E-02	3.63E+02	4.71E+02	4.35E+03	3.50E+02
U-235	5.93E-02	1.19E-02	7.71E+03	7.71E+03	3.52E+03	3.52E+03
U-238	5.93E-02	1.19E-02	7.71E+03	7.71E+03	3.52E+03	3.52E+03
Np-237	5.88E-02	1.18E-02	1.61E+04	3.21E+03	1.20E+03	1.20E+03
Pu-239/40	5.93E-02	1.19E-02	7.71E+03	7.71E+03	3.52E+03	3.52E+03
Pu-242	5.93E-02	1.19E-02	7.71E+03	7.71E+03	3.52E+03	3.52E+03
Am-241	5.58E-02	1.12E-02	1.52E+04	3.05E+03	1.20E+03	1.20E+03
Am-243	5.58E-02	1.12E-02	1.52E+04	3.05E+03	1.20E+03	1.20E+03

TABLE 12

POROSITIES AND TORTUOSITIES

MATERIAL	POROSITY	TORTUOSITY
Compacted Waste	0.50	5.0
Solid Waste (concreted)	0.15	15.8
Buffer (Sand + 10% Illite)	0.33	1.53
Overburden		
Province 2	0.5	1.8
Province 4	0.35	1.6
Province 5	0.45	1.7

TABLE 13

RADIONUCLIDE RETARDATION IN UPPER AND LOWER BEDROCK

	UPPER BEDROCK			LOWER BEDROCK		
NUCLIDE	Prov. II	Prov. IV	Prov. V	Prov. II	Prov. IV	Prov. V
Group 1						
H-3	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Fe-55	2.64E+03	2.64E+03	2.64E+03	2.64E+03	2.64E+03	2.64E+03
Co-60	1.75E+03	1.75E+03	1.75E+03	1.75E+03	1.75E+03	1.75E+03
Sr-90	5.41E+02	3.00E+04	3.20E+04	1.08E+03	6.75E+04	8.41E+02
Pu-238	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03
Pu-241	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03
Cm-243	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03
Cm-244	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03
Group 2						
C-14	2.71E+02	1.20E+05	1.30E+05	5.41E+02	2.70E+05	4.21E+02
Tc-99	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
I-129	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Group 3						
Ni-59	1.75E+03	1.75E+03	1.75E+03	1.75E+03	1.75E+03	1.75E+03
Ni-63	1.75E+03	1.75E+03	1.75E+03	1.75E+03	1.75E+03	1.75E+03
Nb-94	4.64E+03	4.64E+03	4.64E+03	4.64E+03	4.64E+03	4.64E+03
Cs-135	2.70E+04	2.40E+05	2.60E+05	5.40E+04	5.40E+05	4.20E+03
Cs-137	2.07E+04	2.40E+05	2.60E+05	5.40E+04	5.40E+05	4.20E+03
U-235	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03
U-238	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03
Np-237	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03
Pu-239/40	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03
Pu-242	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03	3.52E+03
Am-241	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03
Am-243	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03	1.20E+03

TABLE 14

FLOW/PUMPING RATES
(CUBIC METERS PER YEAR)

CRITICAL FAMILY	:	3.00E+03
POPULATION WELL	:	2.00E+05
SURFACE STREAM	:	4.45E+06

TABLE 15

DILUTION FACTORS COMPARED WITH CRITICAL FAMILY WELL

	PROVINCE		
	II	IV	V
POPULATION WELL ALONE	22.	2.2	44.
SURFACE WATER ALONE	500.	50.	1000.

TABLE 16a

PATHWAY DOSE CONVERSION FACTORS - GROUNDWATER

(mrem/y)/(nuclei/cubic meter)

NUCLIDE	EDE	LNG	LVR	STO	RBM	LLI	BOS	KID	THY
Group 1									
H-3	3.94E-14	0.00E+00	3.94E-14	3.94E-14	3.94E-14	0.00E+00	3.94E-14	3.94E-14	3.94E-14
Fe-55	2.34E-13	2.61E-17	4.52E-13	1.34E-13	1.34E-13	2.74E-13	1.34E-13	1.34E-13	1.34E-13
Co-60	2.24E-11	1.42E-12	1.78E-11	8.54E-12	8.54E-12	7.12E-12	8.54E-12	8.54E-12	8.54E-12
Sr-90	6.13E-12	2.71E-19	3.60E-14	3.60E-14	2.78E-11	4.68E-12	6.66E-11	3.60E-14	3.60E-14
Pu-238	3.45E-11	2.14E-17	1.17E-10	0.00E+00	5.23E-11	2.29E-12	6.46E-10	0.00E+00	0.00E+00
Pu-241	4.18E-12	9.85E-17	1.21E-11	0.00E+00	6.16E-12	6.64E-14	7.62E-11	0.00E+00	0.00E+00
Cm-243	9.02E-11	8.93E-15	3.11E-10	0.00E+00	1.24E-10	8.32E-12	1.62E-09	0.00E+00	0.00E+00
Cm-244	1.06E-10	8.02E-17	3.90E-10	0.00E+00	1.51E-10	1.17E-11	1.88E-09	0.00E+00	0.00E+00

Abbreviations in Tables 16 & 17

EDE : Effective Committed Dose Rate Equivalent

For committed dose rate equivalent for organs:

LNG - lung; LVR - liver; STO - stomach wall; RBM - red bone marrow; LLI - lower large intestine; BOS - bone surface; KID - kidney; THY - thyroid.

Values are shown as 0.00E+00 if data were not available.

TABLE 16b

PATHWAY DOSE CONVERSION FACTORS - GROUNDWATER

(mrem/y)/(nuclei/cubic meter)

NUCLIDE	EDE	LNG	LVR	STO	RBM	LLI	BOS	KID	THY
Group 2									
C-14	1.01E-17	6.86E-20	1.01E-17	1.01E-17	1.01E-17	1.01E-17	1.01E-17	1.01E-17	1.01E-17
Tc-99	1.08E-17	0.00E+00	9.12E-19	9.12E-19	9.12E-19	1.66E-17	9.12E-19	2.32E-17	1.21E-16
I-129	2.52E-17	8.54E-21	2.94E-20	2.94E-20	2.94E-20	4.72E-22	2.94E-20	2.94E-20	8.24E-16
Group 3									
Ni-59	3.22E-18	8.75E-24	1.86E-16	1.86E-18	1.86E-18	1.18E-17	1.86E-18	4.68E-21	1.86E-18
Ni-63	6.14E-15	0.00E+00	3.27E-15	3.27E-15	3.27E-15	3.19E-14	3.27E-15	8.45E-18	3.27E-15
Nb-94	9.48E-15	1.09E-15	7.39E-16	7.39E-16	4.92E-15	3.29E-15	8.22E-15	6.03E-15	7.39E-16
Cs-135	2.72E-18	0.00E+00	2.72E-18	2.72E-18	2.72E-18	0.00E+00	2.72E-18	2.72E-18	2.72E-18
Cs-137	3.84E-12	2.77E-16	1.81E-12	1.81E-12	1.81E-12	1.94E-16	1.81E-12	1.81E-12	1.81E-12
U-235	4.47E-19	3.16E-22	1.27E-20	1.27E-20	3.30E-19	2.44E-19	5.08E-18	2.19E-18	1.27E-20
U-238	6.09E-20	1.93E-24	1.84E-21	1.84E-21	4.90E-20	3.54E-20	8.02E-19	3.29E-19	1.84E-21
Np-237	1.99E-15	2.51E-19	1.62E-15	0.00E+00	3.67E-15	8.83E-17	4.50E-14	0.00E+00	0.00E+00
Pu-239 /40	1.41E-13	4.10E-20	4.55E-13	0.00E+00	2.04E-13	7.77E-15	2.64E-12	0.00E+00	0.00E+00
Pu-242	8.59E-15	7.41E-21	2.82E-14	0.00E+00	1.22E-14	4.80E-16	1.60E-13	0.00E+00	0.00E+00

TABLE 17a

PATHWAY DOSE CONVERSION FACTORS - SURFACE WATER

(mrem/y)/(nuclei/cubic meter)

NUCLIDE	EDE	LNG	LVR	STO	RBM	LLI	BOS	KID	THY
<u>Group 1</u>									
H-3	2.71E-17	0.00E+00	1.96E-17	1.96E-17	1.96E-17	0.00E+00	1.96E-17	1.96E-17	1.96E-17
Fe-55	8.05E-14	8.52E-18	1.48E-13	4.42E-14	4.42E-14	9.02E-14	4.42E-14	4.42E-14	4.42E-14
Co-60	1.00E-10	4.56E-12	5.71E-11	2.76E-11	2.76E-11	2.29E-11	2.76E-11	2.76E-11	2.76E-11
Sr-90	6.76E-14	2.97E-21	3.97E-16	3.97E-16	3.17E-13	5.16E-14	7.34E-13	3.97E-16	3.97E-16
Pu-238	5.91E-12	3.65E-18	2.99E-11	0.00E+00	8.97E-12	3.94E-13	1.10E-10	0.00E+00	0.00E+00
Pu-241	7.16E-13	1.68E-17	2.06E-12	0.00E+00	1.05E-12	1.14E-14	1.31E-11	0.00E+00	0.00E+00
Cm-243	3.06E-12	3.06E-16	1.06E-11	0.00E+00	4.23E-12	2.84E-13	5.51E-11	0.00E+00	0.00E+00
Cm-244	3.60E-12	2.72E-18	1.34E-11	0.00E+00	5.13E-12	4.00E-13	6.41E-11	0.00E+00	0.00E+00
<u>Group2</u>									
C-14	1.37E-15	9.28E-18	1.37E-15	1.37E-15	1.37E-15	1.37E-15	1.37E-15	1.37E-15	1.37E-15
Tc-99	1.10E-18	0.00E+00	9.37E-20	9.37E-20	9.37E-20	1.69E-18	9.37E-20	2.37E-18	1.24E-17
I-129	3.27E-18	1.12E-21	3.85E-21	3.85E-21	3.85E-21	6.16E-23	3.85E-21	3.85E-21	1.08E-16

TABLE 17b

PATHWAY DOS CONVERSION FACTORS - SURFACE WATER

(mrem/y)/(nuclei/cubic meter)

NUCLIDE	EDE	LNG	LVR	STO	RBM	LLI	BOS	KID	THY
Group 3									
Ni-59	9.47E-18	2.74E-23	5.80E-18	5.80E-18	5.80E-18	3.69E-17	5.80E-18	1.47E-20	5.80E-18
Ni-63	1.93E-14	0.00E+00	1.02E-14	1.02E-14	1.02E-14	1.00E-13	1.02E-14	2.66E-17	1.02E-14
Nb-94	2.59E-15	2.37E-16	1.60E-16	1.60E-16	1.07E-15	7.14E-16	1.79E-15	1.31E-15	1.60E-16
Cs-135	3.99E-17	0.00E+00	3.99E-17	3.99E-17	3.99E-17	0.00E+00	3.99E-17	3.99E-17	3.99E-17
Cs-137	3.20E-11	4.17E-15	2.71E-11	2.71E-11	2.71E.11	2.92E-15	2.71E-11	2.71E.11	2.71E-11
U-235	2.74E-20	2.13E-23	8.59E-22	8.59E-22	2.23E-20	1.65E-20	3.44E-19	1.48E-19	3.59E-22
U-238	4.12E-21	1.30E-25	1.24E-22	1.24E-22	3.30E-21	2.37E-21	5.41E-20	2.23E-20	1.24E-22
Np-237	6.78E-16	8.47E-20	5.48E-16	0.00E+00	1.24E-15	2.99E-17	1.52E-14	0.00E+00	0.00E+00
Pu-239/40	2.41E-14	7.01E-21	7.75E-14	0.00E+00	3.50E-14	1.33E-15	4.50E-13	0.00E+00	0.00E+00
Pu-242	1.46E-15	1.27E-21	4.82E-15	0.00E+00	2.09E-15	8.17E-17	2.74E-14	0.00E+00	0.00E+00
Am-241	2.77E-12	7.27E-17	9.20E-12	0.00E+00	3.92E-12	1.62E-13	5.03E-11	0.00E+00	0.00E+00
AM-243	1.60E-13	3.09E-17	5.23E-13	0.00E+00	2.29E-13	9.81E-15	2.94E-12	0.00E+00	0.00E+00

TABLE 18

CONSUMPTION RATES BY ANIMALS

	Food (kg/day)	Water (L/day)	Contaminated Water Fraction
Dairy Cow	1.000E+01	8.000E+01	1.000E+00
Beef Steer	1.000E+01	5.000E+01	1.000E+00
Chicken	1.000E-01	3.000E-01	1.000E+00
Pig	3.000E+00	7.000E+00	1.000E+00

TABLE 19

AVERAGE FOOD CONSUMPTION BY ADULT HUMANS

	Amount (kg/y)	Contaminated Fraction
Leaf Vegetables	1.400E+01	1.000E+00
Fruits & Vegetables	1.100E+02	1.000E+00
Roots	7.900E+01	1.000E+00
Cereal	7.400E+01	1.000E+00
Meat (Beef and Pork)	7.100E+01	1.000E+00
Poultry	1.600E+01	1.000E+00
Milk	1.200E+02	1.000E+00
Milk Prod	5.400E+01	1.000E+00
Eggs	1.400E+01	1.000E+00
Fish (Fresh Water)	5.500E+00	1.000E+00
Fish (Marine)	1.000E+01	0.000E+00
Crustacea	1.500E+00	0.000E+00
Mollusk	1.500E+00	0.000E+00
Seaweed (e.g. Dulse)	3.000E+00	0.000E+00

TABLE 20

GEOGRAPHIC DATA AND HUMAN HABITS

Ca-conc. to get Sr-conc. in fish (g/m^3)	5.000E+01
K-conc. to get Cs-conc. in fish (g/m^3)	1.500E+00
Physical removal constant by sedimentation (/s)	5.730E-07
Eff. dry soil density (kg/m^2)	2.400E+02
Eff. sediment density (kg/m^2)	4.000E+01
Dose reduction for ground surface factor nonuniformity	7.000E-01
Fraction time spent outdoors	2.000E-01
Inhalation rate (m^3/y)	8.400E+03
Intake water (L/y)	7.000E+02
Contamination fraction of human drinking water	1.000E+00
Fraction of meat eaten that is beef	6.600E-01
Fraction of meat eaten that is pork	3.400E-01
Irrigation, annual growing season rate ($L/(m^2.s)$)	2.300E-05
Shoreline occupancy factor	1.000E-02
Water occupancy factor	1.000E-02
Removal constant from plants (/s)	5.730E-07
Fraction left after removal by water treatment	1.000E+00
Ground shield factor by buildings for gamma radiation	4.000E-01
Physical removal constant from soil (/s)	2.200E-10
Shore width factor	2.000E-01

TABLE 21

PEAK ANNUAL EFFECTIVE DOSE EQUIVALENTS
FOR COMBINATIONS OF PROVINCE & TECHNOLOGIES
(mrem/year)

PROVINCE	ABOVEGROUND VAULT	BELOWGROUND VAULT	UNDERGROUND MINED REPOSITORY
II	20	9	15
IV	1	2	0.007
V	39	50	0.01

TABLE 22

MAXIMUM ANNUAL COMMITTED DOSE EQUIVALENTS TO THYROID (mrem/year)

PROVINCE	ABOVEGROUND VAULT	BELOWGROUND VAULT	UNDERGROUND MINED REPOSITORY
II	630	290	470
IV	31	59	0.2
V	1[illegible]00	1600	4

TABLE 23

MAXIMUM ANNUAL COMMITTED DOSE EQUIVALENTS TO
ANY OTHER ORGAN (KIDNEY) (mrem/year)

PROVINCE	ABOVEGROUND VAULT	BELOWGROUND VAULT	UNDERGROUND MINED REPOSITORY
II	0.7	0.3	0.1
IV	0.1	0.007	0.0
V	1	0.009	0.0

TABLE 24

ABOVEGROUND VAULT IN PROVINCE II

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

	GROUP 1 NUCLIDES		GROUP 2 NUCLIDES	
Eff. Dose Equiv.	3.27 -06 (310)	(1)	1.96 +01 (1680)	(3)
Lungs	< 1.0 -06	(1)	9.55 -03 (2080)	(4)
Stomach Wall	3.27 -06 (310)	(1)	6.01 -01 (2240)	(2)
L. L. Intestine	< 1.0 -06	(1)	6.55 -01 (2240)	(2)
Kidney	3.27 -06 (310)	(1)	7.04 -01 (2240)	(2)
Liver	3.27 -06 (310)	(1)	6.01 -01 (2240)	(2)
Red Bone Marrow	3.27 -06 (310)	(1)	6.01 -01 (2240)	(2)
Bone Surface	3.27 -06 (310)	(1)	6.01 -01 (2240)	(2)
Thyroid	3.27 -06 (310)	(1)	6.33 +02 (1680)	(3)

Major Contributors to Dose

(1) H-3
(2) C-14
(3) I-129
(4) C-14 + I-129

TABLE 25

ABOVEGROUND VAULT IN PROVINCE IV

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

	GROUP 1 NUCLIDES		GROUP 2 NUCLIDES	
Eff. Dose Equiv.	6.46 -06 (270)	(1)	1.04 +00 (1440)	(3)
Lungs	< 1.0 -06	(1)	9.47 -04 (1600)	(4)
Stomach Wall	6.46 -06 (270)	(1)	9.71 -02 (1680)	(2)
L. L. Intestine	< 1.0 -06	(1)	1.00 -01 (1680)	(2)
Kidney	6.46 -06 (270)	(1)	1.03 -01 (1680)	(2)
Liver	6.46 -06 (270)	(1)	9.71 -02 (1680)	(2)
Red Bone Marrow	6.46 -06 (270)	(1)	9.71 -02 (1680)	(2)
Bone Surface	6.46 -06 (270)	(1)	9.71 -02 (1680)	(2)
Thyroid	6.46 -06 (270)	(1)	3.14 +01 (1360)	(3)

Major Contributors to Dose

(1) H-3
(2) C-14
(3) I-129
(4) C-14 + I-129

TABLE 26

ABOVEGROUND VAULT IN PROVINCE V

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

	GROUP 1 NUCLIDES		GROUP 2 NUCLIDES	
Eff. Dose Equiv.	1.05 -03 (230)	(1)	3.93 +01 (2000)	(3)
Lungs	< 1.0 -06	(1)	1.75 -02 (2000)	(4)
Stomach Wall	1.05 -03 (230)	(1)	7.10 -01 (2000)	(2)
L. L. Intestine	< 1.0 -06	(1)	8.63 -01 (2000)	(2)
Kidney	1.05 -03 (230)	(1)	9.90 -01 (2000)	(2)
Liver	1.05 -03 (230)	(1)	7.10 -01 (2000)	(2)
Red Bone Marrow	1.05 -03 (230)	(1)	7.10 -01 (2000)	(2)
Bone Surface	1.05 -03 (230)	(1)	7.10 -01 (2000)	(2)
Thyroid	1.05 -03 (230)	(1)	1.26 +03 (2000)	(3)

Major Contributors to Dose

(1) H-3
(2) C-14
(3) I-129
(4) C-14 + I-129

TABLE 27

BELOWGROUND VAULT IN PROVINCE II

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

	GROUP 1 NUCLIDES		GROUP 2 NUCLIDES	
Eff. Dose Equiv.	< 1.0 -06	(1)	8.90 +00 (1840)	(3)
Lungs	< 1.0 -06	(1)	4.47 -03 (2240)	(4)
Stomach Wall	< 1.0 -06	(1)	2.79 -01 (2320)	(2)
L. L. Intestine	< 1.0 -06	(1)	3.04 -01 (2320)	(2)
Kidney	< 1.0 -06	(1)	3.27 -01 (2320)	(2)
Liver	< 1.0 -06	(1)	2.79 -01 (2320)	(2)
Red Bone Marrow	< 1.0 -06	(1)	2.79 -01 (2320)	(2)
Bone Surface	< 1.0 -06	(1)	2.79 -01 (2320)	(2)
Thyroid	< 1.0 -06	(1)	2.87 +02 (1840)	(3)

Major Contributor to Dose

(1) H-3
(2) C-14
(3) I-129
(4) C-14 + I-129

TABLE 28

BELOWGROUND VAULT IN PROVINCE IV

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

	GROUP 1 NUCLIDES		GROUP 2 NUCLIDES	
Eff. Dose Equiv.	3.33 -06 (230)	(1)	1.84 +00 (7840)	(3)
Lungs	< 1.0 -06	(5)	7.68E -04 (6800)	(4)
Stomach Wall	3.33 -06 (230)	(1)	6.74 -02 (1760)	(2)
L. L. Intestine	< 1.0 -06 (230)	(5)	6.99 -02 (1760)	(2)
Kidney	3.33 -06	(1)	7.21 -02 (1760)	(2)
Liver	3.33 -06 (230)	(1)	6.74 -02 (1760)	(2)
Red Bone Marrow	3.33 -06 (230)	(1)	6.74 -02 (1760)	(2)
Bone Surface	3.33 -06 (230)	(1)	6.74 -02 (1760)	(2)
Thyroid	3.33 -06 (230)	(1)	5.93 +01 (7920)	(3)

Major Contributors to Dose

(1) H-3
(2) C-14
(3) I-129
(4) C-14 + I-129
(5) Sr-90

TABLE 29

BELOWGROUND VAULT IN PROVINCE V

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

	GROUP 1 NUCLIDES		GROUP 2 NUCLIDES	
Eff. Dose Equiv.	5.47 -03 (220)	(1)	4.96 +01 (2160)	(3)
Lungs	< 1.0 -06	(1)	1.99 -02 (2160)	(4)
Stomach	5.47 -03 (220)	(1)	5.69 -01 (2240)	(2)
L. L. Intestine	< 1.0 -06	(1)	7.65 -01 (2240)	(2)
Kidney	5.47 -03 (220)	(1)	9.28 -01 (2240)	(2)
Liver	5.47 -03 (220)	(1)	5.69 -01 (2240)	(2)
Red Bone Marrow	5.47 -03 (220)	(1)	5.69 -01 (2240)	(2)
Bone Surface	5.47 -03 (220)	(1)	5.69 -01 (2240)	(2)
Thyroid	5.47 -03 (220)	(1)	1.60 +03 (2080)	(3)

Major Contributors to Dose

(1) H-3
(2) C-14
(3) I-129
(4) C-14 + I-129

TABLE 30

MINED REPOSITORY IN PROVINCE II

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

GROUP 2 NUCLIDES*

Eff. Dose Equiv.	14.5 (14600)
Lungs	4.90 -03 (14600)
Stomach Wall	2.07 -02 (14600)
L. L. Intestine	6.94 -02 (14600)
Kidney	1.14 -01 (14600)
Liver	2.07 -02 (14600)
Red Bone Marrow	2.07 -02 (14600)
Bone Surface	2.07 -02 (14600)
Thyroid	473.3 (14600)

*I-129 is the only significant contributor.

TABLE 31

MINED REPOSITORY IN PROVINCE IV

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

GROUP 2 NUCLIDES*

Eff. Dose Equiv.	6.57 -03 (1,400,000)
Lungs	2.22 -06 (1,400,000)
Stomach Wall	7.67 -06 (1,400,000)
L. L. Intestine	2.72 -06 (540,000)
Kidney	8.60 -06 (890,000)
Liver	7.67 -06 (1,400,000)
Red Bone Marrow	9.05 -09 (1,400,000)
Bone Surface	7.67 -06 (1,400,000)
Thyroid	2.14 -01 (1,400,000)

*I-129 is the only significant contributor.

TABLE 32

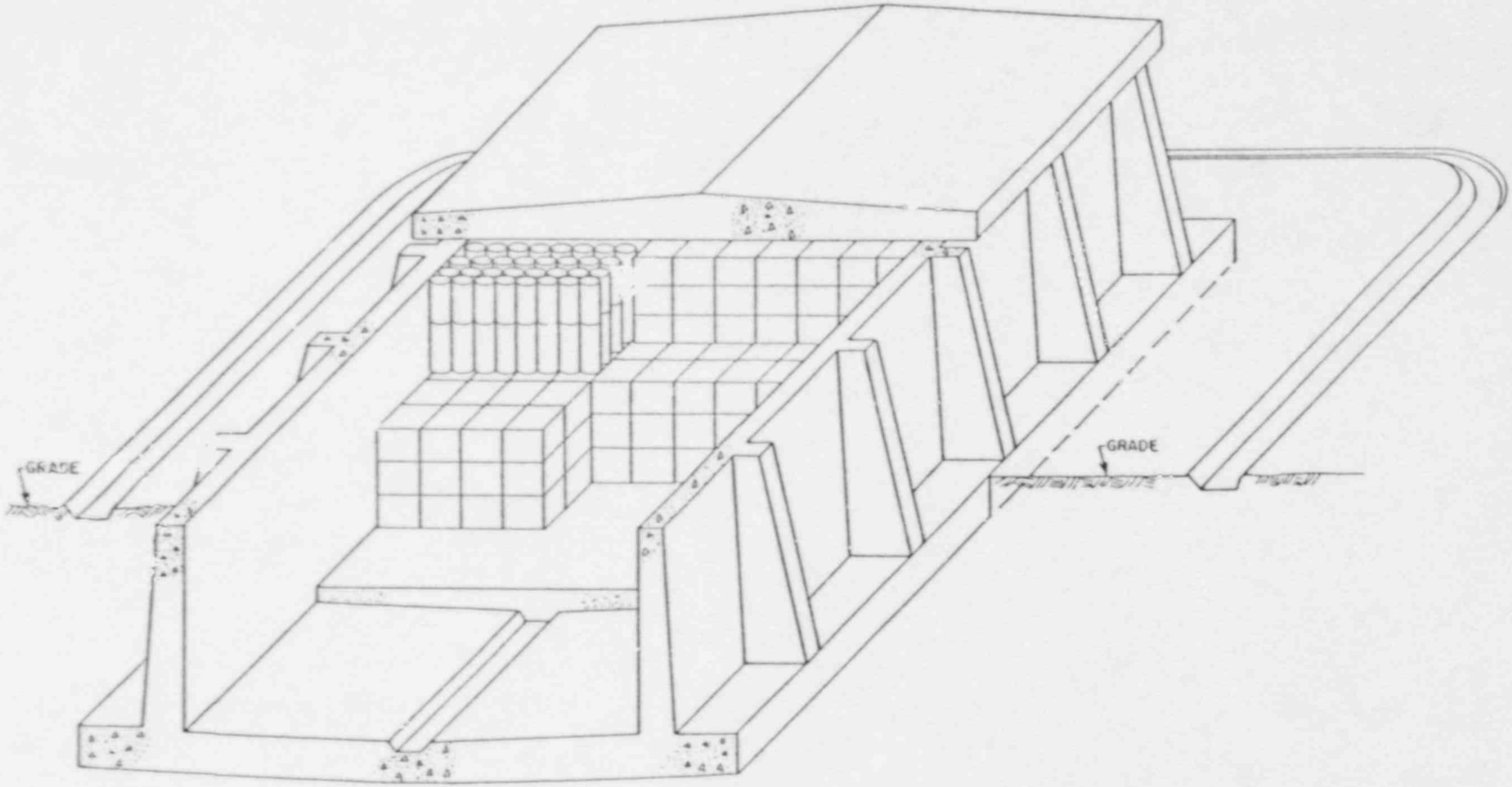
MINED REPOSITORY IN PROVINCE V

DOSE RATES TO ADULTS IN CRITICAL FAMILY
Maximum dose rates in mrem/y, with year of occurrence in parenthesis

GROUP 2 NUCLIDES

Eff. Dose Equiv.	1.20 -01 (1,100,000)
Lungs	4.05 -05 (1,100,000)
Stomach Wall	1.41 -04 (1,100,000)
L. L. Intestine	9.37 -05 (400,000)
Kidney	2.21 -04 (530,000)
Liver	1.41 -04 (1,100,000)
Red Bone Marrow	1.41 -04 (1,100,000)
Bone Surface	1.41 -04 (1,100,000)
Thyroid	3.92 (1,100,000)

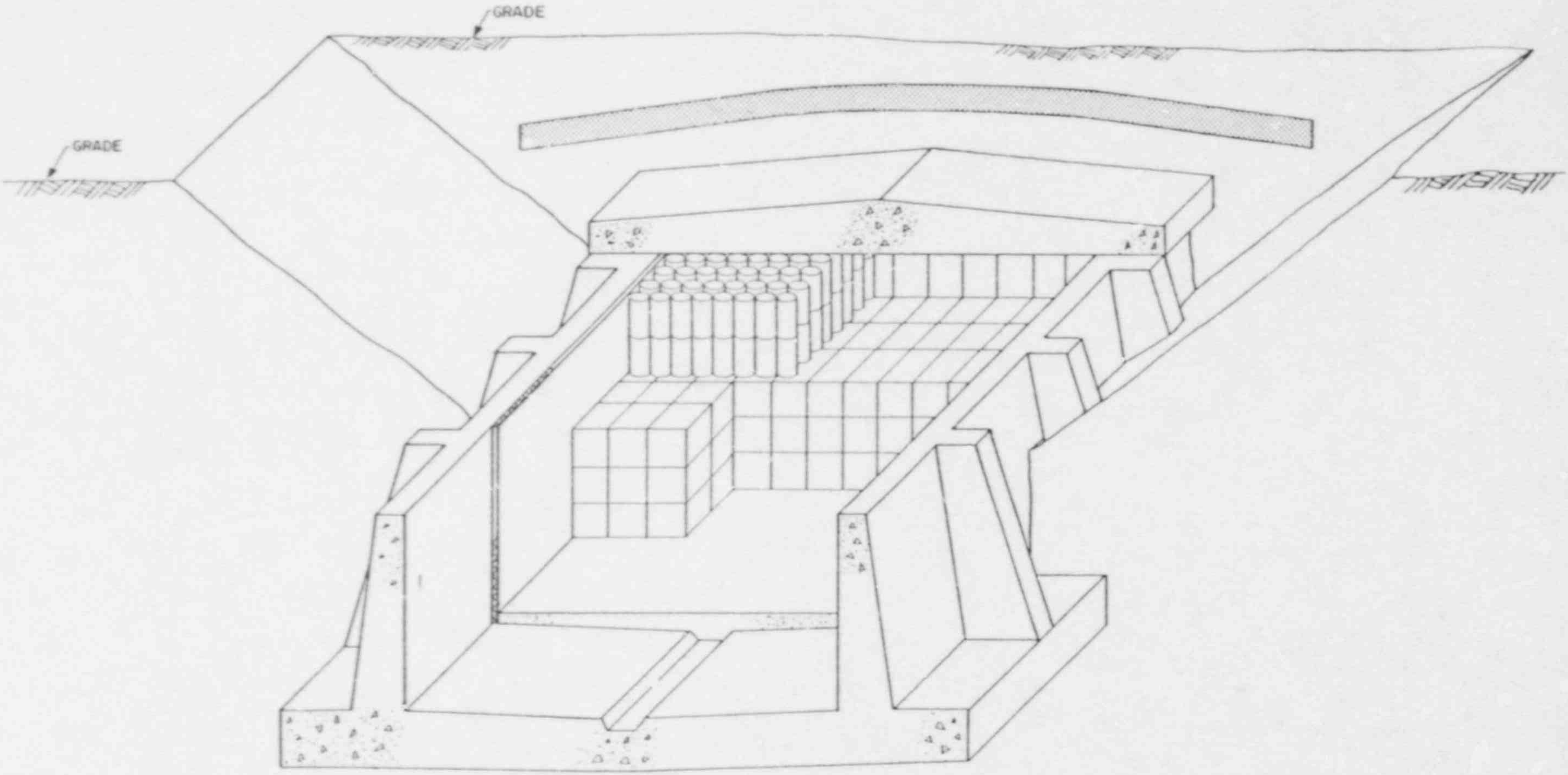
*I-129 is the only significant contributor.



ABOVEGROUND VAULT-CONCEPTUAL DESIGN

FIGURE 1

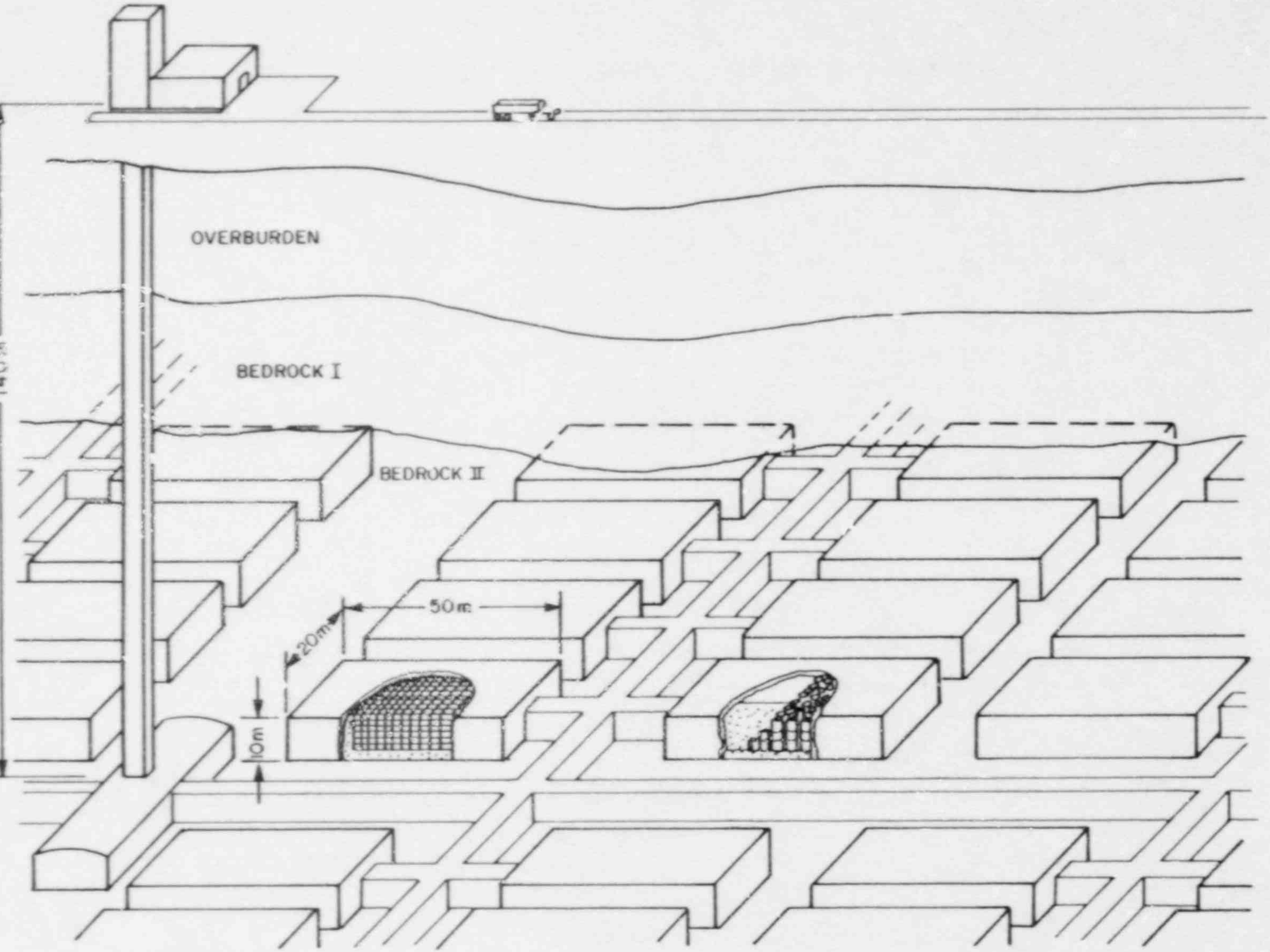
ACRES



BELOWGROUND VAULT-CONCEPTUAL DESIGN

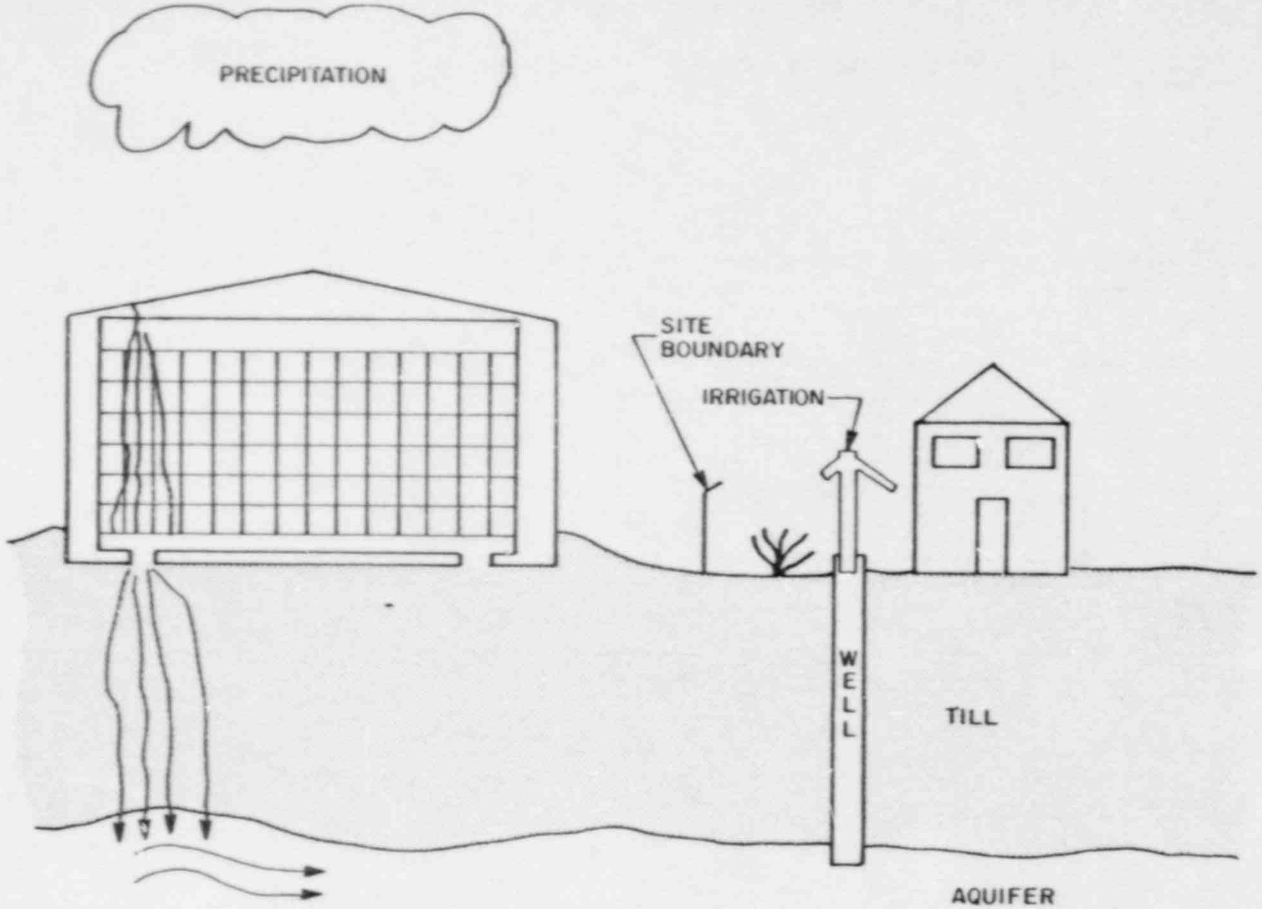
FIGURE 2

ACRES



MINED REPOSITORY - CONCEPTUAL DESIGN

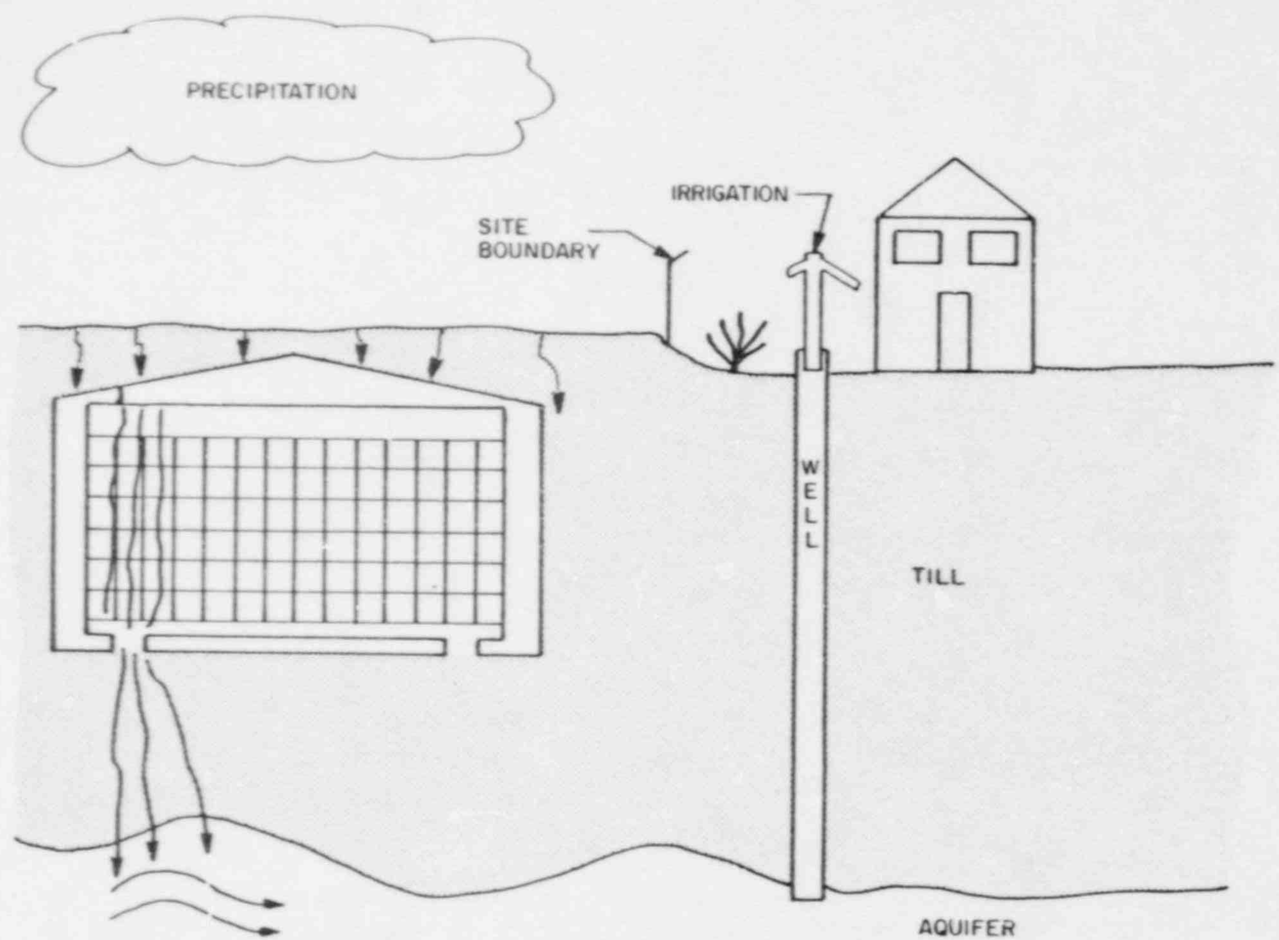
FIGURE 3



SCHEMATIC OF SCENARIO FOR ABOVEGROUND VAULT

FIGURE 4

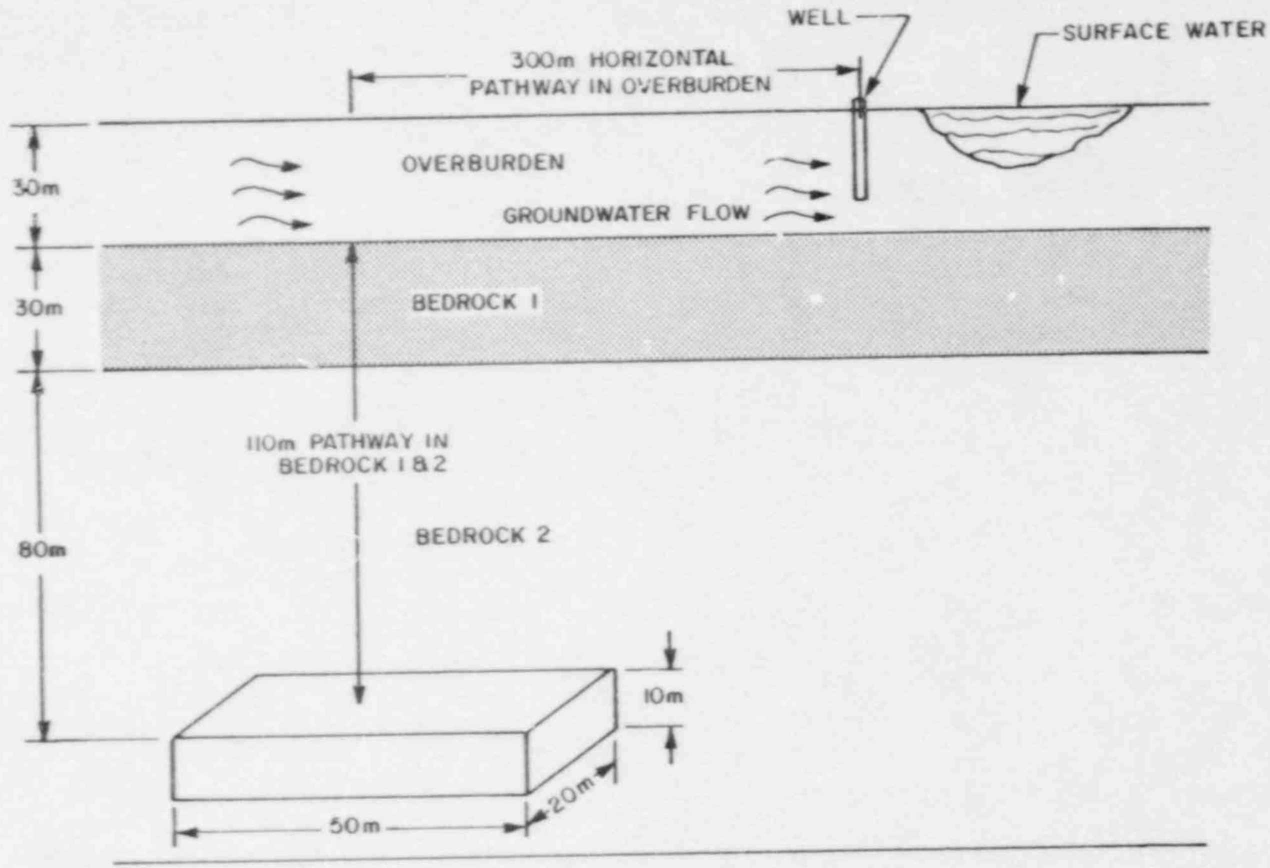
ACRES



SCHEMATIC OF SCENARIO FOR BELOWGROUND VAULT

FIGURE 5

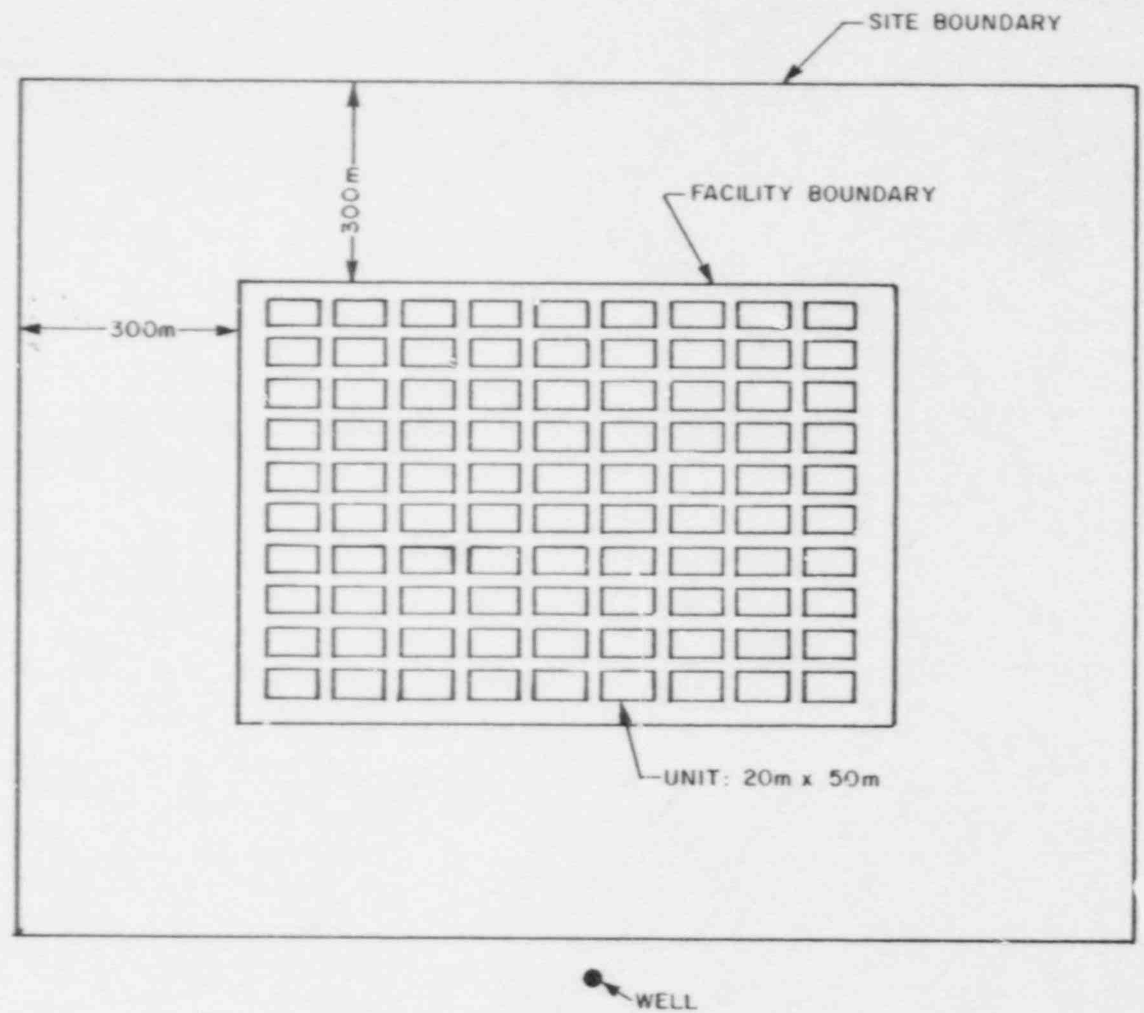
ACRES



SCENARIO PLAN FOR MINED REPOSITORY
IN PROVINCE II

FIGURE 6

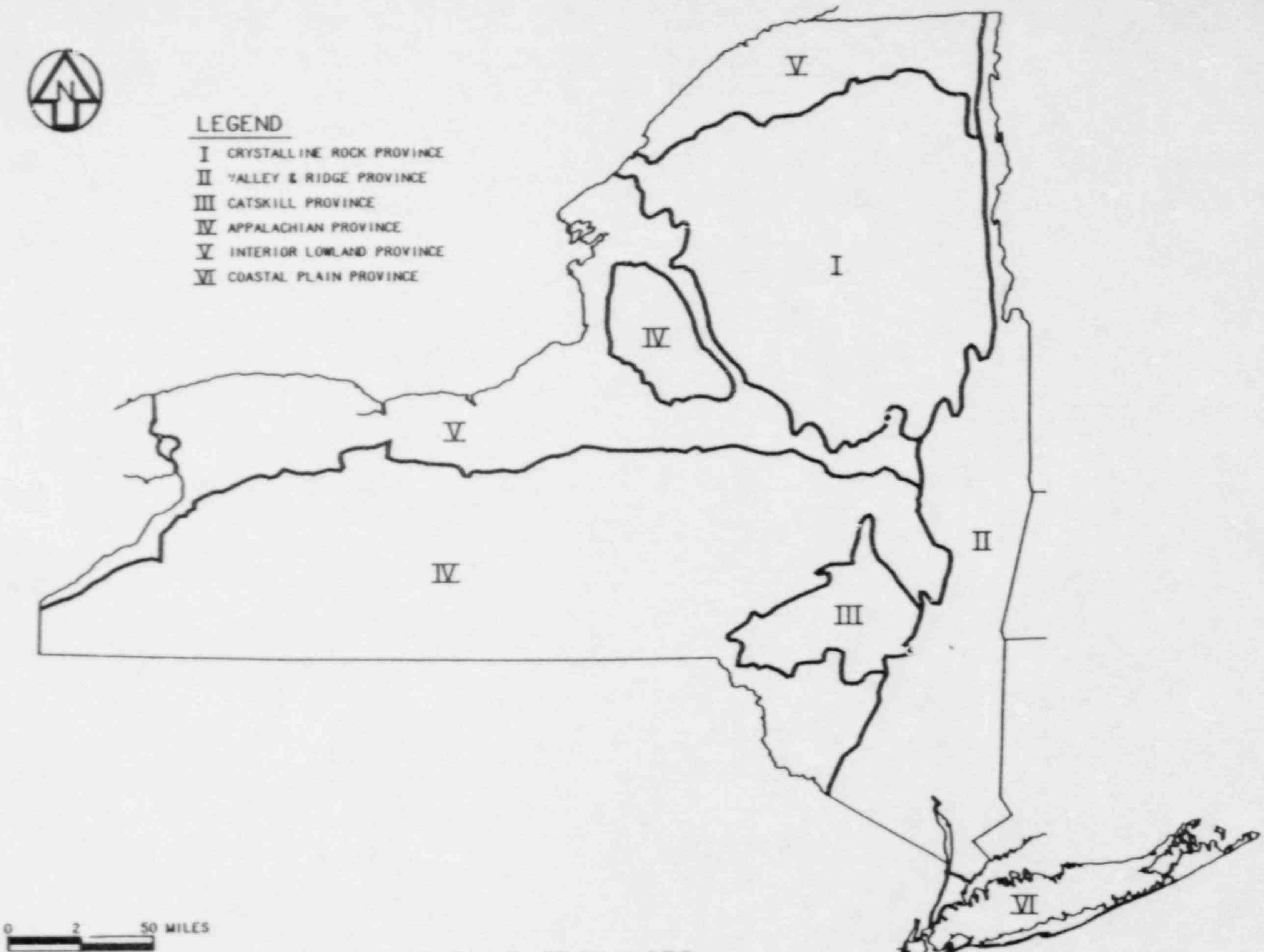
ACRES



SITE PLAN ASSUMED FOR EXPOSURE SCENARIOS
FOR NEAR-SURFACE REPOSITORIES

FIGURE 7

AGRES



PHYSIOGRAPHIC PROVINCES

SOURCE: ADAPTED FROM THOMPSON, 1966

FIGURE 8

ACRES

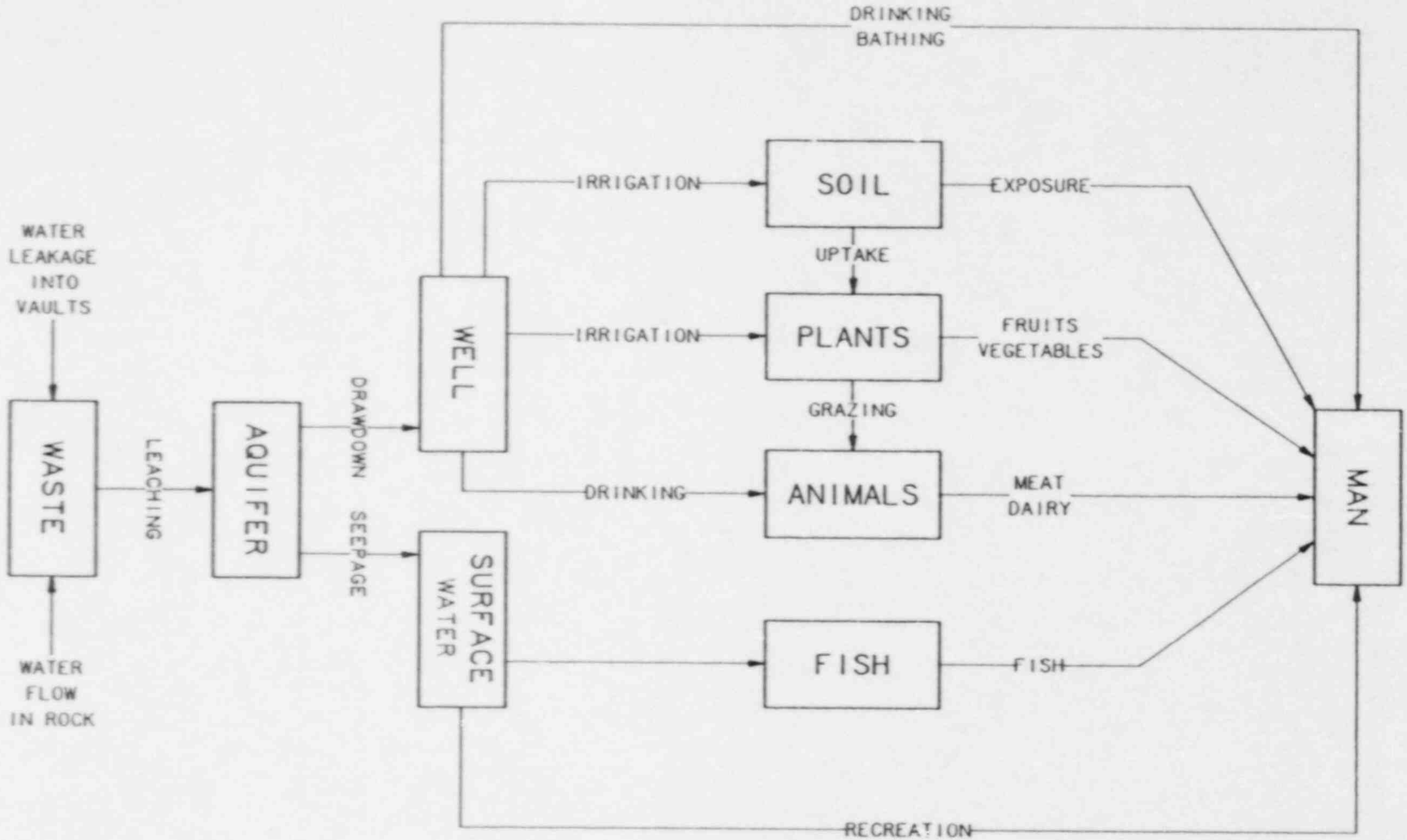
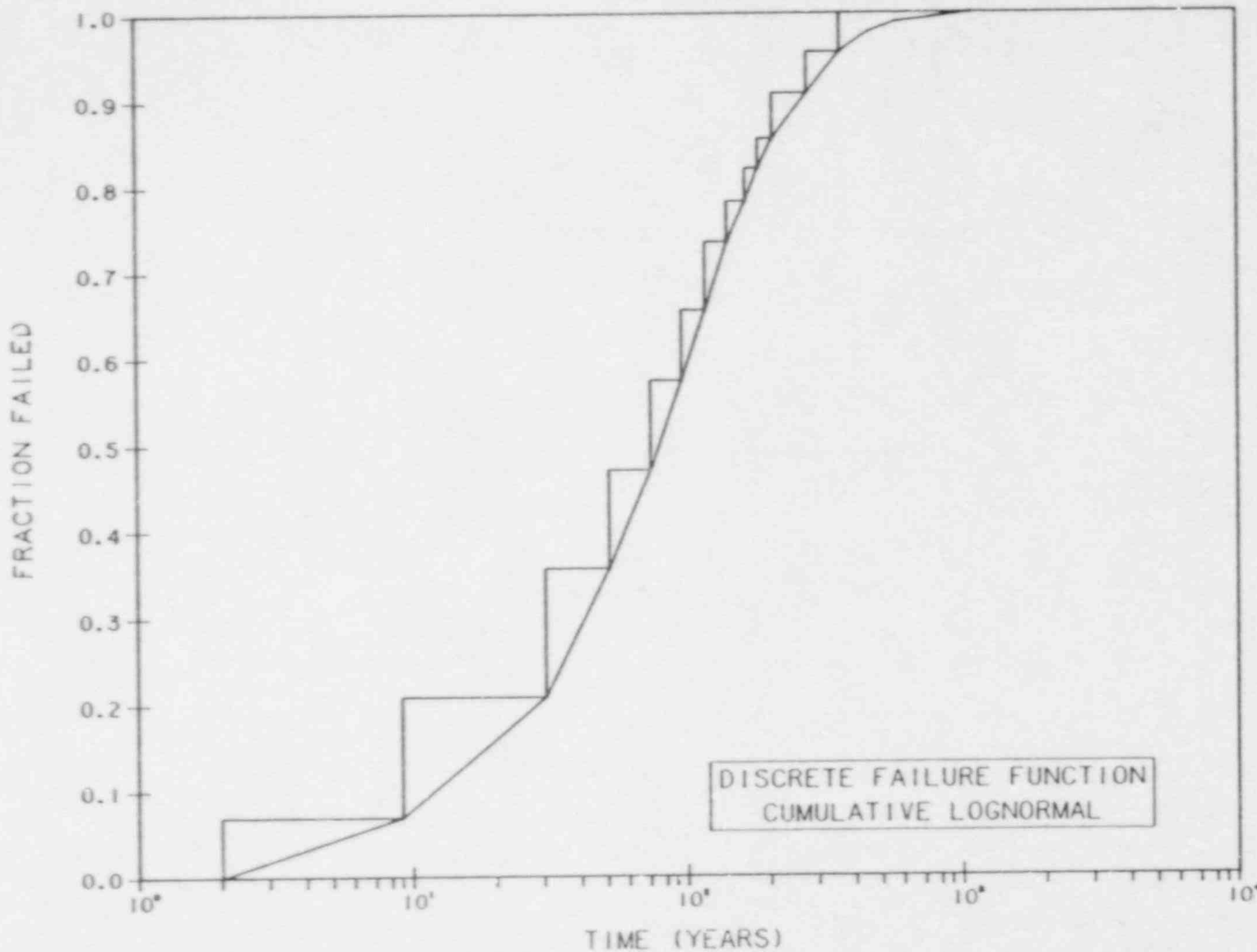


FIGURE 9

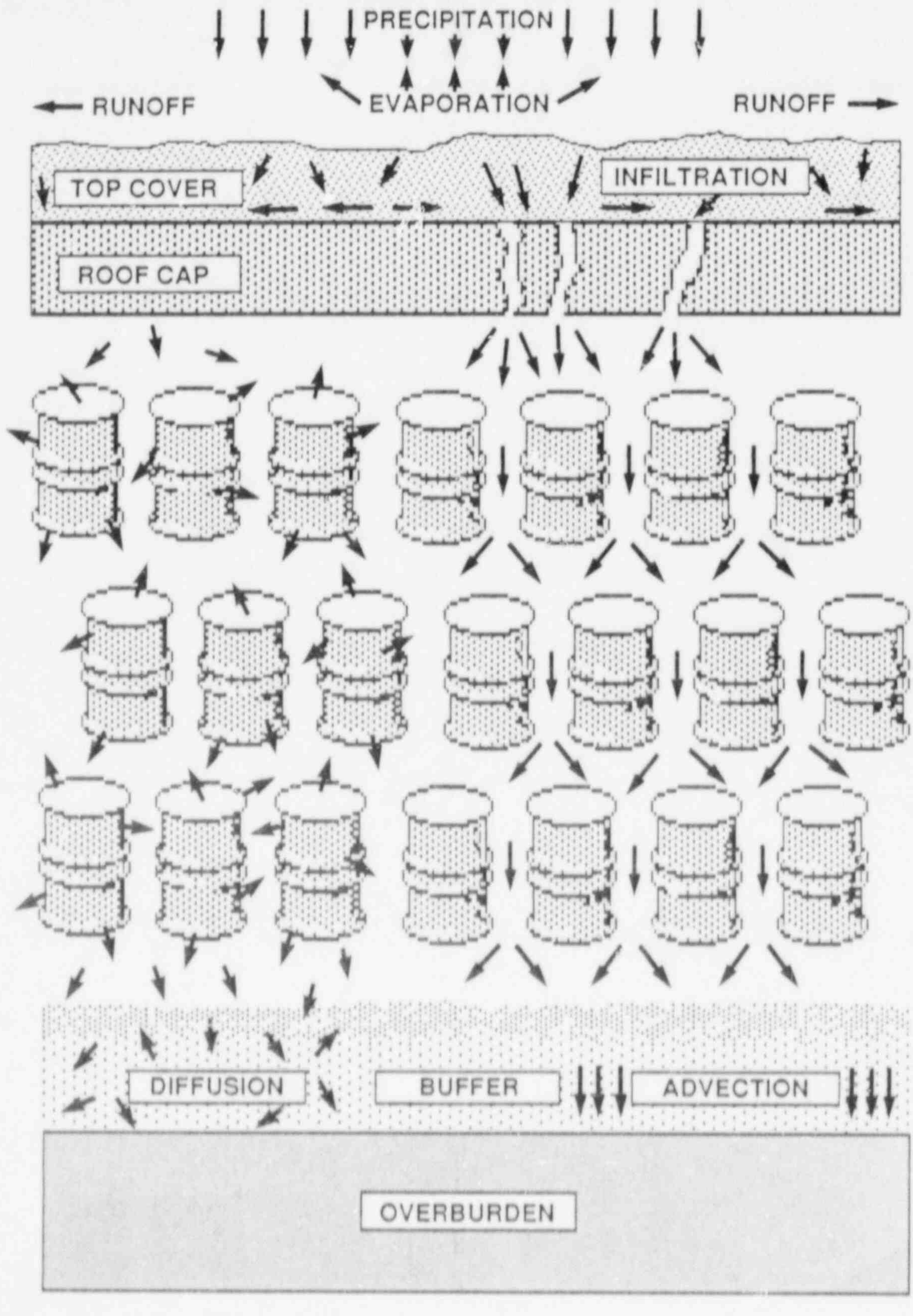
SCHEMATIC DIAGRAM OF MODELLED
NUCLIDE MIGRATION PATHWAYS



TYPICAL CONTAINER FAILURE FUNCTION

FIGURE 10

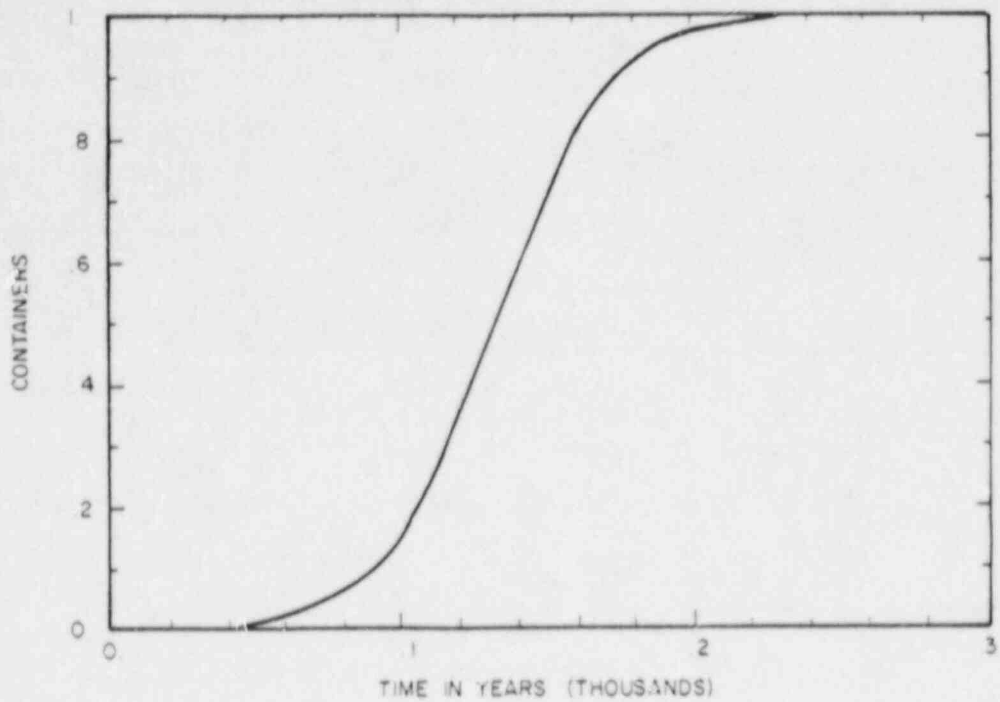
ACRES

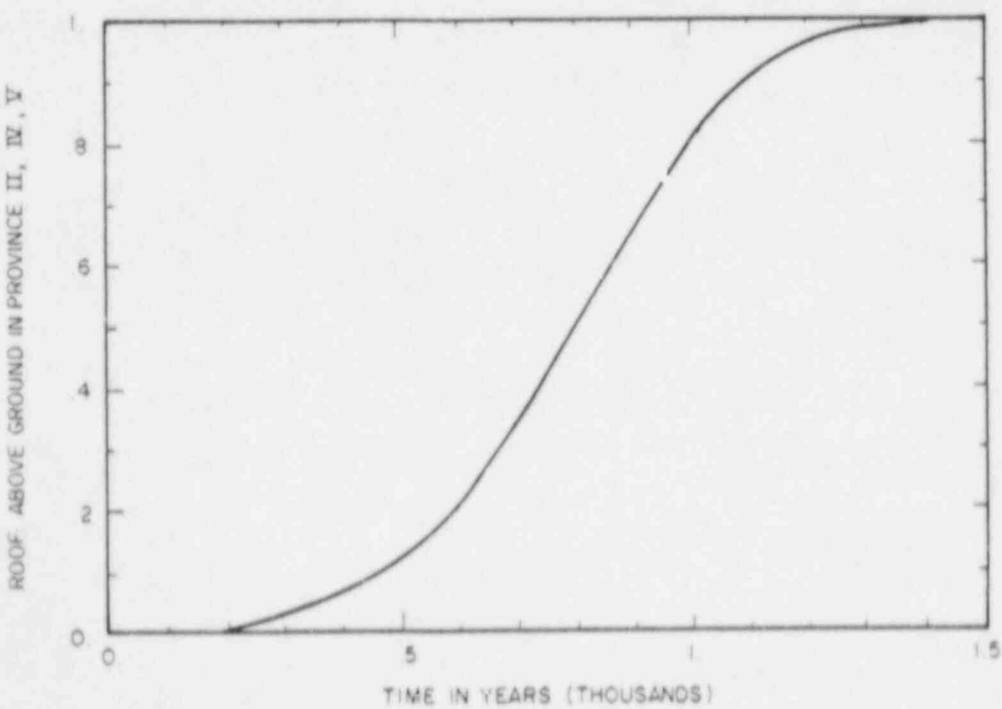


DIFFUSION AND ADVECTION IN THE VAULT

ACRES

FIGURE 11

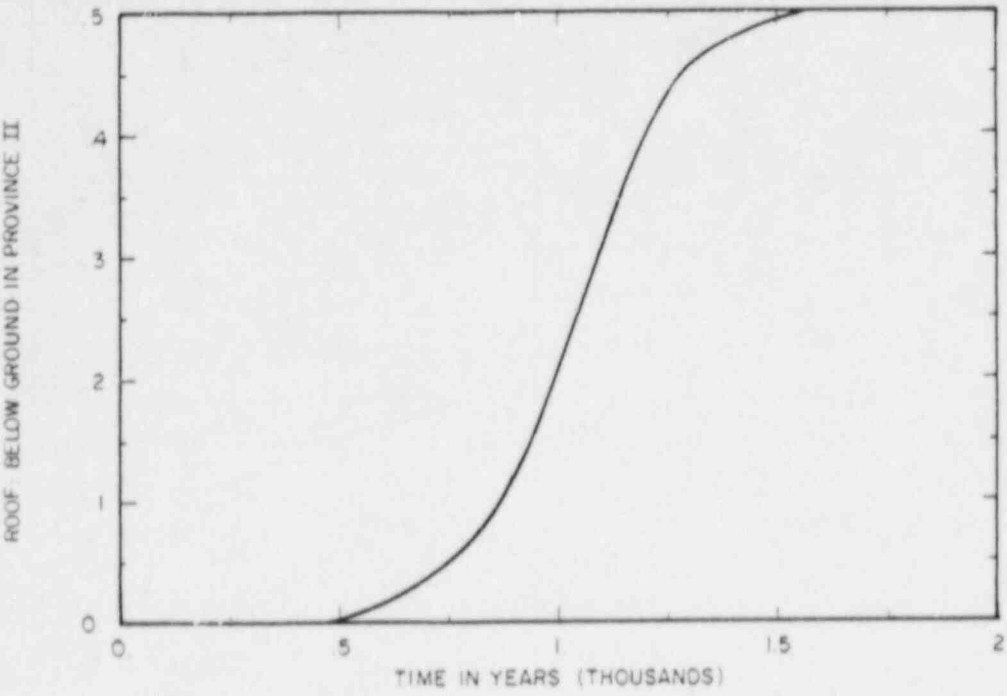


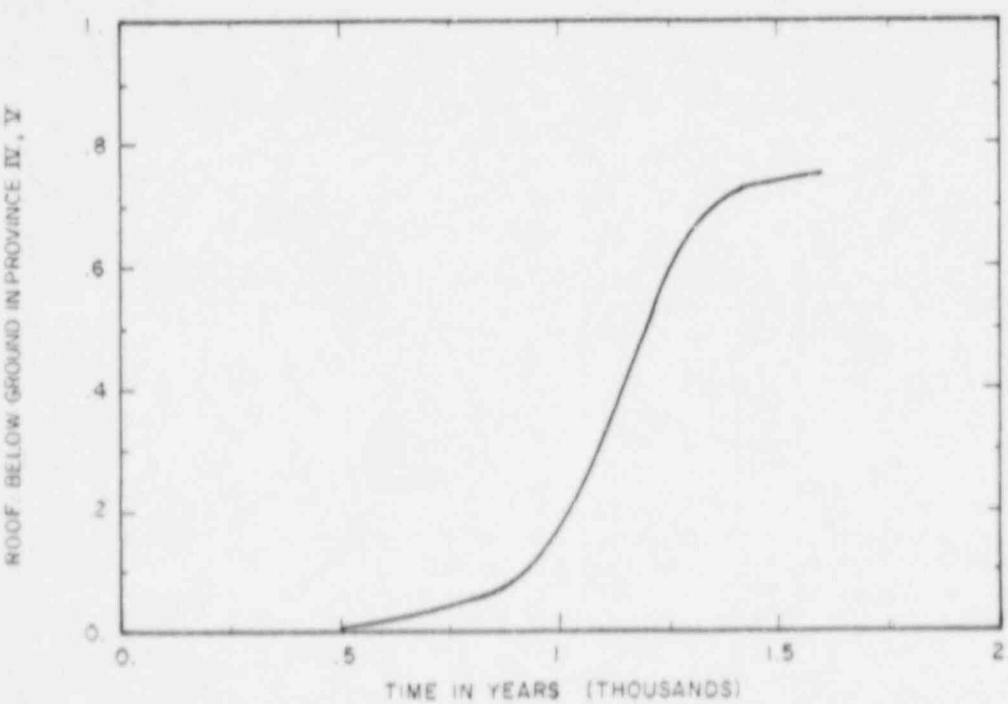


FAILURE FUNCTION FOR CONTAINERS AND FAILURE FUNCTION FOR ROOF OF ABOVEGROUND VAULTS IN PROVINCES II, IV AND V

ACRES

FIGURE 12

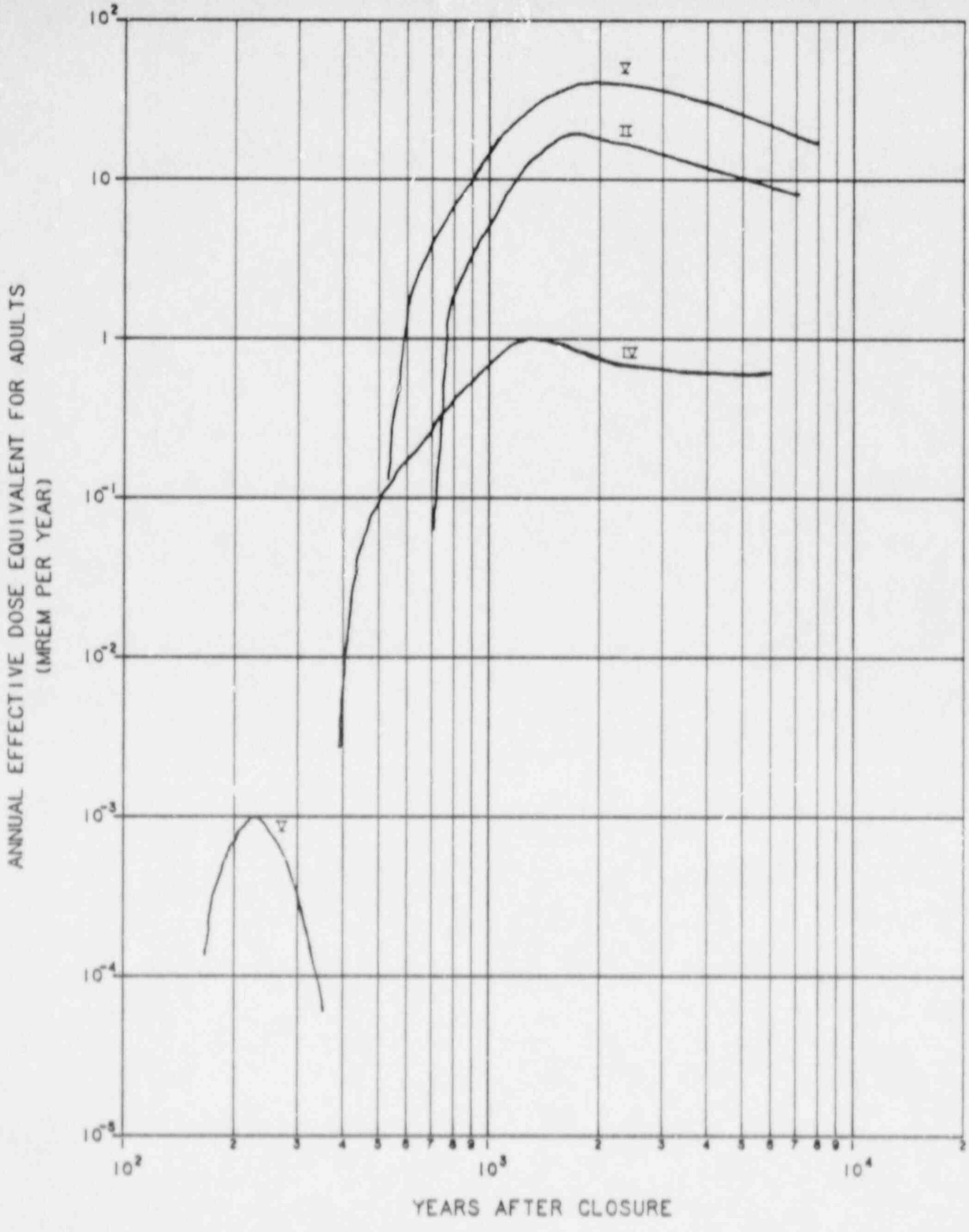




FAILURE FUNCTIONS FOR ROOF OF BELOWGROUND VAULT
(a) IN PROVINCE II, (b) IN PROVINCES IV AND V

ACRES

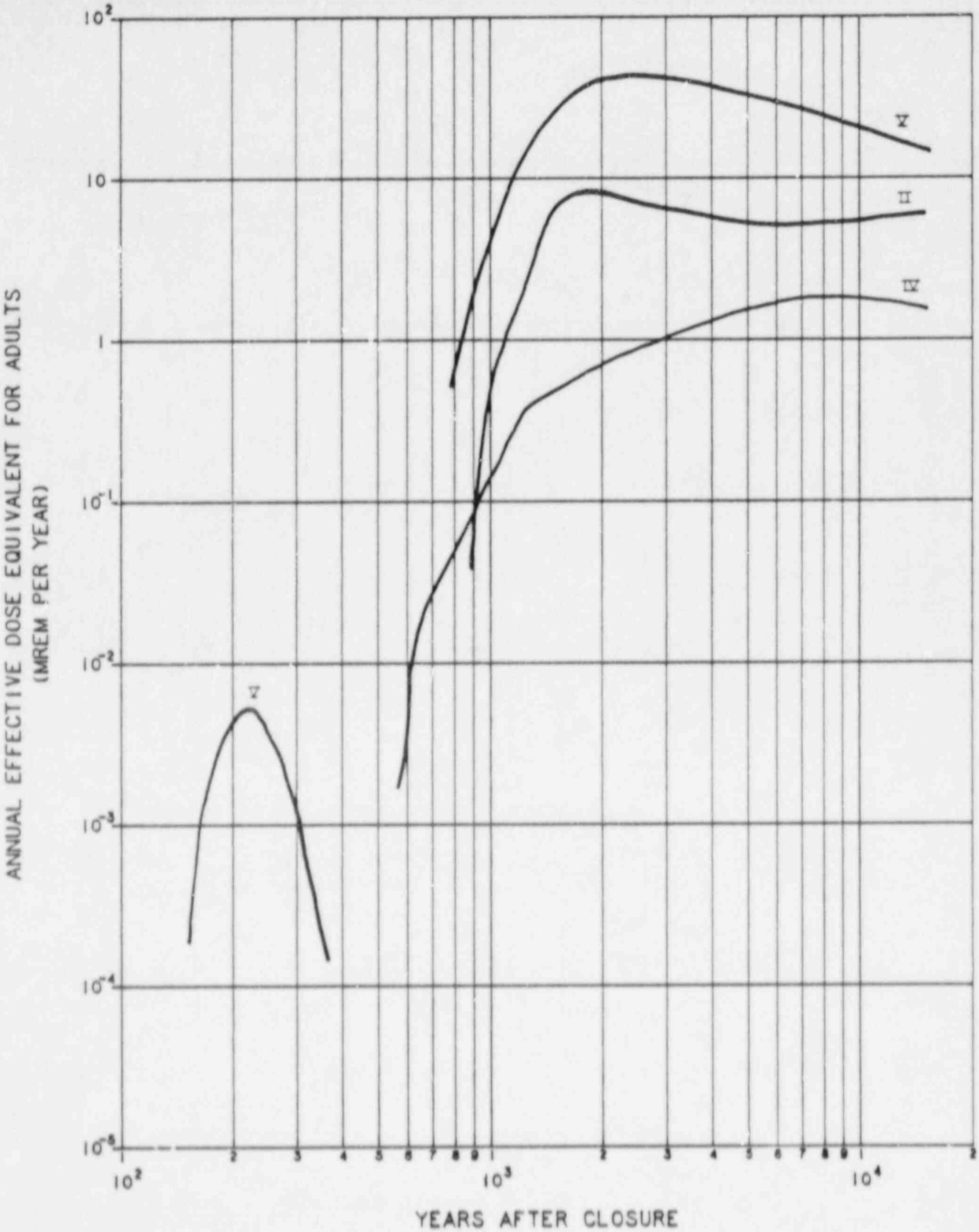
FIGURE 13



COMPARISON OF ABOVEGROUND VAULTS IN PROVINCES II, IV, V

ACRES

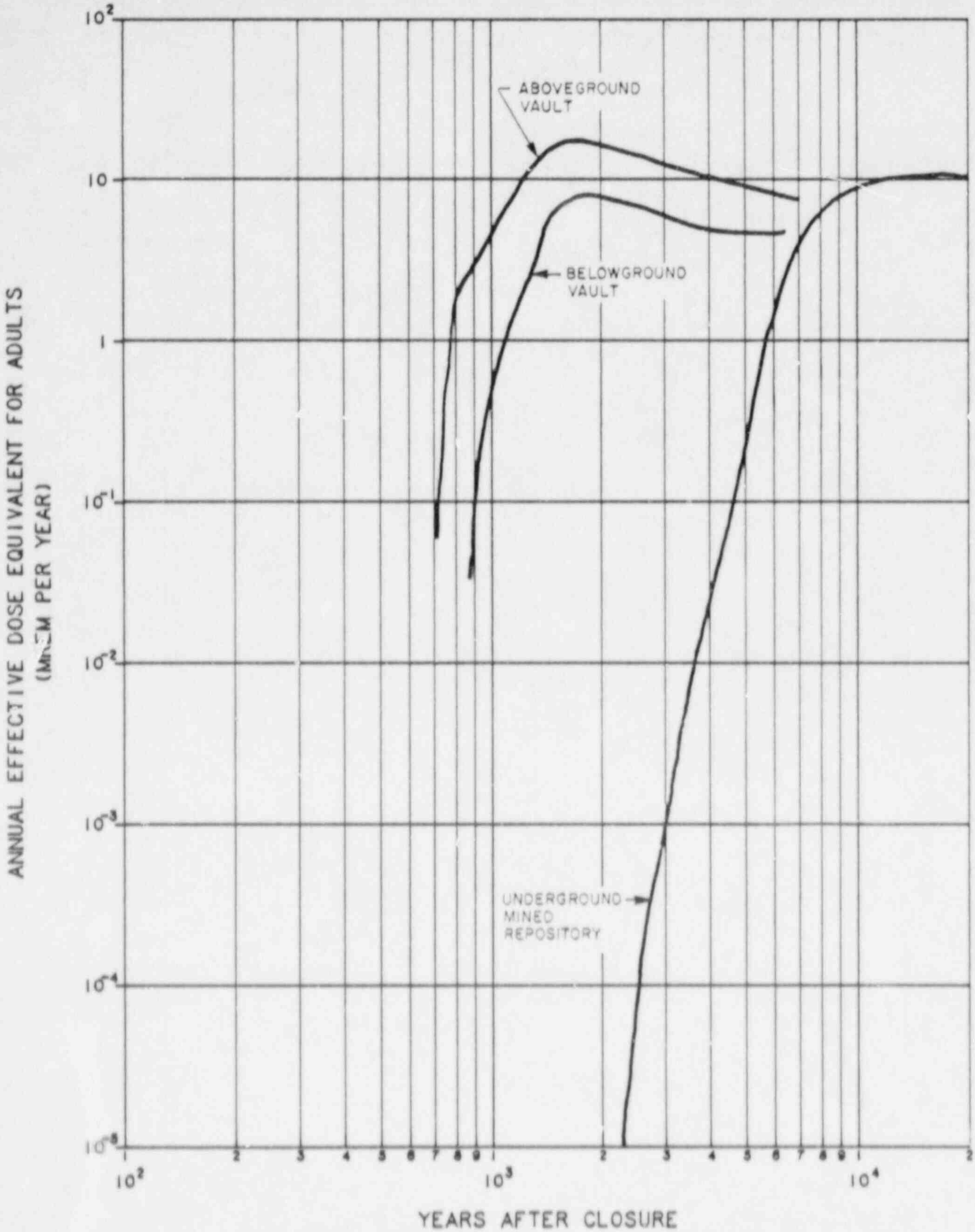
FIGURE 14



COMPARISON OF BELOWGROUND VAULTS IN PROVINCES II, IV, V

ACRES

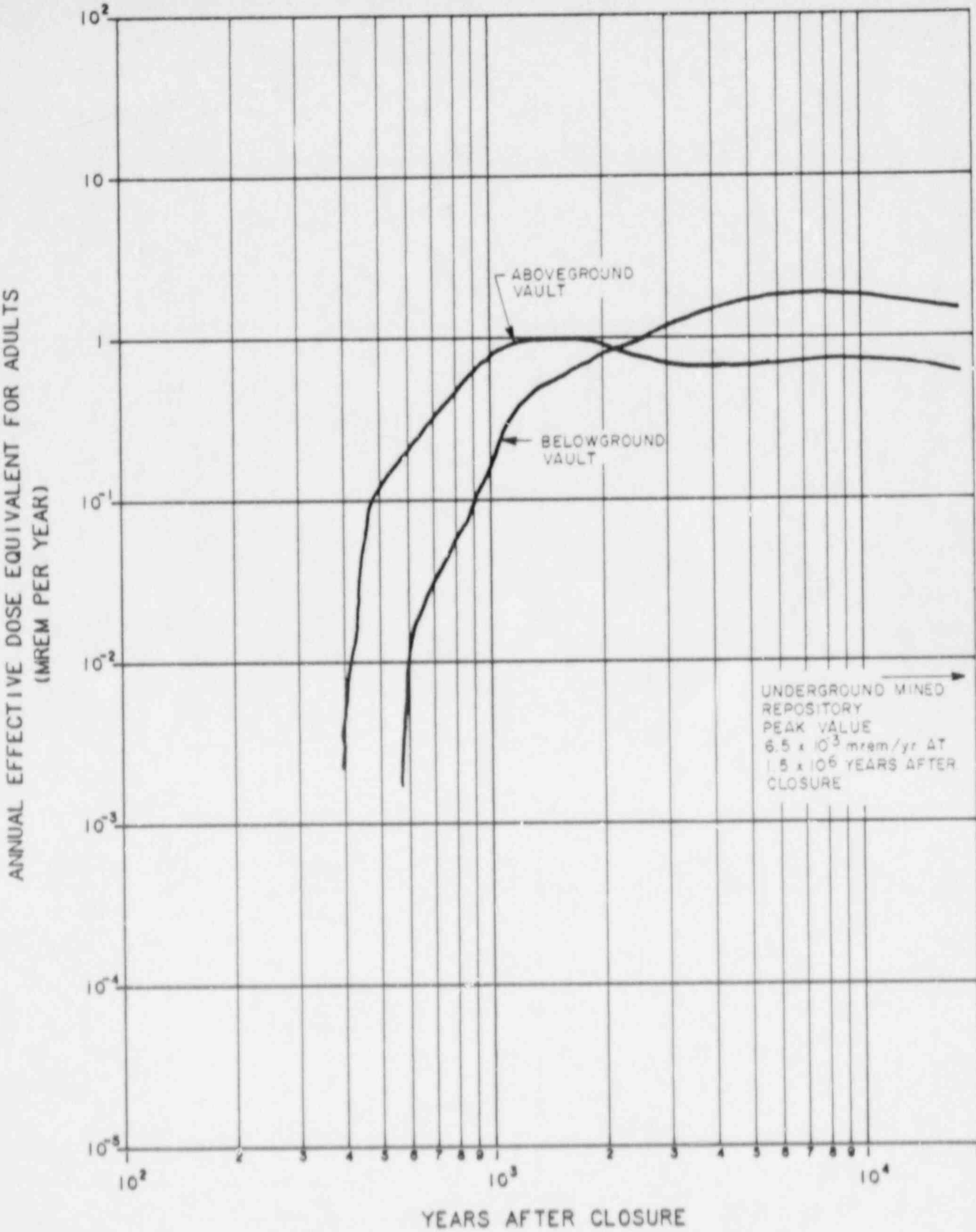
FIGURE 15



COMPARISON OF REFERENCE DISPOSAL METHODS IN PROVINCE II

AGRES

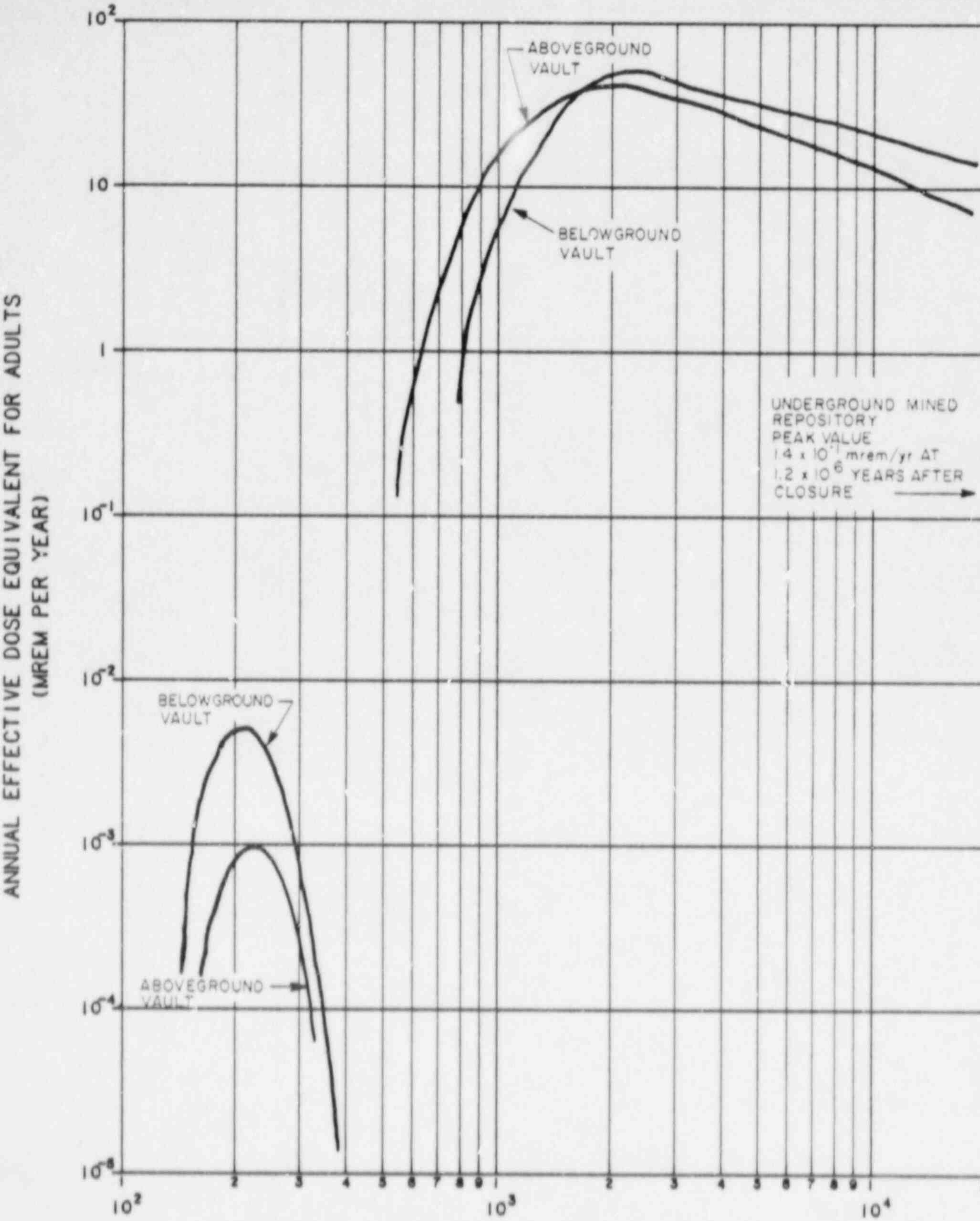
FIGURE 16



COMPARISON OF REFERENCE DISPOSAL METHODS IN PROVINCE IV

ACRES

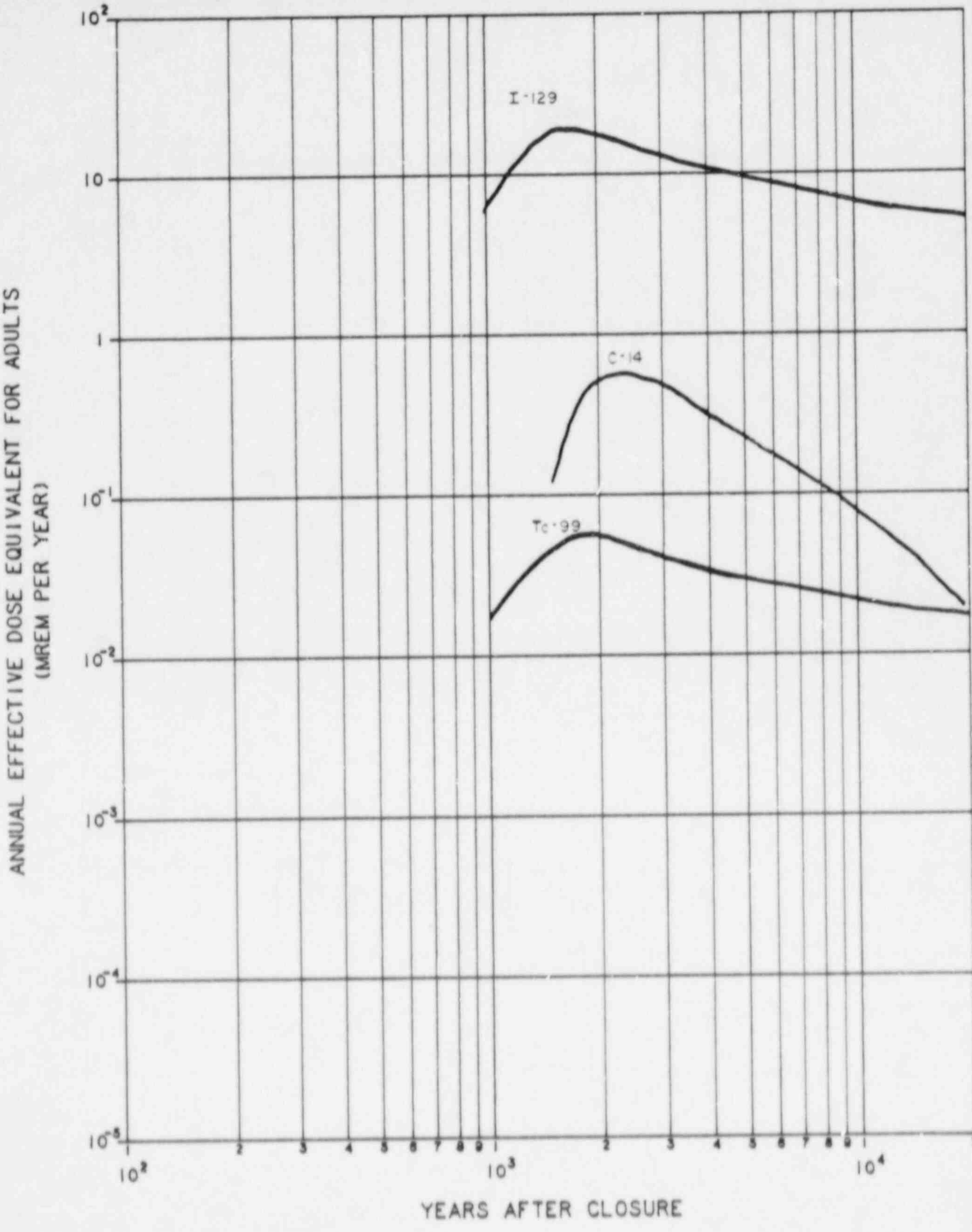
FIGURE 17



COMPARISON OF REFERENCE DISPOSAL METHODS IN PROVINCE V

ACRES

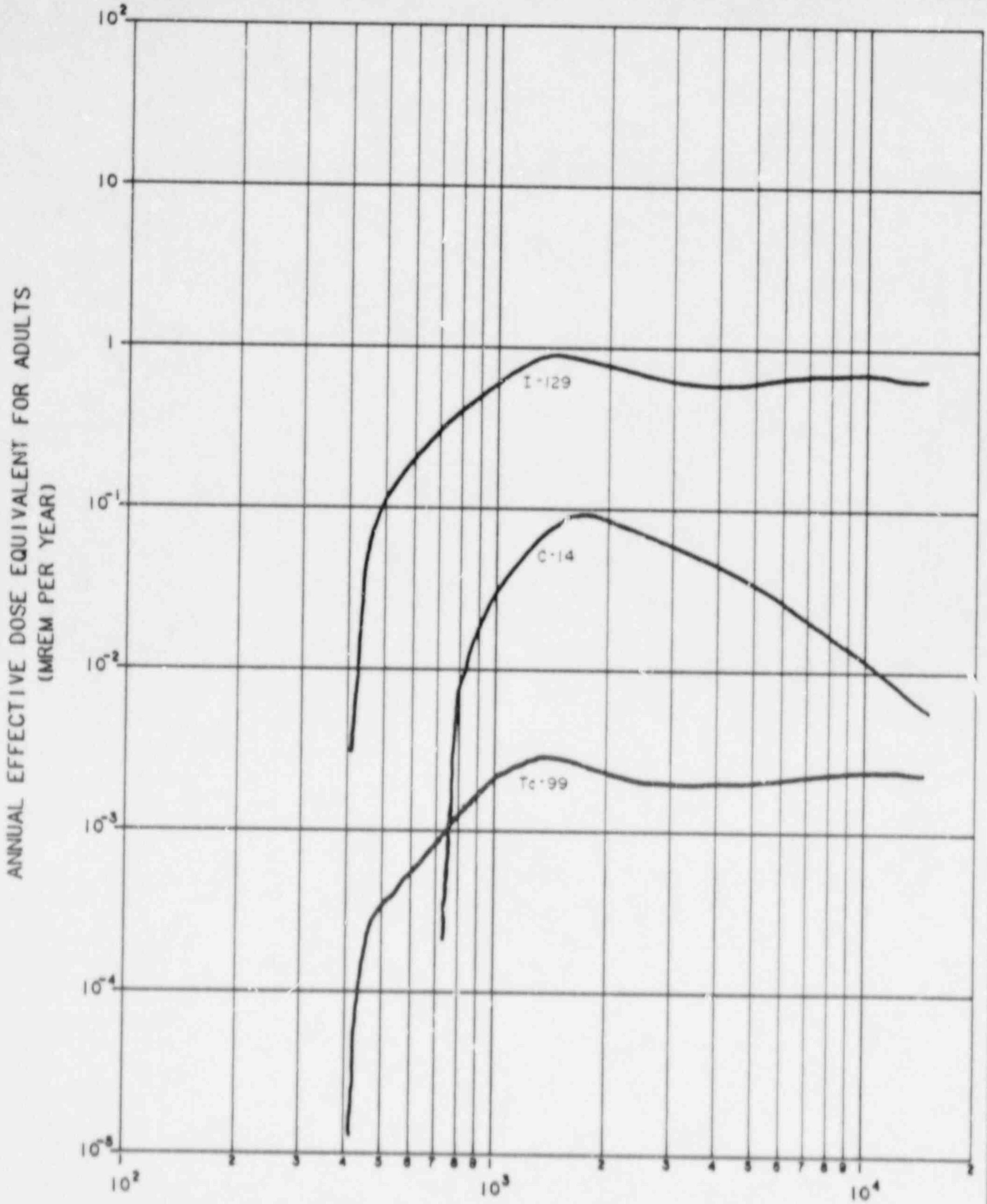
FIGURE 18



ANNUAL EFFECTIVE DOSE EQUIVALENT FOR ADULT
FOR SIGNIFICANT NUCLIDES
ABOVEGROUND VAULT IN PROVINCE II

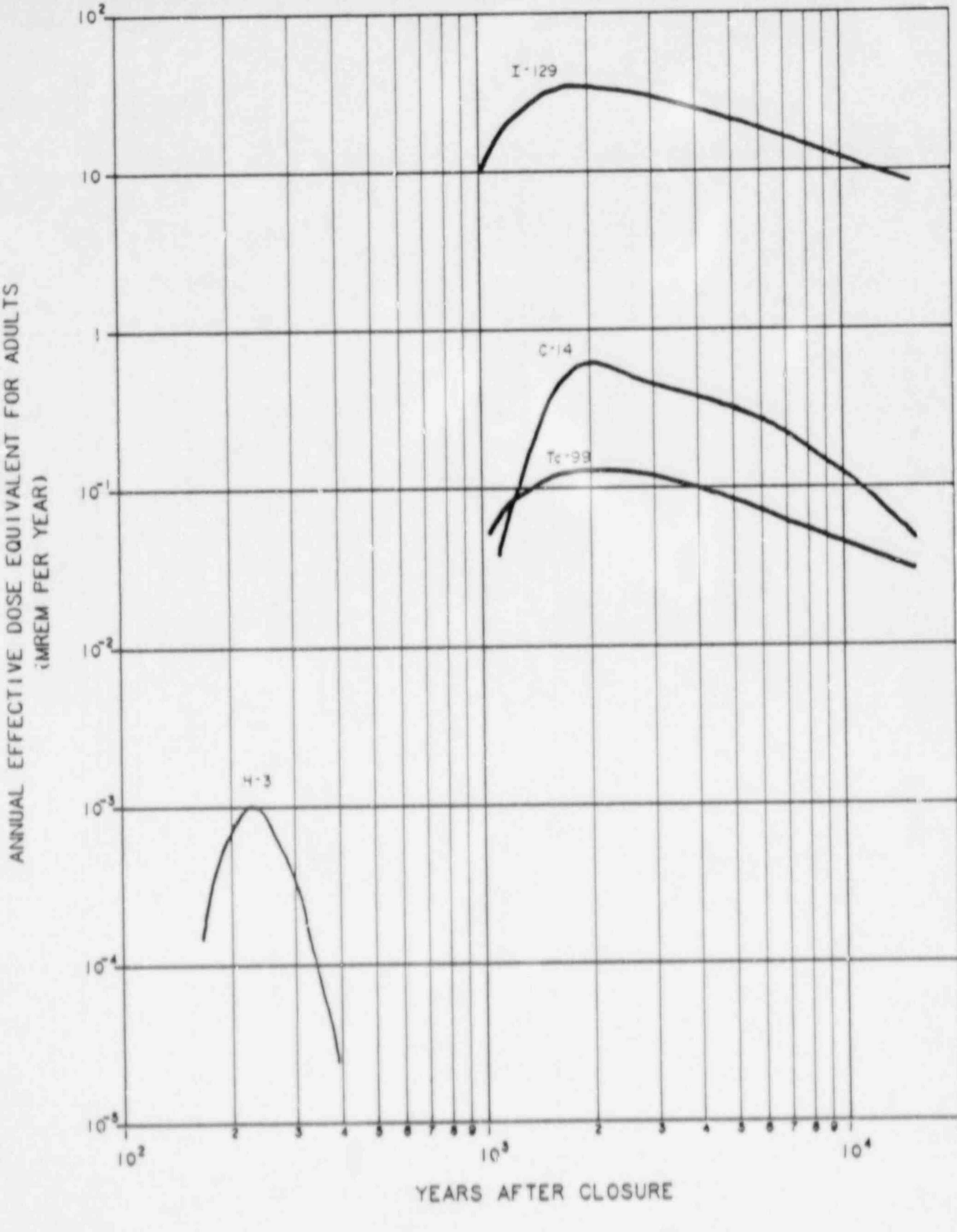
ACRES

FIGURE 19



ANNUAL EFFECTIVE DOSE EQUIVALENT FOR ADULT
FOR SIGNIFICANT NUCLIDES
ABOVEGROUND VAULT IN PROVINCE IV

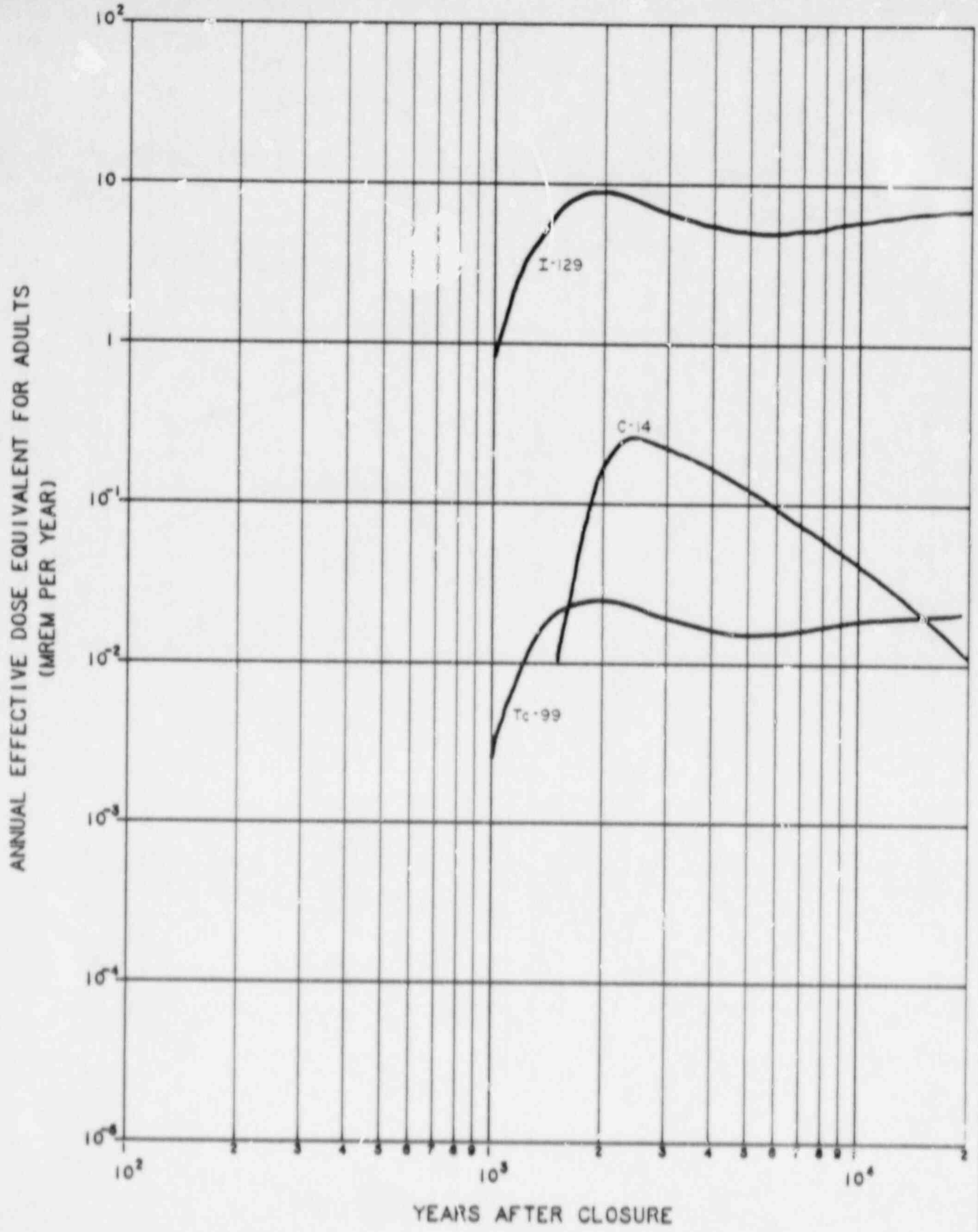
FIGURE 20



ANNUAL EFFECTIVE DOSE EQUIVALENT FOR ADULT
FOR SIGNIFICANT NUCLIDES
ABOVEGROUND VAULT IN PROVINCE V

ACRES

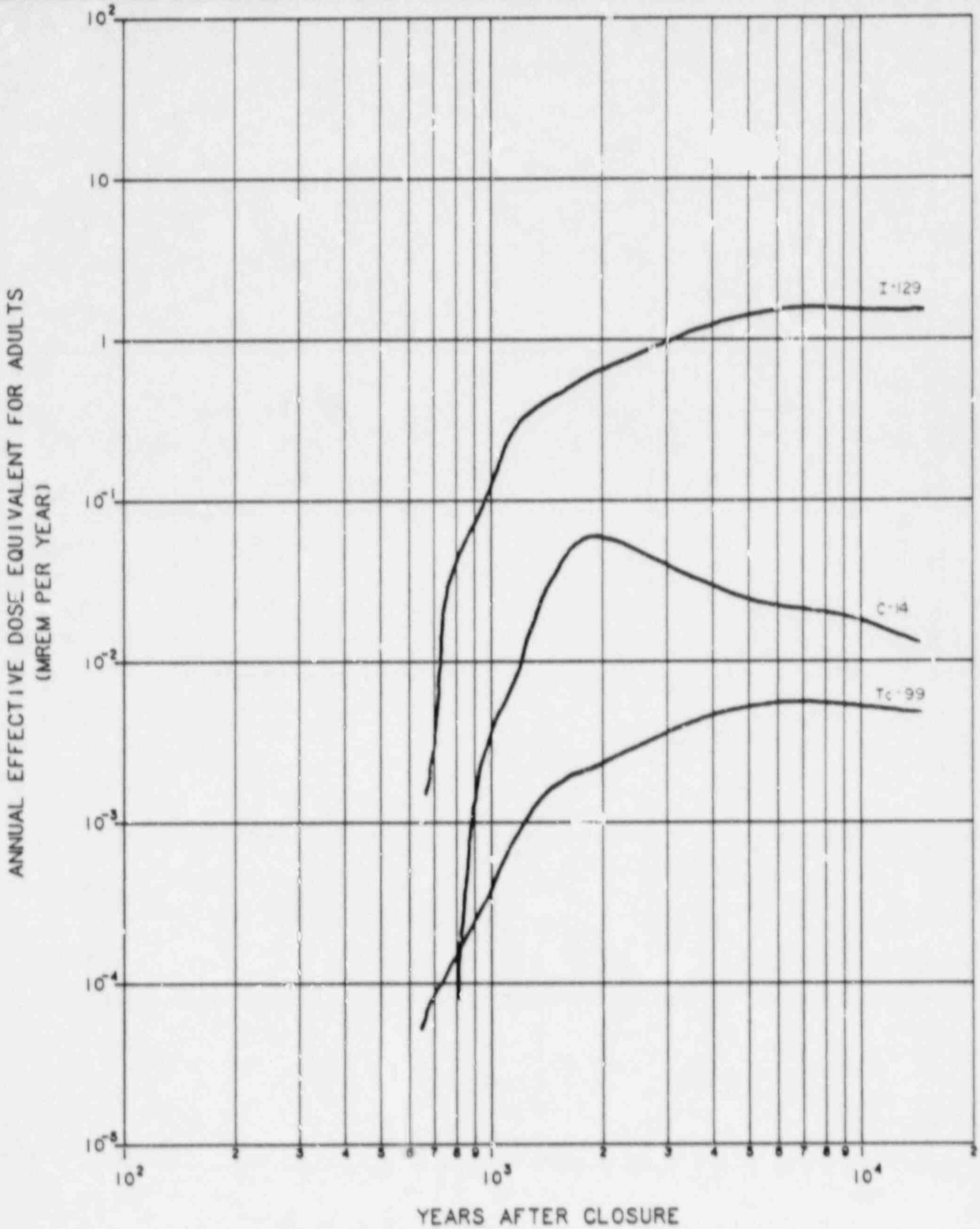
FIGURE 21



ANNUAL EFFECTIVE DOSE EQUIVALENT FOR ADULT
FOR SIGNIFICANT NUCLIDES
BELOWGROUND VAULT IN PROVINCE II

ACRES

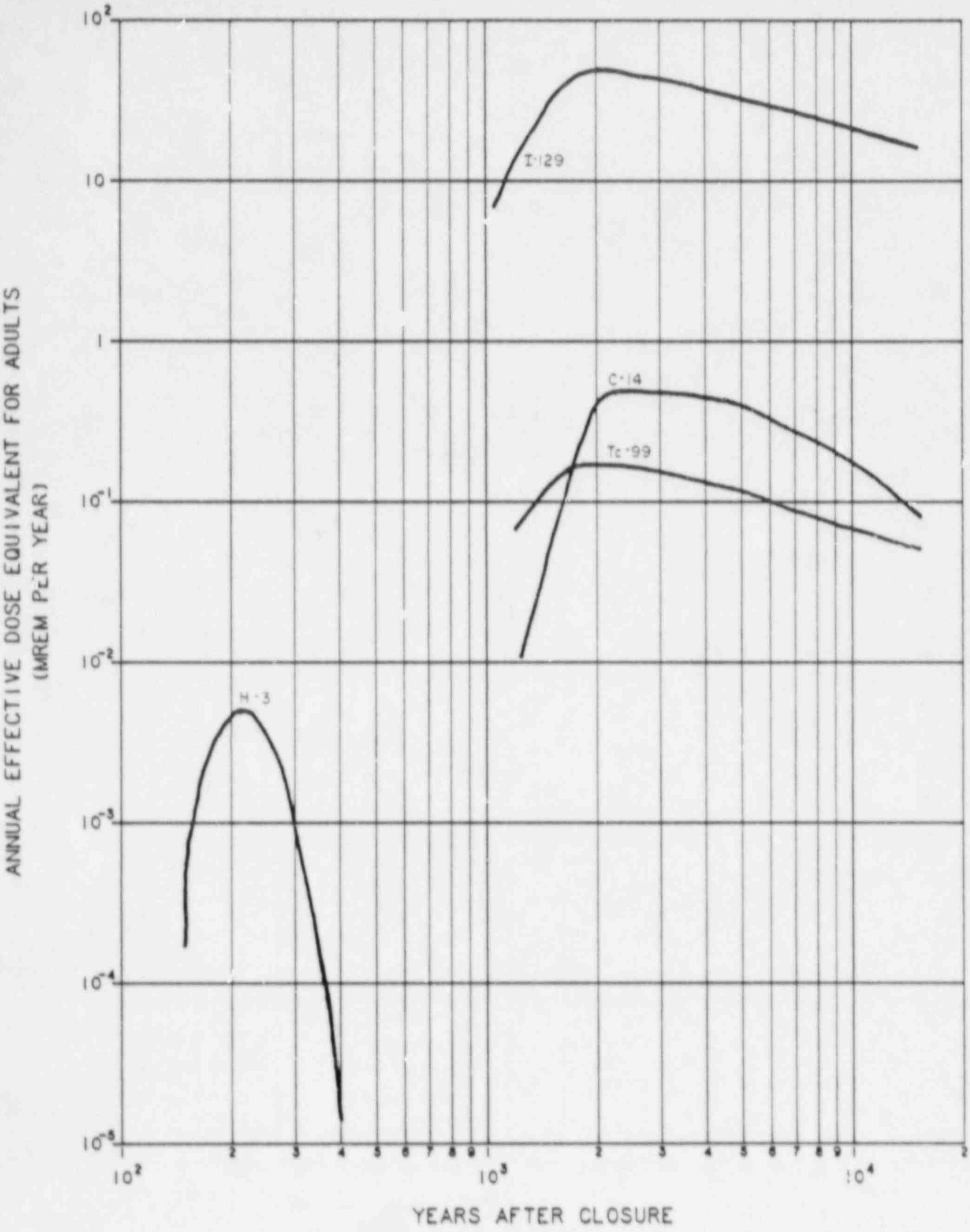
FIGURE 22



ANNUAL EFFECTIVE DOSE EQUIVALENT FOR ADULT
FOR SIGNIFICANT NUCLIDES
BELOWGROUND VAULT IN PROVINCE IV

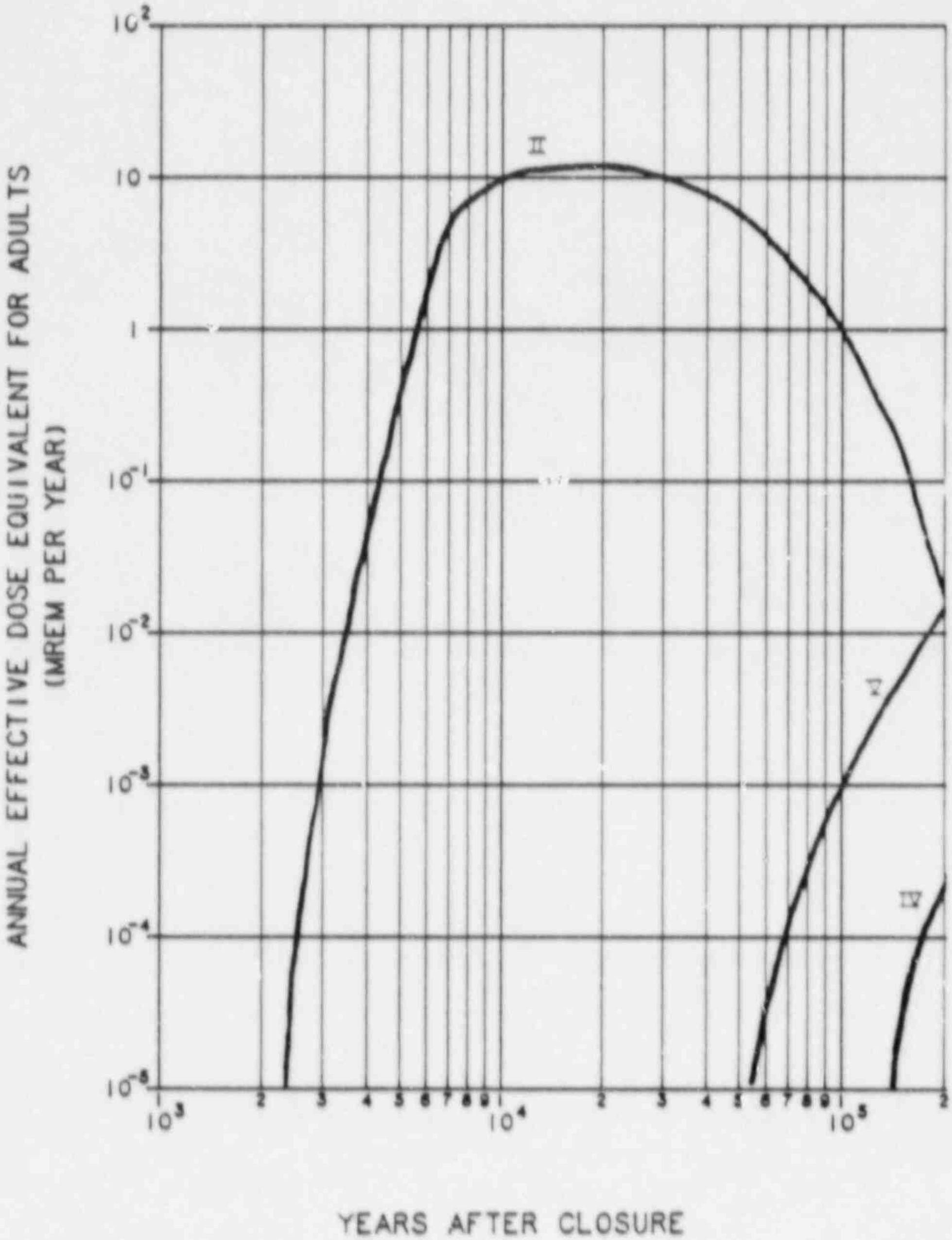
ACRES

FIGURE 23



ANNUAL EFFECTIVE DOSE EQUIVALENT FOR ADULT
FOR SIGNIFICANT NUCLIDES
BELOWGROUND VAULT IN PROVINCE V

FIGURE 24



INDICATION OF LONG-TERM DOSES FROM MINED REPOSITORIES
IN PROVINCES II, IV AND V

ACRES

FIGURE 25

APPENDIX A

Assessment Codes That Were Used

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APPENDIX A

LIST OF FIGURES

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APPENDIX A

The Assessment Codes That Were Used

The following descriptions of the codes are basically the same as in the DEIS. There are some additions to the COSMOS section, in the form of a fuller account of mass-transfer factors and descriptions of two new sub-models. An expanded set of dose-conversion factors has been included in both codes, and now has 'effective committed dose equivalent' and the committed dose equivalent to eight individual organs.

The two safety assessment codes, SYVAC3 and COSMOS-S/D, have been developed by Atomic Energy of Canada Ltd., (AECL) as a complementary pair. SYVAC was originally designed for deep disposal of nuclear fuel waste, and COSMOS for near-surface disposal of low- and intermediate-level nuclear waste. However, both can be used in wider fields and models can be transferred from one structure to the other to improve coverage of a particular problem. This assessment is a good exemple of such a transfer and details are given below, in the section on SYVAC.

Both codes describe complete pathways, starting with escape of radionuclides from containment in a repository, continuing with their migration through ground or atmosphere, and concluding with irradiation of humans, either directly or by way of the food chains. The scenarios that were modelled (with their pathways and repository types) are described and illustrated in Section 4. For these generic types of study, the full generality of the modelling was not needed, although more comprehensive studies might well be appropriate for assessments of specific repositories on actual sites. The major restrictions were as follows:

- only "deterministic" runs were made (see Section 2);
- since gaseous decomposition was assumed to be unimportant, dispersion processes in the atmosphere were omitted (see Section 2);

since the humans at risk in the scenarios are assumed to obtain their water from wells, the only surface water effects that were considered came from recreational activities such as fishing.

COSMOS has been used to assess the aboveground vault and the below-ground near-surface vault, and SYVAC to assess the underground mined repository.

Detailed descriptions of both codes are appearing in the public domain, usually as AECL reports and conference papers. At present, SYVAC coverage is the more complete.

A.1 THE COSMOS-S/D CODE AND ITS MODELS

A.1.1 Code Structure

The COSMOS-S/D (Stochastic/Deterministic) code (Jarvis et al., 1986) is designed for the safety assessment of waste disposal near the surface. In its present form, COSMOS-S/D models water infiltration through a leaking roof; the failure of containers and consequent leaching of their radionuclide contents; migration of the nuclides through saturated and unsaturated media such as buffer, backfill, and layers of ground; and dispersion in the atmosphere or in surface waters, including evaporation and adsorption in sediment. It can describe the transfer of nuclides in groundwater taken by a well, and releases into the atmosphere by irrigation sprays. Finally, it calculates the potential doses to an affected population and the corresponding maximum risk to health.

The code complex COSMOS-S/D consists of five interdependent programs, CHECK, SAMPLE, COSMOS, BIOS and DISPLAY, which share information through files. The heart of the complex, COSMOS, is an assembly of migration models. The structure of the complex is shown in Figure A-1.

Input from users of the code is read and analyzed for completeness and consistency by CHECK, which produces an input file. SAMPLE operates upon the input to produce parameter values that are required by COSMOS, which then creates concentration files. The concentrations are combined with dose/concentration ratios from BIOS, to produce dose and total-dose files. Finally, DISPLAY arranges printing and graphical displays. The code was designed to run on the CDC Cyber System at CRNL, but it is written in ANSI Fortran and, with the exception of machine-dependent operations in SAMPLE and DISPLAY, is easily transportable.

A complete pathway description, from source to irradiation of population, is defined as a "scenario" and COSMOS-S/D can be run in either deterministic or stochastic mode. A deterministic scenario will involve a single run of COSMOS with "best estimate" parameters; a stochastic scenario will require several hundred runs with random values for key parameters with appropriate distributions.

The input file to the program SAMPLE defines the scenario. If the deterministic mode is selected, this basic scenario information is simply written to a parameter file which is accessible to COSMOS. If the stochastic mode is chosen, then in addition to the basic scenario information, SAMPLE writes a specified (large) number of records, each containing a parameter set randomly generated on the basis of selected distributions.

The migration models in COSMOS can access one or some of these parameter sets. Any block of parameter sets stored as sequential records can be accessed to run as a stochastic case. For example, if SAMPLE creates 1000 parameter sets, COSMOS can access the last 250, or any other block of sequential records. In addition, any single parameter set can be accessed and run deterministically so that each stage in the scenario can be examined in detail.

Within COSMOS, appropriate submodels represent each section of a migration pathway from disposal to the environment. The interaction of the migration and dispersion models is indicated in Figure A-2. PEGE is a source-control model that describes container failure and will permit the presence of two different types. LIXY models leaching by diffusion of nuclides from an inner region into a surrounding saturated layer. HADES describes the migration of nuclides, by diffusion and advection, through man-made barriers or layers of ground. STEGI models the mixture of unsaturated and saturated regions that occur in the vault and its immediate neighborhood when water infiltrates through a failing roof. STYX deals with transport by diffusion and advection through underground aquifers. HYDROS models contaminant transport in surface water systems. ATMOS describes dispersion in the atmosphere.

Communication between modules is effected primarily through a 'common block' structure.

Concentrations of each radionuclide are first calculated by LIXY, then recalculated by every submodel that is subsequently invoked. At intermediate points (for example between links in the HYDROS surface water chain) concentrations are optionally printed or stored for later use by the DISPLAY program. At the conclusion of each run, annual doses-per-nuclide and total-doses are calculated from the concentrations and the dose-concentration-ratios are read from files produced by BIOS. These are also stored for use by DISPLAY.

Normally, concentrations, annual doses, and total doses, are stored as the result of a deterministic run, but only the annual total doses from a stochastic case.

Processing, reduction, and plotting of data are controlled by the program DISPLAY, which exists in both deterministic and stochastic versions.

A thorough documentation of COSMOS-S/D is not yet assembled, but the most complete published account is (Jarvis et al., 1986).

A.1.2 Input, Checking and Sampling

When the sources, pathways and appropriate regions have been chosen to represent a particular scenario, an input file is created using a customized user-friendly "input form" or "template". The input file contains "best estimate" parameter values, and, if the code is to be run in stochastic mode, distribution types and characteristics of sampled parameters.

For example, a typical template for a deterministic run will occupy about 8 pages. It starts with general input such as titles, time-range and intervals, tallies of vault and general regions, and tallies of lakes and rivers. This is followed by a list of the radionuclides involved, and their inventories. Geometric data for the vault come next, along with the performance functions to describe roof leakage, and container failures. Then for each waste form and leaching region come diffusion parameters for each nuclide. The geometry of the engineered barriers follows, along with appropriate diffusion parameters for each nuclide, and then similar lists for the layers of overburden and the aquifer. The water-transport data ends with parameters for a well, and properties of the surface water in the lakes and rivers. Atmospheric data are next to describe dispersion and turbulence. The template closes with instructions to the printing and plotting routines.

The input file is submitted to the checker, which scans the file for inconsistencies or omissions, ensuring, for example, that parameters lie within accepted ranges, and computer storage limits are not exceeded. The checker issues both warning and fatal error messages. Warning messages draw attention to the use of non-standard options; in particular, for plotting and printing. Fatal errors filter out non-physical scenarios, and other input errors that will cause a run-time error in the sampling or migration model code. Unless a fatal error is detected in the input file, the checker produces a file that can be used directly for input to SAMPLE.

In the deterministic mode, SAMPLE simply reads the "best estimate" values from the input file and echoes them to a parameter file that is accessible to the COSMOS migration model code.

In the stochastic mode, SAMPLE also creates parameter sets by generating variables according to distributions specified in the input file, using the CDC Fortran function RANF, a pseudo-random number generator. In the current version of COSMOS-S/D, the following scenario parameters are sampled for each nuclide in the scenario.

LIXY: For leaching:

- Initial concentrations of nuclide in source
- Diffusion/retardation of nuclide for inner source region
- Diffusion/retardation of nuclide for outer source region

HADES: For migration in barriers and ground layers:

- Diffusion of nuclide in each layer
- Retardation of nuclide in each layer

HYDROS: For migration in surface waters:

- Sedimentation of nuclide in each link of the surface water chain

Four sampling distributions are supported by the code: Uniform (values are equally likely over a given range); Log-uniform (log-values are equally likely over a given range); Normal (mean and standard deviation provided); Lognormal (mean-of-log, and standard-deviation-of-log provided). It is also possible for any of the parameters described above to be declared "Constant" so it will not be sampled in a stochastic run.

A.1.3 Output

Processing, analysis and plotting of data are carried out by the two programs, DISPLAY and STOCHOS.

(a) DISPLAY-D, for deterministic runs

DISPLAY-D uses the output from a single deterministic run of COSMOS to plot any or all of the following as functions of time:

- concentrations of specified nuclides at intermediate points in the migration pathway;
- dose rates (per nuclide) to whole body or various organs; and
- total dose rates (for all nuclides) to whole body or various organs.

The dose rates are defined in Section 3, and are discussed in more detail in Section A.1.5.

The major result demonstrated by DISPLAY-D is the maximum dose rate and the time at which it occurs.

The maximum dose rate has no significance in a stochastic system because at each run it may occur at a different time. The results are analysed in a statistical fashion, as follows.

(b) STOCHOS, for stochastic runs

When COSMOS is run in a stochastic fashion, several hundred passes of the scenario are made, using sets of input that have been obtained by sampling each of the relevant input parameters from their specified distributions. A set of samples in the form of doses as functions of time, one for each set of input, is therefore generated. Doses are restricted to effective committed dose equivalent.

The routine STOCHOS was developed to analyse this set of samples or dose functions and to calculate the associated risk of fatal cancers plus inheritable disease in the first two generations. This is done by ordering the values of dose from all samples for each time point, in descending order of magnitude. Each sample, and therefore each value of dose at any time point, is assumed equally likely to occur, with the probability of occurrence being equal to the inverse of the total number of samples.

There is hence a sample distribution of dose values for each time point. Various statistical properties are deduced, such as the arithmetic- and geometric-means and standard deviations. Because the doses are ordered and equally likely, one could obtain estimates for confidence limits. If from a sample size N, for example, all but n values fall below a certain level, then the statement that one is (N-n)/N x 100 percent confident that the level is not exceeded can be made. More elaborate statistical tests could also be made on these sample distributions as required.

Risk at a given time can be calculated by using the ordered doses to construct a downward cumulative probability distribution, as a function of dose. If the highest dose of a sample of size N (D_1 say) is assigned a probability $P_1 = 1/N$, the second highest dose D_2 assigned $P_2 = 2/N$, and so forth; then a maximum risk R_{max} is calculated by taking the maximum of the product $D_i x P_i x Q$, where $i = 1,..,N$ and Q is the risk conversion factor per Sievert.

For compliance with Canadian regulations, (Atomic Energy Control Board of Canada, 1987), the annual risk limit of 1.0E-6, where Q = 2.0E-2 health effects per Sievert, is interpreted as meaning that, at any time, the arithmetic mean dose does not exceed 0.05 mSv and not more than 5% of the values in the distribution exceed 1 mSv, where the probability of the scenario occurring is taken as unity.

(c) Storage of output

The results from a deterministic run, and for each run of a stochastic case, are stored on permanent file, along with sufficient details of the input to define the scenario and the actual sampled values of the parameters in the stochastic runs. This permits other forms of output analysis to be specified, or perhaps the sample of parameters to be investigated, without repeating the more time-consuming migration calculations.

(d) SEERA - a correction for vault loading and siting

The conservative assumption is made that all the waste is stored in one row of vaults, within a relatively short period. In practice, there is likely to be a series of rows of vaults and considerable time lapse between the disposal of the first and last waste.

If rows of vaults are used, some of the nuclides in the earlier vaults will have decayed by the time the later vaults are filled. The behavior inside the vaults is likely to be similar on average, as regards container failure, roof failure, leaching, and transport of nuclides downward to the underlying aquifer. However, the transfer to the aquifer will be delayed by the loading delays, and hence the transport times in the aquifer, to the exposed population, will depend on the locations of successive rows.

The modifications to the conservative assumptions will be different for each nuclide - for short-lived, mobile nuclides the dose to the affected population will consist of a succession of smaller peaks, but for the long-lived, retarded nuclides the dose will be a peak that is only slightly lower (albeit somewhat broader).

A less conservative approach would require that the code be run separately for each row of vaults, with differing aquifer lengths, and a final combination made of such results, with appropriate phasing to reflect the delays in loading. Apart from considerably increased complexity, this would require a site-specific knowledge of loading patterns and vault siting.

However, with some simple assumptions, it is possible to make approximate estimates of the conservatism, sufficient for most environmental impact assessments. The estimates are made by the sub-routine SEERA, in the form of a set of correction factors which can be applied to the conservative calculations if desired.

It is assumed that there is a constant time delay T_s, between the closing of one row and the closing of the next. Vaults are arranged in parallel rows across the aquifer, and the later-filled rows are successively further from the exposed population.

For a particular nuclide, the increase in aquifer-transport time is T_w. This will depend on row spacing, the Darcy velocity in the aquifer, and the nuclide-dependent diffusion coefficient, retardation factor, and decay constant.

The total delay for a particular nuclide, between successive rows, is then:

$$T = T_s + T_w$$

One can show quite easily, by summing peaks as they arrive, that if there are n rows of vaults the total effect is,

$$C_n = C_o[1-\exp(-n\ T)]/[1-\exp(-\ T)]$$

The conservative calculation gives a value,

$$(C_n)\max = nC_o$$

and hence the correction for vault loading and spacing is approximately

$$C_n/(C_n)\max = (1/n)[1-\exp(-n\ T)]/[1-\exp(-\ T)]$$

For small T, the ratio approaches unity, from below.
For large T, the ratio approaches (1/n), from above.

A.1.4 Migration and Dispersion Models

The models that make up the COSMOS group to describe the migration of nuclides along the various pathways to the biosphere are shown in Figure 10, together with the present state of the possible links between them.

COSMOS-S/D is designed for low-level waste disposal--that is, with very low content of actinides--and it does not model chain-decay processes. The code thus deals with one nuclide completely before moving on to the next nuclide in the scenario.

The component models of COSMOS are PEGE, LIXY, STEGI, STYX, HADES, ATMOS, and HYDROS, along with two smaller models PNEUMA and KRENE.

In general, the calculations all work in terms of concentrations (the number of radionuclides per unit volume).

Migration is represented as one-dimensional, although the parameter values in a particular region may have been derived by two- or three-dimensional modelling outside the code.

In the migration calculations, the concentration at the exit face of a particular region is used as the source term for calculations in the following region.

(a) Interface Conditions

A variety of concentration-normalizations and mass-transfer coefficients are employed throughout the modelling, to ensure conservation of nuclides and continuity of their flows. They will be described briefly, in the order of their appearance in a scenario. The description refers to a particular nuclide, because some of the normalizations and factors are nuclide dependent.

The total inventory in terms of Bequerels (or curies) is converted into a concentration of nuclides per cubic meter of the effective slab used for leaching calculations. The concept of the slab and the calculations of its dimensions are discussed in the LIXY section below.

The concentration in the pore water in the slab is obtained by dividing by the retardation factor for the particular nuclide - which also varies with the waste form.

Leaching takes place across the interface between an inner and outer region. The mass-transfer coefficients are purely diffusive-continuity of nuclide concentration and nuclide current. The code actually has provision for more sophisticated transport-type conditions, to use more detailed experimental results in the future.

The LIXY calculation gives the concentration at the outer face of the outer region, that is a region with the face area of the effective slab - the cross-sectional area of the vault.

For 'dry' regions of the waste - those not affected by leaks through the roof - the connection to the buffer layer involves equal areas of cross-section, and the mass-transfer is a simple diffusive type continuity of concentration.

For 'wet' regions of the waste - influenced by infiltration through leaky areas of the roof - the transfer is more complicated. First, it is necessary to normalize the effective-slab surface area to the actual area of packages over which transfer to infiltrating water may occur. Second, the transfer must reflect movement from a purely diffusing region into a region with advection. The interface conditions developed by Cheung and Chan (1983) are used. They calculate an effective velocity of transfer and involve the area over which transfer occurs, the length of water travel, and the thickness of the receiving layer. If the layer is very thin, the conditions reduce to continuity of concentration.

From buffer to overburden, the 'dry' regions and 'wet' regions are followed separately, with diffusive-type conservation of concentration, and advective-type conservation of flow, respectively.

For transfer from 'dry' overburden regions to the aquifer, a mixed diffusive-advective condition of the Cheung and Chan type is used, and for transfer from 'wet' overburden regions, a convective-type conservation of flow.

Once the aquifer is reached, transfer between successive layers is simply of the advective type giving conservation of the flow.

(b) LIXY, for leaching

Leaching is assumed to occur in semi-infinite slabs, with symmetry about the zero plane. Initially, the leaching nuclide is contained within a central region, and subsequently leaches into an outer region, across the faces of the central region. In practice, the central region will probably be a block of cement or bitumen containing waste, or a compressed bale of active material, and the outer region will be a surrounding water layer inside the vault or backfill.

It is assumed that both regions are homogeneous and that the leaching process from the inner region, and migration through the outer region, can be described by diffusion processes with retardation factors.

The variable is concentration of leaching nuclide, as a function of time after start of leaching. The Laplace Transform method is used to obtain an analytical form of solution, with considerable savings in computing time. Problems of this sort are usually solved by a solution in series that is valid for relatively short times. However, with considerable difficulty, a solution was derived that is valid for all times. It does not appear in the common literature. It is described in (Fraser and Jarvis, 1985a).

The finite second layer is replaced by an infinite region. The approximation simplifies the mathematics considerably. Its validity, in a comparable situation, is justified in some detail by Shamir and Harleman (1967). The solution at the appropriate distance into the outer region is then taken to represent the concentration at the edge of the second layer, to serve as a source for migration through a following region in the vault. Apart from the simplification, the stratagem has the advantage that solutions of the leaching model are decoupled from solutions of the models for outer regions of the vault and the leaching calculations can be performed separately. The model, its mathematics, and coding, are described in (Fraser and Jarvis, 1985a).

Along with this model goes the problem of deciding the effective thickness of the equivalent semi-infinite slab that represents the inner region. This is resolved by assuming that each waste container leaches into a surrounding (probably thin) region of water. The concept of mean-chord-length of a reasonably smooth body (no sharp penetrations), which was developed for diffusion-type calculations in neutron transport theory, can then be used. The semi-infinite slab is arranged to have the same mean-chord-length as the particular waste containers. The calculation for regular bodies, such as slabs, cylinders, and bales, is particularly simple. Dirac's derivation, (Dirac, 1953) shows that:

$$\text{mean-chord-length} = 4 \times (\text{Volume})/(\text{Surface Area}).$$

The concept of mean-chord-length is also discussed, at some length, by Luikov (1968), for diffusive processes in heat transport.

The concentration of a particular nuclide at the onset of leaching is then calculated to be consistent with the volume of the equivalent slab and the nuclide inventory.

It is assumed that leaching is not inhibited by solubility limits in the second region.

Leaching peaks can vary quite sharply with time, and it is possible that a given set of time intervals could straddle the peak, with the result that the maximum of the concentration would be missed. To avoid this, the peak height and location are calculated separately, and the correct peak height is inserted into the source array at the nearest time point on the lower side. The problem does not appear to be significant in succeeding regions, because diffusion has broadened the peak.

(c) PEGE, for Source Control

PEGE allows for the presence of different streams of waste, in different types of containers, and their differing rates of leaching as the containers fail. At present, two kinds of waste can be present, such as relatively leachable compacted waste, perhaps in steel drums, and leach-resistant encapsulated waste, perhaps in drums or concrete boxes.

In a flooded vault, unprotected compacted waste will start to leach immediately, but waste in a particular container will be protected until that container fails. The containers are likely to be steel drums or boxes and the model describes the statistical process of their failure by corrosion, caused by the moisture in the vault.

It is usually assumed, although this is not necessary, that the cumulative probability of container failure follows an S-type of curve. A typical example is shown in Figure 11, which is obtained from a log-normal distribution with a mean of 100 years, and a standard deviation of 0.35.

For convenience, failures are described in the input by a series of discrete fractions. However, since the failure process is likely to be continuous and, in particular, a container does not fail over the whole of its surface at once, the input failure fractions are interpolated linearly over intervening time points to effect a smoothing.

The different forms of waste can have different failure functions.

When container failures are to be modelled, a "source-shape" leaching curve is first generated with LIXY, for a particular nuclide and is then used to build a composite curve to represent the 'after failure' source of that nuclide. As each fraction fails, and an additional leaching starts, the LIXY shape is added to the composite, starting at the time of the failure and with a magnitude reduced according to the incremental failure that took place and the decay in source nuclide that had occurred by that time.

Since the failure process is still discrete (taking place, as it does, at the standard time intervals), the composite curve is a discrete assembly of peaks and it can show some small ripples that may be undesirable. A final smoothing is effected by taking the points three at a time and performing an area-preserving transformation that reduces small peaks and raises small troughs. The smoothing is 'local' and the major peak in the curve is not affected.

(d) STEGI, for Roof Failure

As the roof on the vault deteriorates, water may begin to leak through from above long before significant loss of structural strength has occurred. Failure is presumably a gradual process, one location at a time, as small areas deteriorate.

In vaults with dimensions of tens of meters, it seems likely that leaks will affect only the packages in the immediate vicinity, until large fractions of the roof are leaking. There might thus be two regimes of moisture coexisting in a vault: normal unsaturated regions with a small percentage of moisture by volume; and localities with water actually flowing, at least during the leaks.

Leaks are likely to be intermittent because of seasonal fluctuations in rainfall and snow melt. However, because all the migration processes take place gradually, it has been assumed that in the fraction of vault affected by leaks, the velocity of the water flowing through will be constant, with a value equal to the yearly average of infiltration velocities. The flow is to continue through layers such as buffer, and through underlving ground layers, to an underlying aquifer. Thus, with time an increasing fraction of roof leaks, and an increasing fraction of the vault cross-sectional area is affected.

The model STEGI draws on a failure function for the roof in much the same way as PEGE uses a failure function for the containers. At any time t, a fraction f(t) of roof is leaking and, at this time, PEGE has estimated that a source S(t) of a particular nuclide is available to migrate after container failure and leaching.

STEGI assumes that two migration processes are proceeding at the same time: diffusion alone from a source [1-f(t)]S(t), through the layers below the regions without leaks; and diffusion plus advection through the layers below the leaky regions. In this way, two source functions are built up over the whole range of t values, and two calls are made on HADES to perform the appropriate migration calculations. The regimes are followed separately through each of the man-made layers, such as backfill and buffer, and through the underlying layer of ground. When the aquifer is reached, the concentrations from the two regimes are combined.

(e) STYX, for Migration Through Underground Aquifers

STYX follows the migration of nuclides through a succession of aquifers, once the immediate surroundings of the vault have been left behind, to eventual seepage into surface waters. Nuclides can leave along the way by extraction in well water and perhaps by entering the atmosphere.

STYX performs the "bookkeeping", as nuclides move through the aquifers, to control conservation of nuclide flow at interfaces between different regions. The anticipated perturbations are changes in aquifer dimensions or removal of part of the aquifer flow by a well.

For a well, the removal is handled by locating the well at an interface. The submodel KRENE is called to take account of the nuclides that may enter the biosphere if the well water is used by humans. If irrigation spray can put radionuclides into the atmosphere, a call is made to the submodel PNEUMA to introduce the radionuclides as a source term for atmospheric dispersion in ATMOS.

For each region, the migration is described by a call to HADES and the concentration at the output face of one region then serves as the input source to the following region, after the appropriate interface conditions have been applied.

(f) HADES, for Migration Through Barriers and Ground

The model describes the migration of nuclides, by diffusion and advection, through a region that can represent a man-made barrier, a layer of ground below the vault, or an aquifer. It is called to model migration through the layers in STEGI and in STYX.

In each region, a one-dimensional equation describes diffusion, advection (if it exists), and retardation processes. A solution is derived analytically, by the Laplace Transform Method, up to the point of a numerical integration over a convolution integral, to yield the time behavior of nuclide concentrations at the outer face of the region. The convolution involves as source term the concentration at the outer face of the previous region and a function that describes the migration processes in the region.

In the development of the solutions, a region is assumed to be infinite in the direction of migration and the solution at the appropriate distance into the region is taken to represent the concentration at the outer edge of a finite region. This substitution has been well discussed by Shamir and Harleman (1967) and is appropriate for the situations envisaged. It has been further checked by other investigations with HADES.

The model is designed for saturated media, but the mathematics are similar to that for certain of the processes in unsaturated regions and if appropriate effective parameter values can be defined, those processes can be modelled.

The mathematics and general structure of the code are described in Fraser and Jarvis (1985b). Since that report was written, the convolution integrations have been changed to use the trapezoidal method rather than Simpson's Rule. The replacement is more "robust" and, although finer time intervals are needed, the overall changes in code structure have permitted a significant improvement in code efficiency.

(g) ATMOS, for Transport and Dispersion in the Atmosphere

The model describes airborne contamination, and its core is a one-wind Gaussian plume model that calculates ground-level air concentrations of contaminant at a single receptor point, from a number of point-sources. Units are contaminant source strength (i.e., a unit amount per unit time, per unit source surface area).

Account is taken of plume depletion of contaminant arising from deposition losses (via the subroutine DPLETE), and vertical dispersion of the plume calculated using the Hosker equations (via the subroutine SIGMAZ). Lateral dispersion is accounted for by assuming a uniform distribution of density in the lateral direction within a sector consisting of one-sixteenth of a circle, one sector for each point of the compass.

ATMOS is called only once to calculate a factor representing ground-level air concentration per unit source strength, for each source. Any stochastic or time variations are assumed to be due entirely to such variations in the sources. When a value for source strength is calculated for a particular time, it is then multiplied by the factor from ATMOS to obtain the actual airborne contaminant concentrations for use with food-chain calculations.

Airborne contaminant sources appear as coding outside of ATMOS. Presently, only two sources are considered in COSMOS: evaporation of tritiated water vapor from lakes, and contaminated water droplets from spray irrigation. Other possibilities for the future include contamination that exists as a gas, such as tritium or carbon-14; pollen released by plants and trees that take up radionuclides through their roots, or smoke and ash in a forest fire; wind suspension of contaminated surface soil by modelling saltation and suspension of soil particles, or by using empirically derived relations such as the Wind Erosion Equations from agricultural lands. Both of the methods of the latter example, however, assume bare soil surfaces and tend to greatly overestimate suspension from forests, lakes and wetlands where one expects radionuclides to emerge in groundwater discharge.

Data required by ATMOS can be divided into three different sets: weather data, consisting of frequency tables describing wind direction, wind speed and weather type; parameters of the terrain representing ground cover for the depletion calculation and terrain roughness for the vertical dispersion calculation; and distances between each source and the receptor point and directions to a given source from the receptor point (e.g., N, NNE, etc.).

The wind and terrain data appear as data statements and may need to be modified since this information is site-specific. The source-receptor distances and directions are input by the user for each source, along with two other values describing the form of contaminant as a gas, particulate, or as contaminated soil particles. Some source information describing changes in the plume height, is set by constants within ATMOS, such as release height and thermal buoyancy effects.

The modelling of ATMOS is a simpler version of the well-established U.S. Code ATM (Culkowski and Patterson, 1976). ATMOS results have compared satisfactorily with a test case that was supplied with ATM. ATMOS is described in Wilkinson (1987a).

(h) PNEUMA, for Releases to Atmosphere

The purpose of this submodel is to handle releases of contaminants into the atmosphere from the vault, from the ground, or from surface waters. It calculates the product of three factors (concentration in water at the point of escape) x (the appropriate atmospheric source factor for the types of release) x (the atmospheric concentration ratio calculated by ATMOS). The result is stored, along with other possible atmospheric contributions at the receptor point, for eventual conversion by BIOS into doses to the affected population.

Atmospheric sources, unlike atmospheric concentration ratios, may depend on stochastically varying parameters. These sources are assumed to occur only within the following subroutines and involve the following processes:

gas generation from biological activity in PEGE (not yet installed);

- suspension by spray irrigation (if specified to occur) in KRENE; and
- evaporation of lake water in HYDROS, presently for tritium only.

Within these routines, atmospheric source factors are calculated and immediately followed by a call to PNEUMA. Atmospheric source factors are defined as the fraction of the current radionuclide concentration available in water that becomes airborne.

(i) KRENE, a Well Model

The purpose of the submodel, KRENE, is to represent the effects of a well (when used with respect to groundwater) or a pump (with respect to surface water). The water drawn up in either of these mechanisms is used in the food-chain calculations of BIOS.

For the case of the well, it is assumed that the contaminant in the groundwater is contained within a plume of known cross-sectional area at the point where the well intercepts it. With the groundwater flow velocity known, the product of velocity times cross-sectional area is interpreted as the incoming flow rate of contaminated water available to the well. If the well pumps at a given rate, the contaminated flow rate remaining in the ground is taken to be the difference between the incoming flow rate available and the pumping rate.

When the pumping rate is less than the flow rate available, the concentration of contaminant in water is assumed to be the same as in the plume just upstream of the well. The decrease in the amount of contaminant in the flow rate remaining is accounted for by assuming that the plume cross-sectional area is reduced by the quotient of pumped rate divided by incoming rate, while the concentration remains the same.

It is possible to pump at a higher rate than the flow rate available since this would mean that clean water, flowing from outside the contaminant plume, is also being pumped. In this case, the contaminant concentration in the well water is equal to the product of that in the incoming flow times a dilution factor, defined as the quotient of pumping rate divided by flow rate available. The amount of contaminant, and hence the flow rate of contaminated water remaining in the ground, is therefore taken to be zero.

The submodel KRENE applies the same process to represent the pumping of surface water, with flow rates being those of a river or lake. The flow rate remaining is equal to the incoming rate minus the pumped rate, where no reference need be made to flow velocity or cross-sections. The probability of pumping rate being higher than the incoming rate is regarded as impossible since this implies that the river or lake is pumped dry.

In the computation, arrays containing contaminant concentrations of the incoming flow, pumped flow, and remaining flow are available to KRENE through different labelled common blocks. If the pumped rate is less than the rate available, the array of incoming flow concentrations is simply copied to the arrays representing pumped and remaining flow concentrations, and the cross-sectional area variable is reduced accordingly. If the pumped rate is greater than the incoming rate, then the product of incoming flow concentration times the dilution factor is stored in the array representing concentration in the pumped flow, and the remaining flow concentration is zero.

(j) HYDROS, for Migration in Surface Water

This routine models radionuclide migration in surface waters in which the compartments considered (river reach and lake) may appear in any sequence. The main mechanism affecting radionuclide concentrations in this model is dilution. However, account is also taken of some delay mechanisms, even though these affect radionuclide concentrations to a comparatively minor degree.

The amount per unit time of a given radionuclide coming out of the ground is available from the groundwater transport routine HADES, and is expressed as the product of (radionuclide concentration) x (groundwater flow velocity) x (cross-section area of the contaminant plume), all taken at the seepage face.

If the mechanism of the river compartment involves only dilution, then all one needs is a value for the outflow rate for the particular river reach. A sufficiently accurate way to obtain this quantity is through the use of Manning's equation, which relates bulk flow properties to the dimensions and characteristics of the river channel. Uniform flow in the river reach is thus assumed, but theory accounting for non-uniform flow could be implemented if necessary.

The bulk flow velocity (calculated as an intermediate step) and the length of the river reach are used to calculate a residence time for water, and hence contaminant levels, within the reach. This is then used to compute the radionuclide decay for river reaches, though the effect in the model is very minimal.

The lake is described mathematically by a one-dimensional ordinary differential equation, solved by analytical and numerical techniques, and coded in a computationally efficient manner as a recursive relation. The principle of this submodel is based upon the total inflow of water to the lake being equal to the total outflow, consisting of evaporation and either groundwater or surface water outflow.

Practically all radioactive decay of contaminant occurs in the lakes. The first of two mechanisms considered is the delay resulting from the time needed to flush the water from the lake. The second and more important mechanism involves the sorption of radionuclides to lake sediment, which is modelled as an irreversible process.

Both submodels assume that contaminant disperses uniformly and instantly throughout the lake volume or river cross-section. One argues that the response time for such dispersion is assumed to be much shorter than the duration of one time-step used in COSMOS. Contaminant plumes, however, can in fact persist for considerable distances in both rivers and lakes. Residence times in a river reach may be as short as days or even hours. In practice then, this assumption may not be conservative and may lead to unrealistically low concentrations.

A simple way to account for these effects might be to define an effective lake volume or river cross-section. These quantities would be less than the actual quantity in both cases, and thus concentration predictions could be made conservative. However, one must assume as a result that water is removed from the river or lake into the food chain at the worst possible location.

HYDROS is described in Wilkinson (1987b).

A ' 5 Dose to a Population at Risk

BIOS calculates the dose to a human from radioactive contamination in air, well water, and surface water. It is based upon the dose/concentration factors in a Standard provided by the Canadian Standards Association, (1987).

The dose concepts have been defined in Section 3. The BIOS routines have now been altered to calculate:

a) effective committed dose equivalent, using the CSA weighting factors; - as discussed in section 3, this is used instead of the 'whole body dose' referred to in USNRC Rules and Regulations 10CFR/Part 61.
b) committed doses equivalent to eight (8) individual organs: lung; stomach wall; lower large intestine; kidney; liver; red bone marrow; bone surface; and thyroid.

The dose conversion factors of the CSA standard were supplemented by calculations of Johnson and Dunford, reported in Johnson et al (1979), Johnson (1982), Johnson and Dunford (1983), Dunford and Johnson (1988).

The BIOS routines work internally in terms of Sieverts and Bequerels but, as explained later, for this study the output doses have been converted to rem.

To understand the model, the following definitions are needed:

- A <u>compartment</u> is a medium that can contain radionuclides (e.g. source, water, air, soil, plants, animals and man).

- A <u>transfer</u> is a process by which radionuclides move from one compartment to another (e.g. sorption of radionuclides to soil particles, plant uptake through the roots, eating, drinking).

- A <u>pathway</u> is a combination of compartments and transfers along which radionuclides are carried from their source to man, where a radiation dose is realized.

- A <u>menu</u> is a network of several pathways that represents all relevant processes in the food chain.

The dose resulting from a given type of radionuclide moving through a menu is given by the following expression:

$$D(h,i,j,k,s) = C(i,j)U(h,i,j)F(h,i,k,s)$$

where, $U(h,i,j)$ is a combination of $P(h,i,j)$ coefficients.

The above functions are defined as follows:

D - Dose rate (Sv/yr);

C - Nuclide concentration in the air, well water or surface water, at the beginning of the food chain (Bq/L in water and Bq/m^3 in air);

U - Menu parameter (nuclide dispersion factor in the food chain, with units of Bq/L for drink or Bq/kg for food, either divided by Bq/L of contaminant in water or Bq/m^3 of contaminant in air);

P - Radionuclide transfer coefficient (defined as the ratio of contaminant concentration i within compartment n, available for further radionuclide transfer, divided by the amount transferred directly to that compartment m, with units depending on the process);

F - Dose-from-concentration conversion factor (in (Sv/yr)/(Bq/kg) for food, (Sv/a)/(Bq/L) for drink, (Sv/yr)/(Bq/m^3) for exposure;

and the indices are:

h - type of human (i.e., infant, adult);

i - type of nuclide (e.g., Strontium-90, one of a total of 71);

j - source of nuclide (i.e., air, well water, river or lake water);

k - exposure type (i.e., internal, external); and

s - dose type (i.e., whole-body, skin, bone surface, thyroid, lower large intestine).

The particular compartments and modules for the radionuclide transfer coefficients used in BIOS are specified in the CSA Standard. As an illustration, however, consider the scenario of a man owning a cow and drinking water and cow's milk. Assuming the relevant compartments are:

- contaminated water;
- contaminated air;
- soil;
- grass;

- milk cow; and
- man.

The radionuclide transfer coefficients are then described (neglecting the h and i indices for simplicity) by

P_{15} - water drunk by the cow;
P_{16} - man drinking contaminated water;
P_{23} - airborne contaminant settling on soil;
P_{24} - airborne contaminant settling on grass;
P_{25} - cow breathing contaminated air;
P_{26} - man breathing contaminated air;
P_{34} - contamination in soil taken up by the grass;
P_{35} - cow ingesting dirt as it grazes;
P_{45} - cow eating grass; and
P_{56} - cow's milk drunk by man.

Possible pathways are represented by the products:

- P_{15} P_{56};
- P_{16};
- P_{23} P_{34} P_{45} P_{56};
- P_{23} P_{35} P_{56};
- P_{24} P_{45} P_{56}; and
- P_{25} P_{56}.

And all pathways (a to f) are summed to arrive at the following (contracted) expressions for the menu parameters:

$$U_1 = P_{16} + P_{15} P_{56}$$
$$U_2 = (P_{23}(P_{34} P_{45} + P_{35}) + P_{24} P_{45} + P_{25})P_{56}$$

The information of BIOS is implemented in two stages by two routines called BIOS and BIOX. The program BIOS is run separately from COSMOS to calculate all radionuclide transfer coefficients based on the input of physical and environmental parameters (e.g. types of radionuclides, parameters of the processes involved in the food chain that could vary, time delays, diets, and contaminated fractions of foods).

The program BIOX, also run separately from COSMOS, uses output from BIOS, input indicating a choice of several pre-programmed menus, and a data file containing the dose-from-concentration conversion factors, to calculate and store a table of values for use in COSMOS, representing the product:

$$U(n,i,j)F(h,i,k,s)$$

In the program COSMOS-S/D, the subroutine DOSNUC is called immediately after the submodels representing the physical systems have calculated radionuclide concentrations in air, well water, and surface water, as functions of time. Using these and the information from BIOX, the doses arising from each radionculide are calculated as the sum:

$$D(h,i,k,s,t) = [C(i,j,t)U(h,i,j)F(h,i,j,k,s)]$$

When the doses have been calculated for each radionuclide in this manner, the subroutine DOSTOT calculates the total dose as the sum:

$$D(h,k,s,t) = (D(h,i,k,s,t)$$

and stores this information in the output files. The biosphere modelling is described in Laurens, (1985).

A.2 SYVAC3

A.2.1 Introduction

SYVAC3 has been developed by AECL for the long-term assessment of the disposal of nuclear fuel waste. SYVAC3 was preceded by two earlier versions, SYVAC1 (Dormuth and Sherman, 1981) and SYVAC2 (Sherman et al., 1986). The executive modules in SYVAC3 consist of about 10 000 lines of code and have been developed using an extensive set of quality assurance procedures. These modules were designed so that the code could be readily adapted to assessments of other than nuclear fuel waste disposal (Andres, 1986). SYVAC was designed to use either single values for the input parameters of the simulation, as done in this project, or distributions, as is often needed for the very detailed studies and sensitivity analyses required in site screening and evaluation.

For this project, SYVAC3 is linked with a vault submodel containing the same equations as those used by the LIXY and PEGE codes in COSMOS (Jarvis et al., 1986) describing the leaching behavior of the waste form, and the failure process for the waste containers, respectively. An additional portion of the submodel simulates the transport of the radionuclides released by the containers through the buffer layer surrounding the waste.

SYVAC3 is also linked to a geosphere submodel which is a modified version of GEONET (Goodwin et al., 1986), a transport code providing the analytical solutions to the equations describing the transport of radionuclides through up to 19 separate pathways, each having up to 9 segments with different transport properties. In this case, three pathways are used to simulate the three relevant geologic provinces of New York State, II, IV and V, and three segments in each pathway are used to represent the three generic layers, overburden, bedrock layer No. 1, and bedrock layer No. 2.

With the submodels installed, SYVAC3 transfers the output of the vault submodel, which is the release rate of each radionuclide leaving the vault versus time after vault closure, to the input of the geosphere submodel. The output of the geosphere submodel is then transferred to a routine which multiplies the maximum concentration of each radionuclide leaving the geosphere, from each type of waste and for each province, by a factor appropriate for converting that concentration to maximum doses to man. The time at which these doses are received is also recorded.

While SYVAC3 and COSMOS-S/D solve essentially the same leaching and transport problem, the method of solution is different. In COSMOS-S/D, radionuclide decay chains are not included, and the entire calculation of radionuclide concentrations at the outer boundary of the vault, for instance, can be completed for one time step before proceeding to the next. The SYVAC system generates a response for the source and for each barrier in the vault for the full time history of the simulation before proceeding to the calculations for the geosphere, where this process is repeated. The current biosphere submodel contains no time dependence; the conversion of geosphere releases to consequences takes place at the conclusion of each simulation run.

As described in the following Section, the SYVAC vault submodel contains the source release functions, describing the process that releases the radionuclides from the waste form into the container through a leaching or dissolution action; a simulation of the container corrosion which releases the radionuclides to the buffer in accordance with a prescribed container failure function, and equations representing the buffer material which surrounds the waste and impedes radionuclide movement out of the vault.

In the SYVAC system, all the time-dependent functions describing the release and movement of radionuclides are converted to series of values for a set of time steps spanning the period and the simulation. Then, in the case of the vault submodel, convolutions are performed for each radionuclide to determine first the combined effect of the source release and container failure functions, then the flow from the vault. The result of convoluting the source release function with the container failure function is convoluted with the solution to the buffer transport equation for an impulse output of a given radionuclide. The latter solution, also converted to a time series, is used, approximating the actual continuous flow of radionuclides into the buffer by a series of discrete impulses with the same amplitude as the continuous flow.

The calculated values of intermediate flows from the waste form, containers and buffer can be stored for reference for a tracer nuclide with chosen properties.

In a similar way to that described above, the flow from the vault is convoluted with the impulse solution to the transport equation for the first segment of each pathway in the geosphere. Further convolutions for the succeeding segments then lead to a value for the flow from the geosphere for each pathway and each radionuclide.

Figure A-3 shows the overall logic diagram for a SYVAC3 system. In contrast to COSMOS-S/D, all of the time-independent parameters needed by the submodel calculations are determined as soon as the data has been obtained from the input files and the arrays initialized. A module named SELECT carries out the functions of initializing all the parameter values for each run in the group (called a case), samples values of distributed parameters, and determines whether a run meets the criteria set for inclusion in the case (e.g., value of some parameters must be greater than a given value). The module DEPPAR calculates the time-independent parameter values needed by the submodels. If a run passes the test for continuation, the time-dependent parameters of the run are calculated by the module SIMLAT, using a number of routines to combine time series in the manner required by the submodel equations. When all the required calculations have been completed, the submodel outputs defined as consequences (dose, for example), the sampled parameter values, and the calculated parameter values are all written in the output files for each run by the model WRDATA.

The simulation runs are continued until the number for the case has been reached, at which time the module FINISH writes a case summary in the output files. The summary contains information on the number of successful runs performed, error or warning messages generated, and how much computer time was required for the case.

A.2.2 Vault Submodel

The vault submodel simulates, in the same way for all three provinces, the escape of radionuclides from the waste form, release by the container after container failure, and the transport of the radionuclides through the buffer layer into the geosphere. All of the processes involved in mobilizing and transporting the radionuclides are mediated by water, which enters the vault, penetrates the containers and leaches the radionuclides out of the waste form. In the case of the vault located in a salt formation, no water would enter the vault unless a seal fails on the shafts or boreholes penetrating the overlying impermeable layer of rock, or a pathway forms in the layer through faulting or subsidence. Neither of those events would be likely for a well-selected site and carefully engineered vault, but there is, nevertheless, a remote possibility of water entering the vault in a salt formation during the period that the waste is still hazardous.

As in the case of the other disposal options, the process of container failure has been simulated in accordance with the observational data on the failure of steel drums through corrosion.

As the containers fail, their contents are assumed to become subject to the leaching action of the water present, resulting in the gradual release of radionuclides into the surrounding water. The simulation of the leaching process for the mined vault is identical to that adopted for the other disposal options. The simulation also assumes that the radionuclides diffuse out of the waste form at a rate characteristic of the waste form, under the influence of the differences in concentration of radionuclides in the waste form and the water. This leaching is simulated by solving analytically a one-dimensional diffusional transport equation for the waste form, with the release rate of each radionuclide outside its surface determined by the transport out of the vault through the buffer. The transport equation used here and in all other parts of the SYVAC model incorporates the decay and isotopic transformation of the radionuclides during the period of the simulations.

Radionuclides reaching the water surrounding the waste form are transported through the buffer layer surrounding the waste by a combination of diffusion and convection processes, under the influence of concentration differences and the slow movement of water through the buffer into the surrounding rock. This is simulated by solving an one-dimensional convection-diffusion transport equation analytically. The release rate of radionuclides at the vault-geosphere boundary is calculated by means of the mass-transfer coefficient approach, which provides a computationally convenient and an adequately accurate representation of the movement of radionuclides through the buffer.

The process of radionuclides through the waste form and buffer is simulated as a time-history of the release rate of each radionuclide at the vault-geosphere boundary. To achieve this, the values for the container failure function, and the functions representing the resulting escape of radionuclides by leaching, and their transport through the buffer, are combined in a convolution process for each of a set of times after the vault. The convolutions are performed by the executive program, SYVAC3, using the data generated by the vault submodel. The output of the submodel is a series of radionuclide release rates at the vault-geosphere boundary for the set of times after closure.

As discussed earlier, the ap, [illegible] and equations used for the leaching and container failure function are the same as those used in the corresponding portions of the COSMOS code, to preserve uniformity of treatment for all the disposal methods.

The leaching of radionuclides from the solidified waste and the functions describing their transport through the buffer are solutions to the one-dimensional convection-diffusion equation for different boundary conditions. For a single-member radioactive decay chain, this equation depends upon the following parameters: R, the retardation factor; D, the diffusion/dispersion coefficient; x, the spatial variable; v, the Darcy velocity in the medium; C, the concentration in the groundwater; and lambda, the radioactive decay constant.

The two-region semi-infinite solution used for the leaching comprises a waste layer of finite thickness and with transport parameters D_1 and R_1. The waste layer lies between two semi-infinite layers of buffer material having parameters D_2 and R_2. The Darcy velocity, v, is assumed to be negligible in all layers. The symbol x denotes the spatial coordinate, a and b are additional spatial coordinates. The origin of the coordinate x is taken at the center of the waste layer, in the plane of symmetry.

The boundary conditions are (1) symmetry at x=0; (2) continuity at x=a; (3) $C_1 = kC_2$ at x=a; and (4) C_2 remains finite for all x=a.

Here, k is the partition coefficient relating the concentration of each radionuclide in the waste form to that in the buffer at the interface, x=a. The COSMOS leaching model can now handle the full generality of these conditions, although the presently available description in Fraser and Jarvis, (1985a), is limited to solutions for k=1 and R=1.

For the transport of radionuclides through the buffer, similar solutions are available (LeNeveu, 1987) using the mass transfer approach. In this approach, the boundary conditions are applied to a layer of thickness, b, located between x=0 and x=b. They involve I, the initial amount of the radionuclide present in the waste, and K, the mass transfer coefficient at the waste form-buffer interface.

A.2.3 Geosphere Submodel

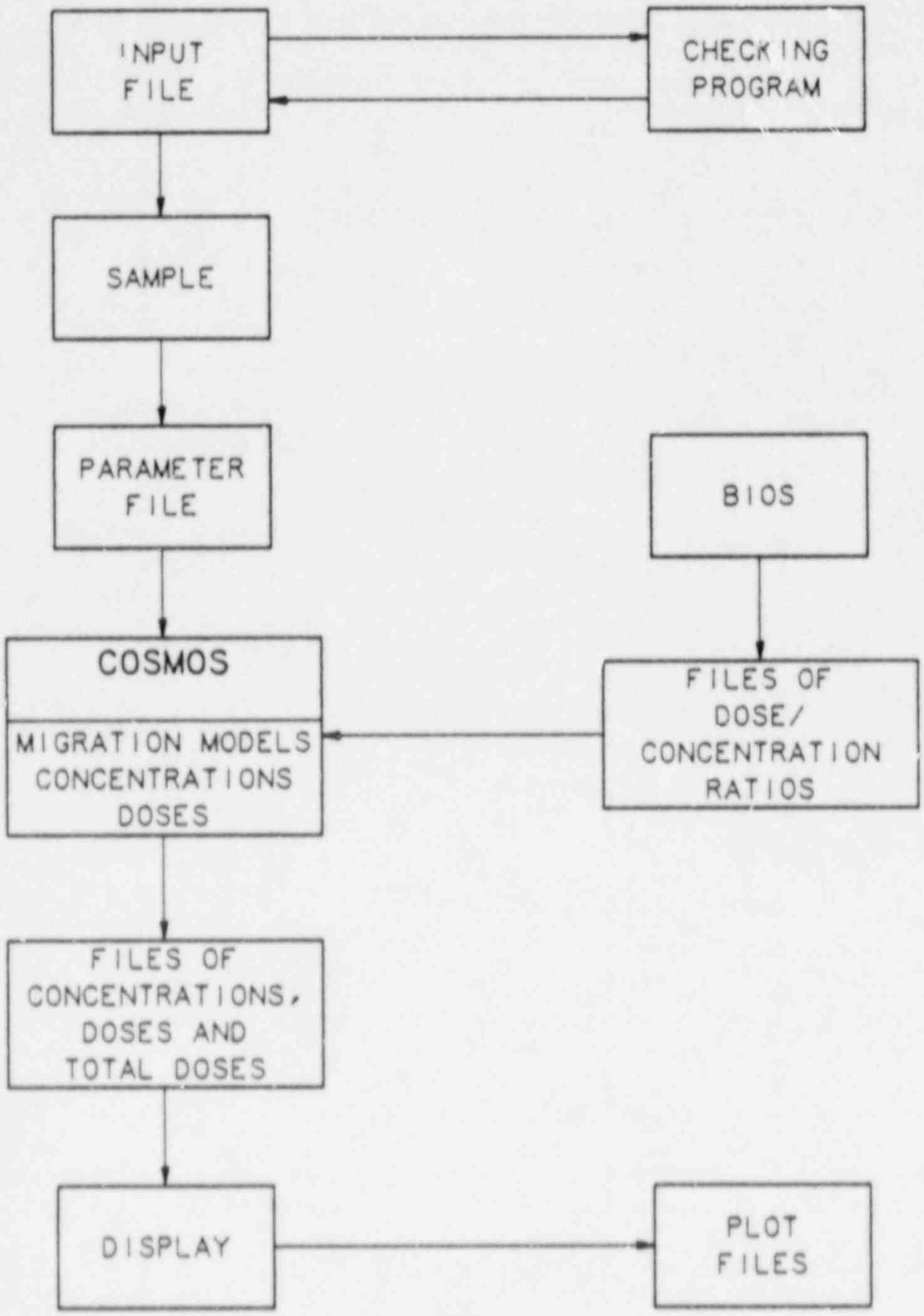
The geosphere submodel is an adapted version of the GEONET code (Goodwin et al., 1987) used as a submodel of SYVAC3-CC3, the program used to simulate the performance of high-activity nuclear fuel waste in a mined vault. The adapted code can simulate the simultaneous movement of radionuclides belonging to decay chains with up to five members, through up to nineteen paths, each pathway having up to nine segments, each with different transport characteristics. The convection-dispersion equation is solved analytically for each segment in each pathway. The SYVAC3 executive program performs the successive convolutions to provide radionuclide flow rates for each pathway at the geosphere-biosphere boundary for a set of times after vault closure. The analytical solutions used are based on semi-infinite boundary conditions at the outlet of each segment, and are published (Heinrich and Andres, 1984). Radionuclide concentrations are converted to concentrations in the groundwater at the geosphere-biosphere boundary using the flow rate of groundwater in the overburden layer.

A.2.4 Biosphere Radionuclide Transfer Coefficients Dose Conversion Factors

The biosphere codes used in SYVAC and COSMOS, while very similar, are not identical. To provide a consistent basis for comparison of the various disposal methods, it was decided to use the COSMOS biosphere code to convert the output of the geosphere submodel to doses to man. The biosphere portion of COSMOS was run as an independent program, with unit concentration input for each radionuclide. Effective committed dose equivalents and committed dose equivalents for the 8 organs listed above were estimated for each of the three relevant provinces of New York State, for both the boundary-well and surface-water scenarios, to produce an array of conversion factors. A routine was linked to the SYVAC3 executive which received the geosphere output concentrations and multiplied each by the appropriate conversion factor from this array to produce the required doses.

A.2.5 Output

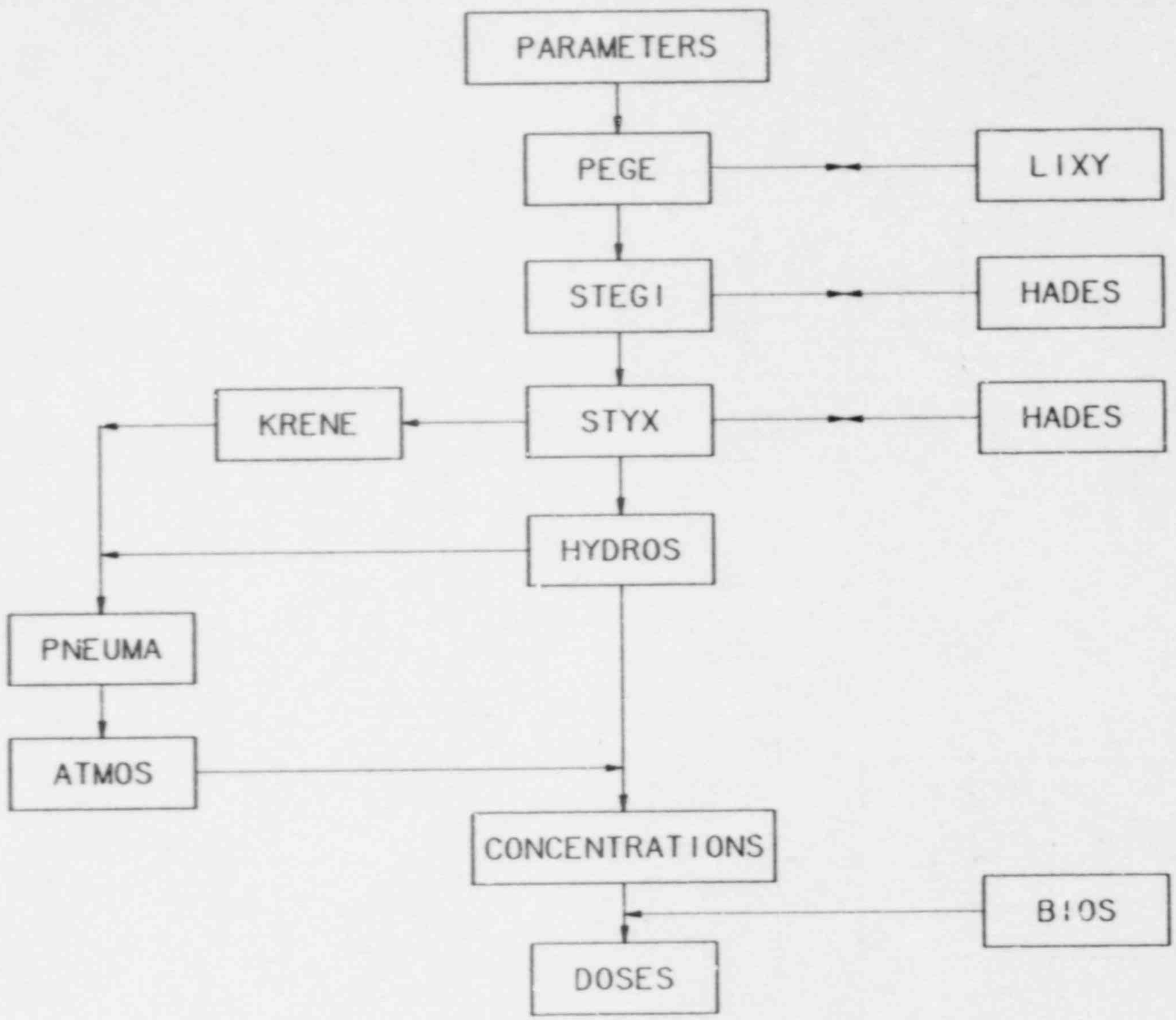
The SYVAC3 executive program contains the codes for performing the time convolutions, linking the input file with the submodels, and providing output to the user. The output is provided as files or as graphical representations of the data. For the present simulations, the results of the simulations are presented in graphical and tabular form indicating the particular scenario for which the simulation was performed, the maximum doses, and the times when these doses were received.



STRUCTURE OF COSMOS
S/D CODE

FIGURE A-1

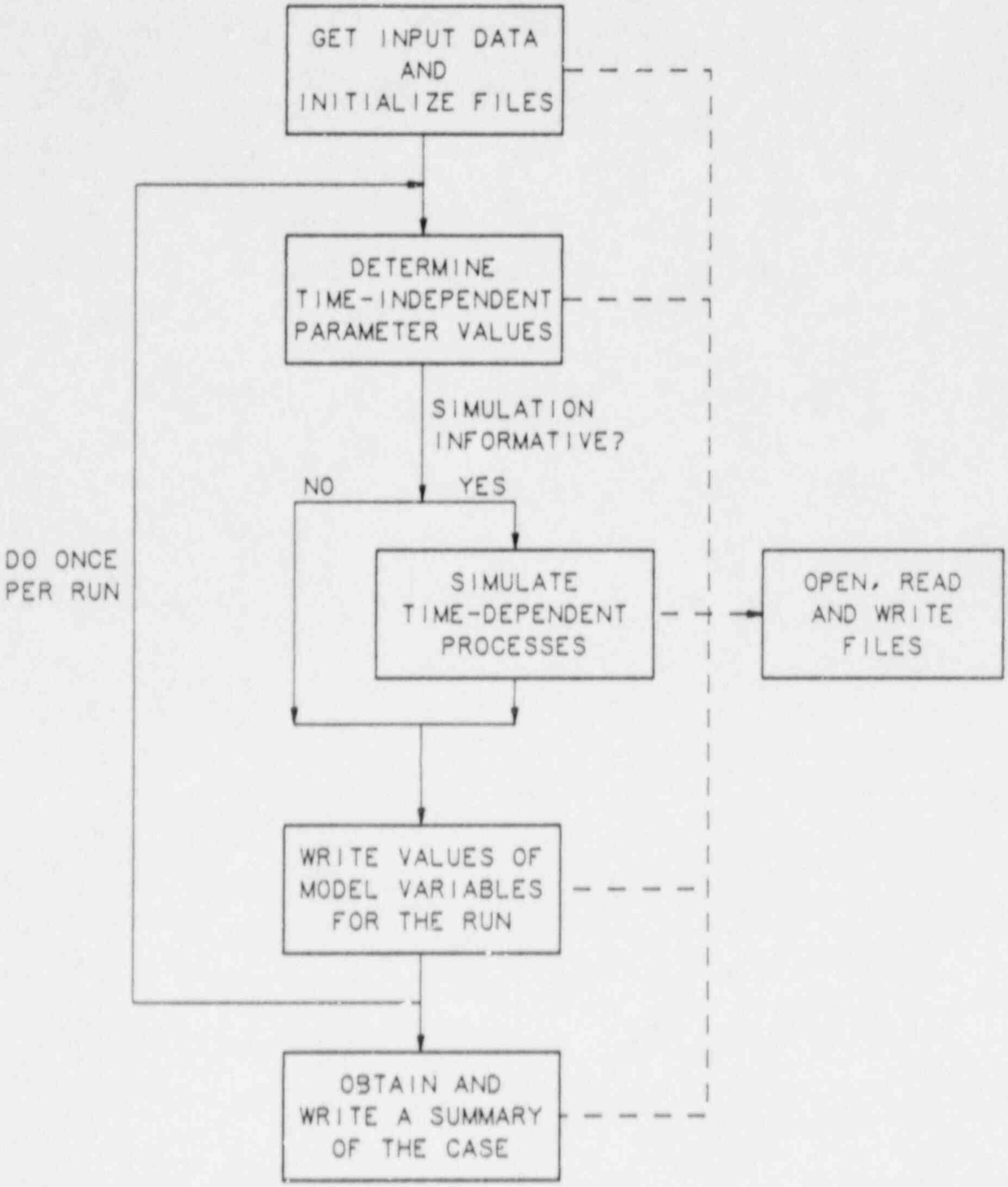
AGRES



MIGRATION MODELS

FIGURE A-2

AGRES



LOGIC DIAGRAM FOR SYVAC3

ACRES

FIGURE A-3

APPENDIX B

Differences Between This Report and the DEIS

A number of factors are responsible for the differences between the results in this report and those in the earlier DEIS.

The models, codes, assumptions, and parameter values were completely reviewed after completion of the DEIS. The COSMOS code was revised in several of its routines to facilitate data input. A new subroutine, SEERA, was developed to allow for groundwater transit times between the rows of vaults assumed for the DEIS. The output was modified so that the committed dose equivalent was calculated for each of eight organs instead of the previous restriction to 'most critical organ'. Although the more important exposures were predicted to occur in the first few thousand years, changes were made so that the code would accommodate various time increments, and thus permit the assessments to be extended to very long times. This was done to provide evidence whether worrisome exposure peaks from the Group 3 nuclides might occur beyond the range of times calculated in the DEIS. The revised codes were thoroughly checked to verify that computing errors were unlikely.

Some of the comments received from readers of the DEIS questioned the appropriateness of some of its modelling assumptions. The assumptions were, therefore, reviewed in the process of developing responses to the comments. Most were judged satisfactory for the purposes of a generic assessment which does not include the effects of the detailed aspects of the site, the design, and the operating procedures. However, major changes were incorporated in the pathways by which carbon-14 could cause exposure of the critical individuals residing at the boundary of the facility.

In the DEIS, it was assumed that well water contaminated with particulate carbon-14 (and the other mobile nuclides) was used to irrigate crops and pasture, as well as for domestic uses and the watering of farm animals. The crops and grass were assumed to take up carbon-14 efficiently through their roots, and thus transfer the nuclide through meat and cereal food chains. As explained in Section 5 (n), advice from the EPA and others led to changes in these assumptions. The combined effect of a change in chemical form, and the elimination of the crop/pasture pathways resulted in about a thousand-fold reduction in the carbon-14 doses from the vault systems (carbon-14 from the mined repository is not important). With the reduction of doses from carbon-14, the dose from iodine-129 became the controlling contribution to the peak effective dose rate equivalent. The peak values, which ranged from 24 to 800 mrem per year for the near-surface vaults in the DEIS, now range from 1.1 to 50 mrem per year. Section 6 contains a complete list of the changes in PDCF's compared with the DEIS values.

The review which followed publication of the DEIS also reexamined the parameter values and other data needed to assess the radiological impacts. Some revisions were made in the generic site characteristics, in particular; the depth to the watertable, in the migration-related parameters, and in the dose conversion factors. The changes in the factors are listed in Section 6.

As mentioned in the previous section, the reduction in the thickness of the unsaturated zone led to a need to mound soil in order to bury the belowground vaults. Aside from a reduction of the length of the migration path, no changes in the model resulted. However, the clear-cut difference in performance which was evident between the aboveground and belowground vaults in the DEIS results, is much less distinct in the revised values. The DEIS and current results are consistent from the point-of-view that the doses arise earlier from the aboveground vaults.

The thyroid doses were still the only significant organ doses in either study, and the effect of the changes varied from case to case. Some decreased slightly, while others increased by factors up to 2.5. However, thyroid doses that exceeded the performance objective in the DEIS estimates, also exceeded it in the revised estimates; and those that met the objective in the DEIS, also met it in the revised ones.

In summary, the new values for the annual effective dose equivalents are much lower than in the DEIS for the vault systems, because of the decreased importance of carbon-14; the results for the underground mined repository are not greatly different. The only significant organ doses in both studies are those received by the thyroid; the values have changed, but are not likely to change conclusions.