Nationale Genossenschaft
für die Lagerung
radioaktiver Abfälle

Société coopérative nationale
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NAGRA

TECHNISCHER BERICHT 82-11

Establishing Storage Criteria for a Swiss Low-Level Waste Repository

December 1982

MOTOR COLUMBUS Consulting Engineers Inc.

Geschäftsstelle Parkstrasse 23 CH-5401 Baden Telephon 056/20 55 11

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FOREWORD

It is the responsability of Nagra to prepare for disposal of all future Swiss radioactive wastes in a manner which will adequately ensure public safety. In the national waste disposal concept three repository types are foreseen for the storage of low, low and intermediate, and high-level wastes (Types A, B, C respectively). It is essential to be able to design the repositories, and to allocate wastes for storage to the appropriate repository type, in such a way as to guarantee compliance with the regulations for ensuring public safety. Consequently, it has been necessary for Nagra to develop a suitable methodology for the allocation of wastes to the appropriate repository type based upon safety criteria derived from Swiss regulatory proposals.

Motor Columbus Consulting Engineers Inc. was requested to carry out the work upon the implementation of the methodology for waste allocation and upon application of this methodology to low-level wastes from decommissioning activities. The present report summarizes the initial phase of this work, as completed at the end of 1982. Subsequent work is being undertaken on improvements to the input data for the Type A repository and on extending the work to the Type B repository.

This report has been prepard by A.L. Smith with assistance from G. Resele and A. Stevens (Motor Columbus Consulting Engineers Inc.) and J.C. Alder, C. McCombie and F. van Dorp (Nagra).

SUMMARY

A methodology based upon safety criteria has been developed for the allocation of future Swiss radioactive waste arisings to one of the three repository types envisaged for the national waste disposal concept. The methodology has been applied to low-level wastes from decommissioning activities, intended primarily for storage in the so-called Type A repository. For this initial application 4 scenarios were considered. Three of these are related to the catastrophic failure of the repository and one to anticipated normal water intrusion.

Maximum average concentrations for 59 radionuclides of interest have been derived for the four scenarios and resulting maximum permissible concentrations for radionuclides, in the waste capable of being stored in a Type A repository have been obtained. The results show 39 radionuclides to be relevant for further investigations and an appropriately grouped specific activity inventory of these 38 radionuclides in decommissioning wastes was established. Using this inventory a first attempt was made to partition the wastes and determine the potential storage capabilities for a Type A repository.

In the work a simple repository model which does not take full account of combined effects of the engineered safety barriers was used. A storage capability of much less than half of the originally envisaged capacity has been arrived at. This result can be accounted for by the differences between present modelling assumptions and parameters and those used previously, by the fact that the partitioning of the wastes was not detailed, and by the conservative assumptions made concerning concentrations of radionuclides in the waste. It is clear that more detailed information on waste arisings and on the behaviour of the technical safety barriers in the repository system can lead to increases in the storage capacity predicted for a Type A repository.

RESUME

Une méthode pour l'attribution des déchets radioactifs produits en Suisse à l'un des trois types de dépôts de stockage ultime envisagés dans le concept national d'élimination de ces déchets a été développée sur la base de critères de sûreté. Cette méthode a été appliquée aux déchets de démantèlement des centrales nucléaires prévus à l'origine pour être stockés, de par leur relativement faible danger potentiel, dans le dépôt dit de type A. Pour cette première application, une analyse conduisant à quatre scénarios potentiels de relâchement des radionucléïdes et d'irradiation résultante a été effectuée. Trois des scénarios sont associés à une défaillance du dépôt par suite d'évènements catastrophiques et le dernier à l'intrusion normale et prévue de l'eau.

La concentration moyenne maximale de 59 radionucléïdes pouvant être impliqués a été déterminée pour chacun des quatre scénarios. La concentration résultante maximale permise pour chaque radionucléïde dans les déchets susceptibles d'être stockés dans le dépôt type A a ainsi été obtenue. Il en ressort que 38 radionucléïdes sont d'importance pour les investigations ultérieures. L'activité spécifique de ces 38 radionucléïdes a été établie par sous-groupes appropriés des déchets de démantèlement. A partir de l'inventaire ainsi fait, un premier essai de répartition des déchets et de détermination de la capacité de stockage du dépôt type A a été entrepris.

Ce travail repose sur un modèle simplifié de dépôt qui ne prend pas en compte tous les effets protectifs combinés des barrières artificielles de sécurité. On obtient une capacité de stockage sensiblement en dessous de 50 % de celle escomptée à l'origine. Ce résultat peut être expliqué par les différences entre les présentes hypothèses et valeurs de paramètre du modèle et celles utilisées auparavant ainsi que par le fait qu'un inventaire des déchets peu détaillé par sous-groupes et conservatif dans les concentrations de radionucléïdes a été utilisé. Il est clair qu'une information plus précise à la fois des déchets et des phénomènes dans la région centrale du dépôt sera nécessaire avant qu'aucune amélioration des résultats présentement obtenus puisse être attendue.

ZUSAMMENFASSUNG

Ausgehend von nuklearen Sicherheitskriterien ist eine Methode zur Zuteilung der zukünftig in der Schweiz anfallenden radioaktiven Abfälle in einem der drei vorgesehenen Endlagertypen entwickelt worden. Die Methode wurde auf den schwachradioaktiven Abfall aus dem Abbruch von Kernkraftwerken angewendet, der hauptsächlich im sogenannten Typ A Endlager deponiert werden soll. Für diese erste Anwendung wurden vier Szenarien, die zur Freisetzung von Radionukliden und zu Strahlenexpositionen führen könnten, analysiert. Drei der untersuchten Szenarien setzten ein katastrophales Versagen des Endlagersystems voraus und eines betrifft das Eindringen von Wasser in das Endlager.

Maximale Durchschnittskonzentrationen von 59 Radionukliden wurden für die vier Szenarien berechnet und daraus die maximalen Konzentrationen dieser Radionuklide im Abfall abgeleitet. Die Resultate zeigen, dass 38 Radionuklide weiter untersucht werden müssen. Das Inventar dieser 38 Radionuklide in Stillegungsabfällen wurde aufgestellt. Mit Hilfe dieses Inventars wurde der in der Schweiz anfallende radioaktive Abfall den Endlagertypen zugeteilt und dabei die erforderliche Kapazität des Endlagers Typ A bestimmt.

Der Arbeit liegt ein vereinfachtes Modell eines Endlagers zugrunde, das die kombinierte Wirkung der technischen Sicherheitsbarrieren nicht voll berücksichtigt. Die vom Lager Typ A aufzunehmenden Abfallmengen entsprechen weniger als der Hälfte der ursprünglich angenommenen Lagerkapazität. Dieser Unterschied lässt sich dadurch erklären, dass Voraussetzungen und Parameter des Endlagermodells anders angesetzt waren als beim früher verwendeten Modell, und dass die Zuteilung des Abfalls nicht aufgrund der Detailkenntnis der Nuklidkonzentrationen im Abfall erfolgte, sondern auf einer groben, konservativen Schätzung des Inventars beruhte. Es ist klar, dass noch weitere Detailangaben sowohl über den anfallenden Abfall als auch über das Verhalten der technischen Barrieren des Lagersystems vorliegen müssen, bevor Verbesserungen der hier gewonnenen Resultate erwartet werden können.

CONTENTS

ESTABLISHING STORAGE CRITERIA FOR A SWISS LOW-LEVEL WASTE REPOSITORY

1 INTRODUCTION

All future Swiss nuclear wastes are assumed to have to be stored within Switzerland. Since it is obviously uneconomic to dispose of the anticipated large volume of low-level decommissioning wastes together with high-level reprocessing wastes in a deep geological repository, for example, a variety of possible repository types therefore require investigation. On the basis of an analysis of the rate of flow of water required to dilute leached radionuclides, from the wastes from a 240 GWe-yr. program, to permissible Swiss drinking-water limits, three categories of wastes have been defined /1/. Three different repository types: A, B, C, have been defined as being appropriate for the storage of the three waste categories labelled: low-, low- and intermediate-, and high-level wastes, respectively.

To summarize the current status of the national waste-disposal concept with regard to repository types, it may be observed that the Type A repository is planned as a horizontally accessed engineered facility, on a hillside, primarily storing decommissioning wastes, plus low-level reactor operational wastes and some wastes from medical, research and industrial applications. The Type B repository is planned as an engineered vault with stable surrounding geological media of up to 100 - 600 m radius. Intermediate-level wastes possibly containing limited quantities of longer-living alpha-bearing nuclides are foreseen as candidate contents. Finally, the Type C repository will be a deep geological facility (>600 m) intended for storing high-level or very long-living wastes.

The organization responsible for the disposal of all future Swiss nuclear wastes in a manner which will adequately ensure public safety is NAGRA (Nationale Genossenschaft für die Lagerung radioaktiver Abfälle). A quantitative radiation protection (safety) objective has been formulated by the Swiss Regulatory Authorities and requires that the maximum possible individual exposure rate from any waste repository shall not exceed 10 millirems per annum /2/. This stipulation imposes constraints for NAGRA on both the inventory of wastes capable of being stored within a specific repository type and the

repository safety-feature engineering-design criteria.

To assist in establishing content constraints and design criteria for the various repository types which will result in economically constructed repositories able to meet the regulatory exposure limit, detailed modelling of the consequences of accidental and/or natural releases from the repositories needs to be undertaken. On the basis of such modelling, it should theoretically be possible to determine the maximum permissible concentration of wastes capable of being stored in the respective repositories, together with safety-barrier requirements. The procedure is, naturally, an iterative one, modelling refinements being incorporated at each iteration and, thereby, successively improved results being obtained for the relevant constraints.

In this report the methodological approach by which storage critera may be established, and required modelling assumptions, are first presented. Thereafter, relevant parameters are specified, maximum average and permissible concentrations of waste material are calculated and the sensitivity of the parameters is investigated. A hypothetical, but plausible, decommissioning waste inventory is established and Type A repository storable waste quantities are derived. Effects of engineered-barrier-relevant parameter variations are considered and conclusions are drawn. The primary objective has been to develop and apply the methodology. Detailed engineering design cannot be based on the results derived herein. Such design-basis information must await more precise input data and modelling refinements before the methodology presented in this report can be applied to the detailed engineering design of an actual repository.

2 METHODOLOGICAL APPROACH

2.1 Background

In view of the Swiss regulatory requirement that the maximum possible individual exposure rate from a waste repository shall not be in excess of 10 mrem per annum, the following procedure for establishing repository storage criteria has been adopted:

i) An inventory of waste types, entailing a practical grouping of wastes with similar nuclide inventories and conditioning, must be established and a first estimation of the likely allocation to the repository type in question must be made.

ii) From the inventory of waste types, a list of potentially relevant radionuclides must be established.

iii) Potential scenarios by which the population might be exposed to radiation doses from the contents of the repository must be defined.

iv) For each individual radionuclide and for each scenario, the maximum average concentration of radionuclide (Ci/m3) capable of being stored in the particular repository, without exceeding the regulatory exposure limit, must be calculated by use of appropriate algorithms. (This procedure is sometimes referred to as "reverse consequence modelling").

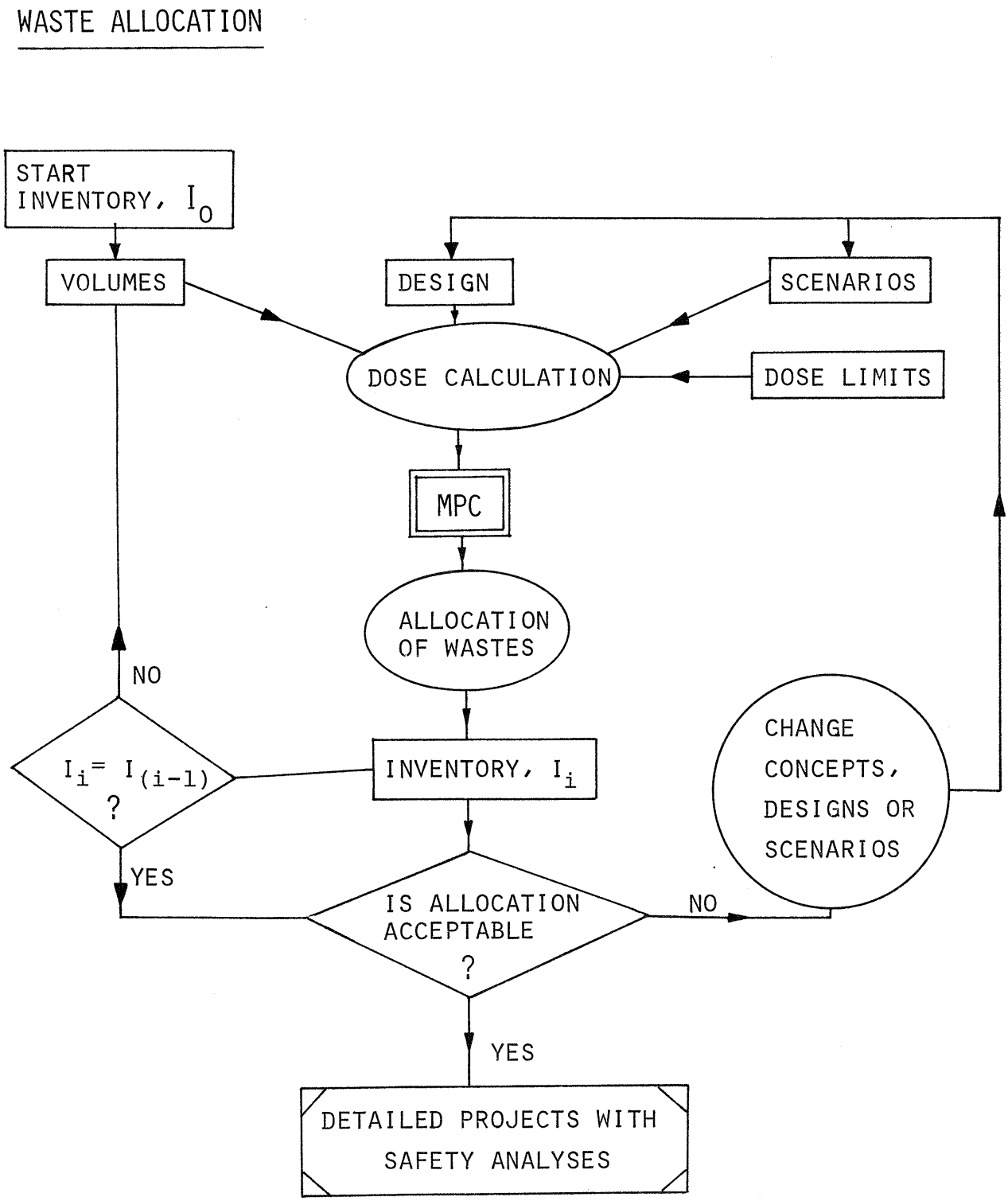
v) From the determined radionuclide maximum average concentrations (MAC), repository and scenario specific maximum permissible concentrations (MPC) may be determined, as explained in the next section. (It should be noted that MAC and MPC are used in this report for concepts other than those to which the terms are normally applied in the radiation protection literature.)

vi) The MPC of the mixture of waste radionuclides can finally be determined (as explained in the next section) and this figure is then compared with the activity concentration of the waste types established at the outset.

This final step in the procedure enables a decision to be reached as to whether the waste inventory estimated at the outset can safely be stored in the repository being considered. If not, the non-storable portion of the inventory must then be allocated to a "superior" repository. Alternatively, possible refinements may be made to the storage concept, repository design (including safety barriers), scenarios or initial inventory and incorporated into a procedural repetition of the methodology, to increase the storable quantities of waste.

In principle, an iterative variational analysis should be undertaken, as depicted in Fig. 1, to derive as much information on repository storage criteria as possible. This procedure should be applied, successively, to the different repository types envisaged, to obtain storage criteria for all three Swiss waste repository concepts. The present study has been restricted to the most basic repository, that intended primarily for the storage of low-level (decommissioning) wastes.

Fig. 1: Iterative Procedure for Determining Waste Allocation



2.2 Definitions of Maximum Average and Maximum Permissible Concentrations

As explained in the previous section, maximum average concentrations for the potentially relevant radionuclides for each scenario may be obtained from reverse consequence modelling. The algorithms used to determine these maximum average concentrations (MAC) are specified in the next chapter. To re-iterate, the MAC for each scenario is the concentration of radionuclide which, when stored in a specified repository which is subjected to the release and exposure scenario considered, would lead to a 10 mrem per annum exposure limit. However, since different scenarios have different applicable MACs and since differences in the probabilities of occurrences of the different scenarios are not considered, it is the lowest value of the MAC for any one radionuclide that is of relevance. This value is termed the maximum permissible concentration (MPC).

From the reverse consequence modelling, a matrix of MACs will be obtained, as shown below, where MAC_{mn} represents the maximum average concentration (in Ci/m3) of nuclide N_m for scenario S_n.

	Scenario			
Nuclide	S_1	S_2	.....	S_n
N_1	MAC_{11}	MAC_{12}	.....	MAC_{1n}
N_2	MAC_{21}	MAC_{22}	.....	MAC_{2n}
.	.	.		.
.	.	.		.
.	.	.		.
.	.	.		.
N_m	MAC_{m1}	MAC_{m2}	.....	MAC_{mn}

The maximum permissible concentration (MPC) of nuclide N_m is defined by

$$MPC\ (N_m) \equiv \min \left\{ MAC_{m1},\ MAC_{m2},\ \ldots\ldots,\ MAC_{mn} \right\}$$

and the MPC of a mixture of m nuclides is defined by

$$1/(MPC)_{mixture} = \sum_{i=1}^{m} \left[f_i \ / \ MPC(N_i) \right] \qquad - (1)$$

where f_i is the fractional radioactivity of the ith nuclide in the mixture. Note that the $(MPC)_{mixture}$ value thus obtained is conservative, since implicit in it is the assumption that radiation doses from all m radionuclides occur simultaneously, which would not necessarily be the case in reality.

2.3 Scenario Analysis

To be able to determine, via reverse consequence modelling, MAC (and thence MPC) values, potential scenarios by which the population might be exposed to radiation doses from the contents of the repository must be defined or identified. The process of identifying relevant scenarios is termed "scenario analysis". A scenario is here defined /3/ as a set of conditions or events which could influence:

i) the release of radionuclides from a waste repository

ii) transport of these radionuclides through the subsurface and surface environment

iii) intake or exposure of these radionuclides by or to humans.

A scenario, therefore, is really a combination of subscenarios, where each of the subscenarios represents one of the following subsets of conditions or events:

1) Engineered safety-feature (barrier) failure events resulting in the potential for radio nuclide release to the geosphere or biosphere. Examples include breaching stresses on the waste immobilizing or containing material, deterioration or failure of the waste-packaging material, etc. Such subscenarios will be referred to as barrier-degradation (or B-type) subscenarios.

2) Subsurface conditions existing in the vicinity of a repository which could result in the transport of potentially releasable radionuclides into the geosphere. Examples include the existence of undetected geological faults, magmatic activity, geological fracturing, tectonic processes, etc. Such subscenarios will be referred to as release (or R-type) subscenarios.

3) Subsurface conditions existing in the vicinity of a repository which could result in the transport of released radionuclides through the geosphere and into the biosphere. Examples are the existence of hydrological pathways, both normal water transport and the effects of perturbations such as thermally

induced groundwater buoyancy or water-table fluctuations due to irrigation activities, etc. Such subscenarios will be referred to as transport (or T-type) subscenarios.

4) Events occurring in the vicinity of a repository which could result in the release of radionuclides directly into the biosphere. Examples include the impact of a meteor in the vicinity of the repository, landslides, glacial erosion, weapons explosions, resource recovery (drilling), etc. Such subscenarios will be referred to as disinterment (or D-type) subscenarios.

5) Environmental conditions existing in the vicinity of a repository which could result in radionuclides which are released into the biosphere impinging (through intake or exposure) on humans. Examples include events such as dust storms, irrigation activities, etc. Such subscenarios will be referred to as exposure pathway (or P-type) subscenarios.

A combination of subscenarios 1, 2, 3, and/or 4, together with 5, represents a set of conditions or events which could result in or affect the release of radionuclides from a repository, their transport through the subsurface and/or surface environment and subsequent human intake or exposure. Consequently, such a combination represents a scenario.

A detailed scenario analysis would involve establishing a list of all possible combinations of subscenarios, from which irrelevant scenarios would be discarded. For each of those remaining, occurrence probability and consequence should then be established using, for example, event-tree techniques. Since reverse consequence modelling is being employed, all consequences are considered to be one-and-the-same (10 mrem per annum individual exposure rate) and, for simplification at the present stage of the analysis, instead of determining probabilities of occurrence, a few so-called "bounding" scenarios have been established for use in the reverse consequence modelling. The establishment of such bounding scenarios is discussed subsequently.

2.4 Influence of Safety Barriers

Safety barriers, both natural (geological formations) and man-made or engineered (packing material, repository liner, etc.) play an important role in isolating the buried wastes from the biosphere. The retention capabilities of safety barriers have a two-fold effect:

i) For the time during which the barriers remain intact, the radioactivity level of the waste decreases, due to natural radioactive decay processes.

ii) When the engineered barriers eventually cease their retaining role (as a result of barrier degradation processes) there will be a distribution over time of barrier failure. Consequently, not all the waste will become available for potential release at the same time. The resultant nuclide transport retardation further reduces ultimate potential exposure levels according, once more, to natural radioactive decay.

In any thorough analysis of storage criteria, a probabilistic consideration of barrier failure should be included. However, since an assessment of the detailed physico-chemical mechanisms controlling engineered barrier degradation processes in a waste repository lies beyond the scope of this investigation, a more straightforward approach has had to be adopted. This approach entails first assuming that at a specified time (to be established subsequently) all engineered barriers fail simultaneously (i.e., all subscenarios, 1, in the previous section, are assigned an occurrence probability of unity). The analysis is then performed making this simplifying assumption. Thereafter, engineered-barrier-relevant parameters are varied in an attempt to assess the influence of barrier integrity upon storage criteria. The determination of absolute values of correlations between barrier behavior and storage criteria was beyond the scope of the present study and must await future improved knowledge of repository site-specific barrier degradation processes.

3 MODELLING ASSUMPTIONS FOR A LOW-LEVEL WASTE REPOSITORY

3.1 Generic Repository Model and Safety Barrier Assumptions

A generic model for the repository is assumed to be a spherical volume of space surrounded by shells, or layers, corresponding to n, various, safety barriers. Obviously, such a simple concept needs to be modified when either repository design features have to be incorporated into the modelling or a high-level waste repository is being considered. However, this concept suffices for present purposes.

Although the low-level (Type A) repository is planned as a horizontally accessed engineered facility built some 200 m deep into the side of a hill, no credit was given in the original concept for the effect of a geological safety barrier. For the Type A repository, the safety barriers are considered to be engineered features only. (For the Type B repository, an additional geological safety barrier is foreseen.) Possible engineered safety barriers include the waste-immobilizing medium, the immobilized-waste containment material, buffer (filling or separating) material, repository liner material and/or repository backfill material.

The following assumptions regarding barriers are made (where n = 1 corresponds to the <u>innermost</u> barrier):

n = 1 (waste-immobilizing medium) represents concrete (possibly bitumen) with a physical thickness which is not relevant, since the waste is assumed to be homogeneously dispersed throughout the immobilizing material

n = 2 (immobilized-waste containment material) represents carbon steel drums with a physical thickness of approximately 2 mm, or, possibly, concrete cylinders

n = 3 (immobilized-waste containment and separating material) represents concrete boxes corresponding to Swiss GT III containers

n = 4 (filling or separating material) represents concrete

n = 5 (repository liner material) represents concrete with a minimum physical thickness of approximately 30 cm.

Backfill is disregarded for the Type A repository and the geological barrier is ignored. The volume of stored, immobilized, wastes is assumed to be approximately 30,000 m3, corresponding to the envisaged net volume of unpackaged immobilized decommissioning wastes from Swiss nuclear power plants already in operation or under construction.

To summarize, the complex, inhomogeneous, contents intended for storage and consisting of drums and GT III containers into which concrete or bituminized immobilized wastes have been placed and which, once in the repository, are surrounded by additional engineered barriers, are here represented by a homogeneous sphere of immobilized wastes, surrounded by a thin layer of steel and a subsequent thick layer of concrete corresponding to barriers 2, 3 and 4, above. The volume of the immobilized waste is approximately 30,000 m3 and the volume of the surrounding concrete is considered to be at least 15,000 m3, the total minimum volume of the repository model therefore being roughly 45,000 m3. Barrier 5 is, for present purposes, ignored.

A final assumption, which can be regarded as a repository "boundary condition", is the assumption of an isolation period, subsequent to sealing the repository, during which <u>at least one</u> safety barrier may be regarded as remaining unbreached and, therefore, ensuring no release of the contents to the biosphere. For the combination of engineered barriers assumed above, and on the basis of present administration policies, a not-unrealistic (minimum) isolation period may be taken to be 100 years.

3.2 Scenario Specifications

In Section 2.3, a scenario was defined as an appropriate combination of barrier-degradation, release, transport, disinterment and exposure pathway subscenarios. For the present series of calculations, the number of subscenarios may be reduced by making the simplifying assumption that at the end of the isolation period, all safety barriers fail simulataneously. This corresponds to ignoring barrier-degradation and release subscenarios which, in turn, implies not giving credit to engineered and geological barriers subsequent to the expiry of the isolation period. It is very conservatively assumed for the present, therefore, that at the termination of the isolation period, all immobilized waste is either released directly into the biosphere or it is introduced into the geosphere in a situation where it can be directly transported through the geosphere.

The following examples of possible subscenarios, applicable for the Type A repository, may be listed:

Transport

T_1 absence of any perturbing features in the surrounding geological medium

T_2 existence of a water-withdrawal well sunk into an aquifer in the surrounding geological medium

T_3 existence of some form of weakness in the surrounding geological medium

Disinterment

D_1 catastrophic failure, e.g. occurrence of a landslide or major earthquake

D_2 inadvertent future intrusion by drilling

D_3 major non-geologic disruptive event, e.g. meteor impact, explosion, etc.

Exposure Pathway

P_1 "normal", undisturbed, atmospheric conditions

P_2 existence of "windy" atmospheric conditions

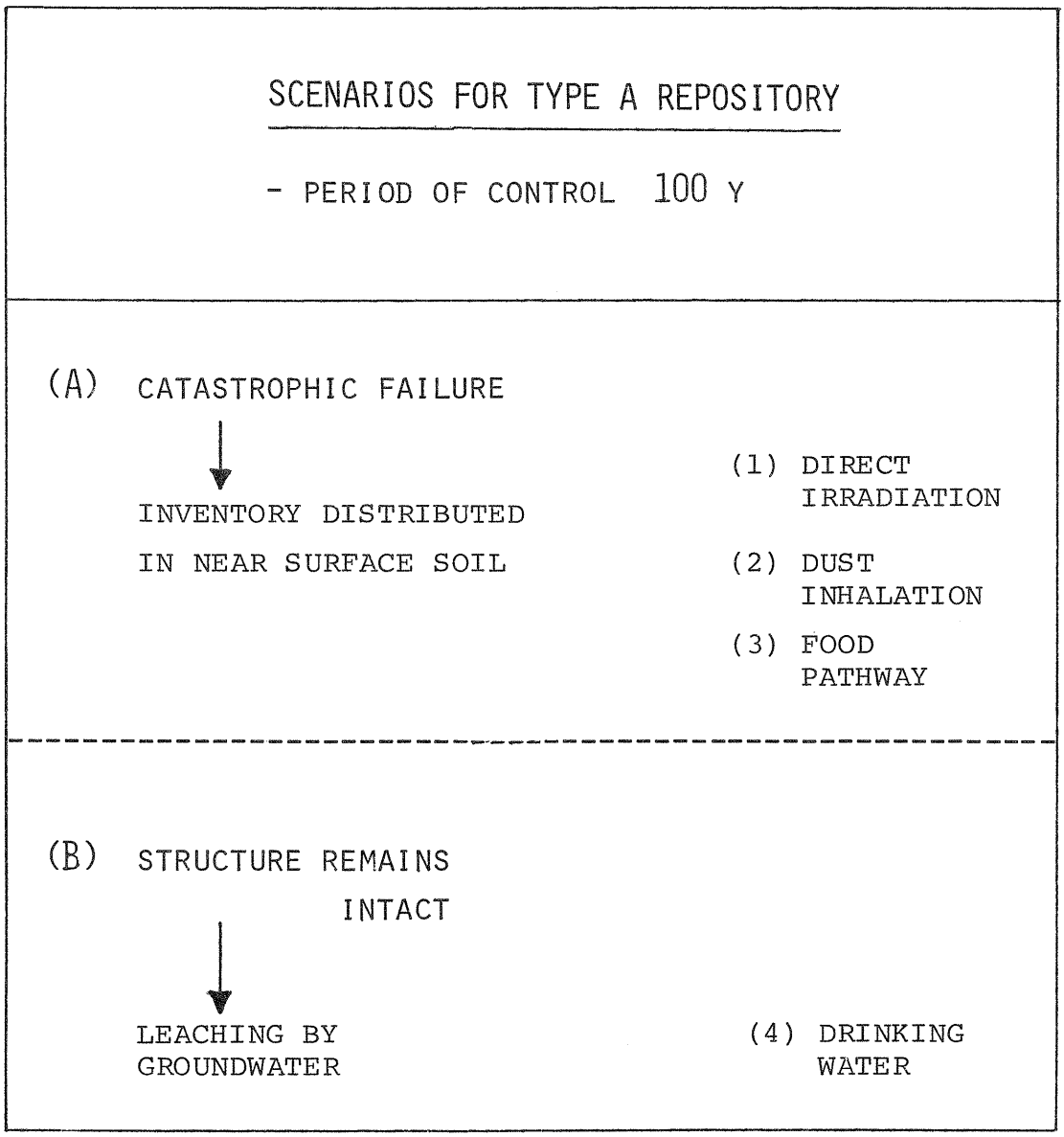
P_3 the utilization of well water from T_2 for drinking purposes

P_4 the utilization of well water from T_2 for irrigation purposes

P_5 occurrence of agricultural activities.

By an application of event-tree methodology, relevant scenarios may be specified from all the various combinations of subscenarios.

Fig. 2: "Bounding" Scenarios



3.3 "Bounding" Scenarios

From the number of possibly relevant scenarios, four "bounding" scenarios have been selected. These are considered to be the most significant since, although occurrence probabilities are expected to be very low, consequences are expected to be the most severe and hence the use of the term "bounding". Three of the scenarios are related to the catastrophic failure of the repository and one to anticipated normal water intrusion. The respective scenarios are depicted in Fig. 2.

The first scenario (direct irradiation) is a combination of subscenarios T_1, D_1, P_1 from the previous section and corresponds to the situation where a landslide or major earthquake disinters effectively all of the waste and distributes it on the ground surface around the former repository. Members of the nearby population are subsequently exposed to direct-gamma radiation from the disinterred waste.

The second scenario (dust inhalation) is a combination of subscenarios T_1, D_1, P_2 from the previous section and corresponds to a situation where, as in the previous scenario, a landslide or major earthquake disinters effectively all of the waste and distributes it on the ground surface around the former repository but, either during disinterment or subsequently, dust (consisting of an admixture of soil with radioactive material) is blown around and subsequently inhaled by members of the nearby population.

The third scenario (food pathway) is a combination of subscenarios T_1, D_1, P_5 from the previous section and corresponds to a situation where, as in the previous two scenarios, a landslide or major earthquake disinters effectively all of the waste and distributes it on the ground surface around the former repository. It is now assumed, however, that after a passage of time, during which the land surrounding the repository is grassed-over, the area is then inadvertently used for agricultural purposes. Any foodstuffs produced on the radio-contaminated land will, themselves, be contaminated and subsequent ingestion will result in internal radiation doses.

The fourth scenario (drinking-water) is a combination of subscenarios T_2 and P_3 from the previous section and corresponds to a situation where radionuclides are leached out of the disposed immobilized waste as a result of eventual water intrusion into the repository. These radionuclides then rapidly migrate into an aquifer and consequently contaminate water in a well sunk into the aquifer.

The above four scenarios are not only "conservative" in the sense that they result in the most severe consequences, but they also incorporate the various differing exposure mechanisms by which doses might be received. In this way, the effects of differing impacts for a specific radionuclide can be compared.

To illustrate the approach used in the selection of the "bounding" scenarios, two examples may be cited. Firstly, scenarios T_2P_3 and T_2P_4, which are similar "normal" water intrusion scenarios but which have differing exposure mechanisms - the former being direct intake, the latter being bioaccumulation followed by ingestion - obviously result in differing individual dose rates. Since radionuclides will be dispersed and have subsequently further decayed in the scenario involving irrigation (T_2P_4), the scenario involving direct drinking (T_2P_3) was regarded as the more "conservative" of the two.

The second example involves scenarios $T_1D_1P_1$ and $T_1D_2P_1$. Both involve direct-gamma radiation exposure from disinterred waste. However, whereas the latter results from a small drilling sample extracted from the repository, the former results from effectively the entire contents of the repository and, therefore, may be regarded as the more "conservative" of the two.

3.4 Maximum Average Concentration Algorithms

Algorithms for determining the MAC (Ci/m3) for each nuclide, N_i, of interest, for the previously specified bounding scenarios are detailed below. In the subsequent discussion, the direct irradiation scenario ($T_1D_1P_1$) will be referred to as scenario S_1, the dust inhalation scenario ($T_1D_1P_2$) as S_2, the food pathway or food uptake scenario ($T_1D_1P_5$) as S_3 and the drinking-water or water uptake scenario (T_2P_3) as S_4.

3.4.1 Direct Irradiation Scenario - S_1

It is assumed that the stored radioactive waste material is completely disinterred and scattered in an approximately uniform distribution on the ground surface around the repository.

The disinterred material consists of a mixture of immobilized waste together with concrete container material and soil. The crude assumptions is made that the situation can be modelled by a homogeneous mixture. The $(MAC)_i$ in such case will be given by

$$(MAC)_{i1} = d \cdot T_i / (f \cdot t \cdot (DF)_i \cdot \eta_i) \qquad - (2)$$

where T_i is the radionuclide decay reduction factor resulting from the isolation period and is given by

$$T_i = \exp (T' \lambda_i) \qquad - (3)$$

where T' corresponds to the repository isolation period (years) and λ_i is the appropriate radionuclide decay constant.

The other symbols have the following usage:

d = regulatory exposure limit (rem/yr.)

f = volume ratio of immobilized waste to immobilized waste plus container and backfill material

t = exposure period per year (h/yr.)

$(DF)_i$ = 1 m body-surface dose-exposure conversion factor (rad(rem) . m3/Ci . h), as tabulated in /4/, for nuclide N_i

η_i = whole-body correction factors, as tabulated in /4/, for nuclide N_i

In this scenario, shielding of the radioactive waste material by any of the original safety barriers and repository overburden of rock and soil is (conservatively) ignored.

3.4.2 Dust Inhalation Scenario - S_2

Assuming, once again, that the stored radioactive waste material is completely disinterred and scattered in an approximately uniform distribution on the ground surface around the repository, the $(MAC)_i$ is given by /5/ as:

$$(MAC)_{i2} = 10^3 \cdot d \cdot T_i \cdot \rho / (A \cdot U \cdot f \cdot f' \cdot t \cdot (DF)'_i) \qquad - (4)$$

where the factor 10^3 is incorporated for consistency in the use of units and the symbols are as used previously, with

ρ = immobilized waste density (g/cm3)

A = dust loading in the air (kg/m3)

U = breathing rate of exposed individual (m3/h)

f' = volume ratio of disinterred immobilized waste present in surface layer of soil to total disinterred immobilized waste

$(DF)'_i$ = 50 yr. whole-body-averaged inhalation dose-rate conversion factor (rem/Ci) as derived from ICRP 30, for nuclide N_i

3.4.3 Food Uptake Scenario - S_3

The same assumptions are made here as were made previously and it is further assumed that after some years subsequent to the landslide which disinterred the waste, the upper layer of soil is cultivated for agricultural purposes and the vegetable and animal produce is then consumed by humans. For this scenario, the $(MAC)_i$ is given by /6/ as:

$$(MAC)_{i3} = 10^3 \cdot d \cdot T_i \cdot \varrho / (B_i \cdot f \cdot f' \cdot F \cdot (u_1 c_{1i} q + u_2 c_{2i} q + u_3) \cdot (DF)''_i) \quad - (5)$$

where the factor 10^3 is incorporated for consistency in the use of units and the symbols are as used previously, with

B_i = vegetative bioaccumulation factor for nuclide N_i

F = fraction of annual food consumption which is produced in the contaminated region

u_1 = usage factor for meat (kg/yr./person)

u_2 = usage factor for milk (kg/yr./person)

u_3 = usage factor for total plant produce (kg/yr./person)

c_{1i} = bioaccumulation factor for meat (day/kg), for nuclide N_i

c_{2i} = bioaccumulation factor for milk (day/kg), for nuclide N_i

q = consumption rate for animals (kg/day/animal)

$(DF)''_i$ = 50-yr. whole-body-averaged ingestion dose-rate conversion factor (rem/Ci) as derived from ICRP 30, for nuclide N_i

3.4.4 Water Uptake Scenario - S_4

It is assumed that, upon termination of the isolation period, the repository is flooded with water and leaching of radionuclides from the waste material commences. In reality, this leaching process will not commence instantaneously since time of commencement (as well as duration) of leaching depends in a complicated way upon the integrity of the engineered safety barriers and their geometrical disposition within the repository, as well as on leachant characteristics such as flow rate, etc.

For present purposes, instantaneous onset of leaching is assumed to occur and it is, further, assumed that engineered barrier degradation is such that the rate of removal of the radionuclides from the immobilized waste can be described by an exponential decay function /7/ where the time-constant for this removal process is related to the standard leach-rate as follows:

$$\mu_i = 365 \cdot L_i \cdot E/(V \cdot \rho) \qquad - (6)$$

where the symbols are as used previously and

μ_i = time-constant for removal of radionuclide N_i ($yr.^{-1}$)

L_i = leach rate for radionuclide N_i (g/cm2 . day)

E = surface area of waste material (cm2)

V = net volume of immobilized waste material (cm3)

Once the radionuclides have been leached out of the waste material, it is further (conservatively) assumed that they are transported almost instantaneously into an aquifer into which a well has been sunk. In other words, no allowance is made for dispersion (and subsequent radioactive decay) and sorption which would normally be expected to occur. The contaminated well water is then drunk by some human and the $(MAC)_i$ is given by /8/ as

$$(MAC)_{i4} = d. T_i \cdot M/(W \cdot \mu_i \cdot \emptyset_i \cdot (DF)''_i) \qquad - (7)$$

where the symbols are as used previously and

M = water flow rate at entry of well into aquifer, expressed in units of waste volume (l/yr./m3)

W = water uptake rate for humans (l/yr.)

$\emptyset_i$ = radionuclide geospheric transport decay reduction factor (not considered here and, therefore, set equal to unity).

4 CALCULATIONAL INPUTS

4.1 Algorithm Parameter Specifications

Parameters (in some cases more precisely than warranted at the current level of analysis) have been specified as follows:

- $T' = 100$ yr.
- $d = 10^{-2}$ rem $yr.^{-1}$ (regulatory limit)
- $f = 0.67$ (Since 30,000 m3 net of immobilized waste is assumed to occupy a total volume of 45,000 m3 including containers, etc.)
- $t = 1{,}000$ h $yr.^{-1}$ for S_1

  5,000 h $yr.^{-1}$ for S_2 (average from /9/ and /10/)
- $\rho = 2.3$ g cm^{-3} (which is the minimum density of concrete from /11/. Since the waste is a mixture of bitumen, concrete and metal, the average bulk density is assumed to be that of concrete)
- $A = 5 \times 10^{-8}$ kg m^{-3} (estimated from values given in /9/ and /10/)
- $U = 0.83$ m3 h^{-1} (ICRP 23)
- $f' = 0.9$ for S_2 (this implies that the content of immobilized waste - excluding containers and backfill material - in the ground surface layer is 90 %)

  0.5 for S_3 (this is more conservative than NUREG analysis assumptions of 0.1 - 0.2 and is decreased from 0.9, to make allowance for additional soil admixture to be present in order to permit agricultural activities to take place)
- B_i as specified in /12/ or /13/ (in the absence of available information, realistic assumptions were made)
- $F = 0.5$ (/9/)

- u_1 = 75.6 kg $yr.^{-1}$ wet weight (/14/)
- u_2 = 159.6 kg $yr.^{-1}$ wet weight (/14/)
- u_3 = 200 kg $yr.^{-1}$ wet weight (/14/)
- c_{1i} as specified in /13/
- c_{2i} as specified in /13/

(Where either c_{1i} or c_{2i} were unspecified, their value was assumed to be 0.05).

- q = 100 kg per day
- E $\sim$ 5 x 10^9 cm2 (for the spherical model assumed here, E $\sim$ 5 x 10^7 cm2. However, in order to enable sufficient leaching to take place in accordance with the assumptions made for S_4, the water-exposed surface has been assumed to be increased by 2 orders of magnitude, through cracking for example)
- V $\sim$ 3 x 10^{10} cm3
- M = 200 l $yr.^{-1}$ m^{-3} (an extremely conservative value, as discussed in /8/)
- W = 440 l $yr.^{-1}$
- $\emptyset_i$ = 1 (as discussed in the previous chapter)

λ_i, η_i, $(DF)_i$, $(DF)'_i$, $(DF)''_i$ and L_i are given in Table 1 where the values of L_i are taken from /8/, /16/, /17/.

TABLE 1: Nuclide-Specific Parameters Required for Scenario Analysis

N_i	$\lambda_i (yr^{-1})$	$T_i = e^{(100 \lambda_i)}$	$(DF)_i$ rem m3/Ci h	η_i	$(DF)_i'$ rem/Ci	$(DF)_i''$ rem/Ci	B_i	Q_i	L_i (g/cm2 d)
H - 3	5.63E-2	2.79E+2	0	0	6.16E+1	6.61E+1	4.8	4.51	(1E-4)
Be - 10	2.77E-7	1.00	0	0	3.44E+5	4.07E+3	(1.0E-1)	(13.76)	(1E-4)
C - 14	1.24E-4	1.01	0	0	2.11E+3	2.11E+3	5.5	6.26	1E-5
Na - 22	2.67E-1	3.94E+11	1.26	6.40E-1	9.25E+3	9.25E+3	5.2E-2	10.66	(1E-4)
Cl - 36	2.31E-6	1.00	0	0	3.70E+3	3.08E+3	(1.0)	(13.76)	(1E-4)
Ar - 39	2.58E-3	1.29	0	0	N.A.	N.A.	N.A.	N.A.	N.A.
Ca - 41	5.33E-6	1.00	0	0	1.85E+3	1.85E+3	(1.0)	(13.76)	2E-5
Ca - 45	1.65	*	0	0	6.17E+3	3.08E+3	(1.0)	(13.76)	2E-5
Sc - 46	3.01	*	1.25	6.51E-1	2.04E+4	5.55E+3	(1.0E-1)	(13.76)	(1E-4)
V - 49	7.67E-1	*	0	0	2.81E+2	5.55E+1	(1.0E-1)	(13.76)	(1E-4)
Mn - 54	8.72E-1	*	5.11E-1	6.53E-1	6.17E+3	2.64E+3	2.9E-2	2.10	(1E-4)
Fe - 55	2.67E-1	3.94E+11	0	0	1.23E+3	6.17E+2	6.6E-4	5.22	(1E-4)
Co - 57	9.37E-1	*	5.34E-2	6.11E-1	3.08E+3	9.25E+3	9.4E-3	3.15	1E-6
Co - 58	3.65	*	5.91E-1	6.28E-1	4.63E+3	3.70E+3	9.4E-3	3.15	1E-6
Co - 60	1.32E-1	5.40E+5	1.57	6.70E-1	4.63E+4	2.64E+4	9.4E-3	3.15	1E-6
Ni - 59	8.66E-6	1.00	?	?	6.17E+2	1.85E+2	1.9E-2	7.08	(1E-4)
Ni - 63	5.54E-3	1.74	0	0	1.85E+3	3.70E+2	1.9E-2	7.08	(1E-4)
Zn - 65	1.03	*	3.56E-1	6.54E-1	1.85E+4	1.85E+4	4.0E-1	10.50	(1E-4)
Se - 79	1.07E-5	1.00	0	0	2.32E+3	2.64E+3	(1.0E-1)	(13.76)	(1E-4)
Sr - 90	2.48E-2	1.19E+1	(2.00E-4)	(7.05E-1)	1.26E+6	1.85E+5	2.0E-1	2.18	5E-4
Mo - 93	1.98E-4	1.02	?	?	1.80E+3	1.85E+3	1.2E-1	3.81	4E-5
Zr - 93	4.62E-7	1.00	0	0	1.03E+5	1.85E+3	1.7E-4	4.58	(1E-4)
Zr - 95	3.89	*	4.39E-1	6.26E-1	1.85E+4	3.70E+3	1.7E-4	4.58	(1E-4)
Nb - 93 m	5.33E-2	2.06E+2	0	0	6.61E+3	4.63E+2	9.4E-3	23.57	(1E-4)
Nb - 94	3.47E-5	1.00	9.59E-1	6.31E-1	4.87E+4	4.63E+3	9.4E-3	23.57	(1E-4)
Tc - 99	3.30E-6	1.00	0	0	1.68E+3	1.85E+3	2.5E-1	36.23	1E-5
Ru - 106	6.93E-1	*	(1.22E-1)	(6.16E-1)	1.03E+5	2.06E+4	5.0E-2	32.24	(1E-4)
Pd - 107	0.99E-7	1.00	0	0	4.63E+2	1.85E+2	(1.0E-1)	(13.76)	(1E-4)
Ag - 110 m	9.99E-1	*	1.73	6.42E-1	3.70E+4	9.25E+3	1.5E-1	11.27	(1E-4)
Sn - 121 m	1.39E-2	4.02	5.60E-3	1.56E-1	4.63E+3	2.06E+3	(1.0E-1)	(13.76)	(1E-4)

TABLE 1: (contd.) Nuclide-Specific Parameters Required for Scenario Analysis

N_i	$\lambda_i (yr^{-1})$	$T_i = e(100 \lambda i)$	$(DF)_i$ rem m3/Ci h	η_i	$(DF)_i'$ rem/Ci	$(DF)_i''$ rem/Ci	B_i	Q_i	L_i (g/cm2 d)
Sn - 126	6.93E-6	1.00	(1.72)	(6.18E-1)	4.63E+4	1.85E+4	(1.0E-1)	(13.76)	(1E-4)
Sb - 124	4.22	*	1.14	6.65E-1	2.11E+4	9.25E+3	(1.0E-1)	(13.76)	(1E-4)
Sb - 125	2.54E-1	1.07E+11	2.46E-1	6.00E-1	4.63E+3	3.08E+3	(1.0E-1)	(13.76)	(1E-4)
Te - 127 m	26.6	*	2.70E-3	5.88E-1	1.32E+4	9.25E+3	1.3	7.98	(1E-4)
I - 129	4.08E-8	1.00	6.50E-3	1.95E-1	1.85E+5	2.64E+5	2.0E-2	3.18	(1E-4)
Ba - 133	7.79E-2	2.42E+2	0	0	6.17E+3	3.08E+3	5.0E-3	2.31	(1E-4)
Cs - 134	3.30E-1	2.15E+14	9.64E-1	6.23E-1	4.63E+4	6.17E+4	3.0E-3	4.22	1E-4
Cs - 135	3.01E-7	1.00	0	0	4.63E+3	6.17E+3	3.0E-3	4.22	1E-4
Cs - 137	2.33E-2	1.03E+1	(3.37E-1)	(6.16E-1)	3.08E+4	4.63E+4	3.0E-3	4.22	1E-4
Ce - 144	8.88E-1	*	8.40E-3	0.552	2.64E+5	2.06E+4	2.5E-3	2.11	(1E-4)
Pm - 147	2.65E-1	3.23E+11	0	0	3.08E+4	9.25E+2	(1.0E-1)	(13.76)	(1E-4)
Sm - 151	7.45E-3	2.11	2.00E-4	8.41E-2	3.08E+4	3.08E+2	(1.0E-1)	(13.76)	(1E-4)
Eu - 152	5.46E-2	2.35E+2	8.57E-1	6.41E-1	2.47E+5	4.63E+3	(1.0E-1)	(13.76)	(1E-4)
Eu - 154	4.33E-2	7.59E+1	7.40E-1	6.49E-1	3.36E+5	9.25E+3	(1.0E-1)	(13.76)	(1E-4)
Eu - 155	3.85E-1	5.25E+16	2.57E-2	5.57E-1	2.31E+4	9.25E+2	(1.0E-1)	(13.76)	(1E-4)
Ho - 166 m	5.78E-4	1.06	9.56E-1	6.18E-1	7.40E+5	7.77E+3	(1.0E-1)	(13.76)	(1E-4)
Ta - 182	2.20	*	8.29E-1	6.55E-1	3.66E+4	5.92E+3	(1.0E-1)	(13.76)	(1E-4)
U - 235	9.84E-10	1.00	6.94E-2	5.70E-1	8.81E+6	2.64E+5	(1.0E-2)	(13.76)	1E-6
U - 238	1.55E-10	1.00	(4.10E-3)	(5.97E-1)	5.00E+6	2.31E+5	(1.0E-2)	(13.76)	1E-6
Np - 237	3.24E-7	1.00	1.25E-2	5.44E-1	4.63E+8	3.70E+7	2.5E-3	2.02	1E-6
Pu - 238	7.89E-3	2.20	6.00E-4	8.25E-2	3.70E+8	3.70E+5	2.5E-4	(13.76)	1E-6
Pu - 239	2.84E-5	1.00	1.00E-4	2.43E-1	3.70E+8	4.63E+5	2.5E-4	(13.76)	1E-6
Pu - 240	1.06E-4	1.01	0	0	3.70E+8	4.63E+5	2.5E-4	(13.76)	1E-6
Pu - 241	4.62E-2	1.06E+2	(7.60E-3)	(5.41E-1)	7.40E+6	9.25E+3	2.5E-4	(13.76)	1E-6
Pu - 242	1.79E-6	1.00	0	0	3.70E+8	3.70E+5	2.5E-4	(13.76)	1E-6
Am - 241	1.60E-3	1.17	7.60E-3	5.41E-1	4.63E+8	2.06E+6	2.5E-4	(13.76)	1E-6
Am - 243	9.40E-5	1.01	1.87E-2	5.83E-1	4.63E+8	2.06E+6	2.5E-4	(13.76)	1E-6
Cm - 242	1.56	*	(6.00E-4)	(8.25E-2)	1.85E+7	6.17E+4	(1.0E-2)	(13.76)	(1E-4)
Cm - 244	3.87E-2	4.79E+1	0	0	2.64E+8	9.25E+5	(1.0E-2)	(13.76)	(1E-4)

4.2 Primary Data

A starting list of 59 potentially relevant decommissioning-waste radionuclides was prepared, as shown in Table 1 (column N_i). Applicable decay constants, λ_i, are listed in the second column of Table 1. In the third column of the table are calculated the appropriate radionuclide decay reduction factors, T_i, resulting from an isolation period of 100 years. In the subsequent columns of Table 1 are listed $(DF)_i$, η_i, $(DF)'_i$ and $(DF)''_i$, the dose-conversion information relevant to the respective exposure mechanisms by which doses might be received.

The asterisks (*) in Column 3 of Table 1 indicate radionuclide decay reduction factors in excess of 10^{20}. In the remaining columns, a value in parenthesis indicates a (conservatively) estimated value; a question mark (?) indicates that insufficient information is available, at present, to enable a reasonable estimate to be made. In Column 4, the estimated values have been derived from information on the daughter of the radionuclide in question. The term N.A. indicates that a particular value is not applicable.

In the third last column of Table 1 are listed vegetative bioaccumulation factors, B_i. In the second last column are listed values of Q_i, where Q_i is defined as

$$Q_i = u_1 c_{1i} + u_2 c_{2i} + (u_3/q) \qquad - (8)$$

This parameter is tabulated for use in the $(MAC)_i$ algorithm relevant to the food uptake scenario, S_3.

In the last column of Table 1 are listed leach-rate values, L_i.

4.3 Reduced Forms of Maximum Average Concentration Algorithms

From equations (2) to (8) and the relevant parameters specified in Section 4.1, the maximum average concentration algorithms may be expressed in reduced, simplified, form as given below.

4.3.1 Direct Irradiation Scenario - S_1

$$(MAC)_{i1} = 1.5 \times 10^{-5} \cdot T_i/(\eta_i \cdot (DF)_i) \qquad -(9)$$

4.3.2 Dust Inhalation Scenario - S_2

$$(MAC)_{i2} = 1.8 \times 10^{5} \cdot T_i/(DF)'_i \qquad -(10)$$

4.3.3 Food Uptake Scenario - S_3

$$(MAC)_{i3} = 1.4 \times T_i/(B_i \cdot Q_i \cdot (DF)''_i) \qquad -(11)$$

4.3.4 Water Uptake Scenario - S_4

$$(MAC)_{i4} = 1.7 \times 10^{-4} \cdot T_i/(L_i \cdot (DF)''_i) \qquad -(12)$$

TABLE 2: $(MAC)_i$ for Scenarios S_1 - S_4 and $(MPC)_i$ Values

N_i \ $(MAC)_i$	S_1 Direct Irradiation	S_2 Dust Inhalation	S_3 Food Uptake	S_4 Water Uptake	$(MPC)_i$ Ci m^{-3}
* H - 3	**	N.A.	2.7E-1	(7.2)	3E-1
* Be - 10	**	5.2E-1	(2.5E-4)	(4.2E-4)	(3E-4)
* C - 14	**	8.6E+1	(2.0E-5)	8.1E-3	(2E-5)
Na - 22	**	**	**	**	**
* Cl - 36	**	N.A.	(3.3E-5)	(5.5E-4)	(3E-5)
* Ar - 39	**	N.A.	N.A.	N.A.	?
* Ca - 41	**	9.7E+1	(5.5E-5)	(4.6E-3)	(6E-5)
Ca - 45	**	**	**	**	**
Sc - 46	**	**	**	**	**
V - 49	**	**	**	**	**
Mn - 54	**	**	**	**	**
Fe - 55	**	**	**	**	**
Co - 57	**	**	**	**	**
Co - 58	**	**	**	**	**
* Co - 60	7.7	2.1E+6	9.7E+2	3.5E+3	8
* Ni - 59	?	2.9E+2	5.6E-2	(9.2E-3)	(9E-3)
* Ni - 63	**	1.7E+2	4.9E-2	(7.8E-3)	(8E-3)
Zn - 65	**	**	**	**	**
* Se - 79	**	7.8E+1	(3.9E-4)	(6.4E-4)	(4E-4)
* Sr - 90	(1.3)	1.7	2.1E-4	2.2E-5	2E-5
* Mo - 93	?	1.0E+2	1.7E-3	2.3E-3	2E-3
* Zr - 93	**	1.8	9.7E-1	(9.2E-4)	(9E-4)
Zr - 95	**	**	**	**	**
* Nb - 93 m	**	5.6E+3	2.8	(7.6E-1)	(8E-1)
* Nb - 94	2.5E-5	3.7	1.4E-3	(3.7E-4)	3E-5
* Tc - 99	**	1.1E+2	8.4E-5	9.2E-3	8E-5
Ru - 106	**	**	**	**	**
* Pd - 107	**	3.9E+2	(5.5E-3)	(9.2E-3)	(6E-3)
Ag - 110 m	**	**	**	**	**
* Sn - 121 m	6.9E-2	1.6E+2	(2.0E-3)	(3.3E-3)	(2E-3)

** corresponds to values greater than E+6
N.A. not applicable

TABLE 2: (contd.) $(MAC)_i$ for Scenarios S_1 - S_4 and $(MPC)_i$ Values

N_i \ $(MAC)_i$	S_1 Direct Irradiation	S_2 Dust Inhalation	S_3 Food Uptake	S_4 Water Uptake	$(MPC)_i$ Ci m^{-3}
* Sn - 126	(1.4E-5)	3.9	(5.5E-5)	(9.2E-5)	(1E-5)
Sb - 124	**	**	**	**	**
Sb - 125	**	**	**	**	**
Te - 127 m	**	**	**	**	**
* I - 129	1.2E-2	9.7E-1	8.3E-5	(6.4E-6)	(6E-6)
* Ba - 133	**	7.1E+3	9.5	(1.3E-1)	(1E-1)
Cs - 134	**	**	**	**	**
* Cs - 135	**	3.9E+1	1.8E-2	2.2E-4	2E-4
* Cs - 137	(7.4E-4)	6.0E+1	2.5E-2	2.8E-4	4E-4
Ce - 144	**	**	**	**	**
Pm - 147	**	**	**	**	**
* Sm - 151	1.9	1.2E+1	(7.0E-3)	(1.2E-2)	(7E-3)
* Eu - 152	6.4E-3	1.7E+2	(5.2E-2)	(8.6E-2)	6E-3
* Eu - 154	2.4E-3	4.1E+1	(8.4E-3)	(1.4E-2)	2E-3
Eu - 155	**	**	**	**	**
* Ho - 166 m	2.7E-5	2.6E-1	(1.4E-4)	(2.3E-4)	3E-5
Ta - 182	**	**	**	**	**
* U - 235	3.8E-4	2.0E-2	(3.9E-5)	6.4E-4	(4E-5)
* U - 238	(6.1E-3)	3.6E-2	(4.4E-5)	7.4E-4	(4E-5)
* Np - 237	2.2E-3	3.9E-4	7.5E-6	4.6E-6	5E-6
* Pu - 238	6.7E-1	1.1E-3	(2.4E-3)	1.0E-3	1E-3
* Pu - 239	6.2E-1	4.9E-4	(8.8E-4)	3.7E-4	4E-4
* Pu - 240	**	4.9E-4	(8.9E-4)	3.7E-4	4E-4
* Pu - 241	(3.9E-1)	2.6	(4.7)	2.0	(4E-1)
* Pu - 242	**	4.9E-4	(1.1E-3)	3.8E-4	4E-4
* Am - 241	4.3E-3	4.6E-4	(2.3E-4)	9.7E-5	1E-4
* Am - 243	1.4E-3	3.9E-4	(2.0E-4)	8.3E-5	8E-5
Cm - 242	**	**	**	**	**
* Cm - 244	**	3.3E-2	(5.3E-4)	(8.8E-5)	(9E-5)

** corresponds to values greater than E+6
N.A. not applicable

5 RESULTANT MAXIMUM AVERAGE AND PERMISSIBLE CONCENTRATIONS FOR A LOW-LEVEL WASTE REPOSITORY

5.1 Base-Case Results

Using equations (9) - (12) and the information contained in Table 1, maximum average concentrations for the four scenarios have been derived and tabulated in Table 2. From these, the maximum permissible concentration (MPC) for each of the 59 radionuclides has been obtained. The MPC values are listed in the final column of Table 2, and have been appropriately "rounded".

Since MPC values $\geqslant 10^6$ Ci/m3 are much higher than the specific activities occurring in radioactive waste intended for a Type A repository /15/, radionuclides with such high MPC values can be regarded, for present purposes, as being "inventory non-significant". The remaining radionuclides are considered to be inventory signficant, and such "inventory significant" radionuclides have been marked with an asterisk (*).

In view of what has been said in the previous paragraph, all numerical values in Table 2 which are equal to, or greater than, 10^6 Ci/m3, have been replaced by a double asterisk (**). In this table, a value in parenthesis indicates a (conservatively) estimated value and a question mark (?) indicates that insufficient information is available even to make a realistic estimate. In such cases, although some MAC values are designated by a question mark, it has, nevertheless, been possible to determine MPC values in all instances except for Ar-39 since, irrespective of the value which should be accorded to the unknown MAC, it can be stated with certainty that it would certainly not be less than that of the applicable MPC.

Finally, for some nuclides, the considered exposure pathways are not applicable and this has been indicated by N.A. in the appropriate entries of Table 2.

5.2 Sensitivity of Maximum Average Concentrations to Parameter Variations

Since the MPC values obtained in Table 2 are determined from the MAC values, changes in MAC values might result in changes to the MPC. In any case, changes in MAC values for any one radionuclide could change the "dominant" scenario (i.e. the scenario which determines the MPC for that nuclide). For this reason, it is necessary to consider the sensitivity of the MAC values (and implications for the MPCs) to variations of algorithm parameters. Each of the four scenarios is considered, in turn, below.

5.2.1 Direct Irradiation Scenario - S_1

From Equation (2) and the parameters specified in Section 4.1, the most sensitive parameter can be seen to be the exposure period per year which might vary between 10 to 5,000 hours per year. The resultant variation in the value of the MAC, from that given in Table 2, would be +2 orders of magnitude or -1/2 order of magnitude.

5.2.2 Dust Inhalation Scenario - S_2

From Equation (4) and the parameters specified in Section 4.1, the most sensitive parameters can be seen to be the exposure period and the dust loading in the air. The former might be as low as 10 hours per year and the latter could vary between 10^{-7} to 10^{-9} kg/m3. The variation in the value of the MAC, as a result of this spread in dust loading, would be roughly one order of magnitude. The variation in the value of the MAC, as a result of the assumed decrease in exposure period, would be roughly +2 orders of magnitude. However, since lower dust loadings are likely to result in longer exposure periods, and vice versa, the two effects together probably result in very little effective change in the overall MAC value.

5.2.3 Food Uptake Scenario - S_3

From Equation (5) and the parameters specified in Section 4.1, it can be seen that the most sensitive parameters are the consumption rate for animals (q), the regionally produced fraction of annual food consumption (F) and the vegetative bioaccumulation factor for nuclide N_i (B_i). q is fairly well determined but F might be very much lower than the value assumed. Nevertheless, although lower values of F would increase the MAC values, there is little possibility of any decrease in the MAC values.

On the other hand, published values of B_i in some instances vary over a range of 6 orders of magnitude for certain nuclides. Therefore, some changes in B_i values might be possible and, consequently, changes in MAC values are feasible. Such changes would not have much significance on the heavy metal MPCs, for example, since these MPCs would still be constrained by MAC values from S_2, even if MAC values from S_3 were to be increased and S_4 has no constraining effect.

5.2.4 Water Uptake Scenario - S_4

From Equation (7) and the parameters specified in Section 4.1, it can be seen that the most sensitive parameters are the water flow and uptake rates (M, W) and the radionuclide "rate of removal time constant" (μ_i). It is, of course, assumed that any surrounding geological media plays no role so that $\emptyset_i = 1$. Variations in μ_i are considered in greater detail, subsequently. W is fairly well determined. M, on the other hand, could be considerably higher than the value assumed in Section 4.1. Values as high as 6,000 l/yr./m3 cannot be excluded, which would result in an increase in the MAC values by at least an order of magnitude.

6 DECOMMISSIONING-WASTE RADIONUCLIDE INVENTORY FOR A FIRST APPLICATION OF THE METHODOLOGY

6.1 Background

As discussed in Section 2.1, before a decision concerning the allocation of waste types to the different repositories can be made, an inventory of waste types, entailing a practical grouping of wastes with similar nuclide inventories and conditioning, must be established. An initial compilation of such an inventory for Swiss nuclear waste arisings is being prepared /15/ and data pertaining to decommissioning wastes have been summarized in /8/. It is only the decommissioning wastes which will be of concern here, since it is primarily these wastes which are intended to be stored in a Type A repository, the MPC values for which have been determined in the previous chapter.

A partitioning involving radioactive wastes arising from sources other than from decommissioning must await a corresponding scenario analysis applied to a Type B repository, which will involve greater attention to geological barriers than has been the case here.

In the next section, a "hypothetical" decommissioning-waste inventory for the approximately 3,000 MWe of nuclear power plant currently in operation or under construction, in Switzerland, is presented. The net immobilized waste volume, excluding containers, is assumed to be approximately 30,000 m3 and the gross waste volume, including containers, is assumed to be ~45,000 m3, as stated in Section 3.1.

6.2 Inventory Assumptions

Only those power plants under construction or currently in operation in Switzerland are here considered. From inventory information relating to Swiss decommissioning wastes /15/ together with other general decommissioning waste information /18/, seven broad categories of waste types have been established, together with net volumes as follows:

A - Low level activated and contaminated concrete (~5,000 m3)

B - Low level activated steels (~2,300 m3)

C - Low level contaminated steels (~7,800 m3)

D - Intermediate level activated steels (~4,300 m3)

E - Intermediate level contaminated steels (~4,300 m3)

F - Low level contaminated tools and other materials used during decommissioning (~3,500 m3)

G - Low and intermediate level concentrates produced as a result of decommissioning activities (~3,000 m3)

Such a categorization corresponds roughly to that already established for Swiss decommissioning wastes /8/ with only some small differences.

Using these categories, together with information on radionuclide activities /15/, /18/ and the estimated waste volumes, a table of specific activities (Ci/m3) for the 38 "inventory significant" nuclides established in the previous chapter was prepared and can be considered, although "hypothetical", roughly representative of the total Swiss decommissioning-waste radionuclide inventory. This inventory is presented as Table 3. Question marks, once again, indicate that insufficient information is currently available to make realistic estimates.

TABLE 3: Specific Activities (Ci m^{-3}) for Waste Types A - G

Nuclide \ Waste Type	A	B	C	D	E	F	G
H-3	2.6E-9	?	?	?	?	?	?
Be-10	?	1.1E-9	?	5.5E-10	?	?	?
C-14	2.0E-3	1.3E-3	?	1.7	?	?	?
Cl-36	?	?	?	?	?	?	?
Ar-39	4.8E-4	?	?	2.7E-8	?	?	?
Ca-41	3.6E-2	?	?	?	?	?	?
Co-60	2.5E+7	9.6E-2	3.3E-2	4.1E+3	1.9E-1	3.8E-4	1.7
Ni-59	3.3E-3	8.4E-4	?	1.1E+1	?	?	?
Ni-63	1.1E-3	9.4E-2	?	1.7E+3	?	?	?
Se-79	?	?	?	?	?	?	?
Sr-90	0	2.2E-4	3.2E-4	1.1E-4	1.8E-5	3.7E-6	4.1E-6
Mo-93	?	1.2E-2	?	1.8E-2	?	?	?
Zr-93	?	?	?	8.2E-6	?	?	?
Nb-93 m	?	5.6E-6	?	1.9E-1	?	?	?
Nb-94	0	1.2E-3	4.5E-5	3.5E-2	2.0E-4	4.3E-7	4.7E-3
Tc-99	?	2.3E-5	?	4.4E-3	?	?	?
Pd-107	?	?	?	?	?	?	?
Sn-121 m	?	?	?	?	?	?	?
Sn-126	?	?	?	?	?	?	?
I-129	?	?	?	?	?	?	?
Ba-133	2.7E-2	?	?	?	?	?	?
Cs-135	?	?	?	?	?	?	?
Cs-137	0	2.3E-3	1.1E-3	1.1E-3	2.3E-2	3.8E-5	4.2E-1
Sm-151	?	?	?	2.1E-2	?	?	?
Eu-152	1.8E-1	?	?	1.1E-3	?	?	?
Eu-154	?	?	?	3.1	?	?	?
Ho-166 m	?	?	?	7.8E-4	?	?	?
U-235	0	4.5E-12	9.3E-12	2.2E-12	8.5E-12	1.0E-13	9.8E-10
U-238	0	8.4E-11	7.8E-6	4.0E-11	9.3E-10	1.9E-12	2.1E-8
Np-237	0	1.0E-10	2.1E-10	4.8E-11	1.1E-9	2.3E-12	2.5E-8
Pu-238	0	7.1E-7	1.5E-6	3.4E-7	7.7E-6	5.0E-9	9.0E-5
Pu-239	0	7.7E-8	1.6E-7	3.7E-8	2.2E-7	1.8E-9	1.9E-5
Pu-240	0	1.2E-7	2.5E-8	5.7E-8	2.3E-7	2.7E-9	3.0E-5
Pu-241	0	1.0E-4	4.4E-5	1.0E-5	2.3E-4	4.7E-6	1.2E-3
Pu-242	0	5.5E-10	5.3E-9	2.8E-10	5.4E-9	7.3E-10	1.2E-7
Am-241	0	2.1E-5	9.0E-6	3.4E-7	7.7E-6	1.6E-8	9.0E-5
Am-243	0	5.7E-9	1.2E-8	2.7E-9	6.1E-8	1.3E-10	1.4E-6
Cm-244	0	3.1E-7	3.9E-6	1.5E-7	3.4E-6	7.0E-9	6.8E-5

7 INITIAL DERIVATION OF STORABLE WASTE QUANTITIES

7.1 Base-Case Results

According to the methodology established in Chapter 2, the MPC for the mixture of "inventory significant" radionuclides should be calculated using Equation (1). This should then be compared with the total specific activity of each of the waste types A - G given in Table 3. For this first application of the methodology, however, because of the lack of available data (as exemplified by Table 3) and because of the usefulness of obtaining nuclide-specific information, a slightly modified approach to the waste partitioning was adopted in order to derive an initial storable volume of waste for the Type A repository. This approach entailed making a nuclide-by-nuclide comparison of the specific activities given in Table 3 with the MPCs derived in Table 2. Only those waste types for which all the specific activities given in Table 3 are lower than the corresponding MPC in Table 2 are regarded, in this (base) case, as being capable of being stored in the Type A repository.

By making such a comparison, it is readily verifiable that only waste type F is capable of being stored in a Type A repository, assuming that no allowance is made for the engineered safety barriers. This implies a net immobilized-waste volume (excluding containers) of some 4,000 m3 which is merely approximately 13 % of the initially foreseen storable net volume.

One serious difficulty relates, of course, to the fact that there are a number of missing entries in Table 3. When, in the future, information is forthcoming on the presently unknown specific activities it might transpire that storable waste quantities are severely constrained by the unavailable specific activities. In the following, however, the assumption is made that any subsequently derived specific actitivies will always be lower than the presently derived MPC values.

7.2 Influence of Waste Characterization

Since, in any realistic decommissioning-waste disposal situation, not all the waste from all Swiss nuclear power plants will be "lumped" together as implied by Table 3 but, rather, the resulting wastes from each individual power plant are more likely to be disposed of individually, it is perhaps preferable to undertake a comparison of the kind proposed in the previous section on a plant-by-plant basis. Making use of such plant specific information on decommissioning-waste radionuclide inventories as was available /8/ at the time of the current investigation, the storable waste volume derived in the previous section is roughly doubled, to a figure of some 20 % of the initially foreseen storable volume. Due to the tentative nature of the inventory data contained in /8/, these data are not reproduced here. Nevertheless, the fact that greater refinement in inventory information (as might be suspected) yields an improvement in the volume of waste capable of being stored in a Type A repository, should provide incentive for obtaining improved inventory information in the future.

7.3 Influence of Isolation Period

Before investigating, in the next chapter, the effects of engineered-barrier-relevant parameter variations on storable volumes of waste, it is worthwhile to consider the influence of changes in the isolation period on MPC values and, ultimately, storable volume. By increasing the length of the isolation period (conceivably by administration control procedures, for example) the effects of the extended decay of the active nuclides can be expected to influence some of the MPC values.

By comparing the MPC values given in Table 2 with the specific activity inventory in Table 3, it would appear that there are 14 nuclides for which the inventory specific activity exceeds the MPC value. These 14 nuclides, together with the waste type for which the MPC is less than the specific activity for the nuclide in question, are presented in Table 4, below, where the relevant entry is indicated by an "X".

Table 4: Nuclides in Waste Types with Specific Activity Exceeding MPC Values (Isolation Period = 100 years)

Waste Type / Nuclide	A	B	C	D	E	F	G
C-14	X	X		X			
Ca-41	X						
Co-60	X			X			
Ni-59				X			
Ni-63		X		X			
Sr-90		X	X	X			
Mo-93		X		X			
Nb-94			X		X		X
Tc-99				X			
Cs-137		X	X	X	X		X
Sm-151				X			
Eu-152	X						
Eu-154				X			
Ho-166m				X			

C-14, Sr-90, Nb-94 and Cs-137 appear to play a dominant role in preventing the decommissioning-waste from being stored in a Type A resository, from the standpoint of this base-case investigation. When the latter three nuclides have decayed to a significantly low level, waste types C, E and G, in addition to F, should be capable of being stored in a Type A repository. By doubling the isolation period, from 100 to 200 years, the MPCs of Sr-90 and Cs-137 are increased by an order of magnitude which, together with the almost equal values of the MPC and specific activity for waste type C for Nb-94 and Sr-90, just permit waste type C to be stored in the Type A repository. The net immobilized-waste volume (excluding containers) of waste type C is some 8,000 m3. Consequently, since with an isolation period of 200 years waste types C and F can be stored in Type A repository, a storable volume of approximately 40 % of the initially foreseen storable volume is attainable in this case.

With a 200 year control period, Table 4 becomes modified to Table 5, below. Within realistic time-scales, no further significant increase in storable waste volume can be attained through an increase in the isolation period.

Table 5: Nuclides in Waste Types with Specific Activity Exceeding MPC Values (Isolation Period = 200 years)

Waste Type / Nuclide	A	B	C	D	E	F	G
C-14	X	X		X			
Ca-41	X						
Ni-59				X			
Ni-63		X		X			
Sr-90			(X)				
Mo-93		X		X			
Nb-94			(X)		X		X
Tc-99				X			
Cs-137					X		X
Sm-151				(X)			
Eu-154				X			
Ho-166m				X			

8 EFFECTS OF ENGINEERED-BARRIER-RELEVANT PARAMETER VARIATIONS

8.1 Background

As explained in Section 2.4, safety barriers play an important role in isolating the buried wastes from the biosphere and, in consequence, improving storage capabilities. Up to this point, the effects of the engineered safety barriers (other than the leach-resistant waste matrix) have been ignored, the assumption having been made that at the conclusion of the isolation period, all barriers are totally degraded. An alternative way of expressing this is to consider all engineered barriers to fail simultaneously after expiry of the control period.

In reality, the engineered barriers will not all fail simultaneously after expiry of the isolation period. It is valuable, therefore, to investigate the effects of any maintenance of barrier integrity upon storage capabilities (which, in turn, implies investigating the effects of engineered-barrier-relevant parameter variations on MPC values). Essentially, maintenance (or partial maintenance) of barrier integrity after conclusion of the control period would have the following implications for the exposure pathways considered in the present analysis:

i) For scenario S_1, any intact engineered barriers will result in a certain amount of shielding and thereby reduce potential exposure levels.

ii) For scenario S_2, any intact engineered barriers will restrict the amount and, therefore, activity of radioactive material admixed with dust and available for inhalation.

iii) For scenario S_3, any intact engineered barriers will restrict the amount and, therefore, activity of radioactive material available for direct biological uptake.

iv) For scenario S_4, the longer the engineered barriers remain intact, the longer the radionuclide release period will be, since leaching will commence later.

In all instances, the net effect is likely to be an increase in the respective MAC (and, ultimately, MPC) values.

For scenarios S_2 and S_3, since an assessment of the effects of physico-chemical barrier degradation processes lies beyond the scope of this investigation, coupled with the fact that it has been claimed /19/ that steel storage drums should no longer be regarded as remaining intact after a 100-year isolation period, the only means of assessing the effect of barrier integrity on storage criteria (through MAC sensitivity analysis) is by arbitrarily varying the volume ratio of disinterred immobilized waste present in the ground surface layer to the total of disinterred immobilized waste (f') in Equations (4) and (5). Because of the arbitrariness of such an approach, it has not been attempted and scenarios S_2 and S_3 will not be considered further. As it happens, merely 4 "inventory significant" nuclides which govern the partitioning of the waste (those listed in Table 4) namely C-14, Ca-41, Tc-99 and Sm-151, have their MPCs restricted by the MAC values determined from either S_2 or S_3.

An attempt will be made, in the next sections, to assess the effects of barrier integrity on storage criteria for scenarios S_1 and S_4.

8.2 Shielding Characteristics

To reduce the conservativeness of ignoring shielding in scenario S_1 (and, thereby, improve the analysis in Section 3.4.1) effects of non-degraded engineered barriers (e.g. intact portions of steel drums, concrete containers, etc.) and any overburden of rock and soil need to be considered. The most simple way of doing this to introduce an attenuation factor, C_i, into Equation 2.

$$C_i = \left[K_1 \exp(-l_1 x) + \ldots\ldots + K_n \cdot \exp(-l_n x)\right] / (K_1 + \ldots\ldots + K_n) \qquad -(13)$$

where

K_n = product of photon energy and emission probability, for photon n

l_n = linear attenuation coefficient for photon n, in shielding material (cm^{-1})

x = thickness of shielding material (cm)

x is a new parameter, variational implications of which for MACs and MPCs can be investigated.

An analysis of the 5 radionuclides for which shielding will be of significance (Co-60, Nb-94, Eu-152, Eu-154 and Ho-166m, from Tables 2 and 4) indicate that at least 50 cm of concrete or soil shielding would be required before the irradiation exposure pathway is no longer of significance.

In the absence of a thorough inhomogeneous analysis, and since such a layer of shielding material cannot be justified, shielding cannot be considered to influence upon storage criteria. In any case, even if shielding to the appropriate extent could be maintained, from Table 4 it can be seen that the end effect on the quantities of storable waste is negligible because partitioning is, ultimately, constrained by nuclides other than those considered here, namely those for which the MPC values are determined primarily by scenario S_4.

8.3 Barrier Retention Characteristics

In scenario S_4, once all barriers external to the waste matrix have failed, radionuclides can be removed by an appropriate removal process, such as leaching by intruding water. As stated in Section 3.4.4, the rate of removal (or release rate) of the radionuclides from the immobilized waste is assumed to be described by an exponential decay function, with time-constant μ_i which is related to the leach rate by Equation (6). The influence on release rate of the maitenance of partial integrity of safety barriers external to the waste matrix can be investigated by varying μ_i. Also, from Equation (7) it can be seen that as μ_i decreases in value (implying an improvement in barrier integrity - fewer cracks in drums, for example), $(MAC)_i$ increases.

According to Equation (6), the two parameters capable of significantly influencing μ_i are the leach rate (L_i) and the surface area to volume ratio (E/V) for the immobilized waste.

It is not clear how E/V will vary, since it will be determined by the ultimate packaging, and storage in the repository, of the immobilized waste material, together with physico-chemical barrier degradation processes. E/V could vary from 0.002 cm^{-1} for the initial state of the spherical repository model assumed for this analysis, to approximately 100 cm^{-1} for the case where the contents of the repository have been degraded to a "powder". For intact steel drums, E/V = 0.1 cm^{-1}, a value which can increase by up to 2 orders of magnitude as the drum cracks and decomposes and the waste matrix is degraded.

In the case of leach rates, numerous values in Table 1 have had to be merely estimated. It is likely that many of the values are extremely conservative. Furthermore, leach rates are dependent on a variety of physical factors, as summarized below, based on /20/ and /21/.

The leach rate is a measurable quantity related to the ability of solid nuclear waste forms to retain radionuclides when the waste forms are exposed to liquid. Factors which influence the leach rate may be categorized as being related to the leaching system, the leachant or the solid being leached.

System Factors (time, temperature, pressure, radiation environment, ratio of surface area to volume of solid)

The leach rate is a function of time and is rarely constant in the early stages of leaching. A decreasing power-law time-dependence is observed, which may be negative square-root. The latter is associated with a diffusion mechanism for leaching, whilst the former is associated with a dissolution or corrosion mechanism.

The leach rate usually increases with increasing temperature.

There are few data available on the isothermal effect of pressure on the leach rate. However, in the range of reasonably expected conditions, the effect of pressure should be minor compared to that of temperature.

Radiation from sources internal to the waste will affect all parts of the leaching system and has the potential for increasing the leach rate. Radiolysis, which has the potential for forming hydrogen peroxide will, undoubtedly, enhance the leach rate.

As the surface area to volume ratio of the solid increases (e.g. the waste form changes from a monolithic to powdered form) one would expect the leach rate to increase. However, concentration effects eventually limit the amounts leached and leach rates from powdered forms of materials have been measured that are 2 - 3 orders of magnitude lower than the leach rates from monolithic forms of the same material.

Leachant Factors (composition, pH, Eh, flow)

Trends in the leach rate with different leachants are difficult to predict. It has been shown that high-purity water is often a more aggressive leachant than highly saline leachants, for example. Leachants containing complexing ions, such as carbonate, may result in higher leach rates than leachants not containing complexing ions. Traces of organic materials, such as humic acid (a powerful complexing agent), are often present in ground waters and can, therefore, enhance leach rates in underground repositories. Trace elements can also play an important role in increasing the leach rate.

The leach rate can be a strong function of pH. Neutral-range (pH 5 - 9) leachants usually result in lowest leach rates and enhancement by factors of 10 or more may occur outside the neutral range.

Oxidation-reduction potential, Eh, of the leachant may plant an important role in the rate of leaching of some waste forms. Actindes, such as uranium, neptunium and plutonium, would be expected to be highly leachable under oxidizing conditions because of the formation of soluble oxo-ions.

The leach rate of a solid in a flowing leachant increases with flow rate. The effect becomes more pronounced with increasing temperature.

Solid Factors (composition, surface condition, porosity)

The most elementary effect on the leach rate is associated with the nature of the solid being leached. Materials such as bitumen, plastics, concrete, glass, ceramics and composites, have diverse leach rates.

The condition of the surface of the solid being leached can have an effect on the leach rate. Waste forms with greater surface roughness displaying greater leach rates, all other conditions being equal.

Materials with open porosity, such as concrete, may have high leach rates simply because the surface area exposed to the leachant is much greater than the geometric surface area. (Porosity may increase the leaching surface area by a factor of 1,000 or more.) The effect is minimized in solids with non-connected pores.

8.3.1 Implications

In view of what has been said, a large variation in true values of μ_i in any actual situation can be expected. The situation is complicated by the fact that, from Equation (6), as E/V increases (as the waste form becomes degraded) there should be an increase in μ_i, but from what has been said about leach rates, as E/V increases, the leach rate decreases, resulting in a decrease in μ_i. It is unclear how these two effects compete. In consequence, potentially large variations in MAC_i values are likely for scenario S_4 resulting from variations in μ_i. The theoretical lower limit of μ_i approaches zero but, for the presently assumed model, any limit lower than approximately 10^{-8} becomes untenable. Although unable to be accurately specified, by making reasonable assumptions about maximum values of E/V and the leach rate, the upper limit of μ_i is $\sim 10^{-2} - 10^{-3}$.

For the nuclides listed in Table 4, MAC values for scenarios S_4 differ by 5 orders of magnitude for values of $\mu_i = 10^{-3}, 10^{-8}$. However, at the lower value of μ_i, the MPC values of only Ni-59, Ni-63, Sr-90 and Nb-94 are increased, to 6E-2, 5E-2, 2E-4 and 1E-3, respectively. Nevertheless, this has no significance for the partitioning of wastes since even at these low values of μ_i, merely waste Type F can be stored in the Type A repository, just as for the base-case in Section 7.1. The reason for this is that even once the MPC values are no longer constrained by the MAC values derived from scenario S_4, there are still at least 10 MPC values constrained by MAC values derived from the other three scenarios.

9 CONCLUSION

This first attempt to partition Swiss radioactive waste arisings and determine potential storage capabilities for a Type A repository has been undertaken based upon a simplified model of the Type A repository which does not take full account of combined effects of the engineered safety barriers.

The results of the present investigation have yielded conclusions different from those of investigations to-date. Instead of a potential storage capacity of some 45,000 m3 of waste including packaging and containers (or double the figure if planned nuclear power plants are included) a storage capability of much less than half of this originally envisaged capacity has been arrived at. The reasons for this are twofold. Firstly, different modelling assumptions and parameters from those used previously have now been employed. Secondly, it has been necessary to work with an extremely rough inventory of waste arisings.

In the process of establishing an inventory of waste arisings, it has been found that certain crucial isotopes are distributed throughout the waste and the fact that they cannot be separated out has important consequences for the partitioning. In view of this, it is clear that more detailed information on both waste arisings and repository "near-field" effects needs to be obtained.

From the results derived it can, nevertheless, be argued that despite conservative modelling assumptions and restrictive waste inventory information, it appears possible to build a Type A repository capable of storing a not insubstantial quantity of waste material. By future improvements in input information required for the methodological approach developed here, improvements in storage capabilities for a Type A repository are likely to be achieved.

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